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U. S. WORKING DRAFT
ON
REPOSITORY PHYSICAL DESCRIPTIONS
IN
A SALT FORMATION

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*Prepared in Support of
INFCE Discussions
at the Direction of the*
DIVISION OF WASTE MANAGEMENT
U. S. DEPARTMENT OF ENERGY

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May 1976

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MAY 1978

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1.0 INTRODUCTION

This preliminary report presents a physical description of a reference deep-geologic repository located in a bedded salt host rock at a depth of about 600 meters. Two reference repository designs are described in this report. One is designed to accept wastes from a light-water reactor (LWR) strategy in which no reprocessing occurs, and the spent fuel itself is considered a waste form. The second is designed to accept wastes from an LWR strategy with reprocessing, in which uranium and plutonium are recycled.

1.1 RADIOACTIVE WASTE ISOLATION IN GEOLOGIC FORMATIONS

The objective of placing radioactive waste into a geologic rock unit is to contain and isolate the waste so that radioactive nuclides from the waste cannot migrate to the biosphere in amounts that will endanger the public welfare. The most important waste form from an LWR strategy in which no reprocessing occurs is the spent fuel itself. The spent fuel will contain all of the volatile and non-volatile fission products: uranium, plutonium, and all other transuranic elements generated in the irradiated fuel elements. On the other hand, the most important waste form from an LWR strategy with reprocessing is the high-level waste (HLW). This high-level waste will contain about 0.1 percent of the iodine, essentially all of the other non-volatile fission products, about 0.5 percent of the uranium and plutonium, and essentially all of the other transuranic elements generated within the irradiated fuel assemblies. For purposes of this repository description, it is assumed that the HLW will be fixed within a glass matrix.

During the initial years of existence of the spent fuel or HLW, the radioactivity associated with the fission products is greater than that associated with the transuranic elements. However, most of these fission products decay at a faster rate than the transuranics, and, after a few hundred years, the radioactivity of the transuranic elements

becomes the most important consideration for isolation. The presence of significant quantities of these long-lived transuranic elements (e.g., ^{237}Np , half-life 2.1×10^6 years; ^{239}Pu , half-life 24,413 years; ^{240}Pu , half-life 6580 years, ^{243}Am , half-life 7340 years) requires waste isolation from the biosphere for thousands of years.

1.2 BASELINE ASSUMPTIONS FOR RADIOACTIVE WASTE ISOLATION IN DEEP GEOLOGIC FORMATIONS

1.2.1 Light-Water Reactor Strategies

Two light-water reactor (LWR) strategies are considered in this report: an LWR strategy with spent fuel disposal, and another with reprocessing to accomplish uranium and plutonium recycle.

1.2.1.1 Spent Fuel Disposal Strategy

In the spent fuel disposal strategy, spent nuclear fuel is discharged from a reactor and is stored within a spent fuel pool for an interim cooling period prior to disposal. After a sufficient cooling period, the spent fuel is placed in canisters and transported to the repository for terminal storage. If the spent fuel packaging facility is located very near the repository, the canisters will be transferred to the repository in a transfer cask. If the packaging is far enough away from the repository that shipment over public transportation routes is involved, then the canisters of spent fuel will be transported in a shielded shipping cask suitable for this purpose.

The only waste form from this LWR strategy which at this time requires terminal storage at a deep-geologic repository is the spent fuel itself. However, for this repository description, a small underground area has arbitrarily been assigned to provide a capability for receiving a nominal quantity of low level waste. Separate spent fuel and low-level waste handling facilities have been provided in this repository design.

1.2.1.2 Reprocessing Strategy

The reprocessing strategy examined in this report involves the recycle of both uranium and plutonium. The spent fuel from LWR's is chemically reprocessed to recover both of these elements. The recovered uranium is then re-enriched in U-235, fabricated into fresh nuclear fuel, and returned to reactors for the production of more electrical power. The recovered plutonium is converted to plutonium dioxide, mixed with uranium dioxide, fabricated into fresh fuel, and also returned to reactors for the production of more electrical power. This latter fuel is often referred to as mixed oxide or MOX fuel.

The radioactive, transuranic-contaminated wastes generated by this complete recycle are diverse in nature. They are the result of the operation of reactors, reprocessing plants, plutonium nitrate to plutonium oxide conversion facilities, plutonium fuel fabrication facilities, and support activities such as transportation, storage, and decommissioning of major facilities. The wastes are categorized as high-level, medium-level, and low-level wastes. These wastes arrive at the repository in canisters or drums. Facilities are provided at the repository to handle each type of waste.

1.2.2 Host Rock Unit - Salt

The reference repositories described in this report are designed for a bedded salt formation. In general, a bedded salt formation consists of salt layers interbedded with other sedimentary rock types.

The important characteristics of salt which make it a potential rock type for isolation of nuclear wastes include: 1) relative abundance in remote locations; 2) relatively high thermal conductivity, which helps to dissipate the heat generated by the emplaced nuclear waste; 3) ease of excavation; 4) impermeability; and 5) plasticity. Salt deforms readily without fracturing. Other factors to be considered in siting a waste repository include the following: 1) salt is highly watersoluble; 2) in the presence of water, salt is corrosive to many materials such as the steels used for waste canisters; and 3) salt often exists in proximity to oil and gas fields, which, with their

attendant drilling operations, pose an added consideration of desirable resources in siting a repository in salt.

1.2.3 Other Major Repository Design and Operational Assumptions

These repository descriptions are based on the existence of a steady-state, nuclear power generating capacity. Three levels of installed capacity have been employed as design bases: 10, 50, and 100 GWe installed capacity operating at 70% load factor and consisting of 35% BWR's and 65% PWR's. Emphasis has been placed on the 100 GWe installed capacity in deriving the basic repository concepts which were scaled down or modified to generate the repository descriptions that would support 10 or 50 GWe installed capacity.

Other assumptions for an equilibrium operation that affect repository design include: 1) waste receipt assumptions; 2) basic repository operational assumptions; and 3) the assumption of no waste retrievability requirement. These assumptions are discussed below.

1.2.3.1 Waste Assumptions

All wastes from the reactor strategies described in Section 1.2.1 are assumed to arrive at the repository in suitable stainless (or carbon) steel canisters or carbon steel drums. The repository design is based on receipt of solid, noncombustible wastes only. If a damaged or excessively contaminated canister is received, such a canister would be overpacked at the repository. No other packaging operations would be required.

The wastes are assumed to arrive at the repository at various ages, that is, at various intervals after the fuel from which the wastes are generated is discharged from a reactor. In the spent fuel disposal strategy, the spent fuel is assumed to be 10 years old at receipt, and the low-level waste is assumed to be 1 year old. Similarly, in the reprocessing strategy, the high-level waste is assumed to be 10 years old, and the cladding, medium-level, and low-level wastes are assumed to be 1 year old.

The age of the cladding, medium-level, and low-level wastes have little or no affect on repository design as long as the level of activity in and on a waste container falls within the range of design values. The age of the spent fuel or high-level waste received at the repository would affect design. The length of the cooling period for these wastes prior to their receipt at the repository determines the heat generation rate of each waste container and, consequently, the amount of heat that will be transmitted to the formation. See Section 3.3 for further discussion.

1.2.3.2 Repository Operational Assumptions

It is assumed that the repository is receiving either spent fuel or reprocessing wastes at a constant rate. This rate is defined by the installed nuclear reactor capacity (GWe) being supported by the repository. Simultaneously with waste handling operations, mine excavation operations are preparing new rooms for waste emplacement, and rooms filled with waste are being backfilled with some of the excavated rock. These latter two operations also are assumed to proceed at constant rates consistent with the assumed steady-state operation.

As indicated above, all wastes are in appropriate containers when they arrive at the repository. All waste handling would be remote or from appropriately shielded vehicles. Although waste emplacement areas are at slightly different depths in this design, no waste emplacement area is allowed to overlie another emplacement area. Each type of waste is placed in an area designated for that specific type of waste (Appendix C for further discussion).

The surface and subterranean acreage requirements for the repository depend on the amount of the installed capacity and the length of time of the support. The surface acreage can be discussed in terms of three regions: 1) the fenced area in which all buildings and surface support facilities are located; 2) an inner controlled area overlying the extent of the subterranean facilities; and 3) an outer controlled area of sufficient extent to ensure no accidental breach of the repository.

1.2.3.3 Waste Retrievability

No waste retrievability requirement is assumed for any of the spent fuel or reprocessing wastes considered in these repository descriptions. Consequently, after a repository room is filled with waste, it will be backfilled immediately or as soon as is operationally expedient.

2.0 REPRESENTATIVE BEDDED SALT SITE DESCRIPTION

This section discusses the bases for selection of the representative bedded salt site for a commercial radioactive waste repository and describes the representative salt site that was chosen. Section 2.1 discusses the basic geologic considerations for selection of a host rock for a waste repository, while Section 2.2 discusses the characteristics of salt that are of interest for waste repository considerations. The representative salt site description is discussed in Section 2.3.

2.1 GENERAL GEOLOGIC CONSIDERATIONS RELATED TO SITE SELECTION

The objective of placing radioactive waste into a geologic rock unit is to contain and isolate the waste in such a way that radioactive nuclides from the waste cannot migrate to the biosphere in amounts that will endanger the public welfare. This goal may be met in different manners at different sites because of inherent differences in geological environments. Accordingly, each prospective repository location must be carefully studied and evaluated in order to comprehend the unique geologic aspects particular to that specific site. The relative suitability of a site for waste disposal can be determined only after these characteristics are adequately understood.

Some geologic factors, however, must meet certain broad requirements before a site can be seriously considered. These criteria are adaptable to any geological setting or region and to any rock unit. General descriptions of these requirements are presented in the following portions of this section.

2.1.1 Depth of Disposal Horizon

A disposal horizon should be located below ground surface at a depth sufficient to isolate wastes and the encapsulating rock unit from phenomena such as fluvial and wind erosion, near-surface circulating groundwater, glacial activity, nuclear warfare, meteorite impacts, and weathering processes. A preliminary estimate of minimum depth is approximately 300 meters.

The maximum depth of the disposal horizon will be governed by the relationship of the lithostatic pressures (the weight of the rock column above the storage facility) to the overall host rock strengths. Lithostatic pressures increase as depth increases. Therefore, with increased depth these pressures will begin to approach the crushing strength of the host rock, and the stability of the repository openings will be endangered by salt creep. From site to site, the lithostatic pressures and host rock strength will vary considerably. Accordingly, the maximum depth at which the disposal facilities could be placed will be site dependent. However, historical experience in potash and salt mines indicates that, at depths greater than about 900 meters, the rate of creep closure in salt begins to increase significantly and may pose safety-related problems because of the great stress caused by lithostatic pressure.¹

2.1.2 Thickness and Lateral Extent of Disposal Horizon

The host rock should have sufficient thickness and lateral extent to provide adequate space to develop and operate the repository, with a margin of undisturbed rock allowed on all sides. In addition, the repository rock unit should be designed with dimensions sufficient to preclude breaching of geological containment by radioactive heat production and possible rock failure or cavity closure after the wastes are emplaced. Relevant studies² indicate that a minimum thickness of approximately 60 meters is considered adequate. However, greater thicknesses, if available, would be more desirable from the standpoints of rock mechanics and hydrology.

2.1.3 Host Rock Properties

The repository site should be located in a host rock unit of low permeability, porosity, and water content. The low permeability of the host rock formation will minimize groundwater movement in and near the repository and create the maximum possible isolation from the groundwater flow systems which could transport and release radionuclides to the biosphere. The porosity and water content should be sufficiently low that the water within the host rock will not be affected chemically

or thermally by the waste-generated heat on a scale large enough to compromise geological containment and operational safety.

The chemical and radiological properties of the host rock should be such that reaction to the stored waste and containers will not jeopardize the geological containment and operational safety of the repository.

Experimental results to date indicate that bedded salt is a suitable host rock for radioactive waste storage.^{2.3}

2.1.4 Hydrogeology

The repository should be located in a region where the groundwater hydraulic gradients are low. This location will aid in reducing the rate at which groundwater will flow through overlying and underlying strata. In addition, the repository should be situated sufficiently distant from any point at which the site groundwater flow system discharges to the biosphere or is utilized by man.

2.1.5 Tectonic Stability, Faulting and Seismicity

A long history of tectonic stability is a prerequisite for a potential repository site. The rate and amount of predictable longterm regional uplift and/or subsidence of bedrock should not pose a threat to the physical integrity of the repository or any overlying rock layers. Faults should not exist at the site or, if present, should have no characteristics that would adversely affect the operational safety or geological containment. The bedded salt repository should not be sited in or near an area in which igneous or volcanic activity has occurred during the post-Pliocene (a period dating from the present to about one and a half to two million years ago). Also, the site should be within a region having a seismic activity level low enough so as not to pose a threat to the safety of the operations or to geological containment. Such a location would be expected to be in a region remote from recorded or historic earthquakes of greater than "moderate" intensity. It should be clearly demonstrated that any structural features found onsite are not associated with potentially active features in adjacent regions.

2.1.6 Relationship to Natural Resources

It is desirable to locate the repository in an area where there has been and will be little interest in the existing natural resources. In addition, it is desirable to minimize man's activities which could endanger the short- and long-term integrity of the waste facilities, such as, borings, mining, well drilling, and land development in the area surrounding the site.

2.1.7 Summary of Geologic Considerations

The important characteristics of salt which make it a potential rock type for isolation of nuclear wastes include: 1) relative abundance; 2) relatively high thermal conductivity which helps to dissipate the heat generated by the emplaced nuclear waste; 3) ease of excavation, salt mining being one of the most efficiently operated underground extractive industries; 4) impermeability; and 5) plasticity, which helps to maintain the impermeability because salt deforms readily without fracturing. Any fractures that could form would tend to "heal." Other important properties of salt include: 1) salt is readily soluble in water; hence, potential dissolution by an adjacent aquifer should be taken into account; 2) in the presence of water, salt is corrosive to many materials such as steel waste canisters; and 3) salt beds formed in the same paleoenvironment in which hydrocarbons formed; thus, oil and gas fields often exist nearby.

2.2 REPRESENTATIVE SITE DESCRIPTION

2.2.1 Stratigraphy

In order to satisfy the geologic guidelines related to the stability of the repository room and long-term containment (Sections 2.1.1 and 2.1.2), a bed of salt at least 60 meters thick above the 600-meter depth will be needed to house the repository. A section that is representative of the general stratigraphy of a bedded salt repository is shown in Figure 2-1.

This reference stratigraphic section was derived by a step-by-step method using data on well-known salt basins. First, using the

depth and thickness requirements, certain basins were eliminated from consideration. Next, representative sections for the remaining areas were developed by combining several sections from published literature into a composite section for each area. Finally, the reference stratigraphic section was obtained by subjectively integrating the composite sections of the representative basins. The sequence of formations in the reference section was found to occur in most of the basins examined. The reference stratigraphic section is considered a typical evaporite cycle⁴.

2.2.2 Hydrogeology

The reference bedded salt section is assumed to be located in a large undeformed geologic sedimentary basin. The strata are nearly flat-lying, and the regional dip is less than one degree. The recharge areas are not topographically much higher than the discharge areas; therefore, the regional groundwater gradient is extremely low.

The reference section contains a sequence of clastic and/or evaporative rock formations with varying permeabilities and porosities. These water-bearing formations, which lie above and under the repository host rock, contain and transmit various quantities of groundwater depending upon the regional gradient and certain hydrogeologic parameters such as porosity, permeability, and formation thickness. The values for these parameters vary considerably from site to site, but are fixed for the reference stratigraphy. A chart listing the ranges of values obtained from a literature search and past experience is shown in Table 2-1.

2.2.3 In-situ Rock Stresses

Due to the rheological character of salt, a lithostatic stress field (equal in all directions) equivalent to the weight of rock above the salt would normally be anticipated. The reason is that, provided sufficient time has elapsed after any tectonic stress has been applied, the salt will creep and readjust until lithostatic equilibrium is reached. Accordingly, the horizontal and vertical in-situ stresses at the repository level of 580 m are assumed to be on the order of 120 to 135 kg/cm².

2.2.4 Host Rock Properties

When considering rock properties, it is important to distinguish between the properties of a small intact piece of rock (dimensions typically in centimeters) and the properties of the rock mass (dimensions typically in hundreds of meters). Intact properties are usually determined in the laboratory, whereas rock mass properties are evaluated in large-scale field tests or, more generally, are estimated in some manner by using the intact properties and the rock mass structure as a basis, as discussed below.

2.2.4.1 Properties of Intact Salt

In order to obtain an understanding of the range of property values of intact salt, a literature search was used as the basis. The types of properties reviewed included index properties, stress-strain relationships, and strength and thermal properties. Means were calculated for each of the properties and maximum and minimum values were determined. The results of this survey are shown on Table 2-2.

2.2.4.2 Rock Mass Properties

Although bedded salt formations may be several hundred meters in overall thickness, they usually consist of irregular sequences of clear to white halite with thin laminae of shales, siltstones, clay, gypsum, and anhydrite. The salt layers range from a few centimeters to tens of meters in thickness. The significant features of the salt are layering of clear and cloudy salt, seasonal layering of shale and anhydrite, with occasional thicker beds of shales, anhydrite and other materials.

Basin salts are characterized by gentle dips. However, they can be folded or faulted. Jointing or fracturing is commonly due to partings along impurity layers. Otherwise, jointing is rare because salt is a relatively plastic material, and joints within the halite tend to seal themselves. For this reason, salt beds are generally impervious to groundwater, since there are few vertical joints through which the water can penetrate. A summary of the structures found in a typical salt formation is included in Table 2-3.

Due to the relative lack of structure in salt compared to other rock types, the rock mass properties do not vary markedly from the intact properties. However, some variations do occur. Based on the methods described above, rock mass properties for the reference host salt rock have been estimated and are listed in Table 2-4.

2.2.4.3 Environmental Factors

The properties of the reference salt formation will be affected by certain environmental factors, including time, temperature, pressure, moisture, and irradiation.

Time. A notable characteristic of salt is its creep behavior with time, where creep is defined as viscous flow under constant stress.

Several stages of creep have been delineated, which depend upon the time span of the observations as well as the magnitude of the applied stress (Figure 2-2). For instance, when the material is initially stressed, an immediate elastic strain is followed by primary creep, in which the strain-time curve is concave downward. This is followed by secondary or steady-state creep, in which the curve has an approximately constant slope. Finally, the tertiary creep corresponds to accelerating strain and leads rapidly to failure. In general, the primary and tertiary stages are very short, usually less than one day, while the secondary stage is comparatively long and depends upon the magnitude of the applied load. From experiments carried out in the laboratory, a number of empirical equations have been developed to describe the creep behavior of salt. The important point, however, is that salt does creep, and a repository cannot be rationally designed unless the creep behavior under the appropriate conditions of pressure and temperature are properly understood.

Temperature. The physical behavior of salt is affected by temperature. For instance, increased temperature increases the rate of creep and decreases the value of Young's modulus. Experiments carried out by Lomenick and Bradshaw⁵ demonstrated that a rise in temperature from 22.5°C to 100°C increased the strain by a factor of seven. The drop in compressive strength with increase in temperature is not as marked, but is reduced by approximately 10% between 22.5°C and 200°C.

The thermal conductivity and heat capacity also decrease with increasing temperature (Table 2-2).

Pressure. The effects of pressure on the properties of salt have not been extensively investigated or reported. However, it is expected that for an increase in deviatoric stress, the creep rate of the salt will increase.

Water/Moisture. Moisture is often trapped internally in vesicles and along partings. Most of the moisture is brine. The trapped brine can be released with considerable energy when heated, and can fracture the rock. In laboratory tests, small salt samples from mines in Kansas fractured when heated to temperatures of 260° to 320°C⁵.

Irradiation. The effect of irradiation on bedded salt was studied by Bradshaw, et al.⁶ It was concluded that radiation exposure doses of 5×10^8 rems produce some changes in structural properties, but the effect on mine stability should be negligible.

2.2.4.4 Summary of Host Rock Properties

In summary, salt has certain characteristics which distinguish it from other rock types. The imperviousness, low moisture content, creep behavior, self-sealing properties, high-thermal conductivity, and lack of fracturing make salt an excellent candidate host rock for isolation of radioactive waste.

2.0 REFERENCES

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FIGURE 2-1 REPRESENTATIVE STRATIGRAPHIC SECTION FOR BEDDED SALT

GENERALIZED CREEP CURVE FOR SALT
UNDER CONDITIONS OF CONSTANT STRESS

FIGURE 2-2

TABLE 2-1

HYDROLOGIC PARAMETERS
FOR GENERIC STRATIGRAPHY - BEDDED SALT

<u>ROCK TYPE</u>	<u>GENERIC HYDRAULIC CONDUCTIVITY</u> (cm/sec)	<u>HYDRAULIC CONDUCTIVITY RANGE</u> (cm/sec)	<u>POROSITY</u> (%)
Terrace Deposits Glacial till	6.8×10^{-6}	$1 \times 10^{-7} - 1 \times 10^{-3}$	20
Calcareous Shale Partly Sandy	1.2×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-3}$	13
Sandstone and Dolomite	8.1×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-2}$	20
Shale Partly Calcareous Some Anhydrite	1.2×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-3}$	13
Cherty Limestone Dolomite, some Anhydrite	5.8×10^{-5}	$5 \times 10^{-8} - 1 \times 10^{-3}$	20
Dolomite	4.6×10^{-5}	$5 \times 10^{-8} - 1 \times 10^{-2}$	30
Salt	Nil	$2 \times 10^{-21} - 1 \times 10^{-8}$	0.5
Interbedded Dolomite and Shale	6.8×10^{-6}	$1 \times 10^{-10} - 1 \times 10^{-3}$	20
Shale	7.1×10^{-7}	$1 \times 10^{-10} - 1 \times 10^{-3}$	16

TABLE 2-2 PROPERTIES OF INTACT SALT

Type of Property	Parameter	Mean	Range	Metric Units
Index	Unit Weight	2080 (39)*	1650-2400	kg/m ³
	Natural moisture content	.08 (12)	0-1.1	%
	Porosity	3.3 (11)	1.4-10.0	%
Stress-strain	Young's Modulus	1.27×10^5 (18)	6.3×10^3 - 5.1×10^5	kg/cm ²
	Poisson's Ratio	0.34 (12)	0.22-0.50	
Strength	Uniaxial Compressive Strength	250 (16)	160-510	kg/cm ²
	Tensile Strength	16 (15)	8-320	kg/cm ²
Thermal	Coefficient of Linear Thermal Expansion	39×10^{-6} (4)	36×10^{-6} - 40×10^{-6}	°C ⁻¹
	Heat Capacity	0.28 (4)	0.19-0.47	cal/gm-°C
	Thermal Conductivity	TEMPERATURE	CONDUCTIVITY	
		°C		cal/sec-cm-°C
		50	1.5×10^{-2}	
		100	1.2×10^{-2}	
		150	1.0×10^{-2}	
		200	8.6×10^{-3}	
		300	7.4×10^{-3}	
		400	5.9×10^{-3}	
			5.0×10^{-3}	

* Number of tests used to derive mean in parenthesis

TABLE 2-3

PARTIAL DESCRIPTION OF TYPICAL STRUCTURES IN BEDDED SALT

	<u>JOINT SET A</u>	<u>JOINT SET B</u>	<u>JOINT SET C</u>
Type of Joint	Bedding	Joint	Joint
Filling	Clay or Anhydrite	Calcite/ Anhydrite	Calcite Anhydrite
Orientation (Dip Jir/Dip)	Horizontal	Dip 80-90°	Dip 80-90° Perpendicular to Set B
Opening Width	Tight	0	0
Spacing	Max. 3/m	Rare	Rare
Waviness	0	Slight	Slight
Extent of Joints (Lengths)	3 to 15m	0.3 to 9m	0.3 to 9m

TABLE 2-4 ESTIMATED ROCK MASS PROPERTIES FOR GENERIC SALT

<u>Type of Property</u>	<u>Parameter</u>	<u>Value</u>	<u>Metric Units</u>
Index	. Unit Weight	2100	kg/m ³
	. Natural moisture content	0.08	%
	. Porosity (rock mass)	0.5	%
Stress-strain	. Young's modulus	1.1x10 ⁵	kg/cm ²
	. Poisson's ratio	0.35	---
Strength	. Unconfined Compressive Strength	230	kg/cm ²
	. Tensile Strength	3	kg/cm ²
Thermal	. Coefficient of Linear Thermal Expansion	40 x 10 ⁻⁶	°C ⁻¹
	. Heat Capacity		
	<u>TEMPERATURE</u>	<u>HEAT CAPACITY</u>	
	0°C	0.20 cal/gm-°C	
	100	0.21	
	200	0.22	
	. Thermal Conductivity		
	<u>TEMPERATURE</u>	<u>CONDUCTIVITY</u>	
	0°C	1.5x10 ⁻² cal/sec-cm°C	
	50	1.2x10 ⁻²	
	100	1.0x10 ⁻²	
	150	8.6x10 ⁻³	
	200	7.4x10 ⁻³	
	300	5.9x10 ⁻³	
	400	5.0x10 ⁻³	

3.0 LWR WASTES TO BE STORED IN THE REFERENCE REPOSITORY

Nuclear wastes arise from operations at various facilities in the nuclear fuel cycle. The amounts of waste generated are computed for three cases of steady-state operation of installed nuclear capacity (10, 50, and 100 GWe) operating at 70% load factor. The quantities of waste generated, stated in the following sections, take into account the 70% load factor for nuclear capacity.

3.1 DESCRIPTION OF WASTE CONTENTS AND FORMS

Two light-water reactor (LWR) strategies are considered as the bases for these repository descriptions. Figure 3-1 graphically presents the major activities of each of these strategies. Reactor strategy 1 (Spent Fuel Disposal) involves the direct disposal of LWR spent fuel. No reprocessing occurs. The intact spent fuel assemblies, with their inventories of uranium and plutonium, are packaged in canisters (prior to delivery to the repository) and disposed of in the repository. The repository for spent fuel is also capable of handling a nominal quantity of low-level waste. Reactor strategy 2 (Uranium and Plutonium Reprocessing) involves the recycle of uranium and plutonium for further use as fuel. Mixed oxide fuel (MOX) resulting from this recycle activity consists of a mixture of UO_2 and PuO_2 . For purposes of these discussions, it is assumed that, in the steady-state operation of this LWR strategy, all LWR's are fueled with a 2:1 ratio of MOX to UO_2 fuel assemblies. The waste arising from reactor strategy 2 includes high-level waste (HLW), cladding waste (CW), medium-level waste (MLW), and low-level transuranic (LL-TRU) waste. This waste will be packaged at the reprocessing plant and disposed of in the repository.

The following sections describe the waste contents and forms for each of these reactor strategies as used in the repository design.

3.1.1 Spent Fuel Strategy (Reactor Strategy 1)

3.1.1.1 Spent Fuel Assemblies

A typical LWR fuel assembly is composed of Zircaloy-clad fuel rods and metallic hardware that hold the rods in the assembly matrix and/or

direct the coolant flow through the fuel assemblies. Characteristic dimensions and other parameters of pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies are listed in Table 3-1.

The spent fuel itself is composed of fission and activation products produced during fuel burnup, as well as fissile and fertile uranium and neutron-induced plutonium. The fission and activation products include certain radioactive volatiles: noble gases, halogens, tritium, and carbon-14. Most halogen and noble gas radionuclides have very short half-lives and decay to negligible quantities before the spent fuel is received at the repository (10 years after reactor discharge). Moderate to long-lived isotopes, such as krypton-85 (10.7 yr half-life), tritium (12.3 yr) and carbon-14 (5730 yr), are present in the waste at the time of burial. In addition to these fuel wastes, the cladding and other structural components of the assembly would be radioactive because of neutron activation of the metal and various corrosion products. The radioactivity of these structural materials is on the order of 3 percent of the total radioactivity in the spent fuel assembly. The total radioactivity of spent fuel is approximately 6.4×10^5 Ci per metric ton of uranium charged to the reactor. This activity corresponds to 10-year cooled spent fuel which is the design age of spent fuel to be disposed of in the repository. A PWR spent fuel assembly contains approximately 0.46 metric tons of uranium with a total activity of about 2.9×10^5 Ci at 10 years. A BWR spent fuel assembly contains approximately 0.18 metric tons of uranium for a total activity of about 1.2×10^5 Ci at 10 years.

Spent fuel arisings are based on a waste generation rate of 25 metric tons of heavy metal per year per gigawatt of installed nuclear capacity (25 MTHM per yr per GWe), with 65 percent PWR and 35 percent BWR; therefore, 25 MTHM corresponds to about 35 PWR spent fuel assemblies and about 49 BWR assemblies.

3.1.1.2 Low-Level Waste

Low-level wastes are generated in the spent fuel reactor strategy. Although there is no present requirement to ship these wastes to a repository, a modest amount of this waste (14 m³ per yr per GWe) has arbitrarily been selected to accommodate customer requests for dispo-

sal at the repository rather than at a burial ground. In addition, this provides for the disposal of LLW generated in the repository surface facilities during waste handling operations. For simplicity, this waste is assumed to be one year old upon arrival at the repository.

3.1.2 Reprocessing-Waste Strategy (Reactor Strategy 2)

3.1.2.1 High-Level Waste

High-level wastes result from aqueous waste generated during reprocessing of irradiated reactor fuels.¹ These wastes are estimated to contain essentially all of the non-volatile fission products, about 0.1 percent of the iodine, about 0.5 percent of the uranium and plutonium, and all of the other actinides originally contained in the irradiated fuel.^{2,3} Because HLW contains actinides, it may also be referred to as transuranic (TRU) waste. The letters TRU designate those wastes which contain more than a specified amount of plutonium or other long-lived alpha emitters. Presently, this specified amount is assumed to be 10 nCi/g of long-lived alpha emitters. This waste, arising from the direct reprocessing of reactor fuel, would contain the same proportion of MOX to UO₂ waste as was originally charged to the reactor (assumed to be 2:1 for these repository descriptions).

HLW is solidified prior to arrival at the repository, and for this study the form was taken to be a borosilicate glass matrix, produced at a rate of 2.1 m³ per yr per GWe. It will be solidified at a specific volume of 0.09 m³/MTHM* in its final glass form. This specific volume results in thermal loadings consistent with thermal guidelines for the repository design in Section 5.0. The activity of HLW is 2.3 x 10⁶ Ci/m³ for 10-year old waste (age upon arrival at the repository).

3.1.2.2 Cladding Wastes

Cladding wastes consist of the fragments of zircaloy, inconel, and stainless steel which remain after the fuel is dissolved, and of

*Refers to the volume of waste generated per metric ton of heavy metal (MTHM) fuel charged to the reactor.

other structural components of the fuel assemblies.^{2,4,5} In addition to neutron-induced radioactivity, the cladding waste has been estimated to contain 0.05% of the actinides and 0.05% of the non-volatile fission products in the spent fuel.² As agreed upon by the INFCE working group 7, the waste may contain up to 0.3% of the plutonium originally in the spent fuel and, because of induced gamma activity, will need biological shielding equivalent to several inches of lead. Since the hulls and hardware are contaminated with transuranics, they may also be referred to as TRU wastes.

Several processes are being considered that will immobilize and stabilize these wastes. The selected process is one in which the final waste product is compacted and is not pyrophoric or combustible. In compacted form, cladding wastes are produced at a rate of 7 m³ per yr per GWe of installed capacity. This waste form will be one-year old upon receipt at the repository.

3.1.2.3 Medium-Level Wastes

The category of medium-level wastes (MLW) designates those wastes, other than high-level and cladding wastes, that arise during reprocessing and require biological shielding or other protective action during handling. They contain long-lived alpha emitters in concentrations greater than 10 nCi/g and can therefore be classified as TRU wastes. The estimated uncompacted volume of the MLW produced is 25 m³ per yr per GWe of installed nuclear capacity. MLW will be one year old upon receipt at the repository.

3.1.2.4 Low-Level TRU Wastes

Low-level TRU wastes are defined as those solid materials that contain plutonium or other long-lived alpha emitters in concentrations greater than 10 nCi/g and yet have sufficiently low external gamma radiation levels to allow them to be handled directly without supplementary shielding (e.g., surface dose rates less than 10 mrem/hr).² LL-TRU wastes will arise primarily at fuel fabrication plants at a rate of 35 m³ per yr per GWe of installed capacity. Several processes are being considered for stabilizing and immobilizing these wastes. The selected process will most likely be one in which the wastes are not combustible and, contain no significant amounts of organic materials.

For INFCE purposes the waste is assumed to be stabilized in concrete, and is assumed to be one year old upon receipt at the repository.

3.2 DESCRIPTION OF WASTE PACKAGING

All wastes associated with reactor strategies 1 and 2 as described in Section 3.1 will be packaged prior to arrival at the repository. In reactor strategy 1, spent fuel will be packaged in canisters and low-level waste will be packaged in drums. In reactor strategy 2, high-level waste, cladding waste, and medium-level waste will be packaged in canisters. Low-level waste will be packaged in drums.

In reactor strategy 1, if the spent fuel is packaged at a facility very near the repository, the canistered spent fuel will be transferred to the repository in a transfer cask. If the spent fuel is packaged at a facility far enough from the repository that shipment over public transportation routes is required, then the canistered spent fuel will be transported in a shielded shipping cask appropriate for such conditions. The canisters and drums will arrive at the repository by truck or rail in an appropriate transportation cask or drum cargo container.

Overpacks are provided for waste packages that arrive at the repository with leaks, or for those that are damaged during handling. Overpacks will be capable of completely enclosing the damaged waste package and will be welded shut.

3.2.1 Reactor Strategy 1 (Spent Fuel Disposal)

3.2.1.1 Physical Description of Waste Packages

The waste canister consists of a cylindrical body enclosed by welded heads at the top and at the bottom. The bottom head is concave and is covered by a metal skirt to permit free standing of the canister on a flat surface. A standard lifting device is located at the top of the canister for remote lifting and handling of all canisters. All canister construction materials will be either stainless or carbon steel. These canisters are packed with 1 PWR assembly or 3 BWR assemblies. The canister dimensions are presented in Table 3-2.

The low-level waste drum is a standard 200-liter (55-gallon) drum with an outside diameter of 60 cm and an overall length of about 88 cm. The drum is fabricated of 16-gauge (0.16 cm thick) carbon steel. The anticipated weight of a full waste drum would not exceed 400 kg. The waste drums are handled by conventional industrial techniques. Drums are stacked on non-flammable pallets when they arrive at the repository and will remain on these pallets during transfer and storage at the repository.

3.2.1.2 Physical Description of Overpacks

Should the canisters or waste drums violate acceptance criteria for the repository upon arrival, or be damaged during handling at the repository, they will be overpacked to assure waste confinement during handling and emplacement operations. Canister and drum failures are considered low-probability events, so the need for and use of overpacks is expected to be minimal. The dimensions of these containers may be altered as a result of additional study. The values presented here appear reasonable and are readily available, but have not been subjected to optimization for supporting a specific nuclear power installed capacity.

Waste canister overpacks consist of a cylindrical body enclosed by welded heads at the top and bottom. The overpacks will be constructed from carbon steel. An overpacked canister would also be handled remotely. The dimensions of the canister overpacks will allow easy installation over the damaged canister, but also maintain a close fit to minimize canister movement that could result in further damage to the canister. Canister overpack dimensions are listed in Table 3-2.

The drum overpack consists of a standard 300-liter (80-gallon) drum with welded flat heads at the top and bottom. All construction materials are to be carbon steel. The drum overpacks will be handled by conventional industrial drum-handling techniques and will be palletized prior to transfer to the storage area. The drum overpack has an outside diameter of about 65 cm and an overall length of about 91 cm.

3.2.1.3 Physical Description of Waste Shipping Containers

The shipment of canistered spent fuel will be accomplished by one of two methods, depending on the location of the facility in which the

spent fuel is packaged. If the spent fuel is packaged in a nearby facility, the canisters will be transported to the repository in shielded transfer casks that are bottom-loaded. These transfer casks are identical to those envisioned for the transfer of canisters down the repository shaft to be loaded onto the waste transporter for transfer to the repository room where the canisters will be placed in holes. If the spent fuel is packaged at a plant that is not in the immediate area, the canisters will arrive in shielded shipping casks. Waste receiving facilities at the repository will be capable of receiving either transfer or the shipping casks by truck or rail.

The reference shipping cask is approximately 6 meters long by 3 meters in outside diameter and weighs about 100 metric tons when fully loaded. Each shipping cask will be equipped with its own lifting yoke that is suitable for crane hook operations. The loaded weight of a cask and its yoke will be less than 125 metric tons. The cask is shipped in the horizontal position, and is placed in the vertical position by the overhead crane for removal from the transporting vehicle.

The cask cavity is assumed to be dry and filled with helium prior to shipment. The cavity diameter is about 1.2 m, with a length of about 5.1 m. With appropriate racks installed, the shipping cask can hold 3 BWR spent fuel canisters (9 BWR assemblies), or 6 PWR spent fuel canisters (6 PWR assemblies).

For the shipment of LLW drums the reference container has cavity dimensions of approximately 1.9 x 1.9 x 4.3 meters long. The containers, when fully loaded with 36 drums stacked vertically on pallets (2 drums high), weighs about 20 metric tons.

3.2.2 Reactor Strategy 2 (Reprocessing Waste Disposal)

3.2.2.1 Physical Description of Waste Packages

The physical description of waste canisters for the packages of HLW, MLW, and cladding wastes are similar to the canisters for spent fuel described in Section 3.2.1.1. Canister dimensions are listed in Table 3.2. LLW drums are similar to the waste drums described in Section 3.2.1.1 for reactor strategy 1.

3.2.2.2 Physical Description of Overpacks

The use of waste package overpacks and their physical descriptions

are similar to those described in Section 3.2.1.2 for reactor strategy 1.

3.2.2.3 Physical Description of Waste Shipping Containers

In this reactor strategy, wastes arrive at the repository in shipping containers approved for public transport. The reference canistered waste cask⁶ is a lead-filled double-walled stainless steel cylinder weighing up to 100 metric tons when loaded. Nine waste canisters can be accommodated in an aluminum insert which fits into the cask cavity. The cask is transported on six-axle rail car designated exclusively for this purpose. The rail car and mounting equipment weigh about 68 metric tons, bringing the total weight of the transport system to about 168 metric tons. The cask is about 4 meters in length and has an external diameter of about 2.1 meters. The cavity diameter is about 1.5 meters and has a length of about 3.35 meters.

Transportation of low-level waste will be accomplished by the use of the shipping container described in Section 3.2.1.3 for reactor strategy 1.

3.3 WASTE PACKAGE THERMAL CONDITIONS

In reactor strategy 1, spent fuel waste generates a sufficient amount of heat to affect repository design and capacity, whereas, in reactor strategy 2, the major heat-generating waste is the high-level waste. In both of these cases, heat generated by the waste originates from the decay of the radioactive nuclides in the waste. The heat-generation histories of these two waste forms are shown in Figure 3-2. Initially, the thermal power of the waste is controlled by the shorter-lived nuclides; therefore, the thermal power decreases rapidly during the first few years after the fuel is removed from the reactor. After several years, the rate of decrease of the thermal power becomes much more gradual as the decay of the longer-lived isotopes become the main heat source. The data in Figure 3-2, indicates that a major reduction in thermal power has occurred by the time that the fuel is ten years old. Because of this reduction, the reference waste for these repository descriptions were chosen to be 10 years old.

Thermomechanical studies have determined specific waste and canister temperature limits which are discussed in Section 5.0. Based on these temperature limits, canister heat loading requirements were developed (Section 5.3). The resulting thermal properties of the waste containers are described in this section.

3.3.1 Reactor Strategy 1

For the disposal of spent fuel, canisters containing 1 PWR or 3 BWR assemblies each have been specified. The heat load for a 10-year old PWR spent fuel assembly is 0.55 kW, and BWR is 0.18 kW. The resulting heat load for canisters of spent fuel is about 0.55 kW/canister for either PWR or BWR assemblies (Figure 3-2). This is an acceptable heat load for canisters in salt (Section 5.0). Low-level wastes associated with this reactor strategy have negligible thermal loading because of their low radionuclide activity. Spent fuel canister surface temperatures will be less than 200°C upon receipt at the repository. LLW drums will be only slightly above ambient.

3.3.2 Reactor Strategy 2

In the reprocessing waste strategy, HLW would be stored in canisters. Each canister contains waste from approximately 1.96 MTHM of reactor fuel. For 10-year old glass HLW, the heat load per can is 3.6 kW (Figure 3-2). This loading coincides with the design value for allowable heat loading described in Section 5.0 for the designated canister design being placed in salt. . The other wastes produced from this reactor strategy (cladding, MLW, and LL-TRU) have negligible thermal loadings compared with the 3.6 kW/canister limit. HLW canister surface temperatures will be less than 250°C upon arrival at the repository. All other wastes will have surface temperatures only slightly above ambient.

3.4 WASTE ARISING

The amounts of waste generated (waste arisings) are based on the production rates and packaging concepts described in previous sections. The waste arisings are presented in Table 3-4 for three cases of steady-state operation of installed nuclear capacity (10, 50, and 100 GWe) of operating at 70% load factor and consisting of 35% BWR's and 65% PWR's.

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FIGURE 3-1 ALTERNATIVE STRATEGIES

..... REACTOR STRATEGY 1
 ————— REACTOR STRATEGY 2

FIGURE 3-2 WASTE THERMAL POWER DECAY

TABLE 3-1

CHARACTERISTIC DIMENSIONS AND WEIGHTS OF UNIRRADIATED
LWR ASSEMBLIES, APPROXIMATE^a

	PWR	BWR
Overall Assembly Length, m.	4.0 ^a	4.5
Cross Section, cm.	21.5x21.5	13.9x13.9
Fuel Element Length, m.	3.8	4.0
Active Fuel Height, m.	3.7	3.75
Void Volume in Fuel Element, cm ³	27 ^b	40
Fuel Element O.D., cm.	0.95	1.25
Fuel Element Wall, cm.	0.06 ^c	0.08 ^c
Fuel Element Array	17x17	8x8
Number of Fuel Elements per Assembly	289	64
UO ₂ , MT per Assembly	0.523	0.214
Uranium Charge, MT per Assembly	0.461 ^d	0.183 ^d
Total Weight, MT per Assembly	0.668	0.279
Zircaloy Weight, MT per Assembly	0.130 ^e	0.057 ^f
Total Metal Weight, MT per Assembly	0.145 ^g	0.065 ^h

a Unless otherwise noted, this information is based on: Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA-76-43, May 1976. Other references could contain other values, but for the purposes of this report these were used as characteristic.

b Lorenz, R. A.; Collins, J. L.; and Manning, S. R., Quarterly Progress Report on Fission Product Release from LWR Fuel for the Period October-December, 1975, Oak Ridge National Laboratory, ORNL-TM-5290, March 1976.

c B&W Standard Nuclear Steam System, PSAR241, Docket STN50481, May 10, 1974. GE Standard Safety Analysis Report, Docket STN50447, July 26, 1977.

d Kisner, R. A., et al., Nuclear Waste Projections and Source Term Data for FY 1977, Office of Waste Isolation/Union Carbide Corp., Nuclear Division, Y/OWI/TM-34, April 1978.

e Includes Zircaloy control-rod guide thimbles.

f Includes Zircaloy fuel-element spacers.

g Includes 10 kg SS nozzles, 5.5 kg Inconel-718 grids.

h Includes SS tie-plates, negligible amount of Inconel springs.

TABLE 3-2
WASTE PACKAGE AND OVERPACK DIMENSIONS

Container	Inside Diameter (cm)	Nominal Wall Thickness (cm)	Overall Length (m)	Active Volume ^d (m ³)
Canister ^a				
HLW	30	1.0	3.0	0.18
MLW	30	1.0	3.0	0.18
Cladding	30	1.0	3.0	0.18
Spent Fuel-PWR	36	1.3	4.9	0.17
Spent Fuel-BWR	41	1.3	4.9	0.07
Drum ^b				
LLW		200 liter		
Overpack ^c				
HLW	35	0.6	3.2	0.31
MLW	35	0.6	3.2	0.31
Cladding	35	0.6	3.2	0.31
Spent Fuel-PWR	41	0.6	5.1	0.67
Spent Fuel-BWR	46	0.6	5.1	0.85
LLW		300 liter drum		

^aMaterial will be stainless or carbon steel.

^bStandard carbon steel drum.

^cMaterial will be carbon steel.

^dIn the case of overpacks, these values refer to total volume.

TABLE 3-3
SHIPPING CASK AND CONTAINER LOADINGS

Waste Type	Waste Packages/ Container
HLW	9 canisters/cask
MLW	9 canisters/cask
Cladding	9 canisters/cask
Spent Fuel-PWR ^a	6 canisters/cask
Spent Fuel-BWR ^b	3 canisters/cask
LLW	36 drums/container

- a. 1 Spent fuel assembly per canister (6 assemblies per cask).
b. 3 Spent fuel assemblies per canister (9 assemblies per cask).

TABLE 3-4
WASTE ARISING

Reactor Strategy	Waste Type	Installed Capacity		
		10 GWe	50 GWe	100 GWe
1 - Spent Fuel	Spent Fuel-PWR	350 canisters/yr.	1750/yr.	3500/yr.
	Spent Fuel-BWR	160 canisters/yr.	800/yr.	1600/yr.
	LLW	680 drums/yr.	3400/yr.	6800/yr.
2 - Reprocessing Waste	HLW	130 canisters/yr.	650/yr.	1300/yr.
	MLW	1400 canisters/yr.	7000/yr.	14000/yr.
	Cladding Waste	390 canisters/yr.	1950/yr.	3900/yr.
	LL-TRU	1750 drums/yr.	8750/yr.	17500/yr.

4.0 DESCRIPTION OF WASTE ISOLATION FACILITIES

A typical repository for Strategy 1 or Strategy 2 wastes consists of a large number of chambers excavated deep within a suitable formation, along with access shafts and various supportive surface structures. Upon arrival at the repository, wastes are handled using special procedures for different waste types. The overall design includes two basic process lines developed to handle the wastes: one for wastes that require shielding and hot-cell facilities (canisters containing spent fuel assemblies, high-level waste, medium-level waste, or cladding) and one for wastes that do not require such protection (drums containing low-level waste). In all cases, the waste containers are inspected for damage that might have occurred during shipping, and remedial action is taken if required. The waste-filled canisters are placed into a hoist cage, lowered into the repository, and emplaced in the appropriate chamber.

The size and complexity of the surface and subsurface facilities depend on the types of waste arisings, their packaging in transport, the quantities received, and the methods of transferring them from the surface to the emplacement location. An artist's conception of a high capacity and complex waste repository (as, for example, for Strategy 2, 100 GWe installed capacity) is shown in Figures 4-1 and 4-2 and a site layout in Figure 4-3.

The surface facilities of the repository, which in the site plan shown occupy approximately 80 hectares of land, are designed to provide for: the receipt and unloading of the waste shipping containers; decontamination and overpack of the containers, if necessary; and the preparation of the waste for the descent to the mine level. In addition to these surface process facilities, the design includes surface support installations such as an administration building, maintenance and warehouse areas, ventilation facilities, radwaste management facilities, and utility systems as well as shafts and hoisting arrangements for transporting wastes, materials, and mine-operating personnel to and from the mine. The underground facilities and equipment receive the wastes, transport them to the

proper storage sites, and place them in storage. The waste-handling facilities allow reserve capacity for any irregularities in delivery schedule. Surge storage capacity is provided in both the above-ground and below-ground facilities.

For the purposes of the design of a reference repository, certain assumptions have been made concerning the way in which waste would be emplaced. These considerations have been discussed in Appendix C.

All repository facilities are designed to protect the public and the operating personnel from radiation and/or contamination hazards, both under normal operations and during emergency or accident situations. Surface facilities containing radioactive materials are designed to maintain their integrity during natural disasters.

The waste packages which are received and stored at the reference repository are described in Section 3.2. Dimensions and quantities of waste arisings from Reactor Strategy 1 (for 10, 50, and 100 GWe installed capacity) are given in Table 4-1. Similar information for Reactor Strategy 2 is given in Table 4-2. Specific receipt considerations are described below.

The reference repository has been designed to receive casks containing waste canisters via truck or rail. Canistered spent fuel from Strategy 1 arrives at the repository in one of two ways. Those spent fuel assemblies that are canistered at a packaging facility near the repository will be transferred to the repository in a shielded transfer cask. Spent fuel assemblies requiring transport over public roads or rail lines will arrive in sealed shipping casks that are in compliance with all applicable regulations. The internal atmosphere of the shipping cask is dry and contains no water in liquid form. If the shipper loads waste canisters into the shipping cask under water, all water is drained from the shipping cask prior to shipment. Air or inert atmosphere in the shipping cask are acceptable at the repository as it is presently shown. Closure and monitoring systems are compatible with the handling system for the shipping casks. A shipping cask contains a number of waste canisters, whereas a transfer cask contains only one canister and is suitable for direct lowering to mine level.

For Reactor Strategy 1, the spent fuel assemblies arrive in sealed metal canisters (one PWR assembly per canister or three BWR assemblies per canister). For Reactor Strategy 2, the canisters contain high level waste (HLW), medium level waste (MLW), and cladding. Canisters which exceed acceptable surface radioactive contamination limits are decontaminated and/or overpacked. All canisters will contain pressurized helium with an absolute pressure of 2 atmospheres.

The low-level waste packed in drums arrives at the repository in special cargo containers. The drums are placed vertically on pallets inside the cargo container, which permits unloading of drums by forklift vehicles. The repository is capable of receiving cargo containers via railroad car or truck. Finally, the atmosphere in the low-level waste cargo container is dry air or an inert atmosphere.

The repository accepts packages containing low-level waste that are in standard size 200-liter drums. Drums that leak may be overpacked to provide additional containment. The radiation level of the waste drums when received at the repository will not exceed 10 mrem/hr total dose rate at the surface of the drum. The waste drums can be handled by conventional industrial drum handling techniques; prior to transfer to the mine storage area, the drums will be removed from the pallets, tested, decontaminated or overpacked (if required) and repalletized.

4.1 FUNCTIONAL FLOW OF WASTE THROUGH FACILITY

4.1.1 Strategy 1 or Strategy 2 Wastes - Shipping Casks Arrival

A functional flow diagram describing the handling of a waste canister that arrives at the repository in a shipping cask is provided in Figure 4-4a, at the end of this section. The handling steps identified on this flow diagram are detailed below, with the numbers preceding each paragraph keyed to the numbers on Figure 4-4a. Handling of spent fuel that arrives at the repository in a transfer cask is described in Section 4.1.2.

(1) Arrival at Site

Waste canisters sealed in high-impact casks arrive at the entrance portal of the facility site via specially designed railroad cars or trucks. These carriers are similar to those now in use in the U.S. for transport of nuclear fuel assemblies.

(2) Preliminary Inspection

The arriving canister and cask are first stationed at a parking area or siding and inspected visually for damage that might have occurred in transit. They are also logged into the recording system of the facility. If visual inspection indicates transit damage, corrective measures are taken.

(3) Cask Preparation and Inspection

After visual inspection, the carrier is moved into an enclosed area where it is monitored externally for surface contamination.

(4) Washdown of Carrier and Cask

The vehicle and cask are washed down to remove road dirt and grime and surface decontamination of the carrier and cask is done if needed. Next the cask tiedowns are removed in preparation for its unloading from the carrier.

(5) Move to Controlled Environment

The over-the-road carrier is next moved through a confinement chamber (airlock) into an inner area of the canistered waste receiving building, where the cask is unloaded.

(6) Unloading the Cask from Carrier

In the cask unloading area an overhead bridge crane is used to lift the sealed cask from the carrier and place it on an at-grade test stand in an upright position.

(7) Test Surface of Cask

The cask identification markings are again verified against shipping documents for content accounting purposes. Then the exterior of the cask itself is re-tested manually for surface contamination and if necessary the exterior is again washed down.

(8) Test Internal Atmosphere of Cask

After the exterior of the cask is shown to be free of contamination, the outer impact lid is removed and the internal atmosphere of the cask is tested for the presence of helium (He), which would indicate a leak in a canister.

(9) Routine Canister Processing

A cask showing no evidence of canister leakage in the purge test is moved by remote controlled mechanical devices to the routine canister processing facilities and is positioned in preparation for canister extraction.

(10) Canister Extraction

The lid of the cask is removed using an overhead bridge crane. This crane will then extract the canisters vertically upward from the opened cask and place them, one at a time, on a stand for subsequent testing.

(11) Inspect Extracted Canisters

After extraction from the cask, each canister is first inspected visually and its markings identified for accounting control. If visual inspection reveals obvious physical damage to the canister, it is re-loaded into the cask and moved with the cask to another area of the building where overpacking can be accomplished. (The visual inspection is carried out by operators stationed behind shielded viewing windows.) If visual inspection reveals no obvious damage, the canister is next weighed and tested for contamination. If surface contamination is discovered, the canister is cleaned externally by steam or water in a wash-down well.

(12) Canister Leakage Testing

After it has been established that a canister has no external contamination, it is lowered into a special, hermetically-sealed well and tested for leakage by evacuating the well and then sampling it for helium buildup. If a canister gives evidence of a leak, it is removed and returned to the shipping cask, which is again sealed and routed to the area of the facility reserved for overpacking canisters.

(13) Canister Overpacking Process

Casks containing faulty canisters (from cask purging tests and from rejections from the canister testing described above) are transferred to the area where facilities are available for overpacking the canisters. The overpacking area also contains the same facilities as those for routine canisters:

(13a) Canister Extraction: See Item 10.

(13b) Canister Inspection and Possible Decontamination: See Item 11.

(13c) Leak Test: See Item 12.

(13d) Faulty Canister Overpacking: If at any of these points a damaged canister is singled out from all the canisters contained in the cask it will be lowered into an overpack. (An overpack is a cylinder of a size to accommodate the damaged canister.) After the canister is lowered into the overpack it would be seal-welded tight by an automatic welding machine.

The overpack would then be retested and thenceforth handled similarly to a routine canister. After final processing of the canisters the lid is replaced on the cask, which can be returned to its shipper.

(14) Canister Transfer to Mine Hoist Cage

After the canisters have been successfully tested, decontaminated, and overpacked, they are inserted into a transfer cask for loading into the mine hoist cage. This transfer cask is then carried to and lowered through a port into a shaft feed room and tracked into the hoist cage for descent to mine level.

(15) Surge Storage

In the event that an interruption should occur in the operation of the mine hoist, a surge storage is provided to temporarily store the processed canisters.

(16) Lower to Mine Level

The hoist cage lowers the shielded canister transfer casks to mine level.

(17) Subsurface Receiving Station

The hoist cage containing the transfer cask arrives at the subsurface receiving station at the base of the canistered waste shaft inside the hoisting cage that is a permanent part of the shaft support facilities. The subsurface canistered waste receiving station is a multi-level facility equipped to remove the transfer cask from the shielded hoisting cage and position it for loading on the transporter. All operations are by remote control and would be directed by operators stationed behind shielded windows. If desired, canisters containing different types of high level waste could be received at different levels in the subsurface facility.

(18) Load Transfer Cask onto Transporter

The transfer cask would be loaded onto the transporter in a sealed room in the subsurface waste receiving station.

(19) Transport to Storage Area

With the transfer cask safely secured on the vehicle, the transporter operator then proceeds to a hermetically sealed isolation portal, which opens to permit the transporter to enter the subsurface waste placement area of the mine. Once in the mine area, the transporter proceeds at a previously established velocity to the designated emplacement location.

(20) Emplacement of Canister

Upon arrival at the emplacement location, the transporter is positioned by its operator over a pre-determined storage site in the floor of the storage room. The operator then remotely lowers the canister into the storage position using the hoisting devices in the transporter. With the canister secured in its emplacement location, the hoist is disengaged by the operator and retracted.

(21) Backfilling of Canister Storage Cavity

After the canister has been emplaced and the transporter hoisting mechanism retracted, the void space in the floor of the storage cavity is backfilled using material introduced by remote control from a vehicle designed for this purpose. This could also be accomplished using a special attachment on the transporter itself.

4.1.2 Alternate Functional Flow for Strategy 1 - Transfer Cask Arrival

In the event that the spent fuel assemblies from the reactors are processed and packed into canisters at a nearby facility it will not be necessary to re-test the waste packages for possible damage during the transport to the site. In this case the canisters will be packed into a shielded transfer cask and delivered directly to the hoist shaft for lowering into the mine. The Flow Diagram for this case is shown in Figure 4-4b and describes the major steps in the movement of this transfer cask and the canister it contains from receipt to storage. The steps identified on the diagram are detailed below with the number preceding each step keyed to those on Figure 4-4b. The subsurface handling is identical to that previously described.

(1) Spent Fuel Arrival at Site

The facility will be designed to receive the canistered spent fuel in shielded transfer casks in the canistered waste receiving building. There it will be unloaded and brought into the canistered waste building, identified as to type and characteristics from the shipping records and markings, and logged into the accounting system of the repository.

(2) Transfer to Hoist Cage

The cask containing a canister will then be transferred by mechanical devices into the shaft transfer room from which it will be loaded into the hoist cage of the shaft servicing the canistered waste handling.

(3) Lower to Mine Level

The hoist will lower the cage with the transfer cask to subsurface receiving station.

(4) Subsurface Receiving Station

The subsurface receiving station will be designed to allow the safe withdrawal of the transfer cask from the hoist cage.

(5) Load Transfer Cask onto Transporter

From the receiving station the transfer cask will be lifted onto a transporter.

(6) Transport to Storage Area

The transporter will carry the transfer cask to the designated storage room.

(7) Emplace Canister

The canister will be emplaced in a designated emplacement hole in the floor of the storage room.

(8) Backfill Canister Storage Cavity

After emplacement of waste canister the storage hole will be back-filled.

4.2 SURFACE FACILITIES

4.2.1 Site Overview

The site as shown in Figure 4-3, which would be suitable for handling wastes either from Strategy 1 or Strategy 2, encompasses approximately 80 hectares enclosed within a fence and further subdivided according to the degree of security required for the surface facilities. All of the buildings for this high capacity facility, except for the administration building, clinic, and the warehouse buildings, are located within a double-fenced isolation zone. The location of facilities has been developed so that the surface space is of optimum use, assuming concurrent mining, placement, and surface operations. Clearing and grading is performed only as required for roads, railroad trackage, parking, surface drainage, security, and fire control.

The facility is serviced by an improved access road from the nearest public thoroughfare. This access road leads to the administrative and receiving area, and continues past the facility-wide parking field to the guardhouse and checkpoint. The road provides access to all other structures on the site within the isolation zone.

Among the major surface facilities inside the isolation area are:

- o Canistered waste building (for Strategy 1 or Strategy 2)
- o Low-level waste building
- o Ventilation supply building
- o Ventilation exhaust building
- o Men and material building

4.2.2 Canistered Waste Handling and Facilities

The Canistered Waste Building provides the capabilities necessary to receive and handle transfer casks containing wastes for Strategy 1 or casks containing canistered waste resulting from Strategy 2. Space is provided for support functions and activities associated with waste handling. This latter space includes a counting room, lab, health physics office, personnel check station, change rooms, shift foreman's office, maintenance and storage room, and mechanical and electrical rooms. Access

from these areas to the operating galleries and to the cask unloading areas is through a containment door.

A layout for a canistered waste handling facility is given in Figures 4-5 and 4-6. These figures show a typical plan of the Canistered Waste Building sized for a high capacity installation as required for waste arisings from Reactor Strategy 2, 100 GWe installed capacity. For Reactor Strategy 1 and for lower installed capacity per year, this building can be reduced as shown in Table 4-3. The handling sequence for canistered wastes in such a facility is shown in Figure 4-7. All operations shown in this figure are conducted in the Canistered Waste Building.

4.2.3 Low-Level Waste Handling and Facilities

The Low-Level Waste Building provides the capabilities necessary to receive and handling all low level waste shipped to the repository in closed cargo carriers (see Figure 4-8). In addition to space for the handling and processing of pallets and individual drums prior to delivery to the mine level, areas are provided to support functions associated with low level waste handling. These areas include a counting room, a personnel check station, a change room, an operations office, a health physics office, and a battery charge area. The hoist house, which will serve the mine access shaft, is located in a separate structure adjacent to the Low-Level Waste Building.

The basic difference between low level waste operations and operations for canistered waste is that low level wastes do not require remote or shielded handling. A schematic flow diagram of low level waste operations is shown in Figure 4-9. All operations, from the receipt of the cargo carrier to the loading of waste containers into the low-level shaft, are conducted in the Low-Level Waste Building. Transfer cars are used to transport cargo carriers; pallets and drums are handled with forklifts.

The interior atmosphere and exterior surface of each cargo carrier are checked for the presence of radioactive contamination upon its arrival at the Low-Level Waste Building. Although it is unlikely that contamination would be present, facilities are provided to decontaminate or overpack drums that are contaminated or damaged. Cargo carriers and pallets can be decontaminated when required.

Pallets containing 200-liter drums are removed from the cargo carriers by forklift to the transfer room where the drums are removed from the pallets and inspected. After the drums are replaced on the pallets, they are transported to the head of the low-level shafts. After being lowered through the shaft, pallets and their contents are transported to and emplaced in the low-level repository. Transportation at mine level is by forklift truck.

4.2.4 Ventilation Supply Building

The ventilation supply air building provides fresh air for the underground mining and nuclear waste placement operations. Outside air is drawn through air intake louvres into the fans and discharged into the downcast (supply air) shaft. Tornado dampers prevent the outflow of air from the mine back through the downcast shaft. The supply air is heated during the winter and cooled during the summer, if necessary, in order to maintain an acceptable minimum temperature depending upon regulations and site-specific considerations.

4.2.5 Ventilation Exhaust Building

The exhaust vent building houses exhaust fans associated with the ventilation of the mining and drilling (or trenching) operations and the fans and filters required for the continuous filtering of exhaust air from the canister placement operations. Monitors would be located in the mining air compartment of the upcast (exhaust air) shaft to indicate the presence of radioactive contaminants. Under normal conditions there should be no contam-

ination in the mining exhaust and, at most, acceptable levels of contamination in the canister placement exhaust. In the event of an accident involving a canister leak or some other event causing a release of radioactive gases from a canister, only the placement exhaust would become contaminated since the two air flows are separated. Although it is highly unlikely that the mining exhaust air would exhibit the presence of radioactive elements, the capability is provided to filter this portion of the exhaust air. Installation of filters and handling (replacement) of potentially contaminated filters would conform to practices used in the nuclear power industry.

4.2.6 Men and Material Building

This building provides separate personnel facilities such as locker rooms and showers for the mining crews and the placing crews, as well as briefing rooms, check-in, and control facilities. It also provides covered access to the Men and Material Shaft Headhouse for personnel, equipment, and materials. The excavated rock is handled through this headhouse, to a conveyor system which transports the rock to railroad cars or to an on-site storage pile.

4.2.7 Other Surface Buildings

The remainder of the surface facilities are described briefly in Appendix A.

4.2.8 Electric Power and Communications

Arrangements for electric power and on-site communications are described in Appendix B.

4.3 SHAFTS

Five distinct shaft structures are sunk, using conventional techniques. These are the Canistered Waste Shaft, the Low-Level Waste Shaft, the Men and Material Shaft, the Ventilation Intake Shaft, and the Ventilation Exhaust Shaft. The orientation of the shafts as well as their basic characteristics, such as size, method of construction, and design, varies according to their purpose and the influence of constraints upon the mine and shaft network construction schedule.

4.3.1 Canistered Waste Shaft

The canistered waste shaft is designed to provide a means for transporting transfer casks from the Canistered Waste Building at the surface to the subterranean storage levels. From a receiving station at the subsurface terminus, the transfer casks will be transported to the emplacement locations. The loading and unloading of the transfer casks into and from the cage used in the canistered waste shaft is by remote control.

4.3.2 Low-Level Waste Shaft

The low-level waste shaft provides a means for transporting LLW pallets from the low-level waste surface facility to the low-level waste subterranean emplacement area. The LLW pallets are handled by forklift vehicles and transported to the storage area.

4.3.3 Men and Materials (M&M) Shaft

A men and material shaft is provided to handle mine and emplacement personnel and mined material removal. The shaft is designed to accommodate three requirements: 1) Provide access for personnel and equipment into and out of the subterranean storage areas, 2) Provide excavated material hoisting for the mining operation, and 3) Serve as a utility access shaft and manway for emergency exit by all personnel. To meet these requirements the

M&M shaft is divided into three main compartments (personnel and equipment cage, excavated material hoisting skips, and a manway and utility access space) and two auxiliary compartments for skip counterweights. The personnel and equipment compartment for the shaft is designed to be sufficiently large to accommodate a double-deck personnel cage. The compartment also is large enough to permit the lowering and raising of large pieces of equipment either in assembled or partly disassembled form.

4.3.4 Ventilation Intake Shaft

Ventilation air for the subterranean area is received through this shaft. Prior to receipt, the air is processed so that it conforms to air quality and temperature requirements.

4.3.5 Ventilation Exhaust Shaft

Ventilation air in the mine-level area is exhausted through this shaft into filtration systems located on the surface. The shaft is lined and divided so that the exhausted air from the mining and placement operations can be processed separately at the surface in the event that there is evidence of contamination.

4.4 SUBSURFACE OPERATIONS

Four operations occur simultaneously in the subsurface portion of the repository -- mining, drilling of storage holes, waste placement, and backfilling of storage rooms. Each waste storage room undergoes these operations in the sequence listed. The four operations must proceed at the same rate in order to ensure a steady-state condition and to utilize personnel and equipment efficiently.

4.4.1 Mining

4.4.1.1 Mine Layout

Room size and waste storage characteristics for the repository are shown in Table 4-4. Eventual mining layouts will be coordinated with appropriate governing regulations. The room dimensions were designed to be compatible with the mining technique assumed as well as with the proposed waste placement scheme. The pillar sizes were based on the approximate heat dissipation capabilities of the geologic medium. In addition, the extraction ratio must ensure overall mine stability. Mine layouts are shown in Figures 4-10 and 4-11. Room cross sections and extraction ratios are shown in Figure 4-12.

Storage room lengths were determined as a result of two considerations. The length of the rooms must be such that the waste placement operations can be completed for each room in a sufficiently short period so that there is not an excessive length of time between the mining and backfilling operations. This constraint is critical when the waste arisings are relatively small (10 GWe installed capacity). Secondly, the rooms should be kept at a length that provides for efficient mining. This is dependent upon the mining procedure and equipment assumed. The basic criteria for determining the storage room width are the requirements for transporter operations. The minimum width required is 5.5 m. The guidelines used in determining room length are listed in Table 4-4.

In order to achieve efficient mining, four development entries or "corridors" are needed. The four main corridors are utilized as follows: one waste placement corridor, one exhaust corridor,

and two Men & Material (M&M) corridors. The M&M corridors accommodate the mined salt conveyor and serve as a travelway for mining equipment, trucks and personnel. The main conveyor is suspended from the ceiling, thus allowing additional mining equipment to travel through this corridor. Supply air flows through all corridors except the exhaust corridor. This system of corridors ensures that the mining and waste placement operations are kept separate. Crossovers are minimized by judicious sequencing of the operations.

The mine layout for Reactor Strategy 1 is shown in Figure 4-10. In this layout ("long room" system) the storage rooms connect directly to the main corridors at right angles. They are grouped into panels of four rooms; successive panels are separated by a massive buttress pillar approximately 25 meters wide. Because of logistics considerations, only one operation can be performed within a panel at a time. Figure 4-10 shows the location of each of the operations at an arbitrary point in time.

The layout for Reactor Strategy 2 (Figure 4-11) employs this same "long room" system for canisters of MLW and cladding and for drums of low-level waste. The scheme used for HLW provides for branch corridors adjoining the main corridors at right angles. Storage rooms, each 100 to 200 meters long, extend perpendicularly from the branch corridor. A branch corridor and its adjacent rooms constitute one panel. Mining procedure for this "short-room" system differs slightly from the long-room's and results in a most-efficient room length of approximately 200 meters. (See Sec. 4.4.1.2) As in the long-room system, only one operation is performed in a panel at a time.

4.4.1.2 Excavation of Rooms

The mining operations will be performed using electrically powered continuous mining units. Extendable conveyor systems will be used to transport the mined material from the back of the mining machine to the main conveyor. The material is carried by the roof-suspended main conveyor to a storage bin near the base of the shaft

and is eventually conveyed to a loading pocket and hoisted up the shaft to the surface. A single M&M shaft can handle the mined material. When the storage rooms are short (200 meters or less) it is more efficient to use diesel-powered trucks that transport the material from the back of the mining machine to either an extendable conveyor or the main conveyor. Table 4-5 shows the number of tons that must be mined for each reactor strategy and installed capacity.

4.4.2 Drilling

Following the excavation of the rooms, storage holes are drilled into the floor using electrically-powered drills. As the hole depth exceeds or nearly exceeds the room height, the drilling will be done in a multistage operation, using a two-piece drilling bit. The drillings will be handled by front-end loaders or large scoop-trams which carry the material to the main conveyor.

Spent fuel canisters (from Reactor Strategy 1) are stored in 7.6 meter deep holes arranged in two rows spaced 1.8m longitudinally. This corresponds to about 315 kW per hectare local areal thermal loading. Figure 4-17 depicts an emplaced canister.

High-level waste canisters (from Reactor Strategy 2) are arranged in single rows of holes 6 meters deep which are spaced 3.7m on centers. This provides a 370 kW per hectare areal thermal loading. Since thermal loading is not a consideration of Medium Level and Cladding Wastes, storage holes for these wastes are spaced as closely as the integrity of the salt will allow. This distance was estimated to be 1.8 meters center-to-center. Five rows of canisters per room can be accommodated. Table 4-6 lists the allowable thermal loadings assumed for each waste type.

Although the allowable local thermal loading for spent fuel canisters is 370 kW per hectare, this could not be achieved because of the limitations on the drilled hole spacing as stated above. For Strategy 1 there is an additional thermal loading criterion of 148 kW per hectare based on the overall repository area, which includes the shaft area, service areas, and miscellaneous other areas. In order to evaluate whether this requirement was satisfied, it was necessary

to assume a total repository size. An area of 400 hectares was assumed. This corresponds to an operating life of 21 years for the 100 GWe installed capacity, 42 years for the 50 GWe installed capacity, and 211 years for the 10 GWe installed capacity. The required central area was computed in each case and pro-rated on an annual basis as shown in Table 4-7. In order to achieve the 149 kW per hectare overall thermal loading, yet maintain the minimum spacing of the canister (for efficient use of mined area), the size of the buttress pillar was varied (see Figure 4-12).

4.4.3 Subsurface Waste Receiving Station

Subsurface receiving and placement operations originate at the subsurface canistered waste receiving station, a tri-level underground facility capable of safely handling the material under shielded control conditions. The three distinct levels -- lower, intermediate, and transporter level -- serve separate functions in the material flow process. An artists conception of a subsurface canistered waste receiving station is given in Figure 4-13. A description of waste handling in the subsurface canistered waste receiving station and the placement of the waste in terminal storage via the shielded transporter is provided in Section 4.1.

4.4.4 Backfilling

Material for backfill is initially provided by freshly mined salt from storage rooms being excavated concurrently. The salt will be transported by conveyor. Since the waste placement operation will occur in a panel between the panel being backfilled and the panel being mined, a crossover will necessarily result. Separation of the two operations will be maintained by installing a conveyor which traverses the waste placement corridor through the ceiling, a common mining practice. Thus, the waste placement operation will not impinge on the backfilling operation. Mined salt that is not needed for backfilling continues along the main corridor to the M&M shaft.

Approximately 60% of the mined salt will be returned to the rooms as backfill. This figure results from the intention that

the rooms be backfilled up to a height 0.6m below the ceiling and that the unit weight of loose salt is 1.42 metric tons per cubic meter. (Salt in place is 2.16 metric tons per cubic meter.) Table 4-5 summarizes the mining and backfill tonnages.

The excavated material is brought to the surface by skip hoists in the Men and Material Shaft. From the receiving bin of the skip hoists, the material is transported by the belt conveyor system to a railroad loading bin where a train of empty cars will be brought by an over-the-road locomotive in reach of a car positioner. The locomotive picks up previously loaded cars and transports them off-site. The railroad handling of the excavated materials does not interfere with the reception of nuclear waste.

4.5 VENTILATION SYSTEMS

4.5.1 Introduction

The basic purpose of the mine ventilation system is to provide an adequate supply of clean, fresh air for human and equipment needs. Requirements for ventilation include satisfying human air supply needs and controlling the chemical and physical quality of the repository environment by providing for the removal of contaminants such as engine exhaust gases, dust, heat, moisture, and smoke generated by equipment and mining processes. Reversibility of fans would be provided if required by applicable regulations. Possible ventilation airflows for this repository include the following:

- o Minimum of 5.7 cubic meters per minute (m^3/min) of fresh air for each man underground;
- o Minimum of 3.5 m^3/min per brake horsepower (metric) of diesel-powered equipment;
- o Minimum of 15 m^3/min per square meter of headings (or 15 linear meters per minute of air velocity in entries);
- o Minimum of 1100 m^3/min of fresh air per mining machine (based on practice used in existing potash mines).

These rates were used to prepare the preliminary estimate of ventilation requirements for mining and placement operations shown in Table 4-8.

4.5.2 Subsurface Ventilation System

The conceptual arrangement of the subsurface ventilation system is shown in Figures 4-14 and 4-15. Figure 4-14 depicts the ventilation flow for the "long room" system, and Figure 4-15 shows the "short room" system (See Sec. 4.4.1.1). As shown in the figures, supply air entering the mine through the Supply Air Shaft is divided into the main corridors. The air is distributed to the working areas of each of the four operations -- mining, drilling, waste placement and backfilling. A small portion of the supply air is used for ventilating the underground mine equipment maintenance area and for dust control at the storage bin sites (not shown). Proper distribution of the ventilation air is accomplished through the installation of stoppings, doors, and crossings in the main airways and line brattices, along with auxiliary fans with ventilation pipes at the working faces.

The layout of the underground ventilation system provides for the separation of the mining operations exhaust airflow from the potentially contaminated exhaust air of the canister placement operations. This separation is accomplished by using one of the main corridors for return of air from the mining, drilling and backfill operations, while collecting the placement operation exhaust air in a separate system of perimeter ventilation drifts. The two exhaust air flows are brought to the surface in separate compartments of the divided Ventilation Exhaust Shaft. The shaft will terminate in the Ventilation Exhaust Building where the exhaust air from the placement operation is HEPA-filtered before being discharged into the atmosphere. The mining and backfilling exhaust air can follow two paths: simple filtering for normal operation (where the air may obtain salt dust) and HEPA-filtering in the event that the monitors indicate contamination (Figure 4-16).

Exhaust from both the backfill and drilling operations must pass over the supply air in the main corridors in order to enter the main exhaust corridor. This is accomplished by ducting the exhaust over the main corridors as shown in the "crossover" detail of Figures 4-14 and 4-15. Developing such crossovers is a common mining practice and ensures the separation of the two air flows.

4.5.3 Ventilation of Surface Structures

4.5.3.1 Air Distribution Requirements

Three major zones of air distribution have been assumed in surface structures where nuclear waste material is being handled, including

- o Zone I - Primary Confinement Zone; areas of high potential contamination;
- o Zone II - Secondary Confinement Zone; areas of low potential contamination;
- o Zone III - Unrestricted Zone; areas of no potential contamination under normal operating conditions.

Each zone is served by a separate, independent supply and exhaust system. Exhaust systems for Zones I and II will be:

- o Provided with at least two sets of HEPA filters, separately Dioctyl Phthalate (DOP) testable in place;
- o Provided with 100% back-up (standby) systems;
- o Connected to the standby electric power system.

Exhaust systems servicing Zone III areas will have at least one set of HEPA filters, but no back-up system or emergency power supply is required. Table 4-9 lists the Confinement Zone classifications for selected areas in the Low-Level and Canistered Waste Buildings. Supply and exhaust control dampers in the ventilation and air conditioning systems would provide for successively greater negative pressures between areas of successively higher contamination potential.

4.5.3.2 Canistered Waste Facilities

Ventilation air supply for the confinement zones of the Canistered Waste Building would be a combination of outside air and recirculated air. The outside air would be drawn through prefilters and HEPA filters, conditioned and then supplied to the operating galleries and cask unloading areas. Ventilation air for primary confinement zone spaces would be HEPA-filtered, recirculated air from the cask unloading areas (secondary confinement zone). Ventilation air for the support facility rooms and the

mechanical equipment room would be supplied by an independent filter-air-conditioning unit. In order to reduce the heating-cooling load on the outside air supply unit, additional filter-air-conditioner units would be installed in the cask unloading areas.

4.5.3.3 Low-Level Waste Facilities

The non-restricted and the confinement zones of the Low-Level Waste Building would be ventilated by independent supply and exhaust systems. The nonrestricted zones would be air conditioned and ventilated with a minimum of six air changes per hour. Outside air would be filtered through a prefilter (30% NBS*) and a main filter (90% NBS), and then heated or cooled before distribution. The non-restricted zone exhaust system would not have auxiliary backup capabilities, nor would it be linked to the standby power network.

*National Bureau of Standards. Percent refers to filter efficiency for particulates.

4-24

SURFACE FACILITIES FOR HIGH CAPACITY INSTALLATION

WASTE ISOLATION FACILITY FOR HIGH CAPACITY INSTALLATION

- 1 CENTRAL CONTROL BUILDING
- 2 MEN AND MATERIALS BUILDING
- 3 EMERGENCY GENERATOR BUILDING
- 4 DIESEL FUEL TANKS
- 5 STORAGE BUILDING
- 6 FIRE HOUSE
- 7 WATER TANK
- 8 LAUNDRY
- 9 CAFETERIA
- 10 MINE OPERATIONS BUILDING
- 11 ADMINISTRATION BUILDING
- 12 CLINIC
- 13 WAREHOUSE AND MAINTENANCE BUILDING
- 14 SUBSTATION
- 15 RAILROAD AND TRUCK LOADING STATIONS

TYPICAL REPOSITORY SITE PLAN FOR HIGH CAPACITY INSTALLATION

Canistered Waste Functional Flow Chart

Canistered Waste Functional Flow Chart
Strategy 1 (Alternate)

FIG. 4-4b

CANISTERED WASTE RECEIVING FACILITY

CANISTERED WASTE SCHEMATIC FLOW DIAGRAM

GROUND FLOOR PLAN

SECTION

LOW LEVEL WASTE RECEIVING BUILDING

LOW LEVEL WASTE SCHEMATIC FLOW DIAGRAM

MINE SCHEMATIC FOR REACTOR STRATEGY 1
 (SCHEMATIC NOT TO SCALE)

MINE SCHEMATIC FOR REACTOR STRATEGY 2.
(SCHEMATIC NOT TO SCALE)

Salt

STRATEGY 1

PWR/BWR: Extraction Ratio = Varies^a

NOTE: All Dimensions Given in Meters

STRATEGY 2

HLW: Extraction Ratio = 23%

MLW & CW: Extraction Ratio = 46%

EXTRACTION RATIO CHART^c

^aThe extraction ratio for spent fuel disposal varies as follows:

20%	for 100 GWe
19%	for 50 GWe
25%	for 10 GWe

^bThe buttress pillar width for spent fuel disposal varies as follows:

44 meters	for 100Gw
50 meters	for 50Gw
26 meters	for 10Gw

^cExtraction ratios are based on rock mechanics considerations.

SUBSURFACE CANISTERED WASTE RECEIVING STATION

LEGEND

VENTILATION DIAGRAM FOR LONG ROOM SYSTEM

VENTILATION DIAGRAM FOR SHORT ROOM SYSTEM

MINE EXHAUST FLOW DIAGRAM - TYPICAL FOR HIGH CAPACITY INSTALLATIONS

CANISTER EMPLACEMENT CONFIGURATION

*In the case of spent fuel, storage rooms will have two rows.

FIG. 4-17

Table 4-1

WASTE ARISING DATA

REACTOR STRATEGY 1

Physical Dimensions

		<u>Inside Dia.</u> <u>(cm)</u>	<u>Length</u> <u>(cm)</u>	<u>Wall Thick.</u> <u>(mm)</u>	<u>Active</u> <u>Volume</u> <u>(m³)</u>
Canisters:	PWR	36	490	13	0.17
	BWR	41	490	13	0.17
Overpacks:	PWR	41	510	6	
	BWR	46	510	6	
Drums:	LLW		200 liters		
Drum Overpack:			300 liters		

Shipping Containers

PWR & BWR
LLW

Casks
Cargo containers

Canisters/Cask*

PWR	6
BWR	3

Yearly Arisings

		<u>10</u>	<u>50</u>	<u>100</u>
Installed Generation Capacity(GWe)				
Canisters:	PWR	350	1750	3500
	BWR	160	800	1600
Drums:	LLW	680	3400	6800

Thermal Power per Canister

PWR	0.55 kW
BWR	0.55 kW

*Not applicable if spent fuel is canistered at a nearby packaging facility.
For this option, a single spent fuel canister is transferred to the repository in a transfer cask.

Table 4-2

WASTE ARISING DATA

REACTOR STRATEGY 2

SALT

Physical Dimensions

	Inside Dia. (cm)	Length (cm)	Wall Thick. (mm)	Active Volume (m ³)
Canisters: HLW	30	300	10	0.18
MLW	30	300	10	0.18
Cladding	30	300	10	0.18
Overpacks: HLW	35	320	6	
MLW	35	320	6	
Cladding	35	320	6	
Drums:		200 liters		
Drum Overpacks:		300 liters		

Shipping Containers

HLW, MLW, & Cladding	Casks
LLW	Cargo containers

Canisters/Cask

HLW, MLW, & Cladding	9
----------------------	---

Yearly Arisings Receipts

	<u>10</u>	<u>50</u>	<u>100</u>
Installed Generation Capacity (GWe)			
Canisters: HLW	130	650	1300
MLW	1400	7000	14000
Cladding	390	1950	3900
Drums: LLW	1750	8750	17500

Thermal Power per Canister

HLW	3.6 kW
MLW	-
Cladding	-

Table 4-3 .

NUMBER OF OPERATING LINES
IN CANISTERED WASTE RECEIVING BUILDING FOR
VARIOUS POWER GENERATION RATES

SALT

	<u>Strategy 1</u>			<u>Strategy 2</u>		
Installed Capacity in GWe	10	50	100	10	50	100
Yearly Receipt of Canisters	510	2550	5100	1920	9600	19200
Number of Unloading Bays Required	1	1	1	1	2	3
Number of Test Cells*	2	2	2	2	4	6

*Not applicable if all spent fuel is canistered at a nearby packaging facility.

TABLE 4-4
DETAILS OF REPOSITORY DESIGN
SALT

Reactor Strategy	Installed Capacity (GWe)	Waste Type	Room Length ^c (meters)	Room Opening w x h (meters)	Approx. ^a Area Per Room (hectares)	Waste Units/ Room	Rooms Per Panel	No. ^b Rows	Long. Spacing (meters)	Storage Configuration
1	10	PWR	165	5.5 x 6.7	0.317	176	4	2	1.8	Canisters in Holes
		BWR	80	5.5 x 6.7	0.154	80	4	2	1.8	Canisters in Holes
		LLW	90	5.5 x 6.7	0.170	2016	4	---	---	Palletized Drums
	50	PWR	800	5.5 x 6.7	1.536	876	4	2	1.8	Canisters in Holes
		BWR	375	5.5 x 6.7	0.720	400	4	2	1.8	Canisters in Holes
		LLW	450	5.5 x 6.7	0.860	10200	4	---	---	Palletized Drums
	100	PWR	800	5.5 x 6.7	1.536	876	4	2	1.8	Canisters in Holes
		BWR	725	5.5 x 6.7	1.392	800	4	2	1.8	Canisters in Holes
		LLW	280	5.5 x 6.7	0.540	6480	4	---	---	Canisters in Holes
2	10	HLW	100	5.5 x 6.1	0.238	24	11	1	3.7	Canisters in Holes
		MLW/CW	340	11 x 6.1	0.683	896	4	5	1.8	Canisters in Holes
		LLW	90	11 x 6.1	0.180	5250	4	---	---	Palletized Drums
	50	HLW	200	5.5 x 6.1	0.477	49	27	1	3.7	Canisters in Holes
		MLW/CW	810	11 x 6.1	1.628	2240	4	5	1.8	Canisters in Holes
		LLW	370	11 x 6.1	0.740	26250	4	---	---	Palletized Drums
	100	HLW	200	5.5 x 6.1	0.477	49	27	1	3.7	Canisters in Holes
		MLW/CW	810	11 x 6.1	1.628	2240	4	5	1.8	Canisters in Holes
		LLW	480	11 x 6.1	0.960	35000	4	---	---	Palletized Drums

^aArea of the storage room plus one interior pillar.

^bRefers to rows of canisters in each room.

^cThe following guidelines regarding room length were used:

- a. Maximum of 2 years required to fill a panel with waste.
- b. Most efficient mining length is 200m for short room system, 800-1100 for long room system.
- c. Minimum room width is 5.5m for transporter operations.

TABLE 4-5
MINING PRODUCTION SUMMARY
 SALT
 (Annual)

Reactor Strategy	Installed Capacity (GWe)	Mined Tonnage (MT x 10 ³)	Drilled Tonnage (MT x 10 ³)	Total Tonnage (MT x 10 ³)	Tons Backfilled (MT x 10 ³)	Tons Hoisted (MT x 10 ³)	Holes Drilled
1	10	119	2	121	25	96	510
	50	295	12	307	125	182	2550
	100	568	23	591	236	355	5100
2	10	264	7	271	86	185	1920
	50	920	36	956	410	546	9600
	100	1836	71	1907	819	1088	19200

TABLE 4-6
LOCAL AREAL THERMAL LOADING
SALT

Reactor Strategy	Installed Capacity (GWe)	Waste Type	Average kW/can ^a	Areal Thermal Loading ^b (kW/Hectare)
1	10	PWR	0.55	306
		BWR	0.55	286
	50	PWR	0.55	314
		BWR	0.55	306
	100	PWR	0.55	314
		BWR	0.55	316
2	10	HLW	3.6	363
	50	HLW	3.6	370
	100	HLW	3.6	370

^a1 PWR spent fuel assembly per canister.
3 BWR spent fuel assemblies per canister.

^bAreal heat load is calculated by dividing the heat of the total number of canisters per room by the area of the room plus one interior pillar.

TABLE 4-7 SALT REPOSITORY UTILIZATION SUMMARY^a (Annual)

Reactor Strategy	Installed Capacity (GWe)	Area Identification	Waste Containers Received	No. of Panels Utilized	Area Utilized ^b (hectares)	Thermal Power of Waste Received (kW)
1	10	PWR	350	0.5	0.94	190
		BWR	160	0.5	0.57	90
		LLW	680	0.08	0.10	----
		Other ^c	-----	-----	0.28	----
		Total	-----	-----	1.89	280
	50	PWR	1750	0.5	4.79	960
		BWR	800	0.5	2.38	440
		LLW	3400	0.08	0.47	----
		Other ^c	-----	-----	1.81	----
		Total	-----	-----	9.45	1400
	100	PWR	3500	1.0	9.06	1930
		BWR	1600	0.5	4.13	880
		LLW	6800	0.25	0.87	----
		Other ^c	-----	-----	4.89	----
		Total	-----	-----	18.95	2810
2	10	HLW	130	0.5	2.0	470
		MLW/CW	1790	0.5	1.8	----
		LLW	1750	0.08	0.1	----
		Other ^c	-----	-----	0.7	----
		Total	-----	-----	4.6	470
	50	HLW	650	0.5	8.0	2340
		MLW/CW	8950	1.0	8.2	----
		LLW	8750	0.08	0.3	----
		Other ^c	-----	-----	6.5	----
		Total	-----	-----	23.0	2340
	100	HLW	1300	1.0	16.0	4680
		MLW/CW	17900	2.0	16.3	----
		LLW	17500	0.12	0.6	----
		Other ^c	-----	-----	12.9	----
		Total	-----	-----	45.8	4680

^aBased on 400 hectare repository.^bProjected surface area.^cIncludes site.

TABLE 4-8

ESTIMATED MINE VENTILATION REQUIREMENTS OF
MINING AND PLACEMENT OPERATIONS

SALT

Reactor Strategy	Installed Capacity (GWe)	Mining Operation Exhaust* (m ³ /min.)	Placement Operation Exhaust (m ³ /min.)	Total Exhaust (m ³ /min.)
1	10	7,400	3,400	10,800
	50	13,300	3,400	16,700
	100	15,600	3,400	19,000
2	10	18,700	3,400	22,100
	50	26,600	4,500	31,100
	100	35,400	5,700	41,000

*Includes drilling and backfilling operation.

CONFINEMENT ZONE CLASSIFICATIONS
AND VENTILATION DESIGN REQUIREMENTS

Canistered Waste Building Criteria

<u>Critical Areas</u>	<u>Confinement Zone</u>	<u>Air Change per Hour</u>
Canister Storage and Feed Room	I	2
Transfer Galleries	I	6
Transfer Cells	I	6
Operating Galleries	II	6
Hoist Machine Room	II	10
Cask Unloading	II	3
Exhaust Air Filter Room	II	2
Mechanical Equipment Room	II	6

Low-Level Waste Building Criteria

<u>Critical Areas</u>	<u>Confinement Zone</u>	<u>Air Change per Hour</u>
Unloading and Parking	II	10
Palletizing and Depalletizing	II	10
Overpack Drum Storage	II	10
Test Cells	II	10
Decontamination and Overpack	II	10
Exhaust Fan Filter Room	II	2
Hoist House	II	10

Note: Values are typical for all Installed Capacities in each Reactor Strategy studied.

5.0 THERMAL RELEASES

5.1 IMPACTS OF THERMAL RELEASES

A major factor associated with geologic storage of radioactive waste will be the heat generated by the high-level waste or spent fuel assemblies. This heat will flow from the waste, through the storage canister and other protective material into the host rock formation, through the rock units surrounding or overlying this formation and, eventually, reach the atmosphere. The heat will have definite impacts on:

- o Integrity of the waste matrix over long periods of time.
- o The integrity of the host rock and the surrounding rock units.
- o The fracture permeability of the host rock and the surrounding rock units.
- o Temperature rises in any overlying aquifers and near the earth's surface.
- o Long-term surficial uplift and subsidence.

To assure that the impact of the heat on the environment will not be detrimental, a systematic determination of the repository design thermal loads should be completed in the following sequence:

1. Establishment of limits for conditions affected by heat.
2. Determination of acceptable thermal loads that will not change the conditions beyond the assigned limits.
3. Development of repository design thermal loads, taking into account engineering and operational requirements.

5.2 THERMAL AND THERMOMECHANICAL LIMITS

The thermal and thermomechanical limits used in these preconceptual design studies are summarized in Table 5-1. It must be emphasized that the limits shown in Table 5-1 are based upon the best available data at

this time. As such, they should be reevaluated as more data become available. These limits definitely require reevaluation on a site-by-site basis prior to detailed site design. The maximum temperature of the high-level waste (500°C) was set to control potential cracking and heavy nuclide migration in a softened borosilicate glass.³ The remaining thermal and thermomechanical limits were chosen because they appear to represent acceptable impacts on the strata overlying the repository. Detailed descriptions of the rationale for these limits are presented in References 1 and 2.

In addition to the limits presented in Table 5-1, other limits might be considered. A limit of 200°C has been proposed for spent fuel temperatures during storage in order to control potential chemical reactions involving the Zircaloy hulls.³ Further, if canister integrity is desirable for longer periods of time (for example, in the case of retrievable storage), then it has been proposed that a maximum desirable temperature for austenitic stainless steel is 375°C.³

5.3 ACCEPTABLE THERMAL LOADS

Acceptable thermal loads have been proposed for a repository located in salt.¹ Although this earlier work assumed a different repository description and different stratigraphy than those presented in this report, supplementary thermal analyses for salt indicate that the results may be applied to define acceptable thermal loads for the salt repository described in Section 4 and the stratigraphy presented in Section 2. The utilization of these previously defined design criteria is probably conservative.

The local areal thermal load (kW/hectare) must be sufficiently low that rock mechanics considerations predict that the temperatures and thermomechanical restrictions of Table 5-1 are not exceeded. Based on the HLW waste temperature restrictions, the maximum allowable canister loading is 3.6 kW. These restrictions indicate that a local loading of 370 kW/hectare is acceptable for the waste from either strategy. At the same time the loading for the spent fuel reactor strategy must not exceed

150 kW/hectare when averaged over the entire repository area. The difference between the design criteria for the average repository loading for each reactor strategy is due to the difference between the decay characteristics for the wastes in each case (see Figure 3-2). The spent fuel heat generation rate falls off more slowly than the HLW heat generation rate; consequently, the long-term energy deposition in the formation will be much higher for the spent fuel strategy than for the reprocessing strategy for the same average loading. Since the far-field thermomechanical stresses depend strongly on the total integrated energy deposition, the local areal thermal loading for reactor Strategy 1 must be significantly less than for Strategy 2 in order to provide acceptable impacts in either case.

5.4 THERMAL LOADINGS IN REFERENCE REPOSITORY DESIGNS

At a specific site, engineering and operational considerations may restrict the thermal loadings to values lower than the limits discussed in the above section. These considerations include such factors as reasonable HLW thermal densities, available canister sizes, permissible hole spacing, or operational limitations on hole arrangements. These considerations were taken into account in the repository designs discussed in Section 4 for the stratigraphy presented in Section 2. In this case, they did not limit the thermal loadings which are presented in Table 5-2.

The local thermal loadings in the spent fuel repository are computed from the canister spacings in the appropriate repository rooms (Table 4-4) and the canister thermal loadings. Since the spent fuel assemblies are 10 years old at emplacement, the individual canister load is 0.55 kW.

The HLW canister thermal loadings are computed from the canister dimensions in Table 3-2, along with a specific volume for the HLW of 0.090 m³/MTHM. The local thermal loading for the HLW is computed from the canister thermal loadings and the canister spacings in the appropriate repository rooms (Table 4-4). The HLW is assumed to be 10 years old at emplacement in the repository.

5.0 REFERENCES

1. Russell, J.E., "Areal Thermal Loading Recommendations for Nuclear Wastes Repository in Salt," Y/OWI/TM-37, Oak Ridge, TN. (unpublished)
2. Science Applications, Inc. "The Selection and Evaluation of Thermal Criteria for a Geologic Waste Isolation Facility in Salt," Y/OWI/SUB-76 107220, Prepared for the Office of Waste Isolation, Union Carbide Corporation, Oak Ridge, TN, 1976.
3. Jenks, G.H., "Maximum Acceptable Temperatures of Waste and Containers during Retrievable Geologic Storage," Y/OWI/TM-42, Oak Ridge, TN, August 1977.

TABLE 5-1

THERMAL AND THERMOMECHANICAL LIMITS FOR PRECONCEPTUAL
DESIGN STUDIESFar-Field Considerations

--Maximum HLW Temperature	500°C ^a
--Maximum Uplift Over Repository	1 to 1.5 m ^b
--Temperature Rise at Surface	<0.5°C
--Temperature Rise in Aquifers	5°C, ALARA ^c

^aSee Reference 3

^bSee Reference 1

^cSee Reference 2, ALARA, "As low as reasonably achievable."

TABLE 5-2

THERMAL LOADINGS IN SALT REPOSITORY (kW/Hectare)

Installed Capacity (GWe/yr)	Reactor Strategy 1			Reactor Strategy 2
	Overall Repository Loading ^a	Local Loading for PWR ^b	Local Loading for BWR ^b	Local Loading for HLW ^c
10	148	306	286	363
50	148	314	306	370
100	148	314	316	370

^aDesign criterion = 148 kW/hectare.

^bBased on 0.55 kW/canister. Design criterion = 370 kW/hectare. Considers projected surface area of a storage room plus one interior pillar.

^cBased on 3.6 kW/canister. Design criterion = 370 kW/hectare.

APPENDIX A
OTHER SURFACE FACILITIES

I. Introduction

The major ancillary surface facilities include:

(Inside the isolation area)

- o Radwaste facilities
- o Central Site Control Building
- o Mine operations building
- o Refrigeration and boiler plant
- o Railroad yards
- o Emergency generator building
- o Diesel Fuel Tanks
- o Storage buildings
- o Firehouse
- o Water Tank
- o Laundry
- o Cafeteria
- o Guardhouse

(Outside the isolation area)

- o Administration Building
- o Clinic
- o Warehouse and Maintenance Building

II. Radwaste Facilities

The purpose of the liquid radwaste building is to solidify the liquid radwaste collected from various on-site sources. Ancillary facilities include palletizing area, truck dock for transporting of empty and full drums of chemicals, cooling towers, and personnel facilities.

The solid waste disposal facilities would be housed in an annex to the Ventilation Exhaust Building. The purpose of these

facilities is to incinerate or compact various solid wastes generated on-site. The compactor section of these facilities would compress non combustible materials such as HEPA filters and residue from the incinerator.

III. Central Site Control Building

This building houses central site communications, monitors for security instruments, remote radiological monitoring control equipment, and ventilation system remote monitoring and control equipment.

IV. Mine Area Operation and Services

Adjacent to the Men and Material Shaft, a service building supplies personnel facilities, such as change room, showers and toilets, check-in area, personnel control, and briefing rooms. Separate facilities are provided for mining and waste placement crews in this building. A separate mine operations building provides facilities for the mining contractor.

V. Boiler and Refrigeration Building

The steam-generating equipment and the refrigeration equipment are anticipated to be housed in separate sections of the Boiler and Refrigeration Building. The boiler plant is assumed to be complete with coal-fired stoker furnaces, boilers, coal storage handling and weighing system, condensate handling, water treatment and feedwater systems, and effluents treatment systems.

VI. Railroads

Railroad tracks are arranged so that nuclear waste material handling, excavated material disposal, and coal handling would be separated in their respective areas to minimize security checking of equipment and personnel. The design of the trackwork serves the following purposes:

- o Receipt of nuclear waste shipments, disposition to the Canistered Waste and Low Level Waste buildings, and return of empty casks and shipping containers.
- o Receipt of shipments for warehouse and mine operations.
- o Disposal of excavated material.
- o Receipt of coal shipments.

VII. Emergency Generator Building

This building would house the diesel turbine-driven emergency generators.

VIII. Diesel Fuel Tanks

Diesel fuel for the diesel-driven emergency generators and the mining and placing operations would be stored in three below-ground 60,000 liter capacity storage tanks. A pump system would fill the day tank for the diesel turbines of the emergency generators. Fuel oil for the mining and placing operation would be lowered to the mine level either in drums or tanks on the men and material hoists.

IX. Storage Building

This building would provide intermediate storage between the warehouse located outside the isolation fencing and short-term requirements of the plant operations. It is accessible by truck and railroad for shipments from outside.

X. Cafeteria

The cafeteria would be located within the isolation area to obviate security checks of the personnel during lunch breaks. The cafeteria would be serviced by a local caterer and would operate for each shift.

XI. Laundry

The laundry building would be equipped to clean work clothes of the mining personnel and personnel other than those engaged

in areas where radiological contamination may occur, such as the Canistered Waste Building, the Low Level Waste Building, and the plating operations. The clothing for these personnel would be of the disposable type and would be incinerated.

XII. Fire House

The fire house would be equipped with pumpers and other fire-fighting equipment.

XIII. Water Tank

A water tank would provide storage of approximately 600,000 liter capacity for standby fire-fighting emergencies.

XIV. Guard House

The main guard house is located at the entrance for personnel and trucks. It would provide all facilities for checking, controlling, and monitoring all personnel, cars, and trucks entering the isolation area. It would also contain an alarm and communications system paralleling the security system in the central site control building.

A second guardhouse at the entrance of the railroad tracks would serve to monitor and control all railroad traffic into the isolation zone.

XV. Administration Building

The administration building, located outside the isolation zone, would house all administrative personnel for the repository not required for the actual operation inside the isolation zone, such as accounting, auditing, computer, contracts, insurance, engineering, mailroom, payrolls, and purchasing.

XVI. Warehouse and Maintenance Building

Materials would be received at the warehouse and maintenance building, also located outside the isolation area, thereby allowing deliveries by truck or rail to enter and leave without elaborate security checks. Material required inside the isolation zone would

be removed from its packaging as far as possible, checked, and forwarded to the storage building inside the isolation area when required. All waste, cartons, or other packaging material would thus remain outside the isolation area and could be disposed of through ordinary means. The maintenance building would be equipped with tools and machines for minor repairs and maintenance. Major repairs are expected to be performed by outside contractors.

XVII. Clinic

The clinic would be located outside the isolation area to permit outside ambulance service to operate this facility without delay caused by security checking. First aid and emergency treatment would be performed in the clinic.

APPENDIX B

ELECTRIC POWER AND COMMUNICATIONS

I. Primary Electric Power

The primary source of electric power for the facility is assumed to be from the overhead medium voltage lines of the local electric utility company. It would supply interior and exterior lighting, ventilation equipment for mining and nuclear waste placement operations, surface waste handling and support facilities, motorized equipment for surface building HVAC requirements, radiation monitoring instrumentation, telephone and radio communications systems, security and surveillance systems and alarms including closed circuit television, hoists, mining operations and conveyors, repository waste placement operations, and maintenance and repair of equipment.

II. On-Site Power Generation

In the event of the failure of primary electric service, standby on-site power generating capacity would be provided to supply the requirements of essential loads. This includes power for limited mine ventilation while evacuating mine personnel, for radiation monitoring instruments, for communications and security systems, for maintaining essential underground and surface facility processes, for fire alarm and protection systems, and for the personnel hoist system serving the men working below ground.

III. Uninterruptable Power Supply

During the interim between the failure of the primary power source and the availability of the secondary standby onsite power source, two requirements for uninterrupted power must be met. The first is for sufficient lighting to aid personnel to exit from structures and for operating personnel to start standby power equipment if manual starting is required. The second requirement is

for sufficient power to maintain the continuity of operation of radiation monitoring equipment, communications, security operations, central control station, guardhouse operations, and key instrument and control systems. Essential and critical equipments that require power continuity will be supported by localized uninterruptible power supplies. Complete lighting blackout can further be precluded by the installation of self-contained battery powered emergency units.

IV. Communications

Normal communications between or within the various areas of the repository, both surface and mine-level facilities, are assumed to use commercial telephone lines. In addition, each guard or watchman on duty would have the capability of maintaining continuous communications with an individual in a continuously manned central alarm station located at the guardhouse adjacent to the personnel portal. A secondary guard communication link would be strategically located telephones with exclusive guard access and connected to the central guard station by dedicated lines. Portable radio or closed circuit television (CCTV) communication may also be desirable. All communications equipment would remain operable after loss of primary power by automatically switching to an uninterruptible supply and then to the onsite standby generator system.

V. Underground Distribution

All wiring between origin and destination of different buildings would be designed to be installed in underground conduits and manholes. Three separate conduit systems would be used. One system would contain wiring for telephone communications, security and alarm wiring, fire alarm wiring and CCTV wiring and cables. The second system would contain power and control wiring originating at the emergency electric power generating plant. The third system would contain power, lighting and control wiring of the normal primary power source supply and distribution.

APPENDIX C

OVERVIEW OF ASSUMED EMPLACEMENT STRATEGY

All canistered high-level waste, middle-level (MLW), and cladding (CW) waste as well as spent fuel assemblies from both the PWR and BWR fuel cycles, are assumed to arrive at the repository in sealed steel canisters. All canisters, regardless of reactor strategy, will be handled remotely and will require shielding. For purposes of this study, the HLW, MLW and cladding canisters of reactor strategy 2 are assumed to be identical. Although the specific dimensions of the spent fuel canisters vary somewhat from the reprocessing waste canisters, a physical similarity does exist (Table 3-2) and was used for purposes of simplicity and ease of assessing impacts of variations in assumed receipt rates and waste types upon repository designs. Because of this similarity among the canisters for all high-level, middle level, cladding, and spent fuel wastes, a single reference handling facility and burial procedure has been developed for all canistered wastes. The surface handling facilities are scaled as appropriate for the assumed installed capacity (i.e., number of receiving bays, hot cells, transfer galleries, and waste shafts).

Canisters are transferred to to subsurface storage area in shielded transfer casks. These transfer casks are loaded on transporters to be taken to the storage rooms.

The burial procedure for canisters is independent of type of waste or installed capacity. All canistered wastes are buried in holes in the storage room floors with about 2.5 meters of backfill material above it. This burial assumption is based on the need for shielding of canisters from operating personnel.

An alternative to emplacing MLW and cladding waste in the floor could be to encase the canister in concrete sleeves and stack the waste on the floor of previously filled high-level waste storage rooms.

Low-level drums from both reactor strategies are assumed to be stored on pallets, stacked in the open on the floor of the storage room. Since burial of these drums is not required, an alternative storage arrangement could be the placement of all LLW in previously filled canistered waste storage rooms.