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THERMAL IMPACT OF WASTE EMPLACEMENT AND SURFACE COOLING ASSOCIATED WITH GEOLOGIC DISPOSAL OF NUCLEAR WASTE

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ABSTRACT

This report studies the thermal effects associated with the emplacement of aged radioactive wastes in a geologic repository, with emphasis on the following subjects:

- the waste characteristics, repository structure, and rock properties controlling the thermally induced effects,
- the current knowledge of the thermal, thermomechanical, and thermohydrologic impacts, determined mainly on the basis of previous studies that assume 10-year-old wastes
- the thermal criteria used to determine the repository waste loading densities, and
- the technical advantages and disadvantages of surface cooling of the wastes prior to disposal as a means of mitigating the thermal impacts.

The waste loading densities determined by repository designs for 10 year-old wastes are extended to older wastes using the near-field thermomechanical criteria based on room stability considerations. Also discussed are the effects of long surface cooling periods determined on the basis of far-field thermomechanical and thermohydrologic considerations. The extension of the surface cooling period from 10 years to longer periods can lower the near-field thermal impact but have only modest long-term effects for spent fuel. More significant long-term effects can be achieved by surface cooling of reprocessed high-level waste.

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ABBREVIATIONS

BWR	boiling water reactor
DHLW	defense high-level waste
DOE	Department of Energy
EIA	Energy Information Administration
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
FBR	fast breeder reactor
GEIS	Generic Environmental Impact Statement
HLW	high-level waste
HTR	high-temperature reactor
HWR	heavy water reactor
INFCE	International Nuclear Fuel Cycle Evaluation
KBS	Kärnbränslesäkerhet, Swedish Nuclear Fuel Safety Program
LWR	light water reactor
MOX	mixed oxide fuel
NRC	Nuclear Regulatory Commission
NWTS	National Waste Terminal Storage Program
ONWI	Office of Nuclear Waste Isolation
OWI	Office of Waste Isolation
PWR	pressurized water reactor
RH-TRU	remotely handled transuranic-contaminated waste
WIPP	Waste Isolation Project Plant

1. EXECUTIVE SUMMARY

This report discusses the thermal effects that could result from the emplacement of radioactive waste in a geologic repository and the potential for mitigation of those effects by surface cooling of the waste prior to disposal. This Executive Summary lists the main points of the various key technical findings in six brief sections. Section 1.1 addresses the factors that control thermal effects. Section 1.2 covers the overall thermal, thermomechanical, and thermohydrologic impacts determined mainly on the basis of previous studies that assumed 10-year-old wastes. Since the waste loading density in a repository is determined by thermal criteria, thermal criteria that have been published by the Department of Energy (DOE) are discussed next in Section 1.3. Section 1.4 evaluates the mitigating effects of different surface cooling periods on thermal perturbations in the canister boreholes, at the repository level, and in the surrounding geologic setting. In a given waste disposal system, the surface cooling period would be determined by a combination of technical, economic, social, and environmental factors. This report examines mainly the technical factors and from that standpoint reviews the advantages and disadvantages of surface cooling in Section 1.5. These specific points are followed by an overview of the main technical conclusions in Section 1.6.

1.1 CONTROLLING FACTORS OF THERMAL EFFECTS

Thermally induced effects are determined by the waste characteristics, repository structure, and rock properties. Analyses in the literature show that

- For the same waste loading density, spent fuel releases more heat at longer times than reprocessed wastes, which have most of the long-lived plutonium and uranium isotopes removed. Thus the spent fuel will induce much larger long-term thermal impacts than reprocessed high-level radioactive waste.

- Waste inventory studies indicate that a significant fraction of the wastes received by repositories will be more than 10 years old, especially with the expected delays in establishing fully operational repositories until close to the turn of the century.

- Many years of research on salt as a repository medium and several recent investigations on hard rocks are enabling investigators to develop detailed designs of the configurations of the emplacement holes and the room-and-pillar structures. Relatively simple repository geometries, however, have been used in most far-field models. Eventually, the exact depth, size, and shape of the repositories will be determined by the site-specific stratigraphic and regional constraints of the geologic setting as well as the environmental impacts of waste emplacement.

- Thermal properties of the rock formations can be measured fairly accurately in the laboratory and in the field. Mechanical properties measured in intact rock samples are not representative of those of fractured rock masses. Hydrologic properties, especially permeability, have a wide range of uncertainty.

1.2 THERMAL EFFECTS DETERMINED MAINLY ON THE BASIS OF STUDIES OF 10-YEAR-OLD WASTES

The very-near-field, near-field, and far-field thermal effects have been extensively studied for 10-year-old wastes. The main findings are that

- Waste package integrity and maximum waste and canister temperatures depend sensitively on the heat power of the emplaced waste and thermal conductivities of the various components, including backfill or air gaps surrounding the canister. Borehole degradation should be controlled to maintain retrievability for a prescribed period of time.

- Thermomechanical stability of the room-and-pillar structure is determined by the stress changes due to both excavation and thermal loading. Controlling thermal loading and ventilation, and utilizing mining engineering techniques, such as roof bolting, will help to ensure mechanical stability and maintain the safety of personnel during the waste emplacement and retrieval operations.

- The long-term, far-field temperature rise depends mainly on the heat capacity of the rock. Heat capacity is the least site-specific and rock-specific property. This allows us to evaluate and predict long-term thermal effects with cautious optimism.

- Surface uplift from thermal expansion of rock is largest for a spent fuel repository in salt. The surface uplift depends on the cumulative heat energy in the rock formation, which persists over thousands of years.

- Buoyancy flow will also persist for more than 1000 years. Distortion of the convection cells by a regional pressure gradient does not significantly change the travel time for water to move from the repository up to the surface.

1.3 RESEARCH NEEDS FOR THERMAL, THERMOMECHANICAL, AND THERMOHYDROLOGIC CRITERIA

Waste loading criteria used by the DOE in repository design have not included the following considerations.

- Thermohydrologic perturbation is not addressed in the existing far-field criteria. Although safety analysis requires including a description of the anticipated response of the bulk hydrogeologic system to the maximum design thermal loading, the repository loading density itself is not explicitly bounded by the considerations of vertical flow from the repository to the surface. Further investigations are required to elucidate the dependence of design waste loading on thermohydrologic impacts and the predictability of long-term thermohydrologic responses.

- The very-near-field criteria are based on limits on the maximum temperature rises occurring at short times. The relationships between these early-time maxima and the long-term waste package integrity and radionuclide release rates should be carefully evaluated. For salt, brine inflow to the borehole has been observed to be low during the heating period and high after the heat power has been turned off. The potential long-term delay in brine migration due to entrapment in microcracks should be considered.

- Thermoelastic analyses do not predict the behavior of nonelastic fractured rock masses. The rock failure conditions should be evaluated with fracture models that consider the couplings of fracture deformation to thermal expansion of rock blocks and to fluid flow through the fractures.

- Thermomechanical instability induced by tension outside the heated zone should be considered in addition to compressive failure. This should be done for both the near-field room-and-pillar structure and the far-field rock formations. Thermally induced tensile stress may open the fractures and change the permeability.

1.4 EMPLACEMENT WASTE DENSITY AND THERMAL LOADING OF AGED WASTES

The effect of different surface cooling periods on thermal impact can be summarized as follows:

- For constant mass of emplaced material per unit area, surface cooling significantly reduces the thermal impact for reprocessed waste, but only modestly for spent fuel. The main effect of surface cooling is to allow a significant portion of the fission products to decay. For spent fuel containing long-lived actinides, the thermal impacts over thousands of years are not significantly changed by extending the period of surface cooling from 10 to 100 years.

- For constant power density at emplacement, longer surface cooling increases the thermal impact significantly for spent fuel, but only modestly for reprocessed waste. Since most of the design studies and economic analyses are expressed in terms of thermal power densities, conclusions based on 10-year-old waste should not be applied to older wastes unless careful re-evaluations are made.

- If a higher waste density is considered in the design of a repository for older wastes, limitations of loading should be carefully determined by imposing both near-field criteria based on thermomechanical stability considerations and far-field criteria based on thermohydrologic perturbation considerations.

1.5 ADVANTAGES AND DISADVANTAGES OF SURFACE COOLING FROM THE STANDPOINT OF THERMAL IMPACT

Repository design and environmental evaluation should carefully consider the effects of different surface cooling periods.

- Surface cooling allows more concentrated waste emplacement and lower thermal loading. The quantitative changes depend sensitively on the waste type and on the thermal criteria used in determining optimal loading.

- For the region in the vicinity of the waste package and the repository room-and-pillar, the lower thermal loadings associated with older wastes could reduce the short-term temperature rise and the thermomechanical instability.

- Reductions in the near-field thermomechanical perturbations are significant for older wastes in salt and especially in hard rocks. If the near-field criteria determine the waste loadings, the creep analyses for salt and the thermoelastic analyses for hard rocks should be carefully evaluated to determine the optimal waste-loading densities for older wastes.

- If far-field criteria are used, the extension of surface cooling periods will allow only a modest increase in waste density for spent fuel. The balance between a modest reduction in repository spatial requirements and the additional expense of the maintenance of surface storage facilities will be the determining factor in optimizing the duration of surface cooling.

- For reprocessed waste, long surface cooling can effectively lower the far-field thermal effects. From a technical standpoint, it may be advantageous to store reprocessed wastes above ground for up to 100 years to allow a significant fraction of the fission products to decay.

1.6 OVERVIEW OF SURFACE COOLING, WASTE LOADING, AND THERMAL CRITERIA

This report studies the sensitive dependencies of thermally induced effects on waste types, surface cooling, and waste loading. The short-term, near-field thermal impacts are determined by the emplacement heat power of the wastes. The long-term, far-field thermal impacts are determined by the cumulative heat in the rock formations. The extension of the surface cooling period from 10 years to longer periods can lower the near-field thermal impact but have only modest long-term effects for spent fuel. More significant long-term effects can be achieved by surface cooling of reprocessed waste.

Detailed evaluations of the thermal, thermomechanical, and thermohydrologic effects are essential to the quantitative regulation and design of repositories for geologic disposal of nuclear wastes. The optimal waste loading design is determined by the thermal criteria. Criteria that have been used by DOE are based on the thermomechanical stability considerations for the waste package and repository structural components, as well as on the allowable surface uplift due to thermal expansion of the surrounding geologic setting. The vertical buoyancy flow from the repository to the surface has not been considered as a bounding criterion in the determination of repository loading densities.

Current maximum design thermal loading is based on existing criteria for 10-year-old wastes. If a higher waste density is considered in the repository design for older wastes, limitations of loading should be determined by careful considerations of both short-term thermomechanical effects and long-term thermohydrologic impacts.

2. INTRODUCTION

2.1 OVERVIEW

Thermal loading is a principal consideration in the design and evaluation of a repository for geologic disposal of nuclear wastes. The age of the wastes--the length of time between their removal from the reactor cores and their final emplacement in a repository--is a significant factor in determining the waste's heat power at emplacement and the thermal effects on the waste's surroundings. Although many studies in the past decade have demonstrated the importance of thermal effects on all components involved--the waste canisters, the repository structure, and the surrounding geologic setting--most of the studies have focused on the effects of 10-year-old wastes. Because no nuclear waste repositories have yet been constructed, a substantial part of the wastes eventually placed in a given repository would have been stored on the surface much longer than 10 years. The emplacement of older wastes may also be preferred if the thermally induced effects are of major concern. Several European countries (Great Britain, West Germany, Sweden, Belgium) plan to allow their nuclear wastes to cool on the surface for longer periods (25, 40, or 100 years) before emplacing the material in a permanent site.

To evaluate the effects of different surface cooling periods and waste loading densities, it is important to specify the spatial scales and the time spans of concern. In the immediate vicinity of a waste canister, it is desirable to maintain the structural integrity of the waste package and to limit the release of radionuclides. In the near-field, the thermomechanical stability of the shafts and underground openings may be adversely changed if the heat power and the density of wastes are too high. The stability of the repository structure is the main concern for the safety of underground operations during the excavation, emplacement, and retrieval periods. On the regional scale, the thermal impacts persist for thousands of years. Thermal loading of the rock formation may induce surface uplift and may change the hydrologic properties of the rocks. The thermally induced buoyancy gradient may cause vertical movement of the groundwater and thus accelerate the transport of radionuclides from the repository to the accessible environment.

The magnitudes of the thermal, mechanical, and hydrologic impacts depend on the heat power and waste density. The waste density is a key parameter in repository design. Although it is desirable to localize the wastes and minimize the size of a repository, it is also necessary to keep the waste density low to limit the thermally induced impacts. The existing thermal design criteria are developed mainly on the basis of many years of research on salt and several recent investigations on hard rocks as possible geologic settings for a repository for 10-year-old wastes.

2.2 OBJECTIVE OF THIS STUDY

The objective of this study is to evaluate thermal guidelines for optimal thermal loadings, emphasizing the effects of the surface cooling periods of the wastes. As part of this effort, a comprehensive review was made of the available research literature on thermal effects, thermal criteria, and material properties of mined geologic repositories and their environment. Such a review of the sensitivity of the repository and the geologic setting to various parameters is designed to elucidate the significance of these parameters in assessing the repository performance.

Evaluation of the effects of different surface cooling periods of the wastes is particularly important for understanding the optimal thermal loading of a repository. This aspect of the thermal loading has received little quantitative examination in the literature. Therefore, additional calculations (formulas in the Appendix) on surface cooling times were performed to supplement the available data in the literature. The results lead to a clearer understanding of the importance of surface cooling in evaluating the overall thermal effects of a radioactive waste repository.

2.3 THERMALLY INDUCED IMPACTS

To assess the thermal impacts in general, and the effects of different surface cooling periods in particular, it is important to specify the thermal loading, the repository design and operation, and the geologic setting. With the specification of these controlling parameters, one can then evaluate the perturbations of the temperature field, the rock deformation, and the fluid flow and radionuclide migration in the engineered structure and its surrounding environs.

2.3.1 Major Parameters

2.3.1.1 Thermal loading

In order to control the thermal effects in and around the repository, it is crucial to determine the thermal loading. The thermal loading is specified by the cooling and emplacement operations:

- duration of surface cooling periods before repository emplacement,
- waste types and heat power at emplacement, and decay thereafter,
- waste distribution and density within the repository, and
- loading sequence of waste emplacement.

Once these initial conditions have been specified, the thermal effects for a given rock type can be readily determined, at least in terms of bounding values from the known heat generation rates of the particular waste types and the available data on properties of the rock formations. Optimizing the surface cooling of the wastes is thus a vital matter, since it will essentially determine the thermal impact in the repository and its surrounding environment over time.

2.3.1.2 Repository structure and operation

The waste emplacement configuration depends on the repository structure. A repository will have several different shafts for the transportation of personnel, equipment, wastes, and the flow of air for ventilation. From the shafts, main corridors and long storage rooms extend into the surrounding rock formation. The important design parameters for thermal considerations are the geometrical dimensions of

- canister emplacement holes,
- hole pattern and arrays,
- room-and-pillar separation, and
- repository extension, shape, depth, and size.

The radius of the hole and the size of the gap between the canister and the rock are important for the determination of waste, canister, and hole temperatures. The hole pitch (separation distance between holes) and row separation determine the region of rock to be heated by each waste canister. The repository size, shape, and depth determine the waste emplacement capacity and the long-term thermal impact of a waste repository.

For the safety of operation during underground excavation and waste emplacement, ventilation will be maintained throughout the operational life of a repository. The ventilation will remove a fraction of the heat released by the waste. This underground cooling is an extension of surface cooling before emplacement. The requirement that the waste be maintained in a retrievable condition may extend the effective cooling period. NRC's proposed regulation (Code of Federal Regulations, Title 10, Part 60, 1981) would require that the geologic repository operations area be designed so that the entire inventory of waste could be retrieved on a reasonable schedule, starting at any time up to 50 years after waste emplacement operations are complete. According to the same proposed regulation, a reasonable schedule is one that requires no longer than about the same overall period of time that was devoted to construction of the repository and emplacement of the wastes.

2.3.1.3 Geologic setting

Heat released from the wastes in the repository will be dissipated in the surrounding rock formations and will eventually be transferred to the ground surface. Depending on the location, the thermal impacts at a point in the rock formation will first increase and then slowly decrease, so that the host rock will eventually be returned to its original ambient condition, since the heat generation rate of the waste continuously decreases and the heat transfer processes slowly remove the heat. The spatial and temporal distribution of the thermal impacts depend on

- waste thermal loading in the repository,
- rock types,
- thermal, mechanical, and hydrologic properties,
- lithology and boundaries, and
- ambient conditions.

Bedded and domed salt, granite, basalt, shale, tuff, and dry alluvium are among the rock formations being considered for waste repositories. The geologic setting is the last barrier to prevent the escape of radionuclides to the accessible environment. Over the geologic time scale of thousands of years, the geologic setting could be effective in providing the required isolation if a stable formation is chosen and a suitable location is found. However, the uncertainties associated with formation inhomogeneity and variation in rock properties require that conservative and stringent criteria be imposed on the magnitude of thermal perturbations to be allowed in the ambient temperature field, the in situ stress field, and the regional groundwater movement.

2.3.2 Main Effects

2.3.2.1 Temperature rise, stress change, and buoyancy flow

The main concern in the assessment of the thermal impact of the wastes is on the thermally induced phenomena at different locations and various periods of time:

- temperature rise in the very-near-field of the canister-borehole region,
- thermomechanical stability in the near-field of room-and-pillar structure, and
- thermohydrologic perturbation in the far-field over the groundwater basin and the recharge-discharge flow pattern.

After the emplacement of a waste canister, the temperatures of the waste, the canister, and the borehole wall increase very rapidly to quasi-steady values that are maintained over long periods. The primary concern about these elevated temperatures is the maintenance of the integrity of the engineered containment. The stability of the room roof and the pillar support is essential for the safety of operations during the excavation, emplacement, and retrieval periods. The development of a buoyancy gradient over thousands of years is directly related to the effectiveness of the geologic setting in isolating the radionuclides.

2.3.2.2 Couplings between heat, fluid, and rocks

Before proceeding with a quantitative discussion of the findings in the literature on these thermal, mechanical, and hydrologic effects, it is appropriate to point out that most of the studies evaluate waste impact on heat, rock, and fluid in a stepwise, uncoupled approach. For the low-permeability, small-deformation effects expected for the waste impact, the simple approach is usually justifiable. However, it should also be realized that studies employing coupling are needed for better understanding of the physical and chemical processes involved, especially in the near-field. The coupling processes of primary concern are

- thermal-buoyancy flow coupling,
- thermal-stress coupling, and
- flow-stress coupling.

Both water and rocks exhibit a certain number of temperature-dependent properties. The density and viscosity of water are well-known functions of temperature. The gravitational force imbalance between the hot fluid near the repository and the ambient cold fluid in the surrounding formation induces a vertically upward driving gradient on the fluid flow. This may perturb the existing pattern of groundwater flow driven by pressure gradients. Also, the various rock types show varying degrees of expansion and deformation behavior as functions of temperature. Different rock formations may respond in a variety of ways to a thermal-stress coupling; e.g., salt flows plastically, granite fractures absorb thermal stress nonlinearly, and shale shrinks above the boiling point. This in turn may have implications for fluid flow through a permeability-stress coupling. The counterbalance between the fluid pressure and the rock stress gives rise to the coupling between fluid behavior and rock deformation.

The coupling effects may be more easily appreciated in schematic form, which includes solute (e.g., dissolved salt) and radionuclide transport (Fig. 2.1). The thermal loading is clearly the main source for the heat transfer, which influences the rock stress, fluid flow, and chemical processes. In these relationships, the fluid flow is likely to have a rather small effect on the temperature field. Except for the near-field, rock formations chosen for the repository should be nearly impermeable. The rock stress may significantly

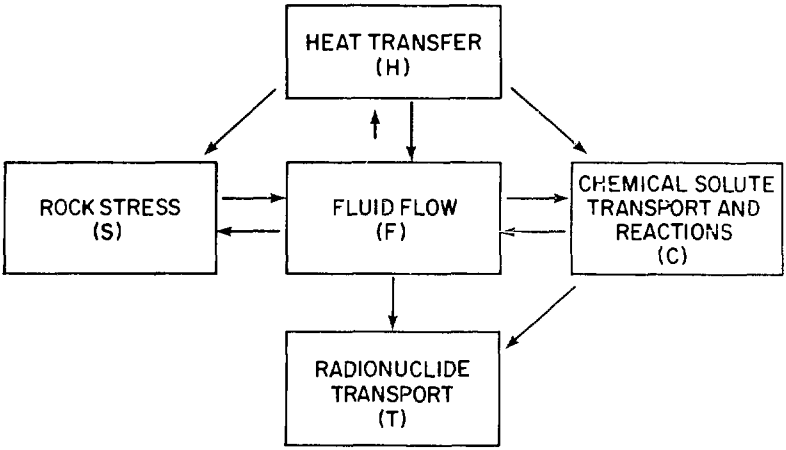

Figure 2.1 Schematic diagram of major factors affecting radionuclide transport from geologic repositories. [XBL 818-3431]

affect the fluid flow through the operation of the permeability-stress coupling, especially for fractured media. In addition, heat may have a strong influence on the rock stress.

The couplings involving chemical solute transport and reactions are important, but will not be included within the scope of this study. The effect of temperature on solubility, sorption, and rates of reaction may be generally known, but further investigation is required to discover their precise degree of influence on the rocks, water, and radionuclides involved in nuclear waste isolation. The couplings between fluid flow and chemical processes are likewise quite important. Fluid flow transports the chemical species, and any dissolution or precipitation will affect the geometry of flow paths, such as the fracture aperture, which in turn will have a significant effect on fluid flow. The heat-rock-fluid-solute-radionuclide couplings are probably critical at the canister-borehole scale. They may also be important at the regional scale, although the processes could be much slower.

2.3.2.3 Effect on radionuclide transport

The focus of all of these interactive processes is the transport of radionuclides through the geologic setting to the accessible environment by fluid flow. It is generally recognized that the principal means of radionuclide transport is the movement of groundwater into, through, and out of the repository. Even though rock stress is important to the understanding of the thermal effects in and around the repository, its influence on radionuclide transport is largely through its effect on permeability for fluid flow. Also, although several chemical processes are quite significant in radionuclide transport (solubility, complexing, sorption, etc.), they basically determine only the extent to which radionuclides can be dissolved and how well they are retarded while being carried by the groundwater. Thus the most conservative way of estimating travel times of the waste to the accessible environment is to calculate the time for the fluid to reach the surface from the repository, assuming perfect solubility of the waste and negligible retardation by complexing, sorption, precipitation, etc.

Fluid transport of the radioactive wastes has important implications for the repository's thermal criteria. Even though the host rock may have a very low permeability, the thermal pulse from the decaying wastes will produce a buoyancy gradient in the groundwater surrounding the repository. The effect of this gradient is to create a driving force for water movement from the repository to the surface at rates above the typical regional flow. Travel paths may become more vertical, and therefore shorter than unperturbed regional flow paths. Such thermohydrologic effects have received inadequate quantitative consideration in waste isolation studies up to now. An important point of the present study is to bring these effects into consideration and to discuss their implications for the thermal criteria of a nuclear waste repository.

2.4 OUTLINE OF THIS REPORT

In Section 3, the temperature rise, thermomechanical stability, and thermohydrologic perturbations due to the presence of a nuclear waste repository are reviewed on the basis of current literature. Most of these discussions are for the case of 10-year-old wastes, which have received the most attention. Abstracts from some of the literature that was surveyed are reproduced in the bibliography at the end of this report. In Section 4, existing thermal criteria for nuclear waste repositories are summarized and evaluated. Both the conceptual and the experimental bases of the criteria are investigated, and further research needs are defined. Section 5 presents a detailed examination of thermal calculations concerning different surface cooling periods. It is shown that the benefit of a longer surface cooling period is substantial for reprocessed high-level wastes but is marginal for spent fuel. Great care must be exercised if one attempts to scale results of repository design studies and calculations based on 10-year-old wastes to older wastes.

3. THERMAL, THERMOMECHANICAL, AND THERMOHYDROLOGIC EFFECTS OF 10-YEAR-OLD WASTES

The need to manage existing and projected radioactive wastes is noted in many studies. Although the effects of the duration of surface cooling periods have been considered in several parametric studies (which are reviewed in Section 5), most studies treat 10-year-old waste as the baseline case. Therefore, this section is devoted mainly to the review of the effects of storing 10-year-old wastes. Since the decay heat characteristics are important for the assessment of waste impacts, several aspects of decay heat generation are discussed in Section 3.1, along with waste inventory projections. Section 3.2 then reviews relevant features of the mined repository design, and Section 3.3 summarizes the thermal, mechanical. and hydrologic properties of the rock formations being considered as hosts for waste repositories. Following this are detailed reviews of the thermal impact of 10-year-old wastes on the temperature rise at the canister-hole scale in Section 3.4, the thermomechanical stability of the room-and-pillar structure in Section 3.5, and the thermohydrologic perturbation of groundwater movement at the regional scale in Section 3.6. This survey of thermal impacts not only summarizes results of current waste heat studies, but also serves as a basis for discussing the effects of different surface cooling periods in later sections.

3.1 WASTE HEAT SOURCE CHARACTERIZATION

The magnitude of thermally induced impacts is controlled by the spatial and temporal distributions of waste heat sources. The amount of heat for a given amount of waste depends on the waste composition, which in turn depends on the nuclear fuel cycle employed. Some details concerning the heat source characteristics of the most widely studied fuel cycles are given in Sections 3.1.1 to 3.1.4. Section 3.1.5 gives a brief account of the projections of waste inventories, since the total amount of waste to be managed depends on the expected number of nuclear reactors needed for electricity generation.

3.1.1 Decay Heat Generation of Nuclear Fuel Cycles

The composition of nuclear waste depends on reactor operation and the nuclear fuel cycle. In the United States, the commercial nuclear reactors used are principally light water reactors (LWR), which use ordinary (light) water both as a circulating coolant and as a neutron moderator. This report primarily discusses the impacts of LWR. The LWR cores contain narrow rods of UO_2 pellets. The uranium fuel consists of 97-98% U-238, 2-3% U-235, and trace amounts of other isotopes. During irradiation, the U-235 fissions and releases energy, and in the process some of the U-238 is converted to Pu-239 by neutron capture; this plutonium isotope can also fission and release energy. Different reactors with alternative fuel compositions, cooling fluids, and modulating materials have also been developed. These include the heavy-water-modulated, natural-uranium-fueled reactor (HWR), the liquid-metal-cooled, fast breeder reactor (FBR), and the high-temperature, gas-cooled, graphitemodulated reactor (HTR). The International Nuclear Fuel Cycle Evaluation Conference (INFCE, 1980) compares the thermal impacts of the different fuel cycles.

After a specified "burnup" time, the fuel is removed from the reactor, and the remaining uranium or plutonium, or both, can be recycled by fuel reprocessing if desired. Three cycles for LWP have been studied extensively by the NRC, the Environmental Protection Agency (EPA), and the DOE for their impact on commercial high-level waste management:

- <u>Spent Fuel Cycle</u>: This is the "throwaway" or once-through cycle. The irradiated fuel assemblies from an LWR core are disposed of directly as waste after being removed from the reactor. The spent fuel (SF) contains both uranium and plutonium as waste components.

- <u>Uranium-only Recycle</u>: This is a "partial" recycle operation. The spent fuel from an LWR is reprocessed by removing most of the uranium and plutonium in order to recycle the uranium for further use as fuel. The reprocessed high-level waste (HLW) contains only trace amounts of uranium and plutonium.

- <u>Uranium and Plutonium Recycle</u>: This is a "total" recycle, in which both uranium and plutonium are recovered from the spent fuel to form a "mixed oxide" (MOX) of UO_2 and PuO_2 for further use as fuel. With effectively more extensive burnup in a MOX cycle, the trace amounts of uranium and plutonium remaining in the MOX reprocessed waste are higher than those contained in the U-only recycle.

These fuel cycles produce waste types of different composition and decay heat generation rates. Figure 3.1 shows the power densities for each of the three cycles. The wastes all originate from the same amount of fuel, MTHM (metric ton of heavy metal U), charged to a pressurized water reactor (PWR). Figure 3.1 also compares the results from several different reports, which are discussed in Section 3.1.3.

3.1.2 <u>Fission Products and Actinides</u>

Radioactive waste from any of the nuclear fuel cycles is composed of two main components: fission products (such as Sr-90 and Cs-137) and transuranic actinides (such as Pu, Am, and Cm). For the first few hundred years, the fission products are the principal contributors to the decay heat generation. This fission product activity is largely independent of the fuel mix, and the heat generation is similar for all three cases of the nuclear fuel cycle. However, the quantities of actinides vary widely according to the fuel cycle. After the first few hundred years, these relatively long-lived heavy transuranic actinides become the major contributors to the decay heat generation. Therefore, in the long term it is the decay heat characteristics of the actinides and their daughters that control the heat generation of the radioactive wastes.

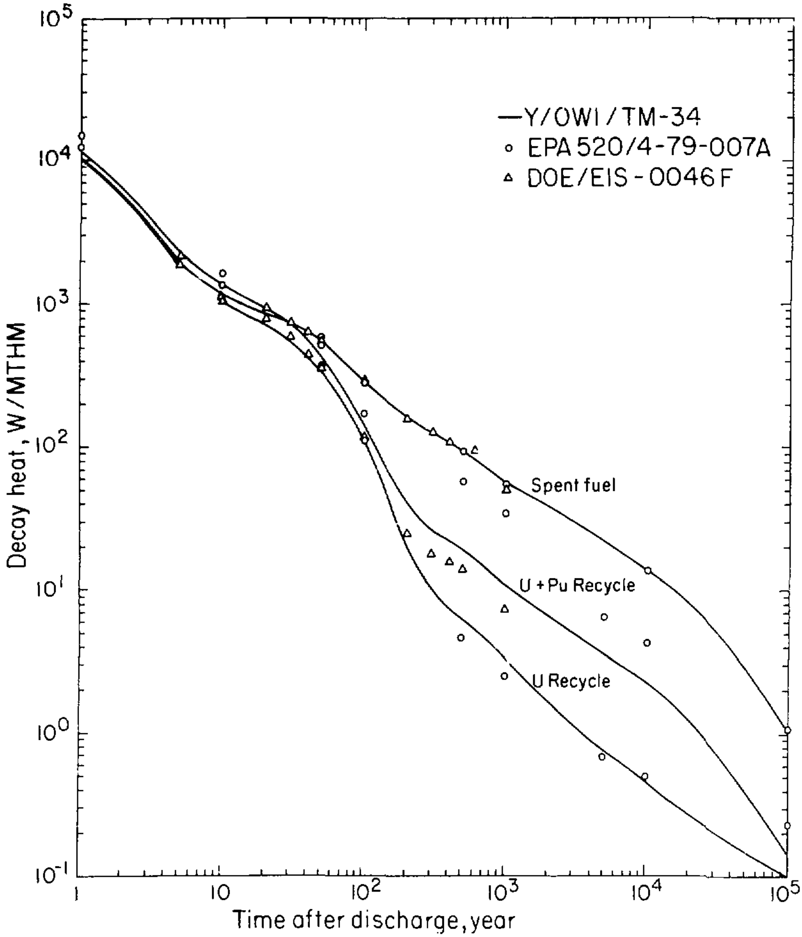

Figure 3.1 Decay heat power from 1 MTHM of a PWR for each of three nuclear fuel cycles (plotted from the data of EPA, 1977; Kisner et al., 1978; DOE, 1980a). [XBL 818-3434]

As a result of reprocessing, it is assumed that 99.5% of the uranium and plutonium originally in the spent fuel is removed, but other actinides formed by the transmutation of uranium and plutonium are retained in the high-level waste. The removal of plutonium especially causes the decay heat rate to be much lower after the first few hundred years. Most of the fission products are retained in the high-level waste, with the exception of all the tritium, the noble gases (He, Kr, and Xe), and 99.9% of the volatile elements (I and Br) that escape during reprocessing.

3.1.3 Details of Different Recycle Options

In Figure 3.1 the solid curves are plotted from tabulated data given in Kisner et al. (1978). The U-only recycle contains only the reprocessed high-level waste (HLW) without the plutonium. The U + Pu recycle is based on the assumed 1:3 blend of MOX fuel and UO_2 fuel. Both recycles correspond to the equilibrium values after many reprocessing charge-discharge cycles. For comparison, Figure 3.1 also includes points plotted from the tabulated data in reports by the EPA (1977) and the DOE (1980a). The spent fuel data are consistent among all three data sets. For the U + Pu HLW cycle, the EPA report assumed a 1:2 MOX:UO_2 ratio. The same study also included the cladding waste together with the HLW. In the draft Generic Environmental Impact Statement (GEIS) on Commercial Waste Management (DOE, 1979b), all three cycles were considered. In the final Environmental Impact Statement (EIS) (DOE, 1980a), the U-only recycle case was deleted because of the low likelihood that it would be implemented. The differences in Figure 3.1 among the results of the three reports reflect the sensitivity of the contents of long-lived plutonium in the wastes to different assumptions in fuel mix, reactor burnup, and reprocessing treatment. Storch and Prince (1979) discuss the different source term data sets and other potential fuel cycles.

Figures 3.2 and 3.3 give a detailed comparison of the cycles of a PWR and a boiling-water reactor (PWR). Although the PWR and the BWR differ in detailed design, modulation, and cooling operations, the power functions are quite similar. The PWR values are slightly higher than the corresponding BWR values because of the higher fuel burnup (PWR: 33,000 MW(t)-d/MTHM, BWR: 27,500 MW(t)-d/MTHM). The PWR is the reference reactor in the previously mentioned EPA and DOE studies and most other studies.

Figures 3.2 and 3.3 both show that there is little difference between the U-recycle and no-recycle curves. The no-recycle curve is a subfuel cycle corresponding to the special case when the discharged waste is reprocessed only once and the reprocessed uranium is not used in fabricating additional fuel elements. Although isotope compositions of reprocessed uranium and fresh uranium are different, the small difference between the U-recycle and no-recycle curves reflects the small contribution of uranium to the total heat generation rate of the wastes.

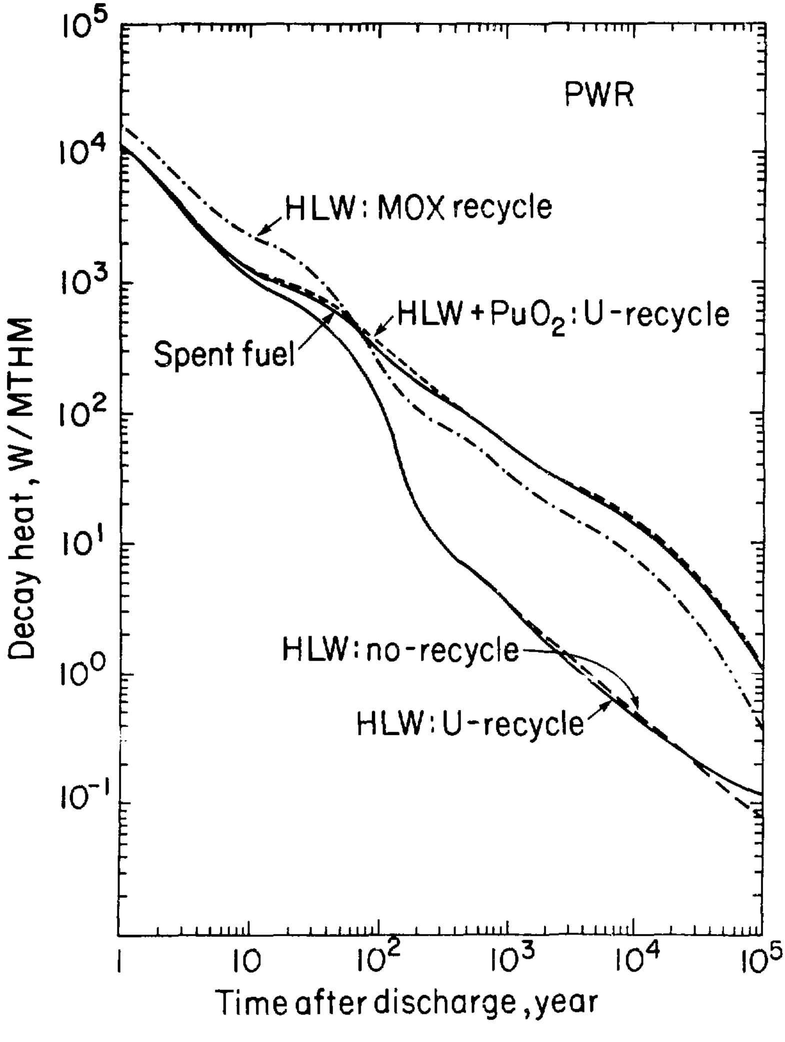

Figure 3.2 Decay heat power for different nuclear fuel cycles for a PWR (plotted from the data of Kisner et al., 1978). [XBL 818-3433]

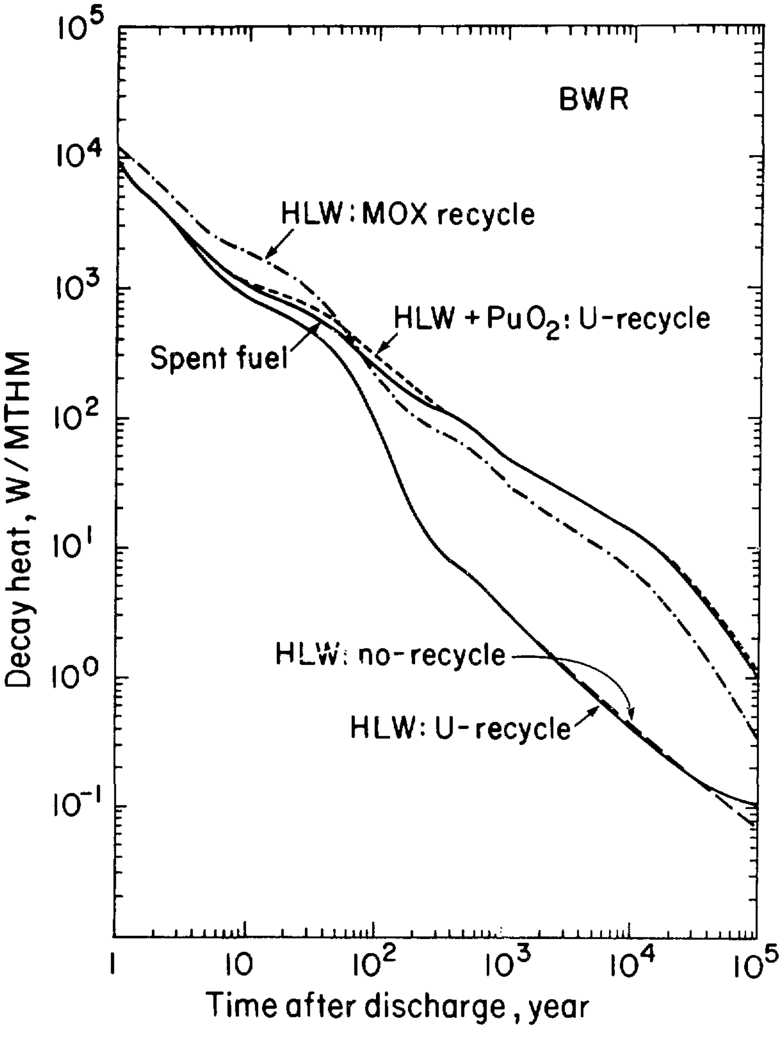

Figure 3.3 Decay heat power for different nuclear fuel cycles for a BWR (plotted from the data of Kisner et al., 1978). [XBL 818-3432]

There is also little difference between the HLW + PuO_2 of U-recycle and the spent fuel curves. The HLW + PuO_2 curve represents the sum of U-only equilibrium recycled waste and the reprocessed and separated plutonium stored together in the repository. The reprocessing treatment of spent fuel does not significantly change the total sum of fission products in the reprocessed waste plus plutonium recovered from the spent fuel.

The long-term heat generation rate is sensitive to the plutonium content. With successive generations of MOX burnup, the total plutonium discharge increases with each cycle (Kee et al., 1976). For the MOX curves in Figures 3.2 and 3.3, an equilibrium condition of the plutonium recycle was assumed (Kisner et al., 1978). The amount of actinides in equilibrium MOX wastes approaches the amount in spent fuel over the whole lifetime of the wastes. The total MOX cycle, however, is not expected to be used in the near future. The MOX-UO_2 ratio of 1:2 in the EPA (1977) study and the 1:3 ratio in the GEIS study (Science Applications Inc., 1976) reflect the likelihood that a low percentage of the MOX cycle will be used for electricity generation.

For reference, Figure 3.4 compares the heat generation rate of defense high-level waste (DHLW), described by Savannah River Laboratory (DuPont, 1980), with that of commercial wastes. The DHLW is assumed to be 15 years out of reactor at emplacement in the repository. The DHLW from plutonium production decays faster than the commercial HLW after 10 years of emplacement.

We will concentrate mainly on the commercial spent fuel cycle. The spent fuel cycle has the highest decay heat rate over long periods of time. For the same amount of wastes, the long-term impact from spent fuel is expected to be most severe. For bounding calculations, however, we will also consider the HLW no-recycle case.

3.1.4 Waste Packages and Thermal Properties

The individual spent fuel canister usually contains one fuel assembly. For a PWR, the waste content of one fuel assembly is 0.4614 MTHM. This means that for 10-year-old wastes, the heat power is 0.55 kW/canister. For a BWR, the waste content is 0.1833 MTHM, and the 10-year-old heat power is 0.18 kW for each spent fuel assembly (Kisner et al., 1978). The possibility of packing more than one fuel assembly into one canister was also considered by several groups; for example, four PWR assemblies per canister by the EPA (1977) and three PWR assemblies per canister by Altenbach (1978).

For reprocessed HLW, the waste content in each canister can be controlled in the reprocessing procedure. Indeed, the maximum canister load is a design parameter that can be determined by a very-near-field thermal criterion. For 10-year-old HLW, high emplacement heat-power values have been used in earlier studies; for example, a 5-kW/canister value in Cheverton and Turner (1972) and Callahan et al., (1975), and 3.95-kW/canister in EPA (1977). However, there is a general trend to lower the canister heat power, reflecting the concern

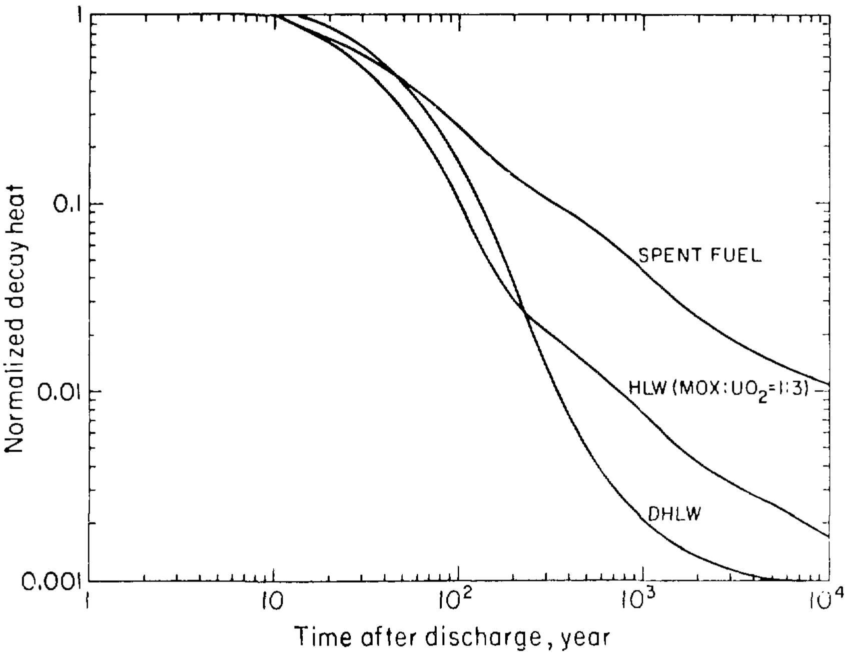

Figure 3.4 Relative decay heat powers of commercial spent fuel, commercial high-level waste (HLW), and defense high-level waste (DHLW) (plotted from the data of Kisner et al., 1978; DuPont, 1980).
[XBL 822-1868]

Table 3.1. Description of Reference Canistered Waste.

Characteristics	SF	HLW	DHLW
Waste Description			
Active length (m)	3.7	2.4	2.3
Active volume (m^3)	NA	0.18	0.63
Age of waste (yr)	10[a]	10[a]	15[a]
Thermal loading (kW/canister)	0.55[b]	2.16[c] or 1.0[d]	0.31[e]
Canister Dimensions			
Outer diameter (m)	0.356[f]	0.324[g]	0.610[h]
Inner diameter (m)	0.337	0.305	0.591
Length (m)	4.7	3.0	3.0
Materials			
Waste	UO_2[j]	glass[j]	glass[j]
Filler in canister	helium	air	air
Canister	carbon steel	304L stainless steel	304L stainless steel

(NWTS, 1980)

[a] At emplacement (years after discharge from reactor).

[b] Heat generation rate for a single PWR assembly 10 years out of the reactor. BWR assemblies would have a lower heat generation rate but this value has been chosen as its use results in predicting maximum temperatures in a repository.

[c] Heat generation rate for commercial HLW used in salt and tuff studies to date.

[d] Heat generation rate for commercial HLW used in granite and shale studies to date.

[e] Maximum expected thermal loading for Savannah River Plant wastes; many canisters will have a lower loading.

[f] Nominal 14-inch schedule 30 carbon steel pipe.

[g] Nominal 12-inch schedule 40s 304L stainless steel pipe.

[h] Nominal 24-inch schedule 20 304L stainless steel pipe.

[j] The choice of waste form for these calculations was based on their advanced state of engineering development. The calculated environments outside the waste forms are insensitive to the details of the waste forms themselves other than heat output and physical dimensions.

over the waste heat impacts. In the interim report generated by the Reference Repository Conditions Interface Working Group in the National Waste Terminal Storage (NWTS) Program (NWTS, 1980), the values 2.16 kW/canister and 1.0 kW/canister were used, as shown in Table 3.1.

The spent fuel waste form consists of an assembly of fuel rods coated with zircalloy cladding. With low thermal power per canister, high cladding thermal conductivity (Table 3.2), and additional radiative heat transfer among the fuel rods (Cox, 1977), the spent fuel temperature rise is lower in most of the cases studied than the reprocessed HLW temperature rise.

The reprocessed HLW waste form depends on the solidification process employed. The low thermal conductivity and leach resistance of the calcined type of HLW (formed by drying liquid HLW at high temperature) can be improved upon by the vitrification process, which incorporates HLW in a glass or ceramic form. Embedding waste-containing beads within a metallic matrix has been suggested to improve the thermal conductivity of reprocessed wastes.

In addition to carbon steel and stainless steel, titanium and copper are potential candidates for canister material. The canister wall may also be thickened to resist corrosion and to withstand the high temperatures. Most studies assume that the canister is a cylinder. However, rectangular-shaped canisters for spent fuel assemblies have also been considered. Outside the canister, an overpack can be added for additional protection. In addition to

Table 3.2. Thermal Properties of Waste Packages.

Material	Thermal Conductivity (W/m-°C)	Specific Heat (J/kg-°C)	Density (kg/m^3)
UO_2	7.01-2.30[a]	234-314[a]	10960
Zircalloy Cladding	14.1-20.4[b]	293-360[b]	6600
Vitrified HLW	1.21	837	2995
Calcined HLW	0.35	837	1297
Stainless Steel 304L	16.4	461	7817
Carbon Steel	45.0	461	7849

(Science Applications, Inc., 1978)

[a] 150-1650°C.

[b] 20-650°C.

metals, ceramics, graphite and carbon materials, a wide variety of glasses and specially selected cements are being studied as potential overpack materials (DOE, 1980a).

3.1.5 Waste Inventory Projections

3.1.5.1 Nuclear capacity growth schedule

Projections of the amounts of radioactive waste are required in order to determine the number and size of waste repositories. The projections are determined by the growth rate of nuclear power generation capacity. Table 3.3 shows the sensitive dependence of the total waste accumulated through year 2040 on the five nuclear power growth scenarios studied in the DOE Final Environmental Impact Statement (EIS) for the management of commercially generated radioactive waste (DOE, 1980a). The power growth curve for Case 3 in the EIS, together with the projections by the Energy Information Administration (EIA) (quoted by Storch and Prince, 1979) and the earlier estimates by EPA (1977) and the Office of Waste Isolation (OWI) (Kisner et al., 1978) are shown in Figure 3.5. After the peak capacity at year 2000, it was assumed that no reactors were to be started up and that the remaining reactors were to be allowed to complete their assumed 40-year life before being decommissioned.

Table 3.3. Total Nuclear Waste Disposal Requirements Accumulated Through Year 2040.

Case	Nuclear Power Growth Assumption	Total Energy Generated (GW(e)-Yr)	Total Spent Fuel Discharged (MTHM)
1	Present inventory only. Reactors shut down in 1980	200	10,000
2	Present capacity (50 GW(e)) and normal reactor life	1300	48,000
3	250 GW(e) by year 2000 and normal reactor life (no new reactors after year 2000)	6400	239,000
4	250 GW(e) by year 2000 and steady-state capacity to year 2040 (new reactors to maintain output)	8700	316,000
5	250 GW(e) by year 2000 and continuously expanding to 500 GW(e) by year 2040	12,100	427,000

(DOE, 1980a)

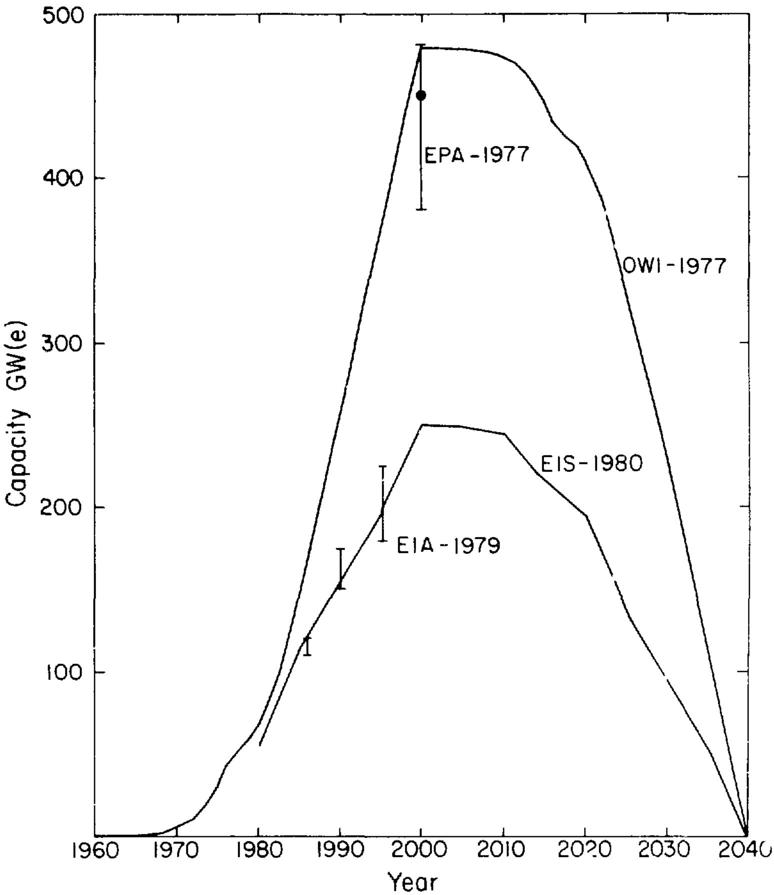

Figure 3.5 Projections of nuclear power growth through 2040 (plotted from the data of EPA, 1977; Kisner et al., 1978; Storch and Prince, 1979; DOE, 1980a). [XBL 818-3435]

The maximum of 480 GW(e) capacity at year 2000 in the OWI projection was based essentially on the estimated available uranium resources in 1977. The U. S. nuclear growth capacity has been lowered in recent years with changes in national energy plans, environmental and regulatory constraints, and electricity demands. The EIA forecasts and the peak capacity of 250 GW(e) at year 2000 considered by the EIS are consistent with the projection of 175.2 GW(e) in 1994 (quoted by Joy et al., 1980) and the projection of 183 GW(e) in year 1995 (DOE, 1979a) based on utility plans for reactors in operation, under construction, or on order.

3.1.5.2. Estimates of repository inventories

Figure 3.6 shows the estimates of cumulative spent fuel discharge from nuclear reactors for the projections of the EIS-Case 3 (DOE, 1980a), the EPA (1977) study, and the projection based on utility plans (DOE, 1980b). The cumulative spent fuel of 239,000 MTHM requires up to four repositories, each having the assumed size of 8.1×10^6 m^2 (2000 acres), a thermal loading of 10 W/m^2 (40 kW/acre) of 10-year-old waste, or a waste density of 8.3×10^{-3} $MTHM/m^2$ (34 MTHM/acre) (DOE, 1980c).

The repository waste inventory depends on the repository startup date, repository receiving capacity, and the backlog of wastes. With the system logistics considered in the EIS, an example of the spent fuel repository inventory is shown in Figure 3.7 for three assumed repository startup dates. (The repository inventory refers to the total inventory of wastes, not the amount of storage in an individual repository.) With repository startup at year 2010, Figure 3.8 shows that most of the spent fuel received by the repository will be much older than 10 years. For the case of a 2030 startup, the minimum age is 19 years. The estimated range of startup dates for the first repository by DOE is 1998-2004 (DOE, 1981).

For the remainder of Section 3 we will assume 10-year-old wastes for the calculations. However, in Section 5, the effects of older wastes will be considered in discussing the optimization of surface cooling periods.

3.2 MINED REPOSITORY DESIGN

The initial United States repository design study was Project Salt Vault in Lyons, Kansas (Bradshaw and McClain, 1971; Cheverton and Turner, 1972). Experiments and design studies in the 305-m (1000-ft) deep bedded salt repository were conducted for reprocessed HLW storage. Since then, modeling studies have been performed to characterize the thermal environment in different media for the storage of spent fuel and reprocessed HLW. These include the engineering design studies of a reprocessed HLW repository in domed salt (Stearns-Roger Engineering Co., 1979), a spent fuel repository in bedded salt (Kaiser Engineers, 1978a,b), and a spent fuel repository in basalt (Kaiser Engineers et al., 1980), as well as generic and scoping studies for granite (Lindblom et

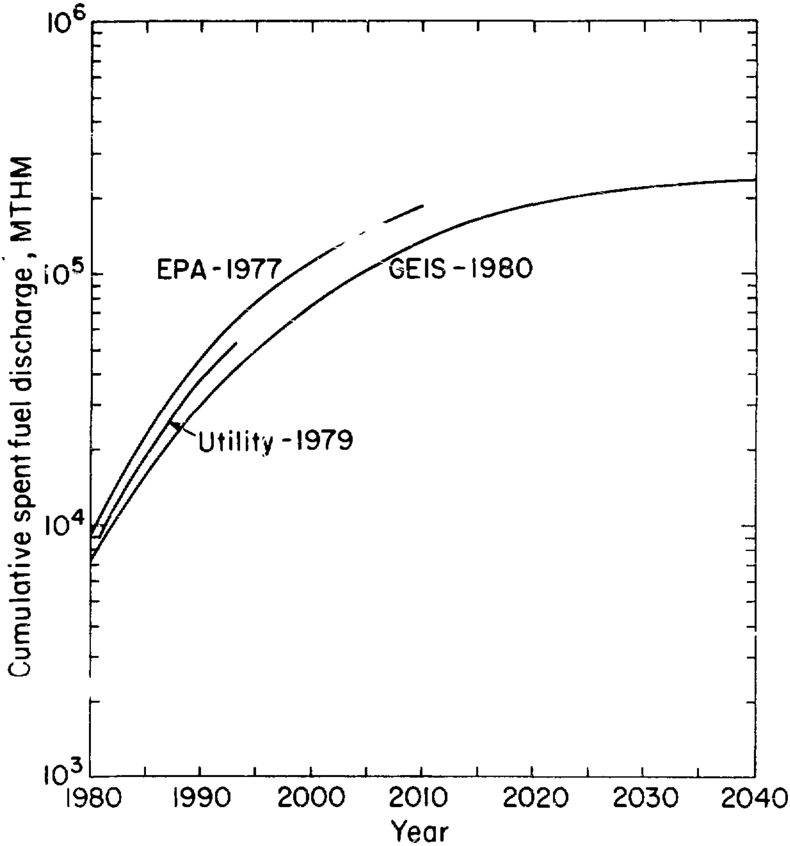

Figure 3.6 Estimates of cumulative SF fuel discharge through 2040 (plotted from the data of EPA, 1977; DOE, 1980a,b). [XBL 818-3436]

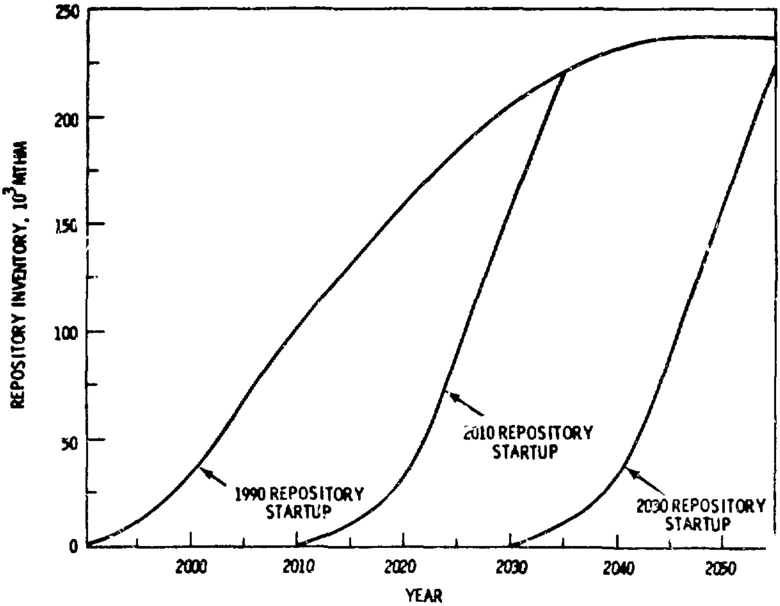

Figure 3.7 Repository inventory accumulation for the SF cycle in EIS Case 3 (DOE, 1980a). [XBL 819-11585]

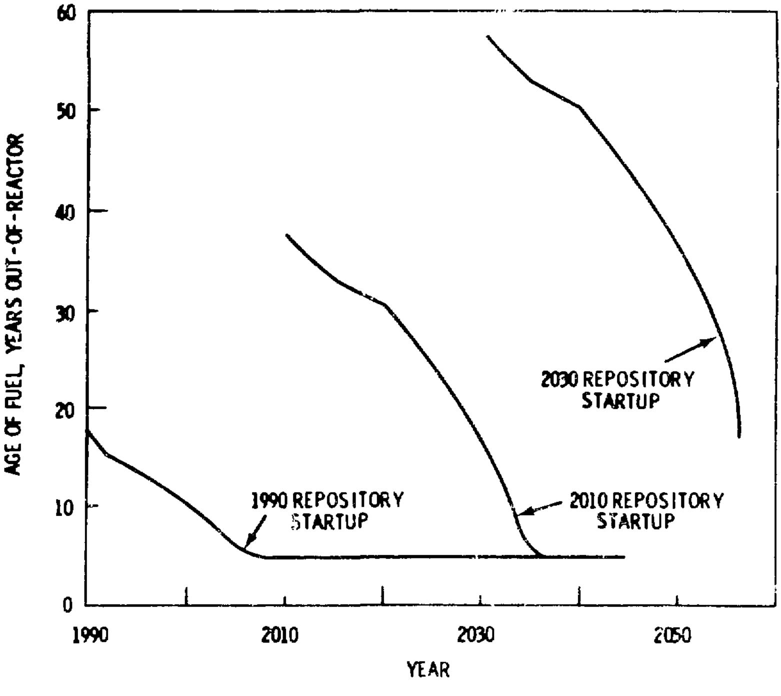

Figure 3.8 Age of fuel entering repository for the SF cycle in EIS Case 3 (DOE, 1980a). [XBL 819-11586]

al., 1977), shale (Thomas et al., 1981), tuff (Bulmer and Lappin, 1980), and dry alluvium (Smyth et al., 1979). Reference repository conditions for the first five rock types are currently being developed in a DOE/NWTS study. The various features of repository structure (emplacement hole, room-and-pillar configuration, overall repository geometry) to be tabulated in this section are from an interim paper that summarizes the activities of the NWTS Reference Repository Conditions Interface Working Group to 1980 (NWTS, 1980). The completion of generic reference conditions for salt, granite, basalt, shale, and tuff will be published as DOE/NWTS reports in the near future (G. E. Raines, personal communication, 1982).

3.2.1 Emplacement Hole Description

Current reference repository designs specify placing waste canisters in vertical holes drilled into the storage room floors. Table 3.4 summarizes the characteristics of the emplacement hole (also referred to as the borehole) for different rock types. One canister per hole is assumed. The SF hole is deeper than the HLW hole to accommodate the longer canister required (see Table 3.1). Additional borehole characteristics not included in Table 3.4 are the carbon steel liner for holes in salt and the use of grout for holes in basalt. After canister emplacement, the hole is plugged with either concrete or backfill material. More detailed designs are being studied to use an engineered backfill and liner to provide additional protection for the waste package and to maintain the retrievability option; usually this will require a larger diameter hole. The main thermal concern is to design engineered structures with high thermal conductivities in order to avoid excessive temperature rises in the canister and waste. Crushed rock and small air gaps are poor thermal conductors (see discussion in Section 3.4.1 on backfill and air gaps around the canister). Typically, the thermal conductivity of crushed rock is one-tenth that of the uncrushed materials.

3.2.2 Room-and-Pillar Dimensions

The geometrical dimensions of the storage rooms and the separation of the tunnels are two important factors in determining the stability of the excavation and its convenience for underground operations. Table 3.5 shows the room-and-pillar dimensions and the arrangement of canisters in the storage room for different rock types. The pillar width has more than doubled from the 7.6-m value considered in Cheverton and Turner (1972) for a bedded salt HLW repository. Many other room-and-pillar arrangements and canister patterns have been studied for the thermal environment. The main concerns of room-and-pillar design have been the room deformation (roof sag, wall convergence, floor heave) in soft rocks and the possibility of rock fracturing in hard rocks during the repository operation-retrievability periods. These are the main factors considered in determining the allowable thermal loading in a repository.

Table 3.4. Emplacement Hole Description.

Rock / Waste	Salt SF	Salt HLW	Granite SF	Granite HLW	Basalt SF	Basalt HLW	Shale SF	Shale HLW	Tuff SF	Tuff HLW
Hole Depth (m)	7.0	5.5	6.7	5.0	6.4	NA	7.0	5.5	8.0	6.0
Hole Diameter (m)	0.54	0.54	0.56	0.52	1.15	NA	0.51	0.51	0.41	0.37
Backfill Thickness (m)	0.051	0.051	0.1	0.1	0.152	NA	0.051	0.051	0.025	0.025
Backfill Material	crushed salt	crushed salt	crushed granite	bentonite	tailored backfill	NA	crushed shale	crushed shale	air	air

(NWTS, 1980)

Table 3.5. Storage Room Description.

Rock / Waste	Salt SF	Salt HLW	Granite SF	Granite HLW	Basalt SF	Basalt HLW	Shale SF	Shale HLW	Tuff SF	Tuff HLW
Room Width (m)	5.5	5.5	7.5	7.5	4.3	NA	5.5	5.5	7.5	5.0
Room Height (m)	6.4	5.5	7.0	7.0	6.1	NA	6.4	5.5	7.0	5.0
Pillar Width (m)	21.3	18.3	22.5	22.5	32.3	NA	18.0	18.0	30.0	20.0
Canister Pitch (m)	1.67	3.66	1.83	2.67	3.66 or 1.22	NA	2.34	2.85	1.19	3.50
Canister Rows Per Room	2	1	2	2	1	NA	2	1	2	1
Row Separation (m)	1.67	NA	2.5	2.5	NA	NA	1.67	NA	2.5	NA

(NWTS, 1980)

3.2.3 Repository Configuration

The areal thermal loading and the repository depth below surface for the NWTS reference repositories are shown in Table 3.6. The average areal loading includes shafts, drifts, and pillars, whereas the local areal loading refers only to the space within the storage room. The thermal loading value of 25 W/m^2 (100 kW/acre) for salt, granite, and tuff is two-thirds of the 37-W/m^2 (150-kW/acre) value that was considered in the mid-1970s. The latter value is adjusted downward from the maximum permissible thermal loading of 39 W/m^2 (158 kW/acre) based on salt temperature considerations (Cheverton and Turner, 1972). The NWTS values for salt and other rocks, however, are high, especially for spent fuel repositories.

Table 3.6. NWTS Areal Loading.

Rock	Salt		Granite		Basalt		Shale		Tuff	
Waste	SF	HLW	SF	HLW	SF	HLW	SF	HLW	SF	HLW
Depth (m)	600	600	1000	1000	1000	NA	600	600	800	800
Local Areal Thermal Loading (W/m^2)	25	25	20	25	12.3	NA	10	10	25	25
Average Areal Thermal Loading (W/m^2)	15	<25	<20	<25	8.2	NA	8	8	<25	<25

(NWTS, 1980)

Table 3.7 presents the GEIS values (Science Applications, Inc., 1978) based on a thermomechanical analysis during the 25-year retrievability period for spent fuel and the 5-year retrievability period for HLW. Long-term, far-field concerns for surface uplift have also imposed a constraint on the allowable thermal loading for salt (Lincoln et al., 1978; Llewellyn, 1978). Areal thermal loading is the most important parameter in determining the repository design. In Section 4, we discuss the controlling factors for determining the allowable thermal loading and present the detailed results on allowable thermal loadings in the draft GEIS and final EIS studies.

A single mined layer at a given desirable depth has been considered in most of the repository design studies. Such a configuration is ideal for layered formations (such as bedded salt), which are relatively thin but areally extensive. For intrusive formations, such as salt domes and granitic plutons, the lateral extension is limited, and either a multilayer repository structure or deeper boreholes with stacked canisters may be required (Just 1978; Kevenaar et al., 1979).

Table 3.7. GEIS Areal Loading Based on Near-Field Thermomechanical Stress Studies.

Rock	Salt		Granite		Basalt		Shale	
Waste	SF[a]	HLW[b]	SF[a]	HLW[b]	SF[a]	HLW[b]	SF[a]	HLW[b]
Areal Loading (W/m^2)	9	37	20	47	20	47	14	30

(Science Applications, Inc., 1978)

[a] 25-year retrievability.

[b] 5-year retrievability.

The repository depths considered in most of the studies range from 300 m to 1000 m. The areal extensions are typically 1-3 km. A disk with a 1.6 km radius covers an area of 2000 acres. The 2000-acre repository has been considered in a number of studies (EPA, 1977; Kaiser Engineers, 1978a; DOE, 1980a).

3.3 ROCK FORMATION PROPERTIES

In addition to the waste heat source and repository dimensions, the thermal impacts depend on the rock properties of the geologic formations surrounding the repository. This section discusses the thermal, mechanical, and hydrologic parameters controlling the waste impacts.

3.3.1 Thermal Properties

The baseline thermal properties of four major rock types (salt, granite, basalt, and shale) considered in the GEIS (DOE, 1979b) and the ranges of values of tuff considered by Tyler (1979) and of dry alluvium considered by *Smyth et al.* (1979) *as candidate repository host rocks are summarized in* Table 3.8. For conductive heat transfer, the controlling parameters are the thermal conductivity and the volumetric heat capacity (specific heat times density). For the short-term temperatures near the canister-rock interface, the temperature rise is inversely proportional to the thermal conductivity. For the average temperature rise at the repository level, the temperature rise is determined by the product of thermal conductivity and volumetric heat capacity. The volumetric heat capacity is the controlling factor for the long-term, far-field thermal response. The rate of heat conduction is governed by the thermal diffusivity, i.e., the ratio of thermal conductivity to volumetric heat capacity.

The thermal properties of the rocks around the canisters can be determined by in situ heating experiments. In most cases, the in situ values agree with the values measured in the laboratory. These thermal properties can be

Table 3.8. Thermal Properties of Rocks.

Rock	Thermal Conductivity[a] (W/m-°C)	Specific Heat[b] (J/kg-°C)	Density (kg/m^3)
Salt	2.08-6.11	840-920	2128
Granite	1.99-2.85	880-960	2640
Basalt	1.16-1.56	710-920	2880
Shale	1.47-1.68	800-880	2560
Tuff (welded)	1.2 -1.9	800-900	2000-2400
(nonwelded)	0.4 -0.8	800-1700	1500-2100
Alluvium (indurated unsaturated)	1.0 -1.2	1000	1700

(DOE, 1979b; Smyth, et al., 1979; Tyler, 1979)

[a] Over the temperature range 0-400°C (except tuff and alluvium).

[b] Over the temperature range 0-200°C (except tuff and alluvium).

functions of temperature. The temperature-dependence of the thermal conductivities and volumetric heat capacities of four different rocks are shown in Figures 3.9 and 3.10. In comparison with thermal conductivities, the volumetric heat capacities are less sensitive to both the temperature and the rock types. The temperature-dependence of the thermal properties is considered in most detailed calculations of the very-near-field temperatures, where the spatial variation (temperature gradient) is high at early times.

The thermal properties of the rocks depend on their mineral compositions. For example, a salt-shale mixture containing 20% shale has a thermal conductivity of 3.2 W/m°C at 100°C compared to 4.2 W/m°C for pure salt at the same temperature (Cheverton and Turner, 1972). The granite in Stripa, Sweden, with its high quartz content, has a high value of 3.23 W/m°C for thermal conductivity at 100°C (Pratt et al., 1977; Chan et al., 1980).

The thermal properties of the rocks also depend on their water content. For example, a nonwelded tuff, which has a high water content, generally has a lower thermal conductivity and higher heat capacity than a welded tuff, which has less porosity and a more dense and compacted structure (Tyler 1979). The unsaturated rocks usually have lower thermal conductivity because the air that

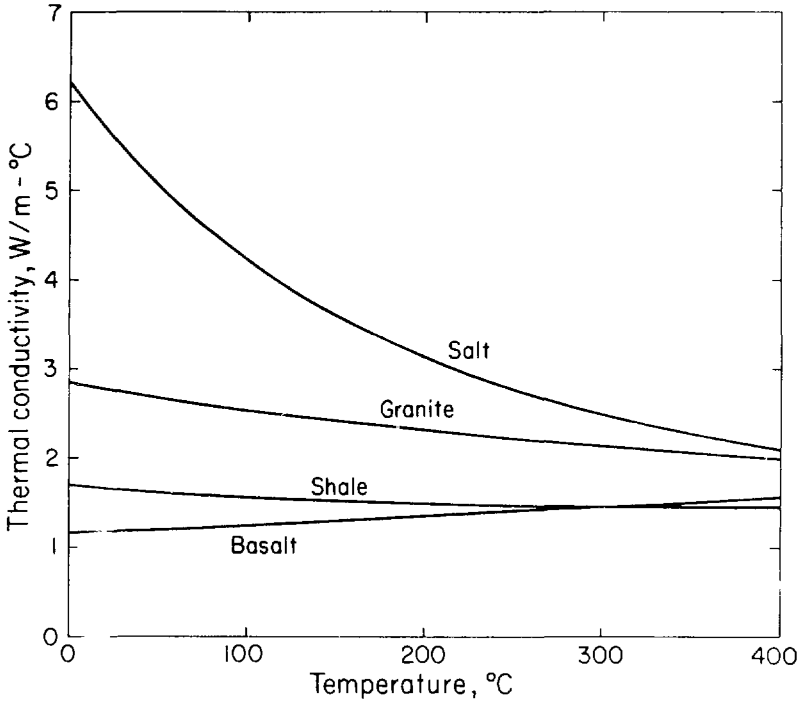

Figure 3.9 Thermal conductivity as a function of temperature for four major rock types (DOE, 1979b). [XBL 818-3441]

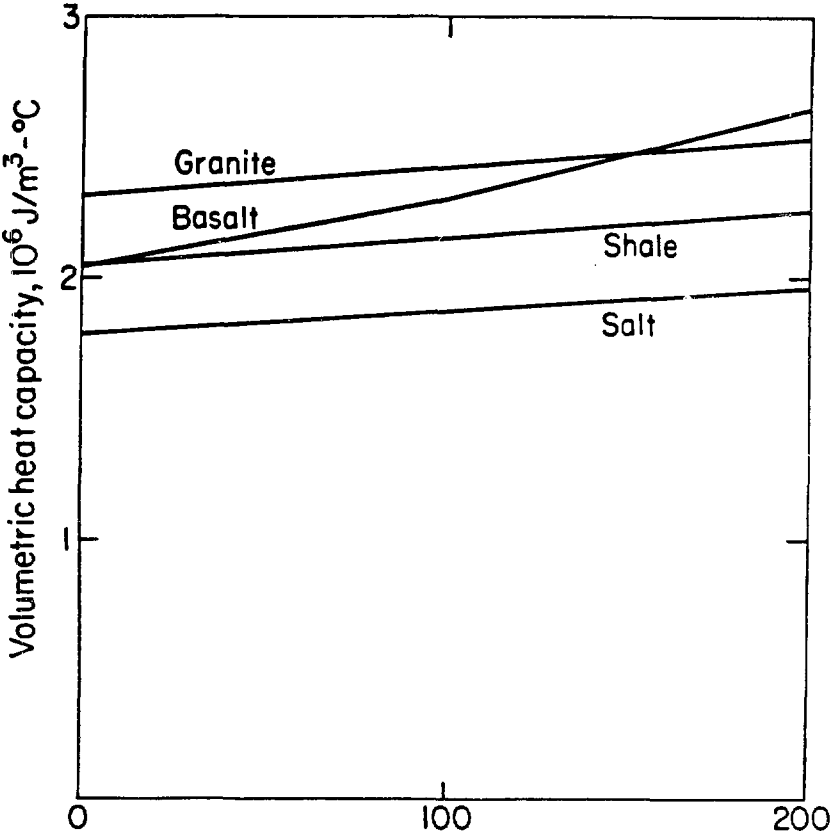

Figure 3.10 Volumetric heat capacity as a function of temperature for four major rock types (DOE, 1979b). [XBL 818-3442]

replaces water in the pores is less conductive. The low thermal conductivity of alluvium or tuff under partially saturated conditions is one of the concerns in considering these formations as potential geologic settings for a nuclear waste repository.

The thermal properties of rocks may also correlate with their mechanical behavior. For example, it was observed in the Eleana Near-Surface Heater Experiment (Lappin et al., 1981) that above 100°C the thermal conductivity of shale decreases as a result of dehydration more than expected on the basis of matrix properties. This result suggests that there is a strong coupling of thermal behavior and volumetric contraction for this rock type, which contains appreciable amounts of expandable clays.

3.3.2 Mechanical Properties

The mechanical properties of five rocks are given in Table 3.9. The values given for the first four rocks are estimates used in the GEIS study (DOE, 1979b). The range of values of the two tuff types illustrates the variability of mechanical properties with rock texture. For loosely packed tuff, the strength may be less than 30 MPa. For densely welded tuff, the strength may approach that of hard rocks like granite and basalt.

Most of the thermomechanical stress analyses assume linear elastic behavior with either isotropic or orthotropic elastic properties. The dependence

Table 3.9. Mechanical Properties of Rocks.

Rock	Young's Modulus (GPa)	Poisson's Ratio	Unconfined Compressive Strength (MPa)	Tensile Strength (MPa)
Salt	11 0	0.35	22.1	0.34
Granite	17.2	0.18	131.0	0
Basalt	12.4	0.26	124.0	0
Shale (horizontal)	4.1	0.15	27.6	0
(vertical)	2.1	0.15	27.6	0
Tuff (welded)	23-41	NA	117.0	NA
(nonwelded)	7-9	<0.1-0.25	7-30	0.1-1.4

(DOE, 1979b; Tyler, 1979)

of elastic properties on temperature, pressure, and time is ignored. However, for realistic representation of the thermomechanical behavior, the nonlinearity of the rocks must be taken into account. The plasticity and creep-flow behavior of salt has received some attention in the literature (e.g., Wahi et al., 1977). Yet the effects of joints and fractures on repository structures in hard rocks have not been well established. One of the difficulties in assessing these effects is the lack of data and the corresponding incomplete understanding of the fracture behavior under normal and shear stresses. Even though hard rock mining experience is extensive, the effects of thermal loading on fracture stability are poorly understood. Blasting during excavation and the resulting stress concentration around the perimeter of the workings introduce local fractures. The instability of these fractures under thermal loadings is one of the main concerns for the safety of repository operations.

Mainly because of fractures, the thermomechanical properties of a large rock mass cannot be represented by the elastic parameters determined by small core samples measured in the laboratory. A rock mass consists of intact blocks of rock separated by sets of fractures. The spacing, width, orientation, extent, filling material, etc., of the fracture sets determine the correction factor that scales the intact block value to the rock mass value of a large volume. The values of the elastic properties of granite, basalt, and shale shown in Table 3.9 have been modified by correction factors to approximate these size effects. In the GEIS values, a 1.5-m (5-ft) cube specimen was assumed to be large enough to represent the rock mass, with the effects of minor joints taken into account for all rock properties. The effects of major fractures are not incorporated in the GEIS study (Dames and Moore, 1978a).

In addition to the size effects, the temperature dependence of the mechanical properties may also be an important consideration, especially for the cases in which the strength of the rock mass reduces with increasing temperature. The temperature dependence of mechanical properties, together with the thermally induced stress from thermal expansion, couples the mechanical processes with the thermal behavior of the rock. The mechanical processes also couple with the hydrologic behavior because of the interaction between rock stress and fluid pressure. The compressibility of the rock matrix affects the storage capacity of the fluid in the rock mass.

3.3.3 Hydrologic Properties

The GEIS study (DOE, 1979b) produced a generic description of the hydrologic stratigraphic sections, including hydrologic properties, for four major rock types. Hydrologic data for tuff and alluvium have also been compiled (Smyth et al., 1979; Tyler, 1979; Bulmer and Lappin, 1980). Table 3.10 lists a typical value and the range for both the permeability and the porosity of the repository host interval or layer of the particular rock type. The table does not properly represent the geologic environments of a repository, since the layers both above and below the host rock intervals are also important in determining the hydraulic flow patterns. The regional groundwater movement

Table 3.10. Hydrologic Properties of Rocks.

Rock Type	Permeability (m^2) Typical	Range	Porosity Typical	Range
Salt	nil	10^{-17} - 2×10^{-30}	0.005	NA
Granite	5×10^{-17}	10^{-13} - 10^{-17}	0.004	NA
Basalt	10^{-17}	2.7×10^{-14} - 10^{-18}	0.006	NA
Shale	7.1×10^{-16}	10^{-12} - 10^{-19}	0.16	NA
Tuff (welded)	10^{-17}	$\leq 1.4 \times 10^{-15}$	0.10	0.02 - 0.25
(nonwelded)	10^{-16}	10^{-15} - 10^{-18}	0.35	0.25 - 0.55
Alluvium	3×10^{-18}	2.4×10^{-14} - 1.2×10^{-21}	0.31	0.16 - 0.42

(Dames and Moore, 1978c,d; DOE, 1979b; Smyth et al., 1979; Tyler, 1979)

also depends on the distant recharge and discharge areas and on the overall regional gradient. Consequently, the generic hydrogeologic conditions are difficult to determine.

The uncertainty and variation in hydrologic properties, especially permeability, is much higher than in thermal and mechanical properties. The permeability value is highly site specific, and the size effect is drastic. The laboratory measurements of intact rock samples are much lower than the borehole measurements, and the borehole measurements may be obscured by the small volume of rock surrounding the borehole. The sampling of a large volume of rock, especially a low-permeability rock formation suitable for a repository, is a major challenge and a significant source of uncertainty ` the assessment of repository impact.

3.4 THERMAL ENVIRONMENT OF THE WASTE PACKAGE

The key factors in determining the very-near-field thermal environment are

- canister thermal loading,
- details of canister-borehole design,
- thermal properties of the wastes, canister-borehole components, and host rock, and
- average canister spacing.

The maximum temperature and temperature gradient occur within the waste package and in the immediate vicinity of the waste canisters. The principal problems associated with the high temperatures are

- waste canister integrity for effective containment of the radionuclides, and

- canister recoverability during the retrievability period.

The following subsections discuss the maximum temperature rises in the waste, canister, and borehole; the structural instability of the various components in the very-near-field; and the long-term hydrological environment expected for the waste package.

3.4.1 Maximum Temperature Rises of Waste, Canister, Borehole

Many studies have been made to model the very-near-field temperatures in various rock formations. As an illustrative example, the results of Claiborne et al. (1980) for the temperature histories in salt containing HLW and SF are shown in Figures 3.11 and 3.12 for temperatures at the waste centerline, the canister surface, and the wall of the emplcement hole. The centerline temperature of HLW peaks at about 320°C between 1 and 2 years after emplacement and decreases to about 120°C at 100 years. The canister temperature peaks at 260°C, and the maximum salt temperature peaks at less than 160°C. These temperatures correspond to a 2.16-kW vitrified waste in a stainless steel canister with an areal loading of 25 W/m^2 (100 kW/acre). The canister is emplaced in a 0.54-m diameter hole with crushed salt backfill between the carbon steel canister overpack and the hole liner. The SF results, with an emplacement power of 0.55 kW have lower salt, canister, and cladding temperatures, and the peaks are later in time than those for HLW (Fig. 3.12). These peaks occur within 60 years after emplacement. Similar calculations have been made for the canister and repository loading and configurations described in Tables 3.1 and 3.4-3.6. The peak temperatures for reference repositories in salt, granite, basalt, shale, and tuff are summarized in Table 3.11.

Many other studies have been made to model the very-near-field temperature. The rock temperature has been shown to be relatively independent of the detailed waste package design. The balance of the heat flux from the waste package and the heat conduction into the rock determines the temperature rise. The temperature rise is proportional to the waste power, is inversely proportional to the rock thermal conductivity, and has weak (logarithmic) dependence on the thermal diffusivity and well-bore radius. The temperature rise at the wall may affect the structural integrity of the borehole.

The temperature rise at the canister surface depends sensitively on the thermal properties and heat transfer modes in the gap between the rock and the canister. With crushed, low-thermal-conductivity rocks backfilling the gap, a high-temperature gradient is induced. The peak canister temperature of a

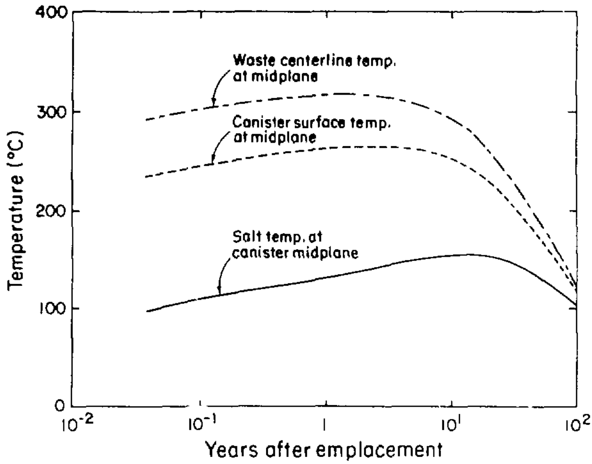

Figure 3.11 Salt temperature histories for HLW loaded at 25 W/m^2 (100 kW/acre) (CLaiborne et al., 1980). [XBL 819-11575]

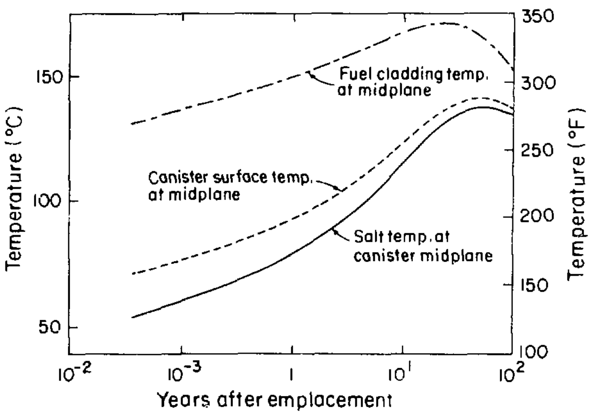

Figure 3.12 Salt temperature histories for SF loaded at 25 W/m^2 (100 kW/acre) (Claiborne et al., 1980). [XBL 819-11574]

Table 3.11. Reference Peak Near-Field Temperatures (°C)[a].

Host Rock	Location	SF[c]	HLW	DHLW
Salt	Host rock	140	160	80
	Canister wall	145	260	90
	Waste[b]	175	320	100
Granite	Host rock	150	165	105
	Canister wall	170	205	115
	Waste[b]	190	225	120
Basalt	Host rock	165	NA	NA
	Canister wall	170	NA	NA
	Waste[b]	185	NA	NA
Shale	Host rock	125	140	125
	Canister wall	140	210	135
	Waste[b]	165	235	140
Tuff	Host rock	185	225	NA
	Canister wall	190	260	NA
	Waste[b]	230	295	NA

(NWTS, 1980)

[a] Assumes initial formation temperature of 34°C for salt, 20°C for granite, 57°C for basalt, 38°C for shale, and 35°C for tuff.

[b] Maximum centerline temperature for HLW and DHLW; maximum cladding temperature for SF. The canister loadings of 10-year-old wastes are 0.55 kW/canister for SF, 2.16 kW/canister for HLW in salt and tuff, 1.00 kW/canister for HLW in granite and shale, and 0.31 kW/canister for DHLW. The area loadings are 25 W/m^2 (100 kW/acre) for SF and HLW in salt and tuff and HLW in granite; 20 W/m^2 (80 kW/acre) for SF in granite; 12.3 W/m^2 (50 kW/acre) for SF in basalt; 10 W/m^2 (40 kW/acre) for SF, HLW, and DHLW in shale; 11.6 W/m^2 (47 kW/acre) for DHLW in salt; and 13.5 W/m^2 (55 kW/acre) for DHLW in granite.

[c] Results for 1 PWR element per canister waste package configuration.

4.61-kW HLW canister with a 0.076-m (3-inch) crushed salt backfill has been modeled to be 194°C (350°F) hotter than would result if the gap were left unfilled (Davis, 1979). If the gap is unfilled, the heat transfer across the gap depends not only on conduction but also on heat transfer by radiation and convection. If there is a large air gap, the significant contribution of air convection and radiation may result in higher equivalent conductivity than if there were perfect thermal contact between the canister and the rock medium. For a small air gap, the equivalent thermal conductivity of the dead air is low. It has been shown that a small gap on the order of 0.025 m (1 inch) should be avoided, since the canister temperature as a function of the gap size is highest at this critical value (Lowry et al., 1980).

3.4.2 Wall Decrepitation and Canister Retrievability

With waste emplacement, the highest thermally induced compressive stresses from rock expansion are expected at the heater borehole wall. When the maximum axial and/or tangential stresses are approaching the compressive strength, spalling of the rock around the borehole will most likely occur, especially in brittle rocks. Decrepitation can also occur in salt at elevated temperatures (>250°C), but in the long term, plastic flow may heal the crushed salt.

The stability of the boreholes has been examined through in situ experiments. For example, the borehole walls surrounding a 5-kW heater in the granite experiment at a depth of 384 m in Stripa, Sweden, did not begin to fail until a temperature in excess of 300°C was reached (Witherspoon et al., 1981). In the near-surface experiment on shale at 23 m depth, no borehole failure was observed for its 3.8-kW heater up to 350°C (Lappin et al., 1981).

At Project Salt Vault (bedded salt), closure was observed along with deterioration of the hole surface due to migrating brine. The maximum hole closure was approximately 0.0127 m (0.5 inch) over the time of the heater experiment, about one and a half years (Bradshaw and McClain, 1971). The heater power varied during the experiment, ranging from 1.6 kW/canister to 4.8 kW/canister, with the average at approximately 3.2 kW/canister.

In order to retrieve the waste, it may be necessary to install liners for the boreholes in bedded or domed salt repositories. The same effect can be achieved by reinforcing the hole by grouting, as has been suggested for the case of the basalt reference repository (Kaiser Engineers et al., 1980). For granite, the inherent strength of the rock may make retrievability relatively easier to attain for long periods of time, so that special measures may not be required.

3.4.3 Brine Migration and the Canister Environment

In the presence of heat, small amounts of saturated brine trapped in small pockets or inclusions in the salt will migrate up the temperature gradient toward the canister, provided that the vapor phase is not present in the inclusion. Waste package corrosion from brine accumulation has been a major isolation concern in salt media, especially bedded salt. This phenomenon has been studied in Project Salt Vault in bedded salt (Bradshaw and McClain, 1971), in the Asse salt mine experiment in a salt anticline (Rothfuchs and Durr, 1980), in the Avery Island salt mine experiment in domed salt (Krause et al., 1980), and in Sandia's salt block experiment with samples from the Waste Isolation Pilot Plant (WIPP) site (Hohlfelder and Hadley, 1979).

A phenomenological model based on the experimentally observed migration rate in single salt crystals has been developed (Jenks, 1979), and the accumulated influx of brine has been calculated (Rickertsen, 1980). Figure 3.13 shows the dependence of total flow into a canister hole on the waste emplacement densities for 10-year-old SF and reprocessed HLW. For a spent fuel repository of 15 W/m^2 (60 kW/acre), the total flow in 1500 years is about 6 liters. This inflow continues to increase after 1500 years because of the slow decay rate of spent fuel. For reprocessed HLW with its higher power at emplacement the inflow is higher, especially at early times. The volumes of the 0.05-m (2-in.)-thick annulus between the overpack and the borehole wall are 560 liters for the 7.6-m (25-ft) SF hole and 400 liters for the 5.5-m (18-ft) HLW hole. The annulus is assumed to be backfilled with crushed salt (Claiborne et al., 1980).

The modeling results incorporate the fluid migration of brine inclusions, but not the vapor-phase transport and the entrapment of fluid in crystal grain boundaries. It was observed that more than 90% of the brine in Project Salt Vault and 40% of the brine in the salt block experiment were released after the heater was shut down--i.e., when the thermal stresses were released upon cooling, the trapped water in the salt was allowed to break free (Bradshaw and McClain, 1971; Hohlfelder and Hadley, 1979). If these observations can be extended to repository operation conditions, the entrapment of brine will delay the inflow into the borehole during the heating phase. It is not clear if the entrapped brine will be released during the slow cooling phase of a repository.

In addition to the fluid inclusions and entrapments, the water chemically bound to minerals in the rock can be released upon heating. The impact of fluid migration and release needs to be assessed for tuff and shale, which contain clay alteration products, and for granite and basalt if the fractures contain clays. If the fractures in the latter two rock types are open, the fluid transmitted through them is likely to be greater than the water released by clays. The long-term integrity of the canister and waste form will be determined mainly by the chemical composition of the hot fluid surrounding the waste packages.

The pressure around a waste canister depends on the borehole seal. The pressure in an imperfectly sealed hole will be essentially equal to the near-atmospheric (0.1 MPa) pressure in the room during repository operations. The

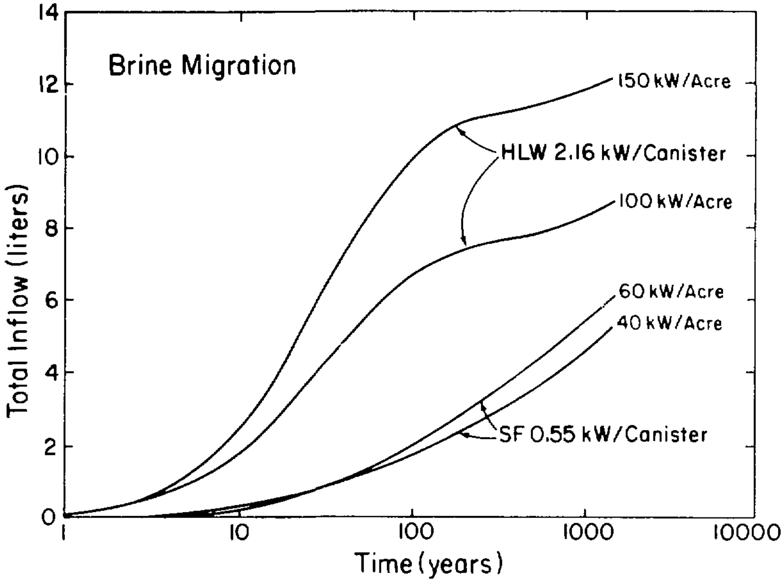

Figure 3.13 Flow into a single hole as a function of time for different waste types and areal thermal loadings (Rickertson, 1980).

[XBL 819-11573]

pressure in a perfectly sealed hole will change as a function of time in response to temperature and rate of brine inflow. For HLW in salt at an areal loading of 25 W/m^2 (100 kW/acre), the pressure peaks at about 3.2 MPa at 10 years after emplacement. For SF at 10 W/m^2 (40 kW/acre), the pressure steadily increases up to 0.45 MPa at 50 years after emplacement (Claiborne et al., 1980). Similar calculations have been made for other nonsalt rocks (NWTS, 1980). A repository located below the water table in these rocks is expected to be slowly filled with water. If the repository excavation is reflooded, the pressure would then rise to hydrostatic. The long-term effects of pressure are likely to be smaller than those induced by chemical and thermal influences on the canister integrity.

3.5 THERMOMECHANICAL ENVIRONMENT OF THE EXCAVATION

Key factors in determining the near-field thermomechanical impacts are

- thermal loading density,
- sizes of rooms and pillars and canister depth relative to the storage rooms,
- thermal and mechanical properties of the rocks, and
- ventilation and heat transfer in the drift.

The main concerns in underground operations are

- adequacy of the thermal environment for the function of personnel and equipment, and
- mechanical stability of the rock for safe operations during the excavation, emplacement, and retrieval periods.

The safety of personnel and maintenance of operating conditions were the focus of earlier design studies. Indeed, as shown in a later section, the existing maximum allowable thermal loadings in the repository designs were determined primarily by thermomechanical stability considerations. However, the thermomechanical analyses are more complex than heat transfer calculations. We first discuss the thermal field in the room-and-pillar region and then present an outline of the thermomechanical analyses and the approximations associated with these analyses. In the last subsection, we briefly review the effects of ventilation in underground cooling.

3.5.1 Temperature Field Surrounding the Storage Room

Following the emplacement of wastes, the rock in the immediate vicinity of the canister heats up. As the heat slowly transfers into the rock mass, heat flows from neighboring canisters interact and the temperatures of the rock masses above and below the canister horizon increase. The temperature evolution surrounding a storage room for the GEIS 47-W/m^2 (190-kW/acre) HLW repository in granite is shown in Figure 3.14.

The main interest in these near-field analyses is the temperature in the pillar center and around the room during the emplacement and retrieval periods. The two-dimensional unit cell was used to get the results shown in Figure 3.14. The repository is modeled as an infinite array of storage rooms with homogeneous heat sources buried below the storage room floors. The effects of surrounding storage rooms are taken into account by the insulating (adiabatic) side boundaries of the unit cell on the centerlines of the room and the pillar. Upper and lower boundaries of the model are sufficiently removed from the canister and room so that they have no effect on the near-field temperatures during the period of interest.

The temperature histories for three locations away from the canister are shown in Figure 3.15. These locations correspond to points near the upper corner of the storage room, mid-pillar at the storage floor level, and mid-pillar at the canister level, respectively. Ventilation in the storage room is not considered in these results. The GEIS report also presents modeling results for SF in salt, basalt, and shale. For the locations away from the waste canister, the two-dimensional temperature results agree fairly well with the three-dimensional results between the canisters.

These temperature distributions are used as input for the thermomechanical calculations. Most thermomechanical calculations to date for room stability have been two-dimensional. Because stress and strain are tensorial quantities, three-dimensional calculations are more complex than two-dimensional modeling. The next subsection discusses two-dimensional thermoelastic results that represent the averages along the storage room and ignore the variations of stress and strain between canisters in the same row.

3.5.2 Rock Deformation and Stress Perturbation

The stability of the room-and-pillar structure depends on both the excavation and the thermal loading. The effects of excavation can be minimized by a small extraction ratio (the ratio of the volume removed to the volume remaining). For a radioactive waste repository, the extraction ratio will be in the 10-20% range, which is low compared to conventional mine operations for removal of mineral ore. The key question is whether the additional thermal loading will induce instability.

For soft rocks like salt, the main concern is for the room closure. Salt under stress can flow slowly, or "creep." With thermal loading, the creep rate increases. It is important to limit roof sag, floor heave, and wall convergence so that emplacement and retrieval equipment can be transported through the tunnel. The heated pillar experiment in Project Salt Vault measured in

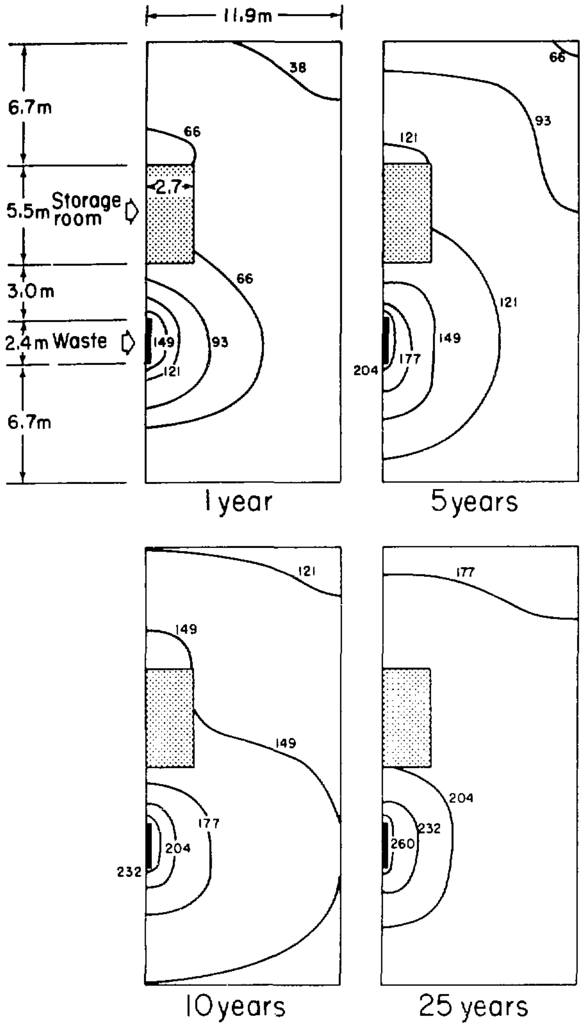

Figure 3.14 Near-field isothermal profiles (°C) for a 47-W/m^2 (190-kW/acre) HLW repository in granite (Science Applications, Inc., 1978).
[XBL 819-11576]

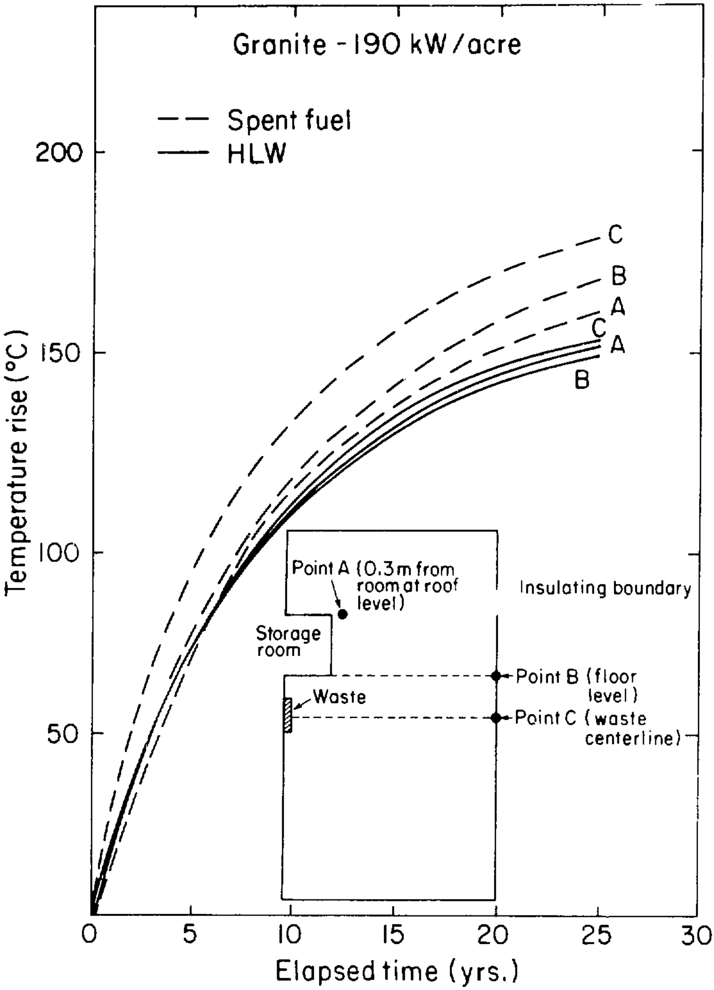

Figure 3.15 Near-field temperature histories for a 47-W/m^2 (190-kW/acre) SF and HLW repository in granite. Inset shows locations of points A, B, C (Science Applications, Inc., 1978). [XBL 819-7428]

situ thermomechanical responses, and this data base has been used in model evaluation studies (Wahi et al., 1978). In the design studies for salt (Stearns-Roger Engineering Co., 1979; Kaiser Engineers, 1978a,b), Lomenick's formula, based on laboratory salt pillar data, has been used to calculate creep. Lomenick's formula shows that strain is proportional to $T^{9.5}$, where T is the absolute temperature expressed in kelvins (K), which indicates the possible temperature dependence of the creep rate. More recent work has indicated that Lomenick's formula may be nonconservative for certain conditions and that the room closure predicted in both conceptual designs may have been underestimated (Stearns-Roger Services, Inc., 1980). Although salt has been extensively studied for the past 20 years, further studies both in in situ experiments and modeling of creep are required to understand the important effects of thermal loading on the stability of a salt repository.

For hard rocks, creep is not a problem. The main concern is the presence of fractures. During excavation, drilling and blasting can shatter the rock around the perimeter of the tunnel. The stress concentration around the excavation may open existing fractures or create new ones. However, most of the modeling studies to date are based on thermoelastic calculations, which neglect the effect of fractures. In addition to the GEIS study (Dames and Moore, 1978a), thermoelastic analyses have been done in granite (Ratigan, 1977), basalt (Hardy and Hocking, 1978), and shale (Thomas et al., 1981). The in situ granite experiment in Stripa, Sweden, shows that thermoelasticity cannot account for the rock displacements induced by the heater, as shown in Figure 3.16 (Hood et al., 1979; Witherspoon et al., 1981). The fractures can absorb the thermal expansion of rocks, and the thermomechanical responses are nonlinear. The significance of fractures for rock displacements in regions away from the heater canister has not yet been assessed, especially their significance for the stability of the storage room.

Although thermoelastic analyses neglecting fractures may not be quantitatively correct, they can qualitatively identify the zones of potential instability. In the GEIS study it was shown that the roof top is the most critically stressed area. Figure 3.17 is an example of thermally induced stresses for the 47-W/m^2 (190-kW/acre) HLW repository in granite. In this example, the maximum compressive stress parallel to the roof surface is 20.7 MPa (3000 psi). The corresponding compressive stress induced at the same point by the excavation is 31.0 MPa (4500 psi). (It is of interest to note that the in situ stress field without the excavation is assumed to be 13.8 MPa (2000 psi) vertical and 20.7 MPa (3000 psi) horizontal.) Therefore, at a roof temperature of 121°C (250°F), the total compressive stress at the roof top due to the sum of the stresses from thermal loading and excavation is 51.7 MPa (7500 psi), one-half of the uniaxial compressive strength 103.4 MPa (15,000 psi). The above numerical example illustrates the procedure used in the GEIS to determine the maximum allowable thermal loading; thermal loading is adjusted so that induced stress is one-half the rock strength. For granite, this procedure gives the value 47 W/m^2 (190 kW/acre). In Table 3.12 the maximum area loadings from this procedure is given for granite, shale, and basalt.

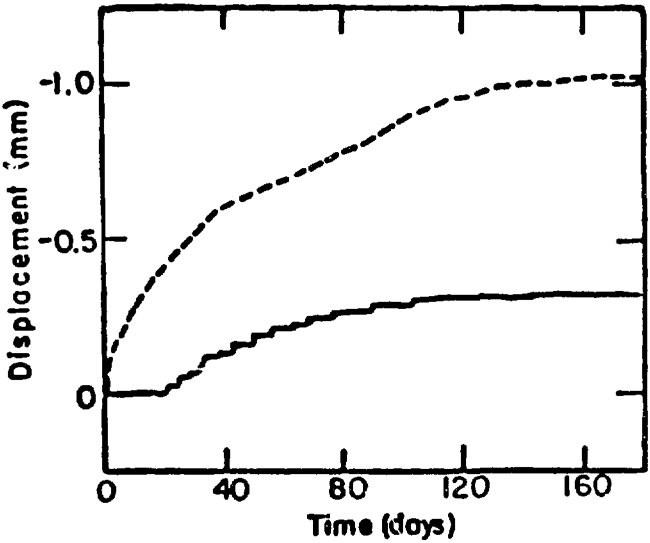

Figure 3.16 Predicted (broken) and measured (solid) vertical displacements between anchor points 3 m above and 3 m below heater midplane (Hood et al., 1979). [XBL 819-11587]

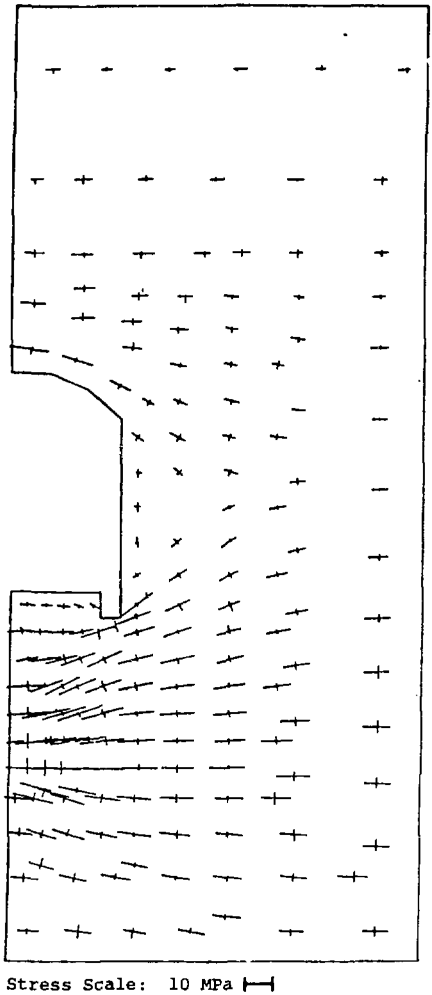

Figure 3.17 Stresses at 5 years after emplacement in granite at a depth of 580 m (1900 ft) for an HLW repository loaded at 47 W/m^2 (190 kW/acre) (Dames and Moore, 1978a). [XBL 819-11588]

Table 3.12. Maximum Areal Loadings Based on Near-Field Thermomechanical Stress Studies.

Rock Type	HLW and SF (5-year retrievability) W/m^2	(kW/acre)
Granite	47	(190)
Basalt	47	(190)
Shale	30	(120)

(Science Applications, Inc., 1978)

In the Swedish Nuclear Fuel Safety Program (Kärnbränslesäkerhet, or KBS), the thermomechanical responses for a repository in granite were modeled (Ratigan, 1977). Like the GEIS study, the KBS calculations were thermoelastic, and stresses were calculated without taking fractures into account. Fracture sets were assumed to exist in the rock mass, however, and after the stress calculations were made, the rock mass was assumed to fail along the fractures. A linear Mohr-Coulomb shear criterion was used to identify the regions of strength failure. This failure criterion is different from the compressive failure criterion used in the GEIS study.

Figures 3.18 and 3.19 illustrate the dependence of strength failure on joint orientation. Each figure shows the progressive failure due to the combined effects of thermomechanical and excavation-induced stresses. To control strength failure in locations where it may be undesirable, the characterization of joints must be carried out. Nonlinear stress analyses that take the inelastic responses of fractures into account should be performed to determine repository stability in hard rock.

The Stripa experiment, the GEIS study, and the KBS calculations discussed above indicate the inadequacy of the thermomechanical analyses used in repository design studies. For the importance entrusted to them, the thermoelastic analyses employed in determining the allowable thermal loading seem to be based on too crude a foundation. A more refined analysis that includes the effects of fractures should be initiated.

Although the current capability in modeling thermomechanical stability is rudimentary and inadequate in several respects, it tends to be the controlling criterion in determining the maximum allowable thermal loading. There is a need to develop analytic approaches that bound the uncertainties. In addition, the safety of the excavation can be maintained by engineering techniques. For example, the use of rock bolts has been included in design studies of bedded salt and basalt. Sound engineering reinforcement and careful monitoring will improve and maintain the margin of safety during the short period of repository operations.

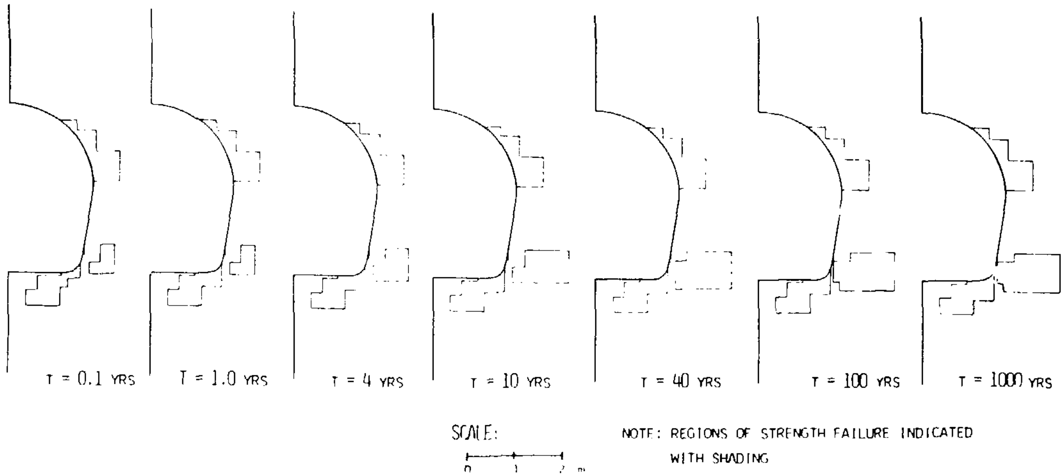

Figure 3.18 Progressive strength failure due to excavation and thermomechanical stresses with joints at 0° and 90° (Ratigan, 1977).
[XBL 819-22589]

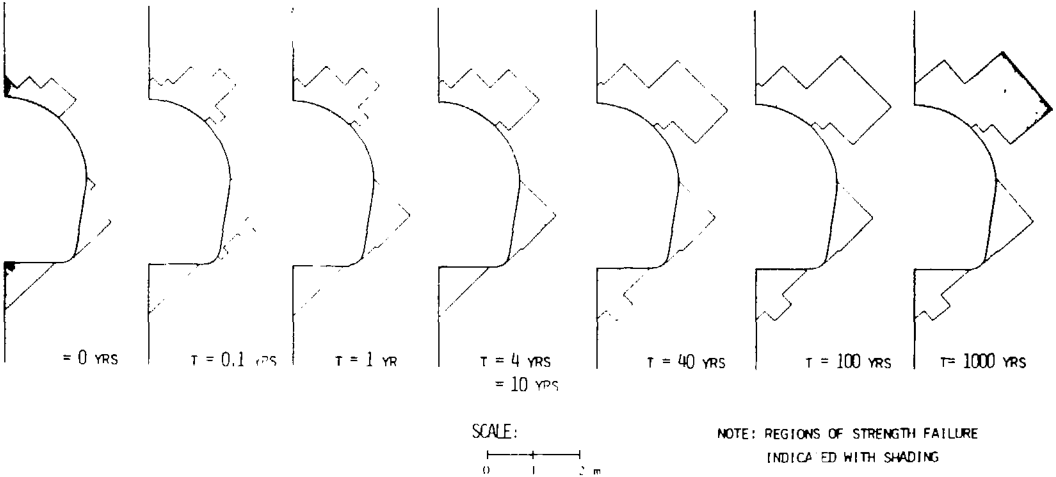

Figure 3.19 Progressive strength failure due to excavation and thermomechanical stresses with joints at 45° and -45° (Ratigan, 1977).
[XBL 819-11590]

3.5.3 Ventilation Cooling

The last topic to be discussed concerning excavation conditions is the effect of ventilation. To maintain the working environment, ventilation will be provided during the excavation and emplacement operations. The heat removed by air circulation effectively prolongs the cooling period of the wastes. If retrieval of the wastes were to be initiated after sealing the tunnel, recooling of the room would be required before the waste canisters could be reached. Both pre-emplacement and post-emplacement ventilation effects have been studied in the literature.

Heat transfer from the floors, roofs, and walls to the rooms and the natural and forced convection of the ventilating air through the rooms remove a large portion of the heat during the ventilation period. Cheverton and Turner (1972) show that at about 5 years after burial, a maximum of 40% of the waste heat can be removed with room ventilation for a 39-W/m^2 (158-kW/acre) HLW repository in salt. Altenbach (1979) shows that up to 90% of the waste heat can be removed after 44 years for a 9-W/m^2 (36-kW/acre) SF repository in salt. The amount of cooling depends on a number of factors:

- flow rate and temperature of the air,
- thermal loading and the burial depth of the waste,
- heat transfer in the rock, and
- heat transfer in the room.

The last factor depends on the roughness of the rock wall (Boyd, 1978). With ventilation, the excavation will experience less strength failure (Ratigan, 1977). Ventilation can substantially lower borehole, canister, and waste temperatures, especially if the borehole is not backfilled with low-conductivity crushed rocks (Claiborne et al., 1980).

When re-entry is desired, a waiting period is required to allow the room floor to cool to an acceptable temperature. Altenbach and Lowry (1980) show that it takes less than 6 months for the floor to cool down after 50 years' storage in a 9-W/m^2 (36-kW/acre) repository (see Fig. 3.20). Boyd (1978) shows that less than 10 years is required for the room temperature to drop to 49°C after 30 years in a 18.5-W/m^2 (75-kW/acre) WIPP drift. The effects of recooling on room stability are not well known.

The principal advantage of ventilation is the cooling of storage rooms during the operations phase of the repository. However, the reduction in the accumulated thermal output of the waste during short periods of ventilation is expected to be small. Ventilation cannot be used to reduce significantly the problems caused by waste heat over thousands of years (Koplik et al., 1979). The underground ventilation cooling is essentially an extension of surface cooling before waste emplacement.

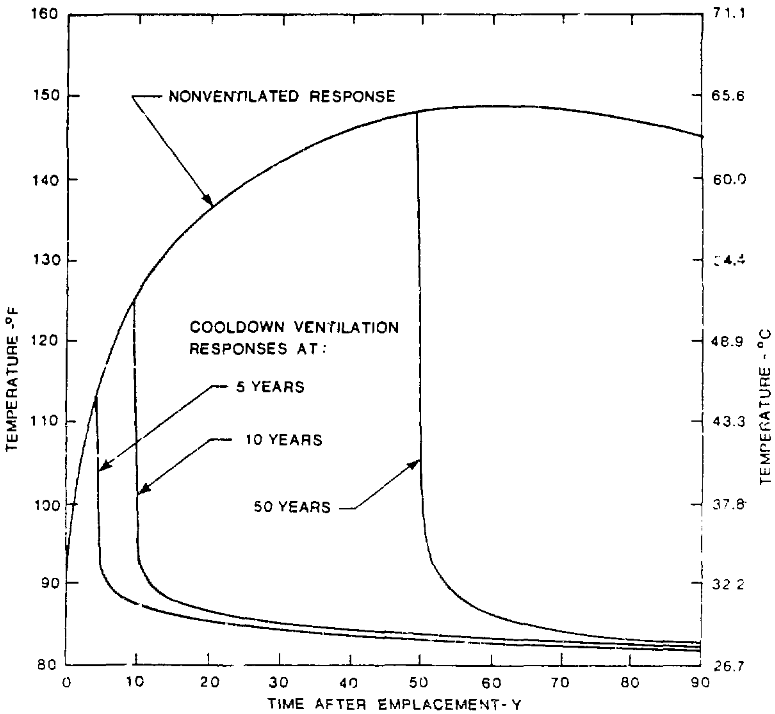

Figure 3.20 Storage room floor temperature histories representing cool-down by 4.72 m^3/s (10,000 ft^3/min) of air from an unventilated state after 5, 10, and 50 years (Altenbach and Lowry, 1980). [XBL 819-11591]

3.6 THERMOHYDROMECHANICAL ENVIRONMENT IN ROCK FORMATIONS

The key factors in determining the far-field thermohydromechanical environment are

- time dependence of waste heat power and thermal loading density,
- repository depth, size, and shape,
- rock properties, and
- in situ environment and boundary conditions.

The main concerns in the geologic setting are

- groundwater buoyancy flow from the repository to the surface,
- surface uplift and integrity of the rock formation, and
- cumulative heat energy that remains in the formation.

The waste repository impact on the rock formation may persist for thousands of years over thousands of meters. Quantitative predictions of the long-term thermal, mechanical, and hydrologic perturbations can be made only through modeling of the repository and its geologic setting. With these predictions one can calculate the degree of long-term isolation. In this section we review the results of far-field thermal modeling and parametric studies, discuss the surface uplift and stress changes, and analyze the hydrologic perturbations due to buoyancy flow.

3.6.1 Long-Term Thermal Perturbations in Rock Formations

Rock formations are poor thermal conductors. It takes hundreds to thousands of years for the waste heat to transfer from a 500- to 1000-m-deep repository to the ground surface. The waste releases heat continuously at a decreasing rate over thousands of years.

The long-term temperature variations and distributions in the rock formation surrounding a repository depend on the thermal loading, the repository characteristics, and the rock properties. The effects of these three controlling factors on the far-field temperature rise will be discussed in the following three subsections.

3.6.1.1 SF repository versus an HLW repository

Figure 3.21 illustrates the evolution of the temperature field in a 10-W/m^2 (40-kW/acre) SF disk-shaped repository in granite (Wang et al., 1981). In this case, the heat leaks out to the atmosphere after 1000 years, distorting

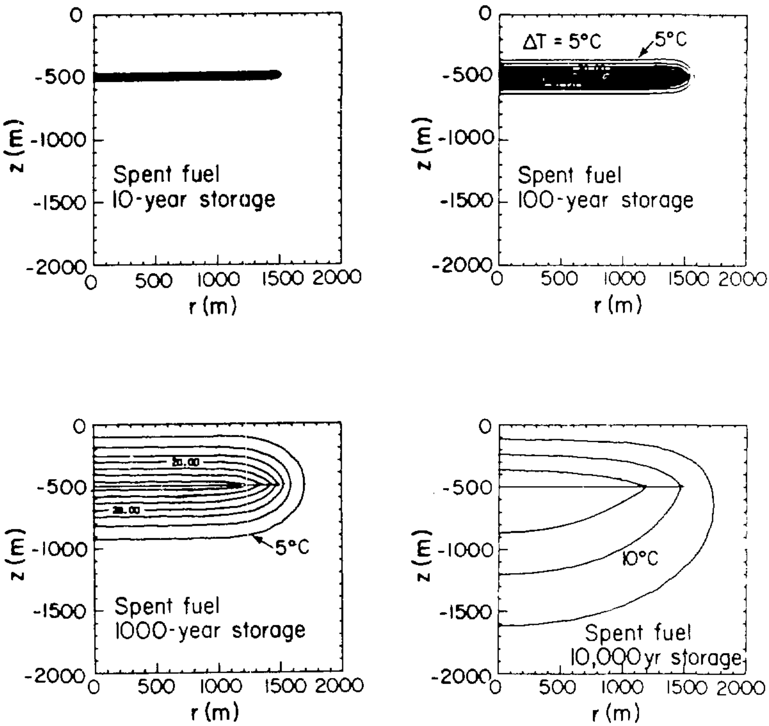

Figure 3.21 Isotherms and profiles of the temperature rise around an SF repository in granite after 10, 10^2, 10^3, and 10^4 years (Wang et al., 1981). [XBL 819-11592]

BIBLIOGRAPHY ON THERMALLY RELATED PROBLEMS IN A NUCLEAR WASTE REPOSITORY

This bibliography is the product of a search of the literature on thermally related problems in nuclear waste storage. Included with each title is the abstract that appeared with the original article. We hope that scientists and engineers interested in these problems will find this compilation useful. As more articles on this subject are brought to our attention, we hope to update the collection. Articles included are arranged according to the following groups:

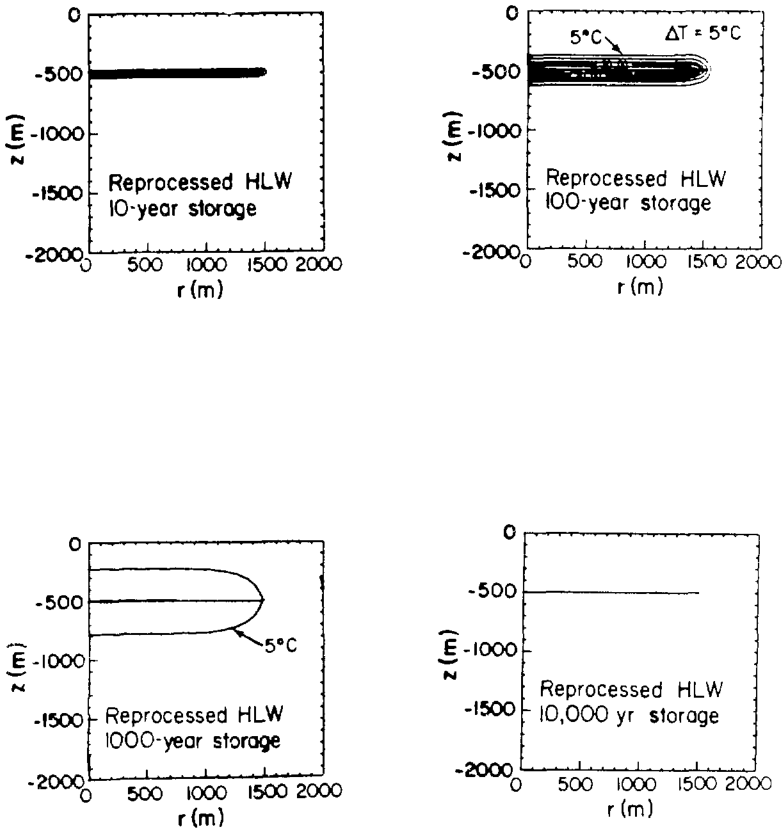

Figure 3.22 Isotherms and profiles of the temperature rise around an HLW repository in granite after 10, 10^2, 10^3, and 10^4 years (Wang et al., 1981). [XBL 819-11593]

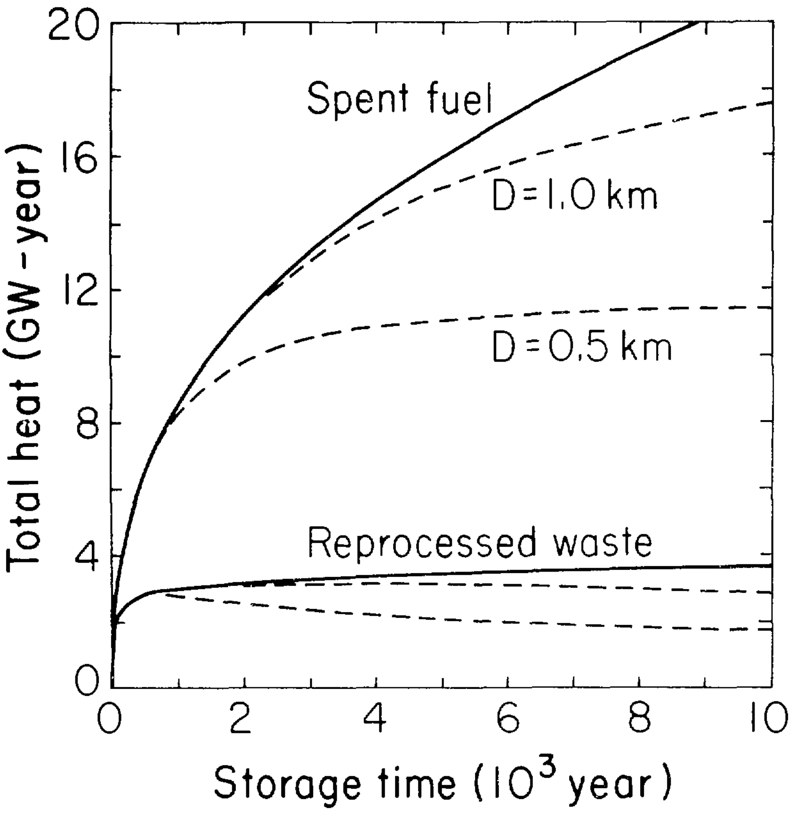

Figure 3.23 Solid curves show heat released by the buried wastes. Broken curves show heat remaining in the granite formation for a disk-shaped repository 3 km in diameter at different depths (Wang et al., 1980). [XBL 808-2736]

hole or using a multilevel repository have been considered for laterally limited rock formations (Hamstra and Kevenaar, 1978; Just, 1978; Tammemagi, 1978). Table 3.13 compares the temperature increases in a single-level repository with a those of a three-level repository. Use of a multilevel burial concept reduces the repository temperature; the thermal interaction among adjoining layers delays the peak temperature. However, the total energy remaining in the formation will increase slightly.

In each of the above results, the repository is approximated by a disk heat source (assumed to be simultaneously loaded), with the waste uniformly distributed over the disk. In more detailed repository designs, more realistic representations of the repository configuration and operations have been made.

For example, in the conceptual design study for the NWTS-1 repository in domed salt (Stearns-Roger Engineering Co. and Woodward-Clyde Consultants, 1978), the waste loading is not instantaneous and a more realistic rate of waste arrival at the repository is assumed. The repository shape is not a simple disk but a ring-shaped region with an inner radius of 344 m (1128 ft) and an outer radius of 1378 m (4522 ft). There is no waste emplacement in the inner-radius region in order to protect the centerline shaft for waste transportation. The repository is loaded from the outermost boundary toward the inner radius. Figure 3.24 shows the evolution of the temperature distribution along the radius of the repository. This result illustrates the sensitive dependence of short-term effects on sequential loading, as well as on repository configuration.

The repository configuration can have a far-field impact. For example, the surface uplift above a rectangular repository with a shaft pillar has been found to be smaller than that above a disk repository (Dames and Moore, 1978a). Surface uplift is discussed in detail in Section 3.6.2.1. In the remainder of this report, the repository is assumed to be a single-level disk for the far-field generic evaluation. It should represent the most compact configuration and induce the greatest impact in most cases.

Table 3.13. Peak Temperature of Single-Level and Three-Level Repositories.[a]

Number of Layers	Domed Salt Temperature Increase (°C)	Domed Salt Time (years)	Granite Temperature Increase (°C)	Granite Time (years)
Single-level	37.0	60	48.1	70
Three-level	34.6	125	38.3	175

[a] Scaled from Just (1978) for a 10 W/m^2 (40 kW/acre) SF repository. The temperature is calculated from a far-field model.

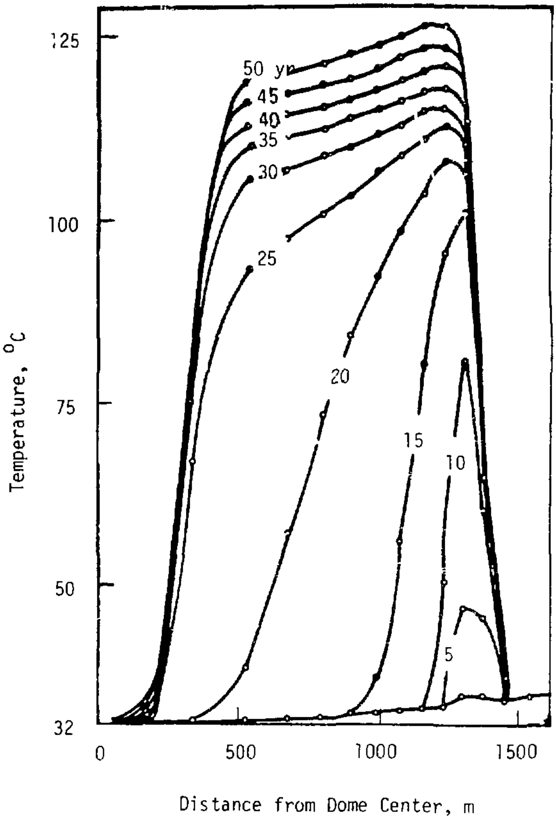

Figure 3.24 Evolution of the temperature distribution for a repository in domed salt loaded sequentially from the outermost boundary toward the dome center (Stearns-Roger Engineering and Woodward-Clyde Consultants, 1978). [XBL 819-11594]

3.6.1.3 Controlling rock properties for far-field temperatures

The temperature field in the repository and in the rock formation depends on

- the rock's capacity to sustain a temperature rise, and
- the heat transfer from the repository to the surrounding formation.

Figure 3.25 shows the dependence of the repository temperature history on the rock type, where temperature represents the areal temperature average between canisters. It produces a conservative estimate of the rock temperature in the pillar. For all rock types, this average repository temperature peaks before 100 years.

The maximum repository temperature rise is approximately proportional to the inverse square root of the product of thermal conductivity and volumetric heat capacity. (This is an exact result if the heat power is a single-term exponential decay function.) The high thermal conductivity of salt is compensated for by its low heat capacity. Table 3.14 summarizes the thermal properties used in Figure 3.25.

Table 3.14. Thermal Properties of Rocks Used in the Far-Field Model.

Rock Type	Thermal Conductivity (K, W/m-°C)	Volumetric Heat Capacity (ρc, J/m^3-°C)	Thermal Diffusivity ($K/\rho c$, m^2/s)	Analytic Approximation[a] (°C(W/m^2)$^{-1}$)
Salt[b]	4.20	1.87×10^6	2.24×10^{-6}	4.02
Granite[b]	2.56	2.43×10^6	1.05×10^{-6}	4.52
Basalt[b]	1.26	2.30×10^6	0.547×10^{-6}	6.62
Shale[b]	1.54	2.15×10^6	0.716×10^{-6}	6.19
Tuff[c]	1.60	1.87×10^6	0.856×10^{-6}	6.52
Alluvium[d]	1.00	1.70×10^6	0.588×10^{-6}	8.65

[a] Analytic approximation for the temperature rise per areal loading in W/m^2 (see Eq. (A-6) in the Appendix: $\dfrac{0.305}{[K\rho c \ell n(2)/t_{1/2}]^{1/2}}$, $t_{1/2}$ = 30 years.

[b] GEIS values at 100°C (DOE, 1979b).

[c] Typical values of welded tuff (Tyler, 1979).

[d] Smyth et al., 1979.

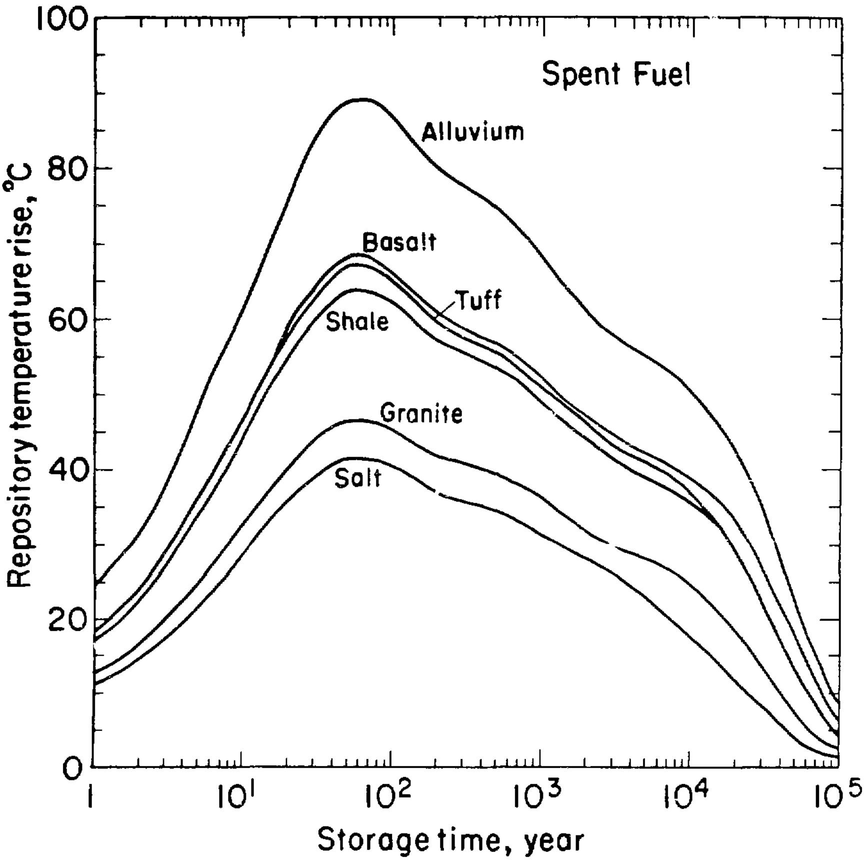

Figure 3.25 History of average SF repository temperature for six major rock types. [XBL 819-11595]

The similarity of the calculated repository temperatures in granite, bedded salt, and domed salt was also noted previously in the EPA (1977) study. The vertical temperature profiles for granite and domed salt from the EPA study are shown in Figures 3.26 and 3.27, respectively. Away from the repository, the temperature distributions are not very sensitive to the different rock types. At early times, the vertical temperature variation (gradient) is high and localized close to the repository. The temperature gradient at the repository level, related to the heat flux released by the waste, is monotonically decreasing. The temperature gradient at the surface, representing the heat flux leaked from the rock to the atmosphere, will be noticeably above its ambient value only after hundreds of years. When the heat flux from the waste is equal to the heat flowing out of the rock, the heat remaining in the rock is at its maximum and the temperature profile from the repository to the surface is nearly linear. This occurs at times on the order of thousands of years, depending on rock type and repository depth.

After this quasi-equilibrium state, the temperature rise from the repository to the surface is determined by the heat in the rock divided by the heat capacity. Although the heat transfer is determined by both thermal conductivity and volumetric heat capacity, the magnitude of the long-term, far-field temperature rise is more sensitive to the latter.

Many modeling studies have been done for different stratigraphies in various rock formations (EPA, 1977; Science Applications, Inc., 1978; DOE, 1980a). Although the rock layers in different stratigraphic models have very different thermal conductivities, the temperature profiles are all quite similar, with only mild kinks across the boundaries. The insensitivity of the temperature profiles to the heterogeneity of the rock layers has stemmed from the similarities among the heat capacities of the different rocks. Among all the rock properties discussed in Section 3.3 (see Tables 3.8 and 3.14), the heat capacity is the least site specific and the least rock specific.

This strongly suggests that we can easily model and understand the long-term temperature changes in the surrounding rock. The uncertainty associated with our incapability to measure rock properties in detail over large rock formations does not seriously limit our capability to assess the long-term temperature changes. A pessimistic viewpoint prevails in the literature that geologic settings are too complex and the uncertainty of the predicted results too great to give meaningful bounding values. However, a reserved optimism may be more appropriate, because, as we have seen above, a reasonable certainty can be assigned to our generic quantitative evaluation of the overall temperature changes in different rock formations.

3.6.2 Far-Field Thermomechanical Perturbations

One of the consequences of a temperature rise in the rock formation is the thermal expansion of the rock. When the rock mass expands, the ground surface rises. Of all the far-field perturbations, surface uplift has received

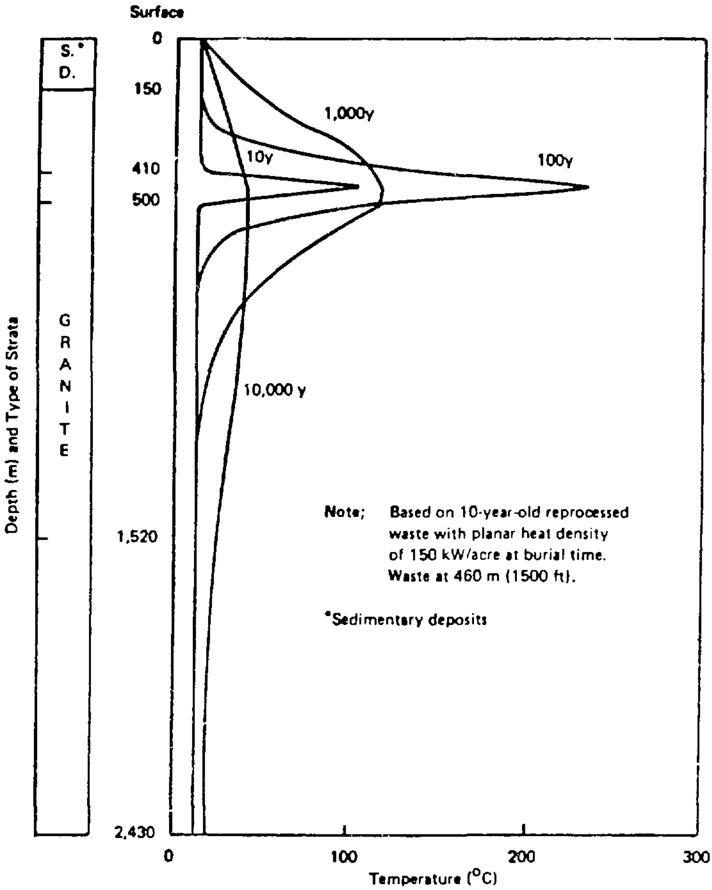

Figure 3.26 Vertical temperature distribution for a repository at 460 m depth in granite, loaded at 37 W/m^2 (150 kW/acre) (EPA, 1977).
[XBL 819-11596]

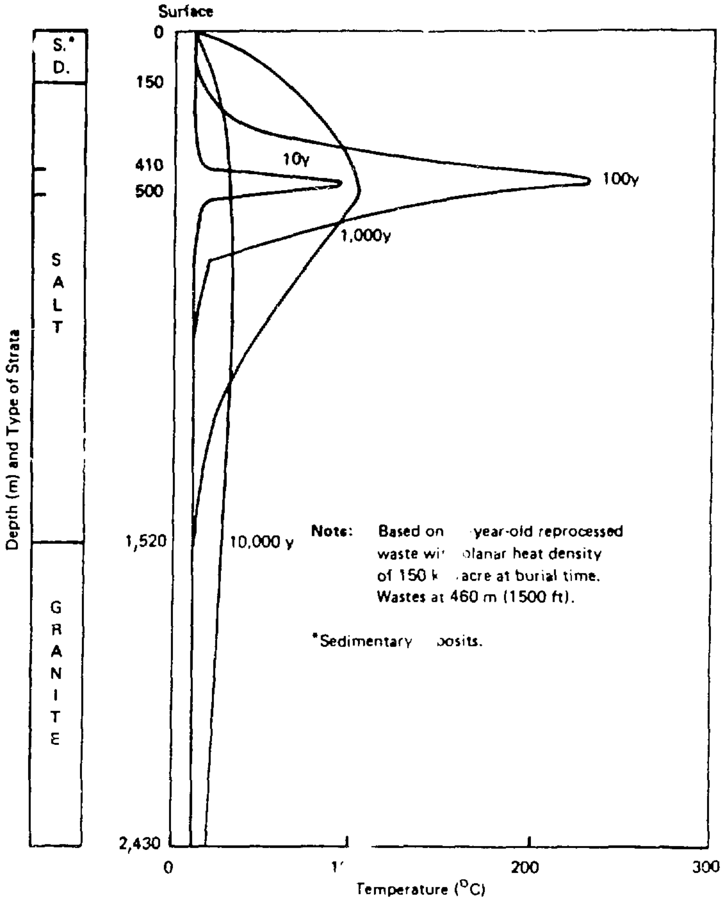

Figure 3.27 Vertical temperature di tribution for a repository at 460 m depth in domed salt, loaded c 37 W/m^2 (150 kW/acre) (EPA, 1977).
[XBL 819-11597]

the most attention, especially in salt, with its high thermal expansion coefficient. Thermal stress may also induce permeability changes. Surface uplift and rock displacement will be discussed in this section.

3.6.2.1 Surface uplift

The volumetric thermal expansion of the rock creates an uplift of the rock mass in which the waste is buried. The maximum thermal uplift will be above the center of the repository, where the maximum temperature rise within the strata will occur. Thermally induced rock displacements are pervasive and cumulative. Surface uplift depends on temperature rise, or equivalently, on the waste heat energy in the whole rock formation. Conservative estimates can be made for surface uplift with rock movement constrained in the axial direction (Callahan and Gnirk, 1978). Table 3.15 summarizes the results of maximum surface uplift for axisymmetric analyses in the GEIS study for salt (Callahan and Ratigan, 1977; Russell, 1979), granite, basalt, and shale (Dames and Moore, 1978a). Also included are the linear thermal expansion coefficients and the Poisson ratio factor of the different rock types.

The thermal loadings in the table are determined by near-field, excavation stability analysis. The concern over large surface uplift for an SF repository in salt has resulted in a lowering of the waste loading from 37 W/m^2 (150 kW/acre) to 15 W/m^2 (60 kW/acre). This is the only case in the literature for which long-term far-field consideration of the waste impact has imposed a constraint on the thermal loading.

Table 3.15. Surface Uplift.

Rock	Salt		Granite		Basalt		Shale	
Waste	SF	HLW	SF	HLW	SF	HLW	SF	HLW
Maximum Surface Uplift (m)	3.45	1.27	0.94	0.34	0.58	0.18	0.79	0.34
Time (years)	1000	150	3000	1000	7000	3000	3500	1800
Thermal Loading, W/m^2 (kW/acre)	37 (150)		47 (190)		47 (190)		30 (120)	
Linear Thermal Expansion Coefficient ($°C^{-1}$)	4×10^{-5}		8.1×10^{-6}		5.4×10^{-6}		8.1×10^{-6}	
Poisson Ratio Factor $(1+\mu/1-\mu)$	2.08		1.44		1.70		1.35	

(Dames and Moore, 1978a; Russell, 1979)

The surface uplift calculated by thermomechanical analyses did not include the opposite effect of subsidence. Subsidence is caused by the eventual closure of the excavation. Even if the repository is backfilled with crushed rock before decommissioning, the deformation of the rock to fill the void will lead to a lowering of the surface, especially in the case of salt, with its tendency to creep. This subsidence could cancel out part of the thermal uplift. Figure 3.28 shows such a phenomenon for an HLW repository in salt at 600 m depth (INFCE, 1980).

Both the uplift and subsidence processes are slow, their effects persisting over long periods of time. In comparison to the ground displacement predicted for the waste repository, much higher subsidence over shorter time periods has been observed in petroleum and geothermal fields. The main concern of surface uplift in repository studies is not the vertical displacement itself but the long-term stability of the geologic formations. This is discussed in the following subsection.

3.6.2.2 Stress perturbations and crack openings

The rock's volumetric thermal expansion can induce both axial and lateral stress changes. The development of thermally induced tensile stresses near the ground surface has been noted in thermomechanical analyses (Dames and Moore, 1978a; Hodgkinson and Bourke, 1980). The tensile stresses exist in the angular direction perpendicular to the radial-vertical (rz) plane. The horizontal components of tensile stress will reduce the normal stress across pre-existing vertical fractures, which will in turn increase their aperture and substantially change the rock mass permeability. If net tension is to be avoided, the rock mass must have large compressive horizontal in situ stresses to counter the induced tensile stresses.

The magnitude of the induced tensile stresses can be controlled by the thermal loading and repository depth. The analytic model of a spherical repository showed that the induced tensile stresses at the ground surface are inversely proportional to the cube of the mean repository depth (Hodgkinson and Bourke, 1980).

The concern over the thermally induced tensile component is not limited to fractured hard rocks but exists also for salt. The large differential of the thermal expansion between bedded salt and the overlying shale layers, or between domed salt and its sheath, may cause a breach in the hydrologic barrier protecting the salt.

Although the significance of the tensile stresses has been noted in thermoelastic analyses, the quantitative assessment of their impact is difficult. The opening of fractures will redistribute the load and stress. The displacement field will be different from continuum analyses. As we noted earlier, the heater experiments in Stripa, Sweden, indicate that fractures under compression can absorb the thermally induced rock displacements in the near-field at early

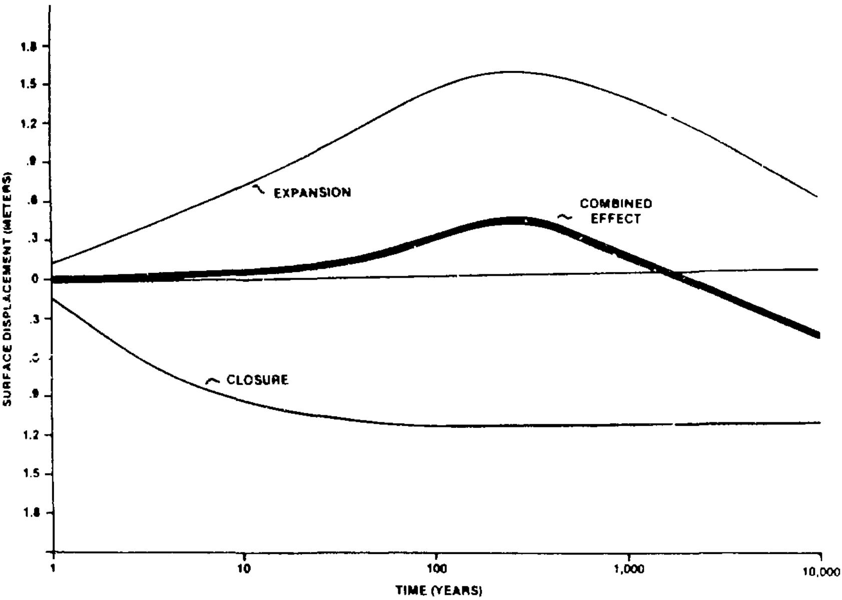

Figure 3.28 Comparison of repository closure and thermal expansion for an HLW repository in salt (INFCE, 1980). [XBL 819-11598]

times. It is not clear to what degree the pre-existing fractures in the far-field under thermally induced tensile stress will alter the displacement of the fractured rock mass.

The main concern of thermomechanical perturbations, both in the near- and far-field, is the induced permeability changes in the rock mass. Permeability may decrease with fracture closure in the compressive regions and increase with fracture opening in the tensile zones. Neither the temperature-stress nor the stress-permeability relationships are well known, especially for large volumes of rock mass. These considerations in coupled processes have contributed somewhat to the uncertainty in the evaluation of long-term, far-field effects. In the next section, other uncertainties in the assessment of hydrologic perturbations will be discussed. It is essential that we understand the origin of these uncertainties in order to base our evaluation and criteria on quantities that can be accurately calculated.

3.6.3 Long-Term, Far-Field Thermohydrologic Perturbations in the Rock Formations

The thermally induced movement of groundwater from the repository to the surface may be the dominant mechanism for radionuclide transport at a particular site. The magnitude of the fluid velocity is difficult to determine because of the variations of the hydrologic parameters in the surrounding rock formations. This section discusses the significance and uncertainty of the thermohydrologic perturbations over thousands of years.

3.6.3.1 Variations in hydrologic conditions

The fluid flow is proportional to the permeability of the rock formation. Permeability may vary over six orders of magnitude even for the same rock type, as shown in Section 3.3.3. The permeability value depends on the scale of the rock volume measured (Brace, 1980). Typically, the value measured in the field is larger than the value measured with small samples in the laboratory.

The velocity of water within the flow channels (fractures or connected pores) is inversely proportional to the porosity of the rock. Of the total void space measured in the laboratory, only a small fraction corresponds to the continuous flow paths (Norton and Knapp, 1977). The uncertainty associated with porosity may be as high as two to three orders of magnitude.

Scale dependence and the heterogeneity of the rock hydrologic properties largely originate from the presence of fractures. Most of the modeling studies use either a porous medium model to represent the average behavior of fractured rock masses (Dames and Moore, 1978b; Hardy and Hocking, 1978; Burgess et al., 1979; Hodgkinson, 1980; Bourke and Robinson, 1981) or a simple fracture model for worst-case studies (Wang and Tsang, 1980; Wang et al., 1981). In view of the uncertainty of the hydrologic properties, these deterministic modeling studies must eventually be supplemented by statistical analyses in order to assess the effectiveness of the rock formation to isolate radionuclides.

3.6.3.2 Buoyancy flow

Although many different generic representations of the hydrologic conditions have been assumed in various modeling studies, they all demonstrate the significance of the vertical buoyancy flow, which has been shown to persist over thousands of years.

Figure 3.29 shows a sketch of a vertical fracture that is assumed to extend from a 10-W/m^2 (40-kW/acre) repository to the surface. The hydraulic aperture of the fracture is assumed to be 1 μm, corresponding to a fracture permeability of 8.3×10^{-19} m^2, and the lateral width of the fracture is assumed to be smaller than the diameter of the repository. The inlet of the fracture at the repository level is assumed to be in direct contact with ambient groundwater. This model represents a very bad hydrologic condition for vertical buoyancy flow, with a continuous fracture connecting the repository to the surface and instant recharge at the repository level. The instant recharge can be maintained only with an effectively infinite permeable zone at the repository level. Although the fracture model is very simple, it should possess the same physical behavior as that of the more complex systems in fractured rock masses.

Figure 3.30 illustrates the temporal dependence of the water velocity in such a vertical fracture as a function of fuel cycle and depth. It shows that the maximum buoyancy flow occurs at thousands of years for SF repositories and at hundreds of years for reprocessed HLW repositories. For the same thermal loading at waste emplacement, the magnitude of the buoyancy flow induced by SF is much larger than that by reprocessed HLW.

Figure 3.30 also shows that the maximum buoyancy flow occurs at a later time for a deeper repository. The buoyancy flow is determined by the integrated temperature distribution from the repository to the ground surface. For a deeper repository, it will take longer for the waste heat to reach the surface. The average temperature rise, or equivalently the amount of heat remaining in the rock formation, reaches its maximum value when the heat flowing out of the rock on the ground surface is equal to the heat released by the waste. The time of occurrence of the maximum value is approximately proportional to the square of the repository depth.

The maximum vertical velocity corresponds to the minimum transit time from the repository to the surface, called in the GEIS study the "minimum surface approach time." In that investigation a porous medium model was used to represent the average hydrologic properties of a fractured rock mass. The results are tabulated in Table 3.16 along with the estimated times of occurrence of these minimum approach times. The porous-medium results agree qualitatively with the calculations from the single-fracture model above, indicating that buoyancy flow can be an important effect regardless of the particular approximation for the flow paths in the rock mass.

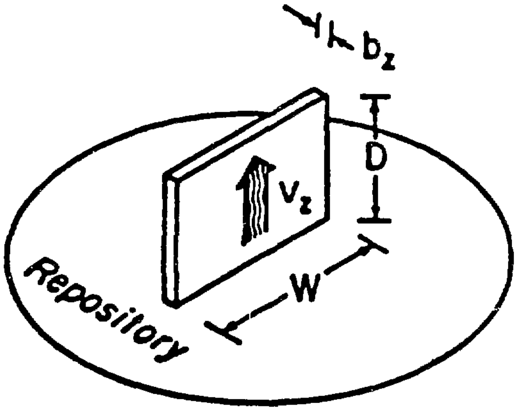

Figure 3.29 Sketch of a vertical fracture from within the repository to the surface with v_z = vertical velocity, D = repository depth, W = fracture width, b_z = fracture aperture (Wang et al., 1981).
[XBL 819-11599]

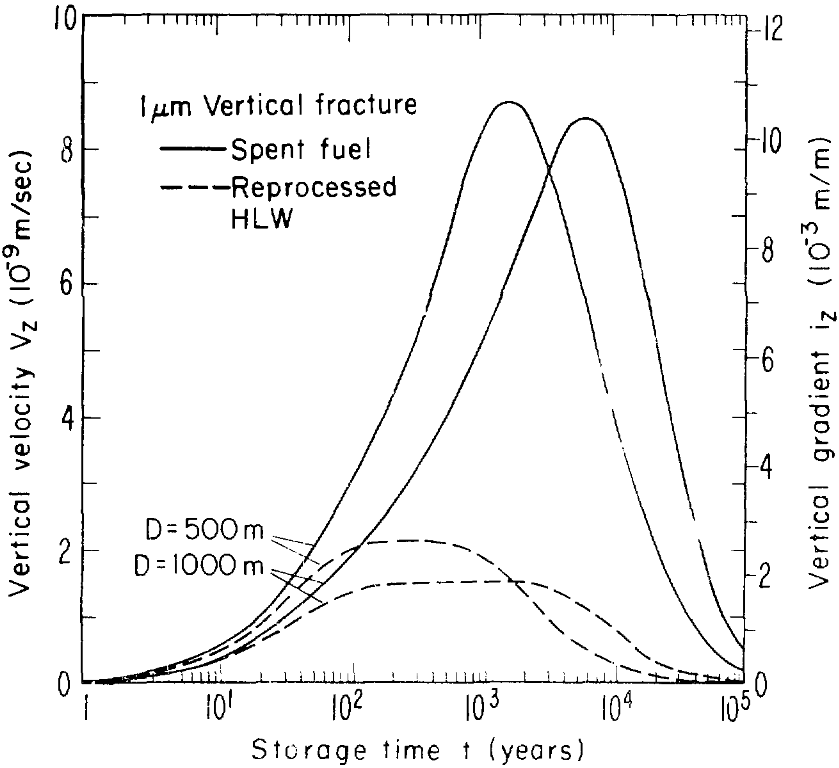

Figure 3.30 Effect of fuel cycle and depth of a repository on the flow velocities, hydraulic gradients, and water movement along a vertical fracture from the repository to the surface (Wang et al., 1981). [XBL 819-7447]

Table 3.16. Minimum Surface Approach Time.

Rock	Granite	Basalt	Shale
Minimum Surface Approach time (years)[a]	100	144–168	80–100
Time of Occurrence (years)	1000–5000	50[illegible]–10000	1000–5000

(Dame and Moore [illegible]78b)

[a] The results are normalized to 25 W/m^2 (100 kW/acre).

It is of interest to note the similarity between buoyancy [illegible] tions discussed here and the surface uplifts descri[illegible] in Section [illegible] effects have maximum values [illegible] long times. Both ef[illegible]ts are determ[illegible] cumulative heat remaining in the rock formation. The [illegible] buoyan[illegible] flow [illegible] from the thermal expansion of the water, and the sur[illegible] uplif[illegible] from thermal expansion of the rock.

3.6.3.3 Distortion of convection cel[illegible]

The repository represents a heat source of finite size [illegible] the host rock. The temperature rise is localized around the rep[illegible] sity contrast between the hot water in the repository vi[illegible] water away from the repository causes the formation [illegible] convecti[illegible] convection cells in an extended vertical fracture (see Fig. [illegible] years are illustrated in Figure 3.32. Two convection cells [illegible]ve develop[illegible] edges of the repository. Heated water flows up from the cent[illegible] the repository, and incoming water is drawn from the rech[illegible] sides (5 km from the repository center) and from the gr[illegible] surf[illegible] from the center of the repository. The diameter of [illegible] order as the depth of the repository at 1000 years. A[illegible] the convection cells are localized in the regions of [illegible] in the immediate repository vicinity. The buoyancy [illegible]w and the [illegible] cells grow as the cumulative heat increases in the rock formations. [illegible] thermally induced phenomena will eventually disappear when the [illegible] by the atmosphere and the rock returns to its original condit[illegible] [illegible]ens [illegible] thousands of years.

The shape of the convection cells depends not only on the heat in th[illegible] rocks but also on the regional groundwater flow driven by the pr[illegible] dient between the recharge zone and the discharge zone. [illegible] distortion of the cells with a horizontal gradient of [illegible] m/m. [illegible] groundwater flow suppresses the convection cell on t[illegible] repository and distorts the cell on the discharge s[illegible] verti[illegible] component of water velocity is only weakly [illegible]

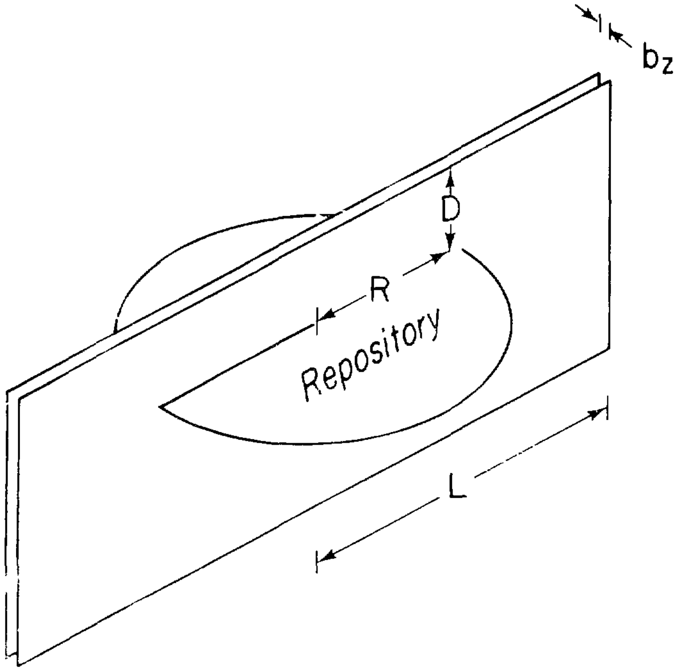

Figure 3.31 Sketch of a vertical fracture through the repository to the surface with D = repository depth, R = repository radius, L = distance to recharge zone, b_z = fracture aperture (Wang et al., 1980). [XBL 819-7269]

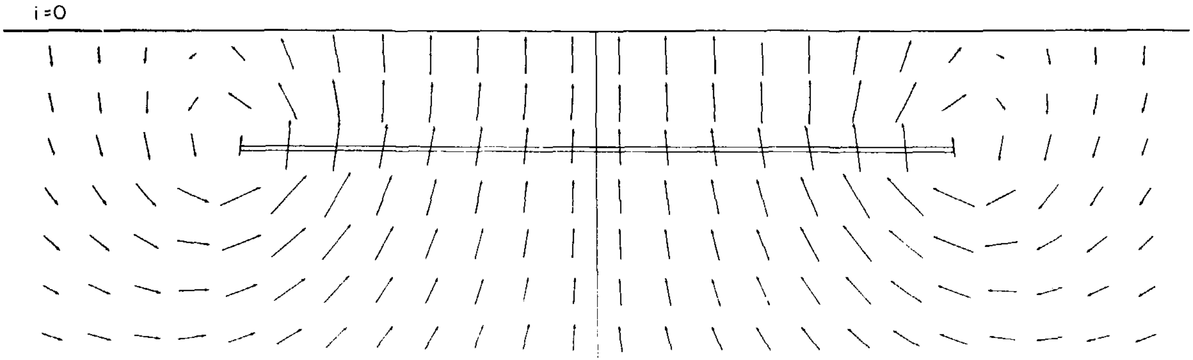

Figure 3.32 Convection cell in an extended vertical fracture around a repository; horizontal gradient = 0.0 m/m (Wang and Tsang, 1980).
[XBL 806-10034]

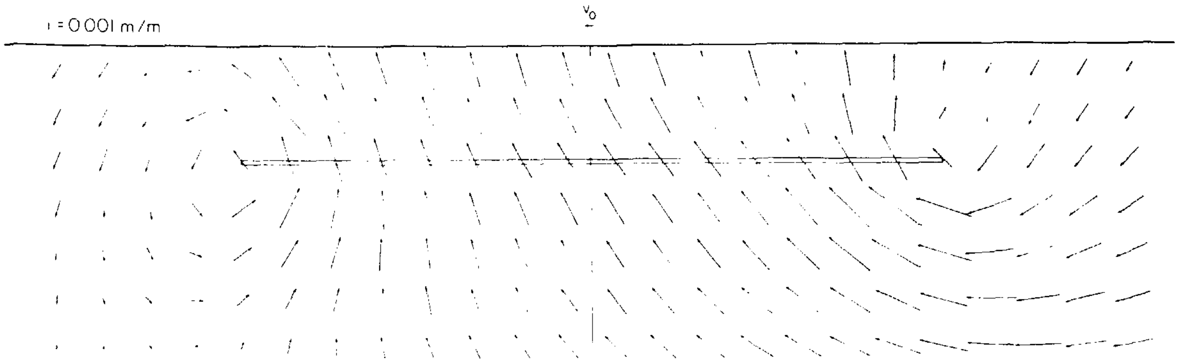

Figure 3.33 Convection cell in an extended vertical fracture around a repository; horizontal gradient = 0.001 m/m (Wang and Tsang, 1980).
[XBL 806-10035]

flow. The flow pattern in Figure 3.33 is essentially a superposition of the horizontal regional groundwater flow upon the flow of the unperturbed cells in Figure 3.32 (Wang and Tsang, 1980).

The stratigraphic sections and the depth dependence of permeabilty also distort the convection cells (Dames and Moore, 1978b; Burgess et al., 1979). The water particles move slowly in low-permeability zones and accelerate when entering high-permeability zones. In the GEIS and KBS studies on granite, permeability decreases monotonically with depth. For shale and basalt (Dames and Moore, 1978b; Hardy and Hocking, 1978), the possibility of high-permeability layers below the repository has also been considered.

3.6.3.4 Thermohydrologic effects: Further study needed

In waste repository studies, modeling of the long-term thermohydrologic impacts have only recently received attention in regard to the possibility of storing waste in hard rocks. Partly because of the uncertainty associated with hydrologic properties and partly because of the traditional focus on room closure in salt, the long-term consequences of hydrologic perturbation and the resulting impacts upon radionuclide migration have been shown to be only of generic importance. To date, the thermohydrologic effects have not been regarded as restrictive enough to be controlling factors in determining the repository thermal loading limit. In future site-specific studies, the anticipated response of the bulk thermohydrologic system to the maximum design thermal loadings must be carefully evaluated to determine if the long-term buoyancy perturbation is significant enough to be a limiting consideration in thermal loading designs.

4. CRITERIA EVALUATION

A number of research studies have developed thermal criteria for geologic radioactive waste repositories. These criteria cover all three levels of barriers in waste isolation: very-near-field (canister scale), near-field (repository scale), and far-field (geologic setting and accessible environment). They include both temperature limits and thermomechanical strength-to-stress ratios. These criteria have been gathered in the EIS document for the management of commercially generated radioactive waste (DOE, 1980a) and its supporting technical report (DOE, 1979b). They are presented in Section 4.1. Section 4.2 makes a careful review of these criteria and examines the data on which they are based and the models or experiments employed. An analysis of these results follows, with comments and notes from various publications in the literature concerning their scope and applicability; also included are the achievable thermal loading limits computed for salt, granite, basalt, and shale. Finally, Section 4.3 identifies current research needs, especially the lack of consideration given to radionuclide transport. Suggestions for further research are also made.

4.1 EXISTING THERMAL CRITERIA FOR NUCLEAR WASTE REPOSITORIES

4.1.1 Brief Description and Summary Tables

The existing thermal criteria for repository design studies are given in Table 4.1. They are grouped according to the general scale of operation.

4.1.1.1 Very-near-field criteria

Typical borosilicate waste glasses have a transition temperature of approximately 500°C, with a slightly higher softening temperature. Above the softening temperature, the heavy waste elements embedded by vitrification may migrate and form separate phases in the glass. Significant increases in cracking and in leach rates might occur. For calcined and sintered glass ceramic, the maximum acceptable temperatures of waste are 700°C and 800°C, respectively (Jenks, 1977). For spent fuel, the maximum pin temperature of 300°C was specified from stress rupture considerations.

Structural integrity of the waste package is also an important consideration. The austenitic stainless steel, probably 304L, that has been proposed for HLW canisters is known to undergo structural changes when heated above 400°C; it also exhibits increased susceptibility to stress cracking when exposed to water.

The maximum rock temperatures suggested will vary with the specific rock type. Generally, approximately 250°C may be appropriate for salt and shale, and 350°C may be acceptable for basalt and granite. However, since these properties depend on the site-specific characteristics of the hard rock, only a general range of temperature has been given.

Table 4.1. Thermal and Thermomechanical Limits for Conceptual Design Studies.

Event	Limits
Very-Near-Field Considerations	
Maximum HLW temperature as vitrified waste	500°C (Jenks, 1977)
Maximum spent fuel pin temperature	300°C (Blackburn et al., 1978)
Maximum canister temperature	375°C (Jenks, 1977)
Maximum rock temperature	250°C to 350°C
Maximum fracture of nonsalt rock	15 cm annulus around canister (Russell, 1979)
Near-Field Considerations	
Room closure during ready retrievability period--salt	10 to 15% of original room opening (Russell, 1979)
Room stability--granite, basalt rock strength-to-stress ratio	2 within 1.5 m of openings (Dames and Moore, 1978a)
Room stability--shale with continuous support rock strength-to-stress ratio	1 within 1.5 m of openings (Dames and Moore, 1978a)
Pillar stability--nonsalt strength-to-stress ratio	2 across mid-height of pillar (Dames and Moore, 1978a)
Far-Field Considerations	
Maximum uplift over repository	1.2 to 1.5 m (Russell, 1979)
Temperature rise at surface	0.5°C (Science Applications, Inc., 1976)
Temperature rise in aquifers	6°C (Science Applications, Inc., 1976)

(DOE, 1979b, 1980a)

An annular zone of crushed rock, due to either the backfilling of crushed rock or to the fracturing of host rock, will occupy the space between the canister and the wall of the borehole. This zone of decreased thermal conductivity and increased permeability is limited to less than 15 cm to control the canister temperature and to limit the susceptibility of the canister to aqueous solutions.

4.1.1.2 Near-field criteria

At the room-and-pillar scale, certain thermomechanical considerations must be taken into account to ensure safety in operation and to maintain retrievability of the waste if a decision on waste retrieval were made. These limits vary according to the thermal and mechanical properties of the host rock types. The retrieval operation can proceed if the rooms and tunnels remain accessible and the excavation is stable.

For salt, calculated room closures of less than 10-15% of the original room height imply that the repository will generally remain structurally stable throughout the retrieval period; however, local rock conditions not accounted for in the analysis may result in some local failure.

The stresses around granite or basalt excavations are limited to no greater than one-half of the rock mass strength to ensure that no minor instabilities occur. If any minor failures were to occur, they would probably take place instantly and give rise to additional and larger failures before repair could be made.

Shale is weaker and more ductile than granite and basalt. Engineered supports may be required to construct excavations and assure stability. With proper and continuous support, the strength/stress ratio of the perimeter of the opening may be less than 2 but should be greater than 1.0 within a 1.5 m skin around the excavation.

To ensure pillar stability, the average stress across the mid-height of the pillar should be less than one-half of the uniaxial compressive strength. This criterion was established from past mining experience in hard rocks (Jaeger and Cook, 1979). If the edges of the pillar are subjected to stresses greater than one-half of its rock mass strength, reinforcement of the excavation sidewalls is required to maintain the integrity of the pillar.

4.1.1.3 Far-field criteria

On the regional scale it is necessary to limit the surface uplift over the repository centerline. A number of modeling studies have shown that 1.2 to 1.5 m would be the maximum uplift to be expected over a repository in domed salt (DOE, 1980a). This range of surface uplift was obtained by linear thermomechanical expansion studies for a 37-W/m^2 (150-kW/acre) repository. In the same study, similar calculations for granite and basalt for loadings of 47 W/m^2 (190 kW/acre) and for shale for 30 W/m^2 (120 kW/acre) show less than 0.4 m of surface uplift. This limit is one that must be evaluated at a specific site, since the effects of the rock mass movement on the hydrology and geology of the region will vary according to the site.

To avoid problems with the biota (both flora and fauna), the temperature rise should be less than 0.5°C at the repository centerline surface. This is also site specific.

The temperature rise in aquifers is limited to less than 6°C to prevent thermal pollution in the region overlying the repository. There have also been proposals to limit the temperature rise in stagnant aquifers 30 and 90 m deep to 8° and 28°C, respectively. This limit must be re-evaluated for each specific site because of variations in geohydrologic and geochemical conditions.

4.1.2 Repository Thermal Load Limits Based on Thermal Criteria

In the DOE literature, a seven-step iterative procedure has been employed using the above thermal criteria to determine acceptable thermal loads for a conceptual repository. The local areal thermal load is first adjusted so that room-and-pillar stability is maintained during a readily retrievable period of at least five years. For reprocessed HLW, the canister load is kept low enough to stay within the very-near-field limits of Table 4.1. Calculations are then made to ensure that the far-field thermal and surface uplift criteria are fulfilled. In these calculations, reasonable estimates were made for the properties of the engineered barriers and geologic formations. Several simplifying assumptions were also made: (1) the repository is loaded simultaneously and instantaneously, (2) the presence of water is neglected, and (3) only the spent fuel (once through) and U + Pu HLW (total recycle) fuel cycles are considered. The results are tabulated in Table 4.2. The thermal load limit and its controlling thermal criterion are given for each scale of operation and for each type of the four geologic media selected as possible repository environments. (Particular attention should be given to the notes below Table 4.2.) The age of the spent fuel and reprocessed HLW are assumed to be 6.5 years.

The thermal load limits given in the table for vitrified HLW are widely accepted values for the canister loading in each geologic media. The areal thermal loading limits are dominated mainly by the criteria for room closure, with the exception of salt. The parameter of surface uplift is strongly restrictive for spent fuel in salt because the additional heat generated by its actinides operates over the long term. Thus, in a spent fuel repository in salt, a thermal loading of 15 W/m^2 (60 kW/acre) is needed to control the surface uplift in the far-field.

Some additional adjustments have been applied to the above figures on the basis of engineering and operational constraints. In order to ensure a conservative estimate of repository capacity, design areal thermal loadings were taken at two-thirds of the value of Table 4.2. Moreover, assuming 6.5-year-old waste rather than 10-year-old waste provided an additional degree of conservatism, since the thermal criteria were based on 10-year-old waste with lower thermal output. The resulting thermal loadings are given in Table 4.3. In each case an asterisk denotes the limiting thermal parameter for the given loading and emplacement medium.

From the table it is apparent that in nearly all cases, the near-field structural limitations are the limiting parameters. For spent fuel in salt, as previously mentioned, the far-field criterion of surface uplift is the

Table 4.2. Thermal Load Limits for Conceptual Repository Designs.

	Thermal Load Limit (controlling factor)[a]			
	Salt	Granite	Basalt	Shale
Canister Limits During Retrieval Period (kW)[b]				
Vitrified glass HLW	3.2(A)	1.7(A)	1.3(A)	1.2(A)
Calcined HLW	2.6(A)	1.6(A)	1.1(A)	1.1(A)
Near-Field Local Areal Thermal Loading Limits (W/m^2)[c]				
5-year retrieval - HLW	37(B)	47(B)	47(B)	30(B)
5-year retrieval - SF	e	47(B)[f]	47(B)	47(B)[f]
Far-Field Average Repository Thermal Loading Limits (W/m^2)[d]				
HLW	37(C)	47(B)	47(B)	30(B)
SF	15(C)	47(B)	47(B)	30(B)

(DOE, 1980a)

[a] Controlling factors: A = canister temperature limit, B = room closure, C = earth surface uplift.

[b] Analysis assumes 15-cm annulus of crushed rock around waste package.

[c] Area includes rooms and adjacent pillars, but not corridors, buttress pillars, and receiving areas. To convert to kW/acre, multiply by 4.05.

[d] Area includes storage area for waste, including corridors and ventilation drifts, but does not include area for shafts or storage areas for other waste types if separate. To convert to kW/acre, multiply by 4.05.

[e] In salt, the emplacement of spent fuel and HLW with plutonium is controlled by the more restrictive 15 W/m^2 (60 kW/acre) far-field thermal limit. Otherwise the near-field limit would be 37 W/m^2 (150 kW/acre).

[f] In order to maintain spent fuel cladding temperatures within the 300°C limit with these areal thermal loadings, the annulus around the canister is left open (no backfill). Heat is transferred across this air space more readily than through crushed backfill material and results in cooler canister and cladding temperatures.

Table 4.3. **Thermal Loadings Achieved at Conceptual Repositories.**

Cycle	Thermal Loading at Emplacement	Salt	Granite	Basalt	Shale
Once-through	PWR				
	kW/canister	0.72	0.72	0.72	0.72
	Near-field local (W/m²)[a]	12	32*	32*	20*
	Far-field average (W/m²)[a]	10*	25	25	16
	BWR				
	kW/canister	0.22	0.22	0.22	0.22
	Near-field local (W/m²)[a]	12	32*	32*	14
	Far-field average (W/m²)[a]	10*	25	25	11
U and Pu Recycle	HLW				
	kW/canister	3.2	1.7	1.3	1.2
	Near-field local (W/m²)[a]	25*	32*	32*	20*
	Far-field average (W/m²)[a]	19	23	23	5
	RH-TRU (hulls)				
	kW/canister	0.32	0.32	0.32	.32
	Near-field local (W/m²)[a]	25*	23	19	
	Far-field average (W/m²)[a]	19	17	15	8

(DOE, 1980a)

[a] To convert to kW/acre, multiply by 4.05.

* Denotes limiting thermal parameter.

restriction. The conceptual designs consider both PWR and BWR canisters in the spent fuel repository and both reprocessed HLW and remotely handled transuranic waste (RH-TRU) for the recycling option repository. In the case of BWR in shale and RH-TRU in nonsalt media, structural limitations on canister placement limit thermal loading.

These tables demonstrate the use of thermal criteria in establishing design limits for a nuclear waste repository.

4.2 COMMENTS AND SUGGESTIONS IN THE LITERATURE CONCERNING THE EXISTING THERMAL CRITERIA

Many comments and suggestions have been raised concerning the existing thermal criteria. These comments are found both in the original research reports from which the criteria were drawn and in similar studies or later publications in which they were reviewed. They have been collected and summarized here to help evaluate the current status of thermal criteria for nuclear waste repositories.

4.2.1 Very-Near-Field Criteria

The canister and waste forms have characteristic thermal properties. In the very-near-field scale, they have led to the criteria limits in Table 4.1.

The limit of 500°C for HLW glass was determined on the assumption that the reprocessed HLW and plutonium are embedded together in the glass. Jenks (1977) noted that above the softening temperature (a few degrees above 500°C), plutonium and other heavy metal phases might separate and migrate. The 1977 EPA report gave a figure of 600°C based on earlier work. Approximately 50% of the plutonium would form a separate phase in the glass if it were above the softening temperature, since the solubility of plutonium in HLW glass may be only about 2% by weight. It should be noted, however, that separate storage of plutonium is commonly considered to be the most likely scheme for the uranium-only recycle case.

The canister temperature limit is also set by Jenks (1977). He cites previous research in which recommended maximum temperatures for austenitic stainless steel ranged from 277°C to 427°C (Mecham et al., 1976; Atlantic Richfield Hanford Company and Kaiser Engineers, 1977). These temperatures were modeled for 100-year retrievable storage in air, whereas Jenks assumed 5-year retrievable storage for HLW and 25 years for SF. He chose 375°C as a reasonable figure, partly because the canister should be at the maximum temperature for only about one year. Retrievable storage for longer periods or long-term containment of radionuclides by waste packages, as would be required by NRC's proposed regulation 10 CFR Part 60, may require a closer examination of the temperature limits for the canister material.

The limit on maximum fracture of nonsalt rock raises further problems related to the presence or absence of a canister backfill. The conceptual waste package is frequently described as a canister with sleeve surrounded by a backfill that may include materials for sorption of radionuclides, inhibiting water invasion, etc. (EPA, 1977; DOE, 1979b, 1980a; Klingsberg and Duguid, 1980). However, such a crushed-rock zone is likely to have a thermal conductivity an order of magnitude lower than that of solid rock, which could lead to unacceptably high canister wall temperatures. Lowry et al. (1980) have discussed the possible dangers of a crushed-rock annulus, whether from backfill or decrepitation. They stated that crushed rock should be avoided because of its effect on canister temperatures and that even a sleeve could have a deleterious effect.

The rock temperature range (250-350°C) is a very general limit. For example, one report suggested a maximum temperature of 500°C to limit decrepitation of basalt (Kaiser Engineers et al., 1980). For salt, some samples showed decrepitation in the 260-320°C range, but samples from another location heated to 400°C produced no decrepitation (Russell, 1979). The stated criteria at least provide a conservative limit, with the provision that the properties of the host media should be determined for the proposed repository site.

Although brine migration into the vicinity of the waste canister has been extensively studied for salt formations (Project Salt Vault, WIPP salt block, Avery Island, Asse salt mines) and the corrosion effects of hot brine on canister and waste have been generally recognized, the existing thermal criteria on waste and canister integrity do not explicitly take these thermochemical effects into account. Both thermal stress rupture and chemical corrosion over long periods of time must be considered for better thermal criteria in the very-near-field.

4.2.2 Near-Field Criteria

Cheverton and Turner (1972) proposed two temperature limits for repositories in salt. One was that temperatures should not be allowed to exceed 250°C in more than 1% of the salt within the boundaries of the waste canister array; this would control plastic flow in the immediate vicinity of the canister. This local criterion can be achieved by appropriate canister thermal loading (EPA, 1977). The other limit was that approximately 25% of the salt between canisters should be kept below 200°C. Concern for room collapse and the breakage of shale layers above the canisters led to the adoption of this limit. The EPA report (1977) realized that collapse is a thermomechanical matter concerning repository excavation design and involving salt mechanics considerations in addition to a general thermal limit. Russell (1979) also recognized this and accounted for different design configurations in his study. He formulated his collapse criterion as a room closure less than 10-15% of the original room opening rather than a mere temperature limit.

For hard rocks, the near-field criteria are framed in terms of strength-to-stress ratios (safety factors) for room stability based on the GEIS studies (Dames and Moore, 1978a). They have been supported by additional analyses in basalt (Kaiser Engineers et al., 1980). These criteria were based on experience in mines not subject to thermal loading. Much more work remains to be done to assess these criteria; the effects of heat in a fractured rock medium must be taken into account. Fractured media deform nonlinearly, and the adequacy of thermoelastic analyses remain to be checked.

4.2.3 Far-Field Criteria

Only three stated far-field criteria have found acceptance so far. The first is the 1.2- to 1.5-m limit on uplift over the repository. The EPA report (1977) points out the significance of this criterion. Geologic stresses could cause a breach of repository integrity. The NRC and Department of Interior have commented that although the "self-healing behavior" of salt is being relied on to preserve an impermeable seal between the repository and overlying aquifers by the DOE, such reliance is questionable and has yet to be fully and thoroughly investigated (DOE, 1980a). The worst case of thermally or mechanically induced fractures affording an entry for water into the salt has not been adequately addressed.

The 0.5°C-surface-temperature limit, chosen to avoid problems with the biota, appears to be reasonably achievable (Cheverton and Turner, 1972; Science Applications, Inc., 1976; EPA, 1977).

The 6°C limit on temperature rise in freshwater aquifers also appears to be achievable (Cheverton and Turner, 1972; Science Applications, Inc., 1976; EPA, 1977). This criterion is based on two limits of 8° and 28°C for aquifers 30 and 90 m deep, which the Science Applications, Inc. study took from the earlier research by Cheverton and Turner (1972). These are the only thermohydrologic criteria accepted so far. However, aquifers at depths of 30 and 90 m are clearly within the region easily accessed by man. The basis for concern about shallow aquifers is the effect of thermal pollution rather than radionuclide transport. The thermohydrologic effects between a repository (at least 460 m deep) and such aquifers (near the ground surface) is not considered in these criteria. The problems of radionuclide transport through the geologic setting by thermally induced buoyancy flow have not been addressed. Consequently, there is an important gap in the present set of existing thermal criteria.

4.3 RESEARCH NEEDS: CRITERIA CONCERNING RADIONUCLIDE TRANSPORT

The primary concern of waste isolation is to limit radionuclide transport from the repository to the accessible environment. The safety of underground operations is also of great importance, but it is generally believed that our scientific knowledge and engineering experience, supported by intensive in situ testing and monitoring, can provide the necessary margin of safety and confidence in the repository operations. However, the long-term prediction of radionuclide transport through rock formations is plagued by the high degree of uncertainty involved in the ability to characterize the relevant rock properties and the geologic processes that affect the movement of radionuclides.

The existing thermal criteria for repository environmental impacts are limited in scope and have only an indirect bearing on radionuclide transport. These criteria scarcely touch upon the far-field hydrologic regime, except for one suggested temperature limit for shallow aquifers. The near-field criteria address the thermomechanical stability around the excavation but not the zones beyond the heated region. The very-near-field criteria for peak temperatures focus on potential failure soon after emplacement. That the existing thermal criteria lack any bearing on radionuclide transport will be discussed later in this section.

4.3.1 Waste-Canister Integrity

By limiting the peak temperatures of the waste, canister, and rock wall, the waste package could be ensured to remain intact during the maximum thermal impact, which occurs within decades after emplacement. However, once the peak temperature is reached, the rate of cooling is much slower than the rate of heating. During the long periods of cooling, the migration of fluids through salt, tuff, and shale and the reflooding of nonsalt rocks via fractures will subject the canisters to hostile aqueous conditions. The major challenge for material scientists is to ensure the integrity of the waste package for a long period of time; i.e., it should be maintained for at least 1000 years, as would be required by NRC's proposed regulation 10 CFR Part 60. After the breaching of the waste package, the controlled release factor of 10^{-5} must be maintained. It is clear that this long-term waste-canister behavior depends, if not exclusively, at least significantly, on the thermal input. The rate of corrosion of the waste canister is expected to depend on the thermal environment. Therefore, it is ideal to minimize temperatures to reduce the corrosion rate. It is not clear that the current thermal criteria are conservative enough to meet these long-term goals.

4.3.2 Rock Stress and Deformation

The repository excavation could be ensured to remain open and safe for emplacement and/or retrieval operations if the near-field criteria for room closure in salt and the room-and-pillar stability criteria for nonsalt rocks are used. Beyond the heated region, however, tension zones exist. As the heat slowly transfers away from the repository, the host rock will undergo a tension-compression cycle. Rocks, especially fractured ones, are known to deform nonlinearly and exhibit hysteresis under cyclic loading. Thus pathways could be created for waste migration. One means of control is to ensure that the host rock would return, as close as possible, to its original state after the thermal pulse. To reduce the possibility of weakening the rock and increasing the permeability from the repository outward during the thermal pulse, limits would have to be placed on the allowable thermal loading. It is not clear whether the rock stress and deformation criteria are sufficient to limit the alteration of rock hydrologic properties beyond the excavation perimeter. Furthermore, most of the present thermomechanical analyses assume that the rock masses are thermoelastic. Thermoelastic analyses may not be sufficient to predict the in situ behavior of fractured rock masses. If it were determined that long-term radionuclide migration should be controlled by means of thermal loading limits, larger safety margins could be necessary to ensure that the existing fractures around the repository will not be opened, new cracks will not be created, and adsorption properties of the rock mass will not be altered to allow radionuclides to escape.

4.3.3 Hydrologic Disturbance

The far-field criteria for the surface uplift and temperature rise are peripheral to the main concern: transport of water and radionuclides to the surface.

Concern about uplift over the repository originated from the consideration of potential fracturing in the rock. For salt, especially domed salt, the thermal expansion of the salt block may lead to fracturing in the overburden and opening of channels for shallow water to reach the salt formation. For hard rocks with smaller thermal expansion coefficients, the uplifts are not regarded as a primary concern; i.e., not serious enough to warrant limiting the thermal loading. The uplift criteria indirectly address the potential perturbation of the hydrologic properties of the rock formation, but not the perturbation of the movement of water and radionuclides.

The limits on the temperature rise in stagnant, shallow aquifers do not directly address the movement of groundwater either. These limits mainly stem from an environmental concern over the effects of thermal pollution in near-surface soils. The issue of radionuclide transport from the repository to the accessible environment through the geologic setting cannot be quantified by the temperature rise at the surface. We have discussed the dependence of the thermohydrologic disturbance on the temperature rise over the whole rock formation rather than the local increase at the surface. Thus even the far-field thermal criteria lack a direct bearing on the central concern of radionuclide transport to the accessible environment.

5. EFFECTS OF THE SURFACE COOLING PERIOD ON THERMAL IMPACTS

For a given amount of waste, a long surface cooling period allows the short-lived radionuclides to decay and the heat power to decrease. In the United States, 10 years has been regarded as the standard cooling period. The basis for the choice of a 10-year cooling period is briefly discussed in Section 5.1. With longer surface cooling periods, the very-near-field temperature rises around the waste canister will be lower, as shown in Section 5.2. The effects of surface cooling on the near-field temperature rise in the repository depend sensitively on the waste density and thermal loading schemes, as discussed in Section 5.3. There is a growing consensus, especially in the European countries considering permanent disposal of reprocessed HLW, that an extension from 10 years to 100 years of cooling time may be beneficial. Section 5.4 discusses the trade-off between a longer surface cooling period and a higher waste loading density as determined by existing near-field thermomechanical criteria. Section 5.5 addresses the question of the cooling period and thermal loading from the long-term, far-field point of view.

5.1 ECONOMIC CONSIDERATIONS OF THE SURFACE COOLING PERIOD

Temporary storage of waste above ground allows heat generation rates and radiation intensities to decrease, thereby reducing subsequent treatment and disposal expenses. However, the maintenance of surface facilities can be expensive. For reprocessed HLW, the cost of waste solidification and waste transportation to and from the processing site depends on the waste age (Dillon et al., 1971). The summation of the predisposal and disposal expenses as a function of waste age at burial was shown to have a minimum of 3 to 10 years, and likely closer to 10 years because of increases in estimated expenses for the repository (Cheverton and Turner, 1972). Most of the subsequent thermal analyses have taken the work of Cheverton and Turner as the benchmark and considered 10 years as the optimal cooling period.

Several factors have since contributed to the uncertainties of the economics of mined geologic repositories. One factor is that SF is now being considered as a potential waste form. Currently, most of the discharged SF assemblies are stored in cooling ponds at reactor sites. Away-from-reactor storage facilities are being considered to accommodate the SF when the on-site capacities are saturated (DOE, 1979a, 1980a). With the delay in the repository start-up dates, the fraction of older wastes to be emplaced in the repository will increase (see Section 3.1.5). The waste form, thermal loading, and time delays are among the key variables in determining the expense of a repository. Uncertainties in repository costs are expected to be quite large (Forster, 1979).

The sensitivity of the expense of a repository to its design parameters has been evaluated in a number of recent studies (Kaiser Engineers, 1978b; Stearns-Roger Engineering Co., 1979; DOE, 1980a,b). The choices of rock type, thermal loading, and repository size all influence repository costs significantly. Table 5.1 gives representative cost figures for basalt and granite repositories for each of two thermal loadings. The total expenses per repository decrease with lower thermal loading. However, undiscounted unit costs increase sharply, indicating that the total cost reductions achieved are not in proportion to the decrease in waste receipts. Table 5.2 shows the effects of the repository size on the expenses for domed salt repositories. The total expenses per repository are insensitive to repository size. The undiscounted unit costs decrease as size increases.

Table 5.1. Estimated Repository Expenses for Varying Thermal Loadings.

Item	Basalt 25 W/m^2 (100 kW/acre)	Basalt 10 W/m^2 (40 kW/acre)	Granite 25 W/m^2 (100 kW/acre)	Granite 10 W/m^2 (40 kW/acre)
Canisters of SF Stored	400,000	160,000	400,000	160,000
MTHM Stored	170,000	68,000	170,000	68,000
Repository Area, km^2 (acres)	8.1 (2000)	8.1 (2000)	8.1 (2000)	8.1 (2000)
MTHM/m^2 (MTHM/acre)	0.021 (85)	0.0083 (34)	0.021 (85)	0.0083 (34)
Canisters/m^2 (per acre)	0.05 (200)	0.02 (80)	0.05 (200)	0.02 (80)
Years of Operation	32	14.9	32	14.9
Total Expense (M$)	3950	3150	3940	3130
Undiscounted Unit Cost ($/kg)	23.20	46.30	23.20	46.00

(DOE, 1980c)

Note: All expenses in 1980 dollars, unescalated, undiscounted.

Table 5.2. Effects of Size on Salt Dome Repository Expenses.

Item	Standardized Repository Size 4.9 km^2 (1200 acre)	8.1 km^2 (2000 acre)	11.3 km^2 (2800 acre)
Assumed Heat Loading W/m^2 (kW/acre)	10 (40)	10 (40)	10 (40)
Canisters of SF Stored	96,000	160,000	224,000
MTHM Stored	41,000	68,000	96,000
MTHM/m^2 (MTHM/acre)	0.0083 (34)	0.0083 (34)	0.0083 (34)
Canisters/m^2 (per acre)	0.02 (80)	0.02 (80)	0.02 (80)
Years of Operation	10.4	14.9	19.5
Total Expense (M$)	2470	2490	2850
Undiscounted Unit Costs ($/kg)	60.20	36.60	29.70

(DOE, 1980c)

Note: All expenses in 1980 dollars, unescalated, undiscounted.

The representative results given in Table 5.2 illustrate the sensitivity of repository expenses. It is important to point out that in the examples of economic analyses presented above, and in the repository design considerations discussed previously in Sections 3 and 4, the repository capacities are expressed in terms of areal thermal loadings. The thermal loading values are referred to a fixed waste age. When wastes of different ages (cooling periods) are considered, the corresponding thermal loading values change drastically (see description in Section 5.2). If the economic analyses of repository design based on 10-year-old wastes are used for different waste ages, the results must be carefully scaled to avoid nonconservative and erroneous conclusions. As will be shown in Sections 5.3-5.5, nonconservative conclusions could be made if long-term, far-field environmental considerations are not included in the economic analyses and design studies.

5.2 REDUCTION OF VERY-NEAR-FIELD THERMAL IMPACTS

In the very-near-field, the short-term rises in temperature around the waste canisters are very sensitive to the duration of the surface cooling period. If the structural integrity of the waste package and emplacement borehole is the primary concern, it is advantageous to allow the highly active waste elements to decay and the temperature to decrease.

5.2.1 Decay of Fission Products

Radioactive decay and neutron-induced nuclear reactions in the waste elements determine the radioactivity and the heat generation rate of the stored wastes. Table 5.3 shows the contributions from the principal radionuclides to the heat power of SF 10 years after being discharged from a PWR. The heat of 10-year-old waste is released mainly by the short-lived fission products. The time dependence of the heat power between 10 and 100 years will be determined mainly by Sr-90 and Cs-137, with half-lives of 28 and 30 years, respectively.

The main effect of a long surface cooling period is to reduce the concentration of the short-lived radionuclides. The heat released by the long-lived actinides Pu-239, Pu-240, and Am-241 will not be reduced significantly with the extension of cooling from 10 years to 100 years. The composition of waste content depends on the fuel cycles discussed in Section 3.1.1. Table 5.4 summarizes the heat generation rates for different waste ages and different fuel cycles from a given amount of waste (1 MTHM). Among the three fuel cycles, the reprocessed HLW with both U and Pu recycled contains the highest concentration of fission products, and the SF contains the highest concentration of actinides. The contrast in the waste compositions among different fuel cycles must be carefully taken into account in studying the effects of different surface cooling periods.

Table 5.3. Radionuclide Heat Generation Rates for 10-year-old PWR SF.

Principal Radionuclides	Half-life (years)	Heat Power	
		W/MTHM	% of Total
Fission Products			
Kr-85	10.76	7.4	0.6
Sr-90	28	71	6
Y-90[a]	7.39×10^{-3}	330	28
Sb-125	2.7	2.5	0.2
Cs-134	2.1	95	8
Cs-137	30	89	7
Ba-137[a]	4.94×10^{-6}	320	27
Pr-144[a]	3.29×10^{-5}	1.1	0.1
Pm-147	2.6	2.7	0.2
Eu-154	16	50	4.2
Actinides			
Pu-238	89	73	6.1
Pu-239	2.4×10^{4}	10	0.8
Pu-240	6.6×10^{4}	15	1.3
Pu-241	13	3.3	0.3
Am-241	458	59	5.0
Cm-244	18.1	47	3.9
Others		14	1.2
	Total	1190.0	100.0

(Arbital et al., 1979)

[a] Short-lived daughter products.

Table 5.4. Initial Heat Generation Rates for Different Waste Ages and Types.

Years after Discharge	Heat Power (W/MTHM)		
	Spent Fuel PWR	HLW: U + Pu Recycle	HLW: No Recycle
1	10430	11542	10274
2	5640	6177	5500
5	2011	2256	1867
10	1189	1340	1032
40	625	540	431
100	289	139	109

5.2.2 Decreases in Temperature Rise

When heat power decreases, the temperature rise decreases. Figure 5.1 illustrates the sensitive dependence of the very-near-field temperature rise on the duration of the surface cooling period. The temperature rises are for the point located at 0.26 m from the canister axis in the midplane of a single HLW canister measuring 0.162 m in diameter and 3 m in length and containing 2.09 MTHM of waste. If the thermal interaction from neighboring canisters is neglected, the very-near-field temperature rises in different rocks are approximately proportional to the inverse of the thermal conductivities, and the times are scaled by the thermal diffusivities controlling the heat conduction. The thermal properties in Table 3.14 are used in the scaling in Figure 5.1.

The very-near-field temperature rise is approximately proportional to the canister emplacement heat power, which decreases sharply with surface cooling duration. The short cooling periods are therefore especially effective in lowering the very-near-field thermal impacts. By applying the very-near-field thermal criteria (Section 4.1), the waste content in each reprocessed HLW canister can be determined. For an SF canister containing one fuel assembly, the waste content (0.4614 MTHM for PWR, 0.1833 MTHM for BWR) is lower than for a reprocessed HLW canister, and the very-near-field temperature rises around a single SF canister will be lower than those shown in Figure 5.1.

For a more concentrated SF canister, the very-near-field temperature rises are correspondingly higher. Table 5.5 lists the unit-cell results of Altenbach (1978) for the maximum salt temperature around a concentrated SF canister containing 650 fuel rods (or approximately 3 PWR fuel assemblies). (The area of the unit cell occupied by one canister is 5.5 m (18 ft) by 23.8 m (78 ft).) To show that the very-near-field temperature rise is approximately

Table 5.5. Peak Salt Temperature for Various Emplacement Times.

Surface Cooling Period (years)	Emplacement Power (kW/canister)	Maximum Temperature Rise (°C)[a]	Temperature to Power Ratio (°C/kW)	Time of Maximum (years)
0.4	28.8	510	18	0.26
1.4	11.6	221	19	0.46
3.4	4.61	88	19	0.59
5.4	2.81	57	20	1.02

[a] Ambient temperature 37°C (98°F) was subtracted from the results of Altenbach (1978).

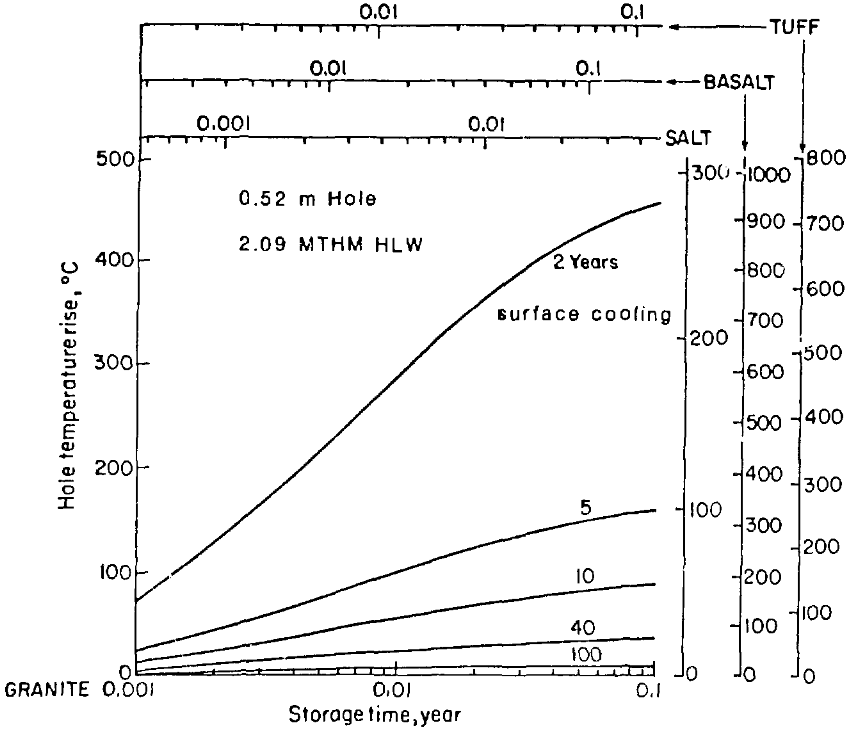

Figure 5.1 Temperature rise at the borehole as a function of surface cooling period. [XBL 817-3278]

proportional to emplacement power, the ratio of borehole temperature rise to initial canister power is also included in Table 5.5. The unit-cell results also take into account the thermal interaction among neighboring canisters. For cases with high areal waste density and close canister spacing, the thermal interaction is an important factor in determining the maximum rock temperature. As a result of thermal interaction, the temperature rise from the combined contribution of the canister array is higher than that induced by a single canister.

These analyses illustrate that the sensitive thermal effects near the waste package at short times are approximately proportional to the emplacement heat power. With a long surface cooling period, the heat power at emplacement decreases and the temperature rise decreases correspondingly. The potential advantage of a long surface cooling period is to lower the likelihood of undesirable thermally induced effects such as canister cracking and borehole degradation.

5.3 WASTE DENSITY AND THERMAL LOADING

If the waste content of each canister and the spacing between the canisters are fixed, it is obvious that longer surface cooling will reduce the thermal impact at all scales of operation and over all times. However, economic considerations may require that the repository design take advantage of the lower thermal impact by utilizing a more concentrated emplacement scheme. Sections 5.3.1-5.3.3 discuss and compare the results of different loading schemes.

5.3.1 Loading at Constant Waste Density

For a standard repository with 10-year-old SF stored at a thermal loading of 10 W/m^2 (40 kW/acre), the corresponding waste density is 0.0083 $MTHM/m^2$ (33.6 MTHM/acre). Figure 5.2 illustrates the dependence of the repository temperature increase on the surface cooling period if this waste density is held fixed.

Each curve in Figure 5.2 has several peaks or bumps at different times. The 1-year cooling curve shows this structure most clearly. These features appear before 10 years and around 10^2, 10^3, and 10^4 years. The short-time features diminish rapidly with longer surface cooling periods. The earlier peaks originate from the short-lived radionuclides. The long-lived radionuclides control the long-term thermal impact. Although the surface cooling periods drastically change the repository peak temperature, the repository will nevertheless have a significant temperature rise above the ambient for over 10^4 years, even for a surface cooling period of 100 years.

The results in Figure 5.2 are calculated with a uniform disk repository model in granite and scaled to other rock formations by the inverse square root of thermal conductivity-volumetric heat capacity products (see Equation

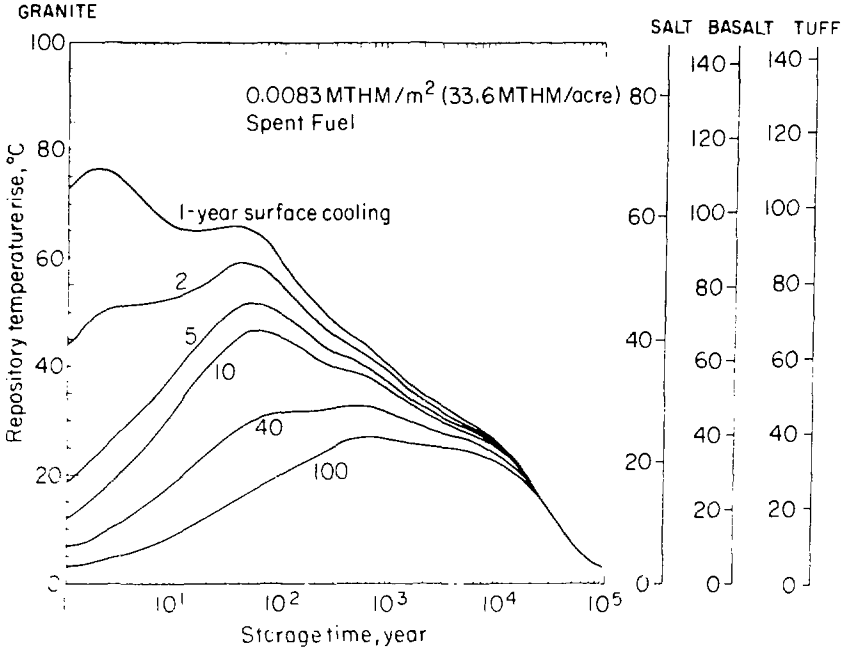

Figure 5.2 Temperature rise in an SF repository as a function of surface cooling period; constant waste density loading. [XBL 817-3274]

(A-6) in the Appendix). This scaling is exact if the heat power is a single-term exponential decay function. The small differences in the shapes of the temperature rise curves among different rocks (see, for example, Figure 3.25) are neglected in the scaling used in Figures 5.2-5.7.

If reprocessed HLW instead of SF is stored with the same waste density, extending the cooling time from 10 years to 100 years can lower the repository temperature much more effectively. This is shown in Figure 5.3. Reprocessed HLW has most of its long-lived actinides removed, and the repository temperature can therefore return to ambient much faster. If an effective half-life of 30 years is assumed for the reprocessed wastes, a 40-year cooling will reduce the activity of 10-year-old waste by half and a 100-year cooling period will lower the heat power of 10-year-old waste by a factor of 8.

If the waste density is fixed, the results for both SF and reprocessed HLW indicate that surface cooling can lower the peak temperature at early times. The long-term thermal impact depends on the actinide content, which is not sensitive to the duration of the surface cooling period.

5.3.2 Loading at Constant Emplacement Power Density

In this section, we will consider the same repositories but with a constant thermal power density instead of a constant waste density. Two reasons have prompted us to present these results. First, the results of most economic analyses and repository design studies are presented in terms of areal thermal loading. It is interesting to study the effects of surface cooling by treating the familiar areal thermal loading (W/m^2 or kW/acre) as a fixed parameter. Second, the very-near-field analysis discussed in Section 5.2 indicates that the rises in temperature around the waste package scale with emplacement heat power. If only the very-near-field impacts are considered, a constant heat power will induce the same thermal impact, independent of the waste age.

Figure 5.4 illustrates the dependence of temperature increase on the surface cooling period of an SF repository with a constant emplacement power density of 10 W/m^2 (40 kW/acre). With the same emplacement power density, the repository temperature rises do not stay the same with different cooling periods. With longer cooling periods, the repository temperature increases. This anomalous trend is especially drastic for surface cooling periods changing from 10 to 40 to 100 years.

By imposing the same emplacement heat power density, the 100-year-old waste, with most of the short-lived fission products already decayed, will contain a substantially larger amount of actinides per unit acre. Therefore, the long-term thermal impact increases accordingly. For reprocessed HLW, the anomalous difference between 10-year cooling and 100-year cooling is less drastic, as shown in Figure 5.5.

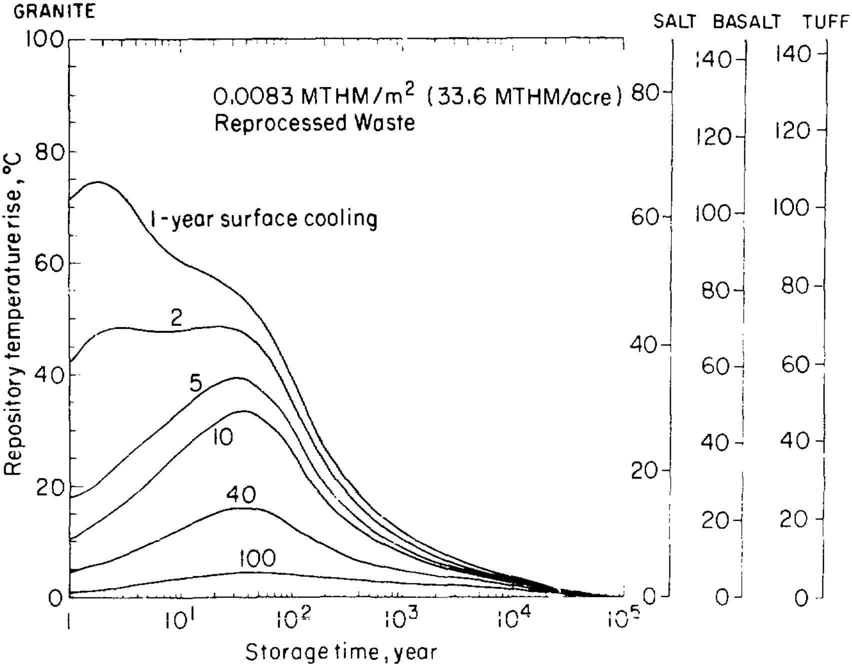

Figure 5.3 Temperature rise in an HLW repository as a function of surface cooling period; constant waste density loading. [XBL 817-3275]

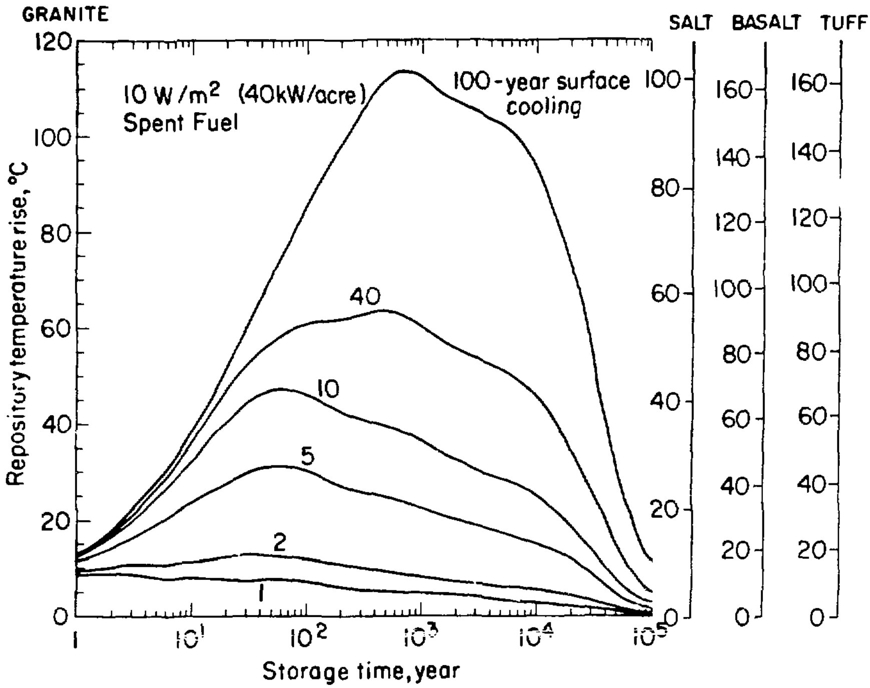

Figure 5.4 Temperature rise in an SF repository as a function of surface cooling period; constant emplacement power density loading. [XBL 817-3277]

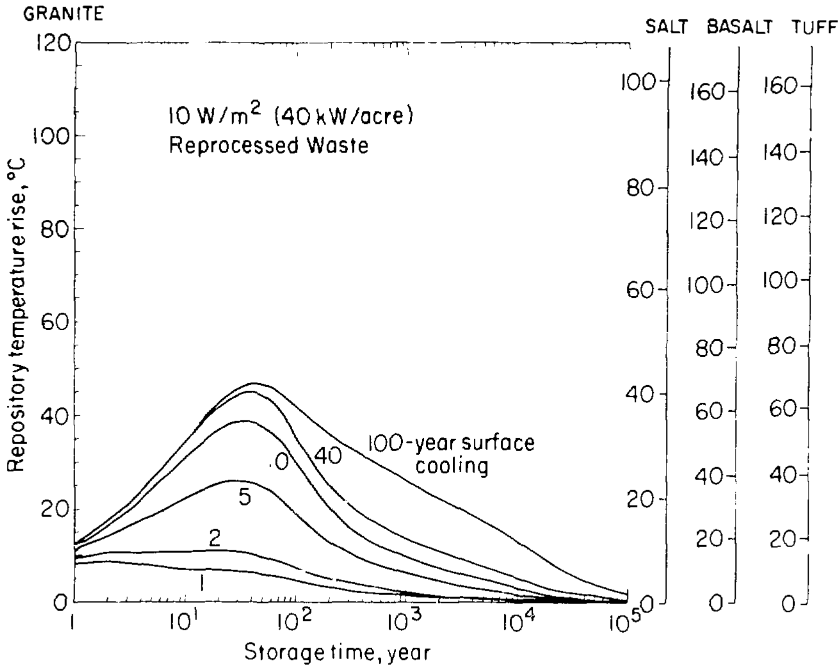

Figure 5.5 Temperature rise in an HLW repository as a function of surface cooling period; constant empla ...ent power density loading. [XBL 817-3276]

Unlike the very-near-field impacts which scale approximately with emplacement heat power alone, the near-field and far-field thermal impacts depend on both the initial power and the subsequent time decay. Since the heat power decay depends sensitively on the waste content, the effects of surface cooling must be assessed for each different fuel cycle.

5.3.3 Comparison of Different Loading Schemes

The results of Section 5.3.1 for constant waste density and the results of Section 5.3.2 for constant power density are summarized in Figures 5.6 and 5.7, respectively. These results illustrate that entirely different conclusions can be made about the effects of surface cooling periods, depending on the parameters chosen and the waste type considered. Comparing the 10-year results with the 100-year results leads to the following conclusions.

- With constant waste density, surface cooling reduces the thermal impacts significantly for reprocessed HLW but modestly for SF.
- With constant emplacement power density, surface cooling increases the thermal impact significantly for SF but very modestly for reprocessed HLW.

We emphasize these points because the thermal loading density has been the most frequently used parameter in the literature. The relationship between thermal loading and waste density has been recognized implicitly in some of the literature. We hope that the above explicit comparison will call attention to this simple but important difference.

5.4 OPTIMIZATION OF WASTE LOADING WITH NEAR-FIELD CRITERIA

Waste loading density is one of the key parameters in the design of a repository. Near-field thermomechanical criteria have been used to determine the optimal waste loading density for 10-year-old wastes (see Sections 3.5.2 and 4.1.2). With longer surface cooling periods to lower the heat power per unit waste, it seems logical to allow a more concentrated waste loading scheme. Section 5.4.1 discusses the allowable waste loading densities based on the temperature rise criteria. In Section 5.4.2, the existing thermomechanical criteria for different rock types are used to extend the design densities for 10-year-old wastes to older wastes.

5.4.1 Allowable Loading with Near-Field Temperature Criteria

In the classical conceptual design study for salt, Cheverton and Turner (1972) used the 1% and 25% salt temperature criteria to determine the maximum permissible loading density. Between the canisters, no more than 1% of the salt was allowed to have a temperature above 250°C and no more than 25% of the salt was allowed to have a temperature above 200°C. The results for a room

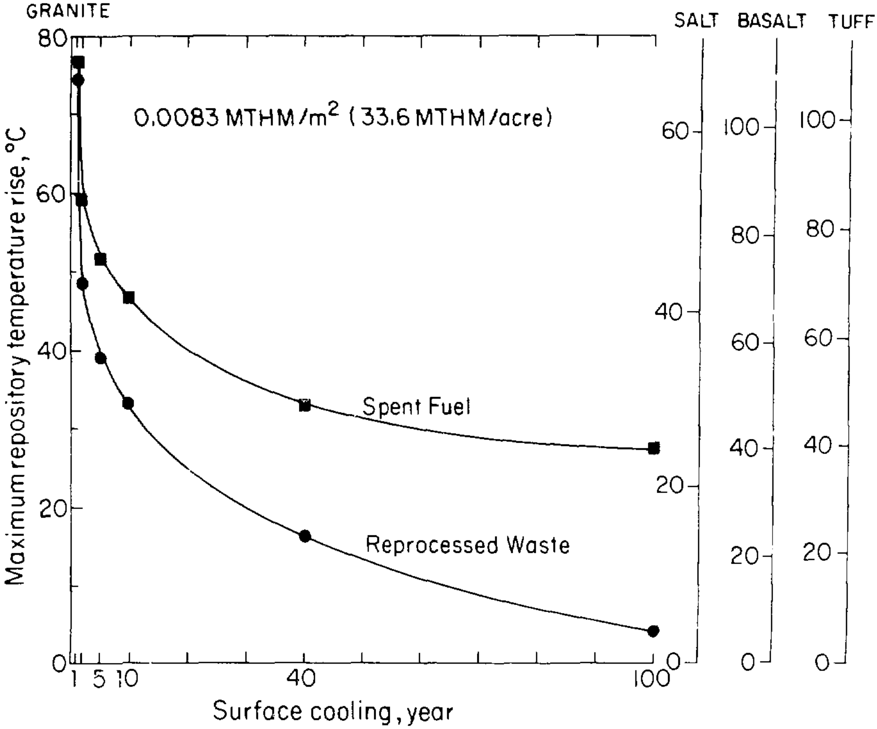

Figure 5.6 Maximum repository temperature rise as a function of surface cooling period; constant waste density loading for SF and HLW. [XBL 817-3270]

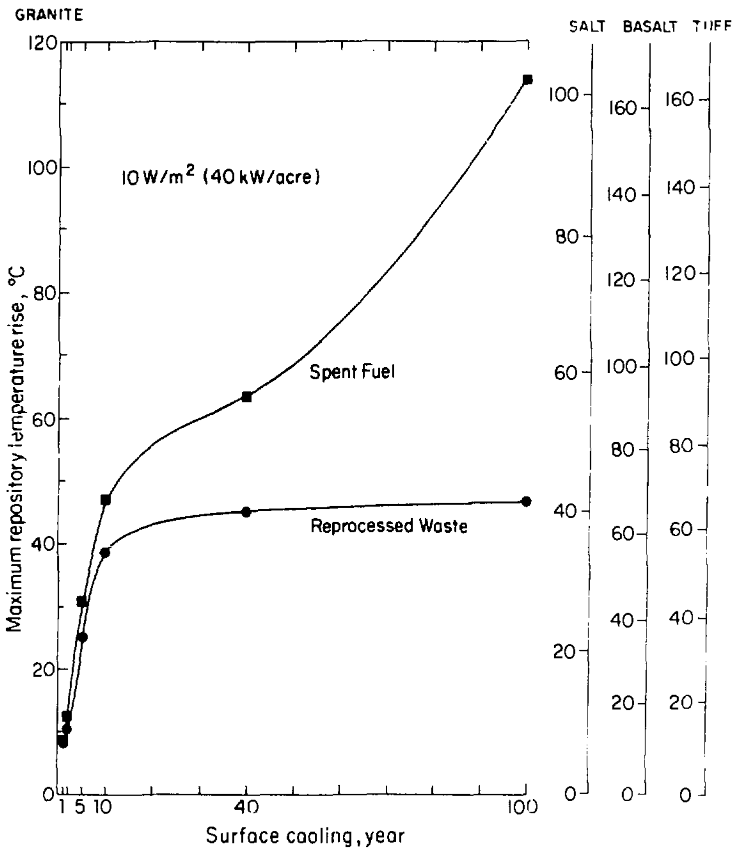

Figure 5.7 Maximum repository temperature rise as a function of surface cooling period; constant emplacement power density loading for SF and HLW. [XBL 817-3271]

4.6 m wide are shown in Figure 5.8. The results were calculated with 1-, 4-, and 10-year-old reprocessed HLW and extrapolated to 100 years on the assumption that the effective half-life of the waste is 30 years. (A constant half-life results in a constant maximum permissible emplacement power level per waste package.) For a waste age of less than 4 years, the 1% salt criterion (very-near-field) becomes limiting and the permissible loading depends on the pitch along a canister row. For older wastes, the 25% salt criterion (near-field) determines the loading.

For older wastes with near-field instead of very-near-field criteria controlling the permissible loading density, we could use a simpler model that assumes a uniform waste loading to calculate the maximum repository temperature and determines the allowable waste densities accordingly. For example, Table 5.6 illustrates the acceptable waste density and the corresponding thermal loading in granite if a 47°C temperature rise limit is imposed as a criterion. The 47°C temperature rise is specified for a granite repository containing 10-year-old spent fuel loaded at a thermal density of 10 W/m^2 (40 kW/acre). With longer surface cooling, it is shown that the same repository can accommodate more waste, but only if emplacement power density is lower.

The maximum repository temperature determines the thermomechanical stability of the mined repositories. In the current thermal design criteria for different rock types, the maximum allowable temperature-rise limits are replaced by thermomechanical criteria. The results of the extension of thermomechanical analyses for 10-year-old wastes to older wastes are discussed in the next section.

Table 5.6. Waste Density and Thermal Loading for Repository Temperature Rise of 47°C in Granite.

	Spent Fuel		Reprocessed HLW	
Surface Cooling Period (years)	MTHM/m^2 (MTHM/acre)	W/m^2 (kW/acre)	MTHM/m^2 (MTHM/acre)	W/m^2 (kW/acre)
10	0.0083 (33.6)	10.0 (40)	0.0117 (47.2)	12.0 (49)
40	0.0118 (47.7)	7.4 (30)	0.0240 (97.2)	10.3 (42)
100	0.0142 (57.5)	4.1 (17)	0.0921 (373)	10.1 (41)

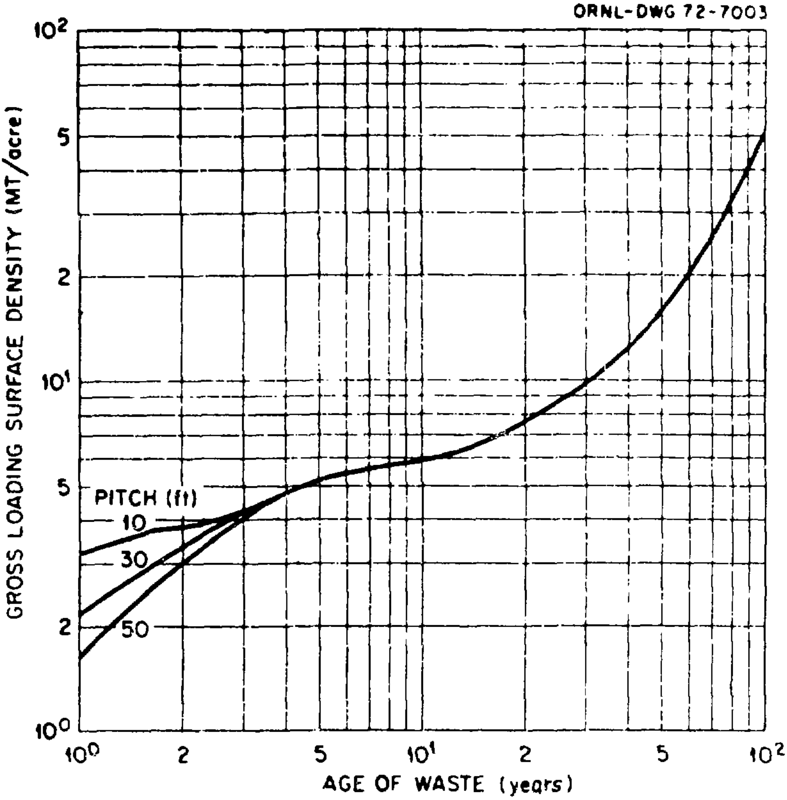

Figure 5.8 Maximum permissible gross loading surface density for a 4.6-m (15-ft) room versus age of waste, based on salt temperature criteria (Cheverton and Turner, 1972). [XBL 819-11600]

5.4.2 Allowable Waste Density with Near-Field Thermomechanical Criteria

The existing thermomechanical criteria are expressed in terms of strain of room closure for salt and strength-to-stress ratios for hard rock repositories (shown earlier in Table 4.1). Imposing these existing criteria on older wastes enables allowable waste densities to be determined. The salt and nonsalt analyses are discussed in the following two subsections.

5.4.2.1 Reduction of strain for room convergence in salt

Room convergence in salt mines depends on the temperature, pillar stress, and time. In Project Salt Vault (Bradshaw and McClain, 1971), the results of model pillar tests of rock salt from the Lyons mine were fitted with an analytic formula called Lomenick's formula (see Appendix, Section A.2). It has been used in the NWTS conceptual designs for domed and bedded salt (Kaiser Engineers, 1978a,b; Stearns-Roger Engineering Co., 1979) and in the NWTS conceptual reference repository description (Bechtel, 1981). In this section we extend the NWTS results for 10-year-old wastes to older wastes.

The NWTS reference salt repository contained 10-year-old wastes emplaced at 37 W/m^2 (150 kW/acre) at 640 m (2100 ft) depth with an average pillar stress of 14.5 MPa (2100 psi). The waste storage rooms were 6.1 m (20 ft) wide and 4.8 m (15 ft 9 in.) high. After 5 years, however, the roof height had shortened by 0.23 m (9 in.). Older wastes stored in the same room and at the same waste emplacement density have a lower average temperature rise at 5 years, thereby reducing the cumulative room convergence (Fig. 5.9). If 0.23 m of room convergence (5% linear strain) is acceptable for safe operations in the repository, the waste emplacement density can be increased. The allowable waste densities and the corresponding thermal densities are tabulated in Tables 5.7 and 5.8 along with the results for hard rocks, discussed in the following subsection.

5.4.2.2 Reduction of strength-to-stress ratios in granite, basalt, and shale

The stress fields around a room in hard rocks such as granite, basalt, and shale depend on the temperature, the in situ stress field, and the change in load due to excavation (see Section 3.5.2). The thermomechanical stability limits for mined repositories in hard rock were established in the GEIS study (Dames and Moore, 1978a; DOE, 1979b). These near-field criteria determine the repository loading density of 10-year-old wastes (see Section 4.1.2).

The near-field thermomechanical criteria are expressed in terms of strength-to-stress ratios, as shown earlier in Table 4.1. The repositories contain 10-year-old wastes stored at a thermal power density of 47 W/m^2 (190 kW/acre) in granite and basalt and 30 W/m^2 (120 kW/acre) in shale. At 5 years after waste emplacement, the sum of the thermally induced stress and the excavation-induced stress within 1.5 m of the openings is half the magnitude of the rock strength for granite and basalt and equal to the rock strength for

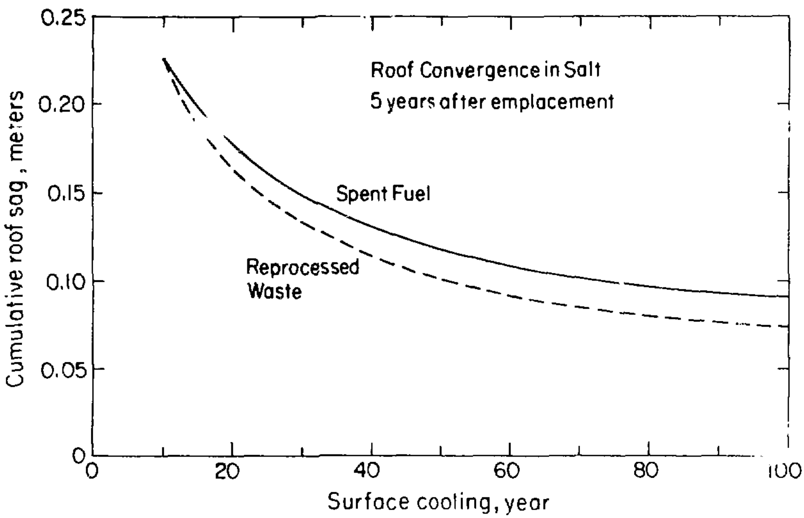

Figure 5.9 Roof convergence in salt as a function of surface cooling period in SF and HLW repositories. [XBL 823-2003]

Table 5.7. Allowable Waste Density Determined by the Near-Field Thermomechanical Criteria.

Surface Cooling Period (years)	Salt SF	Salt HLW	Granite SF	Granite HLW	Basalt SF	Basalt HLW	Shale SF	Shale HLW
	$MTHM/m^2$ (MTHM/acre)		$MTHM/m^2$ (MTHM/acre)		$MTHM/m^2$ (MTHM/acre)		$MTHM/m^2$ (MTHM/acre)	
10	0.0311 (126.0)	0.0359 (145.2)	0.0394 (159.6)	0.0454 (183.9)	0.0394 (159.6)	0.0454 (183.9)	0.0249 (100.8)	0.0287 (116.1)
40	0.0534 (216.1)	0.0713 (288.7)	0.0797 (322.5)	0.1173 (474.6)	0.0699 (282.7)	0.1011 (409.2)	0.0441 (178.5)	0.0639 (258.4)
100	0.0786 (318.2)	0.1112 (450.0)	0.1759 (711.8)	0.4799 (1942)	0.1478 (598.1)	0.3966 (1605)	0.0933 (377.8)	0.2505 (1014)

Table 5.8. Allowable Thermal Loading Density Determined by the Near-Field Thermomechanical Criteria.

Surface Cooling Period (years)	Salt SF	Salt HLW	Granite SF	Granite HLW	Basalt SF	Basalt HLW	Shale SF	Shale HLW
	W/m^2 (kW/acre)		W/m^2 (kW/acre)		W/m^2 (kW/acre)		W/m^2 (kW/acre)	
10	37 (150)	37 (150)	47 (190)	47 (190)	47 (190)	47 (190)	30 (120)	30 (120)
40	33 (135)	31 (124)	50 (201)	51 (204)	44 (177)	44 (176)	28 (111)	28 (111)
100	23 (92)	12 (49)	51 (205)	52 (212)	43 (173)	43 (175)	27 (109)	27 (111)

shale. Older wastes stored at the same waste emplacement density have a lower average temperature rise after 5 years, and the thermally induced stress is less (Fig. 5.10). The temperature rises at the end of 5 years are used to determine the stress values. If the same strength-to-stress ratio criteria can be used for older wastes to ensure mine stability, the waste emplacement density can be increased to accommodate more wastes in the repository, as shown in Table 5.7. The temperature dependence of the rock strength is taken into account. The corresponding thermal loading densities are less sensitive to the surface cooling period, as shown in Table 5.8.

5.4.2.3 Increase of waste emplacement density

The ratios of allowable waste densities of older wastes (tabulated in Table 5.7) to the values of 10-year-old wastes are illustrated in Figure 5.11. It shows that older wastes could be emplaced at more concentrated densities. These results are based on the assumption that the near-field thermomechanical criteria developed for 10-year-old wastes are acceptable independent of the surface cooling period. For reprocessed HLW with a small thermal contribution from the long-lived actinides, these conclusions may be valid. However, for spent fuel repositories, the long-term, far-field effects could become the limiting consideration. This is discussed in Section 5.5.

Figure 5.11 also shows that the increase in allowable waste density is modest for salt compared to the results for hard rocks. For salt, the increase in allowable waste density grows at a slower rate for the longer surface cooling times. Thus the option of a longer surface cooling period may be less attractive for salt than for hard rock repositories. The difference in the form of the curves for salt and for hard rocks results mainly from the different thermomechanical behaviors assumed in the analyses. For salt, the plastic creep strain is proportional to $(T_{amb} + \Delta T)^{9.5}$, where T_{amb} is the ambient temperature in kelvins and ΔT is the waste-induced temperature rise (see Lomenick's formula in the Appendix, Section A.2). For hard rocks, thermoelasticity is assumed for the stress changes, and the thermally induced stress is proportional to ΔT. As longer surface cooling periods lower the temperature rise, ΔT, the nonlinear temperature dependence of the creep for salt shows less sensitivity to ΔT, resulting in a smaller increase in allowable waste density.

The thermoelasticity assumed for hard rocks may be oversimplified in view of the potential nonlinear contributions from the presence of fractures. Additional research beyond the scope of this report is required to study the thermomechanical behavior of fractured rock masses. The temperature dependence of the elastic constants are also not taken into account in the calculations. The dependence of rock strength on temperature, however, is taken into account. Within the temperature range of interest for these calculations (below 120°C or 250°F), granite exhibits a noticeable change in rock strength with temperature, whereas the basalt and shale strengths are almost temperature independent (Dames and Moore, 1978a). For cooling periods of 10 to 100 years, this gain in strength with lower temperature permits an approximate increase of 20% in

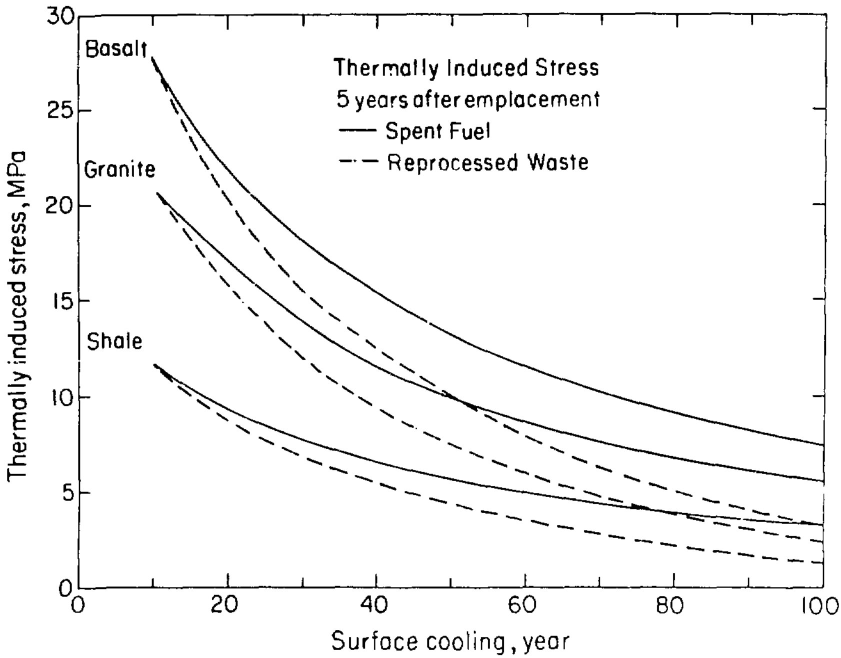

Figure 5.10 Thermally induced stress in granite at 47 W/m^2 (190 kW/acre), basalt at 47 W/m^2 (190 kW/acre), and shale at 30 W/m^2 (120 kW/acre) as a function of surface cooling period in SF and HLW repositories. [XBL 823-2002]

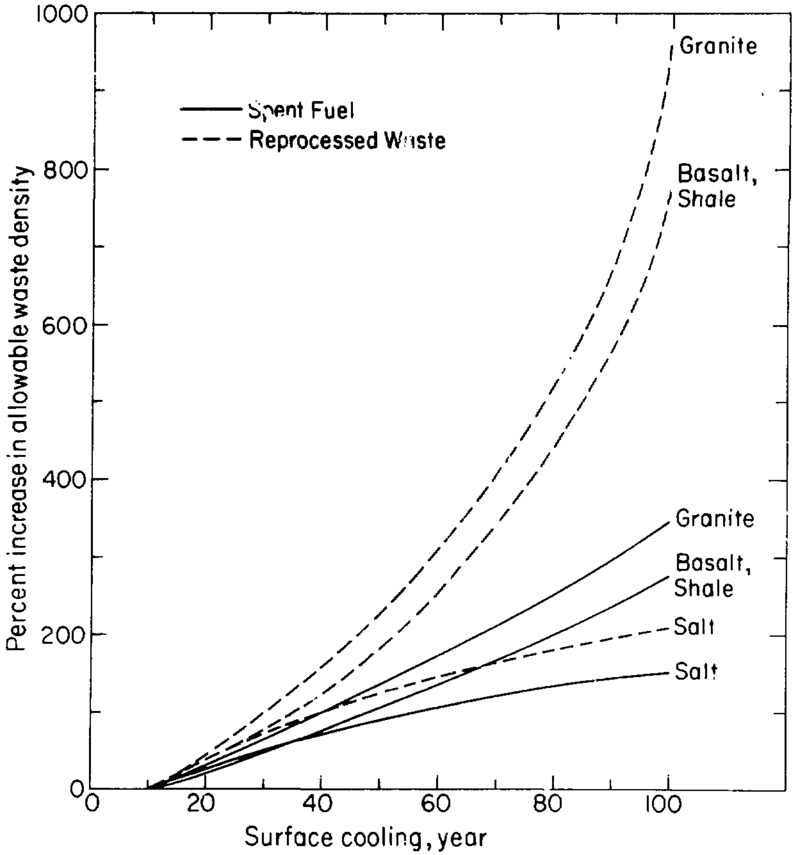

Figure 5.11 Percent increase in allowable waste density in four major rock types as a function of surface cooling period for SF and HLW repositories. [XBL 825-2211]

waste density for granite relative to its allowable limit for a fixed strength at 120°C. Since the mechanical properties and rock strengths are highly site specific, the quantitative conclusions in these calculations should be carefully re-evaluated for any specific rock type and any potential repository site.

5.5 LIMITATION OF LOADING BY LONG-TERM, FAR-FIELD CONSTRAINTS

The long-term, far-field thermohydromechanical effects depend on the temperature rise in the host rock, especially in the region between the repository and the surface. Before we present more detailed results for the optimal loadings based on far-field considerations, it is of interest to present one example of the far-field temperatures in granite induced by a repository uniformly loaded with a constant mass density. Figure 5.12 illustrates the temperature rise at a point midway between the repository and the surface. Similar magnitudes of the maximum temperature rises are expected for other rocks, since the volumetric heat capacities controlling the far-field temperature are insensitive to the differences in rock types. Comparing these results with the near-field results shown earlier in Figures 5.2 and 5.3 makes it evident that the far-field results are less sensitive to the surface cooling period, especially for the SF repository.

The temperature rise from the repository to the surface determines the surface uplift and the buoyancy flow. The repository loading density could be limited by these far-field constraints. These constraints are discussed in the following two sections.

5.5.1 Surface Uplift Considerations

To illustrate that the surface uplift considerations can determine waste loading densities, we will summarize the results of the final EIS (DOE, 1980a) on the effect of waste age for salt, granite, basalt, and shale. The existing thermal criteria discussed in Section 4.1 were used to determine the maximum thermal loading for both SF and reprocessed HLW at 5, 10, and 50 years of age. The loading takes into account the temperature and thermomechanical limitations listed in Table 4.1. These limitations include the maximum allowable temperatures at waste centerline, canister surface, and borehole wall; the maximum room closure for salt; strength-to-stress ratios for hard rocks; and the maximum allowable surface uplifts.

The final thermal loadings used in the EIS study are shown in Table 5.9. The far-field average loading takes into account the unused passive areas for corridors, etc. A safety margin of two-thirds is included in the results. The limiting parameter is denoted by an asterisk. Usually the near-field criteria determine the thermal loadings; however, the far-field surface uplift is also a limiting factor in a number of cases, including not only the SF repositories in salt but also 50-year-old HLW in salt and 50-year-old SF in shale. These results indicate that for older wastes, the far-field criteria become more important in determining the repository loading.

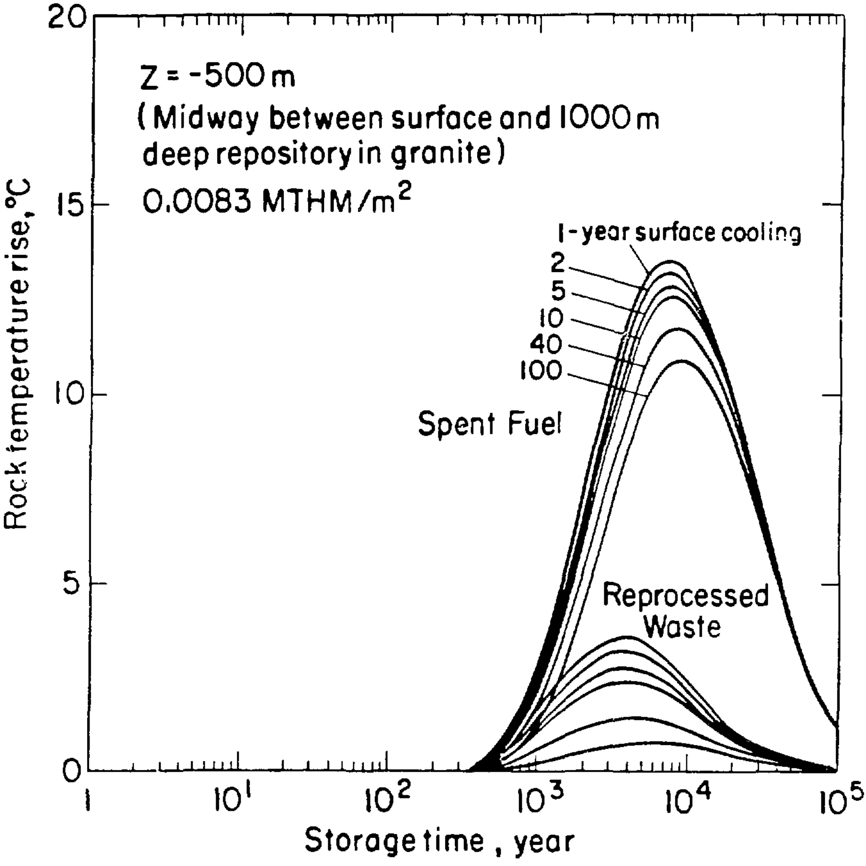

Figure 5.12 Temperature rise at a point midway between the repository and the surface as a function of surface cooling period; constant waste density loading for SF and HLW. [XBL 819-11601]

Table 5.9. EIS Thermal Loadings for Waste Repositories (W/m^2).

Formation	Age of Waste at Emplacement (yr)	Spent Fuel: Near-Field Local Loading	Spent Fuel: Far-Field Average Loading	HLW: Near-Field Local Loading	HLW: Far-Field Average Loading
Salt	5	21	17*	32*	24
	10	12	10*	25*	19
	50	6	5*	17	13*
Granite	5	49*	40	35*	27
	10	32*	26	32*	25
	50	23*	19	30*	23
Basalt	5	49*	40	35*	27
	10	32*	26	32*	25
	50	23*	19	30*	23
Shale	5	30*	24	23*	17
	10	20*	16	20*	15
	50	13	10*	20*	15

(DOE, 1980a)

* Denotes limiting parameters.

The repository waste capacities calculated for these loadings are plotted in Figure 5.13 for SF and in Figure 5.14 for reprocessed HLW for a 8.1 km^2 (2000-acre) repository. The capacity of a salt repository for SF is substantially less than for reprocessed HLW and increases only about 10% from 5 to 50 years. Increases in capacity for the other media range from 30% for SF in shale to 100% for reprocessed wastes in granite.

The sensitive dependence of the repository capacity on fuel cycles and waste age was studied in detail at the International Nuclear Fuel Cycle Evaluation (INFCE) Conference. In the technical appendix of their proceedings, prepared by a joint effort of the Federal Republic of Germany, The Netherlands, and the United States (INFCE, 1980), the waste densities and thermal loadings were calculated for seven fuel cycles and for 10-year-old and 40-year-old wastes in salt. Only the far-field criterion of maximum surface uplift of 1.5 m was used as the limiting parameter for all fuel cycles and waste ages. The near-field and very-near-field temperature profiles were calculated only to assure compliance with near-field and very-near-field temperature criteria.

The results of the INFCE study are shown in Table 5.10. A comparison of the 10- and 40-year-old waste data clearly illustrates the interesting features of allowed areal thermal loadings and emplacement densities that are impacted

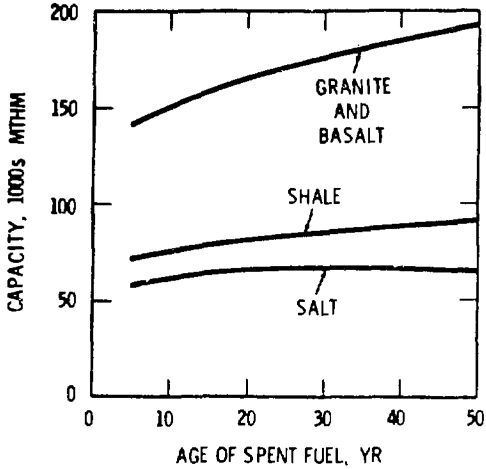

Figure 5.13 Repository capacity as a function of SF age (DOE, 1980a). [XBL 819-11602]

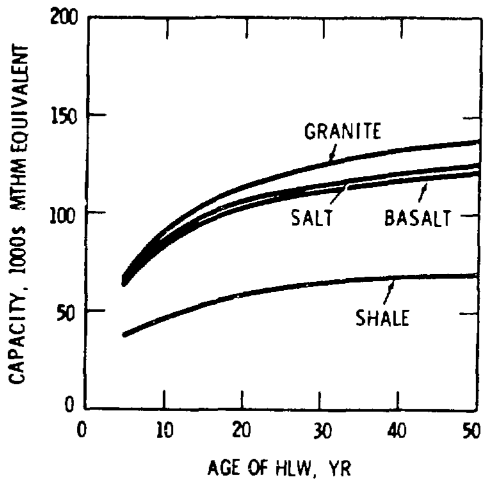

Figure 5.14 Repository capacity as a function of HLW age (DOE, 1980a). [XBL 819-11603]

Table 5.10. INFCE Canister and Repository Thermal Loadings and Waste Emplacement Density.

Fuel Cycle	10-Year-Old Waste: Canister Thermal Loading (kW/canister)	10-Year-Old Waste: Areal Thermal Loading (W/m^2)	10-Year-Old Waste: Emplacement Density ($canister/m^2$)	40-Year-Old Waste: Canister Thermal Loading (kW/canister)	40-Year-Old Waste: Areal Thermal Loading (W/m^2)	40-Year-Old Waste: Emplacement Density ($canister/m^2$)
#1 LWR Spent Fuel	0.56	15	0.027	0.30	9	0.029
#2 LWR U + Pu-Recycle	1.86	37	0.020	0.74	27	0.037
#3 FBR U + Pu-Recycle	1.26	16	0.013	0.64	10	0.016
#4 HWR Spent Fuel	.30	8	0.027	0.17	5	0.027
#5 HWR U + Pu-Recycle	2.29	25	0.011	0.95	14	0.015
#6 HWR U + Th-Recycle	1.56	35	0.022	0.77	35	0.045
#7 HTR U + Th-Recycle	1.09	32	0.029	0.53	29	0.055

(INFCE, 1980)

by surface cooling. Although the allowed thermal loading is reduced by aging the waste an additional 30 years prior to emplacement, the density of emplacement is increased. The increase in the amount of waste (number of canisters in Table 5.10) that can be emplaced per unit area varies from about 1.5% for the SF of heavy water reactors (HWR) to about 103% for the HLW of HWR with both uranium (U) and thorium (Th) recycled. The corresponding enhancement percentages for the light water reactors (LWR), which are the main interest in the United States, are 7% for SF and 83% for HLW with U and Pu recycled.

5.5.2 Surface Cooling, Cumulative Heat, and Far-Field Thermal Effects

The controlling quantity in assessing the far-field thermal effects is the cumulative heat released by the emplaced wastes. Figure 5.15 illustrates the dependence of total heat released by the buried waste in an 8.1-km^2 (2000-acre) repository on the surface cooling period. The waste heat will remain in the rock formation for a long period of time. Although the curves in Figure 5.15 are independent of rock type and characterize only the waste heat source, most of the results on far-field effects presented earlier in this section can be understood from these curves. SF releases more heat over a longer period of time than reprocessed HLW; extension of the surface cooling period removes only a small fraction of the cumulative heat released. On the other hand, the heat from reprocessed HLW is mainly released early. The cumulative heat of reprocessed HLW is much lower than that of SF; the heat removed by surface cooling is a significant fraction of the cumulative heat.

The ratios of cumulative heat released by 40- and 100-year-old wastes to that released by 10-year-old wastes are plotted in Figure 5.16. This figure represents the relative dependence and sensitivity of surface cooling effects over the time range of interest. A lower ratio of cumulative heat energies indicates a greater advantage obtained from longer surface cooling. It is clear from the figure that the effect of surface cooling is more significant for reprocessed wastes than for spent fuel in terms of long-term, far-field effect. The potential advantage of a 100-year cooling period for reprocessed waste is to lower the surface uplift and buoyancy flow to less than half the magnitude of a 10-year cooling period.

European countries, including Belgium, Sweden, the United Kingdom, and West Germany, have also considered longer cooling periods for reprocessed waste (Harmon et al., 1980). The reasons range from near-field concerns over clay stability (Belgium), to backfill stability above 100°C (Sweden), to far-field buoyancy perturbation (United Kingdom). It is of interest to note that an unpublished United Kingdom report quoted by Bredehoeft and Maini (1981) states that "if waste is allowed to cool for 40 to 60 or 70 years, depending on the waste type, the heat would be reduced to the point where buoyancy-induced flow would not be significant."

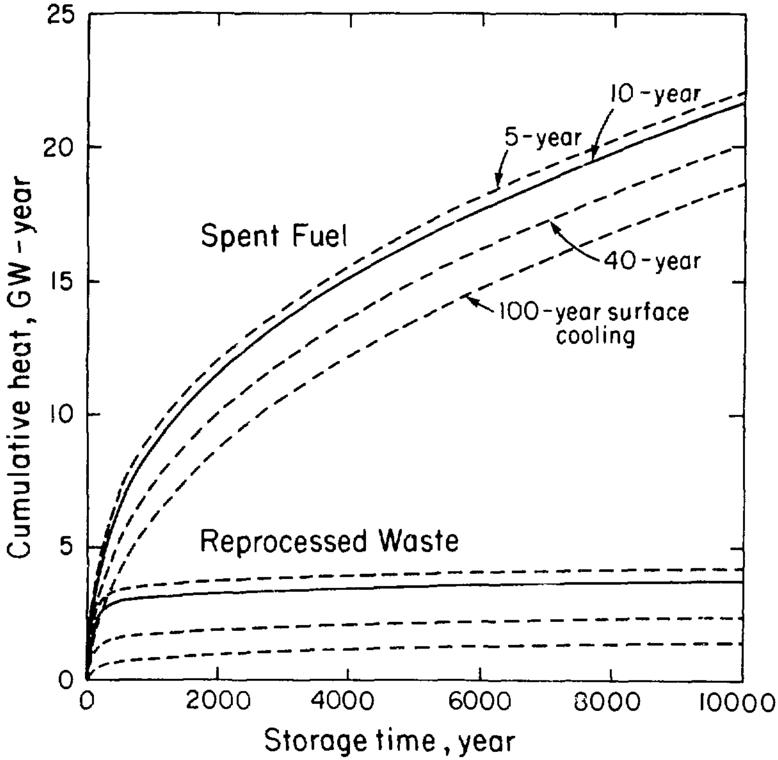

Figure 5.15 Cumulative heat released by SF and HLW as a function of surface cooling period. [XBL 817-3268]

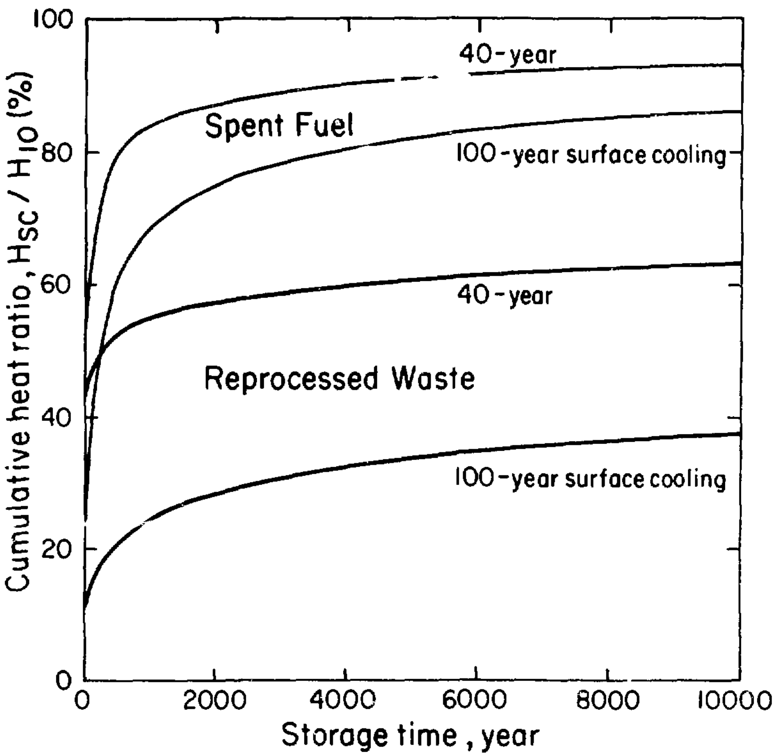

Figure 5.16 Ratio of cumulative heat released by 40-year-old and 100-year-old wastes to that released by 10-year-old wastes. [XBL 817-3269]

5.5.3 The Importance of Surface Cooling Effects

In the evaluation of the thermal effects of waste repositories, waste age is an important parameter contolling waste heat. The heat released into the rocks by the waste is determined by the following three primary parameters: waste age, waste type, and waste loading density. From these parameters the thermal loading density at waste emplacement and the cumulative heat energy can be derived.

Although these derived quantities characterize the thermal effect for scales of operation ranging from the very-near-field to the far-field, and for times ranging from days to thousands of years, the primary waste parameters are more directly related to waste management and economic considerations. The waste loading density especially determines the size of the repository and the excavation costs. To determine the waste heat characteristics, waste age and type must also be specified. The review and evaluation in this report indicates that the effects of waste age (or surface cooling periods) should be carefully studied in repository design and in regulatory evaluation.

6. CONCLUSIONS

Thermal loading is a principal consideration in the design and evaluation of a repository for geologic disposal of nuclear wastes. For a given amount of waste, surface cooling has an important role in determining thermal loading because it can remove a portion of the heat generated by the waste before it is emplaced underground. Most previous studies have assumed uniform loading of 10-year-old wastes, but the concern over thermal effects and the anticipated delay in establishing fully operational repositories require that attention should be given to older wastes.

The effects of surface cooling periods, waste loading, and thermal criteria have been evaluated in this report.

The optimal waste loadings are determined by the thermal criteria. Existing criteria are based on the thermomechanical stability considerations for the waste package and repository structural components, as well as on the allowable surface uplift due to thermal expansion of the surrounding geologic setting. We have extended the waste loading to older wastes using the thermomechanical criteria developed mostly by studies of 10-year-old wastes. The effect of cooling based on far-field criteria has been discussed. The current far-field criteria do not quantitatively address the vertical buoyancy flow from the repository to the surface when applied to the determination of repository loading densities; therefore, the importance of the thermohydrologic effects have also been presented.

The allowable waste densities and thermal loadings for wastes of different ages have been studied on the basis of existing thermal criteria and thermo-hydro-mechanical considerations. Waste loading was found to be sensitive to the waste types. Longer surface cooling periods could be beneficial for reprocessed high-level wastes by minimizing thermal impacts and increasing waste loading density. The far-field effects may be quite important for spent fuel also, but further research is required to evaluate the usefulness of longer surface cooling periods.

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APPENDIX: ANALYTIC FORMULAS

A number of analytic solutions for thermally induced effects are presented in this appendix. These formulas illustrate the functional dependence of temperature rise, thermomechanical deformation, and thermohydrologic perturbation on the thermal loading, rock properties, and repository dimensions.

Simple approximations are made in these solutions so that the controlling parameters in the very-near-field, the near-field, and the far-field can be easily identified.

A.1 VERY-NEAR-FIELD FORMULAS

The maximum temperature rise in the rock on the borehole wall surrounding a radioactive waste canister is approximately given by (Hodgkinson and Bourke, 1978):

$$\Delta T_{rock}^{max} = \frac{q_c(0)}{\pi L K_{rock}} \left[A(L, r_w) - \frac{3L}{8} \left(\frac{8\lambda A(L, r_w)}{\pi L \kappa_{rock}} \right)^{1/3} \right],$$

with

$$A(L, r_w) = \frac{1}{2} \ln \left[\frac{L}{2r_w} + \left(1 + \left(\frac{L}{2r_w} \right)^2 \right)^{1/2} \right], \qquad \text{(A-1)}$$

where

$q_c(0)$ = canister emplacement power, W/canister,
L = canister length, m,
r_w = borehole radius, m,
K_{rock} = thermal conductivity of the rock, W/m°C,
κ_{rock} = thermal diffusivity of the rock, m^2/s,
λ = radioactive decay constant, $\ln(2)/t_{1/2}$, 1/s.

The temperature rise across the cylindrical gap between the canister and the borehole wall may be computed with the steady-state solution

$$\Delta T_{gap} = \frac{q_c(0)}{2\pi L K_{gap}} \ln \left(\frac{r_w}{r_c} \right), \qquad \text{(A-2)}$$

where

K_{gap} = effective thermal conductivity of the gap, W/m°C,
r_c = canister radius, m.

The temperature rise in the waste may be computed using the steady-state solution in an infinitely long cylinder of waste

$$\Delta T_{waste} = \frac{q_c(0)}{4\pi L K_{waste}} , \tag{A-3}$$

where K_{waste} = thermal conductivity of the waste, W/m°C.

These steady-state solutions were used in the GEIS study (Science Applications, Inc., 1978) to validate numerical models.

Equations (A-1) and (A-3) illustrate that the very-near-field temperature rises are proportional to the waste emplacement power and inversely proportional to the thermal conductivities of the various waste components and of the surrounding rock.

A.2 NEAR-FIELD FORMULAS

The temperature rise at the pillar centerline between canister rows can be approximated conservatively by a model with wastes uniformly distributed along the rows at the repository level. The repository temperature in this model (Carslaw and Jaeger, 1959) is given by

$$\Delta T_{\text{repository}} = \frac{1}{2\rho c_{\text{rock}}} \int_0^t \frac{Q_R(t')}{[\pi \kappa_{\text{rock}}(t - t')]^{1/2}} , \qquad \text{(A-4)}$$

where

$Q_R(t')$ = heat power per unit area at time t', W/m^2,
ρc_{rock} = volumetric heat capacity of the rock, or product of the rock density and specific heat, $W/m^3 {}^\circ C$.

The repository temperature for a radioactive decay areal heat source (Beyerlein and Claiborne, 1980; Hodgkinson and Bourke, 1978) is

$$\Delta T_{\text{repository}} = \frac{Q_R(0)}{2K_{\text{rock}}} \left(\frac{\kappa_{\text{rock}}}{\lambda} \right)^{1/2} \text{ImW}[(\lambda t)^{1/2}], \qquad \text{(A-5)}$$

where

$Q_R(0)$ = thermal loading density, W/m^2,
ImW = imaginary part of the error function of complex argument.

The maximum temperature rise is

$$\Delta T^{\max}_{\text{repository}} = 0.305 \frac{Q_R(0)}{K_{\text{rock}}} \left(\frac{\kappa_{\text{rock}}}{\lambda} \right)^{1/2} = 0.305 \frac{Q_R(0)}{(K_{\text{rock}} \rho c_{\text{rock}} \lambda)^{1/2}} . \qquad \text{(A-6)}$$

Equation (A-6) shows that the near-field maximum temperature rise is proportional to the areal thermal loading and inversely proportional to the square root of the product of thermal conductivity, heat capacity, and effective radioactive decay constant.

The temperature rise induces stress and strain changes in the rock medium surrounding the repository. If the rock medium is elastic, the thermally induced stress-strain changes are proportional to ΔT. Thermoelastic analyses were used in the GEIS study (Dames and Moore, 1978a) and in the results given in Section 3.5.2 for hard rocks (granite, basalt, shale). Lomenick's formula, deduced from creep studies in Project Salt Vault (Bradshaw and McClain, 1971), was frequently used in repository designs for salt as a plastic rock medium. In SI units the formula is

$$E = CT^{9.5}\sigma^{3.0}t^{0.3} ,$$

where

E = cumulative strain, m/m,
$T = T_{ambient} + \Delta T$, absolute temperature, K,
σ = average pillar stress, Pa,
t = time, s,
C = 3.4 E-50.

The cumulative strain is therefore a nonlinear function of ΔT.

A.3 FAR-FIELD FORMULAS

The temperature rise around a disk-like repository at any point (r, z) (Carslaw and Jaeger, 1959; Wang et al., 1981) is given by

$$\Delta T(r,z,t) = \frac{1}{\rho c_{rock}} \int_0^t Q_R(t')[V_{-D}(r,z,t-t') - V_{+D}(r,z,t-t')]dt', \tag{A-7}$$

with

$$V_{\mp D}(r,z,t) = \frac{1}{4(\pi\kappa_{rock}^3 t^3)^{1/2}} \int_0^R \exp\left[-\frac{r^2 + r'^2 + (z \pm D)^2}{4\kappa_{rock} t}\right] I_0\left(\frac{rr'}{2\kappa_{rock} t}\right) r'dr',$$

where

$V_{\mp D}$ = instantaneous disk heat source of radius R in the plane $z = \mp D$, 1/m,
r = radial distance from the axis of the repository, m,
I_0 = zeroth-order modified Bessel function of the first kind,
z = vertical coordinate, negative below the ground surface, m,
t = time, s.

The areal heat power function $Q_R(t')$ could be expressed as a series of exponential decay terms. For each decay term, the temperature rise along the z-axis of the repository is

$$\begin{aligned}\Delta T(0,z,t) = & f\{[(z + D)^2/4\kappa_{rock} t]^{1/2},\ t\} \\ & - f\{[(R^2 + (z + D)^2)/4\kappa_{rock} t]^{1/2},\ t\} \\ & + f\{[(z - D)^2/4\kappa_{rock} t]^{1/2},\ t\} \\ & - f\{[(R^2 + (z - D)^2)/4\kappa_{rock} t]^{1/2},\ t\},\end{aligned} \tag{A-8}$$

with

$$f(x,\ t) = \frac{Q_R(0)}{2K_{rock}} \left(\frac{\kappa_{rock}}{\lambda}\right)^{1/2} \mathrm{Im}W[(\lambda t)^{1/2} + ix].$$

The surface uplift due to the thermal expansion of rock between the repository and the surface is given by

$$\Delta Z = \int_{-D}^{0} \left(\frac{1 + \mu}{1 - \mu}\right) \alpha_{rock} \, \Delta T \, dz, \tag{A-9}$$

where α_{rock} = thermal expansivity of the rock, 1/°C, and μ = Poisson's ratio of the rock.

The buoyancy gradient due to the thermal expansion of water for flow from the repository to the surface is approximately given by

$$i_B = \int_{-D}^{0} \alpha_{water} \Delta T \, \frac{dz}{D} , \tag{A-10}$$

where α_{water} = thermal expansivity of water, 1/°C.

Both the surface uplift and the buoyancy gradient are therefore determined by the integrated temperature rise in the rock formation, which is related to the cumulative heat remaining in the rock.

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BIBLIOGRAPHY ON THERMALLY RELATED PROBLEMS IN A NUCLEAR WASTE REPOSITORY

This bibliography is the product of a search of the literature on thermally related problems in nuclear waste storage. Included with each title is the abstract that appeared with the original article. We hope that scientists and engineers interested in these problems will find this compilation useful. As more articles on this subject are brought to our attention, we hope to update the collection. Articles included are arranged according to the following groups:

GROUP B.1: THERMAL EFFECTS

Altenbach, T.J.
Interim report on nuclear waste depository thermal analysis. UCID--17865, Lawrence Livermore Laboratory, University of California, Livermore, CA, 26 p., July 1978

A thermal analysis of a deep geologic depository for spent nuclear fuel is being conducted. The TRUMP finite difference heat transfer code is used to analyze a 3-dimensional model of the depository. The model uses a unit cell consisting of one spent fuel canister buried in salt beneath a ventilated room in the depository. A base case was studied along with several parametric variations. It is concluded that this method is appropriate for analyzing the thermal response of the system, and that the most important parameter in determining the maximum temperatures is the canister heat generation rate. The effects of room ventilation and different depository media are secondary.

Altenbach, T.J.
Three-dimensional thermal analysis of a high-level repository. UCID-17984, Lawrence Livermore Laboratory, University of California, Livermore, CA, 32 p., April 1979.

This report documents the three-dimensional thermal analysis of a high-level waste repository. The analysis used the TRUMP computer code to evaluate the thermal fields for six repository scenarios that studied the effects of room ventilation, room backfill, and repository thermal diffusivity. The results for selected nodes are presented as plots showing the effect of temperature as a function of time.

Altenbach, T.J.; Lowry, W.E.
Advanced three-dimensional thermal modeling of a baseline spent fuel repository. UCID--18660, Lawrence Livermore Laboratory, University of California, Livermore, CA, 50 p., April 1980. (Also in "Heat Transfer in Nuclear Waste Disposal", ASME Proceedings, Vol. 2, p. 43-48, 1980)

A three-dimensional thermal analysis using finite difference techniques was performed to determine the near-field response of a baseline spent fuel repository in a deep geologic salt medium. A baseline design incorporates previous thermal modeling experience and OWI recommendations for areal thermal loading in specifying the waste form properties, package details, and emplacement configuration. The base case in this thermal analysis considers one 10-year old PWR spent fuel assembly emplaced to yield a 36 kW/acre (8.9 W/m^2) loading. A unit cell model in an infinite array is used to simplify the problem and provide upper-bound temperatures. Boundary conditions are imposed which allow simulations to 1000 years. Variations studied include a comparison of ventilated and unventilated storage room conditions, emplacement packages with and without air gaps surrounding the canister, and room cool-down scenarios with ventilation following an unventilated state for retrieval purposes. It was found that at this low-power level, ventilating the emplacement room has an immediate cooling influence on the canister and effectively maintains the emplacment room floor near the temperature of the ventilating air.

The annular gap separating the canister and sleeve causes the peak temperature of the canister surface to rise by 10°F (5.6°C) over that from a no gap case assuming perfect thermal contact. It was also shown that the time required for the emplacement room to cool down to 100°F (38°C) from an unventilated state ranged from 2 weeks to 6 months; when ventilation was initiated after times of 5 years to 50 years, respectively. As the work was performed for the Nuclear Regulatory Commission, these results provide a significant addition to the regulatory data base for spent fuel performance in a geologic repository. Recommendations are made for future directions of thermal analysis efforts, particularly for an expansion of the unit cell concept to treat asymmetrical boundary conditions.

Beyerlein, S.W.; Claiborne, H.C.
Possibility of multiple temperature maxima in geologic repositories for spent fuel from nuclear reactors.
ORNL/TM--7024, Oak Ridge National Laboratory, Oak Ridge, TN,
40 p., January 1980. (Also in "Heat Transfer in Nuclear Waste Disposal", ASME Proceedings, Vol. 2, p. 48-57, 1980)

Heat transfer studies show that two temperature maxima at the disposal horizon could be experienced in CANDU spent fuel repositories--one at about 60 years and another slightly higher one at 13,000 years. Because CANDU spent fuels display a montonically decreasing heat generation rate, it is not immediately obvious why this behavior should occur. This report investigates this behavior, confirms the Canadian results, demonstrates that the double peak phenomenon is due to the presence of the right mixture of short- and long-lived nuclides in the fuel, and concludes that the 13,000 year maximum is largely an artifact of the infinite or very large plane source model. When more realistic repository geometries are used, the second peak disappears for repository sizes less than about 1 km^2. Over the long term, radial and surface heat transfer causes the thermal history of the disposal region to deviate from that predicted by infinite plane (or large finite) source models by reducing the magnitude of the second peak. Beyond a 1000-year time horizon, care should be exercised in modeling spent fuel repositories to include the proper boundary conditions. For the first few centuries after emplacement, however, the infinite source model is consistent with the finite disk source model as well as with arrays of spherical and point sources. The second temperature peak can be avoided by restricting the size of the repository and/or partitioning out the long-lived components of the fuel. When spent fuel from PWRs was examined for multiple temperature maxima, only one peak was found, even for the infinite plane source model.

Boyd, R.D.
Forced-air cooling of a WIPP mine drift.
SAND--78-1122, Sandia Laboratories, Albuquerque, NM,
39 p., July 1978

A one-dimensional model has been developed to predict the time, t, required to cool a waste isolation pilot plant (WIPP) mine drift. Comparisons have been made with a two-dimensional model and are found to be in good agreement for the period between four months to two years. If the mean roughness of the WIPP mine surface is greater than 1/8 inch, the mine floor can be cooled to 120°F in less than one year, depending on the time after waste emplacement at which mine ventilation commences.

Bulmer, B.M.; Lappin, A.R.
Preliminary one-dimensional thermal analysis of waste emplacement in tuffs.
SAND79-1265, Sandia Laboratories, Albuquerque, NM, April 1980

One-dimensional calculations of near-field temperatures resulting from waste emplacement in a multiple-layered tuff stratigraphy are presented. Results indicate a marked sensitivity of peak temperatures to assignment of in-situ fluid pressure, geothermal-heat flux, waste type, and location of waste relative to a specific stratigraphic discontinuity. Under the criterion that allowable initial-power densities are limited by the occurrence of boiling at a distance of 10m from emplaced waste, allowable power densities are calculated to range up to 150 kW/acre or more, depending upon geothermal heat flux and waste type.

Butkovich, T.D; Montan, D.N.
A method for calculating internal radiation and ventilation with the ADINAT heat-flow code.
UCRL-52918, Lawrence Livermore Laboratory, University of California, Livermore, CA, 15 p., April 1980

One objective of the spent fuel test in Climax Stock granite (SFTC) is to correctly model the thermal transport, and the changes in the stress field and accompanying displacements from the application of the thermal loads. We have chosen the ADINA and ADINAT finite element codes to do these calculations. ADINAT is a heat transfer code compatible to the ADINA displacement and stress analysis code. The heat flow problem encountered at SFTC requires a code with conduction, radiation, and ventilation capabilities, which the present version of ADINAT does not have. We have devised a method for calculating internal radiation and ventilation with the ADINAT code. This method effectively reproduces the results from the TRUMP multi-dimensional finite difference code, which correctly models radiative heat transport between drift surface, conductive and convective thermal transport to and through air in the drifts, and mass flow of air in the drifts. The temperature histories for each code in the finite element mesh calculated with ADINAT using this method can be used directly in the ADINA thermal-mechanical calculation.

Callahan, J.L.; Ratigan, J.L.; Russell, J.E.; Fossum, A.F.
Heat transfer analysis of the waste-container sleeve/salt configuration.
ORNL/SUB/4269--7, Office of Waste Isolation, Oak Ridge, TN,
68 p., March 1975

Prior to this investigation, the heat transport considered was only that of straight conduction. The waste container, air gap, and sleeve arrangement was considered to be a single, consistent, time-dependent, heat-generating unit in intimate contact with the salt. The conduction model does not accurately model the heat transfer mechanisms available. Thus radiation and combined radiation and convection must also be considered in the determination of the temperature field. As would be expected, the canister temperatures are higher for the case of radiation across airgap than those that result from conduction when the canister is in intimate contact with the salt. For the radiation case, the canister temperatures rise rapidly to a temperature of approximately 1.140^{o}F and maintain an almost steady state condition for one year whereafter the temperatures slowly decrease. The far field temperatures, near the pillar centerline, are essentially equivalent for all cases. As time proceeds, the far field temperatures of the conduction models are about 15% different.

Carlsson, H. (KBS)
Pilot heater test in the Stripa granite.
LBL-7085, SAC-06, Lawrence Berkeley Laboratory, University of California, Berkeley, CA, 155 p., August 1978

In the Stripa Mine, situated in the central part of Sweden, a pilot heater test has been carried out at 348-m level. The type of rock is a granite with a rather high frequency of fractures. A central main heater with a length of 3 m, a diameter of 30 cm and a total power of 6 kW was placed at the bottom of a 10-m deep borehole. At different radial distances, varying from 0.85 m to 2.95 m from the heater, stress and temperature changes were monitored. Additional measurements of movements along major fractures on the surface and changes of water inflow in boreholes were carried out. The measured temperature distribution compares fairly well with the predicted. A maximum temperature of 333.9^{o}C was measured. The thermal conductivity of the rock mass has been calculated to Lambda = 4.8 W/m^{o}C. The thermally induced stresses in the rock mass do not correspond well with the predicted values. Results of measurements in boreholes of the in situ modulus are found to be about half of the laboratory determinations. Displacements of major fractures on the floor of the test drift are very small.

Chan, T.; Cook, N.G.W.; Tsang, C.F.
Theoretical temperature fields for the Stripa heater project. Vol. 1.
LBL-7082-1/2, SAC-09, Lawrence Berkeley Laboratory, University of California, Berkeley, CA, 65 p., September 1978

The report concerns thermal conduction calculations for the three in-situ heater experiments at Stripa which constitute part of the Swedish-American cooperative program on radioactive waste storage in mined caverns. A semi-analytic solution based on the Green's function method has been developed for an array of arbitrary time-dependent finite line heaters in a semi-infinite medium. This method as well as a three dimensional numerical model using IFD (integrated finite difference) technique has been applied to model the field situations at Stripa. Comparison has demonstrated that the finite line source solution for the rock temperature is in excellent agreement with the numerical model solution as well as with a closed form finite cylinder source solution. It was found that maximum temperature rise in the rock within the two year experiment period will be 178°C for the 3.6 kW full-scale heater experiment, 345°C for the full-scale experiment with a 5 kW central heater and eight 0.72 kW peripheral heaters, and less than 200°C for the time-scaled experiment. The ring of eight peripheral heaters in the second full-scale experiment will provide a nominally uniform temperature rise within its perimeter a few weeks after turn-on. The high temperature zone is localized throughout the duration of all three experiments. Nevertheless, the effect of different spacings on the thermal interaction between adjacent radioactive waste canisters will be demonstrated by the time-scaled experiment. Detailed results are presented in the form of tables, temperature profiles and contour plots. Predicted temperatures have been stored in an on-site computer for real-time comparison with field data.

Chan, T.; Witherspoon, P.A.; Javendel, I.
Heat transfer in underground heating experiments in granite, Stripa, Sweden.
In Heat Transfer In Nuclear Waste Disposal", ASME Proceedings, Vol. 2, pp. 1-8, 1980

Electrical heater experiments have been conducted underground in granite at Stripa, Sweden to investigate the effects of heating associated with nuclear waste storage. Temperature data from these experiments are compared with closed form and finite-element solutions. Good agreement is found between measured temperatures and both types of models, but especially for a nonlinear finite-element heat conduction model incorporating convective boundary conditions, measured nonuniform initial rock temperature distribution, and temperature-dependent thermal conductivity. In situ thermal properties, determined by least-squares regression, are very close to laboratory values. A limited amount of sensitivity analysis is undertaken.

Cheverton, R.D.; Turner, W.D.
Thermal analysis of the national radioactive waste repository: Progress through June 1971. ORNL-4726, Oak Ridge National Laboratory, Oak Ridge, TN, 90 p., December 1971

Solidified high-level fuel reprocessing wastes and alpha-contaminated solid wastes are to be placed in salt formations 1000 ft below the earth's surface at the proposed National Radioactive Waste Repository near Lyons, Kansas. The heat released from these wastes will result in temperature increases throughout the immediate geologic formations. The following several factors influence the permissible temperature rises: (1) thermal instability of the solidified waste; (2) migration of brine; (3) integrity of the mine during operations; (4) integrity of the overlying formations; (5) temperature in freshwater aquifers; (6) temperature of the earth's surface; and (7) temperatures beyond the boundaries of the mine. Preliminary thermal calculations indicated the technical feasibility of the Repository. More recent analyses have considered the Repository in greater detail and have employed updated thermal property and stratigraphic data. The results from these analyses indicate that all of the criteria can be satisfied, and that a heat load of about 130 kW/acre is acceptable for 10-year old high-level waste.

Cheverton, R.D.; Turner, W.D.
Thermal analysis of the national radioactive waste repository: Progress through March 1972. ORNL-4789, Oak Ridge National Laboratory, Oak Ridge, TN, 80 p., September 1972

Thermal studies are continuing in connection with the burial of radioactive wastes in a salt formation similar to that at Lyons, Kansas. Parametric studies pertaining to high-level waste were recently conducted in which room size, waste package array and spacing, waste age, and power per package were variables. The objective of the studies was to aid in determining optimum burial conditions. Room widths of 15, 30, and 50 ft and waste ages of 1 to 100 years at the time of burial were considered. The previously used three-dimensional heat conduction code was revised to include the temperature dependence of the thermal conductivity, and criteria associated with limiting temperatures were modified to achieve a greater degree of consistency for the different room sizes and package arrays.

Results from the analysis show that, independent of waste age, there is an advantage in using the 15-ft room, based on repository mining costs and gross space requirements. In connection with the latter requirements the preference for the smaller room size is restricted to the maximum permissible loading condition.

Cheverton; Turner (continued)

Delaying burial of the waste until about 10 years after the parent fuel is discharged from the reactor also appears to be beneficial.

The calculated maximum permissible loading per gross acre for 10-year-old waste is 6 metric tons of waste nuclides, which corresponds to 158 kW/acre at the time of burial. Considering the lowest expected thermal conductivity and permissible temperature for any of the presently anticipated wastes, and assuming a container diameter of 12 in., the maximum permissible initial (time of burial) power per waste package is 5 kW (independent of waste age). The limiting power for a 6-in.-diam. container is about 3 kW.

A limitation on power per package is also imposed by various waste package handling operations within the repository. Thermal considerations limit the power for a 6-in.-diam. container to 5 kW, independent of waste age, while transporter shielding limits the power to 3.3 kW for 1-year-old waste; however, the permissible level is greater than 5 kW for wastes older than about 3.2 years.

A few calculations were made to determine the effect of varying the waste package arrays within a room. Indications thus far are that there is little difference in permissible loading surface density for one, two, and three rows in a 15-ft room and for two and three rows in a 30-ft room. It is tentatively concluded that the addition of more rows will not introduce significant changes. Thus the parametric analysis applies to very low power levels per package, accommodated by numerous rows, as well as to the higher power levels for which the calculations were made.

Calculations have also been made with regard to temperatures in the high-level repository fringe areas (adjacent rooms, corridors, shafts, etc.), and in connection with the burial of cladding hulls and alpha wastes. Neither of these cases appears to pose particularly difficult thermal problems.

Claiborne, H.C.
Thermal analysis of a fuel cladding repository pilot plant in salt.
ORNL/TM-5221, Oak Ridge National Laboratory, Oak Ridge, TN, 31 p., April 1976

Fuel cladding wastes (hulls) remaining after a chop-leach process used in the recovery of nuclear fuel from spent fuel elements are high level with respect to radiation but rather low level with respect to thermal power. Cost considerations dictate rather large waste canisters with compacted contents; however, such a storage scheme can create thermal problems if the canisters are packed too closely together in a disposal horizon within a geological formation.

The design criteria for a pilot plant for these wastes include retrievability and ready access to the rooms for several years, a condition that restricts the maximum floor temperature to less than 110°F and probably to the order of 100°F. Initial planning calls for waste canisters to be 15 ft long (active length of 13 ft) and made from standard 12-in. steel pipe, with the hulls compacted to near 70% theoretical density. A reasonable arrangement for the canisters is emplacement in 20 in. diameter holes drilled in several rows in the salt comprising the floor of an excavated room. The empty space would be filled with sand, which should facilitate retrievability.

For this study it was assumed that the canisters were filled with PWR fuel hulls, which had a heat generation rate of 0.35 kW per canister in the case of 1-year-old wastes, and that the material loadings were the same for all of the canisters regardless of the age of the waste.

A number of two-dimensional thermal calculations for a unit cell were made to determine the effects of pitch, burial depth, waste age, and canister stacking on the maximum mine floor temperature. It was found that considerably less burial area is required for 10-year-old waste as compared with 1-year-old waste. A concept that utilizes a -ft pitch has been studied from an excavation viewpoint. The results show that the waste must be aged about 7 years in order for the maximum floor temperature not to exceed 100°F when single canisters are buried 15 ft in an array with a 4-ft square pitch. In the case of two canisters stacked vertically with a 5-ft sand-filled separation distance, at least a 4-1/2-ft pitch and an age of 10 years are required for a burial depth of 5 ft.

For selected maximum floor temperatures, the required excavation of salt per waste canister (rooms plus canister holes) varies as some inverse function of waste age, burial depth, and the number of canisters stacked vertically in an array mode. The required excavation per canister is greatly affected by the choice of parameters, particularly the age of the waste. For example, if a 100°F maximum floor temperature is selected, the required room excavation (for the conditions examined) ranges from 200 ft^3 per canister for 0-year old waste contained in two vertically stacked canisters and buried to a depth of 15 ft over the top canister to 650 ft^3 per canister for 1-year old waste in single canisters buried 5 ft deep.

Cox, R.L.
Radiative heat transfer in arrays of parallel cylinders.
ORNL-5239, Oak Ridge National Laboratory, Oak Ridge, TN,
286 p., June 1977

A theoretical and experimental study of radiative heat transfer in arrays of parallel cylinders is presented. Attention is primarily directed toward two geometries common in the industry: square arrays of cylinders on a square pitch and hexagonal arrays of cylinders on an equilateral triangular pitch.

Configuration factors for cylinders on square and equilateral triangular pitches are derived using Hottel's cross-string method. Theoretical equations are presented for configuration factors between rods up to four rows apart for cylinders on triangular spacings and between rods up to three rows apart for cylinders on square spacings.

The usefulness of a formulation of the radiant energy exchange equations in terms of dimensionless variables is demonstrated for the case in which the heat generation rates of the cylinders are known and the temperatures are sought. Each of the major theories for treating radiation exchange within a diffuse-gray enclosure--net radiation, Gebhart's, and Hottel's methods--is examined and compared for utility in handling the steady-state and transient solutions for this case. It is shown that the net radiation method is most convenient for the steady-state problem, while either Gebhart's or Hottel's equations are superior for the transient problem.

Computer programs are presented and described for obtaining both steady-state and unsteady-state solutions for the temperatures of cylinders in hexagonal arrays of cylinders on an equilateral triangular pitch and for square arrays of cylinders on a square pitch. In the particular instance of uniform surface emissivities and uniform heat generation rates in each of the cylinders, steady-state center-rod temperatures are given in terms of dimensionless plots which obviate the necessity of using the computer algorithms.

Experimental measurements were made of the steady-state temperatures in two 217-tube hexagonal arrays having pitch-to-diameter ratios of 1.240 and 1.367 at heat generation rates corresponding to center tube temperatures of 800 and 1000^{o}F. The tests were carried out with the tube bundle in a vacuum to minimize the effects of gaseous conduction and natural convective heat transfer between tubes.

Theoretical calculations based on the assumption of uniform radiosity around the periphery of each tube yielded tube temperatures which were in poor agreement with the experimental observations. Replacing the assumption of uniform radiosity over the entire tube with the assumption of uniform radiosity over 30^{o} segments yielded theoretical temperature profiles across the arrays which differed no more than 7% from the experimental profiles.

Davis, B.W.
Convection and thermal radiation analytical models applicable to a nuclear waste repository room.
UCID--18103, Lawrence Livermore Laboratory, University of California, Livermore, CA, 21 p., January 1979

Time-dependent temperature distributions in a deep geologic nuclear waste repository have a direct impact on the physical integrity of the emplaced canisters and on the design of retrievability options. This report (1) identifies the thermodynamic properties and physical parameters of three convection regimes--forced, natural, and mixed; (2) defines the convection correlations applicable to calculating heat flow in a ventilated (forced-air) and in a nonventilated nuclear waste repository room; and (3) delineates a computer code that (a) computes and compares the floor-to-ceiling heat flow by convection and radiation, and (b) determines the nonlinear equivalent conductivity table for a repository room. (The tables permit the use of the ADINAT code to model surface-to-surface radiation and the TRUMP code to employ two different emissivity properties when modeling radiation exchange between the surface of two different materials.) The analysis shows that thermal radiation dominates heat flow modes in a nuclear waste repository room.

Davis, B.W.
Preliminary assessment of the thermal effects of an annular air space surrounding an emplaced nuclear waste canister.
UCRL--15014, Lawrence Livermore Laboratory, University of California, Livermore, CA, 42 p., April 1979

Modeling results have previously shown that the presence of a large air space (e.g., a repository room) within a nuclear waste repository is expected to cause a waste canister's temperature to remain cooler than it would otherwise be. Results presented herein show that an annular air space surrounding the waste canisters can have similar cooling effects under certain prescribable conditions; for a 16 ft x 1 ft diameter canister containing 650 PWR rods which initially generate a total of 4.61 kW, analysis will show that annular air spaces greater than 11 in. will permit the canister surface to attain peak temperatures lower than that which would result from a zero-gap/perfect thermal contact. It was determined that the peak radial temperature gradient in the salt varies in proportion to the inverse of the drill hole radius. Thermal radiation is shown to be the dominant mode of heat transfer across an annular airspace during the first two years after emplacement. Finally, a methodology is presented which will allow investigators to easily model radiation and convection heat transfer through air spaces by treating the space as a conduction element that possesses non-linear temperature dependent conductivity.

Deane, J.S.; Hollis, A.A.
Practical aspects of heat transfer in a adioactive-waste repository design
AERE-R--9343 Atomic Energy Research Establishment, Harwell, England,
19 p., January 1979

An assessment has been made of the effect on temperature rises of the practical features associated with the disposal of high-level radioactive wastes in a repository constructed within a granite formation. Encapsulation, the use of a backfill material, and reduction in the axial spacing between blocks to 8 m will have little effect on the temperature rises in the granite, and there is clear need for the measurement of granite thermal conductivity at selected sites.

Hamstra, J.; Kevenaar, J.W.A.M.
Temperature calculations on different configurations for disposal of high-level reprocessing waste in a salt dome model.
ECN--42, Stichting Energieonderzdek Centrum Nederland, Petten, Netherlands,
57 p., June 1978

A medium size salt dome is considered as a structure in which a repository can be located for all radioactive wastes to be produced within the scope of a postulated nuclear power program. A dominating design factor for the lay-out of such a waste repository is the temperature distribution in the salt dome resulting from decay heat released from the buried solidified high-level reprocessing waste. Two numerical models are presented for the calculation of both global and local rock salt temperatures. The results of calculations performed with these models are demonstrated to be compatible. Rock salt temperatures related to several types of burial configurations, ranging from two layer repository to deep bore hole concepts varying from 100 to 600 m bore hole depth, can therefore be calculated with one rather simple unit cell model. The results of these calculations indicate that rock salt temperatures can be kept within acceptable limits to realize a respository using standard mining techniques. The temperatures at mine gallery level prove to be a dominating factor in the selection of a repository configuration. More detailed calculations of these temperatures taking into account the loading sequence and the cooling capacity of the mine ventilation are recommended. Finally the apparent advantages of a deep bore hole concept emphasize the need for R and D work with respect to advanced drilling techniques in order to achieve deep dry disposal bore holes that can be realized from a burial mine in the salt dome.

Jeffry, J.A.; Chan, T.; Cook, N.G.W.; Witherspoon, P.A.
Determination of in situ thermal properties of Stripa granite from temperature measurements in the full-scale heater experiments: Method and preliminary results.
LBL--8423, SAC-24, Lawrence Berkeley Laboratory, University of California, Berkeley, CA, 34 p., May 1979

The in situ thermal conductivity and thermal diffusivity of a granite rock mass at the Stripa Mine, Sweden, have been extracted from the first 70 days of temperature data for the 5 kW full-scale heater experiment by means of least-squares fit to a finite-line source solution. Thermal conductivity and thermal diffusivity have been determined to be 3.69 W/(m-^{o}C) and 1.84 x 10^{-6} m^2/S, respectively, at an average rock temperature of 23^{o}C (the average value of the actual temperature data used). These values are only slightly higher than the corresponding laboratory values, i.e., there is no significant size effect in the thermal properties of this rock mass. Since the size and shape of the heater canister used are similar to those considered for nuclear waste canisters and a substantial volume of rock is heated, the thermal properties obtained in this study are representative of in situ rock mass properties under actual nuclear repository operating conditions.

Jenks, G.H.
Maximum acceptable temperatures of wastes and containers during retrievable geologic storage.
Y/OWI/TM--42 Office of Waste Isolation, Union Carbide Corp., Oak Ridge, TN, 7 p., August 1977

Estimates of maximum acceptable temperatures of waste and containers during retrievable geologic storage were needed for use in evaluating and comparing conceptual designs for repositories for wastes from reprocessing and for spent fuel in several different rock types. Estimates of these temperatures and discussions of the bases for the estimates are presented.

Just, R.A.
Heat transfer studies in salt and granite.
ORNL/ENG/TM--14, Oak Ridge National Laboratory, Oak Ridge, TN,
90 p., October 1978

Results are presented of a scoping study on the feasibility of using a multi-layer terminal repository design in both salt and granite formations to store either high-level waste or spent fuel. Calculations have been made to determine te-, -ture profiles within the repository and to provide an estimate of the thermal uplift that can be expected. Near-field models developed to compare temperature profiles in the regions close to the waste canisters indicated that maximum thermal gradients and maximum temperature increases could be significantly reduced by changing from a single to a multi-layer repository design. For both high-level waste and for spent fuel, the maximum temperature increase in the multi-level repositories was reduced to approximately 60 percent of the temperature increase predicted for the single-level repositories at the same areal loading. After the near-field models had verified that maximum thermal gradients and temperature increases could be reduced by using a multilevel repository design, a series of far-field models was developed. The far-field models used to provide qualitative comparisons of the maximum thermal uplift indicate that the thermal uplift is roughly proportional to the energy supplied to the formation. Changing from a single- to a multi-layer repository but keeping the areal loading constant results in increased thermal uplifts.

Kaser, J.D.
Thermal analysis of Hanford defense waste strontium and cesium capsules isolated in basalt.
RHO-LD--78, Rockwell Hanford Operations, Rockwell International Corp., Richland, WA,
33 p., June 1979

Temperatures were calculated for cesium and strontium capsules packaged in steel canisters and isolated in deep basalt formation using the HEATING5 program. For the base case with an initial heat generation rate of 5 kW per canister, the maximum temperature reached at the borehole surface is 760°F, while the canister surface reaches 845°F and the capsule surface reaches 1155°F. A sensitivity analysis showed that the temperature was most affected by changes in capsule power, basalt thermal conductivity, surface emissivity, and borehole diameter. Temperature was much less affected by changes in density and heat capacity of the basalt. Tunnel dimensions, minor changes in canister spacing or pitch had little effect. The discretization error of the numerical calculations is about 30°F.

Krumhansl, J.L.
Preliminary results report: Conasauga near-surface heater experiment.
SAND--79-0745, Sandia Laboratories, Albuquerque, NM, 67 p., June 1979

From November 1977 to August 1978, two near-surface heater experiments were operated in two somewhat different stratigraphic sequences within the Conasauga formation which consist predominantly of shale. Specific phenomena investigated were the thermal and mechanical responses of the formation to an applied heat load, as well as the mineralogical changes induced by heating. Objective was to provide a minimal integrated field and laboratory study that would supply a data base which could be used in planning more expensive and complex vault-type experiments in other localities. The experiments were operated with heater power levels of between 6 and 8 kW for heater mid-plane temperatures of 385°C. The temperature fields within the shale were measured and analysis is in progress. Steady state conditions were achieved within 90 days. Conduction appears to be the principal mechanism of heat transport through the formation. Limited mechanical response measurements consisting of vertical displacement and stress data indicate general agreement with predictions. Posttest data, collection of which await experiment shutdown and cooling of the formation, include the mineralogy of posttest cores, posttest transmissivity measurements and corrosion data on metallurgical samples.

Krumhansl, J.L.
Conasauga near-surface heater experiment. Final report.
SAND--79-1855, Sandia Laboratories, Albuquerque, NM, 98 p., November 1979

The Conasauga experiment was undertaken to begin assessment of the thermomechanical and chemical response of a specific shale to the heat resulting from emplacement of high-level nuclear wastes. Canister size heaters were implanted in Conasauga shale in Tennessee. Instrumentation arrays were placed at various depths in drill holes around each heater. The heaters operated for 8 months and, after the first 4 days, were maintained at 385°C. Emphasis was on characterizing the thermal and mechanical response of the formation. Conduction was the major mode of heat transport; convection was perceptible only at temperatures above the boiling point of water. Despite dehydration of the shale at higher temperatures, in situ thermal conductivity was essentially constant and not a function of temperature. The mechanical response of the formation was a slight overall expansion, apparently resulting in a general decrease in permeability. Metallurgical observations were made, the stability of a borosilicate glass waste form simulant was assessed, and changes in formation mineralogy and groundwater composition were documented. In each of these areas, transient nonequilibrium processes occur that affect material stability and may be important in determining the integrity of a repository. In general, data from the test reflect favorably on the use of shale as a disposal medium for nuclear waste.

Lappin, A.R.; Thomas, R.K.; McVey, D.F.
Eleana near-surface heater experiment final report. SAND80-2137,
Sandia Laboratories, Albuquerque, NM, April 1981

This report summarizes the results of a near-surface heater experiment operated at a depth of 23 m in argillite within the Eleana Formation on the Nevada Test Site (NTS). The test geometrically simulated emplacement of a single canister of High-Level Waste (HLW) and was operated at a power level of 2.5 kW for 21 days, followed by 3.8 kW to 250 days, when the power was turned off. Below 85^{o} to 100^{o}C, there was good agreement between modeled and measured thermal results in the rock and in the emplacement hole, except for transient transport of water in the heater hole. Above 100^{o}C, modeled and measured thermal results increasingly diverged, indicating that the in-situ rock-mass thermal conductivity decreased as a result of dehydration more than expected on the basis of matrix properties. Correlation of thermomechanical modeling and field results suggests that this decrease was caused by strong coupling of thermal and mechanical behavior of the argillite at elevated temperatures.

No hole-wall decrepitation was observed in the experiment; this fact and the correlation of modeled and measured results at lower temperatures indicate that there is no a priori reason to eliminate argillaceous rocks from further consideration as a host rock for nuclear wastes. However, the phenomenological complexities apparent in the test, especially those related to rock-mass dehydration, make it obvious that additional in-situ testing must be done before shales can be adequately charcterized for this purpose.

Lincoln, R.C.; Larson, D.W.; Sisson, C.E.
Estimates of relative areas for the disposal in bedded salt of LWR wastes from alternative fuel cycles.
SAND-77-1816, Sandia Laboratories, Albuquerque, NM, 240 p., January 1978

The relative mine-level areas (land use requirements) which would be required for the disposal of light-water reactor (LWR) radioactive wastes in a hypothetical bedded-salt formation have been estimated. Five waste types from alternative fuel cycles have been considered. The relative thermal response of each of five different site conditions to each waste type has been determined. The fuel cycles considered are the once-through (no recycle), the uranium-only recycle, and the uranium and plutonium recycle. The waste types which were considered include (1) unreprocessed spent reactor fuel, (2) solidified waste derived from reprocessing uranium oxide fuel, (3) plutonium recovered from reprocessing spent reactor fuel and doped with 1.5% of the accompanying waste from reprocessing uranium oxide fuel, (4) waste derived from reprocessing mixed uranium/plutonium oxide fuel in the third recycle, and (5) unreprocessed spent fuel after three recycles of mixed uranium/plutonium oxide fuels. The relative waste-disposal areas were determined from a calculated value of maximum thermal energy (MTE) content of the geologic formations. Results are presented for each geologic site condition in terms of area ratios. Disposal area requirements for each waste type are expressed as ratios relative to the smallest area requirement (for waste type no. 2 above). For the reference geologic site condition, the estimated mine-level disposal area ratios are 4.9 for waste type no. 1, 4.3 for no. 3, 2.6 for no. 4, and 11 for no. 5.

Llewellyn, G.H.
Thermal responses in underground experiments in a dome salt formation.
CONF--770847-6, Oak Ridge National Laboratory, Oak Ridge, TN, 12 p., August 1977

To provide design information for a radwaste repository in dome salt, in-situ experiments with nonradioactive heat sources are planned. Three such experiments using electrical heat sources are scheduled to be carried out in a salt dome. The purpose of these experiments is to acquire rock mechanics data to ascertain the structural deformation due to the thermal load imposed, to study brine migration and corrosion, and to provide thermal data. A data acquisition system is provided with these experiments to monitor temperatures, heat fluxes, stresses, and ground displacement. A thermal analysis was made on models of each of these experiments. The objective of the analysis is to verify the capability of making accurate transient temperature predictions by the use of computer modeling techniques. Another purpose is to measure in-situ thermal conductivity and compare the results with measurements taken from core samples. The HEATING5 computer program was used to predict transient temperatures around the experiments for periods up to 2 years using two-dimensional and three-dimensional heat transfer models. The results of analysis are presented with the associated boundary conditions used in the individual models.

Llewellyn, G.H.
Prediction of temperature increases in a salt repository expected from the storage of spent fuel or high-level waste.
ORNL/ENG/TM--7, Oak Ridge National Laboratory, Oak Ridge, TN,
83 p., April 1978

Comparison in temperature increases incurred from hypothetical storage of 133 mW of 10-year-old spent fuel (SF) or high-level waste (HLW) in underground salt formations have been made using the HEATING5 computer code. The comparisons are based on far-field homogenized models that cover areas of 65 and 25 sq miles for SF and HLW, respectively, and near-field unit-cell models covering respective areas of 610 ft^2 and 400 ft^2. Preliminary comparisons based on heat loads of 150 kW/acre and 3.5 kW/canister indicated near-field temperature increases about 20% higher for the storage of the spent fuel than for the high-level waste. In these comparisons, it was also found that the thermal energy deposited in the salt after 500 years is about twice the energy deposited by the high-level waste. The thermal load in a repository containing 10-year-old spent fuel was thus limited to 60 kW/acre to obtain comparable far-field thermal effects as obtained in a repository containing 10-year-old high-level waste loaded at 150 kW/acre. Detailed far-field and unit-cell comparisons were made between a canister containing high-level waste with an initial heat production rate of 2.1 kW and a canister containing a PWR spent fuel assembly producing 0.55 kW using a three-dimensional unit-cell model, a maximum salt temperature increase of 260°F was calculated for the high-level waste prior to back-filling (5 years after burial), whereas a maximum temperature increase of 110°F was calculated for the spent fuel prior to backfilling (25 years after burial). Comparisons were also made between various configurational models for the high-level waste showing the applicability of each model.

Lowry, W.E.; Davis, B.W.; Cheung, M.
The effects of annular air gaps surrounding an emplaced nuclear waste canister in deep geologic storage. In "Heat Transfer in Nuclear Waste Disposal", ASME Proceedings, Vol. 2, pp. 69-74, 1980

Annular air spaces surrounding an emplaced nuclear waste canister in deep geologic storage will have significant effects on the long-term performance of the waste form. Addressed specifically in this analysis is the influence of a gap on the thermal response of the waste package. Three dimensional numerical modeling predicts temperature effects for a series of parameter variations, including the influence of gap size, surface emissivities, initial thermal power generation rate of the canister, and the presence/absence of a sleeve. Particular emphasis is placed on determining the effects these variables have on the canister surface temperature. We have identified critical gap sizes at which the peak transient temperature occurs when gap widths are varied for a range of power levels. It is also shown that high emissivities for the heat exchanging surfaces are desirable, while that of the canister surface has the greatest influence. Gap effects are more pronounced, and therefore more effort should be devoted to optimal design, in situations where the absolute temperature of the near field medium is high. This occurs for higher power level emplacements and in geomedia with low thermal conductivities. Finally, loosely inserting a sleeve in the borehole effectively creates two gaps and drastically raises the canister peak temperature. It is possible to use these results in the design of an optimum package configuration which will maintain the canister at acceptable temperature levels. A discussion is provided which relates the findings to NRC regulatory considerations.

McVey, D.F., Thomas, R.K. and Lappin, A.R.
Small scale heater tests in argillite of the Eleana Formation at the Nevada Test Site.
SAND79-0344, Sandia Laboratories, Albuquerque, NM, 62 p., November 1979

A series of near-surface heater tests has been run in the Eleana Formation at the Department of Energy's Nevada Test Site, in an effort to evaluate argillaceous rock as a potential emplacement medium for nuclear waste storage. The main test, which employed a full-scale heater with a thermal output approximating that which would be expected from commercial borosilicate waste, was designed to operate for several months. Two smaller scaled tests were run prior to the full-scale test, in order to determine the basic feasibility of operating the full-scale test as planned. This report develops the thermal scaling laws, describes the pretest thermal and thermomechanical analysis conducted for these two tests, and discusses the material properties data used in the analyses. The predictions are compared with qualitative experimental results. In the first test, scaled to a large heater of 3.5 kW power, computed heater temperatures were within 7% of measured values for the entire 96 hour test run. The second test, scaled to a large heater having 5.0 kW power, experienced periodic water in-flow into the heater, which tended to damp the temperature. For the second test, the computed temperatures were within 7% of measured for the first 20 hours. After this time, the water effect became significant and the measured temperatures were 15-20% below those predicted. On the second test, rock surface spallation was noted in the bore hole above the heater, as predicted. The scaled tests indicated that in situ argillite would not undergo major thermostructural failure during the follow-on, 3.5 kW, full-scale test.

Ploumen, P.; Stickman, G.; Winske, P.; Durr, K.; Korthaus, E.; Donath, P.
Studies of the temperature behavior associated with the ultimate storage of high level radioactive waste.
Institut fur Elektrische Anlagen und Energiewirtschaft Achen, W. Germany, Atomwirtsch., Atomtech., v. 24, no. 2, pp. 85-94, February 1979

A West German methodology for storing high level radioactive waste is outlined. Waste produced in fuel reprocessing will be converted into a solid (e.g., borosilicate glass), then permanently stored in a geological salt formation. The heat dissipation associated with the ultimate storage of this waste is examined. Temperature fields are calculated as a function of time and position. Computation techniques and computer programs developed for this purpose are described. Simulation experiments with various geometries and heat outputs at the Asse Mine are also presented. A comparison of test and computed results verifies the accuracy with which the computer programs can predict the temperatures. (In German).

Raines, G.E.; Rickertsen, L.D.; Claiborne, H.C.; McElroy, J.L.; Lynch, R.W.
Development of reference conditions for geologic repositories for nuclear waste in the USA.
Presented at the Material Research Annual Meeting in Boston, MA, Nov. 1980, to be published in Proceedings of the Symposium on "Scientific Basis for Nuclear Waste Management".

This paper summarizes activities to determine interim reference conditions for temperatures, pressure, fluid, chemical, and radiation environments that are expected to exist in commercial and defense high-level nuclear waste and spent fuel repositories in salt, basalt, tuff, granite and shale. These interim conditions are being generated by the Reference Repository Conditions Interface Working Group (RRC-IWG), an ad hoc IWG established by the National Terminal Storage Program's (NWTS) Isolation Interface Control Board (I-ICB).

The reference repository conditions being developed are intended to serve as a guide for: a) scientists conducting material performance tests; b) engineers preparing the design of repositories; c) the technically conservative condition to be used as a basis for DOE license applications; and d) scientists and engineers developing waste forms. Present plans call for the completion of generic reference repository conditions for salt, basalt, tuff, and granite, by December, 1981. Shale has been assigned a lower priority and RRC work on that rock type has been discontinued. However, interim conditions for all five rock types will be published as ONWI reports in the near future.

Ratigan, J.L. (RE/SPEC)
Groundwater movements around a repository. Thermal analyses. KBS-TR--54-02, Kaernbraenslesaekerhet, Stockholm, Sweden, 78 p., September 1977

Conduction baseline thermal calculations have been completed for local near-field and global far-field temperature distributions. The effects of waste age, rock mass thermal conductivity, repository depth, repository ventilation, emplacement sequence, and modeling geometry have been analyzed. These baseline results will be used in further phases of this study to analyze the effects on the rock mechanical situation and subsequent flow permeability perturbations. Additionally, the temperature fields will be utilized to assess thermally induced flow and to quantify the importance of free and forced convective heat transfer and their subsequent effects on the groundwater regime. The temperature rises that have been observed are low in comparison to those arising in the analysis of repositories in other nations. This is mainly due to the waste age and the low GTL being investigated in Sweden. In a qualitative sense, the potential for thermally induced flow appears apparent. However, further study should quantify the magnitude of this flow.

Ratigan, J.; Van Sambeek, L. (RE/SPEC)
Analysis of transient history of underground excavations for radioactive waste isolation.
Y/OWI/SUB--77/22303/P6, 8 p., August 1977.

The constraints and phenomena which must be modeled in realizing a rational prediction of temperature history in a radioactive waste repository are presented. The effects of conductive and radiative heat transfer between the waste package and host rock are presented. Results of numerical investigations are utilized to present specific situations wherein analytical approximations to the waste canister geometry may be utilized. The paper also presents the results of approximations to the mean underground repository temperatures. The reliability of both methods of predicting temperatures is assessed through the comparison of predicted temperatures with measured temperatures from the Project Salt Vault field experiment. The design of experiments for model verification is discussed and a specific heated experiment which has been proposed is presented.

Russell, J.E.
Areal thermal loading recommendations for nuclear waste repositories in salt.
Y/OWI/TM-37, Office of Waste Isolation, Oak Ridge, TN., 43 p., June 1979

The objective of this report is to establish a wider understanding of the background and rationale behind the areal thermal loading recommendations for generic studies, including conceptual designs, of nuclear waste repositories in salt. This objective has been accomplished by (1) defining a reference repository; (2) proposing a set of tentative performance limits for the response of a well-sited generic repository; (3) discussing and documenting the background of the tentative performance limits; and (4) demonstrating that the reference repository can meet the proposed performance limits at the recommended thermal loadings by comparing the results of numerical studies on conservative models within the performance limits.

Throughout the report, it is emphasized that final recommendations for performance limits and areal thermal loading cannot be made before a site is selected. For a given site, the tentative performance limits may have to be modified and additional limits imposed. Such modifications in performance limits may be reflected in different areal thermal loadings than those given herein.

In addition, field, laboratory, and numerical simulation work relevant to nuclear waste repositories in rock salt is continuing and may provide justification for altering the tentative performance limits and/or the thermal loading recommendations.

Within the state of existing technology, a well-sited repository utilizing the appropriate recommended thermal loading will apparently satisfy the tentative performance limits. The final appropriate performance limits are almost certain to be site specific and should be determined prior to final design.

Science Applications, Inc.
Thermal operations conditions in a national waste terminal storage facility.
Y/OWI/SUB--76/49750, Office of Waste Isolation, Oak Ridge, TN, 49 p., 1976

Some of the major technical questions associated with the burial of radioactive high-level wastes in geologic formations are related to the thermal environments generated by the waste and the impact of this dissipated heat on the surrounding environment. The design of a high level waste storage facility must be such that the temperature variations that occur do not adversely affect operating personnel and equipmen.. The objective of this investigation was to assist OWI by determining the thermal environment that would be experienced by personnel and equipment in a waste storage facility in salt. Particular emphas s was placed on determining the maximum floor and air temperatures and without ventilation in the first 30 years after waste emplaceme . The assumed facility design differs somewhat from those previously alyzed and reported. But many of the previous parametric surveys are seful for comparison. In this investigation a number of 2-dimensional 3-dimensional simulations of the heat flow in a repository have een performed on the HEATING5 and TRUMP heat transfer codes. The representative repository constructs used in the simulations are descr ed, as well as the computational models and computer codes. Results of he simulations are presented and discussed. Comparisons are made between e recent results and those from previous analyses. Finally, a summ y of study limitations, comparisons, and conclusions is given.

Science Applications, Inc.
The selection and evaluation of thermal criteria for a geologic waste isolation facility in salt.
Y/OWI/SUB-76/07220, Office of Waste Isolation, Oak Ridge, TN,
92 p., September 1976

Previous design efforts for geologic waste isc tion facilities in bedded salt developed several limiting temperature onditions, or thermal criteria, for use in parametric studies. These c teria, along with other design parameters, must assure that the tem erature variations that occur do not adversely affect operating pers nnel and equipment during normal operations as well as assure conta nment and environmental integrity.

The goals of the present study are to: (1 review past analyses of thermal criteria, (2) determine the factors t t should be considered in defining thermal criteria, (3) suggest ap opriate procedures for determining thermal criteria and (4) suggest additional experimental and computational efforts required to adequ ely determine thermal criteria.

Science Applications, Inc.
Thermal analysis of a ventilated high-level waste repository.
Y/OWI/SUB--76/15527, Office of Waste Isolation, Oak Ridge, TN, 83 p., April 1977

The cooling response of a single ventilated storage room in an unventilated array of rooms is examined. Calculations show that ventilation provides a thermal sink in the heated system inducing temperature gradients in the formation different from the unventilated case. An asymptotic cool-down limit exists for the storage room temperatures: this minimum temperature depends on inlet air temperature, ventilation flow rate, and convective heat transfer coefficient. For inlet air at 75°F and 50,000 CFM and a heat transfer coefficient of 0.8 BTU/H°F ft^2, the limit is about 100°F. A storage room sealed for 5 years will achieve temperatures of approximately 180°F and approximately 4 months would be required in order to cool the storage room floor to a temperature of 120°F with a flow rate of 50,000 CFM at an inlet air temperature of 75°F, assuming a convective heat transfer coefficient of 0.8 BTU/h°F ft^2. Two months would be needed to cool the exhaust air to 120°F. For large air flow rates, the cooling time is independent of the flow rate. Increasing the storage room surface area by 25% over the baseline model depresses the cool-down temperatures by only 4°F and decreases cooling times by 20%. Modifications in canister design or width have virtually no effect on the cooling, but placing the waste deeper beneath the storage rooms and/or using longer canisters can lower the operating temperatures and cooling times. Reducing the canister from 3.5 kW power density for 10-year-old waste (108.5 kW/acre) to 2.0 kW/canister (62 kW/acre) reduces cooling temperatures by more than 20°F and reduces cooling times to a few weeks or less. The cooling times are nearly independent of the conductivity of the geologic formation. The temperature increase in the air brought from the surface down the supply shaft to the storage room level is about 5 to 7°F per 1000 feet. Temperature increases in regions should not be seriously restricted 30 or more feet away.

Science Applications, Inc.
Technical support for GEIS: Radioactive waste isolation in geologic formations. Volume 19. Thermal Analysis.
Y/OWI/TM-36/19, Office of Nuclear Waste Isolation, Oak Ridge, TN, 266 p., April 1978

This volume, Y/OWI/TM-36/19, "Thermal Analyses," is one of a 23-volume series, "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations," which supplements the "Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations," Y/OWI/TM-44. The series provides a more complete technical basis for the preconceptual designs, resource requirements, and environmental source terms associated with isolating commercial LWR wastes in underground repositories in salt, granite, shale and basalt. Wastes are considered from three fuel cycles: uranium and plutonium recycling, no recycling of spent fuel and uranium-only recycling. This volume discusses the thermal impacts of the isolated high level and spent-fuel wastes in geologic formations. A detailed account of the methodologies employed is given as well as selected results of the analyses.

Sisson, C.F.
Predicted temperatures in a bedded-salt repository resulting from burial of DOE high-level nuclear waste canisters.
SAND-78-0924, Sandia Laboratories, Albuquerque, NM,
60 p., February 1978

A 2-D thermal model is defined and used to calculate temperatures within support pillars of a bedded salt repository. The particular configuration considered includes DOE high-level waste canisters (producing 300 W each) buried three abreast below the floor of a 20-foot wide drift. A 3-D "close-in" thermal model is also defined and used to estimate the thermal response of bedded salt near the buried canisters. This model uses boundary conditions from the 2-D "global" model.

Smyth, J.R.; Crowe, B.M.; Halleck, P.M. Reed, A.W.
A preliminary evaluation of the radioactive waste isolation potential of the alluvium-filled valleys of the Great Basin.
LA-7962-MS, Sandia Laboratories, Albuquerque, NM, 22 p., August 1979

The occurrences, geologic features, hydrology. and thermal, mechanical, and mineralogical properties of the alluvium-filled valleys are compared with those of other media within the Great Basin. Computer modeling of heat conduction indicates that heat generated by the radioactive waste can be dissipated through the alluvium in a manner that will not threaten the integrity of the repository, although waste emplacement densities will be lower than for other media available. This investigation has not revealed any failure mechanism by which one can rule out alluvium as a primary waste isolation medium. However, the alluvium appears to rank behind one or more other possible media in all properties examined except, perhaps, in sorption properties. It is therefore recommended that alluvium be considered as a secondary isolation medium unless primary sites in other rock types in the Great Basin are eliminated from consideration on grounds other than those considered here.

Stearns-Roger Engineering Co.; Woodward-Clyde Consultants
Thermal analyses, NWTS repository number 1, conceptual design.
EY-77-C-E5-5367, ONWI LIB 0138, 69 p., March 1978

Knowledge of the changes in thermal environment of a radioactive waste repository is impo_ ant in the design of such a repository. The emplacement of radioactive waste within mined cavities in a salt dome will result in significant increases in the temperature of the salt in and around the repository. These temperature changes will in turn affect the mechanical behavior of the salt mass (e.g., through thermal expansion and creep of the salt), as well as the requirements for ventilation of the mined cavities.

The thermal analyses discussed in this report were begun as an aid in the development of appropriate thermal design bases for the conceptual design of NWTS Repository No. 1. Studies of thermal design criteria for an NWTS repository have been carried out in the past (Cheverton and Turner, 1972). The present work was conducted in order to verify the applicability of previously developed thermal criteria to the characteristics of the Design Basis Salt Dome.

In addition to providing information relating to the selection of appropriate thermal design bases, this study also included consideration of the compatibility between assumed thermal loading and the conceptual design of the mined repository. Specifically, this study has provided temperature distributions and rates of temperature rise within the repository which have been applied in analyses of rock mechanics and ventilation for the repository excavation.

Tammemagi, H.Y.
Preliminary assessment of temperature distributions associated with a radioactive waste vault.
AECL--6308, Atomic Energy Commission of Canada, Ltd. Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, 46 p., 1978.

The temperature distributions of models which simulated radioactive waste vaults were determined, using a finite difference computer code to solve the transient heat conduction equation. Input parameters to the code included thermal properties for granitic rock and heat generation decay data for wastes that would be separated from CANDU fuel if it were recycled. Due to the preliminary nature of the study, only simple models were analyzed. A disc source was utilized to approximate a one-level repository. Various parameters were investigated such as depth of disc, thermal properties of rock, and long-term effects. It was shown that, for a vault at 500 m depth with an initial areal heat flux of 31 W/m^2, a maximum temperature increase of about 80°C occurs at the vault level about 30 years after waste emplacement; maximum increases near the earth's surface occur after about 1000 years and are less than 1°C. Modeling the vault by a number of vertical waste boreholes on one horizontal level, instead of by a disc, with the gross areal heat flux again 31 W/m^2, did not cause serious local temperature increases as long as the initial heat generation rate of each container was less than about 750 W. It was also shown that, by using the vertical dimension available in granitic plutons and constructing either multiple-level vaults or very deep boreholes, initial areal heat fluxes greater than 31 W/m^2 can be utilized without exceeding the 80°C maximum temperature increase anywhere in the vault.

Walsh, E.L.; Bathke, E.A.
Thermal calculations in shale.
Y/OWI/SUB-76/16502; K-76-1080(R), Kaman Sciences Corporation,
Colorado Springs, CO, 36 p., September 1976

Thermal calculations were performed for underground nuclear waste disposal sites assuming shale as the site material. Average thermal properties were used for zero inclined layered shale with waste disposal at a depth of 2000 ft below the surface. Thermal conductivity was assumed constant with temperature and time and no consideration was given for water loss. Vertical conductivity was assumed half of the horizontal conductivity. No allowance was made for formation of brick or ceramic in shale at elevated temperatures. Ambient temperature at a depth of 2000 ft was taken as 93°F. Coolant air was input at 60°F. Time dependent calculations were made for time dependent sources in the assumed geometry within the non-isotropic medium. Thermal loads for times after emplacement of 1/2, 1, 2, 5, 10 and 30 years were calculated. Analysis of ground disposal temperatures resulting from emplacement of a total heat load of 160 kW using steady state calculations indicates: (1) no difference in thermal gradients for 2.25 or 5 kW containers beyond 1 or 2 source diameters, and (2) equilibrium temperature affected areas are at most 2500 ft horizontally and 2000 ft vertically from the source room. Fourier numbers related to the time to attain equilibrium indicate on the order of 1000 years to reach steady state at constant source strength. Thus, steady state temperatures are conservative by at least the source of decay rate in the first century. Time dependent calculations over the first 30 years indicate: (1) maximum temperatures close to the canisters are reached approximately 2 years after the waste canisters are emplaced; (2) significant temperature increases are only within 100-300 ft of the source room; (3) the governing criterion for personnel and equipment operation will be room floor temperature, which reaches 370°F maximum for the unventilated condition; (4) room air temperature can be reduced to reasonable levels with an approximately 2 day air conditioning cycle using 60°F air; (5) continuous room conditioning appears to be prohibitive for the assumed room design since source heat is sucked out of the ground around the room, overloading the conditioning unit.

GROUP B.2: THERMOMECHANICAL EFFECTS

Callahan, G.D.; Gnirk, P.F.
Analytical approximations to the thermoelastic behavior of repository configurations.
Y/OWI/SUB-78-22303/10, RE/SPEC Inc., Rapid City, SD,
35 p., October 13, 1978

The following three sections of this report present three different analytical solutions to approximate the surface uplift of a heated repository. The first solution assumes a temperature distribution which is a function only of the vertical coordinate (z) constant throughout the radius and the lateral strain is assumed to be zero. The second solution assumes a temperature distribution which is a function of the radial coordinate (r) constant throughout the height and assumes no radial displacement at the outer boundary. The third solution is similar to the second solution; the difference being that the outer boundary is assumed to be traction free. The second solution is subsequently derived from the third solution by superposing the elastic solution obtained from imposing a constant radial displacement at the outer boundary of a solid cylinder. The first solution consistently gives the highest vertical displacements due to the lateral constraints.

Chan, T. and Cook, N.G.W.
Calculated thermally induced displacements and stresses for heater experiments at Stripa, Sweden. Linear thermoelastic models using constant material properties.
LBL-7061, SAC-22, Lawrence Berkeley Laboratory, University of California, Berkeley, CA, 125 p., December 1979

Thermally induced displacements and stresses have been calculated by finite element analysis to guide the design, operation, and data interpretation of the in situ heating experiments in a granite formation at Stripa, Sweden. There are two full-scale tests with electrical heater canisters comparable in size and power to those envisaged for reprocessed high level waste canisters and a time-scaled test. To provide a simple theoretical basis for data analysis, linear thermoelasticity was assumed. Constant (temperature-independent) thermal and mechanical rock properties were used in the calculations. These properties were determined by conventional laboratory testing on small intact core specimens recovered from the Stripa test site. Two-dimensional axisymmetric models were used for the full-scale experiments, and three-dimensional models for the time-scaled experiment. Highest compressive axial and tangential stresses are expected at the wall of the heater borehole. For the 3.6 kW full-scale heater experiment, maximum compressive tangential stress was predicted to be below the unconfined compressive strength of Stripa granite, while for the 5 kW experiment, the maximum was approximately equal to the compressive strength before the concentric ring of eight 1 kW peripheral heaters was activated, but would exceed that soon afterwards. Three zones of tensile thermomechanical stresses will occur in each full-scale experiment. One of these, just beneath the floor of the heater drift, persists throughout most of the duration of the experiment, with largest tensile stresses comparable to the in situ stresses as well as to the tensile strength of Stripa granite. Maximum vertical displacements range from a fraction of a millimeter over most of the instrumented area of the time-scaled experiment to a few millimeters in the higher-power full-scale experiment. Radial displacements are typically half or less than vertical displacements. The predicted thermomechanical displacements and stresses have been stored in an on-site computer to facilitate instant graphic comparison with field data as the latter are collected.

Dames and Moore
Technical support for generic environmental impact statement (GEIS): Radioactive waste isolation in geologic formations, vol. 20. Thermomechanical stress analysis and development of thermal loading guidelines.
Y/OWI/TM-36/20; Office of Waste Isolation, Oak Ridge, TN, 23[illegible] p., April 1978

A thermo-mechanical analysis of proposed preconceptual repositories in granite, shale and basalt has been conducted on three different scales: i) Very Near Field (canister scale), ii) Near Field (excavation scale) and iii) Far Field (regional scale). Three numerical methods were used to undertake the thermo-mechanical calculations: the finite element method for thermal stress analysis, the boundary element method [illegible] thermal and thermal stress analysis, and the semi-analytical method [illegible] for thermal and thermal stresses analysis. The properties a[illegible] stratigraphic sections of the generic rock types have been compiled from available literature and used as the input data [illegible]. Because of the [illegible] data on rock mass properties, assumptions where necessary on rock m[illegible] behav[illegible] have been incorporated into the analyses. Very Near Field a[illegible] indicated that within 1/2 year of burial, fracturing of the rock [illegible] to the canister will occur in all rock types for thermal l[illegible] as [illegible] as 1.2 kW/canister. From past experience, it was [illegible] it fr[illegible] would extend about 6 in. into the rock mass, provided a steel [illegible] installed in the storage borehole. Near Field studies of room a[illegible] pillar stability recommend acceptable thermal loads of 190, 120, and [illegible] kW/[illegible] for a 5-year retrievability period in granite, shale, and basalt respectively, for 25-year retrievability, maximum thermal loads are [illegible], 55, and 100 kW/acre, respectively. Far Field analysis shows maximum g[illegible] surface uplifts less than 2 ft and surface temperature rises less than [illegible] at times of 500-7000 years after disposal. Calculated temperature rises at the repository level are 350°F for granite and shale a[illegible] for basalt containing spent fuel; rises of 250°F for granite and [illegible] f[illegible] shale and basalt are calculated for HLW. Preliminary recommend[illegible] are made for further investigations.

Dawson, P.R.; Tillerson, J.R.
Salt motion following nuclear waste disposal.
SAND--77-1226C, Sandia Laboratories, Albuquerque, NM,
23 p., January 1978

The use of thermomechanically coupled secondary creep models to predict deformations resulting from the storage of heat producing radioactive wastes has been presented. The models have been applied to the study of storage alternatives and to the analysis of buoyant movement of heated salt. Thermoelastic models have been used to evaluate the deformations resulting from thermal expansion. The analysis of these problems have been for the purpose of scoping alternatives available to designers and to evaluate the importance of various parameters within the problems. More complete analyses will be performed in the future as ongoing experimental work to provide more accurate constitutive models for the salt behavior.

Dawson, P.R.; Tillerson, J.R.
Nuclear waste canister thermally induced motion.
SAND-78-0566; Sandia Laboratories, Albuquerque, NM, 31 p., June 1978

A thermodynamically-coupled finite element model of viscous flow and heat transfer in salt due to heat-generating radioactive waste canisters has been developed. Temperature-dependent thermal conductivity was included in the analyses, with flow field and temperature distributions coupled through temperature-dependent body forces, temperature-dependent constitutive equations, viscous dissipation, and changes in system geometry. Free expansion of the salt with temperature rises was assumed, leading to the largest density differences and therefore the greatest driving forces for upward salt flow. Separate thermoelastic computations were performed to evaluate the validity of this free expansion assumption. Creeping viscous flow calculations were done using the COUPLEFLO computer code, and the thermoelastic calculations with SANDIA-BMINES program. An axisymmetric region 500 m in radius extending 750 m above and below a canister horizon was analyzed. In the transient creeping flow and heat transfer analyses, the heat source started at 313 K and diminished at a rate based on a 30 year half-life. Steady-state creeping flow and heat transfer were used to predict upper bounds on the magnitude of velocities. For the steady-state analyses, the velocity field and thermal distribution calculations assumed constant thermal output at 313 K. Results of coupled transient analyses using constant salt viscosity (u = 0.5 x 10(E+15) Pa-sec) projected over 10 years, showed the canister sinking, then rising as its heat generates a convective cell. Maximum canister velocity during the 10 year period was 1.5 pm/sec; total displacement was 0.0001 m. Transient analyses with temperature-dependent salt viscosity have been carried out for times up to 150 years. Downward canister velocity was increased due to reduced salt viscosity after heating; after 35 years the velocity became positive, peaking at 0.1 pm/sec upward. After 125 years, convective cell velocity diminished, and the canister sank again. Total displacement over 150 years was 0.0003 m. Steady-state analyses for a number of canister densities, variable and constant salt viscosities, etc. yielded a maximum upward velocity of ~0.5 pm/sec. Displacement of the surface at the repository centerline as predicted by the thermoelastic model was a maximum of 0.15 mm after 30 years.

Dawson, P.R.; Tillerson, J.R.
Comparative evaluations of the thermomechanical responses for three high level waste canister emplacement alternatives.
SAND-77-0388, Sandia Laboratories, Albuquerque, NM,
46 p., December 1979

The structural responses of three room and canister configurations proposed for the underground storage of high level nuclear wastes have been compared. Coupled secondary creep and heat transfer computations indicate that the future retrieval of waste is most readily assured with a design that combines a low extraction ratio (large pillars) with waste emplacement into the floors of each storage room. Thermoelastic computations show minimal room closure in comparison to room closure due to creep deformations.

Gnirk, P.F.; Callahan, G.D.; Pariseau, W.G.; Van Sambeek, L.L.; Wawersik, W.R.
Analysis and evaluation of the rock mechanics aspects of the proposed salt-mine repository concept-III, summary progress report RSI-0012.
ORNL/SUB--3706/7, 65 p., September 1974

Objective is to conduct a comprehensive analysis and evaluation of the rock mechanics aspects of the proposed salt-mine repository concept for the storage of radioactive wastes. Work was done in five areas: experimental quasi-static deformation of salt, experimental creep deformation of salt, transient heat conduction analysis, thermoelasti /plastic analysis of salt loading on waste container sleeve, and heat conduction analysis, thermo/viscoelastic analysis of salt loading on the waste container sleeve. The transient temperature analysis included the room-and-pillar configuration for the Lyons, Kansas, concept, and the New Mexico pilot-plant concept.

Hodgkinson, D.P.
Deep rock disposal of high level radioactive waste: initial assessment of the thermal stress field. AERE R-8999, Atomic Energy Research Establishment, Harwell, England, February 1978

An initial assessment has been made of the far-field thermal stresses arising from fission product heating of the rock in a radioactive waste depository. Analytic solutions to the time dependent stress and temperature distributions are derived for an idealised model of a depository. The solutions show that after about a century from disposal of the waste, nett tensile stresses can occur in the rock overlying the depository. This could lead to cracking and would consequently reduce the effectiveness of the rock as a barrier to migration of the waste. Compressive stresses occur near the centre of the depository, which could lead to collapse of tunnels and therefore to loss of retrievability of the waste.

Hodgkinson, D.P.; Bourke, P.J.
Far field heating effects of a radioactive waste depository in hard rock. Atomic Energy Research Establishment, Harwell, England
Proceedings of OECD, In situ heating experiments in geological formations seminar, Ludvika, Sweden,
pp. 237-248, 13-15 September 1978

Fission product heating of the rock surrounding a depository for high level radioactive waste will result in high temperatures and high thermal gradients over distances of several hundred meters for many centuries. The consequent thermal expansion of the rock leads to stresses which could alter the fracture pattern and therefore the permeability of the rock. These problems are assessed by considering an idealised model of a depository for which analytic solutions to the temperature and stress fields are derived. A related problem is that any water present in the fissures will tend to rise because of its decrease in density on heating. If the water had previously leached away some of the radionuclides in the waste, then this convective transport constitutes a possible leakage path back to the biosphere. For the low permeabilities expected at a depository site, it is possible to linearise the resulting equations and derive analytic solutions for the flow velocities. This procedure has been carried out for the idealised depository model, in order to estimate the magnitude of these effects.

Hodgkinson, D.P.; Bourke, P.J.
Initial asessment of the thermal stresses around a radioactive waste depository in hard rock. Atomic Energy Research Establishment, Harwell, England, "Annals of Nuclear Energy", vol. 7, pp. 541-552, 1980

The disposal of heat emitting radioactive waste into hard rock should result in temperature rises and thermal gradients over distances of several hundred metres for several centuries. The consequent constrained thermal expansion of the rock would induce stresses which have important implications for possible water-borne leakage of radionuclides and for depository design. These problems are assessed by considering a simplified mathematical model for which analytic solutions to the temperature and stress fields are derived.

Hood, M.; Carlsson, H.; Nelson, P.H.
I. Some results from a field investigation of thermo-mechanical loading of a rock mass when heaters are emplaced in the rock. II. The application of field data from heater experiments conducted at Stripa, Sweden, to parameters for repository design.
LBL-9392, SAC-26, Lawrence Berkeley Laboratory, University of California, Berkeley, CA., 39 p., July, 1979

Results are presented of a field experiment to monitor the response of a rock mass to thermo-mechanical loading from electrically heated canisters emplaced in the rock at a depth of 340 m. Measurements made to date of temperature, displacement, and stress fields indicate that heat is transferred through the rock mainly by conduction; discontinuities within the rock mass have a minimal effect on the heat flow. Displacements within the rock from thermal expansion are shown to be much less than those predicted by linear thermoelastic theory. A plausible, though not complete, reason for these reduced displacements is the absorption of the initial rock expansions into discontinuities within the rock mass. Difficulties have been experienced in obtaining reliable stress measurement data using borehole deformation gauges to monitor changes in rock stress. Some data have been obtained and are being analyzed. Rock decrepitation in the heater boreholes is discussed.

Krumhansl, J.L.; Tyler, L.D.
Thermal and mechanical responses in the Conasauga and Eleana formations. SAND-79-0405C, Sandia Laboratories, Albuquerque, NM, presented at 19th Annual ASME Symposium, Geological Disposal of Nuclear Waste, 8 p., March 1979

Two near-surface heater experiments were performed in argillaceous rocks for the purpose of determining the suitability of this rock type for the disposal of heat producing nuclear waste. Site instrumentation included provisions for monitoring both the thermal and mechanical response of the formation. The mechanical behavior of argillaceous rocks was found to be complex and illustrates the necessity of incorporating the dehydration behavior of clays into existing models. The thermal response also reflected the effects of water. Even in the presence of considerable ground water, however, conduction remains the principal method of heat transfer, and computer codes using this assumption give a realistic picture of the in-situ formation behavior.

Osnes, J.D.; Wagner, R.A.; Waldman, H.(RE/SPEC)
Parametric thermoelastic analyses of high-level waste and spent fuel repositories in granite and other non-salt rock types. Y/OWI/Sub--78/22303/12. Office of Waste Isolation, Oak Ridge, TN, 86 p., April 1978

The global thermomechanical behavior of a radioactive waste repository was examined and such parameters as rock type, gross thermal loading, waste type, waste age, repository depth, and in situ stress state were considered. This was accomplished by use of the finite element method and was completed in two phases: (1) a preliminary investigation of the thermal behavior of a repository with respect to host rock type was made using thermal properties typical of a uniform rock mass composed of either granite, limestone, or shale; and (2) a comprehensive thermal and thermoelastic analysis was made using typical thermomechanical properties of granite.

Ratigan, J.L. (RE/SPEC)
Ground water movements around a repository. Rock mechanics analysis.
KBS-TR--54-04, Kaernbraenslesaekerhet, Stockholm, Sweden, 50 p.,
September 1977

The determination and rational assessment of groundwater flow around a repository depends upon the accurate analysis of several interdependent and coupled phenomenological events occuring within the rock mass. In particular, the groundwater flow pathways (joints) are affected by the excavation and thermomechanical stresses developed within the rock mass, and the properties of the groundwater are altered by the temperature perturbations in the rock mass. The objective of this report is to present the results of the rock mechanics analysis for the repository excavation and the thermally-induced loadings. Qualitative analysis of the significance of the rock mechanics results upon the groundwater flow is provided in this report whenever such an analysis can be performed. Non-linear rock mechanics calculations have been completed for the repository storage tunnels and the global repository domain. The rock mass has been assumed to possess orthoganol joint sets or planes of weakness with finite strength characteristics. In the local analyses of the repository storage tunnels the effects of joint orientation and repository ventilation have been examined. The local analyses indicated that storage room support requirements and regions of strength failure are highly dependent upon joint orientation. The addition of storage tunnel ventilation was noted to reduce regions of strength failure, particularly during the 30 year operational phase of the repository. Examination of the local stresses around the storage tunnels indicated the potential for perturbed hydraulic permeabilities. The permeabilities can be expected to be altered to a greater degree by the stresses resulting from excavation than from stresses which are thermally induced. The thermal loading provided by the instantaneous waste emplacement resulted in stress states and displacements quite similar to those provided by the linear waste emplacement sequence.

Ratigan, J.L.; Callahan, G.D. (RE/SPEC)
Evaluation of the predictive capability of the finite element method:
II. Project Salt Vault - Thermo/viscoelastic simulation.
Y/OWI/SUB-78/22303/11; 62 p., July 1978

The finite element method has been and is being extensively utilized in analyses of radioactive repositories. These analyses have included studies of the thermal and thermomechanical response of individual waste packages, storage rooms and pillars and regions which cover several repository depths and several repository diameters. The results of these analyses can be utilized to formulate or modify the emplacement concept, the storage room and pillar dimensions, and the stratigraphic siting of the repository.

Since the finite element method is a member of a class of approximate numerical techniques, it is necessary that the method be validated by comparing computed results to the results of actual laboratory or field situations. This comparison or validation has been utilized in the past in many instances for structures involving man-made materials and for excavations in geological mediums.

However, the validation has not been extensively utilized for comparison to phenomena observed in excavations in geological media involving rock which is heated above its ambient post-mining level. The mechanical response of a room-and-pillar configuration in a radioactive waste repository in salt can be expected to be influenced by the thermal effects. Therefore, a validation of the finite element method involving both excavation and thermal loading is essential.

The purpose of this study is to assess the predictive capability of a two-dimensionnal finite element structural program by performing a simulation of the Project Salt Vault (PSV) experiment and comparing the numerically computed deformations to those which were measured during the PSV experiment. In addition to the prediction of deformations in PSV, an additional study was performed relative to the prediction of the temperature history which was measured during the PSV experiment.

Thomas, R.K.; Lappin, A.R.; Gubbels, M.H.
Three-dimensional thermal and mechanical scoping calculations for underground disposal of nuclear waste in shale. SAND80-2507, Sandia Laboratories, Albuquerque, NM, 1980

Thermal and mechanical scoping calculations were performed to determine the responses of two different shales to the underground disposal of nuclear waste. The geometry under consideration was a three-dimensional model for waste emplacement in a conventional room-and-pillar configuration. The first medium, high-illite shale with no expandable clays, exhibits a positive and nearly linear thermal expansion with increasing temperature. The second medium, eastern Pierre shale with abundant expandable clays, however, will dehydrate and volumetrically contract several percent upon heating to the local boiling point of the pore water. Results of the thermal calculations are presented which show that only the very-near-field rock temperatures within the boiling isotherm are highly sensitive to the rock mass thermal conductivity and, hence, the expandable clay content. The mechanical calculations for the high-illite shale show no evidence of post-excavation instability, and no change in the stress field as a result of waste emplacement. The mechanical response of the eastern Pierre shale, however, is characterized by zones of volumetric contraction and joint opening located within the boiling isotherm and resulting directly from the dehydration shrinkage. This work is part of the Shale Screening Factors Study sponsored by the U.S. Department of Energy through the Office of Nuclear Waste Isolation (ONWI).

Wahi, K.K.; Maxwell, D.E.; Hofmann, R.
Simulation of the thermomechanical response of Project Salt Vault. Final report. Y/OWI/SUB-77/16519/1, 84 p., February 1977

The feasibility of economically and accurately applying Lagrangian explicit finite-difference (EFD) techniques to the analysis of the thermomechanical response of radioactive wastes placed in salt repositories is demonstrated. Three numerical simulations of the Project Salt Vault (PSV) experiment were carried out, using STEALTH 2D, a two-dimensional EFD code. One calculation did not include a model for creep, while the other two calculations used a general model in which creep was included. As expected, when creep was included, it resulted in significantly more pillar shortening and room convergence than when it was not included. The first of the creep simulations (as well as the non-creep simulation) was designed to demonstrate the applicability of the EFD method. The second creep simulation was performed to evaluate the sensitivity of certain numerical parameters, such as zone size and boundary nearness. Numerical data are presented that compare the results of the three simulations to the results of the Project Salt Vault experiment. In the simulations which included creep, the room closure data are in excellent agreement with the shape and magnitude of the experimentally measured floor and roof closures. Temperature histories were also compared at several locations and these data were also in agreement with the experimental values.

Wahi, K.K.; Maxwell, D.E.; Hofmann, R.
Two-dimensional simulation of the thermomechanical response of Project Salt Vault including the excavation sequence. Final report.
Y/OWI/SUB--78/16549/2, 70 p., March 1978

Based on comparisons of the present four-room sequential excavation calculational results with previous two-room simultaneous excavation results and the experimental results, the following may be concluded: 1) the sequence of excavation plays no role in overall deformation response of rooms and pillars, provided that sufficient time (approx. 6 months) lapse exists between the last excavation and the start of the heat source, (2) the assumption of a symmetry plane between rooms 2 and 3 is valid in modeling the Project Salt Vault experiment, 3) in a realistic simulation, one should allow the creep deformations to occur on real time scale even during the period when no thermal source is active (e.g., between standard day 540 and standard day 806). In particular, reference is made to the two-room sensitivity calculation which was started at standard day 806. In that calculation the creep strain rates at day 806, and cumulative strains until day 806 were erroneous. However, the overall thermomechanical response was still in fairly good agreement with the experimental data.

Wahi, K.K.; Maxwell, D.E.; Hofmann, R.
Explicit finite difference simulations of Project Salt Vault.
Presented at 19th Symposium on Rock Mechanics, Lake Tahoe, NV, May 1978

A series of two-dimensional, plane strain simulations of Project Salt Vault (PSV) were computed in order to demonstrate the applicability of the Lagrange explicit finite-difference (EFD) method to the analysis of the detailed stability response of a radioactive waste repository. The PSV field project was chosen for the simulations because it is a well documented experiment for which some materials testing data are available. The PSV experiment was essentially a feasibility study of radioactive waste disposal in an underground salt formation. It included a large-scale experiment performed in an inactive salt mine in Lyons, Kansas, where a new mining level consisting of five rooms was excavated at about 1000 ft depth and approximately 15 feet above an existing level. Heat sources were arranged and activated so that the imposed heating was also essentially symmetric about a vertical plane. The model for salt creep is a generalization of the work performed by Starfield and McClain, and is a general model for a three-dimensional creep response. For the PSV calculations, it relied on the laboratory salt pillar data on Lomerick for its specific constants. The model is stable for discontinuous stress and temperature changes.

GROUP B.3: THERMOHYDROLOGIC EFFECTS

Bourke, P.J.; Hodgkinson, D.P.
Assessment of thermally induced water movement around a radioactive waste depository in hard rock.
Proceedings, Low-flow, low-permeability measurements in -gely impermeable rocks (NEA/IAEA, Paris) Atomic Energy Research Establish t, Harwell, England, p. 221-235, March 1979

An assessment is made of the relative importance of t mally induced and naturally occurring water movement in the vicinity of a har rock depository. The rock is treated as a uniformly permeable medium. Both simple approximations and more comprehensive analysis of the fluid mechanics lead to the conclusion that thermal movement could be the more important for several millenia.

Bourke, P.J.; Robinson, P.C.
Comparison of thermally induced and naturally occurring waterk ie leakages from hard rock depositories for radioactive waste.
Atomic Energy Research Establishment, Harwell, England, "Radioactive Waste Management", vol. 1(4), pp. 365-380, May 1981

The relative importance of thermally induced and naturally occurring flows of water as causes of leakage from hard rock depositories for radioactive wastes is assessed. Separate analyses are presented for involatile, high level waste from reprocessing of fuel and for plutonium contaminated waste from fabrication of fuel. The effects of varying the quantities of wastes, pre-burial storage and the shapes and depths of depositories are considered. It is concluded that for representative values of these variables, thermal flow will remain the major cause of leakage for long times after the burial of both types of wastes.

Eaton, R.R.; Sundberg, W.D.; Larson, D.E.; Sherman, M.P.
Calculated hydrogeologic pressures and temperatures resulting from radioactive waste in the Eleana argillite.
SAND--79-2019C, Sandia Laboratories, Albuquerque, NM., 9 p., November 1979

The SHAFT78 code (multi-dimensional, two fluid phases, porous medium) has been used to begin assessment of the consequences of nuclear waste burial in a 1000 acre repository emplaced in argillite. The repository is assumed to contain spent fuel SF(UO_2) at a loading of 150 kW/acre and to be located at a depth of 600 m. It was found that with perfect backfill, permeability = 1.0 x 10^{-7} Darcys, a maximum fluid pressure in the repository of 770 bars existed at a time of 55 years after burial. Holding all other input variables constant, the maximum fluid pressure in the underground workings never exceeded the local lithostatic pressure when the permeability of the backfill material was increased to 1.0 x 10^{-1} Darcys. The calculated temperature profiles are essentially independent of backfill permeability and porosity indicating that the heat flow is conduction dominated.

Gaffney, E.S.; Nickell, R.E.
Effects of brine migration on waste storage systems. Final report.
UCRL--15102, Pacific Technology, Del Mar, CA, 51 p., May 1979

Processes which can lead to mobilization of brine adjacent to spent fuel or nuclear waste canisters and some of the thermomechanical consequences have been investigated. Velocities as high as 4 x 10^{-7} m/sec are calculated at the salt/canister boundary. As much as 40 liters of pure NaCl brine could accumulate around each canister during a 10-year storage period. Accumulations of bittern brines would probably be less, in the range of 2 to 5 liters. With 0.5% water, NaCl brine accumulation over a 10-year storage cycle around a spent fuel canister producing 0.6 kW of heat is expected to be less than 1 liter for centimeter size inclusions and less than 0.5 liter for millimeter-size inclusions. For bittern brines, about 25 years would be required to accumulate 0.4 liter. The most serious mechanical consequence of brine migration would be the increased mobility of the waste canister due to pressure solution. In pressure solution enhanced deformation, the existence of a thin film of fluid either between grains or between media (such as between a canister and the salt) provides a pathway by which the salt can be redistributed leading to a marked increase in strain rates in wet rock relative to dry rock. In salt, intergranular water will probably form discontinuous layers rather than films so that they would dominate pressure solution. A mathematical model of pressure solution indicates that pressure solution will not lead to appreciable canister motions except possibly in fine grained rocks (less than 10^{-4} m). In fine grained salts, details of the contact surface between the canister and the bed may lead to large pressure solution motions. A numerical model dominated but has a significant convective component.

Hodgkinson, D.P.
A mathematical model for hydrothermal convection around a radioactive waste depository in hard rock. Atomic Energy Research Establishment, Harwell, England, "Annals of Nuclear Energy", vol. 7, pp. 313-334, 1980

A mathematical model of thermally induced water movement in the vicinity of a hard rock depository for radioactive waste is presented and discussed. For the low permeability rocks envisaged for geological disposal the equations describing heat and mass transfer become uncoupled and linear. Analytic solutions to these linearized equations are derived for an idealized spherical model of a depository in a uniformly permeable rock mass. As the hydrogeological conditions to be expected at a disposal site are uncertain, examples of flow paths are presented for a range of different permeabilities, porosities, boundary conditions and regional cross-flows.

Jenks, G.H.
Effects of temperature, temperature gradients, stress, and irradiation on migration of brine inclusions in a salt repository.
ORNL-5526, Oak Ridge National Laboratory, Oak Ridge, TN, 62 p., July 1979

This report reviews and analyzes available experimental and theoretical information on brine migration in bedded salt. The effects of temperature, thermal gradients, stress, irradiation, and pressure in a salt repository are among the factors considered.

The theoretical and experimental (with KCl) results of Anthony and Cline were used to correlate and explain the available data for rates of brine migration at temperatures up to 250°C in naturally occurring crystals of bedded salt from Lyons and Hutchinson, Kansas. It was concluded that the following empirical equation for V/G_s (migration velocity of brine inclusion per unit temperature gradients in the salt) represents the maximum values for this parameter which would result from thermal gradients in the bedded salt crystals in a repository:

$$\log V/G_s = 0.00656T - 0.6036,$$

where T is the temperature of the salt (°C), and V/G_s has the units ($cm^2\ year^{-1}\ {}^oC^{-1}$).

Considerations of the effects of stressing crystals of bedded salt on the migration properties of brine inclusions within the crystals led to the conclusion that the most probable effects are a small fractional increase in the solubility of the salt within the liquid and a concomitant and equal fractional increase in the rate of the thermal gradient-induced migration of the brine. The application of high pressure could reduce the value of the kinetic potential from that prevailing in the absence of the pressure, but this would not affect the maximum rates predicted by the equation shown above.

The presence of stored radiation energy within a salt crystal could affect the rate of brine migration within the crystal if the stored energy causes an increase in the solubility of the salt. However, results obtained in Project Salt Vault suggested that stored radiation energy had little, if any, influence on the rate of brine flow into the emplacement cavities in the salt. No direct information regarding the effect of stored energy on the solubility of salt is available; thus the author recommends that experiments be undertaken to provide such information.

The greatest uncertainty relative to the prediction of rates of migration of brine into a waste emplacement cavity in bedded salt is associated with questions concerning the effects of the grain boundaries (within the aggregates of single crystals which comprise a bedded salt deposit) on brine migration through the deposit. It is likely that the grain boundary trapping will tend to retard brine migration under the conditions expected to prevail with probable repository designs (viz., $G_s \leq 2^oC$ maximum impurities present on grain boundaries, and boundaries compressed by thermal expansion of the salt.)

Jenks, G.H. (continued)

The results of some of the estimates of rates and total amounts of brine inflow to HLW and SURF waste packages emplaced in bedded salt were included to illustrate the inflow volume which might occur in a repository. These estimates, which are based on the results of temperature calculations reported by others, employed the assumptions that (1) the salt contained 0.5 vol % brine inclusions, (2) these inclusions migrated at the maximum rates shown by the equation presented earlier, and (3) grain boundaries had no effect on the migration.

The results of the brine inflow estimates for 10-year-old HLW emplaced at 150 kW/acre indicated inflow rates starting at 0.7 liter/year and totaling 12 liters at 30 years after emplacement. (Temperature calculations did not extend beyond 35 years.)

The results of the estimates for 10-year-old PWR SURF emplaced at 60 kW/acre indicated a constant inflow of 0.035 liter/year for the first 35 years after emplacement. (Temperature calculations did not extend beyond 35 years.)

Lindblom, U.
Groundwater movements around a repository. Phase 1, state of the art and detailed study plan.
KBS-TR-06, Kaernbraenslesaekerhet, Stockholm, Sweden, 113 p., February 1977

The report was prepared as the first phase of a study of the groundwater movements around a repository for spent nuclear fuel in the Precambrian bedrock of Sweden. The objectives of these studies are to provide a state-of-the-art review of groundwater flow in the region of a repository in granitic rock, in order to provide a basis for long term containment assessments and to prepare a detailed study plan for the continuation of the project. The different processes affecting the groundwater situation for containment are given. A state of the art review of the fluid flow, geochemical, heat transfer and rock mechanics processes as they relate to containment is presented. A detailed study plan to provide a comprehensive assessment of the hydrogeological regime around the repository during its lifetime is also presented. The groundwater flow fields will provide a basis for subsequent long term containment studies.

Lindblom, U.E.; Stille, H.; Gnirk, P.F.; Ratigan, J.L.; Charlwood, R.G.; Burgess, A.S.
Groundwater movements around a repository. Final report.
KBS-TR-54-06, Kaernbraenslesaekerhet, Stockholm, Sweden, 127 p., October 1977

The overall goal of this study has been to assess the groundwater flow field in the vicinity of a conceptual high-level radioactive waste repository, situated at a depth of 500 m in the Precambrian bedrock of Sweden. Finite element modelling procedures have been used employing nominal and extrapolated data for initial groundwater conditions and precedent data for material properties. The coupling of thermal, rock mechanics, and groundwater flow effects has been achieved by means of quasi-static techniques. The results of these interelated processes have been analyzed for the following identifiable periods of the repository time frame: (1) pre-construction; (2) construction, but pre-emplacement of the waste; (3) post-emplacement of the waste through the significant portion of the thermal cycle; and (4) long term. Assessment of the results of the analysis efforts lead to the following general conclusions: for the conceptual repository design at 500 m depth with a gross thermal loading of 5.25 W/m^2, the groundwater regime will not be significantly altered by the radiogenic heat dissipation. The long term flow fields will be determined principally by the flow regime prior to construction and can therefore be reliably predicted through establishment of the existing geohydrology.

McCarthy, F.J.; Komarneni, S.; Scheetz, B.E.; White, W.B.
Hydrothermal reactivity of simulated nuclear waste forms and water-catalysed waste-rock interactions.
CONF-781121; Scientific Basis for Nuclear Waste Management,
Vol. 1, F.J. McCarthy (Ed.) The Pennsylvania State University, Materials Research Laboratory, and Department of Geosciences, University Park, PA,
p. 329-340, 1979

A key consideration in the long term safe disposal of nuclear wastes is their stability in the repository environment. If the assemblage of waste phases is not in thermodynamic equilibrium with the mineral assemblage of the host rocks, chemical reactions which are greatly enhanced by the presence of water or brine may occur. Closed-system experiments with three waste forms (all nonradioactive simulations)--spent fuel, a glass, and a crystalline ceramic, and three rock types--basalt, shale, and salt, were done in sealed gold capsules contained in autoclaves or cold-seal pressure vessels, with the pressure at 300 bars, the temperature range 200 to 300^{o}C, and in the presence of excess water so that all phase assemblages included a liquid phase whose composition was determined by the relative solubility of different components. The experiments illustrated three types of hydrothermal waste-rock interactions: reactivity of waste forms in the presence of aqueous solutions (distilled water or NBT-6a bittern brine); reactivity among waste, rock, and water; and potentially important radionuclide fixation reactions. The solution that resulted from the experiments with spent fuel and deionized water contained Cs, Rb, Mo, and U, but no Sr, Ba, La, or Nd. The NBT-6a brine dissolved Cs, Rb, large amounts of Sr, Ba, La, and Nd, and 40 times more U than the deionized water. Less Mo was in solution because it reacted with the Ca in the brine to form powellite (CAMoO4). The results of the experiments with borosilicate glass and deionized water or brine showed that while glass was very reactive in both deionized water and brine, the former converted the glass to a crystalline mineral-like product, while in the latter, major amounts of all the important radionuclide model elements were extracted and remained in solution. In the experiments with the crystalline ceramic, deionized water had very little effect but brine dissolved large percentages of key elements in the ceramic, including Cs and Sr. However, after as much as two months of treatment, several phases, including monazite, were still unaltered, which supports the suggestion that monazite is a potentially ideal host for nuclear waste actinides. Also, the ability of bittern brines to extract from waste forms so much of the heat producing Sr and Cs isotopes, along with substantial U, needs to be given serious consideration in repository selection and engineering design. When basalt was added to the spent fuel-deionized water experiments, most of the Cs was removed from solution. Hence, the availability of oxygen fugacity buffering such as that bound in basalts and many shales appears to be an important attribute of a geological repository. Mechanisms for the fixation of Cs were determined to be formation pollucite and powellite. More attention to optimized synthetic minerals for particular repositories is needed.

McCarthy, G.J.; Scheetz, B.E.; White, S.; Komarneni, S.; Smith, D.K.; Freeborn, W.P.; Barnes, M.W.
Hydrothermal interaction among nuclear wastes, containment, and host rocks in geologic repositories.
RHO-BWI-SA-8-A, Materials Research Laboratory, The Pennsylvania State University, University Park, PA, 3 p., 1979

Radioactive waste from nuclear power plants to be placed under ground in a repository will possess significant amounts of radioactive decay heat during the first few hundred years (the thermal period). Physical and chemical changes analogous to natural geochemical processes can occur in and around the repository during the thermal period which could result in an entirely new object as the "source term" for long time isolation and migration analysis.

If the canister remains sealed during the thermal period it can act as a source of heat which modifies the properties and mineralogy of the surrounding repository rocks, a geochemical process analogous to contact metamorphism. This phenomenon needs to be investigated because it could affect the behavior of the host rocks with regard to migration of long-lived radionuclides away from the immediate repository.

A possible scenario for mobilization of some radionuclides contained in the waste form is dissolution due to the mobilization of indigenous water in the host rocks or intrusion of groundwater into the repository. If this occurs during the thermal period, then the combination of heat and pressure would create a "hydrothermal" environment that could accelerate breaching of the canister and subsequent interactions of the hydrothermal solutions, the waste, the repository rock, and the remains of the canister and any engineered containment structures. It is known that the modest temperatures and pressures expected in the thermal period, up to 400°C and 300 bars pressure, do cause modifications of rocks and minerals in natural low to medium grade metamorphism. It is reasonable then, to expect such modifications in nuclear wastes, especially in metastable forms such as glass.

At the end of the thermal period, an assemblage of rock, waste, and reaction products, larger in volume than the original canister, will constitute the actual waste form subject to low temperature leaching and migration processes over the lifetime of the repository.

The nature and implications of waste-rock interactions is illustrated with results of experimental studies involving four waste forms, basalt and shale rock types, some of the component phases, and fluids typical of both groundwater and brines. Reaction pathways are dictated by the bulk chemical composition of rocks and waste, and by open system variables which may or may not be buffered by the rock, notably pH and Eh.

McCarthy, G.J. (et al) continued

Studies of potential waste-rock interactions can aid in decision making in several ways. First, they may prove to have been no detriment to the effectiveness of waste isolation. For some (and perhaps most) radionuclides, interactions with aluminosilicate host rocks and in reducing environments could result in new thermodynamically stable and low solubility mineral-like phases. This would not be true with salt as a host rock. If interactions are found to result in the release of radionuclides at an undesirable rate, then three options are available:

1. Protect a reactive waste form from hydrothermal conditions by
   (a) allowing the waste to cool on the surface or in a retrievable mode;
   (b) designing a canister to last through the thermal period under the most severe repository conditions.

2. Design an "overpack" that would supply the appropriate chemistry and Eh-pH buffering so that desirable interactions would occur if hydrothermal conditions should arise.

3. Design a waste form which can withstand the most severe hydrothermal conditions. (It is here that the supercalcine-ceramic and multibarrier waste forms show significant promise.)

Ratigan, J.L.; Burgess, A.S.; Skiba, E.L.; Charlw... , R.
Groundwater movements around a repository. Repository domain groundwater flow analyses.
KBS-TR--54-05, Kaerneraenslesaekerhet, Stockholm, Sweden, 127 p., September 1977

The perturbations of the in situ hydraulic permeability caused by (a) stresses resulting from repository storage tunnel excavation and (b) the thermomechanical stresses resulting from the radiogenic heat have been evaluated. Changes in the permeability due to the temperature dependence of the viscosity have also been studied. It was found that the most significant perturbation in the hydraulic permeabilities was caused by the excavation of the storage tunnels. The results of a study of groundwater inflow to the repository, with particular reference to the post-decommissioning period is presented. The objective was to determine the time for the repository to become backflooded (inflow period). For this study, the baseline layout has been used. This consists of rooms approximately 3.5 m diameter, about 1 km in length and spaced at 25 m center to center. The repository has been assumed to be located at a depth of 500 m below ground surface. As part of the field studies for KBS, measurements of inflow rates will be performed at Stripa Mine. It is expected that a regional groundwater gradient of 2×10^{-3} will exist which will cause the convection flows to be swept almost horizontally, indicating that the most likely point of exit from the host rock is into a singular feature at depth and not up to the surface above the repository.

Schimmel, W.P., Jr.; Hickox, C.E.
Application of thermal conduction models to deepsea disposal of radioactive wastes.
SAND77-0752, Sandia Laboratories, Albuquerque, NM, 42 p., March 1978

Thermal problems associated with the emplacement of radioactive wastes in the deepsea sedimentary layer have been studied. In particular, the nature of the temperature field surrounding and the interstitial water velocity arising from a buried cask have been examined. Worst case estimates indicate that the velocity will be extremely weak and thus not likely to provide a primary transport mechanism for the radioactive material. This statement will, of course, only apply for moderately low levels of heat generation by the decaying radionuclides. Because of the low interstitial water velocity, thermal conduction models can be used to predict the temperature field in the surrounding sediments as well as the cask surface temperature. This is equivalent to "decoupling" the energy and momentum conservation relationships thus simplifying the solution of the temperature field.

The present work considers in some detail the temperature field surrounding a vertical circular "cylinder" located a distance below a horizontal, isothermal, plane surface. Actually, the isotherm corresponding to the cask surface is an ellipsoid of revolution but the error will be small for large values of the length to diameter ratio. The resulting expression can be used to estimate temperature of the cask surface for material degradation studies and the effect of temperature upon the ion transport process in the sediments.

Wang, J.S.Y.; Tsang, C.F.; Cook, N.G.W.; Witherspoon, P.A.
A study of regional temperature and thermohydrological effects of an underground repository for nuclear wastes in hard rock.
LBL-8271, Lawrence Berkeley Laboratory, University of California, Berkeley, CA, October 1979

Heat released by the radioactive decay of nuclear wastes in an underground repository causes a long-term thermal disturbance in the surrounding rock mass. The nature of this disturbance for a planar repository 3000 m in diameter at a depth of 500 m below surface is investigated for various waste forms. The effects of changes in the density and viscosity of groundwater caused by the temperature changes on the flow through a simple model of a vertical fracture connected to a horizontal fracture in a rock mass are evaluated. It is concluded that different waste forms and time periods before burial have significant effects on the thermal disturbance and that buoyant groundwater flow is a function of both the vertical and horizontal fracture transmissivities, as well as the changes in temperature. Loaded initially with a power density of 10 W/m^2 of spent fuel assemblies 10 years after discharge from a reactor, the maximum increase in temperature of the repository in granite is about 50^oC and the epicentral thermal gradient about $70^oC/km$.

Westsik, J.H, Jr.; Turcotte, R.P.
Hydrothermal reactions of nuclear waste solids, a preliminary study. PNL-2759, Pacific Northwest Laboratory, Richland, WA, 44 p., September 1978

A simulated high-leel waste glass, Supercalcine, and some common ceramic and metallic solids were exposed to hydrothermal conditions at 350°C and 250°C for time periods ranging from three days to three weeks.[a] Most of the experiments were done in salt brine, but the glass study did include deionized water tests so that the influences of salt could be better understood.

Under the extreme hydrothermal conditions of these tests, all of the materials examined underwent measurable changes. The glass is converted to a mixture of crystalline phases, depending upon conditions, giving $NaFeSi_2O_6$ as the primary alteration product. The rate of alteration is higher in deionized water than in salt brine; however, under equivalent test conditions, 66% of the cesium originally in the glass is released to the salt brine, while only 6% is released to deionized water. Rubidium and molybdenum are the only other fission product elements significantly leached from the glass. Evidence is presented which shows that sintered Supercalcine undergoes chemical changes in salt brine that are qualitatively similar to those experienced by glass samples. High concentrations of cesium enter the aqueous phase, and strontium and molybdenum are mobilized.

Scouting tests were made with a variety of materials including commercial glasses, granite, UO_2. Al_2O_3, steel, and waste glasses. Weight losses under hydrothermal conditions are in a relatively narrow band, with glass and ceramic materials showing 3 to 20 times greater weight losses than 304L stainless steel in the 250°C test used.

The conclusion from these studies is that virtually all solid materials show hydrothermal reactivity at temperatures between 250°C and 350°C, and that these extreme conditions are not desirable. Further work is needed to establish kinetic parameters for the hydrothermal reactions.

(a) These conditions were selected to accelerate the hydrothermal reaction; actual temperature during waste storage, even for a high-activity waste, should be less than 200°C if water is present during the first 100-yr time period following closure of the repository. At 1000 yr the temperature will be less than 10°C for both wet and dry storage.

GROUP B.4: ROCK PROPERTIES

Board, M.P.
Rock mechanics methods and in situ heater tests for design of a nuclear waste repository in basalt.
RHO-BWI-LD-2, Rockwell International, 53 p, June 1978

This paper discusses how in situ data from the Near-Surface Test Facility will be integrated into the overall Waste Isolation Program. It also discusses the rock mechanics methods used for the design of a repository in hard jointed rock and places the Near-Surface Test Facility heater tests into the context of the total picture of the repository design.

The following discussion deals primarily with the application of numerical models to the design of a waste repository. First, the various types of models currently available are discussed with reference to design in basalt. Next, the breakdown of the problem of repository design is given. It is seen that the most efficient method for analyzing repository design is to break the problem down into several problems which are based on physical scale. These include the area directly surrounding a single waste canister (the very near field), the area including many canisters and canister emplacement rooms (the near field), and the area including the entire repository and the rock mass to the free surface (the far field). The methods by which numerical models are used for design are discussed. Flow charts are used to show the basic input data required, the calculational processes used, and the preliminary criteria for judgment of suitable repository performance. It is seen through the preceding discussion that the ultimate design of the allowable gross thermal loading density, and, thus, the layout of the underground workings is highly dependent upon the rock mass properties supplied as base line input data to the numerical models. Of the many input properties required, the thermal conductivity, the thermal expansion coefficient, and elastic moduli of the rock mass have, perhaps, the greatest effect on the calculation of induced temperatures, stresses, and displacements and, thus, repository design. To ensure that the design continues with confidence, actual field (in situ) values of input data must be obtained. Finally, the role of the Near-Surface Test Facility in situ testing in obtaining these basic required data is discussed.

Christensen, A.B.
Physical properties and heat transfer characteristics of materials for krypton-85 storage.
ICP-1128, Allied Chemical Corporation, Idaho Chemical Programs, Operations Office, Idaho, 46 p., September 1977

Collection and storage of about 85% of the krypton-85 produced by nuclear fuel reprocessing is required by Federal regulations for fuel irradiated after 1982. Krypton can be stored as a gas in pressurized cylinders or immobilized either in granular materials, such as zeolites, or in monolithic amorphous or crystalline metals prior to storage. The safety of the storage system depends on the temperatures resulting from the radioactive decay heat generation. For example, a temperature increase in a ^{85}Kr-pressurized cylinder not only increases the gas pressure, but may reduce the wall strength, resulting in possible cylinder failure and release of radioactive krypton. This report compiles the necessary physical properties of krypton and of potential krypton-85 storage materials which are required to evaluate the maximum temperatures developed during storage.

Krypton volumetric data in the pressure and temperature ranges of 0-200 MPa and 273-423K, respectively, were fitted to the Redlich-Kwong (R-K) equation of state. Krypton thermal conductivity data were correlated in the 0.1-95 MPa and 273-608 K ranges. Heat generation and decay rates were given as a function of time.

Granular thermal conductivities depend on the thermal conductivities of the solid and of its surrounding gas, and on the void fraction. Granular thermal conductivities increase with temperature and pressure, but decrease with void fraction; a correlation fitted 76% of the data with an error of $\pm$ 30%. Thermal conductivity, density, void fraction, mean particle diameter, and sorbed-water concentration at room temperature and pressure are tabulated for zeolites; thermal conductivities of silicate, boro-silicate and borate glasses, and of aluminum, nickel, copper, and iron are given. Since no thermal conductivity data exist for amorphous metals, values intermediate to glass and metal were assumed.

Maximum temperatures were calculated (assuming natural convection cooling in air) for a single horizontal cylinder containing krypton-85 fixed in solids or as a gas, and for storage cells containing 104 cylinders. Cylinders containing krypton-85 as a pressurized gas or in crystalline metals yielded the lowest maximum storage temperatures for a given radius and loading. Void fraction changes in granular materials loaded with krypton-85 weakly influenced the maximum temperature. The effects of natural convection cooling of a cylinder in air and water were compared.

Duvall, W.I.; Miller, R.J.; Wang, F.D.
Preliminary report on physical and thermal properties of basalt. Drill hole DC-10, Pomona Flow - Gable Mountain.
RHO-BWI-C-11, Earth Excavation Institute, Colorado School of Mines, Golden, CO, 46 p., May 1978

This report is submitted as a summary of the mechanical and thermal tests on the first series of basalt core received from the Pomona Flow, Drill Hole DC-10 as part of the Colorado School of Mines subcontract from Rockwell Hanford Operations, SA-907. A discussion of the testing procedures and test resutls, with recommendations for future testing, is also included.

Due to the highly fractured and jointed nature of the first series of basalt cores, it was often impossible to avoid selecting a sample in which a discontinuity might affect the test results. Therefore, each sample was characterized for structural features and if such features has, or might have, influenced the test results, it was noted on the control sheet. In some cases, nondestructive tests were repeated, or additional tests performed, to help define test limitations, repeatability and inter-sample variations.

In the following sections, test results will be presented, implications of these tests discussed, and recommendations made based on the test results and on our best engineering judgment. Tests conducted are summarized below:

TEST	TYPE
A	Uniaxial Compressive Strength
B	Brazilian Tensile Strength
C	Triaxial Compressive Strength at Confining Pressures of 500, 1,000 and 2,000 pounds per square inch
D	Modulus of Rupture - Flexural Strength
E	Static Elastic Properties - Young's Modulus Poisson's Ratio Cohesion Internal Angle of Friction
F	Dynamic Wave Velocities Both compression and shear waves in the axial and diametral directions

Excavation Engineering and Earth Mechanics Institute
Final report for fiscal year 1978 on the physical and thermal properties of basalt cores.
RHO-BWI-C-38, Colorado School of Mines, Golden, CO, 78 p., December 1978

As part of the Colorado School of Mines' subcontract from Rockwell Hanford Operations, SA-907, this report is submitted as a summary of the second series of tests performed under that subcontract.

Due to the highly fractured and jointed nature of much of the core, it was difficult to avoid selecting a sample in which a discontinuity might affect the test results. Therefore, each sample was characterized for structural features and, if such features had or might have influenced the test results, it was noted on the control sheet. In some cases, non-destructive tests were repeated, or additional tests performed, to help define test limitations, repeatability, and inter-sample variations.

In the following sections, test results will be presented, implications of these tests discussed, and recommendations made based on the test results and on our best engineering judgment.

Samples used in these tests came from drill holes DC-11, DH-4, DH-5, DC-2, and DDH-3 within the Hanford Site.

Hansen, F.D.; Mellegard, K.D. (RE/SPEC)
Creep behavior of bedded salt from southeastern New Mexico at elevated temperature.
SAND79-70030, Sandia Laboratories, Albuquerque, NM, 126 p., November 1977

This report presents the results of a series of triaxial creep experiments conducted on bedded salt specimens from ERDA Hole 9 in southeastern New Mexico. The salt core and matrix of test conditions were provided by Dr. Wolfgang R. Wawersik, Sandia Laboratories. The purpose of the experiments was to measure creep response of salt at temperatures of 24, 70 and 100°C under confinement pressures of 0, 1500, 2000, 2500, and 3000 psi and differential axial stress levels of 1500, 3000, 4500 and 6000 psi. Test durations ranged from 15 minutes to 500 hours.

The specimens, obtained by recoring four-inch diameter cores in the axial direction, were nominally two inches in diameter and four inches in length. The crystal size ranged from very small to one-half inch diameter; the specimens contained various amounts of clay impurities. A total of 19 specimens were prepared of which 14 were tested.

The collected data included axial and lateral strain, axial and confinement stresses, time and temperature. Periodically, axial stress was adjusted to account for specimen strain in order to maintain a constant differential stress. Frequency of the stress correction was dependent on the rate of deformation; two or more corrections in a 24 hour period were typical. Data were automatically recorded with a printer, manually recoded from the print-out to punched cards and reduced by means of a computer. A preponderance of the data was collected in the transient creep regime. In some tests specimen rupture occurred, while in others an accelerating creep rate brought the specimen in contact with the pressure vessel wall. Also, a considerable amount of data was collected during stress application to creep stress level.

Lappin, A.R.
Preliminary thermal expansion screening data for tuffs.
SAND78-1147, Sandia Laboratories, Albuquerque, NM, 44 p., March 1980

A major variable in evaluating the potential of silicic tuffs for use in geologic disposal of heat-producing nuclear wastes is thermal expansion. Results of ambient-pressure linear expansion measurements on a group of tuffs that vary greatly in porosity and mineralogy are presented here. Thermal expansion of devitrified welded tuffs is generally linear with increasing temperature and independent of both porosity and heating rate. Mineralogic factors affecting behavior of these tuffs are limited to the presence or absence of cristobalite and altered biotite. The presence of cristobalite results in markedly nonlinear expansion about 200°C. If biotite in biotite-bearing rocks alters even slightly to expandable clays, the behavior of these tuffs near the boiling point of water can be dominated by contraction of the expandable phase. Expansion of both high- and low-porosity tuffs containing hydrated silicic glass and/or expandable clays is complex. The behavior of these rocks appears to be completely dominated by dehydration of hydrous phases and, hence, should be critically dependent on fluid pressure. Valid extrapolation of the ambient-pressure results presented here to depths of interest for construction of a nuclear-waste repository will depend on a good understanding of the interaction of dehydration rates and fluid pressures, and of the effects of both micro- and macrofractures on the response of tuff masses.

Lappin, A.R.; Olsson, W.A.
Material properties of Eleana argillite: extrapolation to other argillaceous rocks, and implications for waste management.
ALO-0789-T14, Sandia Laboratories, Albuquerque, NM, 15 p., October 1979

Results of a near-surface heater test in the Eleana argillite suggest the possibility that the high-temperature (> 100°C) thermomechanical response of argillite to waste emplacement may be dominated by behavior of expandable clays. Enough expandable clay is probably present in most argillaceous rocks to cause a similar response. In-situ thermal conductivities may be markedly reduced by even small amounts of clay contraction which results in opening of pre-existing joints. A simple model predicts that such behavior may continue to operate to considerable depths, though several factors affecting determination of this depth remain poorly defined at present.

Martinez-Baez, L.F.; Amick, C.H.
Thermal properties of Gable Mountain basalt cores - Hanford Nuclear Reservation.
LBL-7038, Lawrence Berkeley Laboratory and Petroleum Engineering Laboratory, University of California, Berkeley, CA, 7 p., March 197

This report presents the results and the general methodology of a series of laboratory experiments and calculations used to obtain the thermal properties of a group of basalt cores from the Gable Mountain area in which a model waste repository is proposed.

Thermal properties and its behavior in a range of temperature of 50 to 200°C are evaluated and presented. Thermal conductivities were measured at two temperature levels and one uniaxial stress. Specific heats were calculated at different temperatures in the range from oxide analysis data provided or the samples. Bulk densities were measured at room temperatures and estimated at high temperatures using available thermal expansion data. Finally, thermal diffusivities were calculated from the obtained data.

McKinstry, H.A.
Thermal effects in shales: Measurements and modeling.
Y/OWI/SUB-77/14268; CONF-77049; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA. Pennsylvania State University, Materials Research Laboratory, University Park, PA, 91 p., August 1977

This research concerns thermal and physical measurements and theoretical modeling relevant to the storage of radioactive wastes in shale. Reference thermal conductivity measurements are made at atmospheric pressure in a commercial apparatus; equipment for permeability measurements has been developed. Thermal properties of shales are being determined as a function of temperature and pressure. In one test a 15 mm disk of sample is measured by a steady state technique using a reference material (initially single crystal quartz) to measure the heat flow within the system. The sample is sandwiched between two disks of reference material and the thermal conductivity of the sample at steady state is calculated. A second test determines the effect of temperatures on a larger sample of water-saturated shale (or siltstone). A cylindrical sample 25 cm diameter x 20 cm, is heated electrically at the center while contained in a pressure vessel that maintains a fixed water pressure around it. The temperature is monitored by a microcomputer system connected to 16 thermocouples to record variation in temperature distribution with time. The microcomputer system as currently devised can also control the energy delivered to the heating element. The vapor pulse should thus be measured without heating the outer wall of the pressure vessel. If, as hypothesized, the heat drives the water away from the internal heater the temperature distribution should reflect a dramatic temperature pulse passing through the sample. By using thermal conductivity data from the small sample experiments, permeability measurements, and the results of a finite difference computer model of the heat transfer, the behavior of the sample should be properly interpreted.

Morgan, M.T.
Thermal conductivity of rock salt from Louisiana salt domes.
ORNL/TM-6809, Oak Ridge National Laboratory, Oak Ridge, TN,
21 p., June 1979

The thermal conductivities of dome salt samples from the Avery Island and Jefferson Island Mines in southern Louisiana were determined as part of an evaluation of these formations for use as possible waste repositories. The thermal conductivities ranged from 4 W/mK at 70°C to 2 W/mK at 300°C. The average thermal conductivity of the dome salt at 100°C was about 25% less than that of a single-crystal salt at 100°C. The thermal conductivity was found to be related to the density and fragility of the salt.

The thermal conductivities were measured using a comparative technique in which the sample was sandwiched between standard references of Pyroceram 9606. The precision of the method is within ±10% and the overall error is within ±20%. Considerable attention was given to specimen preparation. Finished surfaces were parallel within 0.005 in. and flat to within 0.0005 in. Chromel-Alumel thermocouples were installed in grooves on interface surfaces to measure temperatures.

Morgan, M.T.; West, G.A.
Thermal conductivity of the rocks in the Bureau of Mines standard rock suite.
ORNL/TM--7052, Oak Ridge National Laboratory, Oak Ridge, TN, 60 p., January 1980

Thermal conductivities of eight rocks from the Bureau of Mines standard rock suite were measured in air over the temperature range 373 to 533°K (100 to 260°C). The thermal conductivities of these rocks were measured to furnish standards for future comparisons with host rock from prospective nuclear waste repository sites. The thermal conductivity at a given temperature decreased by as much as 9% after a specimen had been heated to the maximum temperature (533°K), but additional heating cycles had no further effect. This decrease was smallest in the igneous rocks and largest in the sedimentary types. Variations due to orientation were within the precision of measurements (± 5%). In most cases the thermal conductivities were linear with the reciprocal of the temperature and were within 14% of published data obtained by other methods. Measurements were made by a cut-bar comparison method in which the sample was sandwiched between two reference or metering bars made of Pyroceram 9606 class-ceramic. The apparatus consisted of a Dynatech model ICFCM-N20 comparative thermal conductivity analyzer controlled by a Hewlett Packard model 3052A data acquisition system. A program was written to increment and cycle the temperature in steps between predetermined initial and maximum values. At each step the thermal conductivity was measured after steady-state conditions were established. The rocks furnished by the Bureau of Mines were quarried in large and fairly homogeneous lots for use by researchers at various laboratories. To investigate any anisotropy, cores were taken from each rock cube perpendicular to each of the cube faces. Samples 2 in. in diameter and approximately 0.75 in. thick were prepared from the cores and were dried in a vacuum oven for at least one month prior to taking measurements.

Parsons Brinckerhoff Quade and Douglas, Inc.
Thermal guidelines for a repository in bedrock.
Y/OWI/SUB-76/16504, Office of Waste Isolation, Oak Ridge, TN,
44 p., September 1976

This report summarizes the findings of a study conducted by Parsons Brinckerhoff to develop general thermal guidelines for the underground storage of canisters containing high level nuclear waste. The report contains general thermal guidelines for spacing canisters in three types of rock--shale, limestone, and granite; a survey of thermophysical rock properties and rock mechanical behavior; and a recommended approach for determining rigorously the thermal design criteria for the storage facility. This recommended approach would couple rock mechanics and thermal considerations in a transient, finite element model capable of simulating the excavation and canister emplacement sequence. In recognition of the nascent stage of the waste storage program, we've closed this report with suggestions for future research and for future analytical and experimental efforts.

The canisters of nuclear waste will produce significant quantities of heat from radioactive decay. The storage facility, therefore, must be designed in a manner that will permit this heat to be dissipated safely and efficiently. Crucial to this design is proper canister pitch.

The two major constraints on canister pitch are the maximum allowable temperature of the stored nuclear waste and the overall structural integrity of the underground facility.

Pusch, R.
Water uptake in a bentonite buffer mass. A model study.
KBS-TR--23, Kaernbraenslesaekerhet, Stockholm, Sweden, 27 p., August 1977

Safe deposition of radioactive waste products requires a number of conditions, an important one being the maximum temperature of about 100°C that can be accepted for the buffer mass. This temperature level has been chosen to guarantee the crystal stability of the bentonite component, to restrict the solubility of all mineral components, and to minimize the various negative effects of water vapor in the system. All this means that the heat conductivity lambda of the buffer mass must be sufficiently high. Since lambda varies considerably with the degree of water saturation, the water uptake and the associated temperature changes have been investigated in model tests.

Ratigan, J.L.; Callahan, G.D.
Evaluation of the property of the finite element method; Project Salt Vault thermo/viscoelastic simulation.
Y/OWI/Sub--78/22303/11, 62 p., March 1978

The study presents an assessment of the predictive capability of a two-dimensional finite element structural program by performing a simulation of the project salt vault experiment and comparing the numerically computed deformations to these which were measured during the experiment. The numerical method is evaluated in addition to the geometric and chronological approximations and the salt characterization.

Robertson, E.C.
Thermal conductivity of rocks.
USGS Open File Report 79-356, U.S. Geological Survey, 51 p., 1979

A mathematical model of a physical property of a rock would be most satisfactory for explanation and prediction if based on a physical model incorporating the chemical and physical properties of the pure components of the rock, including the minerals and fluids composing the rock. The calculation of some scalar properties of rocks is easily done (e.g. density and specific heat), but vector properties of second order, e.g. most thermal properties, require more complicated models for calculation from pure mineral properties because of the effects of texture and anisotropy. This present compilation of thermal properties is primarily empirical, not theoretical, due to insufficient data on composition, pore characteristics, and temperature effects. The principal objective of preparing the graphs of this report is to provide a means of estimating thermal conductivity for practical purposes, that is, estimation of conductivity of a rock from a few of its characteristics. It is anticipated that conductivity estimates from the graphs for inaccessible or unmeasurable rocks, and of course as a time-saver, will be useful for radioactive waste, heat flow, and geothermal resource appraisals.

Analysis of measurements of the thermal conductivity of rocks made at about 35°C (300 IO and 50 bars (5MPa) reveals that the effects of porosity, water content, and quartz or olivine content can be combined in a single plot for purposes of estimating the conductivity. Graphs have been prepared for the principal rock types, in the belief that line drawings provide more immediately understood information than tables do, and with the purpose of showing how well the data support the lines drawn. Each figure is described separately herein, and the intercepts of the lines are given so that interpolated values can be calculated. By showing the quartz or olivine data on the plots, an estimate of uncertainty can be made. The important temperature effect on conductivity is shown on a separate set of drawings.

Smith, D.D.
Thermophysical properties of Conasauga shale.
Y-2161, Oak Ridge Y-12 Plant, Oak Ridge, TN, 28 p., December 1978

Thermophysical-property characterizations of five Conasauga shale cores were determined at temperatures between 298 and 673 K. Methods of specimen fabrication for different tests were evaluted. Thermal-conductivity and thermal-expansion data were found to be dependent on the structure and orientation of the individual specimens. Thermal conductivities ranged between 2.8 and 1.0 W/m-K with a small negative temperature dependence. Thermal expansions were between 2 and 5 x 10^{-3} over the temperature range for the group. Heat capacity varied with the composition.

Sweet, J.N.
Pressure effects on thermal conductivity and expansion of geologic materials.
SAND 78-1991, Sandia Laboratories, Albuquerque, NM, 49 p., February 1979

Through analysis of existing data, an estimate is made of the effect of pressure or depth on the thermal conductivity and expansion of geologic materials which could be present in radioactive waste repositories. In the case of homogeneous dense materials, only small shifts are predicted to occur at depths $\leq$ 3 km, and these shifts will be insignificant as compared to those caused by temperature variations. As the porosity of the medium increases, the variation of conductivity and expansion with pressure becomes greater, with conductivity increasing and expansion decreasing as pressure increases. The pressure dependence of expansion can be found from data on the temperature variation of the isobaric compressibility. In a worst case estimate, a decrease in expansion of about 25% is predicted for 5% porous sandstone at a depth of 3 km. The thermal conductivity of a medium with gaseous inclusions increases as the porosity decreases, with the magnitude of the increase being dependent on the details of the porosity collapse. Based on analysis of existing data on tuff and sandstone, a weighted geometric mean formula is recommended for use in calculating the conductivity of porous rock. The effect of pressure on the conductivity of rock with liquid inclusions will be small unless the liquid is boiled away in response to elevated temperature produced by the presence of a nearby heat source. In this case an appreciable decrease in the conductivity could occur. As a result of this study, it is recommended that measurement of rock porosity versus depth receive increased attention in exploration studies and that the effect of porosity on thermal conductivity and expansion should be examined in more detail.

Sweet, J.N.; McCreight, J.E.
Thermal conductivity of rocksalt and other geologic materials from the site of the proposed waste isolation pilot plant.
SAND-79-1665. Sandia Laboratories, Albuquerque, NM, 33 p., March 1980

The measurements first reported by Acton on the thermal conductivity of samples taken from a borehole at the site of the proposed Nuclear Waste Isolation Plant (WIPP) near Carlsbad, NM, have been extended to include additional samples and higher temperature measurements. Samples for measurements were taken from several depths of three wells, including the well AEC 9 from which Acton obtained his samples. These samples ranged from relatively pure rocksalt (NaCl) with small amounts of interstitial anhydrite to essentially nonsalt samples composed of gypsum or clay. The measurements in this latest series were conducted at Sandia, the Los Alamos Scientific Laboratory (LASL), and at Dynatech Corp. In general, the data from the three laboratories agreed reasonably well for similar coarse grained translucent rock salt samples, with the LASL and Sandia results typically being about 20% higher than those of Dynatech. On the basis of these experiments, it is concluded that the thermal conductivity of materials found at the site can be predicted to an accuracy $\pm30\%$ from knowledge of the composition and grain size of these materials.

GROUP B.5: GENERAL REPORTS

Acres Consulting Services Ltd., Toronto, Ontario, Canada
Radioactive waste repository study. Part 2.
AECL-6188-2, Atomic Energy of Canada, Ltd., Whiteshell Nuclear Research Establishment, Canada, Pinawa, Manitopa, 198 p., November 1978

This is the second part of a report of a preliminary study for AECL. It considers the requirements for an underground waste repository for the disposal of wastes produced by the Canadian Nuclear Fuel program. The following topics are discussed with reference to the repository: 1) geotechnical assessment, 2) hydrogeology and waste containment, 3) thermal loading and 4) rock mechanics.

Asher, J.M.
National waste terminal storage program. Progress report, May 1977.
Y/OWI/TM--45/8, Office of Waste Isolation, Oak Ridge, TN, 79 p., June 1977

Status reports are given on the following studies: Gulf coast salt domes, east coast triassic shale basin and gulf coast clays; volcanic rocks; crystalline rocks; Pierre shale; geologic studies at the Nevada test site; heat transfer/thermal analysis; waste/rock interaction; rock mechanics; borehole plugging; safety and reliability studies, shale and clay mineral studies; and data management. Engineering projects reported on include shale in situ tests, dome salt in situ test, cooperative field testing in Sweden, and Nevada test site in situ tests. Design studies of canister transporter, canister emplacement hole drilling, cask scoping study, and SNM assay and accountability. Progress in conceptual design of NWTS repositories 1 and 2 is discussed. Status of the planning and analysis project is given.

Atlantic Richfield Hanford Co., Richland, Wash.
Preliminary feasibility study on storage of radioactive wastes in Columbia River Basalts. Volume 1.
ARH-ST-137 (vol 1), 183 p., November 1976

Geologic, hydrologic, heat transfer and rock-waste compatibility studies conducted at the Atlantic Richfield Hanford Company to evaluate the feasibility of storing nuclear wastes in caverns mined out into the Columbia River basalts are discussed. The succession of Columbia River plateau flood basalts was sampled at various outcrops and in core holes and the samples were analyzed to develop a stratigraphic correlation of the various basalt units and sedimentary interbeds. Hydrologic tests were made in one bore hole to assess the degree of isolation in the various deep aquifers separated by thick basalt accumulations. Earthquake and tectonic studies were conducted to assess the tectonic stability of the Columbia River plateau. Studies were made to evaluate the extent of heat dissipation from stored radioactive wastes. Geochemical studies were aimed at evaluating the compatibility between the radioactive wastes and the basalt host rocks. Data obtained to date have allowed development of a hydrostratigraphic framework for the Columbia River plateau and a preliminary understanding of the deep aquifer systems. Finally, the compilation of this information has served as a basis for planning the studies necessary to define the effectiveness of the Columbia River basalts for permanently isolating nulcear wastes from the biosphere.

Bourke, P.J.
Heat transfer aspects of underground disposal of radioactive waste.
AERE-R-8790, Atomic Energy Research Establishment, Harwell, England, 16 p., December 1977

Heat loss from blocks of vitrified waste buried in granite is described quantitatively and likely block and rock temperatures are roughly estimated. Interactions between the heat transfer and suggested engineering schemes for disposal are considered. The need for a more thorough analysis to determine whether or not maximum safe temperatures would be exceeded is discussed in relation to the overall management of the waste. The setting of different temperature limits for the bulk of the rock, for local hot spots and for the waste is suggested to allow for safety criteria which will have to be specified and met. An experimental program to provide data for confident quantitative prediction of the heat transfer is proposed.

Bourke, P.J.; Chapman, N.A.
Effects of decay heating on rocks and leakage from repositories for radioactive waste. Preprint, Atomic Energy Research Establishment, Harwell, England, 1981

Prediction of heat emissions from high level and plutonium contaminated wastes and of temperature fields produced after burial of representative quantities of these wastes in deep hard fractured rock is reviewed. The near-field effects of heating on corrosion of canisters, leaching of solidified waste and release of radionuclides by water from fractures in the rock are considered. The far-field effects of total heat emission and repository design on water-borne transport of radionuclides to the surface are estimated. Conclusions about the relative importance of these thermal effects and about further research needed to improve prediction of release and transport are presented.

Thermal effects on leakage from burial in plastic and unfractured rocks are assessed separately because the phenomena involved are different. Since study of relevant thermal effects on these rocks is only now beginning in the United Kingdom these are reviewed only briefly.

Brookins, D.G. (University of New Mexico)
Thermodynamic considerations underlying the migration of radionuclides in geomedia: Oklo and other examples.
CONF-781121; Scientific Basis for Nuclear Waste Management, v. 1, G.J. McCarthy (Ed.) p. 355-366, November 1979

The prediction of migration or retention of radionuclides in geomedia is one of the major problems associated with radioactive waste disposal. Use of Eh-pH diagrams based on sound thermodynamic data has proven useful for both the Oklo Natural Reactor and for sedimentary uranium deposits. Appropriate diagrams for temperatures in the range from 25 to 200°C allow predictions to be made which are entirely consistent with available measurements. For geomedia where organics are present and for brines, experimental data must be employed to supplement any predictions made on a purely inorganic thermodynamic data base. Thermodynamic treatment of transport of actinides and/or lanthanides in bedded salt deposits is more problematic, yet first experimental results under high Eh conditions indicate a high degree of retardation of the actinides and lanthanides, lesser retardation of Ru, and very little retardation for Cs, Sr, Tc, I, and Sb. Reducing conditions would increase the retardation for Ru, Tc, and Sb; with clays present Cs, Sr and possibly I should be less mobile.

Callahan, G.D.
Summary on conceptual repository analyses and evaluations and in situ heater experiments for FY 1977.
Y/OWI/SUB-77/22303/8, RE/SPEC Inc., Rapid City, SD, 33 p., November 1977

During the time period October 1, 1976 through September 30, 1977, RE/SPEC Inc. has completed and initiated several tasks related to various aspects of the National Terminal Waste Storage Program. The specific tasks, involving generally rock mechanics efforts, included analytical/numerical simulations of repository concepts utilizing the finite-element method, quasi-static and creep testing of laboratory specimens, and in situ experimentation. The major portion of the overall work effort has been devoted to the analytical/numerical simulations, with the laboratory testing program acting as a support facility in developing material properties and constitutive relations from various salt formations.

The analysis efforts have involved a parametric scoping of the global temperature fields and the potential for global thermal fracturing around repository facilities. Previous analyses in this regard were local in the sense that they were performed for regions about the waste emplacement drillholes and around the room and pillar configurations. Further analysis activities involved an extension of previous work related to the structural analysis of progressively mined solution cavities, completion of the simulation of the structural behavior of the Diamond Crystal Salt Company/Jefferson Island Mine, and assessing short-term room closures for repository configurations in salt. Each of the above analysis efforts were, in general, evaluated in conjunction with rock properties data obtained through the laboratory testing program. Contract negotiations with Akzona/International Salt Company for use of the Avery Island (AI) Mine facilities were not completed until late August, 1977. Thus, the major work effort regarding the in situ testing has involved planning and equipment procurement.

In view of overall supportive activities, time and effort was expended in developing laboratory data acquisition systems; fabricating additional creep testing equipment; generating users manuals for the finite-element computer codes to complement their earlier documentation; and hosting a workshop/review meeting for modeling subcontractors and interested parties for the Office Of Waste Isolation (OWI).

Cheung, H.
Investigations of the performance of solidified high-level nuclear waste forms.
UCRL--52700 Lawrence Livermore Laboratory, University of California, Livermore, CA, 198 p., January 1979

The Lawrence Livermore Laboratory has been providing technical support to the Nuclear Regulatory Commission in the development of regulations, regulatory guides, and branch technical positions for the management of nuclear wastes. Studies of solidified high-level waste during the period from 1976 to 1978, when work was terminated because of shifting of national emphasis onto spent fuel disposal, are presented in this report. The problem of management, i.e., handling, generation, and disposal of solidified high-level waste derived from operation of commercial light-water reactors, requires a thorough systematic solution to protect health and ensure safety. A definition of the problem by describing the components of the waste management system is given: the waste form, the containers, storage and transportation appurtenances, handling equipment, the repository surface and underground facilities, the repository site, and the operations. A systems analysis methodology to assess the hazards of waste management is developed. Data on accident probabilities, waste form characteristics, and geological and hydrological properties of potential repository sites are compiled. A wide range of management scenarios are generated. Limited sensitivity and uncertainty analyses are performed. On the basis of available information, preliminary investigations showed that transportation and interim storage are of most concern. Also identified are areas needing further study: transportation data base, thermal and seismic aspects of interim storage, human factors, geochemical transport of radionuclides, and ground water composition, among others. In addition to the technical solution of the problems, brief consideration to historical and socioeconomic aspects is also given.

Dickey, B.R.; Hogg, G.W.
Heat transfer in high-level waste management.
CONF-790822-1, AIChE National Meeting, Boston, MA, 42 p., August 1979

Heat transfer in the storage of high-level liquid wastes, calcining of radioactive wastes, and storage of solidified wastes are discussed. Processing and storage experience at the Idaho Chemical Processing Plant are summarized for defense high-level wastes; heat transfer in power reactor high-level waste processing and storage is also discussed.

Ensminger, D.A.; Oston, S.G.
Aspects of nuclear waste management. Vol 1. Pre-emplacement risks.
UCRL--15167 (vol. 1), Analytic Sciences Corp., Reading, MA, 58 p., August 1979

Results of the previous one-dimensional thermal analysis of interim storage accidents are confirmed by more detailed two-dimensional calculations. Waste temperatures calculated for interim storage accidents are moderately sensitive to assumptions concerning canister spacing and are much less sensitive to boundary conditions at the canister top. For an individual 100 m from a transportation accident involving solidified high-level waste, the expected dose from gamma radiation is on the order of or smaller than the other possible exposures in the same accident. For the calculation of expected doses to the worst-situated individual in accidents: expected population dose should remain the primary measure of pre-emplacement risk in order to maintain comparability of consequences of different types of accidents.

Envirosphere Company
Review of proposed formats for safety analysis reports for radioactive waste repositories in deep geologic formations, Final report.
Y/OWI/SUB-77/45706, EBASCO Services, 24 p., September 1977

At present, federal regulations and requirements for the licensing or approval of a waste repository facility do not exist. It is anticipated that in the future, Part 60 of Title 10 of the Code of Federal Regulations (10CFR60) will be developed and will contain requirements for a waste repository comparable to those of 10CFR50 for nuclear power plants, including General Design Criteria, (10CFR50, Appendix A).

It is prudent to assume that, prior to the construction of a waste repository facility, the ERDA will be required to submit a Safety Analysis Report (SAR) to the Nuclear Regulatory Commission (NRC). As with SAR's for other nuclear facilities, this report must detail the characteristics of the proposed facility and of the proposed site area, and describe the potential for safety-related interactions between the facility and site. Since the NRC has not yet issued a format guide for the preparation of such a report, OWI has proceeded independently with the development of an outline which will serve as the basis for the ERDA design and facility licensing activities.

Fairchild, P.D.; Jenks, G.H.
Avery Island - Dome salt in situ test.
Y/OWI/TM-55, Office of Waste Isolation, Union Carbide Corporation - Nuclear Division, Oak Ridge, TN, 47 p., June 1978.

A primary objective of this in situ test is to provide quantitative data that can be used to determine whether or not bedded and domal salt will behave similarly in response to heating. Project Salt Vault (PSV), as previously performed in bedded salt, has the potential of significantly contributing to the design of a repository in some salt if the data and subsequent analytical and numerical models developed from PSV can be validated for domal salt by such generic field tests.

The Avery Island (AI) heater test will provide a test case wherein a quantitative a priori prediction using laboratory measured properties and numerical analysis methods can be tested for adequacy of simulating effects in the field. Specific information and data objectives include:

1. determining thermal conductivity in situ for comparison with laboratory measured values;

2. determining the temperatures and temperature gradients in the adjacent salt for comparison with predicted values based on laboratory measured properties; and

3. determining the associated deformations and stresses in the salt for comparison with values predicted by rock mechanis computer modeling techniques.

Data related to the method of canister emplacement and relative to the physical and chemical interactions between the salt and the protective sleeve are needed to better understand and evaluate the design and cost implications of the present retrievability policy and implementation plans. Providing such data (e.g., hole-closure rates, hole-closure pressure, and corrosion attack on protective sleeve, etc.) is a second objective of this in situ test.

Fairchild, P.D.; Russell, J.E.
In situ experiments related to nuclear waste repository design.
Y/OWI/TM-25 (Rev), Office of Waste Isolation, Oak Ridge, TN,
24 p., August 1977

The Office of Waste Isolation of Union Carbide Corporation - Nuclear Division has been charged by the Department of Energy with the responsibility of providing deep, land-based repositories in geologic formations for disposing of nuclear waste from the commercial fuel cycle. The design and construction of waste repositories require information relative to the behavior of rock under high-temperature and -pressure conditions for long periods of time.

Experience has shown that although laboratory data characterizing rock properties are both necessary and useful for preliminary design studies, the bahavior of the rock mass must ultimately be determined by in situ tests, preferably performed at approximately the same depth and in the same formation horizon as would be used for the waste repository. This paper describes types of relevant in situ tests.

Peates, F; Keen, H.
Researching radioactive waste disposal. Atomic Energy Research Establishment, Harwell, England, New Sci., V. 77, No. 1090, p. 426-428, February 1976

At present it is planned to use the vitrification process to convert highly radioactive liquid wastes, arising from the nuclear power programmes, into glass which will be contained in steel cylinders for storage. The UKAEA in collaboration with other European countries is currently assessing the relative suitability of various natural geological structures as final repositories for the vitrified material. The Institute of Geological Sciences has been commissioned to specify the geological criteria that should be met by a rock structure if it is to be used for the construction of a repository though at this stage disposal sites are not being sought. The current research programme aims to obtain basic geological data about the structure of the rocks well below the surface and is expected to continue for at least three years. The results in all the European countries will then be considered so that a firm commitment may be made to select a site for a potential repository, when a far more detailed scientific research study will be instituted. Heat transfer problems and chemical effects which may occur within and around repositories are being investigated and a conceptual design study for an underground repository is being prepared.

Gera, F.
Radioactive waste disposal in geological formations.
IAEA-CN-36/313, Comitato Nazionale per l'Energia Nucleare, Rome, Italy, p. 337-349, May 1977

The nuclear energy controversy, now raging in several countries, is based on two main issues: the safety of nuclear plants and the ability to dispose of long-lived radioactive wastes safely. Consideration of the evolution of the hazard potential of waste as a function of decay time leads to a somewhat conservative reference containment time in the order of 100,000 years. Several concepts have been proposed for the disposal of long-lived waste; at present, emplacement into suitable geological formations under land areas can be considered the most promising option.

It is almost impossible to define detailed criteria for selecting suitable sites for disposal of long-lived wastes. Basically there is a single criterion: that the geological environment must be able to contain the wastes for at least 100,000 years. However, due to the extreme variability of geological settings, it is conceivable that this basic capacity could be provided by a great variety of conditions. The predominant natural mechanism by which waste radionuclides could be moved from a sealed repository in a deep geological formation into the biosphere is leaching and transfer by groundwater. Hence the greatest challenge is to give a satisfactory demonstration that isolation from groundwater will persist over the required containment time. Since geological predictions are necessarily affected by fairly high levels of uncertainty, the author considers that the only practical approach is not a straightforward forecast of future geological events, but a careful assessment of the upper limits of geological changes that could take place in the repository area over the next hundred thousand years. If waste containment were to survive these extreme geological changes, the disposal site could be considered acceptable. If some release of activity were to take place as a result of the hypothetical events, the disposal solution might still be acceptable if the environmental consequences were characterized by sufficiently low radiological risk.

The author concludes that rock salt and argillaceous sediments are considered the most favorable media since they behave plastically, and large masses of these materials should be capable of withstanding significant diastrophism without acquiring secondary permeability. However, it is possible that in areas of great tectonic stability and in particularly favourable geological situations, other materials, less intrinsically advantageous, might also be acceptable.

Hardy, M.P. and staff (Basalt Technology Unit)
Phase I heater test plan for the thermomechanical response of basalt.
RHO-BWI-CO-15 Rev 2, Rockwell Hanford Operations, 98 p.,
December 1978

The Near-Surface Test Facility Phase I testing consists of four separate in situ thermomechanical tests to be performed using electric heaters in a single thick basalt flow called the Pomona flow which is exposed at Gable Mountain within the Hanford Site. The purpose of the Phase I test is to measure the in situ properties of basalt, determine the modes of rock failure under stresses similar to those expected in a nuclear repository, determine the upper acceptable limit of power input to the basalt, demonstrate the interaction of a number of simulated waste canisters, and verify computer models to be used in future design work.

This test plan provides background information, justification, design objectives, and a detailed description of the individual tests.

Kisner, R.A.; Marshall, J.R.; Turner, D.W.; Vath, J.E.
Nuclear waste projections and source-term data for FY 1977. Y/OWI/TM-34, Office of Waste Isolation, Oak Ridge, TN, April 1978

A description of the light-water reactor (LWR) fuel cycle and the nature of the radioactive wastes is basic to the design and evaluation of terminal waste repositories. For these projections, the fuel cycle is represented as a typical system of operations related to the back end of the LWR fuel cycle. Wastes, as prepared for disposal, are described in terms of form, volume, radioactivity, heat generation, and weight. To obtain these waste projections, three fuel management computer codes were used: ORIGEN, KWIKPLAN, and WASPR. A brief description of these codes and their usage is included. Also included are descriptions of the containers assumed to be used for the handling and geologic disposal of the various waste types. The principal purpose of the report is to document the data generated for the Office of Waste Isolation (OWI) and its subcontractors.

Projections of LWR fuel cycle waste are based on OWI modification of the ERDA "mid-case" forecast of 1976 for nuclear power growth in the United States. In this case, the installed nuclear electric capacity rises from a nominal 5[illegible] GW(e) in year 1977 to a nominal 480 GW(e) in the year 2000. The power reactor grid is assumed to consist entirely of LWRS.

The four basic fuel cycle scenarios considered follow the same growth curve. These fuel cycles range from spent unreprocessed fuel (SURF) through full recycle of both uranium and plutonium. The overall objective is to compare the impacts of the alternative fuel cycles rather than to describe the impacts of reactor scenarios or the use of raw materials for power generation.

Ramspott, L.D.; Ballou, L.B.; Carlson, R.C.; Montan, D.N.; Butkovich, T.R.; Duncan, J.E.; Patrick, W.C.; Wilder, D.G.; Brough, W.G.; Mayr, M.C.
Technical concept for test of geologic storage of spent reactor fuel in the Climax, Granite, Nevada Test Site.
UCID-18197, Lawrence Livermore Laboratory, University of California, Livermore, CA, 47 p., June 1979

The spent fuel test in the Climax Granite at the Nevada Test Site is a generic test in which spent fuel assemblies from an operating commercial nuclear reactor are emplaced at, and retrieved from, a plausible waste repository depth in a typical granite. Eleven canisters of spent fuel are emplaced in a storage drift 420 m below the surface along with six electrical simulator canisters. Two adjacent drifts contain electrical heaters which are operated so as to simulate the initial five years of the temperature-stress-displacement fields of a large repository. The site is described, and the pre-operational measurement program and characteristics of the spent fuel are given. Both thermal and mechanical response calculations are summarized. The field instrumentation and data acquisition systems are described, as well as the system for handling the spent fuel.

Sattler, A.R.; Hunter, T.O.
Pre-WIPP in situ experiments in salt.
SAND79-0625, Sandia Laboratories, Albuquerque, NM, 96 p., August 1979

This document presents a systematic and interrelated matrix of in situ (field) experiments to be performed in salt. Most of these experiments will be performed in a specially excavated area of a southeastern New Mexico potash mine. The goals of the program are to obtain scientific and engineering data applicable to the design of emplacement and containment systems for radioactive waste disposal in bedded salt formations in southeastern New Mexico and to acquire information to resolve generic technical issues relevant to isolation of high-level waste in salt formations.

The general objectives of many of these experiments are:

1. To evaluate the thermochemical/mechanical response of the bedded salt to emplaced heat sources at a relatively site-specific scale and geologic situation.

2. To determine, and if present, evaluate unusual or unexpected features of the salt response to a simulated canister heat load.

3. To evaluate the predictive capability of the calculational methods as may be utilized in the scientific modeling and engineering design processes for the WIPP facility.

4. To develop a framework of fundamental information (and associated instrumentation) for the design and installation if in-situ experiments in the WIPP facility.

Storch, S.N.; Prince, B.E.
Assumptions and ground rules used in nuclear waste projections and source term data. ONWI-24, Office of Nuclear Waste Isolation, 124 p., September 1979

Assumptions and ground rules of long term domestic commercial nuclear waste projections published in studies by Union Carbide's Office of Waste Isolation, Arthur D. Little, Inc., and DOE are compared to those of the Commercial Waste Management Impact Statement, prepared by Battelle Pacific Northwest Laboratory. Target capacity growths associated with these projections range from 183 to 570 GW(e) for the year 2000. Each study regards the once-through (no recycle) fuel cycle as a reference case; however, fuel cycles employing reprocessing and various recycle strategies were also considered. The studies are compared with respect to characteristics and packaging/shipment features of spent fuel and wastes generated from reprocessing and other fuel cycle activities. Issues associated with the interim storage of spent fuel are discussed along with the characteristics and issues relating to ore mill tailings and non-fuel cycle wastes. Finally, assumptions and limitations associated with certain computer codes (viz., ORIGEN, KWIKPLAN, WASPR, and DISFUL) employed in the four waste projection studies are outlined. Overall, the report is intended to serve as a guidebook in relating information contained in the published waste projection studies.

Sutherland, S.H.; Allen, G.C., Jr.
High-level defense waste and spent-fuel characterization for geologic waste repositories.
Trans. Am. Nucl. Soc., v. 30, p 283-284, 1978

Sandia Laboratories, Albuquerque, NM, is presently characterizing the thermal and radiation sources of U.S. Department of Energy high-level defense waste (HLDW) and light-water reactor (LWR) spent fuel (SF) which ultimately may be retrievably stored at Federal geologic waste repositories (GWR). This HLDW and SF characterization is of particular importance in evaluating the long-term impact (up to 100,000 yr) of storing the thermally active materials in the GWR. Possible consequences of the thermal energy generation by the HLDW and SF stored in the GWR which are being studied include water migration near the waste canisters or fuel assemblies, thermally accelerated room closure (which is of particular interest if the storage site is located in salt formations), and the stressing of surrounding geologic formations caused by thermal expansion and the heating of aquifers. Studies concerning these effects require detailed thermal histories of the HLDW and SF for many thousands of years.

It is recognized that the long-term thermal energy generation by HLDW and SF is of considerable importance to GWR. The thermal source histories for SF and HLDW are being developed to assist in the design and evaluation of geologic waste repositories.

Thadani, M.
A cost optimization study for geologic isolation of radioactive wastes. Prepared by Teknekron, Inc. for Battelle Pacific Northwest Laboratories, 172 p., May 1979

Current Federal plans for the isolation of high-level radioactive wastes and spent fuel include the possible placement of these wastes in deep geologic repositories. It is generally assumed that increasing the emplacement depth increases safety because the wastes are farther removed from the phenomena that might compromise the integrity of their isolation. Also, the path length for the migration of radionuclides to the biosphere increase with depth, thus delaying their arrival. However, increasing the depth of emplacement adds cost and operational penalties. Therefore, a trade-off between the safety and the cost of waste isolation exists.

A simple algorithm has been developed to relate the repository construction and operation costs, the costs associated with construction and operational hazards, and the costs resulting from radiological exposures to future generations to the depth of emplacement. The cost-optimum emplacement depths are estimated by summing the cost elements and determining the depth at which the sum would be the least.

The relationship between the repository construction costs and the depth of the depository was derived from simplified rock mechanics and stability considerations applied to repository design concepts selected from the current literature and the available data base on mining and excavation costs. In developing the relationship between the repository costs and the depth of the depository, a worldwide cost information data base was used. The relationships developed are suitable for application to bedded salt, shale, and basalt geologies.

Tyler, L.D.
Model development for in situ test results in argillaceous rock. SAND--79-1275C, Sandia Laboratories, Albuquerque, NM, presented at ONWI-LLL Workshop on Thermomechanical Modeling for a Hard Rock Repository, 23 p., June 1979

Near surface heater tests have been conducted in two different geologic settings for argillaceous rocks. The results of these tests have provided the in situ data necessary to develop the thermomechanical models for predicting the response of argillaceous rock to thermal load representative of high-level nuclear waste.

Tyler, L.D.; Cuderman, J.F.; Krumhansl, J.L.; Lappin, A.
Near-surface heater experiments.
SAND--78-1401C, Sandia Laboratories, Albuquerque, NM, presented at OECD/NFA meeting, In Situ Heating Experiments in Geologic Formations, 72 p., September 1978

Full-scale near-surface heater experiments are presently being conducted by Sandia Laboratories in the Conasauga formation at Oak Ridge, Tennessee, and in the Eleana formation on the Nevada Test Site, Nevada. The purposes of these experiments are: (1) to determine if argillaceous media can withstand thermal loads characteristic of high level waste; (2) to provide data for improvement of thermomechanical modeling of argillaceous rocks; (3) to identify instrumentation development needed for further in situ testing; and (4) to identify unexpected general types of behavior, if any. The basic instrumentation of these tests consists of a heater in a central hole, surrounded by arrays of holes containing various instrumentation. Temperatures, thermal profiles, vertical displacements, volatile pressurization, and changes in in situ stresses are measured in each experiment as a function of time, and compared with pretest modeling results. Results to date, though in general agreement with modeling results assuming conductive heat transfer within the rock, indicate that the presence of even small amounts of water can drastically affect heat transfer within the heater hole itself, and that small amounts of upward convection of water may be occurring in the higher temperature areas of the Conasauga experiments.

GROUP B.6: FOREIGN PROGRAMS

Dejonghe, P.
La gestion des dechets radioactifs en Belgique.
Radioactive waste management in Belgium.
IAEA-CN-36/187
Centre d'étude de l'énergie nucléaire (CEN/SCK), Mol, Belgium
pp. 419-435, May 1977

In 1975 the research association BelgoWaste was founded in order to prepare a technical and administrative plan for radioactive waste management in Belgium and to take the preliminary steps for establishing an organization which would be responsible for this activity. The association made a survey of all forecasts concerning radioactive waste production by power reactors and the fuel cycle industry based on various schemes of development of the nuclear industry. From the technical point of view, the reference plan for waste management envisages: purification at the production site of large volumes of low-level effluents; construction of a central facility for the treatment and intermediate storage of process concentrates (slurries, resins, etc.) and a medium-level waste, centralization assuming that adequate arrangements are made for transporting waste before final treatment; maximum recovery of plutonium from waste and treatment of residual material by incineration at very high temperatures; treatment at the production site of high-level effluents from irradiated fuel reprocessing; construction of an underground long-term storage site for high-level treated waste and plutonium fuel fabrication waste (deep clay formations are at present preferred); and disposal of low-level treated waste into the Atlantic Ocean. It is intended to entrust the entire responsibility for treatment, disposal and storage of treated waste to a single body with participation by the State, the Nuclear Energy Research Centre (CEN/SCK), the electricity companies and Belgonucleaire. The partners intend to set up their facilities and services in the area of Mol.

Dyroff, H.; Fleischmann, F.K.; Witte, H.
Radioactive waste in the Federal Republic of Germany.
IAEA-CN 36/121, NUKEM GmbH, Hanau, Federal Republic of Germany,
p. 395-407, May 1977

From 1974 to 1976 a system study was made of radioactive waste in the Federal Republic of Germany. To this study, ordered by the Federal Ministry for Research and Technology and directed by NUKEM, members of the nuclear industry and Government research centres contributed. The results are presented in the paper. The growth of nuclear power and the corresponding nuclear industry, the development of research centres and of the other facilities involved, were projected up to 1990 and are presented with foreseeable trends up to the year 2000. On the basis of this growth, the total amount of radioactive waste to be expected in the FRG through 1990-2000 is calculated and categorized according to physico-chemical and radiological properties and nuclide content. The state of technology and current R&D activities in the fields of interim storage, conditioning, transport and ultimate storage, and their usefulness for the establishment of a comprehensive waste-management system are described and discussed.

Gusev, D.E.; Belitskii, A.S.; Turkin, A.D.
Solutions to the problems of radiation safety and environmental protection in connection with the handling of radioactive waste at nuclear power stations in the USSR.
IAEA-CN-36/349, USSR, p. 183, May 1977

This paper sets forth the basic trends of work on environmental protection in connection with the removal and disposal of liquid and solid radioactive wastes from nuclear power plants in the USSR. Radioactive waste management practices are described, together with the main requirements of the State Health Inspectorate in regard to the radiological safety of the population during waste disposal. Some results are presented of health and radioecological research bearing on the safe processing and disposal of radioactive wastes from nuclear power plants.

Hamstra, J.; Verkerk, B.
Review of Netherlands programme for geological disposal of radioactive waste.
IAEA-CN-36/289, Netherlands Energy Research Foundation (ECN), Petten, The Netherlands, p. 467-479, May 1977

Work started in 1972 on the exploration of possibilities in the Netherlands for permanent disposal of radioactive waste in one of the Zechstein salt formations in the North-eastern part of the country. It was decided to aim at a leached-out cavity for disposal of low- and medium-level solid wastes, which mainly arise at power stations, and at a mine as a repository for solidified reprocessing waste. The latter would be needed at a later date than the cavity, but both had to be located in the same formation. Site selection work, in close co-operation with the State Geological Service, and with the help of information kindly provided by the oil and salt companies, led to the choice of five most promising formations. Preparatory work was done by performing three reconnaissance drillings in the salt dome, which was the first choice. Design of the cavity and infrastructure is described as well as the deep borehole concept developed to make optimal use of the full salt-dome geometry. One of the drillings will be extended to a depth of 2600 m and, in the borehole, convergence measurements will be made to show the feasibility of the vertical storage concept. A review is also presented of the safety analysis performed, and of the programme for the coming years, expected to be carried out under contract with the Commission of the European Communities. Closely related to the safety aspect are soil retention measurements that have been performed with fission-product and actinide nuclides in nearly saturated salt solutions in order to evaluate retention times after a postulated water leaching incident.

Held, C.; Hintermayer, H.P. (Austria)
Comparison of concepts for independent spent fuel storage facilities (Austria).
Seminar on the storage of spent fuel elements, Madrid, Spain, p. 179-195, June 1978

The design and the construction costs of independent spent fuel storage facilities show significant differences. reflecting the fuel receiving rate (during the lifetime of the power plant or within a very short period), the individual national policies and the design requirements in those countries. Major incremental construction expenditures for storage facilities originate from the capacity and the type of the facilities (casks or buildings), the method of fuel cooling (water or air), from the different design of buildings, the redundancy of equipment, an elaborate quality assurance program, and a single or multipurpose design (i.e. interim or long-term storage of spent fuel, interim storage of high level waste after fuel storage). The specific costs of different designs vary by a factor of 30 to 60 which might in the high case increase the nuclear generating costs remarkably. The paper also discusses the effect of spent fuel storage on fuel cycle alternatives with reprocessing or disposal of spent fuel.

Kulichenko, V.V.; Martynov, Y.P.; Krylova, N.V.; Vlasov, V.I.; Kryukov, I.I.
Handling of waste from reprocessing of the fuel elements of a fast neutron reactor.
ERDA-TR-203, IAEA Meeting on LMFBR Fuel Reprocessing, Leningrad, USSR, All-Union Scientific Research Institute of Inorganic Materials, Moscow
6 p., May 17-21, 1976

The report discusses thermal problems that may arise during the organization of the storage of solidified waste from the reprocessing of the fuel of a fast neutron reactor for short cooling times of the fission products. It presents the results of calculations of permissible dimensions of cylindrical blocks under various cooling conditions. The effect of extraction of various isotopes on the storage conditions is demonstrated.

Also discussed are the properties of solid materials suitable for burying the waste from reprocessing of the fuel elements of fast neutron reactors.

Larsson, A.; Hultgren, Å.; Lind, J.
Management of radioactive waste and plutonium in the Swedish perspective.
IAEA-CN-36/554, Sweden, p. 447-454, May 1977

In May 1976 the Governmental Committee on Radioactive Waste (the Aka Committee) submitted its final report to the Swedish Government. The report summarizes a thorough investigation of questions dealing with spent nuclear fuel and radioactive waste. For Sweden the study recommends reprocessing spent fuel as a primary alternative. This should be closely linked with fabrication of mixed-oxide fuel from recovered material for rapid return as fresh fuel in the energy-producing reactors. Such a scheme would have the double advantage of both facilitating waste management and avoiding stockpiling of pure plutonium. The possibility of treating the spent fuel entirely as waste, not utilizing its fuel value, was also considered. Basically, national reprocessing, including possibilities for international, particularly Nordic, regional collaboration is envisaged by the Committee.

The findings and proposals of the Committee are discussed in the light of the recent development on the nuclear scene in Sweden. On the economic side, it is argued that the utilities should include all costs relating to the back end of the fuel cycle in the budgets for their energy production programmes. Reprocessing and waste management neither can nor should be seen as ordinary commercial ventures. Consequently the planning to cover the important needs at the back end of the nuclear fuel cycle is hardly likely to be initiated and undertaken by means of the market mechanism. Careful efforts in this regard are instead required at the national and international levels. The particular sensitivity connected with spent nuclear fuel and plutonium is derived from concern relating to environmental safety and proliferation of nuclear weapons. Together with economic and technical considerations, these two broad categories of concern, including physical security and safeguardability, are crucial in the selection and precise formulation of alternatives to be chosen for the back end of the nuclear fuel cycle. Also affecting national options will be international considerations, such as bilateral or multilateral agreements, including the Non-Proliferation Treaty.

Lopez Perez, B.; Ramos Salvador, L.; Martinez Martinez, A.
Programa nacional Espanol en materia de gestion de desechos radiactivos.
IAEA-CN-36/206, Junta de Energia Nuclear, Spain, p. 437-445, May 1977

The Spanish Energy Programme assumes an installed nuclear electrical power of 23,000 MW(e) by 1985. Spain is therefore making an effort in the management of radioactive wastes that can be summarized as follows: 1) Formulation and review of the regulation on the management of radioactive wastes. 2) Development of the processes and equipment for the treatment of solid, liquid and gaseous wastes from the CNEN "Juan Vigon", as well as those from the Soria Nuclear Centre; solidification studies of radioactive wastes arising from the reprocessing. 3) Evaluation of radioactive waste treatment systems of the new installed nuclear power plants; assistance to the operators of nuclear and radioactive facilities. 4) Increase the storage capacity of the pilot repository for solid radioactive wastes of IAEA categories 1 and 2, located in Sierra Albarrana; studies of adequate geological formation for storage of solid wastes of IAEA categories 3 and 4. 5) Studies of long-term surface storage systems for solidified radioactive wastes arising from reprocessing.

Mandahl, B.; Persson, B.; Wikdahl, E-E.
Handling waste at Swedish nuclear power plants.
IAEA-CN-36/282, Oskarshamn Power Company, Oskarshamn, Sweden, p. 561-572, May 1977

The Swedish nuclear power programme started with a 460-mW BWR at Oskarshamn in 1972. The main practical experience in nuclear waste management originates from this unit. Since 1975, five further reactor units have been taken into use and there are now definite plans for a total of 13 units. The waste handling in Sweden now considered is therefore orientated towards a system with 13 operational units. The paper describes the end products and the waste handling systems currently in use. Present methods and equipment are discussed as well as trends towards modification of these techniques. Estimates are made of the quantities of the end products and their radioactive content. Necessary decay times before the waste can be released as non-active material are also estimated. Lay-out and capacity of the waste stores at some plants and the need for transport equipment at the sites are described. The paper also discusses the need for centralized long-term storage and even methods for centralized waste treatment aimed at reducing the volume of material requiring storage.

Miyanaga, I.; Sakata, S.; Ito, A,; Amano, H.
Development of radioactive waste management at the Japan Atomic Energy Research Institute.
IAEA-CN-36/156, Japan Atomic Energy Research Institute, Tokai, Ibaraki, Japan, pp. 455-466, May 1977

The main efforts in the treatment of low- and medium-level radioactive waste have gone into reducing the volume of waste. For high-level waste, studies are being carried out on solidification and partitioning techniques, in preparation for completion of the fuel cycle in Japan. For marine disposal of low-level wastes planned by the JAEC, significant information has been obtained on the integrity and leaching behaviour of cement-solidified wastes. The paper describes the present state of development of the techniques. For the treatment of low- and medium-level wastes: (1) An incinerator with two-stage ceramic filters has been tested, and the decontamination factor was found to be 10^4 for various nuclides. (2) The reverse osmosis method with a cellulose acetate membrane has been tested for laundry liquid waste. More than 99% of the ^{60}Co was removed together with detergents. (3) Solidification products of spent ion-exchange resin with polyethylene have been proved superior to asphalt products in mechanical properties, water resistance and volume reduction. Homogeneous cement-solidified waste in 200-litre sealed drums did not form any cracks or defects under high hydrostatic pressure. The leaching ratio of ^{137}Cs for the first year was estimated to be lower than 0.3%. For the treatment of high-level wastes: (1) Vitrification using natural zeolite has been developed and the properties of the products proved to be excellent. (2) A partitioning procedure consisting mainly of solvent extraction and ion exchange has been studied. Reduction of the amount of alkaline agent by introducing the denitration technique and reduction of the resin volume by adopting porous type resin was achieved.

Randl, R.P.
Waste management in the Federal Republic of Germany. A survey of policy and research and development.
IAEA-CN-36/98, Federal Ministry for Research and Technology, Bonn, Federal Republic of Germany, p. 157-165, May 1977.

The concept of closing the nuclear fuel cycle in the Federal Republic of Germany is described with special emphasis on waste conditioning and disposal in selected salt formations. The practice and technology of the present status of the treatment of low-level and medium-level wastes are discussed. The different high-level waste-solidification programmes which are at present carried out in the Federal Republic are presented. It is shown that a wide variety of feasible and safe processes are in hand to secure the appropriate product quality for the terminal storage. Experience on the disposal of low-level and medium-level wastes from the Asse experimental programme is described. A large programme on the disposal of high-level waste has been started recently and the future development lines are discussed. The concept for waste management and disposal on an industrial scale, e.g. for the "integrated back-end fuel-cycle park", is presented and the necessary steps to reach the implementation of this concept until the start-up operation of the industrial reprocessing plant are described.

Richter, D.; Korner, W.
Disposal of radioactive wastes produced in nuclear installations in the German Democratic Republic.
CONF-760310; STI/PUB/433; IAEA-SM-207/44; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proc. of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, v. 2, p. 271-286, 1976

As conventional energy sources have decreased in availability, the GDR has been developing its nuclear power production. Studies have been conducted to establish the best methods for disposal of radioactive wastes from the nuclear fuel cycle. Methods investigated included storage above ground, shallow burial, storage in salt mines and formations, and injection into porous bedrock. All types of radioactive waste were considered in the study and the various aspects compared from the standpoints of economics and nuclear safety. As a result, the solution chosen was the secondary usage of abandoned salt mines. Construction of such a repository is currently underway in the Bartensleben mine, located in the cleft salt diapir of the upper Aller Valley. Volume of the storage area is about 5 x 10(E+6) m^3.

Sedov, V.M.; Kolychev, B.C.; Konstantinovich, A.A.; Kulichenko, V.V.; Nikipelov, V.,V.; Nikiforov, A.S.; Martynov, Yu. P.; Oziraner, J.N.; Dolgov, V.V.; Shatsillo, V.G.
Development of methods of solidification and burial of radioactive wastes from the fuel cycle.
IAEA-CN-36/350, USSR, p. 625, May 1977

The report deals with methods for disposal of radioactive wastes from reprocessing plants. Apparatus and flowsheets developed in the USSR for solidification of low- and intermediate-level wastes are discussed. The main research and development results on treatment of high-level waste by two methods are presented. The first comprises production of vitreous materials in an electric furnace without precalcination. The second involves predehydration and calcination of liquid wastes in a fluidized-bed facility followed by vitrification in a ceramic crucible. Processes for intermediate-level waste solidification are also discussed. The study of the behaviour of varicus solidified wastes (vitreous, bitumenized, etc.) in storage has allowed the development of disposal conditions. The paper considers the design of storage places located at fuel reprocessing plants. The main technical and economic data are given on the solidification and storage processes for solids.

Thomas, K.T ; Sunder Rajan, N.S.; Balu, K.; Khan, A.A.
Management of radioactive waste. An overview of the Indian programme.
IAEA-CN-36/388, Department of Atomic Energy, Bombay, India, p. 409 - 417, May 1977

An overview of the management of radioactive waste with particular reference to the Indian nuclear programme is presented. The initial design philosophy of the radwaste management system is discussed in relation to accepting a calculated, minimum discharge of radioactivity to the environment. A brief report of the operational experience with the low and intermediate level radwaste systems is given. Factors that influence the review of the present philosophy for future adoption are presented. Some methods being developed for decreasing release of radioactivity to the environment are discussed. Among techniques considered are solar evaporation, delay and decay of fission rate gases from power reactors, and concentration and storage of ^{85}Kr from fuel reprocessing plants. Problems in the management of high-level and α-bearing wastes are discussed with particular reference to the nature of the waste generated and the policy under implementation for their management. The matrices, solidification processes, modes of interim storage and criteria for selection of site for ultimate disposal of the solidified high-level waste in geological formations are described. An approach to the solution to the problem of management of α-bearing waste is also presented.

Tomlinson, M.; Mayman, S.A.; Tammemagi, H.Y.; Merritt, W. .; Morrison, J.A.; Irvine, H.S.; Vivian, G.A.; Gale, J.; Sanford, B.; Dyne, P.J.
Management of radioactive wastes from nuclear fuels and power plants in Canada.
IAEA-CN-36/178, Atomic Energy of Canada Ltd., Whiteshell Nuclear Research Establishment, Manitoba, p. 167-181, May 1977

The nature of Canadian nuclear fuel and nuclear generating plant radioactive wastes is summarized. Full exploitation of fission energy resources entails recovery of all fissile and fertile material from spent fuel and separating the fission products as wastes for disposal. A plan for final disposal of all the radioactive wastes is a key component of the waste management scheme. Principles of a scheme for safe, responsible disposal of long-lived radioactive wastes deep underground, in isolation from man and the biosphere, are outlined. The status of the development and construction programme is indicated. It is planned to select a site in either a hard rock formation or in a suitable salt bed by 1981 so that a repository can be constructed to begin a demonstration phase in 198[illegible] The repository is to be capable of eventual expansion to accommodate all Canadian nuclear wastes to at least 2050 when in full-scale operation. Extensive geotechnical studies have been initiated in order to select a site, and design and test the repository. The incorporation of fission products in solids that in the short term (17 years) dissolve more slowly than plutonium decays has been demonstrated. Investigations of long-term stability are in hand. The principle of retardation of migration of fission products, so that they decay before surfacing, has been tested. Additional capacity for storage of used fuel prior to reprocessing and disposal is required by 1986 and a preliminary design has been prepared for a pool facility to be located at a central fuel recycling and disposal complex. A demonstration of dry storage of fuel in concrete containers is in progress. The quantities of CANDU generating-station wastes and the principles and methods for managing them are summarized. Methods for volume reduction and immobilization by solidification are well advanced. A radioactive-waste operations site is being developed with several different types of surface storage, each with multiple barriers against leakage. A reactor decommissioning study has been completed. Estimated costs of the various waste management operations are summarized.