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CONCEPTUAL DESIGNS FOR WASTE PACKAGES
FOR HORIZONTAL OR VERTICAL EMPLACEMENT
IN A REPOSITORY IN SALT

FOR REFERENCE IN
THE SITE CHARACTERIZATION PLAN

TECHNICAL REPORT

June, 1987

This document is
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TECHNICAL STATUS

This technical report is being archived in advance of DOE technical review because of Congressional shutdown of the Salt Repository Project. Because of project termination, DOE has expedited the report clearance process to ensure availability of project data. A summary of the review status of this report is attached.

ABSTRACT

This report is a revision and supplement to Waste Package Reference Conceptual Designs for a Repository in Salt, BMI/ONWI-517, by Westinghouse Electric Corporation, 1986. It includes the options of horizontal and vertical emplacement, the addition of a phased repository, an additional waste form (intact spent fuel), revised geotechnical data appropriate for the Deaf Smith County site, new corrosion data for the container, and new repository design data.

The waste package consists of a waste form and canister within a thick-walled, low-carbon steel container surrounded by packing. The container is a hollow cylinder with a flat head welded to each end. The design concepts for the waste container for vertical and horizontal emplacement are identical.

This report discusses the results of analyses of aspects of the reference waste package concept needing changes because of new data and information believed applicable to the Deaf Smith County site. Included are waste package conceptual designs for (1) the reference defense high-level waste form from the Savannah River Plant; (2) intact spent fuel with four pressurized-water-reactor or nine boiling-water-reactor assemblies per package for emplacement during Phase 1 of repository operation; and (3) spent fuel which has been disassembled and consolidated into a segmented cylindrical canister with rods from either 12 pressurized-water-reactor or 30 boiling-water-reactor assemblies per package for emplacement during Phase 2. Sufficient maturity exists in the design data to form a basis for the planned advanced conceptual design; these concepts can be baselined to serve as reference package designs.

FOREWORD

The National Waste Terminal Storage Program was established in 1976 by the U.S. Department of Energy's predecessor, the Energy Research and Development Administration. In September 1983, this program became the Civilian Radioactive Waste Management (CRWM) Program. Its purpose is to develop technology and provide facilities for safe, environmentally acceptable, permanent disposal of high-level waste (HLW). HLW includes wastes from both commercial and defense sources, such as spent (used) fuel from nuclear power reactors, accumulations of wastes from production of nuclear weapons, and solidified wastes from fuel reprocessing.

The information in this report pertains to the waste package design studies of the Salt Repository Project of the Office of Geologic Repositories in the CRWM Program. The concept described in this report is the reference waste package design for the site characterization plan.

Conceptual design is the first phase of a multiphase design effort. Its objective is to develop design concepts that are expected to satisfy the waste package design and regulatory requirements based on available engineering expertise and judgment and on preliminary analysis, evaluations, and data.

This reference design concept will be used to develop the site characterization plan. It also is the basis for advanced conceptual design and development programs until information becomes available that indicates the design should be revised. Revisions to this reference waste package design are expected as the multiphase design program progresses.

The reference conceptual design described in this report is based on a concept described by Westinghouse (1986) in Waste Package Reference Conceptual Designs for a Repository in Salt and incorporates new information from analyses performed using new data appropriate for the Deaf Smith County site, new corrosion data for the container, and revisions in the conceptual design for the repository. The radiation dose rates given in the body of this report and the radiation shielding analyses in Appendix D are based on work performed

by Science Applications International Corporation, Columbus, Ohio. The other analyses and the preparation of the report were conducted by the Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, Ohio.

EXECUTIVE SUMMARY

The objective of the waste package program is to translate functional requirements into designs for effective waste packages for the disposal of spent nuclear fuel and high-level wastes in a repository in salt. Conceptual designs for the waste package have been developed for either vertical or horizontal emplacement in a two-phase repository. During Phase 1, intact spent fuel elements will be packaged and emplaced. In Phase 2, the spent fuel will be disassembled and consolidated prior to packaging for emplacement.

Design data are presented in sufficient detail to support development of a site characterization plan. The conceptual designs were developed from reference conceptual designs reported by Westinghouse in Waste Package Reference Conceptual Designs for a Repository in Salt (BMI/ONWI-517) in 1986. The conceptual design is essentially identical for vertical or horizontal emplacement; however, some new analyses were conducted using new geotechnical data appropriate for the Deaf Smith County site, new corrosion data for the container, and information from revisions in the conceptual design for the repository. The primary areas analyzed were

- Repository-scale thermal performance
- Package internal thermal performance
- Corrosion performance
- Structural performance
- Dose rates at the package and emplacement entry surfaces
- Retrieval operations
- Cost.

Only the areas that are changed from the BMI/ONWI-517 report are discussed in this document.

The waste package includes the waste form and all components between the waste form and the host rock. The waste package must (1) provide waste containment for a period of 300 years or more after closure of the repository; (2) limit the release of any radionuclide following loss of containment to

less than 1/100,000 of the curie inventory of that radionuclide present 1,000 years after closure; (3) provide for the safety of operating personnel and the public during handling, transporting, emplacing, and, if necessary, retrieving the waste; and (4) prevent nuclear criticality within a given package.

Waste package designs are provided for different waste forms and types, defense high-level waste, intact spent fuel, and consolidated spent fuel. Defense high-level waste is borosilicate glass incorporating highly radioactive waste sludge that has been cast into a cylindrical stainless steel canister. Spent fuel from pressurized-water reactors (PWR) and boiling-water reactors (BWR) will arrive at the repository as intact assemblies. During Phase 1 of operations, the repository will package intact fuel assemblies with four PWR or nine BWR (4-PWR, 9-BWR) assemblies per package. Subsequent operations, during Phase 2, will involve disassembling the spent fuel and packing the rods into a canister in a closely packed array. A package may contain rods from 12-PWR or 30-BWR fuel assemblies.

All three types of canistered waste are placed into similar containers made of low-carbon steel. In each case the container is a heavy-walled, cylindrical vessel with welded flat heads. The inside diameter of the container is sized to allow a small clearance for each type of canister. The thickness of the container walls is the sum of the structural thickness required to sustain the 18.0-MPa lithostatic pressure, plus the thickness to allow for corrosion to ensure that the structural thickness will endure through the containment period. Different lengths are necessary to accommodate the different waste forms. The space between the container and the wall of the emplacement hole is filled with a crushed salt packing material.

The waste packages will be spaced in the repository so the areal heat loading is no greater than 10 W/m^2 . Thermal analyses show that for vertical emplacement the maximum temperatures are experienced by the 12-PWR consolidated spent fuel packages with a peak fuel cladding temperature of 369°C and an adjacent peak salt temperature of 134°C . The thermal performance of the package is predicted to be essentially equivalent in either vertical or horizontal emplacement modes. The other spent fuel packages have a lower heat

output than the 12-PWR package and thus have lower temperatures. The defense high-level waste packages have such a low heat output that their spacing in the repository is controlled by repository structural constraints rather than thermal constraints, so their areal heat loading is less than 10 W/m^2 and their temperatures are lower than those of the spent fuel packages.

From the results of the analyses performed on the waste package designs it is concluded that there is reasonable confidence that the waste package concepts will satisfy the design requirements, providing assumptions used in the analyses are supported by research in progress.

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LIST OF CONVERSIONS

<u>To Convert</u>	<u>Into</u>	<u>Multiply by</u>
°C	°F	$(^{\circ}\text{C} \times 9/5) + 32$
cm	in	0.3937
kg	lb	2.205
m	ft	3.281
MPa	psi	145.0377
W/m ²	kW/acre	4.047

1.0 INTRODUCTION

1.1 BACKGROUND

The Office of Nuclear Waste Isolation (ONWI) completed a reference conceptual design for a waste package in which to dispose of different waste forms by vertical emplacement in a generic salt repository (Westinghouse, 1986). The package concept developed consisted of a waste form and canister within a heavy-walled, carbon steel container. The types of waste considered were

- Defense high-level waste (DHLW) incorporated in borosilicate glass that has been poured (while molten) into a cylindrical stainless steel canister
- Commercial high-level waste (CHLW) in the same form as the DHLW
- Spent fuel rods that have been consolidated (CSF) in a cylindrical steel canister (removed from fuel assemblies and grouped into a closely packed array).

Several modifications in the repository concept have occurred and additional data have been collected since the above reference concept study was completed. As a result, the Salt Repository Project Office requested ONWI to reexamine this concept to establish a reference waste package for the site characterization plan. The changes in the program that have resulted in the need to review the reference concept are

1. The repository design effort has been broadened to study both horizontal and vertical emplacement of the waste packages.
2. The repository layout has been changed, based on gassy mine regulations and thermal and rock mechanics requirements.
3. A lower areal heat load for spent fuel has been established to reduce repository room temperatures.

4. A repository depth has been established, based on the Deaf Smith County site, which results in a higher in situ stress.
5. A two-phase repository concept has been introduced. This concept includes disposal of intact spent fuel during Phase 1. In Phase 2 most spent fuel will be consolidated prior to disposal.
6. New salt property data have been selected that are believed appropriate for the Deaf Smith County site.
7. Only the West Valley CHLW with a low heat output is expected in the near future, so CHLW glass is not considered separately from DHLW in this study.
8. Additional corrosion data have been collected for the container material, low-carbon steel.

The update of the reference concept includes only those aspects affected by the program changes discussed above. Topics in Westinghouse (1986) unchanged by the new considerations are not covered in this report. Areas of analysis discussed in this report are as follows:

1. A repository-scale thermal analysis was performed because the salt temperature will be affected by the emplacement mode, the reduced areal heat load, and the revised property data for the salt.
2. An internal thermal analysis of the waste package was performed to determine the effect on package temperature of changes in the temperature at the salt/package interface.
3. Corrosion rates were estimated based on the latest corrosion data from the materials testing program.
4. Structural performance was analyzed for the new in situ stresses for the Deaf Smith County site.

5. Emplacement operations for horizontal emplacement are described.
6. Operations to retrieve a horizontal package are discussed.
7. Canisters for intact spent fuel are described.
8. The cost of the container and canisters for intact and consolidated spent fuel are estimated.
9. Radiation exposure rates from emplaced waste packages were calculated.

Although the new analyses have resulted in some minor changes (increased thickness of the container wall, lower predicted temperatures, and increased costs), the basic package concept remains unchanged. The waste form within a thin-walled, metal canister is placed in a thick-walled, low-carbon steel container. After emplacement the container is surrounded by a packing material in the emplacement hole.

Performance of the waste packages has been divided into three time periods:

1. Operating period (defined as being up to 76 years) during which time all repository waste emplacement activities take place (26 years), followed by an observation period (24 years), and either a decommissioning period or a retrieval period (5 or 26 years). During the operating period, waste packages must provide containment, handling capability, radiological safety, and retrieval capability.
2. Containment period (currently defined as over 300 years after permanent closure of the repository) during which time the waste package is required to provide radionuclide containment.
3. Controlled-release or post-containment period during which the engineered barrier system must provide reasonable assurance that the radionuclide release rate to the geologic medium is maintained at an

acceptably low value, specified by the Nuclear Regulatory Commission (10 CFR Part 60) to be no greater than one part in 10^5 per year of the quantity of certain radioisotopes calculated to be present 1,000 years after permanent closure. The DOE has set as a goal that the waste package will meet this requirement.

The conceptual design is the first of several levels of design which will lead to the final waste package design for a licensed repository. During the design process, a number of analyses occur concurrently, each of which is based upon a physical description of the waste package. Each analysis yields some information that improves that physical description. For the final design, all of the analyses will be performed on the same detailed physical description of the waste package, and the results of these analyses will show the design requirements have been achieved. For the conceptual design, however, that physical description is still under development, and all analyses are not being performed with the exact same physical description since it is continually being improved. Thus the reference conceptual design is a snapshot of the physical description of the waste package at a point in time. The description used for each analytical process is not identical, but it can be shown to be conservative and adequate for the level of information it produces.

1.2 REPORT CONTENTS

Chapter 2 provides the design requirements and data assumptions used as design bases. Chapter 3 describes the waste package designs, addresses fabrication techniques and emplacement features important to package performance, and provides the cost estimates. Chapter 4 gives the results of the design analyses of the waste packages. Assessments of design margins are included together with brief descriptions of the methods and procedures used for the thermal, corrosion, structural, and radiological analyses. Chapter 5 is a discussion of the results. Chapter 6 lists the references used in preparation of the report. Appendixes are included that provide backup information for the thermal, corrosion, structural, and radiation shielding analyses.

2.0 REQUIREMENTS AND ASSUMPTIONS

The waste package conceptual design study (Westinghouse, 1986) was responsive to the overall U.S. Department of Energy (DOE) functional requirements and performance criteria (DOE, 1982) and to the interim performance specifications for conceptual waste package designs (ONWI, 1983).

The DOE prepared these requirements and criteria as interim guidance for the waste package designs until the regulatory agencies completed implementation of their regulations and standards. With the publication of the U.S. Nuclear Regulatory Commission's (NRC) regulations, 10 CFR Part 60, and the U.S. Environmental Protection Agency's (EPA) standards, 40 CFR Part 191, pertaining to the geologic disposal of high-level wastes, the interim guidance was largely superseded. However, with development of the Generic Requirements for a Mined Geologic Disposal System (DOE, 1984) and upon publication of the final regulations and standards, the interim requirements and criteria were proven to have been appropriate guidance for the conceptual design studies.

The DOE published the Generic Requirements for a Mined Geologic Disposal System (DOE, 1984) which contains the requirements and criteria for what is technically needed to achieve the national objective of safe waste disposal in an environmentally acceptable and economic manner. The Salt Repository Project's Requirements Document (SRP/RD) (SRPO, 1987) translates the generic requirements into more specific requirements, criteria, and constraints applicable to the development of a repository in salt. These requirements are consistent with the regulations and standards of the NRC and the EPA.

The SRP/RD contains five functional requirements for the waste package for five basic waste forms. The waste forms are spent fuel, defense high-level waste, West Valley high-level waste, hardware, and nonstandard waste. The functional requirements are statements of what the waste package must accomplish. One or more performance criteria for each requirement define how well the waste package must perform that functional requirement. For each of these performance requirements there are more specific requirements for each of the components of the waste package and criteria that state how well those

requirements must be achieved. Constraints are placed on the waste package components to achieve the desired performance as defined by regulatory limits or by the physical limits established by other parts of the system. The application of these requirements to the design of the waste package is intended to achieve compliance with the regulations of the NRC and the standards of the EPA, as well as the objectives of the DOE.

The EPA standards in 40 CFR Part 191 do not place requirements directly on the waste package. Although DOE's siting guidelines, 10 CFR Part 960, contain general guidance for the waste package they refer to the details of the NRC regulations. Thus, the EPA standards and DOE's siting guidelines are not quoted in the waste package design basis. The applicable NRC regulations in 10 CFR Part 60 apply to two distinct time periods, the preclosure period and the postclosure period (described in Section 1.1). The preclosure waste package requirements apply primarily to the operational phase of the repository and include the prohibition of liquids and chemically reactive materials, containment during handling, waste form criteria, and package identification (see 10 CFR 60.135 b and c). These do not have a significant influence on the conceptual design of the waste package or depend on site characteristics and will be fully addressed in subsequent design phases. Retrievability of the waste for a period of up to 50 years after emplacement is a performance objective given in 10 CFR 60.111 and the criteria in 10 CFR 60.135 b are applicable through the retrievability period. Thus retrievability is a consideration in the conceptual design. The postclosure requirements include the containment requirement and the slow release requirement (see 10 CFR 60.113) and the material compatibility requirement (see 10 CFR 60.135 a). The latter is closely related to the containment requirement and not separately considered in the conceptual design. However, the containment and slow release requirements are major influences on the conceptual design of the waste package.

2.1 WASTE PACKAGE REQUIREMENTS

The Salt Repository Project Requirements Document (SRPO, 1987) contains the five functional requirements and the performance criteria for the waste package and the related requirements and criteria for each component. The

following sections of this report contain a compilation of the above taken from the Requirements Document. Following each requirement is a discussion of its relation to the regulations. Functional Requirements Numbers 1, 2, and 3 relate primarily to the operational period of the repository and were not all specifically addressed in this study. They will be fully addressed in the advanced conceptual design and license application design phases of the waste package. Functional Requirements Numbers 4 and 5 pertain to the containment and slow release characteristics of the waste package and are dominant influences in the conceptual design. For convenience, the functional requirements are identified by number in the order each appears in the Requirements Document; they are not so identified in that document.

2.1.1 Waste Package Functional Requirement Number 1

The waste package shall assist in containing waste during normal operations and design-basis accidents related to unloading, handling, storage, packaging at the repository surface facilities, emplacement, and retrieval, if necessary.

2.1.1.1 Component Requirements

The waste form shall

- Limit the release of radionuclides (1) during handling prior to the addition of other waste package components (for spent fuel), or (2) if container and/or canister containment is lost during handling and emplacement (for other waste forms)
- Control releases of radionuclides during repository design basis events (handling accidents, fires, floods, etc.).

The canister shall

- Be capable of being handled and transported as needed

- Contain radioactive gases and radionuclides prior to emplacement into the container and its subsequent closure
- Maintain the structural integrity needed for retrieval.

The container shall

- Be capable of being handled and transported both above and below ground
- Contain waste during handling, emplacement, and retrieval operations
- Have features that may positively assist in retrieval operations.

The packing shall serve as a radiation shield between the container surface and the tunnel surface.

2.1.1.2 Waste Package Criteria

The waste package must

- Remain structurally intact as a unit that contains the waste and provides for safe handling of the waste through the emplacement process and the retrievability period if retrieval becomes necessary
- Be capable of sustaining normal handling and packaging operational loads without loss of containment, and design basis accidents either without loss of containment or with a limited release of radionuclides as required in 10 CFR Part 20
- Be designed and fabricated to allow the retrieval of the waste by the methods planned and within the time period stated in

Appendix D of the Generic Requirements Document (DOE, 1984),
from the emplacement configuration used in the repository.

2.1.1.3 Component Criteria

The waste form shall

- Be handled and stored in a manner to retain its inherent containment properties (for spent fuel) and to limit the availability and generation of particulates in case of an accident occurring during normal operations (for all waste forms)
- Be in a solid form
- Not contain free liquids to the extent that containment is jeopardized during the operational and containment phases by internal corrosion or interactions; and not contain free liquids to prevent the ready transport of radionuclides from the waste form as a result of (1) accidents breaching the cladding or other package components (for spent fuel), or (2) accidents that might occur during repository operations (for all other waste forms)
- Have the fraction of the waste form existing or converted under accident conditions into particles (having a geometric diameter of 200 μ or less for dispersion control and 10 μ or less for respiratory control) that will (1) be controlled by a combination of cladding and canister where appropriate (for spent fuel), or (2) be limited to a value to be determined in the event of a drop from a specified height of the waste form in its primary canister onto an unyielding surface (for all other waste forms)
- Limit the release of volatile materials in the waste form in quantities to (an amount to be specified) in the event of a fire of 800°C and 15 min duration (for spent fuel and nonstandard waste).

The canister shall

- Withstand a drop of (7 m for defense and West Valley high-level waste) (height to be determined for spent fuel and nonstandard waste) onto a flat, essentially unyielding surface without sustaining a breach
- Withstand a fire test based on 800°C and 15 min (for West Valley high-level and nonstandard waste)
- Have handling features to structurally support the canister during handling and moving and not adversely affect other functions of the canister
- Control the release of gases prior to containerization by containing an internal pressure of 50 kPa for defense and West Valley high-level waste, 69 kPa for nonstandard waste, and (to be determined for spent fuel) at 25°C
- Limit the rate of gas release while unbreached to less than 10^{-7} atm cc/sec helium under dry air conditions at 25°C and with a pressure differential of 1 atm
- Maintain its physical strength and integrity up to 84 years following emplacement as a retrieval contingency
- Be processed and packaged at the geologic repository operations area as necessary to meet the regulatory requirements of 10 CFR Part 60 (for defense and West Valley high-level waste) (for nonstandard waste this applies to nonfuel components only).

The container shall

- Have lifting devices, and the mechanical integrity to sustain routine handling and transport

- Have lifting devices remotely engageable from appropriate orientations
- Be standardized in geometry and size, including handling fixtures, for each waste type to the extent practical to be compatible for handling and disposal
- Have handling features compatible with the configuration and requirements of equipment used to grasp and maneuver waste containers
- Sustain the stresses resulting from normal conditions for transporting, handling, emplacing, and retrieving waste while preserving its capability to contain the radionuclides
- Withstand the drop of 7 m for spent fuel, or 5 m for defense high-level waste, West Valley high-level waste, and nonstandard waste onto an unyielding surface without breach
- Not contain materials capable of sustaining combustion reactions resulting from a fire of 800°C and 15 min duration
- Pass short-term containment qualification tests
- Remain sufficiently intact through the retrievability period so that either the loaded container may be removed, or its head can be removed and the inner canister can be withdrawn.

The packing material between the container surface and the tunnel surface shall limit the radiation level at the tunnel surface to (a value to be determined) consistent with the maximum design basis for dose levels in repository work areas, or less than 20 percent of the allowable dose levels from 10 CFR 20.101 and 10 CFR 20.103.

2.1.1.4 Relation to Regulatory Requirements

Requirement Number 1 for the waste package and its related component requirements and criteria correspond to the NRC performance objectives in 10 CFR 60.111 and to its waste package design criteria in 10 CFR 60.135. The first performance objective is for protection against radiation exposures and releases of radioactive material. To this end, it requires that the repository operations be designed so that until permanent closure is completed, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits of 10 CFR Part 20 and the applicable environmental standards. The waste package design criteria are responsive to this objective.

The second performance objective is for retrievability of the waste. For this objective, the repository operations shall be designed to preserve the option of retrieval through the period of the review of the conformance confirmation program. The functional requirement acknowledges that the waste package shares in this responsibility.

The waste package design criteria require the waste to be in a solid form, to have particulates consolidated, and to have combustible constituents reduced to a noncombustible form. The design criteria also state that the waste package shall contain no explosive, pyrophoric, or chemically reactive materials, and no free liquids. Also, the waste package shall be designed to maintain containment during transportation, emplacement, and retrieval. These requirements do not have a significant effect on the conceptual design but will be fully addressed in the advanced conceptual or license application design.

2.1.2 Waste Package Functional Requirement Number 2

The waste package shall limit the potential for criticality within single waste packages.

2.1.2.1 Component Requirements

The waste form shall limit the potential for criticality within a single package.

2.1.2.2 Waste Package Criteria

The waste packages shall be such that single package nuclear criticality shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions related to nuclear criticality safety. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

2.1.2.3 Component Criteria

The waste form shall have a calculated effective multiplication factor, k_{eff} , less than 0.95 for its normal geometry and any credible changes in geometry due to handling, emplacement, or retrieval accidents (including moderation and reflection).

2.1.2.4 Relation to Regulatory Requirements

The NRC's design criteria for the geologic repository operations area, 10 CFR 60.131, addresses criticality control by requiring all systems for transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste to be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred. The Functional Requirement Number 2 for the waste package places responsibility for criticality control of individual waste packages on the properties of the waste form. Criticality safety among multiple waste packages must be assured by controls on the system within which the waste packages reside.

2.1.3 Waste Package Functional Requirement Number 3

The waste package shall provide a means of unique waste package identification.

2.1.3.1 Component Requirements

The canister shall provide a means of unique identification.

The container shall provide a means of unique visible identification.

2.1.3.2 Waste Package Criteria

The waste package

- Shall have a label or other means of clear visible unique identification
- Identification shall not impair the structural or containment integrity of the waste package and shall be applied in such a way that the information shall be legible at least to the end of retrievability
- Identification shall be consistent with the waste package's permanent records.

2.1.3.3 Component Criteria

The canister shall

- Be uniquely identified using a label or marker alphanumeric code system visible from the top and side that does not adversely

affect other functional performance requirements of the canister (the characters shall be at least 92-point sans serif typeface)

- Have characters that identify the canister that are legible through the period of retrievability.

The container shall

- Be uniquely identified using an external surface alphanumeric code system visible from top and side that does not adversely affect the other functional performance requirements of the container
- Have identification legible through the period of retrievability.

2.1.3.4 Relation to Regulatory Requirements

The NRC's design criteria for the waste package, 10 CFR 60.135, requires that a label or other means of identification be provided for each waste package. The identification must not impair the integrity of the waste package and must be legible at least to the end of the period of retrievability. The identification will associate the waste package with its permanent records.

The component requirements for Requirement Number 3 are placed on the identification of the canister as well as the container. This is to ensure proper identification of the waste until it is placed in its container and to allow identifying the waste if the canister alone is retrieved from the emplaced waste package. This requirement will be fully addressed in subsequent design phases.

2.1.4 Waste Package Functional Requirement Number 4

The waste package shall contain the radionuclides for 300 years or more after permanent closure.

2.1.4.1 Component Requirements

The canister shall enhance the removal of heat generated within the array of spent fuel rods.

The container shall maintain substantially complete containment for a period greater than 300 years after emplacement.

The packing shall

- Conduct heat from the container to the host rock
- Provide a uniform, controlled material in contact with the external container surface.

2.1.4.2 Waste Package Criteria

The waste package shall

- Be designed, assuming anticipated processes and events, so that containment of the radioactive waste will be substantially complete for a period of not less than 300 years after permanent closure of the geologic repository
- Have a limited heat generation rate to ensure that the temperatures reached during the containment period do not act to significantly deteriorate the packages' capabilities to act as isolation barriers during that period
- Be designed so that the in situ chemical, physical, and nuclear properties of the waste package and its interactions with the emplacement environment do not compromise the function of the waste packages or the performance of the underground facility or the geologic setting

- Be designed to include, but not be limited to, consideration of the following factors: oxidation/reduction reactions, corrosion, hydriding, gas generation, thermal effects, mechanical strength, mechanical stress, radiolysis, radiation damage, fire and explosion hazards, thermal loads, and synergistic interactions.

2.1.4.3 Component Criteria

The canister shall redistribute the heat generated within the array of fuel pins in such a manner that maximum allowable waste form temperature of 375°C is not exceeded (for spent fuel).

The container shall

- Sustain the thermomechanical stresses up to 18 MPa imposed by the expected repository conditions while providing containment of the waste
- Be chemically compatible with the expected repository conditions to the degree that containment of the waste can be assured with a high degree of confidence - using materials not subject to localized corrosion - providing a corrosion allowance greater than the expected loss through general corrosion.

The packing shall

- Transfer heat away from the container during the containment period in such a way that the maximum allowable temperature in the waste form or any other package component is not exceeded
- Be a well-mixed homogeneous medium that will minimize compositional variations that might cause increased degradation of the external surface of the waste package container with which it is in contact.

2.1.4.4 Relation to Regulatory Requirements

Functional Requirement Number 4 responds to the NRC's technical criterion for containment in 10 CFR 60.113 that requires the waste package to provide reasonable assurance that the containment of radionuclides will be substantially complete for a period of 300 to 1,000 years after permanent closure of the repository, assuming expected processes and events. This is a dominant influence in the waste package design. The requirement for the container influences the selection of the container material and determines the thickness of the container cylindrical wall and top and bottom heads. This requirement has resulted in a large research effort to investigate the performance of candidate container materials under those conditions anticipated to occur over the planned life of the container. This requirement will also be a large influence in the site characterization effort to define the preemplacement environment for the waste package in the repository.

2.1.5 Waste Package Functional Requirement Number 5

The waste package shall contribute to controlling the release of radionuclides after the containment period.

2.1.5.1 Component Requirements

The waste form shall contribute to controlling release of radionuclides after the containment period.

The container shall contribute to the limiting of radionuclide release following the containment period.

The packing shall contribute to controlling the release of radionuclides.

2.1.5.2 Waste Package Criteria

The waste package shall

- Be designed, assuming anticipated processes and events, so that the release rate of any radionuclide from all of the waste packages following the containment period shall not exceed one part in 100,000 per year of the curie inventory of that radionuclide calculated to be present at 1,000 years following permanent closure; provided that this requirement does not apply to any radionuclide that is released at a rate less than 0.1 percent of the calculated total release rate limit; the calculated total release rate limit shall be taken to be one part in 100,000 per year of the curie inventory of the radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay
- Include, but not be limited to, consideration of the following factors: solubility, oxidation and reduction reactions, corrosion, thermal effects, radiolysis, radiation damage, radionuclide retardation, leaching, thermal loads, and synergistic interactions.

2.1.5.3 Component Criteria

The waste form shall not contain material that has been shown to substantially increase the mobility of radionuclides through the package or the host rock.

The container will contribute to radionuclide release control by promoting an internal brine chemistry that limits transport beyond the container boundary.

The packing shall

- Not impair the ability of the container to meet its performance objectives
- Impede the transport of the radionuclides after the breach of the container.

2.1.5.4 Relation to Regulatory Requirements

This functional requirement responds to the NRC regulation that the release of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present 1,000 years following permanent closure except for any radionuclide which is released at a rate less than 0.1 percent of the total rate limit (see 10 CFR 60.113). The engineered barrier system is the waste package and the underground facility. The controlled release requirement that NRC has placed on the engineered barrier system has been allocated by the DOE totally to the waste package (SRP0, 1987, Criterion 5a., p. 1.2.1-3). The component requirements show that the waste form, the container, and the packing have a role in this functional requirement. The component criteria describe how each of these components will perform its function.

2.2 QUALITY ASSURANCE REQUIREMENTS

A quality assurance program fulfilling all applicable requirements of ANSI (1986) NQA-1 shall be applied to the waste package.

From a design standpoint, baselining will define interface parameters. Changes to any of the parameters will require the documented concurrence of all affected parties. Records of that concurrence will be kept as part of the waste package design file.

The quality assurance effort associated with the fabrication and assembly of the waste package naturally falls into three categories: component fabrication, component subassembly, and waste package final assembly. All of the inspection records generated at the repository will be collected for each waste package and assembled with the waste package fabrication and assembly test records. The fabrication and assembly records will have a complete characterization of each component and will allow tracing the origin of each component to the source. Quality assurance for the waste package is described further by Westinghouse (1986).

2.2.1 Materials, Parts, and Processes

Guidelines on materials, parts, and processes are as follows:

1. Avoid creating an underground deposit of materials that may someday be attractive for salvage.
2. Avoid depleting supplies of critical strategic materials.
3. Reduce the need for prolonged and expensive research and development programs that may delay implementation.
4. Take advantage of historical performance experience with those technologies which might be incorporated into repository performance models.
5. Use production and fabrication processes that have been proven to be amenable to inspection and control from a quality assurance standpoint.

2.2.2 Cost

Specific cost criteria for waste packages have not been specified; however, cost considerations are a design variable. Thus, careful consideration

must be given to tradeoffs between cost and design approaches in meeting the requirements for engineered waste package design.

2.3 DATA AND INFORMATION ASSUMPTIONS

The following summarizes the information used in formulation of the reference conceptual designs of waste packages for disposal of defense high-level waste (DHLW), intact spent fuel assemblies (SFA), and consolidated spent fuel (CSF) in this report. West Valley high-level waste is the only commercial high-level waste expected to be received for disposal in the near future. West Valley waste is approximately the same size and weight as DHLW and has a lower heat load than DHLW (DOE, 1984). Therefore, West Valley waste does not need explicit analysis; and, although it is acceptable for disposal in the repository, it is not considered separately from DHLW in this conceptual design.

Three other types of wastes may be accepted in small quantities for disposal at the repository: contaminated fuel assembly hardware from the consolidation of spent fuel, spent fuel that has been consolidated by the utility at the reactor site, and nonstandard waste. The contaminated hardware will have such low levels of radioactivity from neutron activation and contamination that its heat output will be sufficiently low to not be considered a heat producing waste. It will be placed in a canister and inserted into the same container designed for CSF. Since the CSF container is designed to accommodate the conditions occurring from the higher temperatures of a high-heat-output waste, the use of the package for disposal of hardware waste is very conservative. This waste package is not further addressed in this report.

Some utilities are planning to consolidate some of their spent fuel to increase the capacity of their storage pools at the reactors. It is expected that the rods from two fuel assemblies will be inserted into a box having approximately the same dimensions as one fuel assembly. If this consolidated fuel is shipped to the repository during Phase 1, it will be placed into the canister and container planned for the disposal of intact SFA. With the fuel

rods from two fuel assemblies occupying the space of one intact assembly, this waste package can have twice the heat output of the SFA waste package, or 4,400 W for an eight pressurized-water reactor (8-PWR) package. No explicit analysis was done for this case. However, as discussed in Appendix A, the performance of this waste package is bounded by the 12-PWR CSF package. The package for reactor consolidated spent fuel is not addressed further in this report.

Nonstandard wastes include failed fuel rods and other waste items that have characteristics that deviate in a detrimental manner from stated normal acceptance criteria. The requirements for these wastes have not been fully developed. Concepts for packages for nonstandard wastes have not been considered in this study but will be developed in subsequent design efforts.

2.3.1 Waste Forms and Types

Designs for two waste forms are considered: (1) DHLW from the Savannah River Plant, and (2) spent fuel from domestic commercial power reactors, which may be either intact SFA or CSF consisting of individual fuel rods consolidated into a closely packed array. Key design parameters are summarized in Table 2-1 and Figure 2-1.

2.3.1.1 Defense High-Level Waste

DHLW, containing approximately 28 weight percent high-level waste sludge oxides, is the reference defense high-level waste form (Baxter, 1983). It is borosilicate glass that is poured, while molten, into a stainless steel canister, 61 cm in diameter and 3 m long. The canister, shown in Figure 2-2, is filled to 85 percent capacity to preclude overfilling. Maximum thermal output from this waste form is 470 W per canister. The stainless steel canister is suitable for DHLW handling and short-term surface storage.

Table 2-1. Summary of Waste Form Characteristics

Characteristic	DHLW	SFA		CSF	
		PWR	BWR	PWR	BWR
Canister Diameter (cm)	61	67.3	67.3	62	62
Canister Length (m)	3.0	4.24	4.57	4.00	4.35
Number of Fuel Assemblies	-	4	9	12	30
Net Waste Weight (kg)	1,470	1,840(a)	1,700(a)	5,530(a)	5,670(a)
Waste Form and Canister Weight (kg)	1,940	3,170	2,985	8,470	8,780
Power Output (Initial) (W)	470	2,200(b)	1,620(b)	6,600(b)	5,400(b)

(a) Expressed as kgU for Spent Fuel.

(b) Heat output assumes 10-year decayed spent fuel.

2.3.1.2 Intact Spent Fuel

The canister concept for intact SFA allows 4-PWR fuel assemblies or nine boiling-water reactor (9-BWR) fuel assemblies to fit into a waste container with an inside diameter of 69.2 cm. During Phase 1, no consolidation will occur at the repository and only intact SFA will be packaged for emplacement in the repository. Some intact SFA will be packaged for disposal in Phase 2, also.

2.3.1.3 Consolidated Spent Fuel

CSF consists of spent fuel rods that have been removed from commercial light-water reactor fuel assemblies and consolidated into a closely packed array. The number of rods per waste package depends on the type of fuel assembly. Waste packages for PWR spent fuel contain the rods from up to 12 fuel assemblies, while BWR spent fuel waste packages contain the rods from up to 30 fuel assemblies. CSF will be packaged for emplacement in the repository during Phase 2.

FIGURE 2-1

Battelle Project Management Division	Title <u>NORMALIZED WASTE FORM POWER DECAY</u>		
	CHARACTERISTICS		
	Drawn by <u>J. McLEAN REARDON</u>	Date <u>1-23-86</u>	Drawing Number <u>HE-SK-</u>
	Originator <u>L.A. SMITH</u>	Date Reviewed <u>1-23-86</u>	<u>24104-04</u>
	Date _____	Revision Number _____	

2.3.1.1 Defense High-Level Waste

DHLW, containing approximately 28 weight percent high-level waste sludge oxides, is the reference defense high-level waste form (Baxter, 1983). It is borosilicate glass which is poured, while molten, into a stainless steel canister, 61 cm in diameter and 3 m long. The canister, shown in Figure 2-2, is filled to 85 percent capacity to preclude overfilling. Maximum thermal output from this waste form is 470 W per canister. The stainless steel canister is suitable for DHLW handling and short-term surface storage.

2.3.1.2 Intact Spent Fuel

The canister system for intact SFA allows 4 pressurized-water reactor (PWR) fuel assemblies or 9 boiling-water reactor (BWR) fuel assemblies to fit into a waste container with an inside diameter of 69.2 cm. During Phase 1 no consolidation will occur at the repository and only intact SFA will be packaged for emplacement in the repository.

2.3.1.3 Consolidated Spent Fuel

CSF consists of spent fuel rods which have been removed from commercial light-water reactor fuel assemblies and consolidated into a closely packed array. The number of rods per waste package depends on the type of fuel assembly. Waste packages for PWR spent fuel contain the rods from up to 12 fuel assemblies while BWR spent fuel waste packages contain the rods from up to 30 fuel assemblies. CSF will be packaged for emplacement in the repository during Phase 2.

2.3.2 Geotechnical and Repository Data

Summaries of geotechnical parameters and repository characteristics are given in Tables 2-2 and 2-3, respectively.

766589-1A

Figure 2-2. Defense High-Level
Waste Form

2.3.2 Geotechnical and Repository Data

Summaries of geotechnical parameters and repository characteristics are given in Tables 2-2 and 2-3, respectively.

Table 2-2. Summary of Geotechnical Parameters

Lithostatic Pressure at Repository Level (MPa)	18.0
Salt Decrepitation Temperature (°C)	250
Salt Density (kg/m ³)	2,200
Brine Content (vol%)	4.1
Thermal Conductivity (W/m - K)	4.7
Specific Heat Capacity (J/kg K)	879
Ambient Temperature (°C)	27

Table 2-2. Summary of Geotechnical Parameters

Lithostatic Pressure at Repository Level, MPa	18.0
Maximum Design Pressure ⁽¹⁾ , MPa	18.0
Salt Decrepitation Temperature, °C	250
Salt Density, kg/m ³	2,200
Brine Content, Vol%	4.1
Thermal Conductivity, W/m - K	4.7
Specific Heat Capacity, J/kg K	879

- (1) The maximum design pressure may be the sum of the lithostatic pressure and a thermally induced pressure transient. Determination of this latter pressure is under study by ONWI.

3.0 DESIGN DESCRIPTION

The waste package includes the waste form and all components between the waste form and the host rock. The components for the conceptual designs described in this report are the waste form, a thin-walled metal canister, a thick-walled, low-carbon steel container, and a packing between the container and the host rock. The material for the packing is crushed salt. The packing is not specifically addressed in this report, except in its role in heat transfer in the thermal analyses, its use for shielding in the radiation analysis, and its responsibility in the functional requirement for limiting radionuclide release from the waste package.

Waste package concepts have been developed for three types of waste, defense high-level waste (DHLW), intact spent fuel assemblies (SFA), and consolidated spent fuel (CSF). Descriptions are provided in this chapter for waste packages, and spent fuel canisters' fabrication procedures, emplacement designs, and estimated costs. All of the waste packages are generically similar.

3.1 WASTE CONTAINER

The container is comprised of a cylindrical body with two flat heads as shown in Figure 3-1. The physical size of the waste packages is a function of the container material chosen to meet the containment requirements and the waste form size.

3.1.1 Container Material

The material chosen for the container is a cast steel, ASTM 216, Grade WCA (ASTM, 1987). This low-carbon steel has a predictable corrosion performance and adequate strength while being plentiful, easily formed, and weldable. The selection of this material is described by Westinghouse (1986,

Table 2-3. Reference Repository Characteristics for Vertical and Horizontal Emplacement of Waste Packages

	DHLW		SFA (4 PWR)		CSF (12 PWR)	
	Emplacement Mode		Emplacement Mode		Emplacement Mode	
	<u>Vertical</u>	<u>Horizontal</u>	<u>Vertical</u>	<u>Horizontal</u>	<u>Vertical</u>	<u>Horizontal</u>
Depth Below Surface (m)	750	750	750	750	750	750
Maximum Salt Temperature (°C)	250	250	250	250	250	250
Entry Width (m)	6.10	7.32	6.10	7.32	6.10	7.32
Entry Height (m)	7.32	4.57	7.32	4.57	7.32	4.57
Entry Spacing (m)	36.58	36.58	51.82	51.82	51.82	51.82
Waste Package Pitch (m)	2.65	3.05	4.30	8.59	12.89	25.9
Entry Length (m)	365	365	365	365	365	365

pp. 139-146). Whereas the original selection was based on an earlier version of the specification (the 1977 specification), the slight changes in three impurity levels that were made in subsequent versions are not believed to be significant to this application.

3.1.2 Dimensions

The inside dimensions of the waste container are predicated on the dimensions of the waste forms described in Chapter 2. The physical dimensions of components of the waste package are shown in Figure 3-2. The dimensions were developed following consultation and concurrence with the repository design contractor.

For DHLW the length of the internal container cavity is the canister length plus a longitudinal clearance of 5 cm. This results in a cavity length of 305 cm. The cavity diameter is 2.5 cm larger than the outside diameter of the DHLW canister.

The spent fuel waste forms have a variety of lengths because of different fuel designs (Westinghouse, 1986, p. 61). The CSF canister lifting fixture is assumed to be a socket that fits into void spaces inside the canister so as not to require any additional vertical clearance for a lifting fixture. The SFA canister has a lifting pintle attached to the top cover.

During Phase 1 only one length container will be used for spent fuel. A container with a 445.0-cm cavity will hold intact pressurized-water reactor (PWR) or boiling-water reactor (BWR) assemblies. For the BWR assemblies, it will be necessary to remove the end fittings from the assemblies to fit into this container.

Phase 2 will use two different length containers. A short container with a cavity length of 414.5 cm will be provided for consolidated PWR fuel rods. A longer container with a cavity length of 448.1 cm will be used for consolidated BWR rods and PWR assemblies that cannot be disassembled.

Figure 3-1. Schematic Description of Reference Conceptual Waste Packages

The wall and head thicknesses given in Figure 3-2 were established as the sum of the structural thickness required to resist the lithostatic pressure plus an estimated 1,000-year general corrosion penetration. The depth of general corrosion penetration was calculated on the basis of having adequate brine to support the corrosion process while the corrosion rate is controlled by the kinetics of the process (see Section 4.4.3). The values for corrosion penetration were added to the respective wall thicknesses necessary for structural purposes to obtain the total wall thicknesses for the DHLW, SFA, and CSF container designs (see Section 4.4.4).

Prior to having corrosion data to do the analyses for the corrosion allowances, it was necessary to provide dimensions for the waste package (primarily to estimate container weights) as input to the repository design effort. To determine the dimensions, an estimate, based on previous design studies, was made of the thickness of the steel that would be corroded in 1,000 years. This thickness was added to the wall thickness needed for structural purposes to obtain the total wall thickness for the waste containers. Later, when corrosion analyses were performed using the preliminary method described in Appendix B, it was found that the estimated allowances for the DHLW and SFA containers had indeed been adequate since each estimated corrosion allowance was larger than the calculated value. However, the estimated corrosion allowance of 2.3 cm for the CSF container was less than the calculated value of 2.7 cm. In the next design level, additional corrosion data and improved mechanistic models will be available to more confidently predict the corrosion performance of the container. Since the repository design studies were well along, it was decided to continue to use dimensions based on the estimated corrosion allowances and defer changes in the container dimensions until the next design level. Although the analyses from the currently available data do not support the value of the corrosion allowance used for the CSF container, they do show that only a small correction (less than 5 percent of the total wall thickness) is necessary.

3.1.3 Waste Package Weight

The calculated weights for the waste packages with the reference waste forms are given in Figure 3-2.

3.2 SPENT FUEL CANISTER DESIGN AND FABRICATION DESCRIPTION

The design concepts that have been developed for SFA and CSF canisters are described in this section.

3.2.1 Intact Spent Fuel Canisters

During Phase 1 of the repository operation, intact SFA will be packaged for disposal. Cross sections of typical canisters for four PWR (4-PWR) and nine BWR (9-BWR) assemblies are shown in Figure 3-3.

Each canister consists of four components: a baseplate, a cylindrical shell, a fuel cage, and a cover plate with a lifting pintle. A separate fuel cage design is necessary for PWR and BWR assemblies. A cage is made of three circular plates, each having a square hole to position a 2 by 2 array of PWR assemblies or a 3 by 3 array of BWR assemblies. The 0.3-cm thick plates are spaced longitudinally in the canister by long steel rods. The cylindrical shell is fabricated from 0.6-cm thick steel.

The baseplate and cover plate are cut to size from 1.25-cm thick plate followed by machining their joining surfaces. The baseplate is welded to the canister shell and the fuel cage is inserted into the canister. The handling pintle is attached to the canister cover. The design of the handling device and the handling of the canister assembly will be determined in cooperation with the design of the repository waste handling facility.

The intact SFA is loaded remotely into the canister assembly in the waste-packaging cell. The canister is closed by remotely welding the cover to the canister shell. A typical canister for SFA is shown in Figure 3-4.

3.2.2 Consolidated Spent Fuel Canisters

A cross section of a typical canister for CSF is shown in Figure 3-3. The canister is segmented into six congruent compartments. Each compartment

Waste Form Power Watts	DHLW 470	SFA (4 PWR) 2200	SFA (9 BWR) 1620	CSF (12 PWR) 6600	CSF (30 BWR) 5400
<hr/>					
Canister					
D _C	61.0	67.3	67.3	62.0	62.0
L _C	300.0	422.7	422.7	400.0	435.1
L _H	231.0	424.2*	424.2*	385*	409*
<hr/>					
Container					
D _I	63.5	69.2	69.2	64.5	64.5
D _O	84.8	94.8	94.8	88.4	88.4
L _O	338.3	484.6	484.6	451.1	484.6
T _C	10.7	12.8	12.8	11.9	11.9
T _H	16.8	19.8	19.8	18.3	18.3
<hr/>					
Weights, kg					
Waste Form	1,470	2,670	2,485	8,000	8,280
Canister	470	500	500	470	500
Container	<u>7,250</u>	<u>13,520</u>	<u>13,520</u>	<u>11,340</u>	<u>12,060</u>
Total Package	9,190	16,690	15,505	19,810	20,840

Figure 3-2. Waste Package Dimensions

	<u>NOMINAL SIZE, CM</u>	
Fuel Type	<u>A</u>	<u>B</u>
4-PWR	67.3	46.2
4-BWR	67.3	45.2

Figure 3-3 Cross Section of Intact Spent Fuel Canister
for 4-PWR or 4-BWR Assemblies

FIGURE 3.4 ~~3.4~~ TYPICAL INTACT SPENT FUEL CANISTER

* TWO SIZES PROVIDED

DRAWING NOT TO SCALE

ALL DIMENSIONS IN
CENTIMETER

THE FUEL ELEMENT
CONTAINER IS

will hold the rods from either 2-PWR or 5-BWR assemblies. The compartment dividers not only serve as walls to facilitate incremental filling of the canister, but are effective heat conduction paths which reduce the temperature in the interior of the canisters. The vacant space at the center of the canister cross section accommodates a stem that extends from the canister baseplate for handling purposes. The canister concept is shown in Figure 3-5.

The CSF canisters comprise four basic elements: a baseplate, a handling stem, six identical canister segments, and a cover plate. Fabrication of the baseplate and cover plate will consist of cutting plate stock to size and machining the attachment appurtenances. Similarly, the handling stem will be cut, provided with a handling configuration (interface geometry to be determined) at its upper end, and threaded for attachment to the baseplate at the lower end. The canister segments are designed to be made of 0.2-cm (U.S. 14 gage) thick carbon steel.

Three canister segments will be assembled into a semicircle by welding them together along their interior and exterior seams. Meanwhile the handling stem will have been assembled to the baseplate. Two of the canister segment subassemblies will then be mounted on the baseplate subassembly and welded along their external seams and to the baseplate.

The prepared canister is filled with spent fuel rods in the waste-packaging cell. The cover plate will be installed by remotely welding the cover plate to the circumferential walls of the canister segments and to the handling stem to seal the canister.

3.3 EMPLACEMENT

Waste package emplacement criteria have been evolved during the repository and waste package design processes. An iterative process is required because emplacement is governed both by repository and waste package constraints. Emplacement criteria ultimately affect the waste package performance. Schematic layouts of package emplacement for DHLW, intact SFA, and CSF for both vertical and horizontal emplacement are shown in Figures 3-6 and 3-7,

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Figure 3-5 Typical Consolidated Spent Fuel
Canister Cross Section

WASTE FORM	WASTE PACKAGE SPACING, m Y	ENTRY SPACING, m Z
DHLW	2.65	36.58
SFA	4.30	51.82
CSF	12.89	51.82

3-6
FIGURE 3-7 Repository Dimensions for Vertically
Emplaced Waste Packages

respectively. The layout for the two types of spent fuel waste, SFA and CSF, is based on an areal heat load of nominally 10 W/m^2 . The areal heat load constraint is to obtain reasonable temperatures for occupancy of the repository entries in the case that waste retrieval activities are necessary. Due to the very low power in the DHLW package, it is not possible to place the waste closely enough to reach 10 W/m^2 . The layout for this waste package is controlled by a minimum spacing to retain sufficient rock thickness between emplacement holes to avoid breakout during drilling.

For vertical emplacement, the waste package is lowered remotely into a borehole in the floor. The borehole diameter provides a small clearance on the diameter for the waste container. After emplacement, the container is covered by pouring crushed salt into the borehole to the level of the floor. The depth of the borehole (6.5 m for the 12-PWR CSF package) is determined by the length of the waste container plus the thickness of the crushed salt cover needed to limit the radiation level to 0.5 mrem/hr at the tunnel floor.

All spent fuel containers will be placed in boreholes with the same cover depth, which is based on the radiation shielding for a 12-PWR CSF waste package. The 30-BWR CSF, and the 4-PWR or 9-BWR intact SFA waste packages will all have lower dose rates at the floor surface because they have less fuel loading than the 12-PWR CSF case. The detailed shielding analysis for the 12-PWR CSF vertically emplaced waste package is given in Appendix D.

For horizontal emplacement, the container is to be inserted into a horizontal hole in the entry pillar. A hole is cut into the pillar to receive a container, then a slightly larger diameter counter bore is made at the hole opening to receive the shield plug. A container transfer cask is aligned with the emplacement opening and the container is remotely inserted into the hole. Next, the shield plug is remotely installed. The container should not be exposed at any time that workers are near due to the potential for high radiation exposure. Once the shield plug is installed, dose rates at the wall of the emplacement entry will be reduced to less than 0.5 mrem/hr . The plug for the horizontal emplacement hole is shown in Figure 3-8. Large clearances are provided for the container and shield plug to allow reasonable borehole tolerances and ease of remote installation. Detailed shielding calculations for

PLAN VIEW

SECTION A-A'

WASTE FORM	WASTE PACKAGE SPACING, m Y	ENTRY SPACING, m Z
DHLW	3.05	36.58
SFA	8.60	51.82
CSF	25.91	51.82

FIGURE 3.8 Repository Dimensions for Horizontally
Emplaced Waste Packages

a plug for a 12-PWR CSF waste package are given in Appendix D. These analyses show the shield plug designed for the 12-PWR CSF case should be adequate for all packages (see Section 4.1.3).

3.4 COST ESTIMATES

Estimated costs for the canisters and containers, based on quotations from commercial vendors, are provided in Table 3-1. Estimated costs for the canisters assume

- Material is U.S. 14 gage carbon steel
- The consolidated canisters are composed of four elements as described in Section 3.2.2
- Canister is roll-formed in one piece from tubing or seam-welded continuous strip
- Stem is threaded for attachment to baseplate
- Segments have interior and exterior welds
- Material is ultrasonically tested and welds radiographed
- The weight of all canisters is 370 kg as the reference cost per unit weight
- Material cost is \$2.00/lb
- Quality assurance program equivalent to ANSI/ASME NQA-1 (ANSI, 1986) is applied; no N-stamp required but material intended for an NRC licensable facility
- Contingency allowance is equal to 20 percent of the fabrication cost

FIGURE 3-1 PROPOSED SHIELD PLUG DESIGN
E-E HORIZONTAL EMPLACEMENT

- Canisters are supplied in equal quantities over a 25-year span
- Shipping cost is \$0.05/lb
- Estimate accuracy is ± 30 percent.

The estimates for the containers assume

- The bodies are centrifugally cast, heads are static castings
- Castings are heat treated with furnace charts required
- Castings are machined for inspection and slag removal
- Castings are ultrasonically tested with radiography used to reconcile any anomalies identified in test results
- Castings are machined for weld preparation, top heads are threaded for lifting device
- Contingency allowance is equal to 20 percent of fabrication cost
- Containers are supplied in equal quantities over a 25-year span
- Shipping cost is \$0.05 lb
- Estimate accuracy is ± 30 percent.

Table 3-1. Cost Estimates (Mid-1986 Dollars)

Item	DHLW	SFA		CSF	
		4 PWR	9 BWR	12 PWR	30 BWR
<u>Canister</u>					
Fabrication	0(a)	2,280	2,280	2,080	2,200
Shipping	0	55	55	52	55
QA + Contingency	0	2,280	2,280	2,080	2,200
Total	0	4,615	4,615	4,212	4,455
<u>Container</u>					
Fabrication	24,000	44,700	44,700	37,500	39,900
Shipping	768	1,430	1,430	1,200	1,277
QA + Contingency	12,000	22,350	22,350	18,750	19,950
Total	36,768	68,480	68,480	57,450	61,127
<u>Waste Package</u>					
Total	36,768	73,095	73,095	61,662	65,582
Unit Cost	25.01/kg	39.73/kg of U	43.00/kg of U	11.15/kg of U	11.57/kg of U

(a) The DHLW canister will be supplied by the waste generator.

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4.0 PERFORMANCE DESCRIPTION

This chapter presents a discussion of the performance evaluation of the conceptual designs for the waste package relative to the design requirements given in Chapter 2. The first three functional requirements are related to preclosure functions of the waste package and have only been partially addressed by specific analyses during conceptual design. These requirements will be fully addressed in subsequent design phases. The last two requirements are for postclosure containment and controlled release of radionuclides and have had the greatest influence on the conceptual design. The responses to requirements are described below in the order the requirements are presented in Chapter 2.

4.1 WASTE PACKAGE FUNCTIONAL REQUIREMENT NUMBER 1

The waste package shall assist in containing waste during normal operations and design-basis accidents related to unloading, handling, storage, further packaging at the repository surface facilities, emplacement, and retrieval, if necessary.

The component requirements and criteria for this functional requirement address the ability of the physical nature of the waste forms to restrict mobility of the waste and the ability of the canister and container to contain the waste under expected and potential accident conditions during operations of the repository. The waste form requirements will be applied to the waste generators by waste acceptance requirements being developed for the repository. Design basis accidents will be defined and allocation of requirements to the waste package components will be made during the advanced conceptual designs of the repository and waste package.

4.1.1 Containment

The containment capabilities of the spent fuel canister concepts have not been analyzed but have been an influence in the selection of materials and types of closures described for the canisters. Canisters for glass waste forms have received more design evaluations (Baxter, 1983). Each of the criteria for containment for the canister will be addressed in future design studies.

By the nature of the massive number of container designs that have resulted from the postclosure requirements, it is believed that the container will meet the containment function of this requirement. However, no specific analyses have been performed to demonstrate containment for handling operations. The container criteria also address handling of the container and standardization of designs. Handling appurtenances for the container will be developed in conjunction with the repository design in future design studies. However, the similarity of the designs for the different waste types show that standardization has influenced the conceptual designs.

4.1.2 Retrieval

The first functional requirement, in addressing containment through the possible retrieval process, acknowledges that the container (and possibly the canister) has a role in retrieval. Although retrieval has not been a large influence in the conceptual design, it has been a consideration. A full discussion of the waste retrieval scenario is beyond the scope of this report. However, a presentation of the waste package-related events is made to show how waste retrieval can be accommodated in these waste package designs. Since a retrieval method has not been selected at this time, several possible methods will be outlined.

The creep characteristics of salt are expected to cause the waste containers to be bound into their emplacement holes at the time retrieval would be accomplished. One method of freeing the package and preparing to move it into the entry is overcoring. In this operation a kerf (or slit) is

cut around the package to form a salt core. The core (package and surrounding salt) can be latched by a fixture for moving. It can then be broken free and moved from the emplacement. Different equipment is needed to perform these operations with vertical and horizontal emplacement. These concepts are described below.

Retrieval of a vertically emplaced container by overcoring involves boring an access hole in the floor of the emplacement entry, removing salt from over the container to expose the pintle, and overcoring around the container. This may be done by first using a boring machine to remove salt to create a cylindrical opening directly over a container. Next, a bell housing containing a remotely operated mining machine will be placed over the opening. Using closed-circuit television to observe progress, salt will be removed to expose the pintle. The bell housing is then replaced by an overcoring machine. A large cylindrical cutting body is used to cut a kerf around the container. The bell housing is then repositioned over the container and the pintle is cleaned to prepare for extraction of the container. Lastly, a transfer cask replaces the bell housing above the container and the container is then latched and raised into the cask.

Retrieval of a horizontally emplaced container involves excavating a small alcove or side entry in the wall of the emplacement entry to gain access to the pintle at the end of the container and then cutting a kerf around the container to free it. To do this, a small single-boom continuous miner, called a side entry miner, excavates the side entry. The side entry miner is remotely operated with visual control by closed circuit television. Once the side entry is mined and the mining machine removed, a container extraction machine is positioned in the side entry. The extraction machine cuts a kerf around the upper section of the container. It then engages the disposal container with a clamping mechanism and extracts the container for insertion into a transfer cask.

The physical condition of the container at the time of retrieval will be a concern. The container is designed to maintain its structural integrity under a lithostatic load of 18 MPa for up to 1,000 years. Although higher rates of corrosion are expected during the years immediately after emplacement

due to higher temperatures during that time, some of the corrosion allowance will remain at the time of retrieval, so the structural capacity will not be degraded. Loads on the container commensurate with lithostatic pressure should be acceptable during retrieval. If the pintle is to be used for lifting the container during retrieval, special consideration of the pintle material will be necessary to ensure that it maintains its integrity.

An alternative retrieval method is to retrieve only the waste form canister. Depending upon details of the final container design, the salt above the container would be removed using core bits, at least when nearing the top of the container. Once the top of the container has been exposed, the weld securing the pintle to the head will be removed and the pintle will be removed to provide access to the threaded hole in the head. This hole is then used for attachment of a cutting machine. The cutting machine will rotate around the waste package head, making a trepan groove inside the weld and into a relief undercut in the head below the weld. Once the trepanning is complete, the cutting machine and the container top head can be removed, exposing the waste form canister. The container was designed to resist buckling even with the top head removed; therefore, there will be no restraint to lifting the waste form canister out of the remaining container components. In addition, since the waste form canister has not been exposed to the repository environment, its handling device should not have been degraded.

With the waste form canister exposed, it can be withdrawn into a shielded transfer cask and transported to a shielded storage facility to await further disposition. The overpack components can remain in the repository or be removed by conventional methods, as desired.

4.1.3 Radiation Levels

The first functional requirement addresses the waste package role in radiation safety from the aspect of containing the waste during repository operations. However, the component criterion for the packing requires it to act as a radiation shield in the emplacement hole between the container and the emplacement entry. Personnel working in the vicinity of high-level waste

(HLW) and spent fuel must be protected from radiation emanating from the waste. During shipping, repository transport, emplacement, and most handling operations, this is accomplished through the use of shielded casks and equipment. During packaging and the associated handling, operations will be conducted in a hot cell to provide radiological safety. Shielding requirements and designs for those operations are beyond the scope of this report. However, estimates of the radiation levels at the surfaces of the waste package have been developed for the use of the respective design entities. The radiation analyses are described in Appendix D.

The radiation levels at the side surfaces of the container are shown in Table 4-1. The container surface radiation levels were obtained by computer code calculations of gamma ray and neutron transport through steel around a defense high-level waste (DHLW) form and a 12 pressurized-water reactor (12-PWR) consolidated spent fuel (CSF) waste form. Radiation levels around a backfilled borehole for vertical emplacement and a borehole shield plug for horizontal emplacement with a 12-PWR CSF package are shown in Figure 4-1. The dose rates for horizontal emplacement were based on a preliminary shield plug design having 64 percent of the length and larger diametral clearances than the present design. Thus dose rates using the present design can be expected to be less than those predicted for the preliminary design.

The 30 boiling-water reactor (30-BWR) CSF package will have similar or slightly lower doses than the PWR case because it has a lower radiation source term. The Phase 1 4-PWR spent fuel assembly (SFA) container has a thicker

Table 4-1. Radiation Levels at the Side Surfaces of the Waste Packages

Waste Package	Neutrons (reflected), mrem/hr	Gamma, mrem/hr	Total, mrem/hr
DHLW	$9 \pm .5 \times 10^{-2}$	$2 \pm .5 \times 10^3$	$2 \pm .5 \times 10^3$
CSF (12-PWR)	$5 \pm .5 \times 10^1$	$7 \pm 3 \times 10^3$	$7 \pm 3 \times 10^3$

Figure 4-3. Total Dose Rates At Entry Surface for Emplaced 12-PWR CSF Waste Packages, mrem/hr

wall with a lower waste loading, and therefore will have a lower dose rate than the 12-PWR package. Shielding analysis was not performed for the SFA or BWR CSF packages. The DHLW package has a gamma dose about one third as high as the 12-PWR CSF package at the surface of the container, and the neutron dose from the DHLW package is over 100 times less. In design of the shield plug, neutrons will be the limiting factor due to their greater penetrating power. Therefore, a shield plug designed for the 12-PWR CSF case should be adequate for all packages and was used for all waste types.

4.2 WASTE PACKAGE FUNCTIONAL REQUIREMENT NUMBER 2

The waste package shall limit the potential for criticality within single waste packages.

The component requirements and criteria for this functional requirement apply directly to the waste forms. However, since the geometry of the waste form is important to criticality control, the canister design also is important.

DHLW does not contain sufficient fissionable material to constitute a criticality hazard (Baxter, 1983, p. 18). Intact and consolidated spent fuel in canisters will remain subcritical in the dry state. Consolidation of spent fuel is expected to be performed under dry conditions as is the installation of the loaded canister into the container. In the unlikely event that the canister becomes filled with water, it is estimated, although confirming calculations have not been made as a part of this design phase, that the closely packed fuel rod array of CSF will be sufficiently undermoderated to prevent achieving a critical condition. Based upon this information, CSF should not constitute a criticality hazard during either the repository operating or containment periods. The unlikely event that the canister of SFA becomes filled with water has not been analyzed. This will be addressed in subsequent waste package design phases in conjunction with repository design efforts to consider physical and administrative controls for assured criticality safety of SFA packages.

4.3 WASTE PACKAGE FUNCTIONAL REQUIREMENT NUMBER 3

The waste package shall provide a means of unique identification.

The component requirements and criteria for this functional requirement apply to the canister and the container. The content of each waste container must be traceable to its source through the retrievability period. No specific methods for identification have been specified during the conceptual design studies. However, sufficient corrosion of the low-carbon steel containers may occur during the retrievability period that marking directly on the container is not deemed feasible. An identification plate made of a highly corrosion-resistant material has been recommended (Westinghouse, 1986).

4.4 WASTE PACKAGE FUNCTIONAL REQUIREMENT NUMBER 4

The waste package shall contain the radionuclides for 300 years or more after permanent closure.

This functional requirement has been a dominant influence in the conceptual design of the waste package. The associated waste package criteria relate to limiting the heat output of the packages to obtain acceptable temperatures and designing to achieve physical and chemical compatibility with the anticipated repository conditions.

The anticipated conditions are (1) a rapid temperature rise after emplacement of the container followed by a long period of temperature decline, (2) an ingress of brine resulting in the corrosion of the container, and (3) an early imposition of an external pressure due to the creep of the salt. The heat from the container will cause the temperature in the surrounding host rock to rise. The increased temperature will cause thermal expansion and an increase in the creep rate of the salt, which will result in the application of pressure to the container. The timing and extent of the pressure rise depends on the borehole-to-container clearance and the salt temperature. Pressure loads from thermal expansion of the salt will be mitigated by the

proper size of the gap between the container and the host rock and are finally limited to the lithostatic pressure by the creep of the salt. For analysis of the structural capacity of the container, the lithostatic pressure is assumed to be imposed on the container immediately after emplacement. The changes in temperature and pressure in the host rock may cause brine movement by several mechanisms operating singly or in combination. Brine inclusions within the salt crystals will move up a thermal gradient toward the waste package. Pressure differentials or diffusion may cause brine movement either toward or away from the waste package.

The results of analyses of the above conditions on the conceptual waste package designs are given in the following subsections.

4.4.1 Thermal Performance

The waste package temperature has a direct effect on both the corrosion rate and the structural capability of the container. Calculation of the waste package thermal performance (see Appendix A) is performed in two parts. In the first part, a three-dimensional, finite-element model of the repository, including either a vertically or horizontally emplaced waste package, is used. This model is capable of transient analysis and provides temperatures within the repository and at the interface between the salt and the waste package at specified time intervals after waste package emplacement. The repository model was developed with the conservative assumption that the waste package is within a field of concurrently emplaced identical packages. This assumption gives the highest areal heat load for each waste type and results in the highest predicted temperatures, which is conservative. Therefore, quarter symmetry is assumed and adiabatic boundary conditions are applied to all vertical faces of the repository model.

In the second part, temperature profiles in waste packages were determined at steady state for various surface temperatures. With DHLW a one-dimensional model is adequate. In the CSF package, internal fins were used to reduce waste form temperature. A two-dimensional model is needed for canisters with the internal fins. The temperatures of the intact SFA containers

were analyzed by a one-dimensional numerical correlation. After determining the container surface temperature from the repository calculations, the waste form temperature for this surface temperature was found from the waste package model. Decoupling the calculation at the container surface is valid because the waste package responds much faster than the repository to temperature change. PWR spent fuel packages were analyzed as the upper bound condition since they have higher heat output in all cases than BWR fuel packages.

The waste package concept includes design features that enable heat to be transferred from the waste form to the host rock in such a way that the temperature limits of the materials are not exceeded. These limits are 250°C for the salt to avoid decrepitation (Westinghouse, 1986, Table 2-2), 375°C for spent fuel to promote cladding integrity (Westinghouse, 1986, Table 2-1), and 450°C for DHLW (DOE, 1986a, p. 4) to inhibit devitrification. The limiting temperature for carbon steel is significantly greater than the allowable salt temperature and was therefore not considered. Furthermore, since the waste form canisters will not be exposed to the repository environment under the conditions expected during the containment period, canister temperatures and their effect on interactions with the repository environment are not a concern.

When the container is emplaced in the borehole a gap will exist between the cylindrical surfaces of the container and the borehole. This gap may contain air or, in the case of vertical emplacement, crushed salt. With air in the gap, heat will be transferred by radiation and convection; with crushed salt, the heat will be transferred by conduction. In either case a significant temperature drop can occur across the gap. Within a few years the gap will close by the effects of salt creep and thermal expansion, so good heat transfer will be achieved at the container-borehole interface and no significant temperature drop will occur. The thermal analysis to calculate peak temperatures in the waste package accounts for the effects of the gap. For horizontal emplacement, contact of the container with the salt at the lower side of the borehole will allow some conduction, which will improve heat transfer. This is not accounted for in the model, so the one-dimensional calculation predicts conservatively higher temperatures than can be expected to occur.

FIGURE A-2 DEFENSE HIGH-LEVEL WASTE PACKAGE TEMPERATURES

4.4.1.1 Defense High-Level Waste

The emplacement density for DHLW is controlled by physical rather than thermal limits, see Section 3.3. As a result, DHLW is emplaced at less than the design areal thermal load of 9.88 W/m^2 . With horizontal emplacement, there is effectively a double row of packages, one row in each side wall of the entry. For vertical emplacement, there is a single row of waste packages along the center of the emplacement entry. Thus, with the same package heat output and room spacing, and with the minimum allowed package spacing, horizontal emplacement gives almost double the areal thermal load of vertically emplaced packages. Horizontally emplaced DHLW packages have an areal heat load of 8.34 W/m^2 . Vertical emplacement gives a lower areal thermal load, 4.84 W/m^2 , and therefore lower temperatures.

Figure 4-2 shows the predicted thermal performance of the DHLW waste package during the containment period. For DHLW there is little difference whether the gap is filled with crushed salt or air. For vertical emplacement, the temperature of the salt at the borehole-package interface peaks at about 52°C about 20 years after emplacement. The waste centerline temperature reaches a maximum of approximately 82°C 2 years after emplacement, assuming a 5-cm thick crushed salt gap persists for that time. If the gap closes by salt creep and reconsolidation, the temperature will be slightly lower (5°C). By the end of the containment period the temperatures have declined to about 32°C , which is only slightly above the preemplacement ambient temperature of 27°C . With its higher areal heat load, the horizontally emplaced DHLW package is predicted to have its temperature peak 9°C higher than that for vertical emplacement.

4.4.1.2 Intact Spent Fuel

The intact SFA will be emplaced at nominally 10 W/m^2 . One waste package will contain 4-PWR assemblies with an initial heat generation rate of 2,200 W or 9-BWR assemblies with an initial heat generation rate of 1,620 W. Heat generation rates assume fuel that is 10 years out of the reactor.

FIGURE 4-3 CONSOLIDATED SPENT FUEL (12 FWR) PACKAGE TEMPERATURES

NOTE: The curves represent the temperatures of horizontally and vertically employed packages.

The calculated thermal performance of the salt adjacent to a 4-PWR SFA package during the containment period is described in Appendix A. The peak temperature at the salt-container interface with vertical emplacement is approximately 90°C about 15 years after emplacement. The peak fuel rod temperature was not calculated by the model since the model does not account for heat transfer by convection. However, scoping calculations with a numerical correlation that accounts for radiation and convection showed that the temperature drop in the SFA package is much smaller than that of the CSF package, so the temperature of the SFA package will be bounded by the temperatures for the CSF package given below. The temperatures of the 4-PWR SFA packages are well below the limits of 250°C for salt and 375°C for the fuel cladding. After 1,000 years, the temperature of the waste container has declined to about 60°C.

4.4.1.3 Consolidated Spent Fuel

The CSF packages will be emplaced at a local areal heat load of nominally 10 W/m². The CSF waste packages with the rods from 12-PWR assemblies have an initial heat generation rate of 6,600 W, while those containing the fuel from 30-BWR assemblies generate 5,400 W. A thermal analysis of the lower-heat-generating BWR package was not performed as all temperatures are expected to be less than those of the PWR package.

Figure 4-3 shows the calculated thermal performance of the vertically emplaced 12-PWR CSF packages during the containment period. At the interface between the package and the salt, the peak salt temperature is approximately 134°C about 3 years after emplacement; this is well below the 250°C limit for salt. With no air gap, the temperature of the fuel cladding peaks at 341°C in the first year after emplacement. With a 5-cm gap filled with crushed salt, the cladding temperature peaks at 369°C in the first year after emplacement. The present limit on this temperature is 375°C, based on long-term creep of the fuel cladding. At the end of the containment period, the container temperature has declined to about 63°C while the cladding peak temperature is 85°C. Temperature limits for CSF at the end of the containment period have

not been established. Details of the thermal analyses are given in Appendix A.

The results of the thermal analyses show that for CSF and SFA the waste package temperatures did not significantly differ between horizontal emplacement and vertical emplacement. With horizontal emplacement, the packages are emplaced in both sides of the entry giving effectively a double row in the entry. In vertical emplacement, the packages are placed in a single row down the center of the entry. The spacing between entries is identical for both emplacement modes so the package pitch (the distance between packages) for vertical emplacement is one half the pitch for horizontal emplacement for a fixed areal thermal load. Areal thermal load is a dominant factor in determining the surface temperature of a given size package, but the reduced spacing with vertical emplacement causes the temperature of vertical packages to be slightly higher than for horizontal packages. The peak borehole surface temperature for horizontal emplacement with 12-PWR CSF is about 2°C lower than that for vertical emplacement. The peak borehole surface temperature with intact SFA is about 11°C lower (20 percent of the total temperature increase) in the vertical than in the horizontal case.

4.4.2 Brine Migration

After the waste package is emplaced, a thermal gradient will be established in the surrounding salt. Brine inclusions within the salt crystals will begin to move up the thermal gradient (because of solubility differences across the inclusion) toward the emplacement hole. This brine movement is in addition to brine transported to the borehole by pressure differential and diffusion mechanisms that may have begun before the waste container was emplaced. Some water vapor will evaporate from unconfined brine as it enters the borehole and will escape to the emplacement entry where it will condense and be unavailable for reactions with the waste package.

Thermal expansion and creep of the salt will cause the wall of the emplacement hole to close on the waste package, densifying the crushed salt in the gap. The densified salt will be reconsolidated in time, and the brine

evaporation and pressure-induced brine flow to the waste package will cease. As the heat output from the waste decreases by radioactive decay, the temperatures of the waste package and the surrounding rock will decline. The thermally induced transport of brine inclusions will continue until the temperature gradients in the salt are decreased to such a level that the mechanism ceases. Diffusion of brine by a concentration gradient through the rock will be a remaining mechanism to transport brine to the waste package, although this will be a smaller quantity than that from pressure-differential and thermal effects.

The San Andres Cycle 4 rock salt is about 88 volume percent halite, 4 volume percent anhydrite, and 8 volume percent mudstone with traces of dolomite and polyhalite (DOE, 1986b, p. 3-41). The best estimate of available water is 0.44 weight percent water as brine inclusions in the halite and 1.2 weight percent water associated with the clay minerals (DOE, 1986b, p. 3-105). Although there is no mechanism known to be operable to transport the water from the clay to the waste package, it is conservatively assumed that both sources of water (a total of 1.6 weight percent water or 4.1 volume percent brine) are available to be transported to the waste package.

A recognized mechanism for occluded brine transport, the migration of brine up the thermal gradient, has been mathematically modeled by Jenks (1979). The Jenks equation has been incorporated into a computer model, BRINEMIG, by ONWI (McCauley and Raines, 1987) for calculating the rate and total quantity of brine transported to the waste package as a function of temperature and temperature gradient.

The expected rate and quantity of thermally-transported brine have been calculated for emplaced DHLW, SFA, and CSF waste packages (ONWI, 1987). The calculations were performed with the assumption that brine migration does not stop until there is no temperature gradient in the rock. The quantity of brine calculated to migrate to the emplacement hole is given in Table 4-2.

Potentially greater quantities of brine may be available from pressure-gradient transport than by thermal transport. As indicated above, the San Andres Cycle 4 halite contains about 8 volume percent mudstone both in

Table 4-2. Calculated Available Quantities of Brine
From Thermally Induced Migration

Waste Form	Zero Gradient Cutoff	
	Time, yr	Total Brine, m ³
DHLW	1,000(a)	0.011
SFA (4-PWR)	1,000(a)	0.129
CSF (12-PWR)	1,000(a)	0.435

(a) Calculations were terminated at time - 1,000 years.

discreet interbeds and as chaotic mixtures with halite and anhydrite (DOE, 1986b, p. 3-104). The interbeds may or may not be continuous. If the interbeds are porous and interconnected, a pressure gradient across the interbeds could produce a significant flow of brine to the waste packages. However, as the voids within the excavated regions become filled by creep and consolidation of the salt, the pressure gradients are expected to diminish, so the transport of brine will cease. At this time this is recognized as a potential source of a greater volume of brine than from thermal transport of occluded brine. However, the amount of brine available and the time over which the transport occurs cannot be determined until site-specific data are available.

4.4.3 Corrosion

The corrosion performance was evaluated with the assumption that there would be a sufficient quantity of brine to support the corrosion process. This conservative assumption was taken because of the uncertainty about the quantity of brine available to the waste packages. Experimental data available to date indicate that general corrosion is the principal corrosion process acting on low-carbon steel during the containment period. No pitting, crevice corrosion, stress corrosion cracking, or hydrogen attack have been observed in experiments simulating expected repository conditions. Most of

100

100

100

100

100

The results of these assessments will be compared to the results of the studies in progress of the borehole closure rate that may occur from thermal expansion of the salt adjacent to the waste package. It is anticipated that a properly sized borehole gap will accommodate the expanded salt, so the maximum container stress will occur from lithostatic pressure. Maximum pressure will occur when inward creep of salt densifies the packing in the annulus, and the full lithostatic pressure is applied to the container. Appendix C presents more details of the structural design analysis.

4.4.4.1 Dimensions for Defense High-Level Waste Containers

The cavity diameter of the container for the DHLW form is 63.5 cm. A cylindrical wall thickness of 9.6 cm and a flat head thickness of 15.7 cm are required to sustain the 18.0-MPa lithostatic design pressure. The estimated depth of general corrosion is 1.1 cm during the containment period. This leads to a design thickness of 10.7 cm for the cylinder wall and 16.8 cm for the heads. At the time of emplacement, the cylinder can support approximately 20.8 MPa and the heads 21.8 MPa. As corrosion progresses, these values are predicted to slowly decline to 18.0 MPa at the end of the containment period. As discussed in Section 4.4.3, the more conservative estimated corrosion thicknesses are used in this design.

4.4.4.2 Dimensions for Intact Spent Fuel Containers

The containers for intact SFA have a cavity with a larger diameter, 69.2 cm, than that of the other types of waste and thus, require a thicker wall to sustain the lithostatic pressure. For the SFA containers, the cylinder wall is 10.5 cm thick and the heads are 17.5 cm thick for structural purposes. With the estimated 2.3 cm for a corrosion allowance, the total cylindrical wall thickness is 12.8 cm and the thickness of the heads is 19.8 cm. The initial pressure capacity of the cylinder wall, with the corrosion allowance intact, is 22.6 MPa and 25.8 MPa for the heads. The capacity of the wall is predicted to decline as corrosion proceeds and is 18.0 MPa for both side wall and heads 1,000 years after closure.

4.4.4.3 Dimensions for Consolidated Spent Fuel Containers

The CSF containers have a cavity diameter of 64.5 cm. To withstand the 18.0-MPa lithostatic pressure, the side wall is 9.6 cm thick and the heads are 16.0 cm thick. With a corrosion allowance of 2.3 cm, the total wall thickness is 11.9 cm and the head thickness is 18.3 cm. At the time of emplacement the wall can support 22.5 MPa and the heads 25.2 MPa. After loss of 2.3 cm of thickness by corrosion for 1,000 years, both wall and heads can sustain the pressure of 18.0 MPa.

4.5 WASTE PACKAGE FUNCTIONAL REQUIREMENT NUMBER 5

The waste package shall contribute to controlling the release of radionuclides after the containment period.

The component requirements for this functional requirement place responsibilities on the waste form, the container, and the packing to meet the release requirement. Within the present strategy, the waste form is to be a solid with a low rate of matrix dissolution and low solubility of its components in the expected brines. The container, by its reactions with the brine, is to cause a chemical condition of the brine, which limits the solubility of the radionuclides. The packing is to impede the transport of radionuclides from the waste package. These roles are discussed in the following subsections.

4.5.1 Waste Form Dissolution

The waste form is a barrier to the release of radionuclides during the postcontainment (isolation) period. The waste forms, solidified high-level waste and spent fuel, physically and chemically differ from each other and they each experience somewhat different temperature transients, so their radionuclide release characteristics can be expected to be different.

4.5.1.1 Defense High-Level Waste

DHLW has the lowest decay heat output and, consequently, the lowest temperature of the waste forms analyzed. At the end of the containment period, the peak temperature is predicted to be 32°C. Tests conducted and in progress at the Savannah River Laboratory indicate that the performance of DHLW may meet the proposed Nuclear Regulatory Commission (NRC) release rate criterion at these low temperatures (Baxter, 1981). More recent data (McGrail, 1986) indicates that the rate of matrix dissolution of DHLW in brine approaches a level to meet the release criterion of 10^{-5} per year depending upon the effective surface area of the glass and the chemistry of the brine. As the glass matrix dissolves, individual radionuclides will be exposed to the fluid and will be available to go into solution depending on their solubility. Cesium is highly soluble and will readily go into solution as it is exposed by the dissolving matrix. Other elements such as technetium and the transuranics are expected to be highly insoluble under anoxic conditions, and only a small fraction will dissolve. If brine transport mechanisms (see Section 4.4.2) limit the availability of brine, the corrosion of the container may consume the brine, and dissolution of the glass waste form may not proceed. Thus, from several aspects, DHLW is expected to meet the requirement to contribute to controlling the release of radionuclides.

4.5.1.2 Spent Fuel

The release of radionuclides from spent fuel to the brine that contacts it has not been as thoroughly studied as that from glass waste and cannot be well described at this time. The release from spent fuel is expected to occur by three sequential mechanisms (Johnson et al., 1985). First, after the container fails and the fuel cladding is penetrated there can be an almost immediate release of some fission products, notably cesium and iodine, which have migrated from the fuel to the surfaces of the fuel and cladding during the period the fuel was in the reactor. This has been called the release of the gap inventory. Next, as brine contacts the fragments of spent fuel it can preferentially dissolve fission products that have migrated to the grain boundaries of the fuel during operation in the reactor. During fuel

dissolution experiments, it is usually not apparent when the gap inventory dissolution is complete and grain boundary leaching dominates the process. Because these two mechanisms occur so readily relative to the last mechanism, they are sometimes combined as one process for modeling and may account for the release of up to 10 percent of some fission products.

Following gap and grain boundary leaching, the release from spent fuel is believed to be controlled by the dissolution of the uranium oxide matrix with the associated exposure of the radionuclides to the brine. As the matrix dissolves and the radionuclides are made available to the fluid, their release from the package will be dependent upon their solubility in the brine and on movement of the brine. Solubility of individual radionuclides is a function of their chemical species and will be influenced by the chemistry of the brine, which has been altered by reactions with the container and the spent fuel. Although the predicted temperature of spent fuel is much higher than DHLW after the containment period, spent fuel dissolution is not highly sensitive to temperature (McVay et al., 1981).

It is expected that spent fuel will contribute to controlling the release from the waste package, but the contribution cannot be well defined. Most certainly, other components of the waste package must perform in conjunction with spent fuel to limit the gap release from the package to acceptable levels. The rate of release by matrix dissolution is expected to be significantly lower, but the results of research in progress are needed to define its role in the waste package release function.

4.5.2 Container Corrosion Effects

As discussed in Section 4.4.2, brine is expected to begin to migrate toward the waste package upon emplacement. Prior to closure, air will also be available so the corrosion of the low-carbon steel container can proceed under oxic brine conditions. After closure when air is no longer available, the corrosion process will consume the available oxygen and the brine is expected to become anoxic. The amount of brine available to the waste package is uncertain. If brine migrates primarily by thermal and pressure gradient

mechanisms, brine migration may cease as the repository cools and as voids in the packing and backfill are filled by creep and consolidation of the salt. If other migration mechanisms prevail, brine may be available after the containment period to interact with the waste form. In the latter case, the container corrosion reaction is expected to keep the brine anoxic, which will limit the solubility of most radionuclides (Mendel, 1984, p. 4.1). Thus, as long as low-carbon steel is present, the container can play a role in limiting the release of radionuclides from the waste package, even after container failure.

4.5.3 Packing Performance

During the course of the conceptual design study, the role of the packing has changed from a secondary function for heat transfer and radiation shielding to a primary function of limiting release of radionuclides. At this time, crushed salt, the packing concept for vertical emplacement, has not been evaluated for this new functional requirement, and no response to the requirement can be stated.

The concept of crushed salt as a packing material for vertically emplaced packages resulted from allowing crushed salt to fill the annulus as it is poured into the emplacement hole as a shielding material. The absence of crushed salt in the annulus of a horizontally emplaced container was because no simple system was available to emplace and retain the crushed salt. Since these concepts were adequate for the shielding and heat transfer functions no changes were necessary. With the development of the new functional requirement for the packing it has become necessary to develop a base of information to specify and evaluate materials and geometry for the packing. Experimental studies are currently in progress to gather data on potential packing materials including salt. The properties of primary interest are impedance to fluid flow, absorption (for retardation) of radionuclides, and effect on the local chemistry environment (pH, oxidation state, etc.) in addition to good heat transfer and shielding properties. The design definition and material specification for the packing may change in subsequent design studies.

4.6 QUALITY ASSURANCE

A quality assurance (QA) program has been developed by Battelle Project Management Division to meet the basic and supplemental requirements of the ANSI/ASME NQA-1 (ANSI, 1986), latest edition and addenda, "Quality Assurance Program Requirements for Nuclear Facilities" as interpreted by the Salt Repository Project Office of the U.S. Department of Energy (DOE) and by the Office of Nuclear Waste Isolation (ONWI) to apply to a nuclear waste repository. This QA program is intended to comply with DOE Order 5700.6A, "Quality Assurance"; with the requirements of the U.S. Code of Federal Regulations, Title 10, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants"; and with the Nuclear Regulatory Review Plan: "Quality Assurance Programs for Site Characterization of High-Level Nuclear Waste Repositories" and other QA requirements as directed by the DOE. All applicable QA procedures developed as a part of that QA program have been applied by ONWI and its contractors in the course of this conceptual design study.

4.7 PERFORMANCE SUMMARY

From the results of the analyses performed on the waste package designs there is reasonable confidence that the conceptual waste package designs will satisfy the design requirements, providing the assumptions used in the analyses are supported by research in progress.

5.0 DISCUSSION

This report discusses an analysis of a waste package concept for vertical or horizontal emplacement in a salt repository. The starting point is a waste package concept for vertical emplacement developed earlier and described by Westinghouse (1986). Subsequent work considered the effect of several program changes on the waste package concept. The changes are

1. The repository design effort has been broadened to study both horizontal and vertical emplacement of the waste packages.
2. The repository layout has been changed, based on gassy mine regulations and thermal and rock mechanics requirements.
3. A lower areal heat load for spent fuel has been established to reduce repository room temperatures.
4. A repository depth has been established, based on the Deaf Smith County site, which results in a higher in situ stress.
5. A two-phase repository concept has been introduced. This concept includes disposal of intact spent fuel during Phase 1. In Phase 2 most spent fuel will be consolidated prior to disposal.
6. New salt property data have been selected that are believed appropriate for the Deaf Smith County site.
7. Only the West Valley commercial high-level waste with a low heat output is expected in the near future, so commercial high-level waste glass is not considered separately from defense high-level waste in this study.
8. Additional corrosion data have been collected for the container material, low-carbon steel.

This report describes waste package design concepts for two waste forms to be emplaced in a repository in salt. The waste forms are defense high-level waste and spent fuel (intact spent fuel assemblies or consolidated spent fuel). The accommodations of West Valley high-level waste, reactor consolidated spent fuel, spent fuel hardware, and nonstandard wastes are also recognized.

The waste packages are generically identical, regardless of the waste form and emplacement mode. The containers are right circular cylinders with flat heads. The sizes of the waste forms determine the sizes of the container cavities. The thickness of the container walls is the sum of the thickness required to withstand the crushing force of the lithostatic pressure and the thickness predicted to be consumed by general corrosion through the life of the container. All container components are designed to be fabricated of cast low-carbon steel.

From the results of the thermal, corrosion, structural, and radiation shielding analyses performed on the waste package designs, it is concluded that the waste package concepts are adequate for developing the site characterization plan and can serve as a basis for future design studies. Functional Requirements 4 and 5 (see Sections 2.1.4 and 2.1.5) for waste containment and controlled release of radionuclides have been the major influence in the design studies. A large uncertainty in analyzing the performance of the waste package is associated with the quantity and chemistry of the brine that will be available to interact with the waste package components. These brine parameters will have a large effect on the corrosion of the container and its ability to meet the containment requirement. The brine parameters will also control the dissolution of the waste form components and determine the ability of the waste package to meet the controlled release requirement. With information on these parameters from site characterization and the data from research in progress, it is believed that subsequent studies can result in waste package designs that will meet all of the Functional Requirements for the waste package.

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FEDERAL REGULATIONS AND ORDERS

DOE Order 5700.6A, Quality Assurance.

10 CFR Part 20, Standards for Protection Against Radiation.

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories; Technical Criteria.

10 CFR Part 960, Nuclear Waste Policy Act of 1982, Guidelines for the Recommendation of Sites for the Nuclear Waste Repositories (Final Siting Guidelines) (42 USC Secs. 10101-10226).

40 CFR Part 191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level, and Transuranic Radioactive Wastes; Final Rule, September 19, 1985.

APPENDIX A

THERMAL ANALYSIS

APPENDIX A - THERMAL ANALYSIS

A.1 INTRODUCTION

This appendix describes the thermal analysis methods used to predict repository and waste package temperatures for defense high-level waste (DHLW), intact spent fuel assemblies (SFA), and consolidated spent fuel (CSF) in a repository in salt. Thermal performance is calculated for both vertical and horizontal waste package emplacement. Consolidated spent fuel and intact spent fuel are emplaced at an areal thermal load of 9.88 W/m^2 (40 kW/acre) for both vertical and horizontal emplacement. The thermal load density for DHLW is 8.43 W/m^2 (34.1 kW/acre) for horizontal and 4.84 W/m^2 (19.6 kW/acre) for vertical emplacement.

The calculation is decoupled at the surface of the waste container. The repository is modeled with the waste package as a heat producing element. This calculation gives the midplane temperature at the waste package surface which is then used as a boundary condition for the waste package calculation. The details of the package thermal performance are calculated separately. This approach is conservative since the maximum surface temperature is used as the boundary condition over the entire package. Separating the heat transfer modeling into independent elements is described in ONWI-41 (SAIC, 1979).

The repository is modeled in three dimensions using HEATING5. This program uses the finite difference solution method to solve heat conduction problems. It calculates temperatures throughout the repository as a function of time. The model allows temperature-dependent material properties. Six cases are modeled as follows:

1. DHLW vertically emplaced.
2. SFA vertically emplaced.
3. CSF vertically emplaced.
4. DHLW horizontally emplaced.
5. SFA horizontally emplaced.
6. CSF horizontally emplaced.

The waste package model is one-dimensional for DHLW and two-dimensional for CSF and SFA. A two-dimensional representation of the CSF waste package is needed due to the presence of heat transfer fins in the canister. The DHLW and CSF calculations are done with DOT (INTERA, 1983), a two-dimensional finite element code. The SFA calculations used the Wootton-Epstein correlation to model mixed radiation and convection in a fuel assembly (Bucholz, 1983). Thermal performance of the waste package is independent of its orientation so one thermal analysis for each package was used for horizontal and vertical emplacement.

An estimate of the temperature change across the borehole gap is also made. This gap between the package and the borehole wall is expected to close very quickly. However, it will cause a significant temperature drop shortly after emplacement. Therefore, a one-dimensional calculation was done to estimate borehole temperature drop.

A.2 REPOSITORY THERMAL MODEL

The thermal performance of the waste repository is calculated using a three-dimensional heat conduction finite difference code, HEATING5 (Fowler and Volk, 1959). The same basic model is used for all six repository arrangements. A minimum of 1300 nodes and a maximum of 3400 nodes were used in developing the finite difference mesh for HEATING5. More than one mesh was set up and run for several problems to eliminate errors due to mesh sensitivity. In all cases, the waste package temperature varied by less than one percent with the range of mesh spacings used. The general form of the model is shown in Figure A-1. As shown in Figure A-1, mesh is finer near the waste package. This is the area of the steepest thermal gradient.

To simplify modeling, adiabatic boundaries are taken on all sides. Above and below, the boundaries are placed about 650 m from the edges of the salt bed. These levels are chosen to minimize the effect of the boundaries during the 1,000 year transient calculation. Adiabatic boundaries around the waste packages represent a physical situation where the modeled package is surrounded by other packages of identical geometry, power, and properties. In

Figure A-1. Representative Mesh for Repository Thermal Analysis

horizontal emplacement it assumes that packages are aligned with each other lengthwise. The use of adiabatic boundaries is conservative because it assumes no heat loss from the system to the surroundings. The adiabatic boundaries of the unit cells for the vertical and horizontal cases are shown in Figures A-2 and A-3, respectively.

The repository thermal modeling is based on previous work described in Claiborne (1980) and Westinghouse (1986). The repository model has a simplified geometry. A generic rock designated as "non salt" is placed above and below the repository horizon. The repository horizon is modeled as uniform salt with crushed salt backfill in the emplacement drift. A 30 percent void volume is assumed for crushed salt so its density is 70 percent of solid salt. The crushed salt in the drift does not reconsolidate in the model whereas in reality it would. Since the thermal conductivity of crushed salt is low, 10 percent of the value for solid salt, this simplification is conservative.

No detail of the borehole backfill in vertical emplacement or of the shield plug in horizontal emplacement is modeled. This simplification has minimal effect. The waste package is modeled as a heat generating region of salt. No air gap is modeled between the package and the salt. This is a reasonable assumption because the high salt creep rate will close the gap quickly. The heat capacity of the package and borehole air gap is low and their response time is fast in comparison to the repository so the package can be treated as a "black box" heat source in repository calculations to find the maximum salt temperature. This temperature is then used as a boundary condition on the air gap or container surface to determine the waste form temperature. The repository stratigraphy is shown in Figure A-4 and the material properties are shown in Table A-1. Justification for this approach is given in Section 3 of ONWI-41 (SAIC, 1979).

The waste package is modeled in a rectangular form rather than the actual cylindrical form. By making the perimeter of the square package model equal to the circumference of the actual package, the package side surface area along the waste form heated length is modeled exactly, but the end surfaces are smaller than the actual size. The net effect is that the heat flux at the salt boundary is, conservatively, slightly over predicted. The package geometry as modeled is described in Table A-2.

Unit Cell for Repository
Scale Thermal Calculations
with Vertical Emplacement

Figure A-2. Unit Cell for Repository Scale Thermal Calculations With Vertical Emplacement

Unit Cell for Consolidated
Spent Fuel Repository Thermal
Model for Horizontal Emplacement

Figure A-3. Unit Cell for Repository Scale Thermal Calculations With Horizontal Emplacement

Figure A-4. Stratigraphy for the Repository Thermal Model

Table A-1. Repository Model Material Properties

	Salt	Crushed Salt	Non Salt
Density kg/m ³	2200	1540	2525
Specific Heat J/kg K	879	879	954
Thermal Conductivity W/m K	4.7	0.47	(1)

(1) $k = 3.22 - 9.33E-3 T + 1.16E-5 T^2$

Table A-2. Waste Package Geometry

	Length (m)	Width (m)	Height (m)	Heat Load (W)	Volumetric Heat Generation Rate (W/m ³)
DHLW	3.38	0.632	0.632	470	348
SFA	4.5	0.664	0.664	2,200	1,109
CSF	4.5	0.664	0.664	6,600	3,326

The initial temperature of 27°C was set at all nodes. The effect of geothermal gradient is small and was therefore neglected.

The maximum salt temperature as a function of time is shown in Figure A-5 for vertical emplacement and Figure A-6 for horizontal emplacement.

A.3 WASTE PACKAGE THERMAL MODELS

Waste package internal thermal analysis is decoupled from repository analysis. The salt surface temperature calculated in the repository model (or at early times, the salt surface temperature plus borehole air gap temperature change) is applied as a boundary condition at the waste package surface. To allow the modeling to be done in parallel, the waste package thermal performance is calculated as a function of time for various surface temperatures.

The heat load in the waste package decreases from the initial value as the radionuclides decay. The normalized power as a function of time after emplacement is shown in Table A-3. Because of the rapid response of the waste package and slow rate of change in power level, a pseudo steady state approach is used. The package temperature is not calculated as a function of time from emplacement to 1,000 years. Instead the temperature change through the package is found at selected times as though the package is at steady state with the appropriate power level for the time.

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Table A-3. Normalized Power Characteristics

Time After Emplacement (Years)	DHLW	Spent Fuel
0.0	1.0	1.0
1.0	--	0.966
2.0	--	0.933
5.0	0.845	0.841
10.0	0.766	0.750
15.0	0.681	0.685
20.0	0.606	0.625
30.0	0.485	0.524
50.0	0.315	0.389
70.0	0.203	0.303
100.0	0.115	0.240
300.0	0.010	0.100
500.0	0.004	0.070
800.0	0.002	0.050
1000.0	0.002	0.045

The DHLW package is modeled as an infinite cylinder using a finite element computer code called DOT. This ignores heat loss from the ends and is therefore conservative. It also assumes that the canister is centered in the container. This is reasonable for vertical emplacement. With horizontal emplacement the canister will be off center in the container with no gap at the bottom. Since most of the heat transfer across the gap is by radiation, the larger gap at the upper surface will not significantly reduce heat flow. Contact at the lower side will allow conduction and will improve heat flow so the one-dimensional calculation is conservative for the horizontal case.

The one-dimensional DHLW model is depicted in general form in Figure A-7. The glass waste form is modeled as a uniform heat source in a conducting solid with a thermal conductivity of 1.1 W/m°C. The small temperature drop across the waste canister wall is neglected. Heat transfer across the gap between the canister and the container will occur by a combination of conduction, convection, and radiation. For radiation calculations, the view factor assumes a concentric cylinder configuration with the emissivity of each radiating

FIGURE AE-7 DEFENSE HIGH LEVEL
WASTE PACKAGE THERMAL
MODEL.

DEFENSE HIGH LEVEL WASTE PACKAGE
THERMAL MODEL

LAS/JMR/5-30-86

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surface as 0.6. The convection/conduction heat transfer is calculated using the equivalent thermal conductivity approach described in Eckert and Drake (1959). It considers two concentric cylinders separated by air. Heat transfer from one surface to the other occurs by conduction through the gas and by local convection cells. The gap is sealed at the ends so there is no net gas flow. The carbon steel container wall is modeled as a conducting solid with a thermal conductivity of 45 W/m°C.

Figure A-8 shows the dimensions used in the DHLW calculations. In order to allow these calculations to be done in parallel with the container design, a container wall thickness was estimated. The container wall thickness used in the thermal model does not exactly match the final container design. The temperature change through the container wall is less than 1°C and the waste form temperature for DHLW is much lower than the design objective of 500°C. Therefore, small differences in wall thickness will have no significant effect on the result. The temperature drop from the waste form peak temperature to the container surface for DHLW with various surface temperatures is shown as a function of time in Figure A-9.

The two-dimensional finite-element heat transfer code DOT (INTERA, 1983) is used to model the CSF waste package thermal performance. The waste package which was modeled contains 12 PWR elements consolidated into a canister. The canister is provided with 6 internal heat transfer fins. Each fin is 4 mm thick. The canistered waste form is placed into a thick-walled carbon steel container. A two-dimensional model is needed to account for the internal heat transfer fins. As with the DHLW model end effects are neglected and the canister is assumed to be centered in the container.

A typical mesh arrangement for the CSF model is shown in Figure A-10. Calculations with 72 and 108 mesh points gave identical results. The 108 node mesh is used for all calculations. The carbon steel canister and container wall are modeled as a conducting solid with a thermal conductivity of 45 W/m°C. Radiation, conduction, and convection in the air gap are all modeled by equivalent conduction since DOT does not allow internal radiating

Dimensions for the Defense
High Level Waste Package
Thermal Model

Figure A-8

Temperature Drop in a Defense High
Level Waste Package with Various
Surface Temperatures

Figure A-9

surfaces. Calculation of radiation as an equivalent conduction is described by Fields and Fletcher (1981). The emissivity of the canister and container were taken as 0.6. The thermal modeling was done at the same time as the structural and corrosion allowance calculations. As a result, the container thickness used in the thermal model does not exactly match the actual design. The temperature drop in the container is about 2°C so changes of a few centimeters in wall thickness will have no effect on the final result.

Within a bundle of close-packed fuel rods, heat transfer occurs by a combination of radiation, convection, and conduction with the dominant mode expected to be radiation. In the finite element model, thermal radiation is represented by a conduction model using an effective conductivity in the fuel rod region. The effective thermal conductivity is based on a correlation developed by Cox (1977) which applies to a circular bundle of heat-generating rods. For a triangular pitch rod arrangement and considering only radiation heat transfer, the correlation relates peak bundle temperature to canister temperature, heat load, number of rods, canister diameter, rod diameter and pitch, and rod emissivity. To apply the correlation in the present study involving CSF, the rod-pitch-to-diameter ratio was set to 1.0 and the fuel rod emissivity to 0.8. The effective thermal conductivity is shown in Table A-4. More detail on the derivation of the rod bundle equivalent conductivity is given in Westinghouse (1986).

Table A-4. Effective Thermal Conductivity
of Consolidated Spent Fuel

T °C	K W/m°K
50	0.0640
100	0.0985
150	0.144
200	0.201
250	0.272
300	0.357
400	0.579
500	0.877
600	1.26

The temperature profile along a line midway between two of the six fins is shown in Figure A-11. The effect of changing fin thickness is illustrated in Figure A-12. Figure A-13 shows the temperature change from the peak clad temperature to the container surface for a 12 element PWR consolidated spent fuel waste package.

The SFA was modelled as an infinitely long square array of four 17 x 17 PWR fuel elements in a thin-walled cylindrical canister centered in a thick-walled cylindrical container. The temperature drop in the canister is estimated using the Wooten-Epstein Equation as described by Bucholz (1983). This method accounts for radiation and convection in an intact assembly of fuel pins. The temperature drop across the air gap between the canister and the container is calculated assuming combined radiant and conductive heat transfer. Heat transfer through the container wall is by conduction. The dimensions used in the SFA canister model are shown in Figure A-14. The temperature drop from the peak clad temperature to the container outside wall was calculated at time zero for two container surface temperatures. The SFA and comparable CSF temperature drop is shown in Table A-5. The SFA has a much lower temperature drop than the CSF so no further calculations were done for SFA.

In all cases the waste was modeled as a uniformly distributed heat source. This is clearly acceptable for DHLW but ignores the axial power distribution present in spent fuel. However, conduction in the cladding (and fins in CSF) causes the effective axial conductivity to be higher than the effective radial conductivity. Conduction along the fuel element will smear the power output so that the mid-height temperatures are not significantly altered (McCann, 1980).

A.4 BOREHOLE GAP

For horizontally emplaced waste packages the gap between the container and the wall of the borehole is assumed to be filled with air initially. For vertically emplaced packages, the borehole above the package will be filled

Temperature Profile Between Fins
in a Consolidated Spent
Waste Package

Figure A-11

Effect of Thickness of Internal
Fins on Temperature Drop in a
Consolidated Spent Fuel
Waste Package

Figure A-12

Dimensions for the Intact
Spent Fuel Waste Package
Thermal Model

Figure A-14

Table A-5. Comparison of Temperature Drop in Intact Spent Fuel and Consolidated Spent Fuel Waste Packages

Surface Temperature °C	SFA ΔT °C	CSF ΔT °C
110	102	227
190	85.7	183

with crushed salt which may trickle down and fill the gap. Since the crushed salt-filled gap results in higher temperatures it is considered for vertically emplaced waste packages.

The borehole gap is not included in the models. However, the temperature drop for various width gaps was estimated with crushed salt backfill and with an air gap. Crushed salt is treated as a conducting annular solid with a thermal conductivity of 0.47 W/m°C. Radiation combined with an equivalent conductivity for convection and conduction is used to model the air gap. The results for DHLW are shown in Table A-6 with crushed salt and Table A-7 for air. Temperature drops for CSF with a crushed salt filled gap and air gap are shown in Tables A-8 and A-9, respectively. The crushed salt temperature drop increases with thickness because it is a conduction resistance that increases with thickness. Radiation is independent of distance so the air gap temperature drop is essentially independent of gap size except for small corrections for convection effects.

A.5 WASTE PACKAGE THERMAL PERFORMANCE

Repository temperatures can be combined with the borehole and waste package temperature changes to estimate waste package thermal performance. Calculating the temperature drop across each resistance separately and then using linear combinations is based on the assumption that the waste package acts as a constant heat source. Earlier work has shown this approach to be valid (SAIC, 1979).

Table A-6. Borehole Gap Temperature Drop for
Defense High-Level Waste With
Crushed Salt Backfill

Gap Size (m)	Surface Temperature Values	
	50°C	90°C
0.01	1.2	1.2
0.05	5.5	5.5
0.10	10.4	10.4
0.20	18.9	18.9

Table A-7. Borehole Gap Temperature Drop for
Defense High-Level Waste With
an Air Gap

Gap Size (m)	Surface Temperature Values		
	50°C	70°C	90°C
0.01	8.2	7.0	6.3
0.05	9.8	8.6	7.6
0.10	9.6	8.6	7.6
0.20	9.8	8.6	7.6

Table A-8. Borehole Gap Temperature Drop for Consolidated Spent Fuel With Crushed Salt Backfill

Gap Size (m)	Surface Temperature Values	
	120°C	160°C
0.01	11.6	11.6
0.02	23.0	23.0
0.05	55.6	55.6
0.10	105.5	105.5

Table A-9. Borehole Gap Temperature Drop for Consolidated Spent Fuel With an Air Gap

Gap Size (m)	Surface Temperature Values		
	120°C	140°C	160°C
0.01	46.5	41.8	37.9
0.05	50.5	45.6	41.2
0.20	50.8	46.0	41.5

With either vertical or horizontal emplacement, the borehole gap is expected to close within a few years due to salt creep (Nelson and Fossum, 1985). During all but the first few years after emplacement, the salt will be in contact with the container's outside surface. To calculate the maximum waste form temperature with no gap, the repository temperature at the waste package/salt interface is applied as the waste package surface temperature. The temperature change in the waste package is found for this surface temperature in Figure A-9 for DHLW or Figure A-13 for CSF - 12 PWR by interpolation. The repository temperature is added to the waste package temperature change to give the maximum waste form temperature.

There is potential for a significant temperature change across the gap between the salt surface and the container surface. The effect of this gap on waste form temperature was estimated. For horizontal emplacement, the gap is taken to be an air gap. The bulk of the heat transport is by radiation so the gap size has little effect on the temperature change. With vertical emplacement, the gap was assumed to be filled with crushed salt. Since heat transport is by conduction in a crushed salt filled gap, the temperature change is dependent on gap width. The final requirement for borehole clearance has not been set. Two gap sizes were used for each waste form, 5 cm and 20 cm for DHLW and 2 cm and 5 cm for CSF - 12 PWR. Larger gaps are allowed with DHLW due to the much lower heat output per package.

The calculation of peak waste form temperature with the borehole gap is identical to the calculation without the gap, except the gap temperature change, Tables A-6 through A-9, is added to the salt temperature. The repository salt temperature plus the borehole gap temperature change is then applied as the waste package surface temperature. The gap temperature change is corrected for power reduction as a function of time but does not account for gap closure. Calculations with the gap were done only for the first 20 years. The peak waste form temperature is clearly past by this time and, in any case, the gap should close in 20 years.

The DHLW package is well within thermal performance limits. The temperature in DHLW is low because the heat output of this package is low, 470 watts per package. This dilute heat source gives low temperature drops. There is

also a secondary effect from the physical constraint on minimum package spacing. DHLW packages cannot be emplaced close enough to reach the design heat load of 9.88 W/m^2 (40 kW/acre). The salt temperature and the maximum waste form temperature for DHLW with and without a gap are shown in Figure A-14 for vertical emplacement and Figure A-15 for horizontal emplacement.

The CSF waste form with 12 PWR elements and a power output of 6600 watts is in a package that is essentially the same size as the DHLW package. This higher load per package obviously results in a higher surface temperature and higher temperature changes in the waste package and borehole gap. The higher heat load per package gives a larger package separation for a given areal heat load. Spacing of the packages is not limited by the physical ability to prepare emplacement locations so the CSF packages are emplaced at the design areal heat load of 9.88 W/m^2 (40 kW/acre). The salt temperature and maximum waste form temperature for CSF with and without a gap are shown in Figure A-16 for vertical emplacement and A-17 for horizontal emplacement.

The 12-element PWR CSF package is able to maintain the maximum clad temperature below the limit of 375°C . However, if a crushed-salt-filled borehole gap greater than 5 cm on the radius is used, the clad temperature limit would be exceeded. For vertical emplacement the borehole gap should be 5 cm or less prior to backfilling assuming crushed salt or some other low conductivity backfill is used.

See Insert on p 111.

In no case does either waste form in either emplacement mode cause the salt temperature to rise above 250°C . Salt temperature must be kept below 250°C to avoid decrepitation.

The salt temperature and waste package temperature drop for SFA are both significantly below those for CSF. No calculation of SFA thermal performance was done because CSF is a bounding case.

A reactor consolidated fuel waste may be received for disposal. The details of this waste form have not been specified but essentially it is the

Salt and Waste Form Temperature
for Defense High Level Waste -
Horizontal Emplacement

Figure A-16

Salt and Waste Form Temperature
for Consolidated Spent Fuel -
12 PWR - Vertical Emplacement

Figure A-17

Salt and Waste Form Temperature
for Consolidated Spent Fuel -
12 PWR - Horizontal Emplacement

Figure A-18

Figure A-20 Plot of Radial Temperature Profile From the Package Center to the Host Rock for a 12-PWR CSF Package

Figure A-19 Plot of Radial Temperature Profile From the Package Center to the Host Rock for a DHLW Package

pins from two elements packed into the envelope normally occupied by one element. Four such boxes would be placed into the SFA waste package in the place of four intact fuel elements. This waste form is similar to the consolidated spent fuel form with the boxes acting as the pie-shaped segments of the CSF canister. The boxes provide the same number of heat transfer fins per elements as the CSF canister so the temperature drop in the two waste forms should be the same. The major difference is that the boxes are not attached to the canister. In the CSF canister the fins conduct to the canister wall which then radiates heat across the gap between the canister and the container. With reactor consolidated boxes the box must radiate to the canister which then radiates heat across the gap. Lack of intimate box to canister contact adds another radiative resistance so the gap resistance would effectively double. However, the heat load in the reactor consolidated waste package is 8/12 of the CSF waste package and the SFA canister area is about 1.2 times the CSF canister area. Thus, the overall effect would be about a 10 percent increase in air gap temperature drop or roughly a 5°C increase in waste package temperature drop. The salt temperature will decrease by more than 5°C due to the 8/12 power reduction so the reactor consolidated fuel will have lower cladding temperatures than the CSF waste form.

Insert for p 107.

Plots of the radial temperature profiles for HLW and CSF packages for 1, 10, 100, and 1000 years after emplacement are shown in Figures A-19 and A-20. These assume the gap between the container and host rock is closed.

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APPENDIX B

BRINE MIGRATION ESTIMATES FOR HORIZONTAL EMPLACEMENT

APPENDIX B

BRINE MIGRATION ESTIMATES FOR HORIZONTAL EMPLACEMENT

B.1 INTRODUCTION

This appendix provides estimates of brine migration rates and cumulative brine flows toward waste packages emplaced horizontally in a salt repository. The three types of waste are defense high-level waste (DHLW), intact spent fuel (SFA), and consolidated spent fuel (CSF). The model for brine migration assumes the migration is a function of temperature and temperature gradient so that temperature estimates are also provided. Similar calculations were performed for vertically emplaced CSF packages by McCauley and Raines (1986).

B.2 DATA AND COMPUTER CODES

The calculations were performed with the TEMP (Wurm et al., 1985) and BRINEMIG) (McCauley and Raines, 1986) codes. TEMP was used to calculate the temperature distribution in the direction vertical to the plane of emplacement at the center of the waste package. This is the direction having the steepest temperature gradient and prior calculations (McCauley and Raines, 1986) indicated that temperature gradient has a dominating effect on brine migration. BRINEMIG was then used to calculate the brine migration due to this temperature profile.

The input data required for these calculations are given in Tables B-1 and B-2. The decay characteristics of the waste that were used in the calculations were built into the TEMP code and are slightly different than those used for the thermal calculations in Appendix B. At early times, the estimated temperatures using the TEMP heat rates should be no more than 3 percent less than the temperatures estimated with the Appendix B heat rates and be conservative overestimates at later times.

Table B-1. Salt Properties

Density, kg/m ³	2,200
Thermal conductivity, W/m°C	4.7
Heat capacity, J/kg°C	879
Initial temperature, °C	27
Brine content, Vol%	4.1

Table B-2. Waste Package and Repository Description

	DHLW	SFA	CSF
Container:			
Outside diameter, m	0.808(a)	0.845(a)	0.845(a)
Outside length, m	3.385(a)	4.500(a)	4.465(a)
Initial heat load, W	470	2,200	6,600
Repository:			
Drift pitch, m	36.6	51.8	51.8
Emplacement hole pitch, m	3.0	8.6	25.9
Drift height, m	4.6	4.6	4.6
Drift width, m	7.3	7.3	7.3
Container inset, m	2.0	2.0	2.0

- (a) The dimensions are preliminary values and are slightly less than those given in Chapter 2. This causes the calculated temperatures to be higher giving somewhat larger quantities of brine than if the final dimensions were used. The error, although conservatively higher, is not believed to be significant.

B.3 RESULTS

The principal results of this study are the temperatures at the waste package-salt interface at mid length, the brine migration rates at this location, and the cumulative brine flow.

B.3.1 Temperatures

The temperatures at the waste package-salt interface at mid length (borehole surface temperatures) are shown in Table B-3 for the three types of waste. Figure B-1 is a plot of the data in Table B-3. The temperatures reach maximum values within 20 years of emplacement and then slowly fall toward the ambient value (27 C). As expected, the waste with the highest heat rate (CSF) produces the highest temperatures.

Table B-3. Temperature Versus Time at the Borehole for the Three Different Types of Waste

<u>Time (yrs)</u>	<u>DHLW</u>	<u>PWR FUEL</u>	
		<u>SFA</u>	<u>CSF</u>
0.0	27.0	27.0	27.0
1.0	44.5	68.2	134.6
2.0	47.7	71.2	134.9
5.0	53.0	75.3	133.4
10.0	57.3	78.6	130.7
20.0	60.5	81.0	124.5
50.0	60.0	79.0	106.7
100.0	56.4	73.1	89.4
200.0	52.2	66.0	75.3
500.0	41.9	60.2	66.1
1,000.0	32.2	55.6	59.9

The BRINEMIG code requires estimates of temperature and temperature gradient as a function of distance from the center of the waste package. Temperatures for the three waste forms as a function of distance at four selected times after burial (1, 10, 100, and 1,000 years) are shown in

Figure 1. Temperature vs. Time for the Three Waste Forms

Figures B-2, B-3, and B-4. A significant characteristic of these figures is that the temperature gradient at late times is not zero even out to 300 meters. This characteristic contrasts with vertical emplacement where the gradient is zero at a distance midway between the package (30 meters or less) (McCauley and Raines, 1986). The implications for brine migration are that migration for horizontal emplacement may continue for longer periods of time than for vertical emplacement. However, it should be noted that the temperature levels will be somewhat higher for vertical emplacement and may partially compensate for the gradient effect.

B.3.2 Brine Migration

Brine migration rates and cumulative brine flows were estimated with the BRINEMIG code and the results are shown in Tables B-4 through B-6. Figures B-5 through B-14 are plots of these data. The brine migration model assumes there is a temperature gradient below which brine will not migrate. Two bounding values of this cutoff gradient were assumed for these calculations: zero and 0.125°C/cm. For DHLW, the temperature gradient never exceeded 0.125°C/cm for a distance large enough to set-up a measurable brine flow. Consequently, the model estimated zero flow for this case. However, measurable flows were estimated for DHLW when no cutoff gradient was assumed. Measurable flows were estimated for the two other waste types with both cutoff gradients.

BRINEMIG stops calculating when the brine flow rate goes to zero and outputs the time at which this occurs (51.8 years in Figure B-7 and 212.2 years in Figure B-11). The brine migration rates and cumulative flows are relatively low. The highest values (as expected) are for CSF with no cutoff gradient. The maximum rate is about $3.1 \times 10^{-10} \text{ m}^3/\text{sec}$ and the cumulative flow is about 0.60 m^3 . With a cutoff gradient of 0.125°C/cm, the maximum rate is the same for CSF but the maximum cumulative volume is only about 0.40 m^3 . Comparable estimates for vertical emplacement were not made in this study, however, a previous report (McCauley and Raines, 1986) contains estimates using the TEMP and BRINEMIG codes for the same waste type but different

Figure 2. Temperature vs. Distance for DHLW at four Times

Figure 8. Temperature vs. Distance for SFA at four Times

Figure 4. Temperature vs. Distance for CSF at four Times

Table B-4. Brine Migration for Defense High-Level Waste Packages

Time, yrs	Flow Rate, m ³ /sec x 10 ¹¹		Total Flow, m ³ x 10 ²	
	0.125°C/cm Cutoff	0°C Cutoff	0.125°C/cm Cutoff	0°C Cutoff
1.0	NA(a)	0.602	NA(a)	
2.0		0.616		
5.0		0.617		
10.0		0.580		
20.0		0.479		
50.0		0.259		0.687
100.0		0.143		0.993
200.0		0.080		1.33
500.0		0.019		1.74
1,000		0.002		1.89

(a) A gradient of 0.125°C/cm was not sustained in a sufficient volume of salt to cause brine migration.

Table B-5. Brine Migration for Intact Spent Fuel Packages

Time, yrs	Flow Rate, m ³ /sec x 10 ¹¹		Total Flow, m ³ x 10 ²	
	0.125°C/cm Cutoff	0°C Cutoff	0.125°C/cm Cutoff	0°C Cutoff
1.0	3.87	3.87		
2.0	3.89	3.89		
5.0	3.80	3.80		
10.0	3.61	3.61		
20.0	3.15	3.15		
25.0	2.86		2.71	
50.0	1.68	1.95	4.47	4.56
51.8	0.00		4.56	
100.0		1.10		6.88
200.0		0.555		9.36
500.0		0.301		13.21
1,000		0.166		16.73

Table B-6. Brine Migration for Consolidated Spent Fuel Packages

Time, yrs	Flow Rate, m ³ /sec x 10 ¹¹		Total Flow, m ³ x 10 ²	
	0.125°C/cm Cutoff	0°C Cutoff	0.125°C/cm Cutoff	0°C Cutoff
1.0	31.2	31.2		
2.0	30.0	30.0		
5.0	26.3	26.3		
10.0	22.1	22.1		
20.0	16.1	16.1		
50.0	7.44	7.44	24.3	24.3
100.0	3.46	3.46	32.2	32.2
200.0	1.06	1.57	39.1	39.7
212.2	0.00		39.5	
500.0		0.81		48.9
1,000		0.44		59.7

spacing between waste packages. The corresponding maximum rate was 2.1×10^{-10} m³/sec and the corresponding cumulative flows are 0.62 m³ for zero cutoff gradient and 0.34 m³ for 0.125°C/cm. This comparison tends to indicate that brine migration is primarily dependent on the heat load of an individual package rather than the heat load per unit area.

B.4 CONCLUSIONS

Brine migration rates and cumulative flows were estimated for horizontal emplacement of three different waste types and two bounding values of the temperature gradient at which brine migration ceases. The estimated rates and volumes are relatively low with the highest rate being 3.1×10^{-10} m³/sec and the highest volume being about 0.60 m³ with no cutoff gradient and about 0.40 m³ for 0.125°C/cm. The estimated rates and volumes are similar to previous estimates for the same waste type but with vertical emplacement and different spacing between packages. This comparison tends to indicate that brine migration is primarily dependent on the heat load of an individual package rather than the heat load per unit area.

Figure 5. Flowrate vs. Time for DHLW, Cutoff=0.0

Figure 6. Total Flow vs. Time for DHLW, Cutoff=0.0

Figure 7. Flowrate vs. Time for SFA , Cutoff=0.125

Figure 8. Total Flow vs. Time for SFA, Cutoff=0.125

Figure 9. Flowrate vs. Time for SFA, Cutoff=0.0

Figure 10. Total Flow vs. Time for SFA, Cutoff=0.0

Figure 11. Flowrate vs. Time for CSF, Cutoff=0.125

Figure 12. Total Flow vs. Time for CSF, Cutoff=0.125

Figure 13. Flowrate vs. Time for CSF, Cutoff=0.0

Figure 14. Total Flow vs. Time for CSF, Cutoff=0.0

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APPENDIX C

GENERAL CORROSION PENETRATION

APPENDIX C

GENERAL CORROSION PENETRATION

The allowance for general corrosion of the waste package container over the approximately 1,000-year containment period was estimated by means of a corrosion penetration model. This model is based upon experimental data for the weight loss of steel coupons in surrogate Permian Basin salt with 20 weight percent water added as high-magnesium (~50,000 parts per million) brine. General corrosion measurements of the A216 grade WCA low carbon steel in autoclave-type testing apparatus were performed at Pacific Northwest Laboratories (PNL). The data were collected from tests at 90°C, 150°C, and 200°C over time periods of 1 month to 1 year. High magnesium brines were used in the tests to account for the accelerated corrosion rates which are expected to occur if the container is exposed to brines of the high magnesium composition comparable to those present as small inclusions in the Permian Basin salt. Tables C-1 through C-7 summarize the data accrued to date.

The corrosion product layer formed on A216 carbon steel test coupons in high-magnesium salt brines at all test temperatures was found in the testing performed up to one year in duration to be an amakinite compound, i.e.,

This compound, when mechanically constrained in wet salt tests against the base metal coupon, formed a layer which somewhat protected the metal; that is, the corrosion rate decreased with time. The postulated reaction stoichiometry for the formation of this corrosion product is:

Table C-1. Corrosion Rate of A216 Steel in PBB3 Brine
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
90	1 month	as-cast	1.8	(45)	2.23	(56.7)
			2.7	(68)		
		normalized	2.1	(53)	2.38	(60.4)
			2.7	(67)		
		weldment	1.9	(49)	1.78	(45.2)
			1.6	(41)		
90	3 months	as-cast	2.3	(29)	1.90	(48.2)
			1.5	(26)		
		normalized	1.4	(23)	1.41	(35.9)
			1.4	(25)		
		weldment	1.5	(26)	1.19	(30.2)
			0.91	(31)		
90	6 months	as-cast	2.2	(56)	2.22	(56.3)
			2.2	(56)		
		normalized	1.6	(39)	1.59	(40.3)
			1.6	(41)		
		weldment	0.72	(26)	1.21	(30.7)
			1.7	(43)		
150	1 month	as-cast	46	(1200)	45.1	(1140)
			45	(1100)		
		normalized	59	(1500)	59.8	(1520)
			60	(1500)		
		weldment	38	(970)	34.1	(865)
			30	(770)		
150	3 months	as-cast	41	(1000)	41.1	(1050)
			41	(1000)		
		normalized	53	(1300)	52.7	(1340)
			53	(1300)		
		weldment	29	(740)	28.6	(728)
			28	(720)		
150	6 months	as-cast	37	(940)	37.1	(943)
			37	(950)		
		normalized	48	(1200)	47.8	(1220)
			48	(1200)		
		weldment	28	(720)	27.8	(706)
			27	(690)		

Table C-1. Corrosion Rate of A216 Steel in PBB3 Brine
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
200	1 month	as-cast Lot 1	220	5500	220	(5600)
			230	5700		
			220	5600		
			210	5400		
200	1 month	normalized Lot 1	260	6500	250	(6400)
			260	6600		
			250	6300		
			250	6400		
200	1 month	weldment Lot 1	130	3300	130	(3200)
			140	3500		
			130	3200		
			120	2900		
200	1 month	as-cast Lot 2	170	4300	170	(4300)
			170	4300		
			170	4300		
			170	4300		
200	1 month	normalized Lot 2	160	4000	160	(4000)
			160	4100		
			160	4100		
			160	4000		

Table C-2. Corrosion Rates of A216 Grade WCA Steel in PBB1
Salt With PBB3 Brine at 150°C

Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
		mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
1 month	as-cast	32	810	33	830
		33	850		
	normalized	38	970	40	1000
		42	1100		
	weldment	24	610	21	520
		17	440		
3 months	as-cast	31	800	31	780
		30	760		
	normalized	59	1500	55	1400
		52	1300		
	weldment	11	280	12	300
		13	320		
6 months	as-cast	29	750	31	780
		32	810		
	normalized	47	1200	47	1200
		48	1200		
	weldment	7.9	200	8	200
		8.1	210		

Table C-3. Corrosion Rates of A216 Grade WCA Steel in Surrogate Site-Specific Salt With PBB3 Brine
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
90	1 month	as-cast, lot 1	1.1	28	1.2	31
			1.3	34		
		normalized, lot 1	1.3	32	1.3	33
			1.3	34		
	3 months	as-cast, lot 1	1.5	37	1.5	38
			1.5	38		
90	3 months	as-cast, lot 1	1.2	30	1.1	28
			1.0	26		
		normalized, lot 1	0.91	23	0.95	24
			0.99	25		
	3 months	as-cast, lot 2	1.5	37	1.5	37
			1.5	38		
90	3 months	as-cast, lot 2	1.4	36	1.4	36
			1.4	36		
		normalized, lot 2	1.4	36	1.4	36
			1.4	36		
	6 months	as-cast, lot 1	2.4	60	2.1	53
			1.9	47		
90	6 months	as-cast, lot 1	1.9	47	2.1	53
			2.3	59		
		normalized, lot 1	1.9	47	0.96	24
			2.3	59		
	1 month	as-cast	0.83	21	0.96	24
			1.1	28		
150	1 month	as-cast	27	680	27	690
			27	700		
		normalized	32	810	33	840
			34	870		
	1 month	weldment	13	340	14	360
			15	390		

Table C-3. Corrosion Rates of A216 Grade WCA Steel in Surrogate
Site-Specific Salt With PBB3 Brine
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate mils/yr (μm/yr)		Mean Corrosion Rate mils/yr (μm/yr)	
150	3 months	as-cast	37	940	37	930
			36	920		
		normalized	52	1300	52	1300
	52		1300			
	weldment	7.4	190	18	460	
		29	730			
150	6 months	as-cast	35	900	34	870
			33	840		
		normalized	47	1200	47	1200
	48		1200			
	weldment	5.6	140	6.6	170	
		7.5	190			

Table C-4. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB3 Brine, 5% by Weight Water
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
90	3 months	as-cast	0.47	12	0.52	13
			0.65	17		
			0.43	11		
		normalized	0.64	16	0.63	16
			0.65	16		
			0.61	15		
		weldment	0.66	17	0.79	20
			0.80	20		
			0.90	23		
90	6 months	as-cast	0.97	25	0.76	19
			0.78	20		
			0.54	14		
		normalized	0.83	21	0.92	23
			0.93	24		
			1.01	26		
		weldment	0.93	24	0.89	23
			0.78	20		
150	3 months	as-cast	17	440	14	370
			13	330		
			13	320		
		normalized	28	710	30	750
			26	670		
			35	880		
		weldment	4.8	120	8.7	220
			7.4	190		
			14	350		
150	6 months	as-cast	15	370	14	360
			15	370		
			14	340		
		normalized	38	960	30	760
			24	600		
			28	710		

Table C-4. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB3 Brine, 5% by Weight Water
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
150	6 months	weldment	15	370	9.6	240
			6.7	170		
			7.5	190		
200	3 months	as-cast	35	880	35.6	905
			35	880		
			37	950		
		normalized	25	650	24.1	613
			22	560		
			26	650		
		weldment	26	670	23.7	604
			21	540		
			24	600		

Table C-5. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB1 Brine, 5% by Weight Water
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
90	3 months	as-cast	0.12	3.1	0.12	3.0
			0.12	3.1		
			0.11	2.9		
		normalized	0.13	3.2	0.11	2.9
			0.11	2.8		
			0.11	2.7		
		weldment	0.13	3.4	0.14	3.5
			0.15	3.9		
			0.13	3.2		
90	6 months	as-cast	0.082	2.1	0.060	1.5
			0.044	1.1		
			0.053	1.4		
		normalized	0.063	1.6	0.073	1.9
			0.085	2.2		
			0.072	1.8		
		weldment	0.036	0.92	0.066	1.7
			0.084	2.2		
			0.078	2.0		
150	3 months	as-cast	0.25	6.3	0.28	7.0
			0.26	6.7		
			0.32	8.2		
		normalized	0.30	7.5	0.27	7.0
			0.28	7.1		
			0.25	6.3		
		weldment	0.82	21	0.49	12
			0.33	8.3		
			0.33	8.3		
150	6 months	as-cast	0.35	8.9	0.43	11
			0.51	13		
			0.42	11		
		normalized	0.31	7.8	0.40	10
			0.74	19		
			0.14	3.5		

Table C-5. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB1 Brine, 5% by Weight Water
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate mils/yr (μm/yr)		Mean Corrosion Rate mils/yr (μm/yr)	
150	6 months	weldment	0.77	20	0.56	14
			0.45	11		
			0.46	12		
200	3 months	as-cast	4.3	110	3.65	92.7
			3.5	89		
			3.2	81		
		normalized	3.1	78	3.29 ± 1.14	83.6
			3.0	75		
			3.8	97		
		weldment	2.9	73	2.99 ± 0.72	75.8
			2.8	71		
			3.3	84		

Table C-6. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB3 Brine, 20% by Weight Water
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate mils/yr (µm/yr)		Mean Corrosion Rate mils/yr (µm/yr)	
90	3 months	as-cast	2.1	54	2.3	59
			2.2	55		
			2.7	68		
		normalized	1.9	49	1.9	49
			1.8	45		
			2.1	54		
		weldment	1.7	43	1.5	39
			1.8	45		
			1.1	29		
90	6 months	as-cast	3.1	79	3.2	81
			3.2	80		
			3.3	83		
		normalized	3.6	92	2.5	64
			1.7	43		
			2.2	56		
		weldment	1.7	43	1.6	40
			1.6	41		
			1.4	35		
150	3 months	as-cast	19	490	19	480
			22	560		
			15	390		
		normalized	39	990	38	960
			37	940		
			38	960		
		weldment	20	500	14	360
			12	310		
			10	260		
150	6 months	as-cast	15	390	16	410
			17	430		
			17	430		
		normalized	40	1000	40	1000
			42	1100		
			38	970		

Table C-6. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB3 Brine, 20% by Weight Water
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate mils/yr (µm/yr)		Mean Corrosion Rate mils/yr (µm/yr)	
150	6 months	weldment	7.4 17 3.8	190 440 95	9.5	240
200	3 months	as-cast	32 42 32	810 1100 810	35	910
		normalized	70 70 64	1800 1800 1600	68	1700
		weldment parent metal	12 26 22	300 660 560	20	510
		weld metal	32 42	810 1100	37	960
		interface	50 58 88	1300 1500 2200	65	1700
200	6 months	as-cast	21 25 26	530 640 660	24	610
		normalized	45 60 60	1100 1500 1500	55	1400
		weldment parent metal	4 5 4	100 100 100	4.3	110
		weld metal	23 24 14	580 610 360	20	520
		interface	23 38 26	580 970 660	29	740

Table C-7. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB1 Brine, 20% by Weight Water
(Page 1 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
90	3 months	as-cast	0.31	7.8	0.33	8.3
			0.35	8.8		
			0.33	8.4		
		normalized	0.33	8.3	0.34	8.5
			0.32	8.2		
			0.36	9.2		
		weldment	0.27	7.0	0.30	7.5
			0.27	6.8		
			0.35	8.8		
90	6 months	as-cast	0.11	2.8	0.15	3.8
			0.16	4.1		
			0.17	4.4		
		normalized	0.17	4.4	0.17	4.2
			0.16	4.0		
			0.16	4.2		
		weldment	0.14	3.5	0.15	3.7
			0.15	3.8		
			0.15	3.7		
150	3 months	as-cast	0.97	26	0.84	21
			0.82	21		
			0.70	18		
			0.99	25		
			0.74	19		
		normalized	1.0	26	0.83	21
			0.87	22		
			0.66	17		
			0.76	19		
			0.84	21		
		weldment	0.79	20	0.73	19
			0.71	18		
			0.80	20		
			0.65	17		
			0.83	21		
			0.58	15		

Table C-7. Corrosion Rates of A216 Steel in Surrogate Site-Specific
Salt With PBB1 Brine, 20% by Weight Water
(Page 2 of 2)

Temp (°C)	Test Duration	Material Condition	Corrosion Rate		Mean Corrosion Rate	
			mils/yr	($\mu\text{m}/\text{yr}$)	mils/yr	($\mu\text{m}/\text{yr}$)
150	6 months	as-cast	0.46	12	0.42	11
			0.43	11		
			0.36	9.1		
		normalized	0.49	12	0.36	9.2
			0.17	4.3		
			0.42	11		
		weldment	0.48	12	0.54	14
			0.50	13		
			0.65	16		
200	3 months	as-cast	0.38	9.5	0.47	12
			0.30	7.5		
			0.75	19		
		normalized	0.79	20	0.54	14
			0.26	6.5		
			0.58	15		
		weldment	1.1	28	1.1	27
			1.1	28		
			1.1	27		
200	6 months	as-cast	0.35	9.0	0.36	9.0
			0.39	10		
			0.32	8.1		
		normalized	0.42	11	0.37	9.5
			0.32	8.2		
			0.38	9.7		
		weldment	0.67	17	0.61	15
			0.53	14		
			0.62	16		

The waste package emplacement scenario involves the insertion of crushed salt (or other packing material) around the waste package in the borehole. This packing material is expected to consolidate with time as the borehole contracts due to salt creep around the waste package. If the corrosion layer on the A216 container can thereby be assumed to resemble the amakinite layer of the wet salt tests, the following corrosion rate analysis may be performed.

Assuming that the corrosion (amakinite layer formation) rate for the A216/high-magnesium brine is proportional to the magnesium ion concentration at the corrosion layer/metal interface, i.e.,

$$\frac{dp}{dt} = k C_i \quad (C-2)$$

where

C_i = Concentration of Mg^{++} ion at interface
 p = Depth of corrosion (amakinite layer)
 k = Rate constant for reaction

and that the magnesium ion migrates to the corrosion layer/metal interface via diffusion, i.e.:

$$J = D \frac{C_0 - C_i}{l} \quad (C-3)$$

where

C_0 $\equiv$ Concentration of Mg^{++} in the brine
 l $\equiv$ Depth of corrosion layer
 $= bp$
 D $\equiv$ Diffusion coefficient of Mg^{++} in corrosion layer
 J $\equiv$ Flux of Mg^{++} ions to metal surface

and that all Mg^{++} ions reaching the metal surface react to form the amakinite corrosion product, i.e.,

$$J = v \frac{dp}{dt} \quad (C-4)$$

where

v = Stoichiometric reaction coefficient

then

$$v \frac{dp}{dt} = D \frac{C_o - C_i}{bp}$$

$$p \frac{dp}{dt} = \frac{D}{vb} (C_o - C_i)$$

$$= \frac{D}{vb} C_o - \frac{1}{k} \frac{dp}{dt}$$

$$p \frac{dp}{dt} + \frac{D}{vbk} \frac{dp}{dt} = \frac{DC_o}{vb}$$

$$pdp + \frac{D}{vbk} dp = \frac{DC_o}{vb} dt$$

(C-5)

integrating

$$\frac{p^2}{2} + \frac{D}{vbk} p = \frac{DC_o}{vb} t$$

or

$$p = -\frac{D}{vbk} + \frac{D^2}{v^2 b^2 k^2} - 2 \frac{DC_o}{bv} t^{\frac{1}{2}}$$

$$= -\frac{D}{vbk} + \frac{D}{vbk} \left[1 - \frac{2 vbk^2 C_o}{D} t^{\frac{1}{2}} \right]$$

$$= \frac{D}{vbk} \left[1 - \frac{2 vbk^2 C_o}{D} t^{\frac{1}{2}} - 1 \right]$$

$$\text{let } B = \frac{D}{\sqrt{b}k} \text{ and } A = \frac{DC_o}{bv} \quad (C-6)$$

The rate kinetics for high-magnesium brine general corrosion of A216 steel was therefore represented by the expression:

$$p = B \left(1 + \frac{2A}{B^2} t \right)^{\frac{1}{2}} - 1 \quad (C-7)$$

where p is the corrosion layer penetration depth, t is the time, and A and B are fitting constants derived from the experimental corrosion rate data.

This expression was derived assuming that the rate of corrosion (amakinite layer formation) at the metal/corrosion layer interface is controlled by the diffusion of magnesium ion from the brine through the amakinite layer to this interface. Both 'brine-only' and 'wet salt' corrosion tests showed that the high-magnesium brine corrosion product to consist solely of this amakinite-type product.

Corrosion penetration tests on as-cast A216 coupons in Permian Basin surrogate salt with 20 percent water added as high-magnesium brine at 150°C over a 1-year period (Table C-6 and Figure C-1) resulted in general corrosion penetration depth data which gave the following values for the fitting constants 'A' and 'B':

$$A = 7.3 \times 10^{-8} \text{ cm}^2/\text{hr}$$

$$B = 0.003 \text{ cm}$$

Subsequently, corrosion tests at 90°C, 150°C, and 200°C with 20 percent water as high-magnesium brine were conducted at 3 and 6 month durations were performed in a similar test apparatus. These data are also shown in Table C-6.

Both A and B are parameters proportional to the diffusion coefficient (D) for the magnesium ion across the amakinite corrosion layer which forms for

Figure C-1 Corrosion of A216 Cast Steel in PBB3 Brine, 20% Water, at 150°C

A216 in high-magnesium brines. If it is assumed that the diffusion coefficient has an Arrhenius-type temperature dependence, of the form

$$D = D_0 e^{-E_D/RT}$$

and the reaction rate constant also has an Arrhenius-type temperature dependence of the form

$$k = k_0 e^{-E_k/RT}$$

The temperature dependence of the coefficients 'A' and 'B' may be represented by the expressions:

$$A = A_0 e^{-E_D/RT}$$

$$B = B_0 e^{-(E_D - E_k)/RT}$$

If it can be further assumed that the activation energy for the reaction (C-1), E_k , is much lower (approximately 1-4 kcal/mole) than the activation energy for diffusion of the magnesium ion through the corrosion layer, E_D , (approximately 10-20 kcal/mole, then the 'A' and 'B' constants may be represented as

$$A = A_0 e^{-E/RT}$$

$$B = B_0 e^{-E/RT}$$

$$\text{where } E = E_D \quad (E_D - E_k)$$

Corrosion tests for as cast A216 steel in Permian Basin surrogate salt with 20 weight percent water as magnesium-rich brines performed at 90°C, 150°C and 200°C for a 3-month duration (Table C-6) resulted in a derived value for the activation energy, E , of 17.75 kcal/mole, with $A_0 = 108.35 \text{ cm}^2/\text{hr}$ and $B_0 = 4.4529 \times 10^6 \text{ cm}$.

The calculation of a corrosion allowance (p) for the 1,000-year repository period is approached as follows:

$$p = \int_0^{1,000} R dt$$

where R is the instantaneous corrosion rate at any time t over the repository period. For the corrosion penetration model detailed above this rate, R_C may be given by:

$$\begin{aligned} R_C &= \frac{dp}{dt} = \frac{d}{dt} (2At + B^2)^{\frac{1}{2}} - B \\ &= \frac{d}{dt} (2At + B^2)^{\frac{1}{2}} \\ &= \frac{1}{2} (2At + B^2)^{-\frac{1}{2}} \frac{d}{dt} (2At + B^2) \\ &= A (2At + B^2)^{-\frac{1}{2}} \end{aligned} \quad (C-9)$$

and the corresponding general corrosion allowance model becomes:

$$p = \int_0^{t_m} A (2At + B^2)^{-\frac{1}{2}} dt \quad (C-10)$$

where t_m is the time period over which the rate model is applied. p may be approximated by:

$$p \approx \sum A (2At_i + B^2)^{-\frac{1}{2}} \Delta t_i$$

and since $A = A_0 e^{-E/RT}$ and $B = B_0 e^{-E/RT}$

$$\begin{aligned} R &= A_0 e^{-E/RT} (2A_0 e^{-E/RT} t + B_0^2 e^{-2E/RT})^{-\frac{1}{2}} \\ &= A_0 (2A_0 t e^{E/RT} + B_0^2)^{-\frac{1}{2}} \end{aligned}$$

$$= \frac{2t}{A_o} e^{E/RT} + \frac{B_o}{A_o}^2 - \frac{1}{2} \quad (C-11)$$

so the general corrosion penetration for the time period t_m becomes

$$P = \sum_{i=1}^{t_m} \frac{2t_i}{A_o} e^{E/RT} + \frac{B_o}{A_o}^2 - \frac{1}{2} \Delta t_i \quad (C-12)$$

where

$$A_o = 108.35 \text{ cm}^2/\text{hr}$$

$$B_o = 4.4529 \times 10^6 \text{ cm}$$

$$E = 17,520 \text{ cal/mol}$$

Corrosion penetration curves for 90°C, 150°C, and 200°C calculated using these values are plotted in Figure C-2 for times up to 1 year. Also plotted are experimentally measured corrosion penetration values at 3 and 6 months.

The temperature of the waste package/borehole interface for 12 PWR consolidated spent fuel (CSF), intact spent fuel (ISF), and defense high-level waste (DHLW) in both vertical and horizontal configurations has been calculated by means of the HEATING5 code for the emplacement conditions detailed in this report. For the consolidated spent fuel and the intact spent fuel the higher overall temperature history is obtained for the vertical emplacement configuration; for the defense high-level waste the higher overall temperature history is obtained for the horizontal emplacement configuration. Table C-8 summarizes these temperature histories.

The significant difference in the temperature histories of these three types of waste, and the experimentally indicated strong dependence of the corrosion reaction rate on the temperature, indicate that different general corrosion allowances for the three waste forms are indicated. The temperature histories and the expression for the calculation of the general corrosion penetration are combined with the following assumptions to generate a corrosion allowance for the A216 waste containers for the repository period:

FIGURE C-2

PREDICTED PENETRATION VS. EXPERIMENTAL
20% PBB3/PBB1

- There is enough magnesium-saturated brine present immediately after package emplacement that the corrosion reaction is not stoichiometrically limited, that is the corrosion environment resembles that of the corrosion test conditions.
- The corrosion reaction kinetics indicated by Equation C-7 hold for the first year of emplacement, roughly corresponding to the period for which test data are now available.
- After 1 year, the reaction rate depends only on the borehole temperature. The corrosion penetration at times greater than 1 year is calculated by setting the time, t , in the equation C-12 at 1 year and using the actual time-dependent temperature history given in Table C-8. This usage corresponds to the conservative assumption that the corrosion layer adherent to the metal reaches a maximum effective thickness (i.e., protective effect) at 1 year, after which it remains constant.

Using these assumptions, Table C-9 gives the corrosion penetration as a function of time for the three waste forms obtained using Equation C-12. The corrosion penetration rate at the 1 year point is 0.0126 cm/yr for CFF, 0.0027 cm/yr for ISF, and 0.0006 cm/yr for DHLW.

Table C-8. Temperature, °C

Time (yrs)	CSF (vert.)	ISF (vert.)	DHLW (Horiz.)
0.5	126.35	74.88	42.17
1.0	131.18 (0.0126 cm/yr)	80.60 (0.0027 cm/yr)	44.54 (0.0006 cm/yr)
2.0	134.03	85.17	47.54
3.0	134.29	87.05	49.57
4.0	133.57	87.93	51.08
5.0	132.31	88.28	52.25
6.0	131.94	88.86	53.37
7.0	131.47	89.35	54.41
8.0	130.87	89.69	55.30
9.0	130.11	89.89	56.11
10.0	129.24	89.98	56.82
11.0	128.78	90.20	57.43
12.0	128.29	90.40	57.97
13.0	127.75	90.54	58.43
14.0	127.14	90.61	58.82
15.0	126.49	90.63	59.16
16.0	125.87	90.64	59.47
17.0	125.22	90.62	59.75
18.0	124.52	90.55	59.98
19.0	123.78	90.45	60.18
20.0	123.01	90.30	60.34
25.0	119.72	89.65	60.99
30.0	115.85	88.43	61.12
40.0	110.77	86.88	61.06
50.0	104.78	84.44	60.09
60.0	100.90	82.81	59.04
70.0	96.73	80.87	57.65
80.0	94.76	80.02	56.56
90.0	92.81	79.17	55.45
100.0	90.74	78.18	54.20
200.0	79.40	72.08	46.16
300.0	75.50	68.22	42.07
400.0	71.00	66.51	39.91
500.0	68.45	64.85	38.54
600.0	67.21	63.86	37.59
700.0	65.88	62.98	36.86
800.0	64.68	62.03	36.25
900.0	63.97	61.45	35.76
1000.0	63.30	61.00	35.37

Table C-9. General Corrosion Penetration (cm)

Time (yrs)	CSF (vert.)	SFA (vert.)	DHLW (Horiz.)
0.1	.0041	.0005	.0001
0.5	.0113	.0017	.0004
1.0	.0174	.0029	.0007
5.0	.0625	.0140	.0035
10.0	.1160	.0298	.0080
50.0	.4026	.1447	.0472
100.0	.6149	.2595	.0899
500.0	1.3629	.8561	.3073
1,000.0	1.9924	1.4253	.4940

APPENDIX D

STRUCTURAL ANALYSIS

APPENDIX D

STRUCTURAL ANALYSIS

This appendix discusses the methods used to determine the thicknesses of the cylindrical walls and top and bottom heads for the conceptual designs of the waste packages described in this report. The cylinder walls and heads were analyzed by using the rules and methods of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. Paragraph NB-3133.3, with the applicable curves in Appendix VII, was used for the cylinder walls. Paragraph NC-3225.2, with the allowable stress table in Appendix I, was used to analyze the heads.

The structural analysis of the containers was based on the cavity diameter of the specific container, the design uniform lithostatic pressure of 18.0 MPa (2,610 psi), and the allowable stress for the low-carbon steel at the maximum temperature of service. (Stresses from nonuniform loads were assumed to be lower than the stresses from the full lithostatic load because the non-uniform loads should occur early in the borehole closure process, long before the full lithostatic loads would be imposed.) Since the structural analysis was performed in parallel with the thermal analysis, a maximum service temperature of 250°C was assumed. Later, the results of the thermal analysis showed that for all waste packages the maximum service temperature was < 150°C (see Appendix A). As a result the allowable stress used in the calculations was less than necessary and the walls are conservatively thicker than necessary for the conditions of service.

For the cylinders, the allowable pressure is the lesser of P_{a1} or P_{a2} , where:

$$P_{a1} = \frac{2.167}{D/t} - 0.0833 \quad B$$

$$P_{a2} = \frac{2S}{D/t} \left(1 - \frac{1}{D/t} \right)$$

where

D = outside diameter

t = wall thickness

B = value from Table VII-1101-2

S = value from Appendix I.

For all values considered, P_{a1} was less than P_{a2} . The calculation involves the use of the container outside diameter and wall thickness. Since the container size is based on the cavity diameter, the actual calculation proceeds by assuming a wall thickness from which the outside diameter is calculated. Then, for the outside diameter corresponding to a given wall thickness, the allowable external pressure is calculated. This procedure is repeated for several different wall thicknesses in an iterative manner for each container until the minimum wall thickness to sustain the design pressure is determined.

Once the minimum wall thickness is determined the design thickness is obtained by adding to the structural thickness the thickness predicted to be needed to allow for material loss by general corrosion for 1,000 years (see Appendix C). The initial pressure capacity of the total wall thickness prior to any thinning of the wall by corrosion can be computed directly with the above formula.

The thickness of the container heads required for the design pressure is calculated by the expression

$$t = d (CP/S)^{0.5}$$

where

t = head thickness, in.

d = cavity diameter, in

C = 0.33, the most conservative head attachment factor

P = external pressure, psi

S = the allowable stress for cast low-carbon steel

The minimum head thickness was calculated for each container and the thickness for corrosion allowance was added to obtain the total design head thickness.

APPENDIX E

SHIELDING ANALYSES

APPENDIX E - SHIELDING ANALYSES

E.1 INTRODUCTION

The radiation shielding calculations of the horizontal emplacement concept and a vertical emplacement concept are presented in this report. The reference data provided for the required shielding analysis are provided by ONWI (Albert, 1987). The horizontal emplacement concept is illustrated in Figure E-1 and consists of two concentric horizontal boreholes, the smaller (1.05 meter diameter) containing the waste package, and the larger (1.35 meter diameter) containing a shield plug (1.25 meter diameter). Four waste forms are described in the Reference Data for emplacement in the repository. These waste forms are defense high level waste and consolidated spent fuel from 6, 9, and 12 pressurized water reactor fuel assemblies.

The vertical emplacement concept is illustrated in Figure E-2. The configuration is similar to the horizontal emplacement concept but exhibits some important differences. The waste container and waste canister are concentric with the emplacement borehole. Furthermore the salt plug is flush with the top of the waste canister with no clearance gap between the salt plug and the emplacement borehole.

The shielding analysis consists of two parts, the package radiation analysis and the shield plug analysis. In the package radiation analysis the surface dose rates on the sides and ends of the waste canister are calculated. In the shield plug analysis the dose rates around the shield plug and the face of the horizontal and vertical emplacements are evaluated.

The shielding analysis has been performed using the MORSE multigroup Monte Carlo radiation transport code (Straker, 1970 and 1972). The Monte Carlo Code was used for both the package radiation analysis and the shield plug analysis. A verification of the program for the emplacement shielding analysis was performed and reported in a separate report (Albert, 1986).

E-1
Figure 1.1: Horizontal Emplacement Concept

E-2
Figure 1-2: Vertical Emplacement Concept

E.2 HORIZONTAL EMPLACEMENT ANALYSIS

E.2.1 Calculational Approach

For the analysis of the surface dose rates on the sides and ends of the waste packages, the MORSE code was configured to estimate the doses using a boundary crossing estimator. This approach results in a surface dose estimate which is an average over the scoring surface. During the same calculation, the location, energy, and direction of particles escaping from the waste package was written to a file for subsequent use as a surface current source for the shield plug analysis.

Separate calculations were made for the neutron and gamma ray sources in the waste packages. These sources were calculated separately in order to provide as much insight into the components of the radiation doses in the horizontal emplacement concept and to facilitate the application of variance reduction techniques.

E.2.1.1 Geometrical Modeling

Figure E-3 gives the specified dimensions of the waste package and Figure E-4 shows the internal configuration of the four waste forms. The configuration and dimensions of the horizontal emplacement and the shield plug are shown in Figure E-5.

The geometry specifications for the waste packages, the horizontal emplacement, and the shield plug were reduced to a combinatorial geometry model for use in the shielding analysis. Figures E-6 through E-9 show the cross section of the four waste form configurations as modeled in the combinatorial geometry model. Two of the waste forms (CSF9 and CSF12) contain a hexagonal central void which was approximated by cylindrical void of equal cross sectional area. No other approximation was made in the geometry model used in the radiation analysis.

	DHLW	CSF 6 PWR	CSF 9 PWR	CSF 12 PWR
Power, Watts	470.0	2,300.0	4,950.0	6,600.0
Canister				
Dc, cm	61.0	45.4	54.3	62.0
Lc, cm	305.0	470.0**	400.0**	400.0**
Lh, cm	231.0	365.0**	365.0**	365.00**
Tc, cm	0.95	0.2	0.2	0.2
Container				
Dc, cm	62.5	47.5	56.8	64.5
Lc, cm	80.8*	67.9*	76.8*	84.5*
Lh, cm	336.5*	446.5*	446.5*	446.5*
Tc, cm	8.62*	10.0*	10.0*	10.0*
Th, cm	16.7*	18.3*	18.3*	18.3*
Weights				
Waste Form				
MTU	N/A	2.78**	4.17**	5.55**
Net	1,460.0	3,960.0**	5,940.0**	7,520.0**
Gross	1,940.0	4,192.0**	6,290.0**	8,390.0**

* Approximate value for use in thermal model; varies in shield model.

** Representative value only varies with fuel design.

E-3
Figure 2-1: Waste Package Dimensions

2.1 CALCULATIONAL APPROACH

7

Defense High Level Waste

Consolidated Spent Fuel

6 elements

9 elements

12 elements

(All dimensions in centimeters)

E-4

Figure 2.2: Waste Form Configurations

E5

Figure 2.3: Horizontal Emplacement Shield Plug Geometry

E-6

Figure 2.4: DHLW Waste Package - Horizontal Configuration

E-7

Figure 2-3: CSF-6 Waste Package - Horizontal Configuration

E-8
Figure 2.6: CSF-9 Waste Package - Horizontal Configuration

E-9

Figure 2.7: CSF-12 Waste Package - Horizontal Configuration

The combinatorial geometry models were checked for correctness using the PICTUR geometry debugging facility of the MORSE code. This feature of the code produces a visual display of a defined cross section of the geometry model. Since the same ray tracing routines are used to produce the display as are used in tracking particle trajectories in the radiation analysis, this check of the geometry model also identifies undefined regions in the geometry model.

E.2.1.2 Material Compositions and Nuclear Cross Sections

In order to determine the nuclide number densities for calculation of the macroscopic cross sections for the several waste forms a procedure was followed which conserved the heavy metal mass, waste form volume, and gross mass for each waste form. For the consolidated spent fuel waste forms, the uranium number density was calculated from the specified mass and calculated volume of the waste form. The oxygen number density corresponding to this uranium number density was then calculated. The remainder of the mass necessary to give the total waste form mass was assumed to be zirconium. The calculated nuclide number densities are given in Table E-1.

The nuclear cross sections used in the package radiation analysis were the Cask cross section library which was developed at Oak Ridge National Laboratory for use in bulk shielding calculations. The Cask library is a coupled neutron and gamma-ray library containing 22 neutron groups and 18 gamma ray groups. The energy group structure for this library is given in Table E-2.

E.2.1.3 Radiation Source

The neutron spectrum used in the radiation analysis is shown in Table E-3. This spectrum was regrouped into the 22 group structure as shown in Table E-4. The gamma ray source spectrum was obtained from a nuclide depletion analysis (ORIGEN) with 10-year cooled values used for the consolidated spent fuel gamma ray source. The source term for DHLW is for 5-year-old

Table E-1. Number Densities for Macroscopic Cross Sections

Material	Element	Partial Density (Kg/m ³)	Atomic Mass (gm/mole)	Density (Kg/m ³)	Atomic Density (atoms/barn*cm)
Waste Form (DHLW)	O	943.5	15.9994	2350	3.551 x 10 ⁻²
	Si	463.2	28.0860		9.931 x 10 ⁻³
	Na	943.5	22.9898		2.471 x 10 ⁻²
Waste Form (CSF-6)	U	4794.6	238.0300	6829	1.213 x 10 ⁻²
	O	644.3	15.9940		2.426 x 10 ⁻²
	Zr	1390.0	91.2200		9.176 x 10 ⁻³
Waste Form (CSF-9)	U	5012.0	238.0300	7136	1.268 x 10 ⁻²
	O	673.8	15.9940		2.537 x 10 ⁻²
	Zr	1450.0	91.2200		9.572 x 10 ⁻³
Waste Form (CSF-12)	U	5110.8	238.0300	7287	1.293 x 10 ⁻²
	O	686.6	15.9940		2.585 x 10 ⁻²
	Zr	1490.0	91.2200		9.836 x 10 ⁻³
Iron	Fe	7800.0	55.8470	7800	8.411 x 10 ⁻²
Salt	Na	860.0	22.9898	2190	2.259 x 10 ⁻²
	Cl	1330.0	35.4530		2.259 x 10 ⁻²
Salt Plug (High ρ)	Na	687.2	22.9898	1750	1.805 x 10 ⁻²
	Cl	1062.8	35.4530		1.805 x 10 ⁻²
Salt Plug (Low ρ)	Na	430.0	22.9898	1095	1.130 x 10 ⁻²
	Cl	665.0	35.4530		1.130 x 10 ⁻²

Table E-2. Energy Group Structure for Cross Sections

Group #	Upper Energy Boundary, (eV)	
	Neutron	Gamma Ray
1	1.5000×10^7	1.00×10^7
2	1.2214×10^7	8.00×10^6
3	1.0000×10^7	6.50×10^6
4	8.1873×10^6	5.00×10^6
5	6.3600×10^6	4.00×10^6
6	4.9658×10^6	3.00×10^6
7	4.0657×10^6	2.50×10^6
8	3.0119×10^6	2.00×10^6
9	2.4659×10^6	1.66×10^6
10	2.3500×10^6	1.33×10^6
11	1.8268×10^6	1.00×10^6
12	1.1080×10^6	0.80×10^6
13	5.5020×10^5	0.60×10^6
14	1.1109×10^5	0.40×10^6
15	3.3546×10^3	0.30×10^6
16	5.8294×10^2	0.20×10^6
17	1.0130×10^2	0.10×10^6
18	2.9020×10^1	0.05×10^6
19	1.0677×10^1	0.02×10^6
20	3.0590	
21	1.1253	
22	4.1400×10^{-1}	
	1.0000×10^{-1}	

Table E-3. Specified Neutron Source Spectrum

Upper Energy (MeV)	Fraction of Neutrons in Group
14.62	0.00045
12.21	0.00181
10.0	0.00555
8.187	0.01512
6.703	0.02764
5.488	0.04074
4.993	0.05567
3.679	0.10400
3.012	0.09150
2.466	0.09054
2.019	0.08910
1.653	0.08390
1.353	0.06848
1.108	0.06636
0.907	0.09921
0.608	0.06068
0.408	0.08611
0.1119	0.01165
0.0150	0.00141

Table E-4. Regrouped Neutron Source Spectrum

Upper Energy (MeV)	Fraction of Neutrons in Group
15.0	4.49×10^{-4}
12.2	1.81×10^{-3}
10.0	5.55×10^{-3}
8.19	2.29×10^{-2}
6.36	6.17×10^{-2}
4.97	3.81×10^{-2}
4.07	1.20×10^{-1}
3.01	9.15×10^{-2}
2.47	2.35×10^{-2}
2.35	1.14×10^{-1}
1.83	1.95×10^{-1}
1.11	1.83×10^{-1}
5.50×10^{-1}	1.29×10^{-1}
1.11×10^{-1}	9.32×10^{-3}
3.35×10^{-3}	3.09×10^{-3}
5.83×10^{-4}	4.53×10^{-4}
1.01×10^{-4}	6.79×10^{-5}
2.90×10^{-5}	1.73×10^{-5}
1.07×10^{-5}	7.15×10^{-6}
3.06×10^{-6}	1.79×10^{-6}
1.13×10^{-6}	6.56×10^{-7}
4.14×10^{-7}	4.17×10^{-7}
4.14×10^{-7}	

sludge-supernate waste. The gamma ray spectrum extracted from these data are given in Table E-5 and the regrouped gamma ray source spectrum are given in Table E-6.

The neutron and gamma ray source strength were obtained from the same ORIGEN analysis. The specific neutron and gamma ray source strength was multiplied by the waste mass to obtain the absolute source strength necessary for normalization of the neutron and gamma ray radiation analysis. The resulting source strengths are given in Table E-7.

E.2.1.4 Dose Response Function

The package radiation analysis was performed with the Snyder-Neufeld Neutron flux-to-dose response function and the Henderson gamma ray response function. These response functions are given in Table E-8.

E.2.2 Package Radiation Analysis

E.2.2.1 Calculation Matrix

A number of calculations were performed in order to obtain the information required for the package radiation analysis. These calculations, identified as leakage calculations, consisted of a calculation with the neutron source and a separate calculation with the gamma ray source. In each of these calculations two primary results were obtained. The neutron and gamma ray current escaping from the waste package was recorded for subsequent use as the leakage source for the shield plug analysis. In addition the radiation dose averaged over the side and end surface of the waste container was obtained. The calculation setup for the neutron and gamma ray source was essentially the same except that an expectation boundary crossing estimator was used for the gamma ray source while an analog boundary crossing estimator was used for the neutron source.

Table E-5. Specified Gamma Ray Source Spectrum
Consolidated Spent Fuel

Mean Energy (MeV)	Gammas/sec/MTHM			
	Activ. Prod.	Actinides	Fiss. Prod.	Totals
0.015	5.645×10^{11}	4.016×10^{13}	1.851×10^{15}	1.892×10^{15}
0.025	3.137×10^{12}	1.587×10^{12}	4.055×10^{14}	4.102×10^{14}
0.0375	8.215×10^{11}	2.303×10^{11}	4.898×10^{14}	4.909×10^{14}
0.0575	4.751×10^{10}	2.315×10^{13}	3.560×10^{14}	3.792×10^{14}
0.085	2.044×10^{10}	6.128×10^{11}	2.220×10^{14}	2.226×10^{14}
0.125	2.797×10^{10}	5.313×10^{11}	2.083×10^{14}	2.089×10^{14}
0.225	2.809×10^{11}	4.139×10^{11}	1.839×10^{14}	1.846×10^{14}
0.375	1.660×10^{12}	2.728×10^{10}	8.873×10^{13}	9.042×10^{13}
0.575	2.134×10^{12}	1.107×10^9	3.260×10^{15}	3.262×10^{15}
0.85	3.174×10^9	1.133×10^9	2.605×10^{14}	2.605×10^{14}
1.25	7.194×10^{12}	4.697×10^8	1.043×10^{14}	1.115×10^{14}
1.75	6.392×10^7	1.053×10^8	3.051×10^{12}	3.051×10^{12}
2.25	3.812×10^7	3.509×10^7	7.250×10^{10}	7.257×10^{10}
2.75	1.179×10^5	3.554×10^8	4.425×10^9	4.781×10^9
3.5	2.784×10^{-7}	1.830×10^7	5.733×10^8	5.916×10^8
5.0	3.454×10^{-8}	7.826×10^6	6.063×10^{-5}	7.826×10^6
7.0	2.237×10^{-9}	9.022×10^5	3.934×10^{-6}	9.022×10^5
11.0	1.416×10^{-10}	1.036×10^5	2.488×10^{-7}	1.036×10^5
Totals	1.589×10^{13}	6.672×10^{13}	7.433×10^{15}	7.516×10^{15}

Table E-5. Specified Gamma Ray Source Spectrum
(Continued)

Defense High Level Waste

Mean Energy (MeV)	Gammas/sec/(lb glass)			Totals
	Activ. Prod.	Actinides	Fiss. Prod.	
0.015	1.070×10^8	1.677×10^9	3.522×10^{11}	3.540×10^{11}
0.025	1.687×10^7	2.814×10^6	7.743×10^{10}	7.745×10^{10}
0.375	9.304×10^6	4.936×10^6	8.461×10^{10}	8.462×10^{10}
0.575	1.031×10^7	4.102×10^7	6.758×10^{10}	6.763×10^{10}
0.850	4.048×10^6	1.030×10^6	4.377×10^{10}	4.378×10^{10}
0.125	1.555×10^6	7.695×10^4	4.086×10^{10}	4.086×10^{10}
0.225	5.112×10^5	1.058×10^5	3.555×10^{10}	3.555×10^{10}
0.375	1.434×10^5	7.937×10^3	1.863×10^{10}	1.863×10^{10}
0.575	8.235×10^3	1.249×10^4	3.924×10^{11}	3.924×10^{11}
0.85	1.303×10^5	6.873×10^3	9.696×10^9	9.696×10^9
1.25	3.489×10^9	2.643×10^2	5.601×10^9	9.090×10^9
1.75	0.0	3.373×10^1	3.732×10^8	3.732×10^8
2.25	1.849×10^4	1.873×10^1	8.010×10^8	8.010×10^8
2.75	5.723×10^1	1.019×10^4	6.122×10^6	6.132×10^6
3.50	0.0	8.945	7.107×10^5	7.107×10^5
5.0	0.0	3.596	6.094×10^{-9}	3.596
7.0	0.0	3.855×10^{-1}	3.954×10^{-10}	3.855×10^{-1}
11.0	0.0	4.245×10^{-2}	2.501×10^{-11}	4.245×10^{-2}
Totals	3.639×10^9	1.727×10^9	1.130×10^{12}	1.135×10^{12}

Table E-6. Regrouped Gamma Ray Source Spectrum

Group #	Energy (MeV)	Group Fraction (gammas/group)	
		DHLW	CSF
1	10.00	0.00000	0.00000
2	8.00	0.00000	0.00000
3	6.50	0.00000	0.00000
4	5.00	0.00000	0.00000
5	4.00	0.00000	0.00000
6	3.00	0.00000	0.00000
7	2.50	0.00000	0.00001
8	2.00	0.00002	0.00008
9	1.66	0.00007	0.00027
10	1.33	0.00231	0.00933
11	1.00	0.00210	0.01186
12	0.80	0.00291	0.02521
13	0.60	0.31309	0.38290
14	0.40	0.05174	0.06027
15	0.30	0.01512	0.01216
16	0.20	0.04027	0.03579
17	0.10	0.07109	0.05896
18	0.05	0.50125	0.40312
Totals		0.99997	0.99995

Table E-7. Calculation of Neutron and Gamma Ray Source Strength

DHLW		CSF		
		6 Elements	9 Elements	12 Elements
Net Glass Weight (lb)	3256			
(kg)	1480			
Net Waste Weight				
	(MTHM)	2.78	4.17	5.55
Gammas/sec/(lb glass)	1.14×10^{12}	7.52×10^{15}	7.52×10^{15}	7.52×10^{15}
Gammas/sec	3.71×10^{15}	2.09×10^{16}	3.14×10^{16}	4.17×10^{16}
Neutrons/sec/(lb glass)	3.31×10^2	1.83×10^8	1.83×10^8	1.83×10^8
Neutrons/sec	1.08×10^6	5.09×10^8	7.63×10^8	1.02×10^9

Table E-8. Neutron and Gamma Ray Response Functions

Group No.	Neutron Energy (eV)	Neutron Response (rem/neutron/cm ²)	Gamma Ray Energy (eV)	Gamma Ray Response (rem/gamma/cm ²)
1	1.50 x 10 ⁷	7.00 x 10 ⁻⁹	1.00 x 10 ⁷	2.42 x 10 ⁻⁹
2	1.22 x 10 ⁷	7.00 x 10 ⁻⁹	8.00 x 10 ⁶	2.07 x 10 ⁻⁹
3	1.00 x 10 ⁷	7.08 x 10 ⁻⁹	6.50 x 10 ⁶	1.76 x 10 ⁻⁹
4	8.18 x 10 ⁶	6.72 x 10 ⁻⁹	5.00 x 10 ⁶	1.59 x 10 ⁻⁹
5	6.36 x 10 ⁶	6.03 x 10 ⁻⁹	4.00 x 10 ⁶	1.27 x 10 ⁻⁹
6	4.96 x 10 ⁶	5.43 x 10 ⁻⁹	3.00 x 10 ⁶	1.08 x 10 ⁻⁹
7	4.06 x 10 ⁶	4.83 x 10 ⁻⁹	2.50 x 10 ⁶	8.75 x 10 ⁻¹⁰
8	3.01 x 10 ⁶	4.48 x 10 ⁻⁹	2.00 x 10 ⁶	7.35 x 10 ⁻¹⁰
9	2.46 x 10 ⁶	4.33 x 10 ⁻⁹	1.66 x 10 ⁶	6.44 x 10 ⁻¹⁰
10	2.35 x 10 ⁶	4.23 x 10 ⁻⁹	1.33 x 10 ⁶	5.30 x 10 ⁻¹⁰
11	1.82 x 10 ⁶	3.96 x 10 ⁻⁹	1.00 x 10 ⁶	4.45 x 10 ⁻¹⁰
12	1.11 x 10 ⁶	3.30 x 10 ⁻⁹	0.80 x 10 ⁶	3.50 x 10 ⁻¹⁰
13	5.50 x 10 ⁵	1.73 x 10 ⁻⁹	0.60 x 10 ⁶	2.56 x 10 ⁻¹⁰
14	1.11 x 10 ⁵	7.00 x 10 ⁻¹⁰	0.40 x 10 ⁶	1.77 x 10 ⁻¹⁰
15	3.35 x 10 ³	6.07 x 10 ⁻¹⁰	0.30 x 10 ⁶	1.22 x 10 ⁻¹⁰
16	5.83 x 10 ²	6.72 x 10 ⁻¹⁰	0.20 x 10 ⁶	6.60 x 10 ⁻¹¹
17	1.01 x 10 ²	5.35 x 10 ⁻¹⁰	0.10 x 10 ⁶	3.90 x 10 ⁻¹¹
18	2.90 x 10 ¹	3.88 x 10 ⁻¹⁰	0.05 x 10 ⁶	8.37 x 10 ⁻¹¹
19	1.07 x 10 ¹	3.42 x 10 ⁻¹⁰	0.02 x 10 ⁶	8.37 x 10 ⁻¹¹
20	3.059	3.27 x 10 ⁻¹⁰		
21	1.125	3.22 x 10 ⁻¹⁰		
22	0.414	3.20 x 10 ⁻¹⁰		
	0.100			

E.2.2.2 Results of Package Radiation Analysis

The results of the package radiation analysis are given in Table E-9. The dose as a function of container thickness is given in Figures E-10 through E-13.

E.2.3 Shield Plug Analysis

E.2.3.1 Calculation Approach

The MORSE code was used to calculate the dose at several positions in the horizontal emplacement. The calculation was coupled to the package radiation analysis by using the package leakage current as the radiation source for the emplacement radiation analysis. The calculations were performed using a flux at a point estimator.

E.2.3.2 Geometrical Modeling

The geometry model used in the emplacement radiation analysis was identical to the model used in the nominal package radiation analysis. The same geometry model was used to eliminate as many opportunities for error or inconsistency as possible. In the package radiation analysis, all regions outside of the waste package were specified as void regions; while in the emplacement analysis, the appropriate cross section media were specified.

E.2.3.3 Material Compositions

The material compositions are the same as used in the package radiation analysis and with the same nuclide densities given in Table E-1.

Table E-9. Results of Package Radiation Analysis - Horizontal Concept

Container/ Surface	Neutron	Surface Dose (mRem/hr) Gamma	Total
DHLW			
Top	$9\pm 2 \times 10^{-3}$	$5\pm 3 \times 10^{-1}$	$5\pm 3 \times 10^{-1}$
Bottom	$4\pm .5 \times 10^{-2}$	2 ± 1	2 ± 1
Side	$9\pm .5 \times 10^{-2}$	$2\pm .5 \times 10^3$	$2\pm .5 \times 10^3$
CSF-6			
Top	4 ± 1	$1\pm .5 \times 10^{-1}$	4 ± 1
Bottom	$1\pm .2 \times 10^1$	7 ± 4	$2\pm .5 \times 10^1$
Side	$4\pm .5 \times 10^1$	$4\pm 2 \times 10^3$	$4\pm 2 \times 10^3$
CSF-9			
Top	6 ± 2	$1\pm .5 \times 10^{-1}$	6 ± 2
Bottom	$2\pm .4 \times 10^1$	4 ± 2	$2\pm .4 \times 10^1$
Side	$4\pm .5 \times 10^1$	$7\pm 3 \times 10^3$	$7\pm 3 \times 10^3$
CSF-12			
Top	6 ± 2	$2\pm 1 \times 10^1$	$3\pm 1 \times 10^1$
Bottom	$1\pm .3 \times 10^1$	4 ± 2	$2\pm .4 \times 10^1$
Side	$5\pm .5 \times 10^1$	$7\pm 3 \times 10^3$	$7\pm 3 \times 10^3$

E-10

Figure 2-8: Dose vs Container Thickness - DHLW

E-11

Figure 2.9: Dose vs Container Thickness - CSF6

E-12

Figure 2-10: Dose vs Container Thickness - CSF9

E-13

Figure 2.11: Dose vs Container Thickness - CSF12

E.2.3.4 Radiation Source

The radiation source used for the emplacement radiation analysis was the leakage file from the shielding calculation of the four waste packages. Separate calculations were performed for the neutron and secondary gamma ray leakage and the primary gamma ray leakage for the waste packages. The leakage file written in the package radiation analysis calculations contained the state vector of each particle escaping from the waste package. The state vector included the particle position, its direction of travel and its energy. In the emplacement analysis, the source was selected from the appropriate leakage file and the state vector for each 'source' particle read from the leakage file. In order to improve the efficiency of the emplacement analysis, source region biasing was used for the selection of source particles from the leakage tapes. An appropriate biasing schema must come from a priori knowledge of the relative importance of source particles to the problem solution. One can argue that neutrons and gamma rays which escape from the waste packages on the front surface are more likely to contribute to the dose at the entrance to the emplacement borehole than those which escape from the back end. Similarly, it is apparent that those that escape from the upper half of the cylindrical surface of the waste package are relatively more likely to encounter the gap between the shield plug and the borehole than those that escape on the bottom surface. Following this reasoning, an importance was assigned to surface areas of the waste package as illustrated in Figure E-14. Although the biasing schema is a 'fair game' in the sense that total particle weight is conserved, because only a finite number of source particles are tracked, biasing which is too severe can distort the results if an adequate sampling of the source distribution is not obtained. In order to select the biasing factors a series of test calculations were performed to establish biasing schema which would reduce the variance without distorting the results. The source selection algorithm was designed to reject initial particles with the bias factors shown in Figure E-14. The statistical weight of particles which survived the source region biasing were then corrected to account for the survival probability, thus preserving the total leakage probability.

NOTE: ALL LEAKAGE TOWARD TUNNEL FROM CYLINDRICAL SURFACE AS SHOWN; AWAY FROM TUNNEL DIVIDED BY X. FOR ALL ANALYSES, $X = 4$.

Figure E-14. Source Region Importance for Emplacement Radiation Analysis

E.2.3.5 Dose Response Function

The emplacement radiation analysis ~~was~~ used the same response functions as the package radiation analysis. These response functions are tabulated in Table E-8.

E.2.3.6 Results of Emplacement Radiation Analysis

Four sets of calculations were performed for each waste form consisting of cases for the neutron and gamma sources and for high and low salt plug density. Dose levels were calculated at seven locations in the emplacement, as indicated in Figure E-15. The results of the emplacement radiation analysis are given in Table E-10.

An initial examination of Table E-10 shows some unexpected and even 'counter-intuitive' results. Perhaps most surprising is the observation that the dose levels for the CSF-6 container are greater than for the CSF-9 container even though the source strength is lower by 50%. Additional calculations were performed in order to understand or resolve this apparently

E-15

Figure 2-18: Location of Dose Points in Emplacement Radiation Analysis

Table E-10. Results of Emplacement Radiation Analysis - Horizontal Concept

Waste Form/ Dose Point	Dose - Low Density Plug (mRem/hr)			Waste Form/ Dose Point	Dose - High Density Plug (mRem/hr)		
	Neutron	Gamma	Total		Neutron	Gamma	Total
DHLW				DHLW			
#1	$7+2 \times 10^{-3}$	$4+1.5$	$4+1.5$	#1	$5+2 \times 10^{-3}$	$4+1.5$	$4+1.5$
#2	$7+2 \times 10^{-4}$	$2+1.2 \times 10^{-4}$	$9+2 \times 10^{-4}$	#2	$4+1.2 \times 10^{-4}$	$4+2 \times 10^{-4}$	$7+3 \times 10^{-4}$
#3	$1+5 \times 10^{-5}$		$1+5 \times 10^{-5}$	#3	$1+5 \times 10^{-5}$		$1+5 \times 10^{-5}$
#4	$8+4 \times 10^{-7}$		$8+4 \times 10^{-7}$	#4	$6+3 \times 10^{-7}$		$6+3 \times 10^{-7}$
#5	$1+5 \times 10^{-6}$		$1+5 \times 10^{-6}$	#5	$4+2 \times 10^{-6}$		$4+2 \times 10^{-6}$
#6	$1+5 \times 10^{-6}$		$1+5 \times 10^{-6}$	#6	$1+5 \times 10^{-6}$		$1+5 \times 10^{-6}$
#7	$2+1 \times 10^{-7}$		$2+1 \times 10^{-7}$	#7	$2+1 \times 10^{-7}$		$2+1 \times 10^{-7}$
CSF-6				CSF-6			
#1	$2+4 \times 10^1$	$7+3 \times 10^1$	$9+4 \times 10^1$	#1	$1+2 \times 10^1$	$8+4 \times 10^1$	$9+4 \times 10^1$
#2	$7+2$	$3+1 \times 10^{-1}$	$7+2$	#2	$2+6$	$2+1 \times 10^{-1}$	$2+6$
#3	$4+2 \times 10^{-1}$		$4+2 \times 10^{-1}$	#3	$1+5 \times 10^{-1}$	$1+5 \times 10^{-4}$	$1+5 \times 10^{-1}$
#4	$3+1.5 \times 10^{-2}$		$3+1.5 \times 10^{-2}$	#4	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$
#5	$6+3 \times 10^{-2}$		$6+3 \times 10^{-2}$	#5	$4+2 \times 10^{-2}$		$4+2 \times 10^{-2}$
#6	$7+3.5 \times 10^{-2}$		$7+3.5 \times 10^{-2}$	#6	$4+2 \times 10^{-2}$		$4+2 \times 10^{-2}$
#7	$3+1.5 \times 10^{-2}$		$3+1.5 \times 10^{-2}$	#7	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$
CSF-9				CSF-9			
#1	$1+2 \times 10^1$	$1+4 \times 10^2$	$1+4 \times 10^2$	#1	$1+2 \times 10^1$	$1+2 \times 10^2$	$1+2 \times 10^2$
#2	$2+1$	$1+5$	$3+1.5$	#2	$5+2.5$	$3+1.5 \times 10^{-1}$	$6+3$
#3	$1+4 \times 10^{-1}$		$1+4 \times 10^{-1}$	#3	$2+1 \times 10^{-1}$		$2+1 \times 10^{-1}$
#4	$3+1.5 \times 10^{-2}$		$3+1.5 \times 10^{-2}$	#4	$3+1.5 \times 10^{-2}$		$3+1.5 \times 10^{-2}$
#5	$5+2.5 \times 10^{-2}$		$5+2.5 \times 10^{-2}$	#5	$4+2 \times 10^{-2}$		$4+2 \times 10^{-2}$
#6	$6+3 \times 10^{-2}$		$6+3 \times 10^{-2}$	#6	$5+2.5 \times 10^{-2}$		$5+2.5 \times 10^{-2}$
#7	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$	#7	$1+5 \times 10^{-2}$		$1+5 \times 10^{-2}$
CSF-12				CSF-12			
#1	$2+4 \times 10^1$	$6+3 \times 10^3$	$6+3 \times 10^3$	#1	$2+4 \times 10^1$	$6+3 \times 10^3$	$6+3 \times 10^3$
#2	$1+2 \times 10^1$	$2+8 \times 10^{-1}$	$1+2 \times 10^1$	#2	$6+2$	$2+1 \times 10^{-1}$	$6+2$
#3	$9+3 \times 10^{-1}$		$9+3 \times 10^{-1}$	#3	$1+3 \times 10^{-1}$		$1+3 \times 10^{-1}$
#4	$1+5 \times 10^{-1}$		$1+5 \times 10^{-1}$	#4	$1+5 \times 10^{-2}$		$1+5 \times 10^{-2}$
#5	$1+5 \times 10^{-1}$		$1+5 \times 10^{-1}$	#5	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$
#6	$1+5 \times 10^{-1}$		$1+5 \times 10^{-1}$	#6	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$
#7	$3+1.5 \times 10^{-2}$		$3+1.5 \times 10^{-2}$	#7	$1+5 \times 10^{-2}$		$1+5 \times 10^{-2}$

anomalous result. Although a complete quantitative understanding of this result was not obtained, a sufficient understanding was obtained to suggest the validity of the results.

There appear to be three main factors which influence the relative dose levels in the emplacement for the consolidated spent fuel waste forms:

- Neutron and gamma ray source strength,
- Radiation streaming in the gap between the borehole and the waste container, and
- The angular distribution of the neutron and gamma ray leakage from the waste container surface.

The source strength (neutrons and gamma rays per second) is increasing with waste form volume. Due to self shielding in the waste form and attenuation in the container case, the leakage from the waste container would be expected to increase approximately with the radius of the waste form.

Since the total leakage from the side (cylindrical surface) of the waste container is so much larger than the top or bottom surfaces (see Table E-9) one would expect that the size of the eccentric gap might effect the dose levels in the emplacement. Such 'radiation streaming' effects have been observed in many other problems with large void regions. In order to investigate the magnitude of this effect, the CSF-6 case was recalculated with a reduced gap size equal to the gap size for the CSF-12 problem. With this reduced gap size the results were observed to be reduced by approximately a factor of two, indicating that the gap size has a significant effect on the emplacement radiation dose levels.

A third factor is the angular distribution of the radiation leakage from the waste container. Because the containers are relatively thick, the radiation leakage is strongly forward peaked relative to the normal to the container surface. By looking at the direction of particles escaping from the

leakage file, it was observed that the total leakage was proportional approximately to $\cos(\theta)^3$, where θ is the cosine of the angle from the normal to the surface. An examination of the geometry of the dose point locations relative to the CSF-6, CSF-9, and CSF-12 waste containers suggests that this is certainly a factor to be considered in understanding the relative emplacement radiation dose levels for the CSF-6, CSF-9, and CSF-12 cases.

Although a complete and quantitative explanation of the results reported in Table E-10 was not possible within the scope of the present calculations, a careful reexamination of the results and calculations revealed no reason to believe that the reported results are erroneous.

E.3 VERTICAL EMPLACEMENT ANALYSIS

A series of calculations similar to those performed for the horizontal emplacement concept were also completed for the vertical concept illustrated in Figure E-2. The sequence of calculations was identical to that described for the horizontal emplacement analysis:

- Calculations of the leakage current on the top, bottom, and side surfaces of the four waste packages, and
- Subsequent transport calculations in the emplacement using the surface leakage source from the package calculations.

E.3.1 Package Radiation Analysis

The package radiation calculations were identical to the horizontal emplacement analysis in all aspects except the geometrical configuration. The same material composition and cross sections, the same source terms, and the same response functions were used.

E.3.1.1 Geometrical Modeling

The waste package geometry model was changed from that of the horizontal configuration to reflect a slightly different configuration. For the vertical concept the waste form and canister were assumed to be concentric. Figures E-16 through E-19 show the dimensions of the waste packages as modeled for the package radiation leakage calculations. The required changes to the combinatorial geometry model were straightforward and consisted of relocating the central axis of several of the cylindrical bodies.

E.3.1.2 Results of Package Radiation Analysis

The results of the package radiation calculations are given in Table E-11. The results are given in mrem/hr for the top, bottom, and side surfaces of the waste packages. The results are normalized to the source strength of the respective waste forms given in Table E-7.

E.3.2 Backfill Salt Shield Analysis

The shielding analysis of the vertical concept backfill shield was slightly different than that performed for the horizontal concept. The salt backfill was assumed to completely fill the diameter of the emplacement borehole (as compared to the eccentric gap left between the plug and the borehole in the horizontal concept). The salt backfill was assumed to fill the borehole down to the top of the waste canister but not to fill in the space between the canister and the borehole wall. Rather than calculating the dose for a specified thickness of the salt backfill, it was desired to determine the thickness of the plug required to give a dose rate of 0.5 mrem/hr in the emplacement drift. Therefore, calculations were performed to give dose rates for salt backfill thickness of 50, 100, and 150 cm. thickness which were expected to bracket the required thickness to give 0.5 mrem/hr. The required thickness can then be estimated by interpolation.

E-16

Figure 3-1: DHLW Waste Package - Vertical Configuration

E-17
Figure 3.2: CSF-6 Waste Package - Vertical Configuration

E-18
 Figure 2-8: CSF-9 Waste Package - Vertical Configuration

E-19
 Figure 3.4. CSF-12 Waste Package - Vertical Configuration

Table E-11. Results of Package Radiation Analysis - Vertical Concept

Container/ Surface	Neutron	Surface Dose (mRem/hr)	
		Gamma	Total
DHLW			
Top	$9+2 \times 10^{-3}$	$5+3 \times 10^{-1}$	$5+3 \times 10^{-1}$
Bottom	$4+5 \times 10^{-2}$	$2+1$	$2+1$
Side	$9+5 \times 10^{-2}$	$2+5 \times 10^3$	$2+5 \times 10^3$
CSF-6			
Top	$3+1.5$	$1+5 \times 10^{-2}$	$3+1.5$
Bottom	$8+2$	$7+4$	$2+5 \times 10^1$
Side	$4+5 \times 10^1$	$4+2 \times 10^3$	$4+2 \times 10^3$
CSF-9			
Top	$6+2$	$1+5 \times 10^{-1}$	$7+2$
Bottom	$1+2 \times 10^1$	$4+2$	$2+4 \times 10^1$
Side	$4+5 \times 10^1$	$7+3 \times 10^3$	$7+3 \times 10^3$
CSF-12			
Top	$2+5$	$2+1 \times 10^1$	$2+1 \times 10^1$
Bottom	$2+4 \times 10^1$	$4+2$	$3+5 \times 10^1$
Side	$5+5 \times 10^1$	$7+3 \times 10^3$	$7+3 \times 10^3$

The calculational procedure, material compositions, cross sections, radiation source, and response function were the same as that used in the horizontal emplacement calculations.

E.3.2.1 Results of Emplacement Radiation Analysis - Vertical Concept

The results of the emplacement calculations are tabulated in Table E-12 and are plotted in Figures E-20 through E-27. The results in Table E-12 are given for the neutron and gamma ray source and for high and low density shield plug thickness of 50, 100, and 150 cm. (The 0 thickness results are obtained from the top surface results of the waste package calculations.) The results are given in mrem/hr and have been normalized to the neutron and gamma ray source strength of the several waste forms given in Table E-7.

An examination of the results indicates some unexpected and perhaps 'counter-intuitive' results. For the CSF-6 container, the dose level is not decreasing from the 50 cm thickness to the 100 cm thickness as rapidly as might be expected. A more careful examination of the results indicates that the statistical uncertainty in the results is 'distorting', in fact accentuating, a real effect. The explanation is analogous to the situation for the CSF-6 container observed in the horizontal emplacement calculations. For the vertical emplacement, the larger gap for the CSF-6 container is an important factor in explaining the observed results. The dose points reported in Table E-12 are for points along the axis of the cylindrical waste container. In order for the dose levels for 50 cm and 100 cm backfill to be approximately the same, it would be necessary for radiation streaming in the concentric cavity to be an important effect. In order to verify this explanation, an additional calculation was performed for 50 cm backfill with the dose point off-axis at a radius over the concentric gap. The dose level over the gap was observed to be a factor of ten greater than the dose level on axis (both dose points at 50 cm backfill). Thus as in the horizontal emplacement, the results for the CSF-6 waste container are suspected to be complicated by the competing effects of source strength, gap size, and angular distribution of the source.

Table E-12. Results of Emplacement Radiation Analysis - Vertical Concept

Waste Form/ Backfill Thickness		Dose - Low Density Backfill (mRem/hr)		
		Neutron	Gamma	Total
DHLW				
	50 cm	$2+4 \times 10^{-4}$	$3+1.5 \times 10^6$	$2+4 \times 10^{-4}$
	100 cm	$3+6 \times 10^{-5}$	$2+1 \times 10^{-9}$	$3+6 \times 10^{-5}$
	150 cm	$2+1 \times 10^{-5}$	$4+2 \times 10^{-12}$	$2+1 \times 10^{-5}$
CSF-6				
	50 cm	$2+1 \times 10^{-1}$		$2+1 \times 10^{-1}$
	100 cm	$6+3 \times 10^{-2}$		$6+3 \times 10^{-2}$
	150 cm	$2+1 \times 10^{-2}$		$2+1 \times 10^{-2}$
CSF-9				
	50 cm	$5+2.5 \times 10^{-1}$		$5+2.5 \times 10^{-1}$
	100 cm	$3+1 \times 10^{-2}$		$3+1 \times 10^{-2}$
	150 cm	$2+5 \times 10^{-3}$		$2+5 \times 10^{-3}$
CSF-12				
	50 cm	$3+1 \times 10^{-1}$		$3+1 \times 10^{-1}$
	100 cm	$5+1.5 \times 10^{-2}$		$5+1.5 \times 10^{-2}$
	150 cm	$9+4 \times 10^{-3}$		$9+4 \times 10^{-3}$

Table E-12. Results of Emplacement Radiation Analysis - Vertical Concept
(Continued)

Waste Form/ Backfill Thickness		Dose - High Density Backfill (mRem/hr)		Total
		Neutron	Gamma	
DHLW				
	50 cm	$7+3.5 \times 10^{-4}$	$4+2 \times 10^{-8}$	$7+3.5 \times 10^{-4}$
	100 cm	$2+1 \times 10^{-5}$	$6+3 \times 10^{-12}$	$2+1 \times 10^{-5}$
	150 cm	$6+3 \times 10^{-7}$	$2+1 \times 10^{-15}$	$6+3 \times 10^{-7}$
CSF-6				
	50 cm	$2+.5 \times 10^{-1}$		$2+1 \times 10^{-1}$
	100 cm	$3+1.5 \times 10^{-1}$		$3+1.5 \times 10^{-1}$
	150 cm	$1+.5 \times 10^{-1}$		$1+.5 \times 10^{-1}$
CSF-9				
	50 cm	$3+1 \times 10^{-1}$		$3+1 \times 10^{-1}$
	100 cm	$2+.6 \times 10^{-2}$		$2+.6 \times 10^{-2}$
	150 cm	$8+2.5 \times 10^{-4}$		$8+2.5 \times 10^{-4}$
CSF-12				
	50 cm	$2+.8 \times 10^{-1}$		$2+.8 \times 10^{-1}$
	100 cm	$1+.5 \times 10^{-2}$		$1+.5 \times 10^{-2}$
	150 cm	$1+.5 \times 10^{-3}$		$1+.5 \times 10^{-3}$

Note: 0 thickness values are averaged over the top surface of the container, 50, 100, 150 cm values are point values on the axis of the waste container.

E-20

Figure 3-5: Dose vs Shield Plug Thickness, DHLW, High Plug Density

Note: 0 thickness values are averaged over the top surface of the container, 50, 100, 150 cm values are point values on the axis of the waste container.

E-2/

Figure 3-6: Dose vs Shield Plug Thickness, DHLW, Low Plug Density

Note: 0 thickness values are averaged over the top surface of the container, 50, 100, 150 cm values are point values on the axis of the waste container.

E-22

Figure 3.7: Dose vs Shield Plug Thickness, CSF-6, High Plug Density

Note: 0 thickness values are averaged over the top surface of the container, 50, 100, 150 cm values are point values on the axis of the waste container.

E-23

Figure 2-8: Dose vs Shield Plug Thickness, CSF-6, Low Plug Density

E-24

Figure 3-9: Dose vs Shield Plug Thickness, CSF-9, High Plug Density

E-25

Figure 2.10: Dose vs Shield Plug Thickness, CSF-9, Low Plug Density

E-26

Figure 2-11: Dose vs Shield Plug Thickness, CSF-12, High Plug Density

E-27
Figure 3.12: Dose vs Shield Plug Thickness, CSF-12, Low Plug Density

The results are believed to be valid, but there is not sufficient information to be extracted from the calculations to be able to provide a detailed explanation of the radiation transport characteristics of the emplacement.

E.4 REFERENCES

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