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Limits on the Thermal Energy Release From Radioactive Wastes in a Mined Geologic Repository

Technical Report

March 1983

Julia A. Scott

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Battelle Memorial Institute
Columbus, OH 43201

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Office of NWTS Integration
BATTELLE Project Management Division

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ABSTRACT

The thermal-energy release of nuclear wastes is a major factor in the design of geologic repositories. Thermal limits need to be placed on various aspects of the geologic waste-disposal system to avoid or retard the degradation of repository performance because of increased temperatures. The thermal limits in current use today are summarized in this report. These limits are placed in a hierarchical structure of thermal criteria consistent with the failure mechanism they are trying to prevent.

The thermal-criteria hierarchy is used to evaluate the thermal performance of a sample repository design. The design consists of disassembled BWR spent fuel, aged 10 years, close packed in a carbon steel canister with 15 cm of crushed salt backfill. The medium is bedded salt. The most-restrictive temperature for this design is the spent-fuel centerline temperature limit of 300 C. A sensitivity study on the effects of additional cooling prior to disposal on repository thermal limits and design is performed.

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EXECUTIVE SUMMARY

Background

Many of the technical uncertainties associated with the disposal of nuclear waste in mined geologic repositories are related to the thermal energy released by the emplaced waste and its effect on the repository system. As the heat generated by the emplaced waste migrates from the waste form through the canister and emplacement area into the host formation and through the overlying rock, it will have definite impacts on the integrity and the potential retrievability of the waste packages, the stability of the rooms and pillars, the integrity of the host rock, surrounding rock, and any overlying aquifers, the long term uplift and subsidence of the overlying rock, and ultimately the habitability of the biosphere.

Because of these effects, quite a few thermal limits have been suggested for the various aspects of the geologic disposal system, such as the waste form, canister, and host formation. Unfortunately, these thermal limits are currently scattered through the literature and there is a need for centralizing this information.

To help alleviate this problem, a survey of the literature was undertaken to determine the thermal limits in current use. In reviewing the literature, the potential ways in which a repository could fail

because of increased temperatures were identified. Thermal limits have been suggested for a subset of these failure mechanisms in order to prevent their occurrence. These thermal limits were also identified in the literature review.

Thermal Criteria Hierarchy

By identifying the various ways in which thermal energy released by nuclear waste can cause a repository to fail, a set of thermal criteria can be developed. It is then possible to arrange the thermal criteria into a hierarchical structure in order to provide a systematic approach to understanding the effects of thermal energy on the overall waste disposal system.

The first step in this process is to define the mined geologic disposal system. Figure ES-1 shows the components of the mined geologic disposal system, their definitions and functions, as they relate to the thermal energy released by the emplaced waste. As can be seen from Figure ES-1, and inferred from the above statement, not all aspects of a geologic disposal system are shown. Those which are not significantly affected by increased temperatures, such as the rock handling system, were not included.

In order for any part of the mined geologic system shown in Figure ES-1 to operate successfully, the systems in the lower levels must successfully complete their assigned functions. The functions required by a higher order system, therefore, will appear as constraints on the lower order systems. Knowing the functions each part of the waste disposal system must perform, it is possible to place constraints on them

FIGURE ES-1. HIERARCHY OF A MINED GEOLOGIC DISPOSAL SYSTEM

*Not all repository subsystems are shown.

with regard to their tolerance to the heat generated by the emplaced waste. Each of the lowest order systems in Figure ES-1, i.e., Items 6 through 12, can be further expanded to include the thermal criteria, and the thermal limits which apply to each system.

Geologic System Hierarchy

As an example of how Items 6 through 12 of Figure ES-1 can be expanded, Figure ES-2 shows the geologic system, Item 10 of Figure ES-1, in more detail and the thermal criteria that apply to it. The uppermost entry in Figure ES-2 specifies the component of the geologic disposal system under consideration. The second level in the hierarchy, Items 2 through 4 of Figure ES-2, lists the thermal criteria, as developed from the possible ways this particular component of the waste disposal system could fail because of the thermal energy release. The last layer in the hierarchy lists the thermal limits which have been recommended to satisfy the thermal criteria.

In the geologic system, Figure ES-2, four potentially adverse effects of the thermal energy release were identified. These are the loss of physical integrity through shear failure in the rock mass because of thermally induced changes in its state of stress, excessive vertical deformations on the surface resulting from the thermal expansion of the rock mass, the creation of an excessive perturbed fissure zone (also due to the thermal expansion of the rock mass), and an excessive ground surface temperature rise. The first three potential failure modes in the geologic system have the capability to directly alter the performance of the geologic system, while the last failure

mode, an excessive ground surface temperature rise, may be environmentally unacceptable. The thermal criterion for each of these failure modes is shown in the second level of the hierarchy in Figure ES-2.

The thermal limits suggested for each thermal criterion are shown in the last level of the geologic system hierarchy. Only the limit on the ground surface temperature rise is actually expressed as a temperature. The remaining thermal limits for the geologic system are derived from the thermomechanical effects of the thermal energy release on the rock mass.

Summary of the Thermal Criteria and Limits

A summary of the thermal criteria and limits that apply to the mined geologic disposal of radioactive waste is presented in Table ES-1. These thermal criteria and limits correspond to the lower two levels in the waste form system, containment system, waste handling system, sealing system, geologic system, hydrologic system, and geochemical system hierarchies.

Reading Table ES-1 from left to right, the first column identifies the system to which the thermal criteria and limits apply. These systems are the same as those shown in the lowest level in Figure ES-1, i.e., the waste form, containment, waste handling, sealing, geologic, hydrologic, and geochemical systems. The second column lists the thermal criteria that have been developed for that particular system. In any system there may be a series of alternatives, such as bedded salt or basalt in the geologic system. These alternatives appear in the third column. Finally, the last column identifies the precise manner in

FIGURE ES-2. HIERARCHY OF GEOLOGIC SYSTEM LIMITS

**TABLE ES-1. SUMMARY OF THERMAL CRITERIA AND LIMITS FOR THE DISPOSAL OF RADIOACTIVE WASTES
IN GEOLOGIC REPOSITORIES**

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Waste Form	<p align="center">PREVENT ALTERATION OF CHEMICAL COMPOSITION</p> <p>Limit the temperature of the waste form to less than that required to cause changes in the chemical composition of the waste form</p>	Spent fuel	Maximum Temperature, C 250	Oxidation of ceramic fuel UO_2 to U_3O_8
		Spent fuel	200	Degradation of protective oxide film on Zircaloy clad
		Borosilicate glass HLW	500	Softening of glass
		Borosilicate glass HLW	500	Transition
		Thermosetting or urea formaldehyde resins TRU	300	Decomposition of waste form
	<p align="center">PREVENT COMBUSTION</p> <p>Limit the temperature of the waste form to less than that required for combustion of the waste form</p>			Combustion
	<p align="center">PREVENT LOSS OF PHYSICAL INTEGRITY</p> <p>Limit the temperature of the waste form to less than that required to cause structural changes in the waste form</p>	Spent fuel	Maximum Temperature, C 390	Rupture of clad by internal pressure
		Spent fuel	380	Stress-rupture of the clad
		Spent fuel	300	Stress-corrosion cracking of clad
		Borosilicate glass HLW	500 375	Devitrification
		Borosilicate glass HLW	500 - 800	Cracking of waste glass
		Cement TRU	150	Microcracking
	<p align="center">PREVENT VOLATILIZATION</p> <p>Limit the temperature of the waste form to less than that required for volatilization of the waste form. The resulting accumulation of gases could lead to explosive conditions</p>	Calcine HLW	Maximum Temperature, C 700	Cesium volatility
	<p align="center">PREVENT SPONTANEOUS IGNITION OF GASES</p> <p>Limit the temperature to less than that required for spontaneous ignition of generated gases</p>			Spontaneous ignition of gases

Use of Table Reads left to right, example The waste form can fail through changes in its original composition which for spent fuel can occur above 250 C due to the oxidation of the ceramic fuel from UO_2 to U_3O_8

TABLE ES-1. (Continued)

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Containment System	<p>PREVENT COMBUSTION OF CONTAINMENT SYSTEM</p> <p>Limit the temperature of the containment system to less than that required for combustion of the engineered barriers</p>			Combustion
	<p>PREVENT CORROSION OF CONTAINMENT SYSTEM</p> <p>Limit the temperature of the containment system to prevent the corrosion of the engineered barriers by either the waste form or the emplacement environment</p>			Corrosion
	<p>PREVENT INTERNAL PRESSURIZATION</p> <p>Limit the temperature of the waste form to less than that required for volatilization of the waste form in order to prevent internal pressurization of the canister which could lead to a breach in containment</p>	Calcine high-level waste	<p>Maximum Temperature, C</p> <p>700</p>	Significant cesium volatility
	<p>PREVENT LOSS OF PHYSICAL OR CHEMICAL INTEGRITY</p> <p>Limit the temperature of the containment system to less than that required to cause any changes in the physical integrity or chemical stability of the engineered barriers</p>	<p>304L SS</p> <p>Bentonite</p> <p>Zeolite</p>	<p>Maximum Temperature, C</p> <p>375</p> <p>100 – 300</p> <p>< 300</p>	Structural changes leading to increased susceptibility to stress cracking
Waste Handling System	<p>PREVENT LOSS OF PILLAR INTEGRITY</p> <p>Limit the temperature and temperature induced stress of the host rock to prevent any instability of the pillars</p>	<p>Basalt</p> <p>Granite</p> <p>Shale</p>	<p>Strength-to-Stress Ratio</p> <p>> 2 across the midheight of the pillar</p> <p>> 2 across the midheight of the pillar</p> <p>> 2 across the midheight of the pillar</p>	Pillar instability
	<p>PREVENT LOSS OF ROOM INTEGRITY</p> <p>Limit the temperature and temperature induced stress of the host rock to prevent any instability of the rooms</p>	<p>Basalt</p> <p>Granite</p> <p>Shale</p>	<p>Strength-to-Stress Ratio</p> <p>> 2 within 1.5 meters of room opening</p> <p>> 2 within 1.5 meters of room opening</p> <p>> 1 within 1.5 meters of room openings with continuous support</p>	Room instability

TABLE ES-1. (Continued)

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Waste Handling System		Bedded salt Domed salt	Percentage of Room Closure 10–15 10–15	Room instability
	PREVENT LOSS OF RETRIEVABILITY Limit the temperature of the host rock in the immediate vicinity of the waste package to prevent any structural changes which could prohibit the retrievability of the waste	Salt Granite Basalt Shale	Maximum Temperature, C 250 350 350 250	Decrepitation Fracture Fracture Fracture
	PREVENT LOSS OF SAFE WORKING CONDITIONS Limit the air temperature in the repository to provide a safe, comfortable working environment for personnel	Open rooms Retrieval operations in previously sealed rooms	Maximum Temperature, C 50 28	Excessive air temperature
	PREVENT EXCESSIVE WALL AND FLOOR SURFACE TEMPERATURES Limit the temperature of the walls, floors, etc., of the repository with which men, equipment, or materials may come in contact	Retrieval operations	Maximum Temperature, C 49	Excessive temperature
	PREVENT LOSS OF SHAFT INTEGRITY Limit the temperature of the rock mass such that the shaft and its components undergo no fracturing due to the induced thermo mechanical stresses and experience no significant ground water leakage			Fracturing and ground water leakage
	PREVENT LOSS OF SHAFT WASTE HANDLING CAPABILITY Limit the temperature of the rock mass such that any deformations of the shafts or shaft liners will be sufficiently small to allow for routine operations without major remedial work.			Shaft instability
Sealing System	PREVENT LOSS OF HYDROLOGIC INTEGRITY Limit the temperatures of the repository so that any adverse changes in the permeability of the shaft and borehole seals, or backfills is prohibited			Increased permeability

TABLE ES-1. (Continued)

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Sealing System	<p>PREVENT LOSS OF PHYSICAL INTEGRITY</p> <p>Limit the temperatures of the repository to prevent any adverse structural changes in the shaft and borehole seals, or the backfills.</p>			Structural changes
Geologic System	<p>PREVENT EXCESSIVE GROUND SURFACE TEMPERATURE RISE</p> <p>Limit the temperature rise at or near the ground surface to prevent any adverse effects on the biosphere.</p>	<p>All Media</p> <p>Bedded salt Domed salt Basalt Granite Tuff Shale</p>	<p>Temperature Rise, C</p> <p>< 4 within 3 meters of the surface</p>	Excessive temperature increase
	<p>PREVENT THE CREATION OF EXCESSIVE PERTURBED FISSURE ZONE</p> <p>Limit the thermomechanical stresses in the rock mass to prevent the creation of an excessive perturbed fissure zone.</p>	<p>All Media</p> <p>Bedded salt Domed salt Basalt Granite Tuff Shale</p>	<p>Extent of Perturbed Fissure Zone</p> <p>< 15 percent of the repository depth downward</p>	Fracture
	<p>PREVENT LOSS OF PHYSICAL INTEGRITY</p> <p>Limit the thermomechanical stress in the rock mass to prevent the fracture of the host rock or the overlying stratigraphy.</p>	<p>Bedded salt</p> <p>Domed salt</p> <p>Basalt</p> <p>Granite</p> <p>Tuff</p>	<p>Strength-to-Stress Ratio</p> <p>1 in the aquitards</p> <p>1 in the aquitards and cap rock</p> <p>1 everywhere in the far field</p> <p>1 in the middle 70 percent of the rock column between the repository and the surface</p> <p>1 in the middle 70 percent of the rock column between the repository and the surface</p>	Shear failure
	<p>PREVENT EXCESSIVE VERTICAL DEFORMATIONS</p> <p>Limit the thermal energy release to prevent excessive uplift or subsidence of the geologic system</p>	<p>All Media</p> <p>Bedded salt Domed salt Basalt Granite Tuff Shale</p>	<p>Vertical Displacement</p> <p>< 3 meters</p>	Uplift and/or subsidence

TABLE ES-1. (Continued)

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Hydrologic System	<p>PREVENT CREATION OF CONVECTION CELLS</p> <p>Limit the temperature of the rock mass to prevent the creation of convection cells which could increase the velocity of the flow system, or decrease the effective path length to the surface for radionuclide migration</p>			Convection cells
	<p>PREVENT GROUND WATER MIGRATION TOWARD HEAT SOURCES</p> <p>Limit the temperature gradients in the rock mass to prevent the migration of ground water toward the heat sources</p>			Brine migration
	<p>PREVENT HYDROTHERMAL REACTIONS</p> <p>Limit the temperature of the waste form to prevent hydrothermal reactions (after containment)</p>			Hydrothermal reactions
	<p>PREVENT EXCESSIVE TEMPERATURE RISE IN NEIGHBORING AQUIFERS</p> <p>Limit the temperature rise in the neighboring aquifers to prevent any adverse effects on the biosphere</p>	<p>All Media</p> <p>Bedded salt Domed salt Basalt Granite Tuff Shale</p>	<p>Temperature Rise, C</p> <p>< 6 in stagnant aquifers</p>	Excessive aquifer temperature rise
Geochemical System	<p>PREVENT LOSS OF GEOCHEMICAL INTEGRITY</p> <p>Limit the temperature of the rock mass to prevent any adverse changes in the chemical composition of the rock mass which could affect its capability to retard the migration of radionuclides</p>	<p>Granite</p> <p>Bedded salt Domed salt Basalt Tuff</p>	<p>Maximum Temperature, C</p> <p>125 from the near field outward to a distance equal to 15 percent of the repository depth</p> <p>100 outside the region defined above</p> <p>100 from the near field outward to a distance equal to 15 percent of the repository depth</p> <p>75 outside the region defined above</p>	Chemical changes in rock mass
	<p>PREVENT ADVERSE GEOLOGY/CONTAINMENT/WASTE FORM INTERACTIONS</p> <p>Limit the temperatures of the geochemical system to prevent any adverse chemical reactions between the rock mass, engineered barriers, or waste form</p>			Adverse geology/containment/waste form interactions

TABLE ES-1. (Continued)

Applicable System	Thermal Criteria	Alternative	Limit	Failure Mechanism
Geochemical System	<p>PREVENT LOSS OF SORPTION PROPERTIES</p> <p>Limit the temperatures of the rock mass and engineered barriers to prevent the loss of any sorption properties they may have.</p>	Zeolites	<p>Maximum Temperature, C</p> <p>< 300</p>	Loss of sorption properties

which the system alternative can fail. As an example, the waste form system can fail through changes in its original composition, which for spent fuel can occur above approximately 250 C due to the oxidation of the ceramic fuel from UO_2 to U_3O_8 .

As can be seen from Table ES-1, not all of the thermal criteria developed from the literature survey could be assigned thermal limits. These missing thermal limits include such items as the effect of increased temperatures on canister corrosion, the volatility of the wastes, the migration of ground water towards heat sources, and media-canister-waste form interactions. These gaps in the table occur because thermal limits for the corresponding thermal criteria were not found in the literature reviewed. Also, in some instances, more than one documented thermal limit was found for a thermal criterion.

Applications

The primary purpose of the thermal criteria hierarchy is the identification of those aspects of the waste disposal system which are adversely affected by the thermal energy released by the nuclear waste and the interrelationships among these areas. This approach will help to evaluate the thermal acceptability of a given repository design in a systematic fashion.

To illustrate how the thermal criteria hierarchy could thus be used, the thermal acceptability of a simple repository design was evaluated. Table ES-2 describes the repository configuration chosen for this purpose, a spent fuel BWR repository in salt. A set of maximum permissible temperatures for this repository design was then obtained

Table ES-2. Repository Characteristics

Characteristics	Specifications
Waste Description	
Age of Waste, yr	10
Thermal Loading, w	2.86
Mass Loading, MTHM	0.0029
Diameter of Pin, m	0.0062
Active Length, m	4.0
Canister Description	
Inner Diameter, m	0.23
Thermal Loading, kW	2.6
Backfill Thickness, m	0.15
Room Description	
Rows per Room	2
Row Separation, m	10.67
Hole Pitch, m	25.65
Adjacent Pillar Thickness, m	22.97
Canisters per Hole	1
Repository Depth, m	600
Materials	
Waste	BWR Spent Fuel Pins
Canister	Carbon Steel
Backfill	Crushed Salt
Medium	Bedded Salt

from the thermal criteria hierarchy. These are listed in Table ES-3. Where temperature limits were not available, such as the corrosion limit for the canister, they were not included.

The results of the thermal analysis of this repository design are shown in Figure ES-3. As can be seen from the radial temperature distribution of the spent fuel canister, the containment waste form, canister, and salt emplacement temperatures are all below their maximum values. At a distance of 0.86 meters from the centerline, the temperature of the salt is well below 75 C and declining, thereby assuring that the near-field and far-field limits will be met. At 100 years after disposal, the spent fuel centerline temperature is below 100 C and declining, as can be seen by examining the variation of the spent fuel centerline temperature with time. Assuming a required containment period in excess of 100 years, the postcontainment spent fuel temperature limit will thus be met.

Based strictly on temperature considerations, the chosen repository design is acceptable. A thorough analysis, however, would have required the evaluation of a complete set of thermal limits. Any missing thermal limits, therefore, would need to be determined, and the analysis would need to include not only temperature calculations, but also thermo-mechanical and hydrological calculations.

Recommendations

The thermal criteria hierarchy developed here should be periodically reviewed to incorporate other failure mechanisms and the

Table ES-3. A Partial Set of Restrictive Temperatures
for the Repository Configuration Shown in
Table ES-2(a)

Waste Form/Canister/Medium	Maximum Temperature, C
Spent Fuel	
during containment period	300
after containment period	100
Carbon Steel Canister	375
Salt	
emplacement area	250
near field	100
far field	75

(a) Where temperature limits were not available for the failure mechanisms developed in Chapter 3, they were not included.

RADIAL TEMPERATURE DISTRIBUTION OF SPENT FUEL WASTE PACKAGE

VARIATION OF SPENT FUEL CENTERLINE TEMPERATURE WITH TIME

FIGURE ES-3. RESULTS OF THE THERMAL ANALYSIS OF THE EXAMPLE REPOSITORY CONFIGURATION

missing thermal limits. Care should be taken in updating the hierarchy and determining the missing thermal limits in order to maintain internal consistency.

1 INTRODUCTION

The purpose of this paper is first to provide a summary of all the available thermal criteria for nuclear waste isolation in geologic repositories and their bases in current use, and second to identify those areas where either the criteria do not exist or the existing criteria are inadequate. To accomplish this, a hierarchy of thermal criteria will be developed. Analyses will then be conducted to demonstrate the interrelationships of thermal criteria in the hierarchy.

In this first chapter a discussion of why thermal criteria need to be developed will be presented. Following this, a brief description of the mined geologic system, and the waste types and their sources is given. The last section of this chapter will outline the organization of the rest of this report.

1.1 THE NEED FOR THERMAL CRITERIA

One of the major factors to be considered in the isolation of nuclear waste is the heat generated by the waste. Many of the technical uncertainties associated with the disposal of nuclear wastes in mined geologic repositories are related to this thermal energy release, the resulting thermal environment, and the impact this thermal environment has on the waste disposal system. The heat generated by the nuclear wastes will eventually flow from the waste form, through the canister and emplacement area, into the host formation, through the

surrounding and overlying rock to the surface and then out into the atmosphere. This migration of thermal energy will have impacts on:

- The integrity and the potential retrievability of the waste packages
- Mine room and pillar stability
- The integrity of the host rock and surrounding rock
- Any overlying aquifers or ground water flow
- The long-term uplift and subsidence of the overlying rock, and
- The biosphere.

Physical and chemical changes in the waste form or canister resulting from increased temperatures could jeopardize the integrity of the waste package if they were not controlled. As the surface temperature of the canister increases so does its susceptibility to corrosion. Should corrosion progress to the point where the canister is breached, the waste form will then be exposed to an environment which might contain ground water. Ground water provides the primary mechanism by which radionuclides can be transported to the environment. Any physical or chemical changes that may have occurred in the waste form are therefore important, as such changes will usually tend to decrease the waste form's leach resistance and generally increase the mobility of the waste. Leach rates for waste glasses, as an example, have been found to increase by approximately a factor of 10 for an increase of 100 C in temperature (U.S. Department of Energy, 1979). Stress-rupture of spent fuel pin cladding may occur at temperatures greater than 380 C, thereby decreasing the spent fuel's leach resistance.

As the heat generated by the wastes is transferred to the host formation the question of mine stability and integrity of the host rock

and surrounding medium becomes important. An added concern for repositories in salt is the decrepitation of the salt. The introduction of heat into this system will induce stresses in the host rock and the surrounding formation. These thermally induced stresses must be considered in the design of a repository to insure its structural stability. If the structural integrity of the host rock is lost, the possibility of ground water intrusion is increased. If the loss of structural integrity occurs in the near term, the ability to emplace and retrieve the nuclear wastes may also be endangered. In the stratigraphy overlying the repository, the heat generated by the emplaced wastes will cause the rock mass to expand, thus inducing positive vertical surface displacement. Then, as the heat-generation rate decreases, negative vertical surface displacements will occur. This deformation of the overlying strata may cause fracturing in the rock, thereby causing perturbations in the natural hydrological flow. Again, this could provide a path for the transport of radionuclides to the environment.

To assure that the energy generated by the emplaced wastes will not have a detrimental effect on a repository, a systematic determination of repository thermal design is required. This includes the establishment of limits for conditions affected by heat and, from these, the determination of acceptable thermal loads that will not violate safety, engineering, and operational requirements. These design limits can be specified in terms of temperature; thermo-mechanical, thermohydrological, and thermochemical criteria; and performance constraints or limits.

Currently, it is not clear which part of the waste disposal system will have the most restrictive temperature limit. This limiting temperature could be the maximum waste form temperature, the waste form surface temperature, the container surface temperature, the temperature of the undisturbed rock immediately surrounding the container, or the temperature at some point in the surrounding geology (the near or far field). Also, the location of this limiting temperature will probably be different for each potential site.

1.2 THE MINED GEOLOGIC DISPOSAL SYSTEM

The concept of a mined geologic repository for the disposal of nuclear wastes is briefly summarized in this section. Figure 1-1 shows an artist's conception of a geologic repository and its support facilities. Surface facilities at the site provide for unloading the waste from trucks and/or rail cars, preparation of the waste for emplacement, and transfer of the waste to the underground workings. Excavated rock storage, general maintenance, administrative, and support areas are also located at the surface. The surface facilities will be connected by several shafts to the underground system.

The repository will provide for the receipt, preparation, transfer underground, and the emplacement and containment after closure of the radioactive wastes. In addition, the repository will provide the capability to monitor and, if necessary, retrieve the wastes.

A key element in the mined geologic disposal concept is the waste package. Figure 1-2 shows a possible schematic waste package and emplacement configuration for a spent fuel assembly. The function of the waste package is to provide for handling and emplacement and to

FIGURE 1-1. ARTIST'S CONCEPT OF A GEOLOGIC REPOSITORY AND ITS SUPPORT FACILITIES
(U.S. DEPARTMENT OF ENERGY, 1980)

FIGURE 1-2. SCHEMATIC DESIGN OF A WASTE PACKAGE FOR SPENT FUEL
(U.S. DEPARTMENT OF ENERGY, 1980)

insure retrievability of the wastes. During the thermal period the package must also provide for containment.

The primary purpose of the waste disposal system is to contain and isolate the radioactive waste. This goal is achieved by containing the waste for a specified period of time and controlling the release of the radionuclides after the containment period. The concept is to provide a series of independent barriers to the release of radionuclides to the biosphere. The multiple barriers that form this system fall into two categories:

- Geologic or natural barriers, and
- Engineered barriers.

Geologic barriers provide effective, long-term, naturally occurring barriers which isolate nuclear waste from man and the biosphere. Engineered barriers contain the waste by delaying the time of release and retarding the rate of release of radionuclides to the geologic environment. These man-made barriers must also be designed to withstand the more severe thermal conditions encountered initially.

The nature of the host rock is basic to the determination of the other barriers in the multibarrier system, and several media types are being considered. Properly chosen rock structures provide physical and chemical properties that contribute to repository strength. Sufficient repository depth and lateral extent of the rock mass contribute to the isolation capability of the repository. Geologic formations currently being investigated include bedded salt deposits and salt domes, basalt, and crystalline rocks such as granite, shale, and tuff. Figure 1-3 shows the locations of these geologic formations in the continental United States. Some regions of interest are shown in Figure 1-4.

ROCK SALT DEPOSITS

BASALTS

GRANITIC FORMATIONS

ARGILLACEOUS FORMATIONS

FIGURE 1-3. GEOLOGIC FORMATIONS BEING INVESTIGATED AS POSSIBLE HOST MEDIA

FIGURE 1-4. REGIONS OF INTEREST FOR TERMINAL STORAGE OF NUCLEAR WASTE

1.3 WASTE TYPES AND SOURCES

Sources of nuclear waste originate not only from electrical power generation but also from military, medical, industrial, and agricultural activities. Nuclear waste can be divided into six major categories depending on their origin, level of radioactivity, and potential hazard. These are spent (unreprocessed) fuel, commercial and defense high-level waste, transuranic wastes, cladding and hardware wastes, low-level wastes, and tailings.

Spent fuel is that fuel discharged from light water reactors (LWRs). Fuel for LWRs is enriched uranium dioxide (UO_2). The uranium dioxide is then sintered into the form of right circular cylinders approximately 1.5 centimeters in length and between 0.75 and 1.25 centimeters in diameter, depending on the reactor design. These pellets are then placed into long narrow tubes of 366 to 376 centimeters in length to form the fuel rods. A typical fuel rod is shown in Figure 1-5. The fuel rods are then arranged in a square array held together by spacer grids, end nozzles, and other structural materials to form a fuel assembly. The physical dimensions vary according to LWR type (pressurized water reactors, PWRs, or the boiling water reactors, BWRs), the four reactor suppliers and the many different reactor designs. Figure 1-6 shows a typical PWR fuel assembly.

High-level wastes originate when spent fuel, either from commercial or defense reactors, is reprocessed to recover uranium and plutonium. These wastes are characterized by high energy radiation. High-level waste consists of 99 percent fission products and actinides, approximately 0.1 percent iodine and bromine, and 0.5 percent uranium

FIGURE 1-5. TYPICAL PWR FUEL ROD (O'HARA ET AL, 1979)

FIGURE 1-6. PWR FUEL ASSEMBLY (O'HARA ET AL, 1979)

and plutonium. In the uranium-only recycle option, however, the plutonium content could be significantly higher. Present regulations require that all-liquid high-level waste be converted to a dry solid within five years after reprocessing.

As with high-level wastes, most transuranic wastes result from reprocessing. They consist of materials contaminated with elements heavier than uranium. Transuranic waste is characterized by medium to low energy radiation. Transuranic elements are often very long-lived.

Cladding and hardware wastes also originate with reprocessing. They consist of leached hulls, support rods, control rod drives, end fittings, support grids, and assorted springs and spacers. The major concern in cladding and hardware waste disposal is the capability of the materials to spontaneously ignite in air or other environs.

Low-level wastes are those radioactive wastes not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material. These wastes typically consist of small amounts of radioactive material in large volumes. Similarly, tailings are the by-products of the uranium mining and milling and consist of large volumes of rock and soil that have residual natural radioactivity. Neither of these waste types requires disposal in mined geologic repositories.

The temperatures which result with the disposal of these various waste types will not be the same in all cases. They will vary due to the differences in thermal output and activity decay among the waste types. With spent fuel the heat generation rates of discharged PWR and BWR fuel assemblies vary with burnup, specific power, and the composition of the assemblies when put into the reactor. The specific power

level affects the short-term heat generation rate, while the burnup affects the long-term heat generation of the assemblies. Heat generation rates for high-level wastes will vary with the reactor type (PWR or BWR), fuel type (UO_2 or mixed oxide fuel), and the reprocessing cycle.

1.4 REPORT ORGANIZATION

Following the introduction, Chapter 2 will identify the existing thermal criteria and their sources. Chapter 2 groups the thermal criteria as they apply to the very-near-field, near-field, and far-field regions of the repository. The development of the thermal criteria hierarchy is presented in Chapter 3. After a general discussion of the thermal criteria hierarchy, detailed discussions of each of the geologic disposal subsystems and their thermal criteria are given. Chapter 4 outlines some possible uses of the thermal criteria hierarchy and presents an example. The interrelationships among the criteria are also discussed. Chapter 5 provides recommendations for further work.

2 LITERATURE REVIEW

For purposes of analysis, the repository has been divided into three overlapping regions, the very near field, the near field, and the far field. The very near field consists of the waste form and its associated canister and emplacement area. The near field consists of the actual excavation area--the rooms and pillars. The far field consists of the formation and overlying stratigraphy at distances far removed from the repository. Without exception the existing thermal criteria have been defined with respect to these regions. For the purposes of this report the same repository divisions will also be used. The thermal constraints which apply to the very near field, near field, and far field are discussed in Sections 2.1, 2.2, and 2.3, respectively.

2.1 VERY NEAR FIELD

In the very near field the major concern is the maximum temperature of the waste form, canister, sleeve, overpack, and backfill. The influencing factors include:

- The canister and canister well design
- The very-near-field thermal properties
- The short-term thermal decay rates of the waste
- The individual canister thermal load, and
- The average canister spacing.

Peak temperatures are usually reached in the first five to ten years.

Table 2-1 lists the recommended peak temperatures for the waste forms that are under consideration. The predominant source of information for waste form temperatures is G. H. Jenks (1977). For the case of high-level waste, Jenks did not make the distinction between defense and commercial sources. The supporting arguments for the temperature limits are presented in the succeeding paragraphs.

For typical borosilicate waste glasses Jenks (1977) chose a temperature limit of 500 C based upon the following. The transition temperature for typical borosilicate waste glasses is approximately 500 C, with a slightly higher softening temperature. Significant increases in cracking and leach rates have been observed in test glasses that were heated for periods of a few months in the temperature range 500 to 800 C (Jenks, 1977). And, the migration of heavy, separable phases in the glass might take place above the softening temperature. Later work by Turcotte and Wald (1978) concerning the effects of temperature on the rate of devitrification of the waste glass during retrievable geologic storage has prompted Jenks to suggest the 500 C limit be reduced to 375 C during retrievable geologic storage. Turcotte and Wald (1978) have shown that considerable devitrification will occur at times less than one year at a temperature of 500 C.

Devitrification is the transition of the molecular structure of glass from the unordered or amorphous state to an ordered or crystalline state (O'Hara et al, 1979). There is a concern that when waste glasses devitrify the impurities in the glass may end up in the intergranular boundary areas where they may be easily leached. Devitrification has been observed to increase the leach rate of glass by a factor of two or three (Mendel, 1978).

Table 2-1. Recommended Maximum Temperatures
for the Waste Form

Waste Form	Maximum Temperature, C
Borosilicate Glass High-Level Waste	500 (Jenks, 1977) 375 (Turcotte and Wald, 1978)
Calcine High-Level Waste	700 (Jenks, 1977)
Sintered Glass Ceramic High-Level Waste	800 (Jenks, 1977)
Spent Fuel	200 (Jenks, 1977) 370 (Blackburn, 1978)
UO ₂	250 (Cordfunke, 1969)
Glass Transuranic Waste	500 (Jenks, 1977)
Cement Transuranic Waste	110 (Jenks, 1977) 150 (Lokken, 1978 and the American Society for Testing and Materials, 1978)
Bitumen Transuranic Waste	80 (Burns, 1971)(a)
Thermosetting Resin Transuranic Waste	300 (Gilmore, 1977)
Urea-Formaldehyde Resin Transuranic Waste	300 (Columbo and Neilson)
Slagging Pyrolysis Incinerator Transuranic Waste	(b) (U.S. Department of Energy, 1979)
Fused Cladding and Hardware Waste	400 (Jenks, 1977)
Compressed Cladding and Hardware Waste	200 (Jenks, 1977)
PuO ₂	500 (Jenks, 1977)
Gaseous Fission Product Waste	(c) (Miller and Wurm, 1980)

Table 2-1. (Continued)

-
- (a) The maximum allowable temperature for bitumen is very dependent on the waste type and quantity of waste incorporated in it. For bitumen containing salts from reactor water cleanup, a temperature limit of 80 C is recommended to avoid the separation of any incorporated salts.
 - (b) A temperature limit for the waste form from the slagging pyrolysis incinerator has not yet been reported. The temperature limit could be quite high, however, since the processing temperature of the incinerator is 1650 C.
 - (c) Solid waste forms for gaseous fission products are silver zeolite, calcium carbonate, metal matrices, and polymer matrices. Temperature limits for these waste forms have not yet been established.
-

Jenks (1977) specifies a temperature limit of 700 C for calcine waste since significant cesium volatility occurs at temperatures greater than 750 C. Calcine is a granular substance consisting of the simple oxides of waste nuclides formed by heating high-level liquid waste in air to high temperatures.

Preliminary evidence has shown sintered glass ceramic high-level waste to be stable to at least 800 C (Jenks, 1977). Glass ceramics are formed by a controlled devitrification, thereby eliminating the potentially detrimental effects caused by uncontrolled devitrification. The final product consists of small crystals held together with a glassy phase.

Several temperature limits have been suggested for spent fuel, all with the same intent -- maintaining the integrity of the fuel. Jenks (1977) has reported that at a temperature of about 200 C the protective oxide film of the Zircaloy[®] cladding will degrade in the absence of an oxidant. Blackburn et al (1978) have suggested that

stress-rupture of spent fuel pins will occur at a temperature of 380 C in PWR fuel assemblies and at a slightly higher temperature in BWR fuel assemblies. As a result of this, Blackburn et al (1978) recommended a maximum spent fuel temperature of 370 C. The Management of Commercially Generated Radioactive Wastes (U.S. Department of Energy, 1980), using Blackburn's work as a data source, chose a maximum spent fuel temperature of 300 C because of some residual uncertainty concerning possible stress-corrosion cracking.

In the presence of oxygen and high temperatures, UO_2 can be oxidized to U_3O_7 or U_3O_8 . Up to 250 C, UO_2 can be oxidized to U_3O_7 with little change in volume. Above 250 C oxidation to U_3O_8 results in a large volume expansion which crumbles the uranium oxide to powder and could split the Zircaloy cladding (Cordfunke, 1969).

Jenks (1977) chose the same maximum temperature, 500 C, for transuranic waste glass as he did for borosilicate high-level waste glass. Jenks, however, also noted that transuranic waste would probably have a relatively low thermal power and as a result the actual maximum temperature would be less than 500 C.

For cement transuranic waste the immobilization of the waste occurs by mixing aqueous solutions, slurries, and dry particulates with cement. The optimum proportions of cement and waste will vary and the water content must be carefully controlled to assure concretization without creating excess free water. Jenks (1977) has suggested a maximum temperature of 110 C for cement, which is the Atlantic Richfield et al (1977) maximum temperature for normal conditions in structural concrete in long-term, 1,000 years, surface storage of

high-level waste. Lokken (1978) and the American Society for Testing and Materials (1978) have recommended a temperature limit of 150 C for cement to avoid dehydration and thermal stresses which lead to micro-cracking and a reduction in structural properties.

Bitumen transuranic waste is created by mixing wet and solid wastes with bitumen at elevated temperatures to evaporate the water. The bitumen waste mixture is then usually discharged into drums. Depending on the type of bitumen, the softening point varies from 34 to 65 C (International Atomic Energy Agency, 1970). At the softening point several effects may occur: phase separation will occur near the softening temperature; mixing of tetraborates, iron, and aluminum salts causes hardening of the bitumen; and heating above the softening point can cause emission of gas or aerosols. The maximum temperature for bitumen is also very dependent on the waste type and loading. For example, 80 C is the recommended maximum temperature for bitumen containing salts from reactor water cleanup to avoid the separation of any incorporated salts (Burns, 1971).

The principle of the thermosetting resin process is to disperse the radioactive material in a polyester resin dissolved in styrene, obtain a homogeneous mixture, and then induce polymerization. A maximum temperature of 300 C has been recommended for thermosetting resins to avoid any decomposition of the material (Gilmore, 1977).

Liquid and dry transuranic waste is mixed with urea-formaldehyde resin and the reaction is catalyzed by homogeneous additions of NaHSO_4 . Urea-formaldehyde polymerizes and traps the waste in a solid structure with little heat or volume expansion. Columbo and Neilson

(1977) have recommended a maximum temperature of about 300 C to avoid decomposition of the polymer.

A temperature limit for the waste form from the slagging pyrolysis incinerator at the Idaho National Engineering Laboratory has not yet been reported. The maximum temperature could be quite high, however, as the processing temperature of the incinerator is 1,650 C.

Jenks (1977) has recommended a maximum temperature limit of 400 C for fused cladding and hardware wastes based on an observed increase in surface reaction rates at high temperatures.

As with fused cladding and hardware wastes, the surfaces of the Zircaloy hulls may become very reactive in the compressed cladding and hardware waste form as a result of the degradation of the protective oxide film. However, as the area-to-volume ratio of these wastes will be much larger than for the fused wastes, and accordingly, the possibilities for rapid surface reactions will be greater, Jenks (1977) suggests a temperature limit of 200 C.

Jenks (1977) has noted that high-fired, 1,000 C, PuO_2 is very refractory and is difficult to dissolve by normal techniques. Also, its vapor pressure is very low, $\ll 10^{-10}$ atm at 500 C. Accordingly, he has recommended a temperature constraint of 500 C.

Some of the solid waste forms developed for gaseous fission products are silver zeolite, calcium carbonate, metal matrices, and polymer matrices. Temperature limits for these waste forms, however, have not yet been established (Miller and Wurm, 1980).

Table 2-2 lists the recommended maximum canister temperatures. Jenks (1977) gives the only source of maximum canister temperatures based on actual structural changes. Other sources recommend maximum

Table 2-2. Recommended Maximum Canister
Temperatures

Canister	Maximum Temperature, C
304L Stainless Steel	375 (Jenks, 1977)
Carbon Steel	375 (Jenks, 1977)

canister temperatures, usually 300 to 500 C, designed to limit the corrosion rate of the canister should it come in contact with ground water.

Using austenitic stainless steel, probably 304L, as his canister material, Jenks (1977) noted that it underwent structural changes during long-term exposure in air at temperatures in the range of 400 to 900 C. The observed effect is an increased susceptibility to stress cracking when the steel is subsequently exposed to aqueous solutions. Jenks also noted that carbon steel is probably less susceptible to significant changes in structure and corrosion during long-term heating at temperatures greater than 400 C (1977). Not having any information to confirm this, however, he chose the same temperature limit for carbon steel as for austenitic stainless steel, 375 C for five year retrievable geologic storage.

Table 2-3 lists some recommended maximum backfill temperatures. In the multibarrier system the backfill has two major functions:

- To minimize contact between the waste package and the host rock
- To control the release of radionuclides into the host environment in the event of containment failure by the previous barriers.

The first function of the backfill can be further divided into two areas. First, the backfill is required to minimize contact between the waste package and the ground water to prevent any corrosion. Second, the backfill must protect the waste package from the impact of falling rock and from undue stress resulting from minor bedrock movement. The important backfill properties are permeability, bearing capacity, shear strength, plasticity, and sorption.

Table 2-3. Recommended Maximum Backfill
Temperatures

Backfill Material	Maximum Temperature, C
Bentonite	100 (Smith, 1980) 300 (Miller and Wurm, 1980)
Zeolite	300 (U.S. Department of Energy, 1979)

Bentonite has been extensively studied by Swedish scientists as a potential backfill material. However, they have expressed concern over the physical and chemical stability of pure bentonite in a repository environment at temperatures over 100 C (Ahlstrom, 1979). Miller and Wurm (1980), on the other hand, report that bentonite retains its properties to at least 300 C. Zeolites have also been proposed as a backfill material primarily because of their high-cation-exchange capabilities. Unfortunately, zeolites lose their water and therefore their cation-exchange capacity at temperatures well below 300 C (Miller and Wurm, 1980).

2.2 NEAR FIELD

The primary concern in the near field is the stability of the rooms and pillars as judged by the media temperatures and the rock strength-to-stress ratios. For a repository in salt this usually translates into an acceptable amount of creep, whereas in nonsalt repositories this usually corresponds to minimizing the possibility of failure by rock fracturing. The influencing factors include:

- The average waste emplacement density
- The time-dependent heat generation rate
- The average depth of the heat sources relative to the storage room floor
- The host rock thermal and mechanical properties, and
- The in situ stresses in the formation due to the geologic structures.

The above parameters and ventilation conditions also determine the personnel working conditions.

Table 2-4 lists the recommended peak media temperatures for salt, granite, basalt, and shale. A recommended maximum temperature for tuff was not found.

Cheverton and Turner (1972), in a study where the repository medium was bedded salt, specified a thermal constraint on the basis of a unit cell. They stated that no more than 1 percent of the salt volume in this unit cell should be at a temperature above 250 C. This limit was based upon experimentally observed decrepitation (fracturing) of the salt above 250 C. Jacobsson (1977) concluded from laboratory tests done on several unconfined rock samples from several locations

Table 2-4. Recommended Maximum Media
Temperatures

Medium	Maximum Temperature, C
Salt, bedded or domed	250 (U.S. Department of Energy, 1980)
Granite	350 (U.S. Department of Energy, 1980)
Basalt	350 (U.S. Department of Energy, 1980)
Shale	250 (U.S. Department of Energy, 1980)

that the decrepitation temperature of salt was between 260 and 320 C. Samples from other locations, however, did not show any decrepitation when heated to 400 C. Jacobsson (1977) has also recommended a limit of 350 C for granite and basalt, although a good basis for determining the maximum permissible temperature for hard rocks has not yet been established. For shale, a maximum of 250 C may be reasonable although, as with hard rocks, a good basis for determining the maximum permissible temperature has not been established.

J. E. Russell (1977), studying salt as a possible repository medium, defined a constraint based on stress and mine geometry parameters. From this thermomechanical analysis, Russell, using as his criterion the amount of room closure that would still allow access to the rooms at the end of the retrievability period, set a limit on the room closure of 10 to 15 percent of the original room height (see Table 2-5). An earlier work by Cheverton and Turner (1972), also studying bedded salt, defined a near-field constraint based on a unit

Table 2-5. Recommended Permissible Amounts of
Room Closure in Salt

Medium	Permissible Amount of Room Closure, %
Bedded Salt	10-15% (Russell, 1977)
Domed Salt	10-15% (Russell, 1977)

cell. They stated that no more than 25 percent of the salt volume in the unit cell should exceed 200 C.

Near-field constraints have also been given in terms of strength-to-stress ratios for the rooms and pillars by Dames and Moore (1978) (see Table 2-6). Strength-to-stress ratios are a way of determining rock stability. If the strength-to-stress ratio is less than one, then there is a potential for fracture failure according to the Mohr-Coulomb failure criteria. Should the strength-to-stress ratio be greater than one, however, then the rock mass should be mechanically stable. Dames and Moore considered three media types: granite, basalt, and shale. For a room in granite or basalt, they set a required strength-to-stress ratio of 2 within 1.5 meter of the opening to the room. Whereas for a room in shale, with continuous support, they set a required strength-to-stress ratio of 1 within 1.5 meters of the room opening. Dames and Moore also required a strength-to-stress ratio of 2 across the mid-height of the pillars for pillar stability in granite, basalt, and shale.

The strength-to-stress ratios, shown in Table 2-6, and the permissible amounts of creep for salt, shown in Table 2-5, can be translated into a recommended allowable waste emplacement density for a specified

Table 2-6. Recommended Maximum Media Strength-to-Stress (s/s) Ratios

Medium	Strength-to-Stress Ratio
Granite	2 within 1.5 m of opening (Dames and Moore, 1978) and 2 across the pillar mid-height
Basalt	2 within 1.5 m of opening (Dames and Moore, 1978) and 2 across the pillar mid-height
Shale	1 with continuous support (Dames and Moore, 1978) within 1.5 m of opening and 2 across the pillar mid-height

medium and waste type, as shown in Table 2-7. J. E. Russell (1977) developed the suggested waste emplacement density for salt. The waste emplacement densities for basalt and shale by Science Applications, Inc. (1976), are actually two-thirds of their calculated density. The Management of Commercially Generated Radioactive Waste (U.S. Department of Energy, 1980) also suggests allowable waste emplacement densities for the various media and waste types. In calculating the emplacement densities shown in Table 2-7, the following assumptions were made:

- Waste emplaced 10 years after removal from reactor
- The repository is loaded simultaneously and instantaneously
- Effects of stress upon thermal properties were not included
- Presence of water was neglected in the thermal analysis
- Only simplified horizontal stratigraphies were assumed.

Table 2-7. Recommended Average Waste
Emplacement Density, kW/acre

Waste Type Medium	Commercial High- Level Waste		Spent Fuel
Salt	150(a) 100	36(b) --	(Russell, 1977) (U.S. Department of Energy, 1980)
Granite or Basalt	190(a) 130	80(b) 130	(Science Applications, Inc., 1978) (U.S. Department of Energy, 1980,
Shale	120(a) 80	55(b) 80	(Science Applications, Inc., 1978) (U.S. Department of Energy, 1980)

(a) Average waste emplacement density for 5-year retrievability.

(b) Average waste emplacement density for 25-year retrievability.

During waste emplacement and retrieval operations the working conditions of the personnel will be of concern. Kaiser Engineers (1980) recommends a maximum air temperature of 50 C for retrieval operations in unsealed rooms. For retrieval operations in rooms that have already been backfilled, however, they recommend a maximum air temperature of 28 C, see Table 2-8. The National Institute for Occupational Safety and Health (NIOSH) (1972), recommends a WBGT (wet-bulb globe temperature) of 26 C be maintained in the repository room during the retrieval operations. The WBGT is a physiological heat stress index adopted by NIOSH which reflects the combination of air temperature, humidity, radiation, and wind speed. A rock temperature of 49 C has been suggested as the highest desirable temperature for the safe operation

Table 2-8. Recommended Maximum Working
Temperature

Working Conditions	Maximum Temperature, C
Retrieval operations in unsealed rooms	50 (Kaiser Engineers, Inc., 1980)
Retrieval operations in rooms from which backfill has been removed	28 (Kaiser Engineers, Inc., 1980) 26 (NIOSH, 1972) 49 (MIDES-WG)

of retrieval equipment by the Mine Design Study Working Group (MIDES-WG) of Sandia National Laboratories in its 1980 activity work package on blast cooling and continuous cooling.

2.3 FAR FIELD

The primary concern in the far field is to maintain the physical, chemical, and hydrologic integrity of the host rock and the overlying stratigraphy. Another major concern is to limit the temperature rise in the aquifers and at the surface of the earth. The influencing factors for this region include:

- The average waste emplacement density
- The long-term time dependence of the heat generation rate, and
- The average far-field thermal properties.

By assuring the integrity of the formation a long-term barrier between man and the environment and the radioactive waste is provided.

By defining the minimum strength-to-stress ratio required to maintain the stability of the far-field rock the physical integrity of the formation can be assured. Table 2-9 lists the currently recommended strength-to-stress ratios for domed salt, bedded salt, granite, basalt,

Table 2-9. Recommended Maximum Strength-to-Stress Ratio

Medium	Strength-to-Stress Ratio - NWTs Working Group on Far-Field Performance Constraints
Bedded Salt	> 1 in the aquitards
Domed Salt	> 1 in the aquitards or caprock
Granite Tuff	> 1 in the middle 70% of the rock column lying between the repository horizon and ground surface
Basalt	> 1 everywhere in the far field

and tuff developed by the NWTs Working Group on Far-Field Performance Constraints in its draft on repository performance constraints in the far-field domain. To determine the strength-to-stress ratios in Table 2-9, the in situ stress, mechanical properties, constitutive law, and initial temperature gradient in the formation must be known.

In general, the changes in stress in a formation caused by repository excavation and the thermal energy release, are compressive in nature. Therefore, there is a potential for shear failure in those regions where the rock mass shear strength is exceeded. For granite and tuff it has been suggested that the strength-to-stress ratio be greater than one in the middle 70 percent of the rock mass between the repository and the surface, see Figure 2-1 from the NWTs Working Group on Far-Field Performance Constraints draft. For basalt it has been recommended that the strength-to-stress ratio be greater than one everywhere in the far-field region. This is to ensure that the repository basalt flow remains intact. For bedded and domed salt it

FIGURE 2-1. SCHEMATIC CROSS SECTION OF A REPOSITORY IN GRANITIC ROCK SHOWING ZONES OF POTENTIAL FRACTURING (NWTS WORKING GROUP ON FAR-FIELD PERFORMANCE CONSTRAINTS)

has been recommended that the strength-to-stress ratio be greater than one in the aquitards or caprock.

A related parameter, known as the perturbed fissure zone, also has been defined by the NWTS Working Group on Far-Field Performance Constraints. As the rock mass undergoes thermal expansion, horizontal tensile stresses may be established in a zone near the surface. These tensile forces have the potential for opening vertical fractures, hence the name--the perturbed fissure zone. Consequently, it has been recommended that the maximum vertical extent of the perturbed fissure zone should not extend downward from the surface more than 15 percent of the repository depth in any medium.

Another parameter used to gage the far-field rock stability is the amount of vertical displacement, uplift, or subsidence experienced by the ground surface during the thermal period. Table 2-10 lists two recommendations for the maximum permissible vertical displacement. It should be noted that it is possible to meet the maximum vertical displacement limit and still fracture an underlying layer.

Table 2-10. Recommended Maximum
Vertical Displacement

Medium	Vertical Displacement, m	
All media--Bedded Salt Domed Salt Granite Basalt Tuff Shale	< 3	(NWTS Working Group on Far-Field Performance Constraints)
Bedded Salt Domed Salt	1.2 - 1.5	(Russell, 1977)

The NWTS Working Group on Far-Field Performance Constraints, in its draft on repository constraints on the far-field domain, recommends a maximum vertical displacement of less than 3 meters. They required the maximum vertical displacement to be less than those variations which occur by natural processes such as glacial rebound and erosion. Also, the vertical displacement must take place in a relatively smooth fashion.

J. E. Russell (1977) recommended a tentative range of 1.2 to 1.5 meters for the maximum permissible uplift over a repository. The 1.2 to 1.5 meter limit was based upon the assumption that rock-mass movements due to uplift would not be any worse than movements caused by subsidence over mines in sedimentary rocks, which are sometimes twice the suggested limit.

Because of the key role the shafts play in the operation of a repository, it is important that they maintain their integrity and suffer no deformations. After repository closure the main concern will be the long-term integrity of the shafts and borehole seals with regard to ground-water leakage. Table 2-11 lists some possible shaft and borehole constraints which are related to the thermal energy release of the emplaced wastes.

In order to preserve the natural flora and in general protect the biosphere some attempts have been made to determine the maximum permissible temperature rise at or near the ground surface, see Table 2-12. Science Applications, Inc. (1978) has recommended a maximum temperature rise of 0.5 C at the surface. The NWTS Working Group on Far-Field Performance Constraints also has suggested the maximum temperature increase within 3 meters of the ground surface be less than 4 C.

Table 2-11. Recommended Shaft and
Borehole Constraints

Medium	Constraint - NWTS Working Group on Far-Field Performance Constraints
All media--Bedded Salt Domed Salt Granite Basalt Tuff Shale	1) The shaft and its components undergo no fracturing due to thermo-mechanical stresses and no significant water leakage. 2) Deformation of shafts and shaft liners be small enough not to interfere with operations or require remedial repair work.
Basalt Bedded Salt Domed Salt	During postclosure the permeability of sealed shafts and boreholes be the same as the aquitards in zones where aquitards are intersected.
Granite Tuff	During postclosure the effective permeability of sealed shafts and boreholes be approximately the same as the host rock permeability for 70% of the vertical column between the repository and the surface.

Table 2-12. Recommended Maximum Surface
Temperature Rise

Medium	Maximum Temperature Rise, C
Bedded Salt	0.5 (Science Applications, Inc., 1976)
All Media--Bedded Salt Domed Salt Granite Basalt Tuff	< 4 within 3 meters of the surface (NWTS Working Group on Far-Field Performance Constraints)

This limit was derived from comparison to a natural analogue, the Marysville geothermal area. From limited environmental studies, it appears that the region has not suffered any environmental damage as a result of the higher heat flux there. Both recommended temperatures are well within the daily, seasonal, and geographical temperature variations in the United States.

Temperature rise in aquifers is also a concern because the flow velocity could conceivably carry the higher temperature water outside the repository area. Also, temperature rises and temperature gradients can influence ground-water flow patterns which could provide a transport mechanism for the radionuclides. Table 2-13 lists some recommended hydrologic constraints.

Science Applications, Inc. (1978), suggested a maximum temperature rise of 6 C in the aquifers. The NWTS Working Group on Far-Field Performance Constraints, in its draft on repository performance constraints in the far-field domain, approached the problem on an overall basis by determining two ways in which the ground-water regime could be perturbed by the thermal energy release of the emplaced wastes. First, convection cells could be established, giving rise to an increase in the flow velocity and/or the development of a vertical component to the flow. Second, the thermomechanical stresses could alter the effective permeability of the rock mass. Consequently, the NWTS Working Group recommended that (1) no significant thermally induced convection cells contact the salt formation (bedded or domed) in which a repository is located; and (2) for granite, basalt, and tuff the transit time for ground water to travel from the repository to the surface as a consequence of thermal convective forces be greater than 1,000 years.

Table 2-13. Recommended Hydrologic Constraints

Medium	Constraint
All media--Bedded Salt Domed Salt Granite Basalt Tuff Shale	6 C maximum temperature rise in aquifers (Science Applications, Inc., 1976)
Bedded Salt Domed Salt	No significant thermally induced convection cells contact the salt formation in which the repository is located (NWTS Working Group on Far-Field Performance Constraints)
Basalt Granite Tuff	The transit time for ground water to travel from the repository to the surface as a consequence of thermal convective forces is greater than 1,000 years (NWTS Working Group on Far-Field Performance Constraints)

Increased temperatures may also lead to chemical effects. Minerals may become unstable, and reactions may occur which reduce the volume or the radionuclide retention capability of the minerals. For nuclear waste disposal in salt the primary concern is the dissolution of the salt formation. For granitic rocks the micas can become unstable at as low as 350 C (NWTS Working Group on Far-Field Performance Constraints). Clay minerals, occurring as clay seams in salt strata and as secondary minerals in igneous rocks, can undergo dehydration and other changes at temperatures as low as 100 to 200 C. Table 2-14 lists some recommended media temperatures for geochemical stability.

Table 2-14. Recommended Maximum Media Temperatures
for Geochemical Stability

Medium	Maximum Temperature, C	
Granite	125 in the near field to 15% of repository depth	(NWTS Working Group on Far-Field Performance Constraints)
	100 outside the above region	
Bedded Salt	100 in the near field	(NWTS Working Group on Far-Field Performance Constraints)
Domed Salt	to 15% of	
Basalt	repository depth	
Tuff		
Shale	75 outside the above region	

Based on the above constraints, waste emplacement densities for a specified waste type and medium can be determined (see Table 2-15).

The following assumptions were made in calculating the emplacement densities:

- Repository is loaded simultaneously and instantaneously
- Effects of stress upon thermal properties were not included
- Presence of water was neglected in the thermal analysis
- Only simplified horizontal stratigraphies were assumed.

Table 2-15. Recommended Average Waste
Emplacement Density, kW/acre

Medium	Waste Type	Age of Waste at Emplacement, yr	Emplacement Density, kW/acre
Salt	Spent fuel	5	67 (U.S. Department of Energy, 1980)
		10	60 (Russell, 1977),
			40 (U.S. Department of Energy, 1980)
		50	20 (U.S. Department of Energy, 1980)
	High-level waste	50	54 (U.S. Department of Energy, 1980)
Shale	Spent fuel	50	42 (U.S. Department of Energy, 1980)

3 A HIERARCHY OF THERMAL CRITERIA

3.1 AN OVERVIEW

In order to understand the interrelationships among the various thermal criteria and to identify dominant criteria, a hierarchy of interaction was developed. The DOE/NWTS-33 series of documents (U.S. Department of Energy, 1981; 1982a, b, c) and the U.S. Nuclear Regulatory Commission (NRC) (1981) Proposed Rulemaking, 10 CFR Part 60, were used as a starting point in structuring the thermal criteria hierarchy. The DOE/NWTS-33 documents provide guidance to the National Waste Terminal Storage (NWTS) program in the form of "program objectives, functional requirements, performance criteria, and specifications intended to ensure that the program is consistent with national policy and that it results in the safe and environmentally acceptable disposal of radioactive waste". The NRC (1981) 10 CFR Part 60 specifies technical criteria for disposal of high-level radioactive wastes in geologic repositories.

The DOE/NWTS-33 criteria documents divide the mined geologic disposal system into three subsystems; the site, the repository, and the waste package. DOE/NWTS-33(1) addresses program objectives and criteria for the system as a whole. DOE/NWTS-33(2), (3), and (4) present the site, repository, and waste package performance criteria, respectively. The program objectives, as outlined in DOE/NWTS-33(1), are:

- Effective waste isolation
- Institutional and societal acceptability
- Technical conservatism
- Multiple regional repositories
- Waste accommodation
- Effective resource utilization, and
- Use of near-term technology.

The criteria presented in DOE/NWTS-33(1) cover the system and sub-system functions and system performance. These criteria are summarized below.

- The system must provide for waste receipt and emplacement and must provide containment and isolation adequate to ensure no releases occur that might result in unacceptable doses to the public.
- The site must provide natural barriers to contain and isolate the waste and assist in keeping man away from the waste.
- The repository must provide for waste handling, emplacement, and retrievability; and contribute to long-term containment and isolation by enhancing the natural system through engineered barriers, providing monitoring, and providing measures to protect against human intrusion.
- The waste package must provide waste containment for a specified period and a long-term barrier to waste release into the geologic environment beyond that period.
- The system must satisfy NRC and EPA requirements for public health and safety and preserve the quality of the environment.

Figure 3-1 shows the hierarchy of NWTS objectives, functional requirements, criteria, and specifications. Functional requirements establish the capabilities that the mined geologic disposal system must provide to achieve the program objectives. Performance criteria are designed to measure the performance of the disposal system and ensure that the functional requirements are met. Specifications will further define the performance for a particular site or design option.

As stated previously, the NRC's Proposed Rulemaking 10 CFR Part 60 specifies the disposal of high-level radioactive wastes in geologic repositories. It addresses the siting, design, and performance of a geologic repository, and the design and performance of the waste package. As with the DOE/NWTS-33 documents, the concept of multiple barriers is used to promote technical conservatism. The primary function of the waste package and underground facility is to control the release of radionuclides into the geologic environment. The present public draft criteria would require the engineered system to be designed so that the waste is contained within the waste package for the first 1,000 years following emplacement. After 1,000 years, the site must provide the primary barrier for the isolation of wastes from the environment. Figure 3-2 depicts a plausible pictorial representation of a geologic repository according to 10 CFR Part 60.

Both the DOE/NWTS-33 documents and the NRC's Proposed Rulemaking 10 CFR Part 60 follow the general guidelines of the Environmental Protection Agency (EPA). The current guidelines of the EPA call for the prompt disposal of radioactive wastes once adequate methods are available. Because they must be effective for so long, the EPA

FIGURE 3-1. HIERARCHY OF OBJECTIVES, REQUIREMENTS, CRITERIA, AND SPECIFICATIONS (U.S. DEPARTMENT OF ENERGY, 1981; 1982a, b, c)

FIGURE 3-2. A PLAUSIBLE HIERARCHICAL STRUCTURE OF THE GEOLOGIC REPOSITORY ACCORDING TO 10 CFR PART 60 (U.S. NUCLEAR REGULATORY COMMISSION, 1981)

requires disposal systems to offer as much protection as is reasonably achievable. Also, disposal systems should reduce the consequences of possible mistakes in selection, design, or construction by using several different types of barriers, both engineered and natural. Active institutional controls, while useful, should not be expected to last more than 100 years after disposal. The EPA guidelines are designed to cover radionuclide releases to the accessible environment for a period of 10,000 years after disposal.

To achieve the guidelines outlined in the DOE/NWTS-33 series of documents, the NRC's Proposed Rulemaking 10 CFR Part 60, and the EPA, an organized approach to analyzing the complex mined geologic disposal system is necessary. The draft document, "Nuclear Waste Disposal--A Description", which is still in preparation, is a first step toward realizing this need. It describes the development of a relevance tree for the nuclear waste disposal system. Relevance trees are simple graphical representations between a basic function and its sub-functions. These hierarchical structures are constructed by placing the highest order functions at the top of the tree and all visualized ways of accomplishing this basic function at the next level of the tree.

In the actual development of a relevance tree for the nuclear waste disposal systems, the draft document "Nuclear Waste Disposal--A Description" found it more meaningful to display the results in terms of well-known subsystems concepts. Nevertheless, the hierarchy was developed and arranged according to a logical decomposition of the system's functional requirements. Figure 3-3 shows how the primary

FIGURE 3-3. CONCEPTUAL STRUCTURE OF THE NUCLEAR WASTE DISPOSAL SYSTEM

objective of the NWTS program, the long-term isolation of nuclear waste from the biosphere, can be broken down into the more specific functional requirements that must be provided by the subsystems of the nuclear waste disposal system.

In the draft paper, "Nuclear Waste Disposal--A Description", functional requirements are seen as simple statements of what the subsystem must do. They are derived from the next higher level system by specifying how the higher order function can be accomplished. A specific variable or parameter, known as a performance measure, is then used as an indication of how well the function is being performed. Closely tied to performance measures are performance criteria which explicitly define how well a subsystem must perform.

3.2 DESCRIPTION OF THE THERMAL CRITERIA HIERARCHY

A system is defined as a functionally related group of elements, whereas a hierarchy is a body of entities arranged in a graded series. Thus, for a system at any level in a hierarchy to function properly, all lower levels must function properly. Also, the functions required by a higher order system will appear as conditions or constraints on operation of lower order systems. Different system configurations at any level will also influence the functions which the lower level must provide. Therefore, the hierarchy will become more and more specialized as different design options are chosen at each level.

As can be seen from the DOE/NWTS-33 documents, the NRC's Proposed Rulemaking 10 CFR Part 60, the EPA guidelines, and the "Nuclear Waste Disposal--A Description" draft working paper, there are several

plausible hierarchical structures for a geologic repository. This situation exists because of the inconsistent use of many terms when discussing geologic disposal of radioactive wastes and especially when talking about system descriptions and hierarchies. To avoid any ambiguities in this paper a glossary of terms has been included in Appendix A. Some terms, however, are critical to understanding the following sections. They and their definitions follow:

- hierarchy - a body of entities arranged in a graded series
- system - a functionally related group of elements
- function - an assigned duty or activity
- criterion - a standard or rule on which a judgment or decision can be based
- constraint - something that restricts, limits, or regulates
- performance - the way in which something functions.

In order to determine the appropriate thermal criteria for the geologic disposal of radioactive waste, the impact of their thermal energy release on the disposal system must be evaluated. But to evaluate the possible problems, the geologic disposal system itself must be understood. Figure 3-4 defines the broad aspects of a mined geologic repository. The tasks of receiving, inspecting, and emplacing the waste are not included.

The first element in the hierarchy is the environmental system. This primary system includes man, his current and future activities, and the biosphere of which he is a part. It is this system which the mined geologic disposal system is designed to protect, although the actual linkage between the two is unclear, as represented by the dashed line.

The mined geologic disposal system is a passive system with respect to the environmental system. The environmental system is controlled, to a limited extent, through institutional controls. These, however, will lose their effectiveness with time. The mined geologic disposal system, however, does have a direct influence on the lower systems in the hierarchy. By defining the geologic repository, constraints are immediately placed on any lower levels. This is true for any level in the hierarchy.

At each level in the hierarchy the lower systems are designed to allow the higher order systems to successfully complete their functions. Of necessity this responsibility to the higher order systems specifies the functions the lower order systems must perform. Figure 3-5 shows the nuclear disposal system in terms of the functions each element must fulfill.

The environmental system's primary function is to assure public health and safety by isolating existing and future nuclear waste, derived from both military and civilian activities, from the biosphere. The currently preferred method of accomplishing this goal is the concept of a mined geologic repository. As a result, the primary function of the mined geologic disposal system is the long-term isolation and containment of radioactive wastes in mined geologic repositories. The functions of all lower levels are thus generally defined, with the definition becoming more narrowed with each succeeding level.

As can be seen from Figures 3-4 and 3-5 the mined geologic disposal system can be divided into three subsystems: the waste package system, the repository system, and the site system. The waste package

FIGURE 3-4. A HIERARCHY OF A MINED GEOLOGIC DISPOSAL SYSTEM

*Not all repository subsystems are shown.

FIGURE 3-5. A HIERARCHY OF FUNCTIONS FOR A MINED GEOLOGIC DISPOSAL SYSTEM

*Not all repository subsystems are shown.

system consists of the waste form, the canister, any overpack materials, and backfills. Its main functions are to contain the wastes for a specified period of time, to provide for the slow release of the waste after the containment period, and if necessary to provide for the retrieval of the wastes. The repository system consists of the shafts, and the rooms and pillars. Its functions include emplacing and if necessary retrieving the wastes, and maintaining safe, comfortable working conditions for onsite personnel. The site system consists of the geologic formation and surrounding rock, the hydrologic regime, and the geochemical regime. Its primary function is to provide a site suitable for the long-term isolation of radioactive wastes, and whose physical, chemical, and hydrologic properties will act to retard the migration of radionuclides.

The following sections, 3.2.1, 3.2.2, and 3.2.3, will discuss in more detail the waste package system, the repository system, and the site system, respectively. These sections will explore the effects of the thermal energy release and suggest some possible thermal criteria along with the supporting reasons.

3.2.1 Waste Package System

The waste package system has been defined as an engineered system consisting of the waste form, its container, and any other engineered components separating the waste form from the geologic formation. As a system, the waste package has three functions: contain the waste within the package for a specified period of time, provide for the slow release of the wastes after the containment period, and provide for the retrieval of the wastes.

The requirement for containment for a specified period of time is based on the decay characteristics of the wastes. This period is a function of the time the heat generated by the wastes is dominated by the decay of relatively short-lived fission and activation products. During this period the toxicity of the fission products decreases by several orders of magnitude thereby mitigating, to some extent, the danger to man if the nuclides were to be transported to the biosphere. The requirement for retrievability arises from the recognition that some degree of uncertainty may exist in defining and implementing long-term isolation of radioactive wastes in mined geologic repositories. Currently proposed regulations would require, if adopted, a containment period of 1000 years with a limited release for the following 10,000 years at a nominal rate of less than one part in 10,000 per year and a retrieval option including the operational period plus 50 years (U.S. Nuclear Regulatory Commission, 1981).

The heat generated by the encapsulated wastes can endanger the effectiveness of the waste form, the canister, and any other engineered components as isolation barriers. To prevent this, the containment materials and the waste form should be maintained at sufficiently low temperatures to retard those effects which deteriorate the isolation barriers provided by the waste package. The waste package system can be divided into two lower order systems to facilitate the determination of these temperature limits. These two systems are the waste form system and the containment system. The waste form system consists of the radioactive waste materials and any encapsulating or stabilizing materials placed inside the canister. The containment system consists

of all the engineered components of the waste package, excluding the waste form.

3.2.1.1 Waste Form System

The waste form system is represented as a hierarchy in Figure 3-6. (The following discussion will refer to Figure 3-6.) The function of the waste form system is to immobilize the radionuclides by containment as part of a solid. Several waste forms have been proposed, but with few exceptions, all fall into one of three categories: spent fuel, glass, or tailored ceramics. Although an acceptable geologic disposal system can probably be designed for any given waste form, the choice of waste form will affect the complexity, cost, feasibility, and size of a geologic repository.

Maximum allowable waste form temperatures will be different for the various waste forms. Maximum waste form temperatures are a function of the radionuclide content and specific activity of the package, the chemical composition and form of the waste, the package geometry, the characteristics of the engineered components of the package, the effective thermal conductivity of the geologic medium, and the characteristics of the heat sink. After emplacement of the waste package in the repository, the most likely vehicle for radionuclide release from the waste form into or through the package system and into the surrounding geology is intruding ground water. Waste form leaching, Item 2 in Figure 3.6, is therefore of primary concern when addressing waste mobilization. Before leaching by ground water can occur, physical and/or chemical reactions must occur to breach the engineered

barriers of the containment system. These failure mechanisms will be discussed in the following section on containment systems.

The next level in the hierarchy identifies the primary release mechanisms of radionuclides. These release mechanisms occur with increased temperatures and will tend to decrease the leach resistance of the waste form and thereby increase the potential mobility of the waste. The failure mechanisms can be grouped into four general categories: alteration of the original chemical composition of the waste form, combustion of the waste form, volatilization of the waste form, and the loss of physical integrity of the waste form.

Item 3 of Figure 3-6 requires the maintenance of the original chemical composition and form of the waste form. Chemical composition is the complete description of the chemical compounds and their relative abundance which make up each waste form. The form describes how the waste has been immobilized, such as glass, spent fuel rods, or ceramics, to name a few. There are several possible failure mechanisms for the various waste forms in this category.

For spent fuel, there are two possible failure mechanisms under the chemical composition and form element in the waste form system hierarchy, as shown in Table 3-1 (Figure 3-6). The first failure mechanism is the oxidation of the ceramic fuel from UO_2 to U_3O_8 with increased temperatures. This reaction has been shown to occur at 250 C (Cordfunke, 1969). With the conversion of UO_2 to U_3O_8 there is a large volume expansion which could crumble the uranium oxide to powder and split the Zircaloy cladding of the spent fuel. The presence of oxygen, however, which is required for this reaction to occur is not

FIGURE 3-6. HIERARCHY OF WASTE FORM LIMITS

expected to be abundant in the waste package. A maximum temperature of 250 C is recommended to prevent this type of failure (Cordfunke, 1969).

The second failure mechanism is the degradation of the protective oxide film on the Zircaloy cladding of spent fuel in the absence of an oxidant at elevated temperatures. This reaction has been reported to occur at 200 C (Jenks, 1977). Pending additional work, Jenks has recommended a maximum of 200 C to prevent this type of failure.

Table 3-1 (Figure 3-6) also shows the possible failure mechanisms for borosilicate high-level waste through changes in the chemical composition and form. Both failure mechanisms are directly related to the temperature rise experienced by the waste glass. The transition temperature for borosilicate glass is 500 C, while the softening temperature occurs at a slightly higher temperature (Jenks, 1977). Borosilicate glass has been reported to flow easily at 900 C (NWTs Working Group on Far-Field Performance Constraints in its draft on repository performance constraints in the far-field domain). At these temperatures there is concern that the migration of heavy, separable phases might occur. To prevent these failure modes from occurring a maximum temperature of 500 C is suggested for borosilicate glass high-level waste (Jenks, 1977).

The two waste forms for transuranic waste which have a potential for failure in their chemical composition or form are thermosetting resins and urea-formaldehyde resins. Both fail through decomposition of their materials (Cohen et al, 1977, and Gilmore, 1977). A maximum temperature of 300 C is, therefore, recommended for both thermosetting resins and urea-formaldehyde resins (see Table 3-1 of Figure 3-6) (Miller and Wurm, 1980).

The last waste type in Table 3-1 is cladding and hardware wastes. Compressed cladding and hardware wastes and fused cladding and hardware wastes are the two waste forms for this waste type. Both fail in the same manner, degradation of the protective oxide film in the Zircaloy cladding leading to an increase in surface reaction rates (Jenks, 1977). The fused waste form can tolerate higher temperatures, however, because of its smaller surface area-to-volume ratio. Based on this failure mechanism, maximum temperatures of 200 C and 400 C are recommended for the compressed and fused waste forms, respectively.

Item 4 of the waste form hierarchy (Figure 3-6) requires the prevention of the combustion of the waste form. The temperature at which combustion occurs is a function of particle size, surface area, and chemical composition and form. Once a fire has started, it promotes the mobility of the waste. Of the various waste forms, transuranic wastes are composed, either partly or entirely, of combustible materials.

Item 5 of Figure 3-6 requires the prevention of the volatilization of the waste form. This failure mechanism is related to the combustion failure mechanism in that the gases generated through volatilization of the waste form could experience spontaneous ignition. Thus, to avoid an explosive concentration of gases, it is necessary to restrict the maximum temperature to less than that required to volatilize any component of the waste form. At a temperature of 750 C, calcine commercial high-level waste has been reported to experience significant cesium volatility (Jenks, 1977). For this waste form, therefore, a maximum temperature of 700 C has been recommended by Jenks (see Table 3-2 of Figure 3-6).

Item 6 in the waste form hierarchy (Figure 3-6) requires the maintenance of the physical integrity of the waste form. As with the first element in the level, chemical composition and form, there are several failure mechanisms in this category, all of which are dependent on the waste type and waste form. Table 3-3 (Figure 3-6) lists some of the possible failure mechanisms as a function of waste type and form.

For spent fuel, there are three primary failure mechanisms under the physical integrity element in the waste form hierarchy. The first is the rupture of the clad by internal pressure (Blackburn et al, 1978). Fuel rod design for reactor service limits pressure-induced stress to less than $2/3$ the material yield stress or $1/2$ the material ultimate strength, whichever is lesser. Since the principal loading of fuel rod cladding comes from internal pressure, the relationship between this biaxial loading condition and the uniaxial loading conditions used to determine the material yield stress and the material ultimate strength must be established. Blackburn et al (1978) indicate that ratios of 1.1 and 1.0 are appropriate for biaxial to uniaxial yield stress and biaxial to uniaxial ultimate strength, respectively. Thus, service hoop stress level should be limited to $(2/3 \times 1.1)$ the material yield stress or $1/2$ the material ultimate strength, whichever is smaller. From this, Blackburn et al (1978) have shown that the maximum allowable temperature for PWR spent fuel is 396 C and >427 C for BWR spent fuel (see Figure 3-7).

The second failure mechanism for spent fuel is the stress-rupture (Sr) of the clad. Spent fuel rods stored at elevated temperatures are potentially susceptible to failure by stress-rupture since the internal pressure in the rod produces a tensile hoop stress in the cladding

FIGURE 3-7. MAXIMUM ALLOWABLE TEMPERATURE TO AVOID RUPTURE OF FUEL ROD CLAD BY INTERNAL PRESSURE (BLACKBURN ET AL, 1978)

wall. Failure by stress-rupture is guarded against by restricting design stress levels to less than $2/3$ the minimum stress to produce rupture in some time, t . Using curves of $2/3 S_r$ versus temperature, various storage conditions, and the design stress level curve, the diagram in Figure 3-8 can be constructed. From the points of intersection the maximum allowable cladding temperature for isothermal storage can be determined. Isothermal conditions, however, are not expected in dry storage of spent fuel because the decay heat varies over the storage life. The variation of decay heat and cladding temperature for a PWR assembly is shown in Figure 3-9. By using a creep damage rule the effect of varying the stress and temperature utilizing the isothermal stress-rupture data can be assessed.

Figure 3-10 shows the accumulation of creep damage for various initial cladding temperatures. As the temperature decreases during storage, the rate of damage accumulation slows, so that indefinite storage is possible at a starting temperature of 380 C (Blackburn et al, 1978).

Stress corrosion cracking is the third failure mechanism for spent fuel. The maximum allowable temperature for this failure mechanism is dependent on the initial flaw size. For flaw sizes that may be representative of as-fabricated cladding temperature, limits of 350 C and 340 C are obtained (Blackburn et al, 1978). The two crack sizes used were a 8μ crack depth with a crack depth to crack length ratio 0.9 and a 8μ crack depth with a crack depth to crack length ratio of 0.2, respectively. Cladding which has experienced pellet-cladding interactions during reactor service may have cracks extending halfway through the wall with an associated temperature limit of 250 C.

FIGURE 3-8. MAXIMUM ALLOWABLE TEMPERATURE FOR SPENT FUEL STORAGE AT CONSTANT TEMPERATURE (BLACKBURN ET AL, 1978)

FIGURE 3-9. DECAY HEAT AND FUEL CLADDING TEMPERATURE AS A FUNCTION OF STORAGE TIME (BLACKBURN ET AL, 1978)

FIGURE 3-10. ACCUMULATION OF CREEP DAMAGE FOR VARIOUS STARTING TEMPERATURES (BLACKBURN ET AL, 1978)

Based on this information, a maximum temperature of 300 C has been recommended to avoid stress-corrosion cracking failure (U.S. Department of Energy, 1980).

There are two possible ways in which borosilicate glass high-level waste can lose its physical integrity. These are devitrification and cracking of the waste glass. Devitrification is the transition of the molecular structure of glass from the amorphous to a crystalline state. The basic concern with this change is that devitrified glass will have a higher leach rate than glass which has not devitrified. A devitrification temperature of 500 C has been reported for borosilicate high-level waste glass (Jenks, 1977).

A more recent study, however, has suggested the maximum temperature for waste glass should be set at 375 C to avoid devitrification (Turcotte and Wald, 1978). The lower limit, 375 C, recommended when considerable devitrification in a zinc borosilicate waste glass, was observed to occur at times less than one year at a temperature of 500 C. As the rate of devitrification tends to be highest at the center of the waste glass where the highest temperatures are found, and lowest at the glass surface where leaching takes place, a maximum centerline temperature of 500 C with a maximum surface temperature of 375 C is recommended to avoid devitrification (Jenks, 1977; Turcotte and Wald, 1978).

As mentioned previously, the second way for borosilicate high-level waste glass to lose its physical integrity is through cracking of the glass at elevated temperatures. Significant increases in cracking and, therefore, in leach rates have been observed in test glasses heated for a few months in the range 500 C to 800 C (Jenks, 1977).

Based on these tests a maximum temperature of 500 C is recommended to avoid cracking of the waste glass.

Another waste form which can lose its physical integrity at elevated temperatures is cement transuranic waste. At elevated temperatures, cement has been shown to experience dehydration and thermal stresses which lead to microcracking and reduction in structural properties (American Society for Testing and Materials, 1978; Lokken, 1978). A maximum temperature of 150 C is, therefore, recommended to avoid this form of failure.

3.2.1.2 Containment System

The containment system is represented as a hierarchy in Figure 3-11. (The following discussion will refer to Figure 3-11.) The function of the containment system is to contain the wastes for a specified period of time, to control the release of radionuclides after loss of containment, and to provide for the retrievability of the waste. Several engineered barriers actually make up the containment system. These include the canister, the overpack, the hole liner, and the backfill. The canister is the primary physical barrier to water intrusion following the thermal period. The overpack acts as a barrier to the intrusion of ground water through the thermal period. Both the overpack and the hole liner provide a means for retrieving the waste should it become necessary or desirable. The backfill is the primary chemical barrier to radionuclide migration. It also acts to inhibit the access of water to the canister.

The second level in the containment system hierarchy lists four general ways the containment system can fail. These are the loss of

FIGURE 3-11. HIERARCHY OF CONTAINMENT SYSTEM LIMITS

physical or chemical integrity, corrosion, combustion, and internal pressurization.

Item 2 of Figure 3-11, loss of physical or chemical integrity is a rather broad failure mechanism. Generally stated, the containment system temperatures must be limited to avoid undesirable chemical or physical reactions in the containment system, between the waste form system and the containment system, or between the containment system and the emplacement environment. Tables 3-4 and 3-5 (Figure 3-11) give some possible failure mechanisms in this class for the canister and the backfill.

At elevated temperatures it is possible for the canister to lose its physical integrity. When 304L stainless steel is heated to temperatures greater than 375 C, structural changes occur which lead to an increased susceptibility in the alloy to stress cracking on subsequent exposure to water (Jenks, 1977). Because of the effect, Jenks has recommended a maximum temperature of 375 C for 304L stainless steel. It has also been suggested that, while carbon steel can probably tolerate higher temperatures than 304L stainless steel, its maximum temperature also be set at 375 C (Jenks, 1977).

Swedish scientists have expressed concern that pure bentonite backfill will not retain its physical and chemical integrity in a repository environment at temperatures greater than 100 C (Ahlstrom, 1979). Other sources, however, report that bentonite retains its properties to at least 300 C (Miller and Wurm, 1980). The maximum allowable temperature for bentonite could, therefore, be as low as 100 C or as high as 300 C (see Table 3-5 of Figure 3-11).

Another possible backfill material is zeolite. At temperatures well below 300 C, however, zeolites tend to lose their water and, therefore, their cation exchange capability (Miller and Wurm, 1980). Based on this, a maximum temperature no higher than 300 C is recommended for zeolites (Miller and Wurm, 1980).

Since corrosion of the containment system is such an important failure mechanism for the containment system, it is included in the containment system hierarchy as a separate entry instead of being included in the physical and chemical stability entry. Corrosion of the containment system can be initiated by either the waste form or the elements of the emplacement environment. The maximum allowable temperature for the containment system based on corrosion failure will vary with the waste form, the containment material, and the medium.

High-temperature corrosion by the waste on the canister depends on the materials involved. For a calcine waste form in a steel canister, not much is known. Cesium oxide corrosion of 304L stainless steel indicates that intergranular corrosion occurs in a linear fashion with time (Westinghouse, 1983). Maximum temperatures of 800 C and 100 C have been recommended for 1020 carbon steel and 304L stainless steel, respectively (Westinghouse, 1983).

Corrosion of the canister by ground water is also important. Ground water contacting the canister will have high ionic strength, low pH, and will probably contain very low starting concentrations of the container materials (Ad Hoc Panel of Earth Scientists, 1978). These conditions impose a very severe corrosive environment on the canister. And, as the surface temperature of the canister increases so will the corrosion rate of the canister.

Data for the influence of temperature on the rate of corrosion for conditions of interest, however, are lacking. Rough estimates of how corrosion rates are influenced by temperature can be seen in Figures 3-12 and 3-13. Braithwaite and Molecke (1979) have shown that the alloys Hastelloy C-276, Inconel 625 and 600, Incoloy 825, and several titanium-base alloys are resistant to corrosion by oxygen-free brine solutions to 250 C. A corrosion rate of 0.7 mils/year for 304L SS exposed to deoxygenated brine at 250 C has been measured. Under the same conditions, however, low carbon steel has shown a corrosion rate of 68 mils/year, which is considered unacceptably high in brine (Rockwell International, 1979).

The third possible canister breach mechanism is combustion. In general, the temperatures of the containment system should be kept below the temperature required for combustion to prevent the possible dispersal of the waste. The temperature at which combustion will occur is a function of the surface area and the material properties of the containment system.

The last possible canister breach mechanism is the rupture of the canister by internal pressure. This failure mechanism is directly related to the encapsulated waste form. Volatilization of the waste form must be prevented so that overpressurization of the containment system, especially the canister, does not occur. Calcine high-level waste is a potential instigator of this failure mechanism, as it experiences significant cesium volatility at temperatures greater than 750 C (Jenks, 1977) (see Table 3-6 of Figure 3-11).

As can be seen in Figures 3-6 and 3-11, some of the failure mechanisms identified are lacking thermal-thermomechanical limits.

FIGURE 3-12. THE ESTIMATED TEMPERATURE DEPENDENCY OF NICKEL ALLOY CORROSION IN OXIC BRINE AND ANOXIC SEAWATER (WADDELL ET AL, 1981)

FIGURE 3-13. TEMPERATURE DEPENDENCY OF COMMERCIALLY PURE TITANIUM CORROSION IN OXIC BRINE AND ANOXIC SEAWATER (WESTINGHOUSE, 1983)

Since these limits are not available in the literature, work should be undertaken to determine them. Only when this is done will the thermal criteria hierarchy be applicable to all possible repository designs.

3.2.2 Repository System

The repository system has been defined as the conduit which allows the waste packages to be emplaced deep within the geologic formation. The functions of this system are to provide for (1) the operational safety and comfort of onsite personnel, (2) emplacing the wastes, and (3) retrieving the wastes should it become necessary or desirable. The heat generated by the encapsulated wastes can endanger the ability of the repository to perform its functions. Repository temperatures should, therefore, be maintained at sufficiently low levels to retard those effects which are detrimental to the repository functions.

The repository system can be further subdivided into the waste handling system and the sealing system. The physical components of the waste handling system are the shafts and the rooms and pillars. The sealing system is an engineered system, consisting of the shaft and borehole seals and the backfills, designed to maintain waste isolation after repository closure.

3.2.2.1 Waste Handling System

Figure 3-14 shows a hierarchy representation of the waste handling system. (The following discussion will refer to Figure 3-14.) The functions of this system are to provide access from the surface to the repository by means of shafts, to provide access to the emplacement

FIGURE 3-14. HIERARCHY OF WASTE HANDLING SYSTEM LIMITS

rooms, to provide a safe, comfortable working environment for personnel and equipment, and to provide for the retrieval of the wastes. The waste handling system dominates the repository system during the operational phase of the repository.

As can be seen from Figure 3-14, the primary source of unsafe working conditions is excessive temperatures. Temperatures in the storage rooms are affected by the volume of rock mined and depend on the heat capacity and density of the host rock. The radiative heat transfer properties of storage room surfaces also influence the room temperatures. To assure a safe working environment, in terms of both air and surface temperatures, a maximum working temperature of 50 C has been recommended for retrieval operations in open rooms (Rockwell International, 1979). For retrieval operations in rooms from which backfill has been removed, a temperature of 28 C is recommended (Rockwell International, 1979) (see Table 3-7 of Figure 3-14).

The surface temperatures that the personnel, equipment, and materials come in contact with are also important, in their own right, as they can lead to elevated air temperatures. Figure 3-15 shows how the floor temperature of a repository in salt for high-level waste in stainless steel canisters at a 155 kW/acre thermal loading will vary over time. The initial temperature at the mine level is 93 F. Both the air temperature and the wall and floor temperatures can be reduced with mine ventilation (see Figure 3-15). A maximum rock temperature of 49 C has been suggested for the safe operation of retrieval equipment by the Mine Design Working Group of Sandia National Laboratories in the draft activity work package on blast cooling and continuous cooling (see Table 3-8 of Figure 3-14). The capability of a repository to

FIGURE 3-15. COMPARISON OF PEAK FLOOR TEMPERATURES
(RUSSELL, 1977)

handle the wastes can be seriously jeopardized if the shafts from the surface to the underground emplacement area suffer damage as a result of the thermal energy release of the wastes. For all the candidate media, it has been recommended that any deformations of the shafts and shaft liners be sufficiently small so that routine shaft operations are not impeded (NWTs Working Group on Far-Field Performance Constraints in the draft on repository performance constraints in the far-field domain). Potentially, shaft deformations will be much greater for salt repositories than for other media.

In terms of waste isolation, the loss of shaft integrity is a serious concern. Should water leakage occur, a primary pathway for radionuclides to the surface would be created. To prevent this failure mechanism it has been recommended that, in any rock type, the shaft and its components undergo no fracturing due to the thermomechanical stresses induced in the repository and have no significant water leakage (NWTs Working Group on Far-Field Performance Constraints). The prevention of significant water leakage through the shaft is an important consideration.

Another critical element in the waste handling system is the stability of the rooms. The stability of rooms in hard, brittle rocks, such as granite or basalt, is felt to be assured if the stresses in the rock around the room opening are limited so that even the possibility of minor instabilities is eliminated (Dames and Moore, 1978). In shale, a weaker, more ductile rock, it may not be possible to construct rooms with the required degree of stability without continuous engineered support. Provided the effect of temperature on rock mass strength and stress-strain behavior is taken into account, the

strength-to-stress ratio should be, at the minimum, 2 within 1.5 meters of the room opening, whereas for shale the minimum strength-to-stress ratio is 1 within 1.5 meters of the room opening with continuous engineered support (Dames and Moore, 1978).

The measure of room stability in salt is the percentage of room closure at the end of the retrievability period. This parameter is used, instead of the strength-to-stress ratio, since salt tends to creep at elevated temperatures. As with granite, basalt, and shale, this performance measure is directly related to the stress and mine-geometry parameters of the repository. A maximum closure of 10 to 15 percent is recommended for bedded and domed salt in order to maintain access to the rooms throughout the retrievability period (Ad Hoc Panel of Earth Scientists). Closure constraints, however, neglect the fact that salt can undergo brittle failure.

Related to the room stability is the integrity of the associated pillars. Based on past experience at zero thermal loading, pillar stability is considered assured if the average stress across the mid-height of the pillar is less than one-half of the uniaxial compressive strength of the pillar (Dames and Moore, 1978). Provided the effect of temperature in rock mass strength and stress-strain behavior is taken into account, the strength-to-stress ratios shown in Table 3-11 (Figure 3-14) are recommended for pillar stability. For granite, basalt, and shale a minimum strength-to-stress ratio of 2 across the mid-height of the pillar is suggested (Dames and Moore, 1978).

The last element in the waste handling hierarchy is the loss of retrievability of the waste packages. To assure retrievability of the

canister, extensive rock fracturing around the canister cannot be allowed. Small pieces of fractured rock could jam the canister. The maximum rock temperatures shown in Table 3-12 (Figure 3-14) are recommended to prevent this. In the case of salt, both bedded and domed, decrepitation tends to occur at temperatures greater than 250 C. For shale, 250 C may also be reasonable. In the hard rocks, granite and basalt, 350 C may be acceptable (U.S. Department of Energy, 1979).

3.2.2.2 Sealing System

A hierarchy representation of the sealing system is shown in Figure 3-16. (The following discussion will refer to Figure 3-16.) The sealing system consists of the shaft and boreholes seals, and backfill for the rooms, corridors, and underground areas. The primary function of the sealing system is to provide for the long-term integrity of the shaft and borehole seals, and backfills with respect to ground water leakage.

There are two ways in which the sealing system can fail. These are the loss of physical integrity and the loss of hydrologic integrity. To prevent the first failure mechanism, loss of physical integrity, it is recommended that the shaft and its components undergo no adverse structural changes such as fracturing due to the elevated temperatures and the resulting thermomechanical stresses (NWTS Working Group on Far-Field Performance Constraints in the draft repository performance constraints in the far-field domain). The second failure mechanism can be prevented by requiring sufficiently low temperatures so that adverse changes in the permeability of the shaft and borehole

FIGURE 3-16. HIERARCHY OF SEALING SYSTEM LIMITS

seals, and the backfills, do not occur. For basalt and bedded and domal salt, it is recommended that the permeability of the sealed boreholes and shafts be approximately the same as the aquitards in those zones where aquitards are intersected. In granite and tuff, it is recommended that the effective permeability of the sealed shaft and boreholes be approximately the same as the host rock permeability for the middle 70 percent of their length (NWTS Working Group).

It is apparent from Figures 3-14 and 3-16 that some of the failure mechanisms identified by the waste handling and sealing systems are lacking thermal-thermomechanical limits. This is due to the absence of such limits in the literature. For the thermal criteria hierarchy to realize its full potential these limits must be determined.

3.2.3 Site System

The site system consists of the geologic, hydrologic, and geochemical systems that provide natural isolation barriers to the migration of radioactive wastes. As a system, the site has three functions:

(1) provide a site suitable to the long-term isolation of the radioactive wastes; (2) provide a site whose physical, hydrologic, and chemical properties will act to retard the migration of radionuclides; and (3) provide for the retrieval of the wastes, should that be necessary. The heat generated by the emplaced wastes could endanger the ability of the site system to provide natural isolation barriers to radionuclide migration. The site system should therefore be maintained at sufficiently low temperatures to retard those effects which deteriorate the isolation barriers of the site system.

The site system can be further divided into three lower order systems to facilitate the determination of maximum temperatures or temperature-related limits. These three systems are the geologic system, the hydrologic system, and the geochemical system. The geologic system is composed of any acceptable igneous, sedimentary, or metamorphic formation, and its surrounding stratigraphy into which the radioactive waste is emplaced. The hydrologic system is defined as any soil, rock unit, or zone which has a distinct influence on the storage or movement of ground water by virtue of its porosity or permeability. The geochemical system consists of any interaction between the waste, rock, or ground water within the geologic formation. It is the geochemical system which binds the geologic and hydrologic systems together.

3.2.3.1 Geologic System

An ordered representation of the geologic system is shown in Figure 3-17. (The following discussion will refer to Figure 3-17.) Its functions include providing an environment suitable for the long-term isolation of radioactive wastes, providing an environment which will retard the migration of radionuclides, and providing for the retrievability of the wastes. To accomplish its functions the geologic system should be maintained at sufficiently low temperatures to retard those effects which are detrimental to its functions. The second level in the geologic system hierarchy lists the ways in which the geologic system could fail to perform its functions.

The first means by which the geologic system can fail is to lose its physical integrity. Excavation of a repository and the release of

FIGURE 3-17. HIERARCHY OF GEOLOGIC SYSTEM LIMITS

thermal energy by the emplaced wastes causes a change in the stresses in the rock mass. These are generally compressive in nature through most of the rock mass, therefore leading to a potential for shear failure in those regions where the rock mass strength is exceeded. The potential for shear failure is greatest near the repository and diminishes with distance. Minimum strength-to-stress ratios, shown in Table 3-13 (Figure 3-17), are recommended to prevent shear failure in the rock mass.

For basalt, the minimum strength-to-stress ratio everywhere in the far-field region should be greater than one (NWTS Working Group on Far-Field Performance Constraints in its draft on repository performance constraints in the far-field domain). In granite and tuff the minimum strength-to-stress ratio should remain greater than one in the middle 70 percent of the rock column lying between the repository horizon and the ground surface (NWTS Working Group). The analysis of fracturing is more complex for salt than for the other media since the nonlinear plastic creep of salt and other sedimentary strata must also be taken into account. A minimum strength-to-stress ratio of one is recommended for the aquitards and the caprock of bedded and domed salt, respectively (NWTS Working Group).

As the thermal energy of the radioactive waste is released, the rock mass will vertically deform upward because of thermal expansion. As this thermal energy is exhausted, the rock mass will subside. While surface ground displacements will not form a breach of the natural geologic barrier, it may be of sufficient magnitude to be environmentally unacceptable. Extensive vertical deformation could cause changes in runoff patterns or distress surface structures. The maximum vertical

displacement for any medium is recommended to be less than 3 meters (see Table 3-14 of Figure 3-17) (NWTs Working Group). Due to room closure, after repository decommissioning, the maximum subsidence will probably occur earlier than maximum uplift.

A failure mechanism directly related to vertical displacement and fracturing is the creation of a perturbed fissure zone (NWTs Working Group). When the rock mass bulges upwards due to thermal expansion, horizontal tensile stresses may be created in the upper portion of the rock mass. Since most geologic materials have very low tensile strengths, there is a potential for creating new vertical fractures or for opening existing ones. For all rock types, it is recommended that the perturbed fissure zone should not extend downward from the surface more than 15 percent of the repository depth (see Table 3-15 of Figure 3-17) (NWTs Working Group).

The last mode of failure for the geologic system is excessive ground surface temperature rise. While this will not directly alter the performance of the geologic system, it may be environmentally unacceptable. A maximum temperature rise of less than 4 C within three meters of the surface is recommended (NWTs Working Group). A depth of 3 meters was selected to include most, but not all, biological systems such as roots, burrowing animals, and insects.

3.2.3.2 Hydrologic System

A representation of the hydrologic system as a hierarchy is shown in Figure 3-18. (The following discussion will refer to Figure 3-18.) The basic functions of the hydrologic system are to provide an environment which will prevent or retard the migration of radionuclides and to

FIGURE 3-18. HIERARCHY OF HYDROLOGIC SYSTEM LIMITS

prevent the introduction of ground water into the repository. The hydrologic system should, therefore, be maintained at sufficiently low temperatures to retard those effects which deteriorate its effectiveness as an isolation barrier.

The creation of convection cells is a primary means of failure for the hydrologic system. The thermal energy released by the emplaced wastes could cause the creation of convection cells, giving rise to an increase in the flow velocity and/or the development of a vertical component to the flow (NWTs Working Group on Far-Field Performance Constraints in its draft on repository performance constraints in the far-field domain). For repositories in salt there is the additional concern that the convection cells might cause the dissolution of the formation. It is therefore recommended that the transit time for ground water to travel from the repository to the ground surface as a consequence of thermal convective forces be greater than 1,000 years for basalt, granite, and tuff (NWTs Working Group). For repositories in bedded or domed salt, it is recommended that no significant thermally induced convection cells contact the salt formation (NWTs Working Group).

The migration of ground water towards a heat source is an important failure mechanism in bedded salt. Bedded salt usually contains a small number of brine inclusions. All fluid brine-filled cavities have been shown to migrate up any existing temperature gradients. This migration of brine results from an increase in the solubility of the salt with increased temperature and from thermal effects within the brine. The rate of brine migration will vary with temperature, as seen

in Figure 3-19. The quantity of brine and its arrival rate are important parameters for bedded salt.

Dissolution of the waste form with the subsequent release of radionuclides can occur if the waste form comes in contact with ground water at the high temperature and pressure conditions of a geologic repository. Most waste forms are susceptible to hydrothermal reactions at temperatures above 100 C (Cohen et al, 1977). The extent of any waste form dissolution will be significantly influenced by the chemical composition of the ground water, particularly its pH and the oxidizing or reducing capacity of the repository environment. In spent fuel, the most leachable elements under oxidizing conditions appear to be cesium, rubidium, molybdenum, and strontium (Rockwell International, 1979). For borosilicate glass, the most leachable elements, under oxidizing conditions, appear to be sodium, silicon, molybdenum, uranium, and cesium (Rockwell International, 1979).

While excessive temperature increases in neighboring aquifers will not cause a breach in the geologic isolation barriers they may be environmentally unacceptable. Temperature rises in aquifers could result in higher temperature water being carried outside the repository area. This in turn could cause adverse effects in the biosphere. A maximum temperature rise of less than 6 C has been recommended for stagnant aquifers (Ahlstrom, 1979). For aquifers in areas of low heat flux, a small flow rate would significantly reduce the temperature rise.

FIGURE 3-19. MIGRATION SPEEDS PER UNIT TEMPERATURE GRADIENT OF ALL-LIQUID INCLUSIONS (OLANDER ET AL, 1980)

3.2.3.3 Geochemical System

The last subsystem in the site system is the geochemical system. A representation of the geochemical system as a hierarchy is shown in Figure 3-20. (The following discussion will refer to Figure 3-20.) The primary function of the geochemical system is to maintain the geochemical stability of the rock mass in which the repository is located. The geochemical system should, therefore, be maintained at sufficiently low temperatures to retard those effects which are detrimental to its effectiveness as an isolation barrier.

Elevated temperatures can lead to a number of chemical effects in the rock mass. Most of these are detrimental to the geochemical system stability. Minerals may become unstable and reactions may occur which reduce the radionuclide retention capability of the formation.

The primary thermochemical concern for a repository in salt is the dissolution of the salt formation. As the temperature increases, so does the solubility of salt. Kaufman (1960) has shown that the mass of NaCl in a saturated solution increases by approximately 6 percent as the temperature increases from 30 C to 100 C. A possible beneficial effect of the variation of salt solubility with temperature is the potential for plugging fractures and ground water pathways downstream from the repository when the ground water cools and precipitates some of the salt from solution.

Most of the primary and secondary minerals which form granitic rocks appear to be stable at the pressures and temperatures expected in the rock mass. One reaction, however, which does take place at low temperatures and pressures is dehydration. Most of the common

FIGURE 3-20. HIERARCHY OF GEOCHEMICAL SYSTEM LIMITS

rock-forming minerals are either nonhydrous or have tightly bonded water and therefore are stable in the expected conditions. The micas are the only mineral family which can become unstable below 500 C and low pressures. Some micas may become unstable at temperatures as low as 350 C (NWTS Working Group on Far-Field Performance Constraints in the draft on repository performance constraints in the far-field domain). Clay minerals, which occur widely in sedimentary sequences as clay seams in salt strata and as secondary minerals on igneous rocks, can undergo dehydration and other changes at temperatures as low as 100 to 200 C (NWTS Working Group).

A maximum temperature of 125 C for granite and 100 C for the other rock types is recommended for the region which extends from the repository near field outward to a distance equal to 15 percent of the repository depth (NWTS Working Group). Anywhere outside of the region defined above, a maximum temperature of 100 C for granite and 75 C for the other rock types is recommended (NWTS Working Group).

Compatibility between the geology and the waste package system is also an important consideration in the geochemical system. The temperatures in the geochemical system should be kept at sufficiently low values to prevent any adverse chemical interactions between the rock mass, engineered barriers, and the waste form. Preliminary experiments show that waste-rock interactions, in the presence of ground water and under moderate temperature and pressure conditions, will be rapid and extensive if the container is breached (McCarthy et al, 1978). Ground water would be expected to transport species to and from the waste form and the host rock. This would produce a zone around the waste canister in which new minerals could develop (McCarthy et al, 1978).

There is also the potential for beneficial chemical reactions occurring in the geochemical system. The adsorption of migrating radionuclides on the overpack, backfill, host rock, secondary minerals, or sediments is one such favorable reaction. The loss of these sorption properties should be prevented.

Materials selected for their sorption properties should be thermally stable, have good heat-conducting properties, and have low permeability to ground water in the repository environment. Zeolites, such as chabazite and clinoptilolite, are examples of inorganic sorption materials. At temperatures well below 300 C, however, zeolites tend to lose their water and, therefore, their sorption properties (Miller and Wurm, 1980).

From Figures 3-17, 3-18, and 3-20 it can be seen that some of the failure mechanisms are lacking thermal-thermomechanical limits. These gaps in the thermal limits prevent the thermal criteria hierarchy from realizing its full potential. Since these limits could not be found in the literature, work should be undertaken to determine them.

4 APPLICATION AND IMPACTS OF THE THERMAL CRITERIA HIERARCHY

There are several possible uses for the thermal criteria hierarchy developed in Chapter 3. Some of these potential uses are to determine the thermal-thermomechanical limits applicable to a given repository design or to determine if the thermal performance of a given repository design is acceptable, and to evaluate the impact on a given repository design of a change in the thermal limits or to evaluate the impact on the thermal limits of a change in a given repository design. Section 4.1 demonstrates the selection of temperature limits for an example waste type and form, canister, backfill, and medium. Then an example repository design for the given waste package and medium is evaluated to determine its thermal performance. Section 4.2 briefly discusses the impacts of various repository design alternatives. As an example of an alternative, the waste type in Section 4.1, 10-year-old BWR, is cooled an additional 90 years prior to permanent disposal. The possible impacts associated with this design variation are briefly described.

4.1 APPLICATION OF THE THERMAL CRITERIA HIERARCHY

Over the years, several design studies have been prepared for the disposal of nuclear waste in geologic repositories. Each of these studies developed a set of thermal limits by which to evaluate their designs. The thermal constraints chosen for spent fuel in salt by three of these studies are listed in Table 4-1.

Table 4-1. Comparison of Thermal - Thermomechanical Limits Chosen by the Management of Commercially Generated Radioactive Wastes (U.S. Department of Energy, 1980), the Westinghouse Engineered Waste Package Design Studies (Westinghouse, 1983), and the Areal Thermal Loading Recommendations for Nuclear Waste Repositories in Salt (Russell, 1977)

Constraints	Management of Commercially Generated Radioactive Waste (U.S. Department of Energy, 1980)	Westinghouse Engineered Waste Package Design Studies (Westinghouse, 1983)	Areal Thermal Loading Recommendations for Nuclear Waste Repositories in Salt (Russell, 1977)
Maximum Spent Fuel Temperature, C	300	375 (during containment) 100 (after containment)	200
Maximum Canister Temperature, C	375 (304L SS)	250 (titanium)	375
Maximum Salt Temperature, C	250	250	260-320
Maximum Room Closure	10-15% of original room opening		10-15% of original room opening
Maximum Uplift, m	1.2 - 1.5		1.2 - 1.5
Maximum Temperature Rise at Surface, C	0.5		0.5
Maximum Temperature Rise in Aquifer, C	6.0		6.0 and ALARA ^(a)

(a) ALARA = as low as reasonably achievable.

As can be seen from Table 4-1, there is considerable variation in the thermal constraints chosen and in the values assigned to them. This variation in the choice of thermal limits reveals the need for an organized approach to the evaluation of the thermal effects on the waste disposal system that result from the thermal energy release of the waste. The hierarchy of thermal criteria, described in Chapter 3, was developed for just such a purpose.

Table 4-2 describes a possible repository configuration for disassembled BWR spent fuel in salt. A set of maximum permissible temperatures for the repository configuration shown in Table 4-2 can be obtained from the thermal criteria hierarchy developed in Chapter 3. These are listed in Table 4-3. In some instances, such as the corrosion limit for the canister, where temperature limits were not available for the failure mechanism, they were not included. A complete analysis of the thermal criteria hierarchy would require the determination of these missing thermal limits.

An analytical program, ARRAYF (Klett et al, 1980, see Appendix B), was used to determine if the repository configuration shown in Table 4-2 violated any of the temperature constraints in Table 4-3. ARRAYF (Klett et al, 1980), is a computer program developed to solve the analytical equations for the conduction of heat from finite-length vertical heat sources through a semi-infinite medium with one isothermal surface. It includes up to 12 heat sources at a given distance below the isothermal surface. The isothermal plane and adjacent heat sources are added by using the method of superposition. An eight-point Gaussian quadrature technique is used to evaluate the equations.

Table 4-2. Repository Characteristics

Characteristics	Specifications
Waste Description	
Age of Waste, yr	10.0
Thermal Loading, W	2.86
Mass Loading, MTHM	0.0029
Diameter of Pin, m	0.0062
Active Length, m	4.0
Canister Description	
Inner Diameter, m	0.232
Thermal Loading, kW	2.622
Backfill Thickness, m	0.15
Room Description	
Rows per Room	2
Row Separation, m	10.67
Hole Pitch, m	25.65
Adjacent Pillar Thickness, m	22.97
Canisters per Hole	1
Repository Depth, m	600.0
Materials	
Waste	BWR Spent Fuel Pins
Canister	Carbon Steel
Backfill	Crushed Salt
Medium	Bedded Salt

Table 4-3. A Partial Set of Restrictive Temperatures
for the Repository Configuration Shown in
Table 4-2^(a)

Waste Form/Canister/Medium	Maximum Temperature, C
<hr/>	
Spent Fuel	
during containment period	300
after containment period	100
Carbon Steel Canister	375
Salt	
emplacement area	250
near field	100
far field	75

(a) Where temperature limits were not available for the failure mechanisms developed in Chapter 3, they were not included.

There are two options for the finite-length vertical heat sources available in ARRAYF: the canister option and the solid cylinder option. The canister option shown in Figure 4-1 was selected for the analysis of pins close packed in the canister, as specified by Science Applications, Inc. (1979). Since the temperature drop across the canister wall is considered to be negligible it was replaced with crushed salt backfill. The temperature is calculated at each of the 10 points shown in Figure 4-1. Table 4-4 shows the thermal conductivities, emissivities, and heat capacities assumed in the analysis. All physical property data are assumed to remain constant.

There are four possible canister arrays available in ARRAYF: a square array, a triangular array, a single-row array, and a double-row array. The double-row array was used in the analysis of the repository

FIGURE 4-1. CANISTER CONFIGURATION USED IN ARRAYF

configuration shown in Table 4-2. Figure 4-2 shows the arrangement of the canisters in the room.

The results of the temperature analysis of the repository configuration in Table 4-2 are shown in Figures 4-3 and 4-4. Figure 4-3 shows the radial temperature distribution of the spent fuel canister. Figure 4-4 shows the variation of the spent fuel centerline temperature with time after emplacement. As can be seen from these figures, all the temperature limits listed in Table 4-3 are easily met. The waste containment, canister, and salt emplacement temperatures are all below their maximum values. At a distance of 0.86 meters the medium temperature is well below 75 C and declining, thereby assuring that the near- and far-field limits will be met (Figure 4-3). The spent fuel centerline temperature 100 years after disposal is already below 100 C and

Table 4-4. Assumed Thermal Conductivities, Emissivities, and Heat Capacities

Waste Form/Backfill/ Medium	Thermal Conductivity, kW/mC	Emissivity	Heat Capacity, kJ/m ³ C
Close-packed array of BWR spent fuel pins(a)	0.00088	0.6	Neglected
Crushed Salt Backfill(b)	0.0003	0.6	Neglected
Salt	0.0060	0.6	2200.0

(a) A close-packed array as defined in Appendix B of Storage Room Design Parameters for the ONWI Spent Fuel Disposal Study: Nine Case Studies in Salt (Science Applications, Inc., 1979).

(b) The emissivity assumed for salt was also assumed for the crushed salt backfill.

FIGURE 4-2. ROOM AND PILLAR GEOMETRY

FIGURE 4-3. RADIAL TEMPERATURE DISTRIBUTION OF SPENT FUEL WASTE PACKAGE

FIGURE 4-4. VARIATION OF SPENT FUEL CENTERLINE TEMPERATURE WITH TIME AFTER EMPLACEMENT

declining, thereby assuring that at the end of the containment period the 100 C spent fuel temperature limit will be met (Figure 4-4).

The U.S. Department of Energy (1980) specifies a maximum average thermal loading of 60 kW/acre based on far-field limitations, for the case of spent fuel in salt with a 15 cm backfill thickness. Although salt can accept higher thermal loadings based on room and pillar stability considerations, the long-term heat contribution from the plutonium in spent fuel results in unacceptable amounts of surface uplift. The additional heat contribution of plutonium does not affect room stability. In order to meet the far-field limit of 60 kW/acre, the maximum near-field density that can be achieved for spent fuel is 75 kW/acre (U.S. Department of Energy, 1980). At this loading, the temperature constraints on the surface and aquifers are also met.

The local thermal loading for the repository configuration in Table 4-2 is 19.7 kW/acre. This is well below the 60 kW/acre far-field limit and the 75 kW/acre near-field limit. Since the number of canisters or their size does not affect the far field, it is reasonable to assume that at 19.7 kW/acre the far-field thermomechanical limits in Chapter 3 are satisfied.

Room and pillar stability, however, is also dependent on the extraction ratio. For the calculations in this chapter the repository configuration described in a draft on interim reference repository conditions for salt was used as a starting point. The row separation and hole pitch dimensions were then varied while the pillar width remained constant. This leads to extraction ratios for the lower emplacement densities, such as that of the example, which exceed the recommended 25 percent limit for salt. The temperatures, however, for

such cases are then limiting since the rooms would become narrower and/or the pillars wider to achieve an acceptable extraction ratio. This would lead to lower temperatures and thermomechanical stresses.

It is tempting to use the thermal loading as a design point, since all the thermal constraints seem to be met at 19.7 kW/acre. This is misleading, however, as it is possible to have a thermal loading of 19.7 kW/acre and still violate a thermal constraint (Figure 4-5). By altering the amount of waste per package and the spacing of the canisters as per Table 4-5, the spent fuel temperature limit was exceeded while the thermal loading remained at 19.7 kW/acre (20 MTHM/acre).

4.2 IMPACTS OF VARIOUS ALTERNATIVES

An important aspect of the waste disposal program is the recognition, definition, and evaluation of the interfaces that exist between the engineered waste package and the other components of the mined geologic repository system. Changing any one element in the system can affect the other components of the system. The actual design of the waste package will have an impact on virtually all aspects of the waste disposal system, from the waste form configuration to repository operations and decommissioning. In terms of the self-shielded waste package versus the borehole waste package, the design of the self-shielded waste package allows for hands-on operations and eliminates the need to have individual boreholes for each package (Cohen et al, 1977). The choice of medium will affect the size of the repository and how much waste can be emplaced in it. Cost will also vary as the components of the waste disposal system change.

FIGURE 4-5. RADIAL TEMPERATURE DISTRIBUTION OF SPENT FUEL WASTE PACKAGE

Table 4-5. Repository Characteristics

Characteristics	Specifications
Waste Description	
Age of Waste, yr	10.0
Thermal Loading, W	2.86
Mass Loading, MTHM	0.0029
Diameter of Pin, m	0.0062
Active Length, m	4.0
Canister Description	
Inner Diameter, m	0.28
Thermal Loading, kW	3.849
Backfill Thickness, m	0.15
Room Description	
Rows per Room	2
Row Separation, m	10.67
Hole Pitch, m	37.68
Adjacent Pillar Thickness, m	22.97
Canisters per Hole	1
Repository Depth, m	600.0
Materials	
Waste	BWR Spent Fuel Pins
Canister	Carbon Steel
Backfill	Crushed Salt
Medium	Bedded Salt

Table 4-6 lists the results for some variations in canister loading and canister spacing of the repository configuration outlined in Table 4-2. In all cases if the 300 C spent fuel limit was met, so were all the other temperature constraints. This leads to an hourglass view, Figure 4-6, of the interaction of the thermal criteria and possible design variations for a given situation. In this scheme, the applicable thermal limits for a given set of specifications, waste type, medium, repository location, etc., are obtained from the thermal criteria. The most restrictive thermal limits(s) can then be determined from these. After the most restrictive thermal limit(s) have been determined, various design options can be analyzed to find the optimum design. The most restrictive temperature limit for the repository configuration in Table 4-2 is the 300 C spent fuel limit during containment.

Cost is an important parameter in the optimization of the waste disposal system. Knowing the most restrictive thermal limit it is possible to attempt to optimize a given repository design with respect to cost. One parameter which could influence the cost of a repository is the age of the waste at disposal. Because the thermal power of the waste decreases over time, there may be a benefit from longer term storage of waste, prior to disposal. Figure 4-7 shows the decrease in peak spent fuel centerline temperature for the repository configuration in Table 4-2.

There are two possible ways to take advantage of the lower thermal power of aged spent fuel: (1) placing more waste in a canister, or (2) placing the canisters closer together. The benefit could take the form of reduced costs in mining and/or in packaging. Figure 4-8 illustrates

Table 4-6. Canister Load and Spacing Variations in the
Repository Configuration of Table 4-2

Waste Age, yr	MTHM/ Package	Inner Radius of Canister, m	Row Sep- aration, m	Hole Pitch, m	Temperature, C		
					Centerline	Canister	Medium
10	3.9	0.280	15.67	40.66	330	247	41.4
	3.9	0.280	10.67	37.68	336	253	46.9
	3.9	0.280	5.67	9.67	362	278	73.6
	2.7	0.232	15.67	27.71	259	202	38.9
	2.7	0.232	10.67	37.68	262	205	41.6
	2.7	0.232	15.67	40.66	259	201	37.7
	2.7	0.232	10.67	25.65	263	206	41.3
	2.7	0.232	5.67	9.67	280	223	60.0
	2.7	0.232	4.67	17.46	279	222	58.5
	2.7	0.232	4.67	14.25	280	223	60.1
	2.7	0.232	5.67	6.51	286	286	65.0
	3.3	0.257	4.67	17.46	320	250	64.9
	2.7	0.232	10.67	37.70	92.4	77.2	33.6
	2.7	0.232	10.67	25.65	93.9	78.3	33.8
100	14.5	0.540	4.67	17.49	276	192	74.3
	14.5	0.540	5.67	9.56	278	195	77.2
	14.5	0.540	10.67	37.70	252	167	48.7
	18.0	0.601	15.67	40.66	284	179	45.5
	18.85	0.615	21.67	47.28	289	179	40.8
	18.85	0.615	46.67	66.60	277	170	34.4
	23.0	0.680	15.67	40.66	337	202	49.0
	23.0	0.680	10.67	37.70	346	211	58.8

FIGURE 4-6. AN HOUR-GLASS VIEW OF THE INTERACTION OF THE THERMAL CRITERIA AND THE DESIGN OPTIONS

**FIGURE 4-7. VARIATION OF SPENT FUEL CENTERLINE TEMPERATURE WITH TIME
FOR THE REPOSITORY CONFIGURATION IN TABLE 4-2**

FIGURE 4-8. EFFECT OF SPENT FUEL AGE ON REPOSITORY CAPACITIES FOR THE ONCE-THROUGH CYCLE AND 2,000-ACRE REPOSITORIES

the possible advantage of placing the canisters closer together. Additional cooling of the spent fuel could reduce the area required to dispose of the same number of spent fuel elements by 5 to 30 percent, depending on the geologic medium (U.S. Department of Energy, 1979).

Changing only the age of the waste to 100 years, the repository configuration in Table 4-2 was again analyzed. The results of this analysis are shown in Figures 4-9 and 4-10. Figure 4-9 shows the radial temperature distribution of the spent fuel canister. Figure 4-10 shows the variation of the spent fuel centerline temperature with time after emplacement. The peak spent fuel centerline, canister, and emplacement area temperatures are well below their maximum permissible values shown in Table 4-3. At a distance of 0.86 meters the medium temperature is well below 75 C and declining, thereby assuring that the near- and far-field medium limits will be met (Figure 4-9). Since the maximum spent fuel centerline temperature is less than 100 C it is reasonable to assume that the spent fuel temperature at the end of the containment period is less than the 100 C limit.

The results presented in Figures 4-3 and 4-4 indicate that the amount of waste placed in the canister could be increased slightly without violating the temperature limits in Table 4-3. Figures 4-9 and 4-10, however, indicate that a potentially significant increase in the amount of waste that can be placed in the canister could be realized by cooling the waste for an additional 90 years. To evaluate the possible increases in the amount of waste per canister, the spacing between the canisters and the amount of waste per canister were varied until the most restrictive temperature limit, the 300 C spent fuel temperature, was reached. Figures 4-11 and 4-12 show the results of this analysis

FIGURE 4-9. RADIAL TEMPERATURE DISTRIBUTION OF SPENT FUEL WASTE PACKAGE

FIGURE 4-10. VARIATION OF SPENT FUEL CENTERLINE TEMPERATURE WITH TIME AFTER EMPLACEMENT

FIGURE 4-11. MAXIMUM AMOUNT OF BWR SPENT FUEL, COOLED 10 YEARS, THAT CAN BE PLACED IN A WASTE PACKAGE WITHOUT VIOLATING THE SPENT FUEL CENTERLINE TEMPERATURE LIMIT

FIGURE 4-12. MAXIMUM AMOUNT OF BWR SPENT FUEL, COOLED 100 YEARS, THAT CAN BE PLACED IN A WASTE PACKAGE WITHOUT VIOLATING THE SPENT FUEL CENTERLINE TEMPERATURE LIMIT

for 10- and 100-year-old BWR spent fuel in salt. The cross-hatching in each of the figures indicates the point at which the total energy released into the formation becomes the most restrictive thermal limit.

As can be seen from Figure 4-11, the maximum amount of 10-year-old BWR spent fuel that can be placed in a single canister without violating the spent fuel centerline temperature limit ranges from 3.3 MTHM/package (1,138 pins) at 20 MTHM/acre to about 2.9 MTHM/package (1,000 pins) at 50 MTHM/acre. Most current design studies, therefore, could increase the amount of waste they place in a single package. Approximately six times more waste could be placed in a single canister, however, if the BWR spent fuel were cooled an additional 90 years (Figure 4-12). This could significantly reduce the number of packages required to dispose of the same amount of BWR spent fuel cooled only 10 years. The cross-hatching in both Figures 4-11 and 4-12 indicates the point at which the far-field limit dominates.

With the current design of long-lived packages, which represents a large portion of the total cost of a repository, a cost savings of only a few percent would be realized by reducing the area required for a repository. By placing more waste in a package, however, a potentially significant cost savings could occur. Mining of boreholes, package fabrication, and package handling costs are reduced when fewer packages are required. For spent fuel cooled 100 years, a potential cost savings of approximately 30 percent has been calculated (Waddell et al, 1981). Over one-third of the cost reduction was due to savings in package materials. However, any cost advantages realized by this additional cooling of the waste may be counterbalanced by the added cost and hazard of long-term storage prior to permanent disposal.

By understanding the thermal constraints that apply to a given repository configuration, it is possible to evaluate the acceptability of the repository design, as was shown in the first section of this chapter. It is also possible to vary the repository characteristics in an attempt to optimize the design, as was shown in the latter part of this chapter. Although this analysis dealt with the impacts of waste age, similar analyses could be performed to evaluate the impacts of different waste package designs, canister materials, backfill materials, canister spacings, room and pillar designs, media, etc. In all cases the thermal criteria hierarchy developed in Chapter 3 can be used to determine the appropriate thermal constraints by which to evaluate a particular design.

5 CONCLUSION AND RECOMMENDATIONS

Two major points are provided by this report towards understanding the thermal impacts on the mined geologic disposal system that result from the thermal energy release of the emplaced wastes. The first is the summary of the thermal limits in current use (presented in Chapter 2) which have been placed on various aspects of the waste disposal system. The second point is the development of the thermal criteria hierarchy in Chapter 3. By identifying the aspects of the waste disposal system that are adversely affected by increased temperatures and how these areas interrelate, it is possible to evaluate the thermal acceptability of a given repository design.

Directly related to the thermal criteria hierarchy in Chapter 3 is the parametric analysis shown in Chapter 4. This type of analysis can identify those aspects of repository design that are heavily influenced by the thermal energy release of the emplaced wastes. Through this information, possible repository designs could be optimized in terms of effectiveness, safety, cost, etc.

Before the thermal criteria hierarchy developed in Chapter 3 can realize its full potential, the concept of such a hierarchy of thermal limits must be accepted and used. Also, for this thermal criteria hierarchy to realize its full usefulness, the missing thermal limits must be determined. These missing thermal limits include such items as the effect of increased temperatures on canister corrosion, the volatility of the wastes, the migration of ground water towards heat

sources, and medium canister waste form interactions. Care should be taken in determining these temperature limits so that they are consistent with other limits in the hierarchy. As other failure mechanisms are identified they should be incorporated into the thermal criteria hierarchy.

The limits, of course, are not enough in themselves. One must also have accurate physical property data to evaluate site-specific repository designs.

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APPENDIX A

GLOSSARY OF TERMS

*U.S. Department of Energy, Management of Commercially
Generated Radioactive Waste, Vol 1,2, DOE/EIS-0046F,
October 1980.

GLOSSARY OF KEY TERMS AND ACRONYMS

Abiotic: characterized by the absence of life.

Abyssal Hill: relatively small topographic feature of the deep ocean floor ranging to 600 to 900 m high and a few kilometers wide.

Actinides: Radioactive elements with atomic number larger than 88.

Activation: The process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles.

Activity: A measure of the rate at which radioactive material is emitting radiation; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The special unit of activity is the curie (Ci).

AFR: Away-from-reactor (spent fuel storage concept).

Aging: Usually refers to time to permit decay of short-lived radionuclides.

ALAP: As low as practicable, now generally replaced with ALARA (as low as reasonably achievable).

ALARA: As low as reasonably achievable. ALARA refers to limiting release and exposure and is used by the NRC (10 CFR 50.34) in the context of ". . . as low as reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations. . ."

Allowance Item: A number, arrived at by judgement, that represents material or equipment cost that cannot be developed otherwise because of the absence of design detail.

Alluvial Fan: A sloping, fan-shaped mass of loose rock material deposited by a stream at the place where it emerges from an upland onto a broad valley or a plain.

Alluvium: All detrital material deposited permanently or in transit by streams.

Alpha Particle: A positively charged particle emitted by certain radioactive material. It is made up of two neutrons and two protons; hence it is identical with the nucleus of a helium atom.

Amphibole: A group of dark, rock-forming, ferromagnesian silicate minerals which are closely related in crystal form and composition and which have abundant and wide distribution in igneous and metamorphic rocks.

Andesitic: A volcanic rock composed primarily of the plagioclase feldspar andesine and one or more mafic constituents.

Anion: An ion that is negatively charged.

Anticline: A fold, the core of which contains stratigraphically older rocks, which in simplest form is elongate and convex upward with the two limbs dipping away from each other.

APS: Atmospheric protection system.

Aquifer: A water-bearing layer of permeable rock or soil that will yield water in usable quantities to wells.

Aquitard: A natural rock or soil of low permeability which is stratigraphically adjacent to one or more aquifers and through which water movement is markedly retarded or impeded.

Argillaceous: Containing or pertaining to clay.

Artesian: When pertaining to an aquifer, it is one that is confined so that its hydraulic head rises above the top of the aquifer unit; thus an artesian water body is one that is confined under hydraulic pressure.

Atom: An electrically neutral particle of matter, indivisible by chemical means.

Atomic Number: The number of protons within an atomic nucleus.

Atomic Weight: The mass of an atom relative to other atoms.

Back End of the Fuel Cycle: Includes spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, and waste management.

Background Radiation: The radiation in man's natural and undisturbed environment. It results from cosmic rays and from the naturally radioactive elements of the earth, including those from within the human body.

Basement Rock: A complex of undifferentiated rocks that underlies the oldest identifiable rocks in the area.

Basin: A depressed area generally having no outlet for surface water.

Batholith: A shield-shaped mass of igneous-intruded rock, greater than 100 km² in area, extending to great depth and whose diameter increases with depth.

Bedrock: A solid rock formation usually underlying one or more other loose formations.

Benthic: Refers to the bottom of a body of water.

Bentonitic: Pertaining to rock containing bentonite, a clay formed from the decomposition of volcanic ash.

Biosphere: The part of the earth in which life can exist, including the lithosphere, hydrosphere, and atmosphere; living beings together with their environment.

Biota: The animal and plant life of a region.

Biotite: A complex silicate of aluminum, potassium, magnesium, and iron with hydroxyl that is a widely distributed and important rock-forming mineral of the mica group.

Block-Faulting: A type of vertical faulting in which the crust is divided into structural or fault blocks of different elevations and orientations.

Boiling Water Reactor (BWR): A reactor system that uses a boiling water primary cooling system. Primary cooling system steam turns turbines to generate electricity.

Borosilicate Glass: A silicate glass containing at least 5 percent boric acid and used to vitrify calcined waste.

Breccia: A coarse-grained clastic rock composed of large, angular, and broken rock fragments cemented together in a finer grained matrix.

Burial Grounds: Areas designated for disposal of containers of radioactive wastes and obsolete or worn-out equipment by near-surface burial.

Calcine: Material heated to a temperature below its melting point to bring about loss of moisture and oxidation.

Canister: A metal container for radioactive solid waste.

Cask: A container that provides shielding and containment during transportation of radioactive materials.

Catastrophic: A violent, sudden or unexpected event which results in failure of the predicted performance of a system or component.

Cation: An ion that is positively charged.

Cation Exchange Chromatography (CEC): A process for separating several cations using the differences in the rate they travel on an ion exchange column.

Cermet: A material made by combining a heat resistant ceramic with a metal usually made by powder metallurgy.

CH-TRU: Contact-handled TRU waste.

Clastic: Pertaining to or the state of being a rock or sediment composed principally of broken fragments derived from preexisting rocks or minerals.

Colocated: Refers to location of facilities at a common site.

Concentration Guide: The average concentration of a radionuclide in air or water to which a worker or member of the general public may be continuously exposed without exceeding radiation dose standards.

Consolidated (material): In geology, natural materials that have been made firm, cohesive, and hard.

Contact-Handled Waste: Waste package having surface dose rate less than 0.2 R/hr. Such packages can be handled by workers without extensive shielding. Contact-handled wastes were termed low-level wastes in DOE/ET-0028 and DOE/ET-0029.

Containment: Confining the radioactive wastes within presented boundaries, e.g., within a waste package.

Contingency (cost): The amount of money added to the estimated cost of a project to cover certain areas of cost uncertainty and reduce the probability of understating the project cost estimate. With the contingency added, there is a more nearly equal probability of a cost underrun or overrun.

Cost of Money: Weighted cost of debt and equity financing. Cost of money is used synonymously with cost of capital.

Critical Mass: The mass of fissionable material of a particular shape that is just sufficient to sustain a nuclear chain reaction.

Criticality: The condition in which a nuclear reactor is just self-sustaining.

Crystalline Rock: An inexact but convenient term designating an igneous or metamorphic rock, as opposed to a sedimentary rock.

Curie (Ci): A special unit of activity where 1 Ci equals 3.7×10^{10} spontaneous nuclear disintegrations per second.

Daughter Nuclide: A nuclide formed upon disintegration of a parent radionuclide.

Decommissioning: Preparations taken for retirement from active service of nuclear facilities, accompanied by the execution of a program to reduce or stabilize radioactive contamination. The objective of decommissioning is to place the facility in such a condition that future risk to public safety from the facility is within acceptable bounds.

Decontamination: The selective removal of radioactive material from a surface or from within another material.

Decontamination Factor (DF): The ratio of the original contamination level to the contamination level after decontamination.

Deep Continental Geologic Formations: Geologic media beneath the continents and isolated from the land surface by several hundred to thousands of meters of overlying rock material.

Depositional Environment (sedimentary environment): A geographically restricted environment where sediment accumulates under similar physical, chemical, and biological conditions.

Devitrification: The process by which glassy substances lose their vitreous nature and become crystalline.

Diapirism: The piercing of overlying rocks by an upward-moving mobile core or material, such as a salt body or an igneous intrusion.

Discharge: In ground-water hydrology, water that issues naturally or is withdrawn from an aquifer.

Disposal (radioactive waste): The planned release of radioactive waste in a manner which is considered permanent so that recovery is not provided for.

Dome: A dome-shaped landform or rock mass; a large igneous intrusion whose surface is convex upward with sides sloping away at low but gradually increasing angles; an uplift or an anticlinal structure, either circular or elliptical in outline, in which the rock dips gently away in all directions, for example, a salt dome.

Dissolution: In this context it refers to the dissolving of spent fuel by nitric acid as a process step in fuel reprocessing.

Dose: Herein generally means the more rigorous term "dose-equivalent." The latter, expressed in units of rem, implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure.

DOT: U.S. Department of Transportation.

Dry Storage: Storage of waste packages without liquid cooling.

EIA: Energy Information Administration.

EPA: U.S. Environmental Protection Agency.

Epeirogeny: The broad movements of uplift and subsidence which affect whole or large portions of continents or ocean basins.

Fault: A fracture or fracture zone along which there has been displacement of the sides relative to one another parallel to the fracture.

Fault Block: A crustal unit either completely or partly bounded by faults.

Fault System: A system of parallel or nearly parallel faults that are related to a particular deformational episode.

Feldspar: Any of a group of common rock-forming minerals that are silicates of alumina and some other base, such as potash, soda, or lime.

Fission (nuclear): The splitting of a nucleus into two or (rarely) more fragments; usually limited to heavier nuclei such as isotopes of uranium, plutonium, and thorium.

Fission Product: Any radioactive or stable nuclide produced by fission, including both primary fission fragments and their radioactive decay products.

Fissionable Material: Actinides capable of undergoing fission by interaction with neutrons of all energies.

FPF: Fuel packaging facility.

Fracture: breaks in rocks caused by intense folding or faulting or the process of breaking fluid-bearing strata by injecting a fluid under such pressure as to cause partings in the rock.

Freshwater Lens: A body of fresh water roughly shaped like a lens formed as a result of injecting freshwater into a salt water body or occurring naturally when precipitation infiltrates a saline aquifer.

Fuel (nuclear reactor): Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel Cycle: Mining, refining, enrichment, and fabrication of fuel elements, use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, reenrichment of the fuel material, refabrication of new fuel elements, and management of radioactive waste.

Fuel Element: A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

Fuel Reprocessing Plant (FRP): Plant where irradiated fuel elements are dissolved, waste materials removed, and reusable materials are segregated for reuse.

Fuel Residue Waste (FRW): Solid wastes consisting of the residue (fuel element hardware and chopped cladding material) after the bulk of fuel core material, including most of the actinides and fission products, has been dissolved in nitric acid.

Gamma Ray: Electromagnetic radiation, similar in nature to x-rays, emitted by the nuclei of some radioactive substances during radioactive decay.

GEIS: Generic Environmental Impact Statement.

Geohydrology: The study of the character, source, and mode of occurrence of underground water.

Geothermal: Pertaining to the heat of the interior of the earth.

Geothermal Gradient: The increasing temperature of the earth with depth.

GESMO: Generic Environmental Impact Statement on use of Mixed-Oxide fuel in LWRs.

Granitic: Of or pertaining to granite. Granite-like.

Granitoid: A textural term indicating grain size and mineral distribution typical of granite.

Ground Water: Water that exists or flows within the zone of saturation beneath the land surface.

Grout: A mortar fluid combined with liquid waste to provide a matrix for isolation of the waste and to seal the waste from the environment.

GWe: Gigawatts (billions of watts) of electrical generation; a rate of energy production.

Gyre: A large closed ringlike system of ocean currents which rotates in a circular motion in each of the major ocean basins.

Half-Life: a) physical--the time required for quantity of a radioactive substance to decay to one-half of its original quantity. b) biological--time required for half of an ingested or inhaled substance to be eliminated from the body by natural process. c) effective--time required for half of an ingested or inhaled radioactive substance to be eliminated from the body by the combination of radioactive decay and natural processes; mathematically equal to product of the physical and biological half-lives divided by the sum of the physical and biological half-lives.

Head End of the Fuel Cycle: Mining, milling, enrichment, and fabrication of UO_2 fuel.

HEPA: High-efficiency particulate air (filter).

High-Level Liquid Waste (HLLW): The aqueous waste resulting from operation of the first cycle solvent extraction system (or its equivalent) in a facility for reprocessing irradiated reactor fuels as well as concentrated wastes from subsequent cycles.

High-Level Waste (HLW): DOE management directives define high-level waste to include high-level liquid wastes, products from solidification of high-level liquid waste, and irradiated fuel elements if discarded without reprocessing. A proposed NRC regulation (10 CFR 60.3) defines high-level waste to include irradiated fuel, high-level liquid waste, and products from its solidification. In the GEIS there are instances, however, where discarded spent fuel and high-level waste (as wastes from the reprocessing of spent fuel) are cited separately.

Highly Enriched Uranium (HEU): Uranium containing 5% or more of added ^{235}U .

HM: Heavy metal, generally uranium and plutonium.

Hornblende: A common member of the amphibole group of minerals.

Hot Cell: A facility which allows remote viewing and manipulation of radioactive substances.

Hydraulic Gradient: The change in static head per unit of lateral distance in a given direction.

Hydrologic: Pertaining to the study of the properties, distribution and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

Hydrostatic Pressure: The pressure exerted by the water at any given point in a body of water at rest.

ICPP: Idaho Chemical Processing Plant.

ICRP: International Commission on Radiological Protection.

ILLW: Intermediate-level liquid waste.

Immobilization: Treatment and/or emplacement of the wastes so as to impede their movement.

Interim Storage: Storage operations for which a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or final disposition is expected.

Interstices: In geology, small openings between solid particles in a rock or unconsolidated material; may be a void or pore and often contains ground water. Interstitial permeability is used to differentiate interconnected pore permeability from fracture permeability.

Ion Exchange: Replacement of ions adsorbed on a solid, such as a clay particle, or exposed at the surface of a solid by ions from solution, usually in natural water. The phenomenon is known to occur when natural water moves through clays, zeolitic rocks, and other materials of the earth's crust.

ISFS: Independent spent fuel storage.

ISFSF: Independent spent fuel storage facility.

Isolation: Segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

Joint: A fracture or parting in a rock, along which little or no displacement of rock material has occurred.

Kaolinite: A common clay consisting mainly of hydrous aluminum silicate and closely related in chemical composition and crystal structure.

Kilowatt-hour (kWh): Use of electricity for one hour at a rate of 1000 watts.

Levelized Unit Cost: Capital and operating charges translated into an equivalent constant (or level) annual unit cost.

Light Water Reactor (LWR): May be either a BWR or PWR; uses as coolant ordinary water (H_2O) instead of heavy water (D_2O).

Lithification: The conversion of unconsolidated sediment into solid rock by processes such as compaction, cementation, and crystallization.

Lithology: The study of rocks. Also the character of a rock: its structure, color, mineral composition, grain size, and arrangement of its component parts.

Lithostatic pressure: The confining pressure at depth in the crust of the earth due to the weight of the overlying rocks.

Littoral: Belonging to, inhabiting or taking place on or near the shore of a body of water.

Low Enriched Uranium (LEU): Uranium containing less than 5% by weight but greater than 0.72% by weight ^{235}U .

M&M Shaft: Men and Materials shaft at a mined repository.

Mafic: Pertaining to or composed dominantly of magnesium rock-forming silicates.

Magmatism: The development, movement, and solidification to igneous rock, of magma, a naturally occurring mobile rock material, generated within the earth and capable of intrusion and extrusion.

Maximum Individual, Maximum-Exposed Individual: A person whose location and habits tend to maximize his radiation dose.

Megawatts (MW): Millions of watts.

Mica: A group of silicate minerals of aluminum and other bases, especially potassium, magnesium, and iron, and characterized by great perfection of cleavage in one direction, that produces thin, tough, elastic plates.

Mixed-Oxide Fuel Fabrication Plant (MOX-FFP): Plant where uranium oxide and plutonium oxide are mixed and fabricated into fuel elements for use in nuclear power plants.

MOX: Mixed oxides (of uranium and plutonium).

MTHM: Metric tons of heavy metal (usually refers to reactor fuel, in which the heavy metals are uranium and plutonium).

Mucking and/or Settling Ponds: Ponds next to drilling operations where the excavated mud or slurry is placed; the sediment that settles at the bottom of these ponds is called muck.

Multibarrier: A system using the waste form, the container (canister), the overpack, the emplacement medium, and surrounding geologic media as multiple barriers to isolate the waste from the biosphere.

NAS: National Academy of Sciences.

NASA: National Aeronautics and Space Administration.

NCRP: National Council on Radiation Protection and Measurement.

Neutron: Stable particle in a nucleus of very slightly greater mass than a proton but without nuclear change.

NO_x: Oxides of nitrogen, specifically NO and NO₂.

NRC: Nuclear Regulatory Commission.

Nucleus: The inner core of the atom, consisting primarily of neutrons and protons, which make up almost the entire mass of the atom but only a minute part of its volume.

Nuclide: A species of atom characterized by its mass number, atomic number, and nuclear energy state; to be regarded as a distinct nuclide the atom must be capable of existing for a measureable lifetime in its nuclear energy state.

Olivine: An olive-green, common rock-forming ferromagnesian silicate mineral of mafic, Ultramafic, and low-silica igneous rocks.

ONWI: Office of Nuclear Waste Isolation at Battelle Memorial Institute, Columbus, Ohio; under contract to DOE.

Operations: Broad classification of waste management activities in terms of their basic function (e.g., waste storage, treatment, transportation or disposal).

ORNL: Oak Ridge National Laboratory.

Overpack: Secondary (or additional) external containment for packaged nuclear waste.

Outcrop: A part of a body of rock that appears, bare and exposed, at the surface of the ground.

Parent Nuclide: A radionuclide that upon disintegration yields a specified nuclide, either directly or as a later member of a radioactive decay series.

Partition: To separate one (or more) element(s) from one (or more) other element(s). Examples include the separation of uranium and plutonium from each other, the separation of actinides and fission products in the waste, and the separation of one fission product from the other fission products.

Perihelion: The point in the orbit of a celestial body that is closest to the sun.

Permeability: The quality or state of being permeable. The relative ease with which a porous medium can transmit a liquid under a hydraulic gradient.

Peridotite: A coarse-grained plutonic igneous rock composed chiefly of the mineral olivine but also containing considerable amounts of other ferromagnesian minerals.

Plagioclase: The group of common rock-forming feldspar minerals; silicates of varying mixtures of sodium and calcium.

Pluton: A body of intrusive igneous rock of any shape or size.

Pluvial: Pertaining to a period of time in which rainfall or precipitation is abundant.

PNL: Pacific Northwest Laboratory operated for DOE by Battelle Memorial Institute.

Porosity: That property of a rock or soil which enables the rock or soil to contain water in voids or interstices, usually expressed in percentage or as a decimal fraction of void volume as compared to total volume.

Pressurized Water Reactor (PWR): A reactor system that uses a pressurized water primary cooling system. Steam formed in a secondary cooling system is used to turn turbines to generate electricity.

Primary Wastes: Untreated initial wastes resulting from operation of fuel cycle facilities other than waste management facilities (wastes from operation of waste management facilities are secondary wastes).

Pyroxene: A group of dark rock-forming silicate minerals closely related in crystal form and analogous in chemical composition to the amphiboles; found chiefly in igneous rocks.

Rad: Radiation absorbed dose, the basic unit of absorbed dose of ionizing radiation. A dose of 1 rad is equivalent to the absorption of 100 ergs of radiation energy per gram of absorbing material.

Recharge: In hydrology, a source or means for replenishment of water withdrawn or discharged from an aquifer.

rem (roentgen equivalent man): A quantity used in radiation protection to express the effective dose equivalent for all forms of ionizing radiation. It is the product of the adsorbed dose in rads and factors related to relative biological effectiveness.

Remotely Handled Waste: Waste package having surface dose rate greater than 0.2 R/hr. Such packages require extensive shielding and/or remote handling to protect operating personnel. Remotely handled wastes were termed intermediate-level wastes in DOE/ET-0028 and DOE/ET-0029.

Repository (Federal): A Federally owned and operated facility for storage or disposal of specific types of waste from DOE sites and/or licensees.

Retrievability: Capability to remove waste from its place in isolation with approximately the same level of effort and radiation exposure as required to place the waste.

RH-TRU: Remotely handled TRU waste.

Risk (mathematical): Product of the consequences and the probability of the event's occurrence.

Roentgen: A unit for measuring gamma or "x-ray" radiation. The Roentgen is defined by measuring the effect of the radiation on air. It is that amount of gamma or x-rays required to produce ions carrying 1 electrostatic unit of charge in 0.001293 g of dry air under standard conditions; $1 \text{ R} = 2.58 \times 10^{-4} \text{ coulomb/kg}$.

RWSF: Retrievable waste storage facility.

Scrubbers: An apparatus that chemically removes impurities from exhaust gas emissions.

Secondary Wastes: Wastes that result from applying waste treatment technologies to primary wastes.

Sedimentary Basin: A geologically depressed area that has thick sediments in the interior and thinner sediments at the edges.

Seismicity: The phenomenon of earth movements as manifested by earthquakes.

SFPF: Spent fuel packaging facility.

Shield: A continental segment of the earth's crust which has been relatively stable over a long period of time and which has exposed crystalline rocks mostly of Precambrian age; in general, representing the oldest rocks of the continent.

Shielding: A material interposed between a source of radiation and personnel for protection against the danger of radiation. Commonly used shielding materials are concrete, water and lead.

Shipping Cask: A specially designed container used for shipping radioactive materials.

SHLW: Solidified high-level waste.

Short-Lived Nuclides: Radioactive isotopes with relatively short half-lives. Usage for some isotopes varies with the concept being considered (e.g., isotopes with 5-50 year half-lives are short lived in the context of geologic disposal but long lived in the context of predisposal operations).

Slurry: A fluid mixture or suspension of insoluble material.

Solidification: Conversion of liquid radioactive waste to a dry, stable solid.

Source Terms: The quantity of radioactive material (or other pollutant) released to the environment at its point of release (source).

Spent Fuel (SF): Nuclear reactor fuel that has been used to the extent that it can no longer be used efficiently in a nuclear power plant.

Stock: An igneous intrusion less than 100 km² (40 mi²) in surface exposure.

Storage: Retention of waste in some type of manmade device in a manner permitting retrieval.

Strain: Deformation resulting from applied stress; proportional to stress.

Stratum: Sedimentary bed or layer, regardless of thickness, of homogeneous or gradational lithology.

Syncline: A fold, the core of which contains stratigraphically younger rocks, and which, in simplest form, is elongate and concave upward with the two limbs dipping toward each other.

Tailings: The part of any ore that is regarded as too poor to be treated further.

Tails: In the case of uranium it refers to the depleted uranium left after enrichment operations.

TBR: Tributyl phosphate, a solvent used in the PUREX fuel reprocessing process.

Technologies: Specific methods for implementing concepts. An example is calcination of liquid high-level waste by using a spray calciner.

Tectonic: Of, pertaining to, or designating the processes causing, and the rock structures resulting from, deformation of the earth's crust.

Tectonism (diastrophism): Crustal movement produced by earth forces, such as the formation of plateaus and mountain ranges; the structural behavior of an element of the earth's crust during, or between, major cycles of sedimentation.

Theoretical Density (TD): Maximum density attainable for any given material.

Thermal Regime: The area adjacent to a heat source which is affected by that source.

Trajectory: The curve that an object describes in space in traveling from one point to another.

Transmissivity: Volume of water flowing through a 1-ft width of aquifer of given thickness under a unit gradient (1 ft vertically for each 1 ft laterally) and at the viscosity prevailing in the field. Mathematically, it is the product of permeability and aquifer thickness.

Transmutation: A nuclear process in which one nuclide is transformed into the nuclide of a different element. This can be accomplished by bombardment with neutrons or other nuclear particles.

Transportation: Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic (TRU) elements: Elements with atomic number greater than 92. They include, among others, neptunium, plutonium, americium, and curium.

Transuranic Waste: Waste material measured or assumed to contain more than a specified concentration of transuranic elements. For purposes of this Statement, TRU waste is waste from locations that might cause contamination levels above 10 nanocuries of transuranic alpha activity per gram of waste.

Treatment: Operations intended to benefit safety or economy by changing the waste characteristics.

Ultramafic: Pertaining to igneous rocks composed chiefly of ferromagnesian dark minerals.

Uplift: A structurally high area in the crust, produced by movements that raise or upthrust the rocks, as in a dome or arch.

Vital Areas: The code of Federal Regulations (10 CFR 73), defines equipment items, systems, devices, and materials whose failure, destruction or release could directly endanger the public health and safety by exposure to radiation defined as "vital". Areas containing such items or materials (e.g., spent fuel or high-level waste) are defined as "vital" areas and subject to special protection measures.

Waste Immobilization: Process of converting waste to a stable, solid and relatively insoluble form.

Waste Isolation Pilot Plant (WIPP): A Defense repository proposed for a site in Southeastern New Mexico.

Waste Management: The planning, execution and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, transportation, storage, surveillance, and isolation.

Water Table: The upper surface of the zone of water saturation in the subsurface, at which the pressure is equal to atmospheric pressure; the upper surface of an unconfined aquifer.

APPENDIX B

COMPUTER CODE ARRAYF*

*Klett, R. D., Hertel, E. S., Molly, A. E., 1980. System Engineering Programs for Geologic Nuclear Waste Disposal, SAND80-0440, Sandia National Laboratories, Albuquerque, New Mexico, June.

Finite Heat Generating Canisters With Effects of
the Earth's Surface and Adjacent Cylinders

Two computer programs have been developed which solve analytical equations for the conduction of heat from spherical and cylindrical, exponentially decaying heat sources, through a uniform semi-infinite medium with one isothermal surface. The MUFTI program describes the temperature field around single spherical or finite length vertical heat sources a given distance below the isothermal surface. ARRAY is an extension of MUFTI which includes up to 10 heat sources in a plane parallel to the isothermal surface. Both programs are based on the work of J. Mufti with the isothermal plane and adjacent heat sources added by using the method of superposition. An eight-point Gaussian quadrature technique developed by D. Amos is used to evaluate the equations. This technique is very rapid and inexpensive, making it well suited for systems studies. All thermophysical property, geometric, and time input data and all outputs are dimensionless, which makes it possible to analyze several geometries and surrounding media with each computer run by multiplying the output by several simple algebraic expressions.

Using the design sequence outlined in this report, the nuclear waste canister size and the disposal media are defined by using the TMAX series and modified by logistic, environmental, economic, manufacturing, and other constraints. Modified and extended versions of the MUFTI and ARRAY programs are used as inputs to the programs described in the next sections for repository optimization and for waste for sensitivity studies. For these applications, it is more convenient to input dimensional properties, geometries, and times,

and have a dimensional output. In addition, most repositories have canisters, liners, and gas-filled annuli between the waste and the disposal media. The modifications to ARRAY described in this section add the dimensional input and output, simplify the input, add multiple canisters and gaps, allow for an annular waste configuration, add canisters in a vertical column, change the output format, and make MUFTI an option of ARRAY, to form a new program ARRAYF.

The modifications are based on a single isolated canister or either of two canister arrays and two configurations. Figure B-1 shows the square and triangular (hexagonal close pack) layouts that would be used for geologic dry well and subseabed repositories. Figure B-2 is the configuration used to approximate a multilayered canister located a finite distance (D) from the earth's surface. Also shown are 10 preselected locations for temperature calculations. Figure B-3 is the configuration used to compute temperatures of up to 10 locations in the disposal medium.

FIGURE B-1. CANISTER LAYOUTS FOR GEOLOGIC DRY WELL AND SUBSEABED REPOSITORIES

FIGURE B-2. VERTICAL FINITE CONCENTRIC WASTE CYLINDERS FILLED WITH AN ANNULAR HEAT SOURCE, BURIED A FINITE DISTANCE FROM THE EARTH'S SURFACE IN THE $x = 0$ PLANE

FIGURE B-3. VERTICAL FINITE CYLINDRICAL HEAT SOURCES BURIED A FINITE DISTANCE FROM THE EARTH'S SURFACE IN TRIANGULAR OR SQUARE ARRAYS

DISTRIBUTION LIST

ALABAMA DEPT OF ENERGY
 CAMERON MCDONALD
ALABAMA STATE GEOLOGICAL SURVEY
 THORNTON L NEATHERY
ALLIED GENERAL NUCLEAR SERVICES
 P F HIGHBERGER
 M A KOLB
ALLIS-CHALMERS
 GARRICK J SOLOVEY
AMERICAN NUCLEAR INSURERS
 DOTTIE SHERMAN
ANALYSIS & TECHNOLOGY INC
 T MAZOUR
ANALYTIC & COMPUTATIONAL RESEARCH INC
 B SAGAR
APPLIED MECHANICS INC
 JOHN R WILLIAMS
ARGONNE NATIONAL LABORATORY
 SHERMAN M FRIED
 WYMAN HARRISON
 MARTIN SEITZ
 MARTIN J STEINDLER
 W S WHITE
ARINC RESEARCH CORP
 H P HIMPLER
ARIZONA PUBLIC SERVICE COMMISSION
 HENRY W RILEY, JR
ARTHUR D. LITTLE INC
 AVIVA BRECHER
 CHARLES R HADLOCK
ATKINS RESEARCH & DEVELOPMENT
 T W BROYD
ATOMIC ENERGY CONTROL BOARD
 J L WALLACH
ATOMIC ENERGY COUNCIL, ROC
 KUO-YUEH LIU
ATOMIC ENERGY OF CANADA LTD
 M O LUKE
 ANN QUINN
ATOMIC ENERGY RESEARCH ESTABLISHMENT
 P J BOURKE
 R L NELSON
 JOHN RAE
AUSTRALIAN ATOMIC ENERGY COMMISSION
BABCOCK & WILCOX
 INFORMATION SERVICES
BATTELLE COLUMBUS DIVISION
 JOHN T MCGINNIS
 STEPHEN NICOLSI
 THOMAS M TRAINER
 KENNETH R YATES
BECHTEL GROUP INC
 THOMAS S BAER
 LESLIE J JARDINE
 N A NORMAN
 GERALD L PALAU
 RICHARD J TOSETTI
BENDIX FIELD ENGINEERING CORP
 BILL GRAHAM
 CHARLES A JONES
 JOHN C PACER
BLACK & VEATCH
 M JOHN ROBINSON
BOEING ENGINEERING AND CONSTRUCTION COMPANY
 R B CAIRNS
BRUNNEN SYSTEMPLANUNG
 D BRENK
BRITISH NUCLEAR FUELS LTD
 R S WILKS

BROOKHAVEN NATIONAL LABORATORY
 M S DAVIS
 SANDRA G LANE
 PETER SOO
 HELEN TODOSOW (2)
BUNDESANSTALT FUR GEOWISSENSCHAFTEN UND ROHSTOFFE
 MICHAEL LANGER
 HELMUT VENZLAFF
BUNDESMINISTERIUM FUR FORSCHUNG
 ROLF-PETER RANDL
BUREAU DE RECHERCHES GEOLOGIQUES ET MINIERES
 PIERRE F PEAUDECEF
BURNS AND ROE INDUSTRIAL SERVICES CORP
 JOHN PIRRO
CALIFORNIA DEPT OF CONSERVATION
 PERRY AMIMITO
CANYONLANDS NATIONAL PARK
 PETER L PARRY
CAPITAL AREA GROUND WATER CONSERVATION COMMISSION
 A N TURCAN JR
CARNEGIE-MELLON UNIVERSITY
 INDIRA NAIR
CASTUS VALLEY ORCHARDS
 CARL ANDERSON
CAYUGA LAKE CONSERVATION ASSOCIATION INC
 D S KIEFER
CENTRE D ETUDE DE L ENERGIE NUCLEAIRE
 RENE HEREMANS
CENTRE D INFORMATIQUE GEOLOGIQUE
 GHISLAIN DE MARSILY
CONNECTICUT SITING COUNCIL
 GLORIA DIBBLE POND
CONSOIL ASSOCIATES
 DAVID A LEE
COPPE/UFRJ
 LUIZ OLIVEIRA
CORNELL UNIVERSITY
 JOHN BIRD
 DUANE CHAPMAN
 FRED H KULHAWY
CORTLAND COUNTY HEALTH DEPT
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 KARL J ANANIA
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 O L OZTUNOLI
DAPPOLONIA CONSULTING ENGINEERS INC
 LISA K DONOHUE
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DAWCON MANAGEMENT CONSULTING SERVICE
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DEPT OF THE ENVIRONMENT
 F S FEATES
DIVISION OF ENVIRONMENTAL SAFETY RESEARCH
 HARUTO NAKAMURA
DMGA, IPT
 C DINIS DA GAMA

DRAVO ENGINEERS AND CONSTRUCTORS
 KEN BEALL
DUKE UNIVERSITY
 THOMAS DAVIS
E.R. JOHNSON ASSOCIATES INC
 E R JOHNSON
EARTH SCIENCE AND ENGINEERING INC
 LOU BLANCK
EARTH TECHNOLOGY CORP
 KENNETH L WILSON
EAST TENNESSEE STATE UNIVERSITY
 MICHAEL R SMITH
EBASCO SERVICES INC
 ZUBAIR SALEEM
 RAYMOND H SHUM
ECOLOGY CENTER OF LOUISIANA
 ROSS VINCENT
EDISON ELECTRIC INSTITUTE
 R E L STANFORD
EDS NUCLEAR INC
 C SUNDARARAJAN
EG & G IDAHO INC
 GEORGE B LEVIN
 T H SMITH
ELECTRIC POWER RESEARCH INSTITUTE
 CHAIM BRAUN
ELSAM
 ARNE PEDERSEN
EMBASSY OF THE UNITED STATES (SWEDEN)
 MAUD LI HAJJLEN-RYLANDER
ENERGY FUELS NUCLEAR INC
 DON M PILLMORE
ENERGY INC
ENERGY RESEARCH GROUP INC
 MARC GOLDSMITH
ENERGY RESEARCH LABORATORY HITACHI INC
 MAKOTO KIKUCHI
ENVIRONMENTAL POLICY INSTITUTE
 DAVID M BERICK
ENVIRONMENTAL RESEARCH GROUP INC
 PETER G COLLINS
ENVIROSPHERE COMPANY
 ROGER G ANDERSON
 BOB HAINES
ERIE COUNTY ENVIRONMENTAL MANAGEMENT COUNCIL
 JOAN P SCHMIDT
ERTEC WESTERN INC
 KENNETH WILSON
EXXON NUCLEAR IDAHO COMPANY INC
 GARY WAYMIRE
F.B.A.B.
 OTTO BROZEN
FENIX & SCISSON INC
 JOSE A MACHADO
 CHARLENE SPARKMAN
FLORIDA DEPT OF ENVIRONMENTAL REGULATION
 HAMILTON OVEN
FLORIDA INSTITUTE OF TECHNOLOGY
 JOSEPH A ANGELO, JR
FLUOR ADVANCED TECHNOLOGY DIVISION
 JOAN V MCCURRY
FLUOR ENGINEERS & CONSTRUCTORS INC
 RAYMOND J DUGAL
FORD, BACON & DAVIS UTAH INC
 DARRELL H CARD
 PRESTON H HUNTER
 BURTON J THAMER

FOSTER-MILLER ASSOCIATES INC
 NORBERT PAAS
GENERAL ATOMIC COMPANY
 ROBERT J CAMPANA
 H C CARNEY
 MICHAEL STAMATELATOS
GENERAL COURT OF MASSACHUSETTS
 TIMOTHY J BURKE
GENERAL ELECTRIC COMPANY
 BRUCE HUTCHINS
GEOLOGICAL SURVEY OF CANADA
 JEFFEREY HUME
 LIBRARY
GEORGIA INSTITUTE OF TECHNOLOGY
 MELVIN W CARTER
 GEOFFREY G EICHHOLZ
 ALFRED SCHNEIDER
GEOTECHNICAL ENGINEERS INC
 RICHARD W TURNBULL
GEOTRONS
 JAMES MERCER
GESELLSCHAFT F. STRAHLEN U.
UMWELTFORSCHUNG M.B.H.
 WOLFGANG BODE
 HERMANN GIES
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