WASTE PACKAGE CONCEPTUAL DESIGNS FOR SALT

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ABSTRACT

The conceptual design phase to develop waste packages for spent fuel, commercial high-level waste, and defense high-level waste for emplacement in a salt repository has been completed.(1) This paper describes the reference package concepts for a salt repository, package design requirements, and package performance.

INTRODUCTION

The conceptual design phase to develop waste packages for spent fuel (SF), commercial high-level waste (CHLW)1 and defense high-level waste (DHLW) for emplacement in a salt geologic repository has been completed.(1)

A conceptual design is that phase of the design process in which all the identified criteria, specifications, interfaces, and data are used as a foundation for formulating concepts which will meet performance and design objectives. These concepts are developed into designs. The designs are carried to the point where engineering judgment can determine which are feasible and practical, and can be expected to meet the design criteria and requirements. This process includes the performance of scoping analyses (stress, thermal, shielding, etc.); selection of materials and estimates of materials performance; production of general sketches containing essential dimensions; consideration of fabrication feasibility, cost, and materials availability; and incorporation of design features to satisfy interface requirements with other elements of the repository and waste disposal system.

WASTE PACKAGE PERFORMANCE REQUIREMENTS

A nuclear waste isolation system based on the concept of a geologic repository is intended to protect the public from the hazards of the radioactive waste by interposing a system of barriers, both natural and man-made, between the waste and the accessible environment. The natural barriers of interest include the geological, hydrological, and geochemical characteristics of the site and environs, while the engineered barriers are the mined repository and the waste package. Thus, the primary performance objective of the waste package is to work with other components of the engineered repository system to limit the release of radionuclides to the accessible environment.

Draft regulations by the NRC under which the geologic repository system would be licensed assign values for this release to various times in the isolation system lifetime:

- The preclosure period during which the repository is open and waste is being received, packaged, emplaced, and retrieved if deemed necessary for safety reasons prior to repository closure and decommissioning.
- The containment period (currently defined as up to 1,000 years after permanent closure of the repository) during which waste must be essentially contained within the physical boundaries of the package.
- The isolation period extending out to 10,000 years after emplacement during which radionuclide release from the engineered system (either the package alone or in conjunction with the repository) must by maintained at a low value, currently specified at one part in 10^5 per year of the radionuclide content existing at 1,000 years after emplacement.

Existing and proposed federal regulations which are expected to apply to the waste package and which have directed the design are:

10 CFR 60	Technical Criteria for Regulating Geologic High-Level Radioactive Waste - Federal Register/Vol. 45, No. 94/Tuesday, May 13, 1980 Advance Notice of Rulemaking and Federal Register/Vol. 46, No. 130, Wednesday, July 8, 1981 Proposed Rules.
10 CR 191	Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste - Working Draft/December 10, 1979.
10 CFR 20	Standards for Protection Against Radiation
10 CFR 50	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants - Appendix B
10 CFR 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
49 CFR 173	Shippers - General Requirements for Shipments and Packagings
30 CFR 57	Health and Safety Standards - Metal and Nonmetallic Underground Mines.

Other DOE National Waste Terminal Storage Program documents utilized in the package design were the DOE Statement of Position on Waste Confidence Rulemaking (2) and NWTS 33-1(3).

From these regulations and documents, a set of design requirements was developed to guide the design, as summarized here:

Containment. Provide reasonable assurance that radionuclides will be contained for at least 1,000 years after repository decommissioning under expected repository conditions.

Isolation. Provide reasonable assurance that after the containment period, the annual radionuclide releases to the geologic setting do not exceed a predetermined level currently defined as one part in 10^5 per year of the maximum postcontainment inventory of any nuclide which is in excess of 0.1 percent of the total annual Curie release.

Retrievability. That portion of the engineered waste package containing the waste form shall be retrievable intact for a period after initial emplacement (for design purposes, a time requirement of 150 years).

Radiation Safety. The engineered waste package, when emplaced and in conjunction with the geologic repository design prior to decommissioning, shall comply with 10 CFR 20 exposure to workers as it applies to nuclear fuel facilities.

Criticality Safety. Assure that the K_{eff} is less than or equal to 0.95 and no nuclear criticality accident can occur unless at least two unlikely independent and concurrent or sequential changes have occurred.

Fire Safety. The engineered waste package as a whole or package components shall not be capable of initiating or sustaining combustion nor shall it be capable of creating explosions under reasonably expected conditions.

Handling Safety. The containerized waste and the retrievable portions shall be capable of surviving a free vertical fall for a distance at least 2.0 times the package length without release of radionuclides or loss of retrieval capability. In addition, the containerized waste and the retrievable waste package shall be suitable for lifting, transferring, and orienting both on and off the geologic repository site.

Transportation. The containerized waste and the retrievable portions of the engineered waste package shall be capable of transportation by conventional rail and truck systems in conjunction with suitable shipping containers.

Accountability. The canistered waste and retrievable package shall have a unique marking for identification to assure traceability of the package contents from first fabrication until repository decommissioning.

Waste Package Environment

The waste package environment which will affect package performance and hence influence package design varies with the geologic medium considered for the site of the repository. Since exact salt sites and in situ geologic/geochemical data were not available and will not be until specific sites are identified and fully characterized, generic geological and geochemical parameters representative of that geology were developed. It is expected that by the time waste package preliminary design activities are initiated, more site-specific data will be available. Tables I and II summarize the geotechnical parameters utilized for the salt package conceptual designs.

Table I. Summary of Geotechnical Parameters for Salt.

Depth of Repository	600 m
Design Basis Earthquake	0.3 g
Lithostatic Pressure at Repository Level	16.2 MPa
Hydrostatic Pressure at Repository Level	6.0 MPa
Maximum Design Pressure	16.2 MPa
Salt Decrepitation Temperature	250 C
Density of Salt	2.2 g/cm^3
Porosity of Salt (Volumetric)	0.5%-1.7%
Moisture Content of Salt (Volumetric)	0.5%
Permeability of Salt	0.1 md
Brine Constituents	See Table IV
Thermal Conductivity of Salt 25 C	6.0 W/mK
Thermal Expansion of Salt	4.10^{-5} K^{-1}
Specific Heat Capacity of Salt	1.0 kJ/kgK
Creep of Salt at 25 C	0.65 cm/year

Table II. Composition of Salt Brines. (mg/liter, ±3%)

Ion	Inclusion Brine	Dissolution Brine
Na^+	42,000	115,000
K^+	30,000	15
Mg^{2+}	35,000	10
Ca^{2+}	600	900
Fe^{3+}	2	2
Sr^{2+}	5	15
Li^+	20	–
Rb^+	20	1
Cs^+	1	1
Cl^-	190,000	175,000
SO_4^{2-}	3,500	3,500
$B(BO)_3^{3+}$	1,200	10
HCO_3^-	700	10
Br^-	400	400
I^-	10	10
pH at 20 C	6.5	6.5

PACKAGE CONCEPT DESCRIPTION

The waste package designs for SF, CHLW, and DHLW consist of the waste forms and production canister and an overpack. The overpack is a thick-walled cylindrical container fabricated from carbon steel components. Its function is to provide containment of the waste during the operations period (transfer, emplacement, and possible retrieval) and up to 1,000 years after emplacement.

Tables III and IV show reference waste form characteristics for spent fuel and commercial high-level waste.

WASTE FORMS

A major interface for the waste package design is the nuclear waste itself; the waste package design must accommodate the waste form. The specification of the waste form is an iterative process requiring interfacing between the salt repository program (waste package design) and the waste form producer. Although the types, quantities, and toxicity of the waste radionuclides are fixed by the source of generation, waste form compositions, geometries, and properties are subject to interface agreements where necessary between the salt repository program and the waste form producer. It accomplishes this function by being of sufficient thickness to resist crushing from external pressure loads from lithostatic pressure in addition to any expected corrosion processes occurring during the containment period.

Figure 1 presents package dimensions for SF, CHLW, and DHLW. Figure 2 gives the nominal repository dimensions and emplacement geometries for the three waste types.

The waste forms (spent fuel, CHLW, DHLW) will arrive at the repository and are expected to be in conformance with applicable waste form product specifications. The overpack consists of three components: the bottom head, body, and top head with pintle. The body and bottom head will arrive at the repository already joined and inspected. The insertion of the waste form into the overpack, placing of the overpack lid, the final seal weld, and its inspection will all be done in a remote hot cell. The external surface of the package will be checked for surface contamination and will, if necessary, go through a decontamination process.

The lifting pintle located at the top of the package provides a means of handling the package. When it is used in conjunction with a positive fail-safe grapple, requirements for safe transfer of the waste package can be accomplished. The overpack is of sufficient thickness to retain its containment features for normal surface and subsurface handling incidents. Until definitive design basis accidents are established, the package is being designed to withstand a drop of at least 2.0 times its length without loss of containment of the overpack component.

Table III. Characteristics of Spent Fuel Rods.

	PWR		BWR	
	Max.	Min.	Max.	Min.
Diameter (mm)	11.18	9.5	14.48	12.0
Length (m)	4.1	3.48	4.1	3.95
Weight of U per Rod (Kg)	2.15	1.63	2.97	2.75
	Nominal		Nominal	
Thermal Conductivity (W/mK)	5.5		5.5	
Specific Heat (J/kgK)	264		264	
Density (g/cm^3)	2.0		2.4	
Heat Generation per Rod[a] (W)	2.08		3.02	
Emplacement Limiting Temperature (C)	375		375	
Limiting Temperature to Minimize Leaching (C)	100		100	
Leach Rate (g/cm^2-day)	10^{-5}		10^{-5}	
Average Fuel Burnup (MWd/MT)	33,000		27,500	
Radioactivity Content[a] (Ci)	1,515		952	

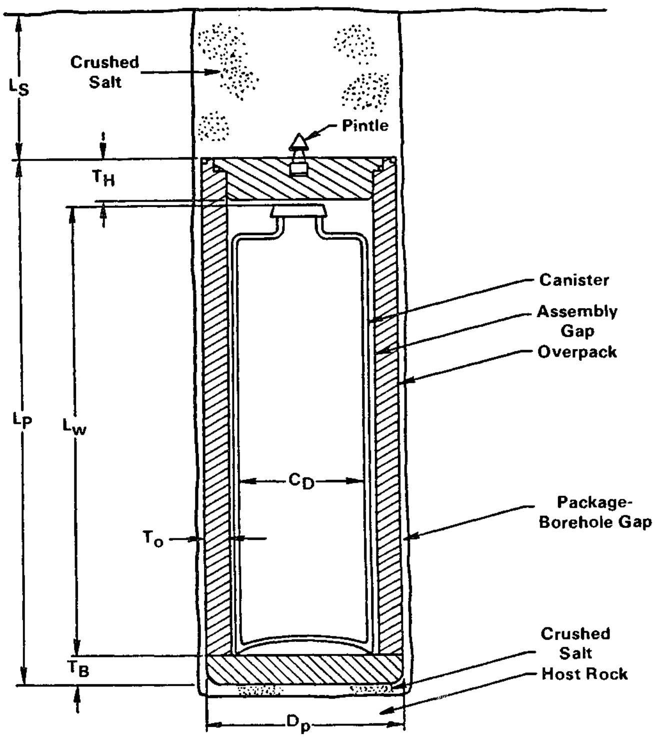
(a) 10-year-old fuel.

Table IV. CHLW Characteristics.

Canister Dimensions	0.324 m diam x 3.0 m long 0.560 m diam x 4.09 m long – salt alternate
Glass[a] Thermal Conductivity	0.8-1.3 W/mk (0 to 500 C)
Glass Specific Heat	> 700-800 J/kgK
Glass Density	3.1 g/cm^3
Canister Maximum Heat Output at Emplacement	2.2 kW 9.5 kW – salt alternate
Limiting Glass Temperature During Package Design Life[b]	500 C
Limiting Glass Temperature After Package Design Life	100 C
Glass Total Weight	595 kg 2583 kg – salt alternate
Canister Radioactivity Content	6.58×10^5 Ci – 10 years out of reactor 28.57×10^5 Ci – salt alternate
Canister Active Glass Volume	0.19 m^3 0.85 m^3 – salt alternate
Leach Rate	2.0×10^{-6} g/cm^2/day

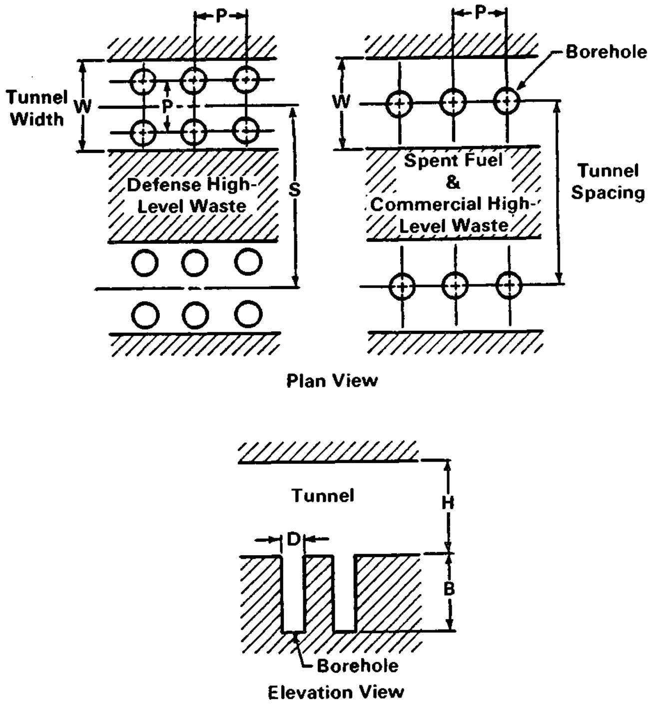
(a) Glass refers to HLW in glass matrix. Assumes 10 years out of reactor.
(b) This limit represents the glass softening point beyond which glass devitrification is possible.

WASTE PACKAGE DIMENSIONS FOR SPENT FUEL, CHLW, AND DHLW



All Dimensions in cm	L_P	L_S	L_W	T_H	T_O	T_B	D_P	C_D
Spent Fuel (PWR)	450	140	410	20	12	18	84	57
Commercial High-Level Waste	457	173	409	21	15	21	89	56
Defense High-Level Waste	338	112	300	18	10.4	18	84	61

Fig. 1. Waste Package Dimensions for Spent Fuel, CHLW, and DHLW



	All Dimensions in Meters					
	B	D	H	P	S	W
Spent Fuel	5.9	0.93	7.4	10.4	27.5	4.0
Commercial High-Level Waste	6.3	0.94	7.2	10.0	31.6	4.0
Defense High-Level Waste	4.5	0.89	6.1	2.4	25.8	5.0

Fig. 2. Reference Repository Emplacement Dimensions

If retrieval of the waste package is required, the package has design features to enable either of two retrieval methods to be employed: (1) removal of the entire package either directly or by overcoring or (2) cutting of the overpack lid and removal of the waste form.

Following emplacement of the waste package into the repository, the overpack is expected to act as a containment barrier to the radionuclides for several hundred to 1,000 years. Major processes which act to destroy this containment are the closure force from salt creep acting to restore lithostatic pressure and corrosion resulting from the presence of moisture in the salt. Both processes are affected by the thermal performance of the emplaced package.

Figure 3 shows the predicted thermal performance of the package during the containment period. Table V gives performance parameters for spent fuel, CHLW, and DHLW packages including expected quantity of migrated brine, quantity of metal reacted, equivalent thickness when uniform corrosion takes place, total overpack thickness, and thickness required for crush resistance. Also shown is the thickness of metal reacted under the postulate of the brine acting on only a small area of the metal (this is equivalent to an unlimited supply of brine for corrosion reaction).

Following loss of containment, radionuclides are free to be released from the waste package by inter-

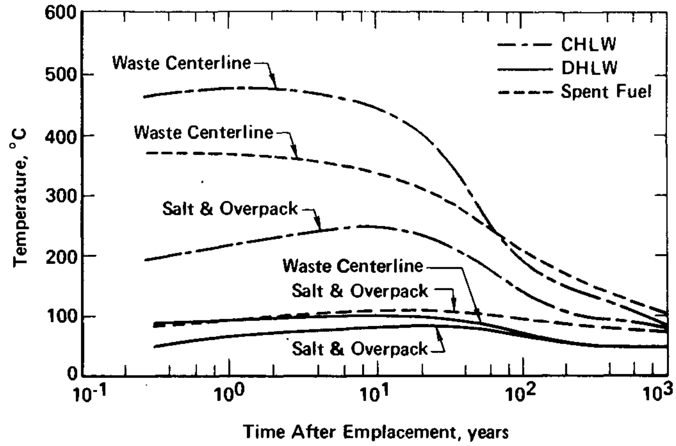


Fig. 3. Thermal Performance of Waste Packages During Containment Period

Table V. Waste Package Performance Parameters.

	Spent Fuel	CHLW	DHLW
Brine Quantity Per Package, liters	80.0	80.0	13.0
Metal Quantity Reacted, kg	104.0	104.0	17.0
Equivalent Uniform Thickness of Corrosion, cm	0.1	0.1	0.03
Total Overpack Thickness, cm	12.0	15.0	10.4
Crush Resistance Thickness Required, cm	9.5	10.0	9.5
1,000-Yr Corrosion for Limited Area Reaction (Unlimited Brine Assumption), cm	2.5	5.0	0.9

action with brine. No function is being assigned to the failed overpack to partially restrict the inflow of brine or the egress of radionuclides. Release rates will be covered by the performance of the waste form alone taking into account alteration (if adverse) of the brine by the degradation products of the overpack and canisters. Release rates will be governed by either solubility or leaching considerations, depending on the quantity and rate of brine coming in contact with the waste form.

REFERENCES

1 ONWI-438, Engineered Waste Package Conceptual Design Defense High Level Waste, Commercial High Level Waste and Spent Fuel Disposal in Salt, Westinghouse Electric Corporation, February 1983.

2 DOE/NE-0007, Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (Waste Confidence Rulemaking), Washington, D.C., 15 April 1980.

3 DOE/NWTS-33(1), NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Program Objectives, Functional Requirements, and System Performance Criteria, Battelle Project Management Division, Columbus, OH, March 1982.