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An AP-2.14Q review of this document was completed on March 12, 2004.

This design report addresses the current event sequences as determined by Preclosure Safety. The Naval Nuclear Propulsion Program (NNPP) Resident Manager has advised that new event sequences are currently being developed.

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00A	This design analysis supersedes and revises the previous document ANL-VDC-ME-000001 REV 00 to report the results of the finite element analyses and evaluations of the latest naval waste package. All pages are affected. Change bars are not used because the changes are extensive.	98	98	M.M. Lewis 03/15/04	D.G. McKenzie IV 15MAR2004	D.J. Tunney 3/15/2004	M.J. Anderson	3/15/04

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1. PURPOSE

A design methodology for the waste packages and ancillary components, viz., the emplacement pallets and drip shields, has been developed to provide designs that satisfy the safety and operational requirements of the Yucca Mountain Project. This methodology is described in the *Waste Package Design Methodology Report* (Mecham 2004 [DIRS 166168]. To demonstrate the practicability of this design methodology, four waste package design configurations have been selected to illustrate the application of the methodology. These four design configurations are the 21-pressurized water reactor (PWR) Absorber Plate waste package, the 44-boiling water reactor (BWR) waste package, the 5-defense high-level waste (DHLW)/United States (U.S.) Department of Energy (DOE) spent nuclear fuel (SNF) Co-disposal Short waste package, and the Naval Canistered SNF Long waste package. Also included in this demonstration is the emplacement pallet and continuous drip shield.

The purpose of this report is to document how that design methodology has been applied to the waste package design configurations intended to accommodate naval canistered SNF. This demonstrates that the design methodology can be applied successfully to this waste package design configuration and support the License Application for construction of the repository. In this document, the results of design calculations are summarized and used to show that the design configurations are in compliance with the applicable criteria in *Naval Spent Nuclear Fuel Waste Package System Description Document* (BSC 2003 [DIRS 165427]) and *Project Design Criteria Document* (Minwalla 2003 [DIRS 161362]).

2. QUALITY ASSURANCE

The naval waste packages are classified as Safety Category items (BSC 2003 [DIRS 165179], Table A-2, p. A-4). Therefore, this document is subject to the requirements of *Quality Assurance Requirements and Description* (DOE 2003 [DIRS 162903]). This document was developed in accordance with AP-3.12Q, *Design Calculations and Analyses*.

3. USE OF SOFTWARE

No computer software or models were used in the generation of this report. Contributory calculations provide descriptions of software used.

4. DESIGN INPUTS AND ASSUMPTIONS

Generic design inputs and assumptions that are used in contributory calculations to this report may be found in *Waste Package Component Design Methodology Report* (Mecham 2004 [DIRS 166168], Sections 4 and 5). Specific design inputs and assumptions may be found in the supporting calculations.

5. GENERAL DESCRIPTION

5.1 GENERAL CONFIGURATION

Section 114(a)(1)(B) of the Nuclear Waste Policy Act of 1982 (NWPA), as amended (42 U.S.C. 10134(a)(1)(B)) (NWPA 1982 [DIRS 101681]), requires "a description of the waste form or packaging proposed for use at such repository, and an explanation of the relationship between such waste form or packaging and the geologic medium of the site." This section describes the waste forms to be disposed, along with their packaging. An explanation of the important parameters considered in the design of the waste package is included in this section, as is a summary of the expected performance of the waste package design. This section:

- Presents an overview of the waste forms and the waste package design configurations
- Describes the waste package, its design bases, and its functions
- Discusses in detail the waste forms, the parameters considered in designing the waste package (and its variations), and the evaluations performed on the designs
- Describes the material selection and fabrication of the waste package
- Presents the results of design evaluations of the waste package.

Waste Form Overview—Waste forms to be received and packaged for disposal include spent nuclear fuel from commercial power reactors, spent nuclear fuel owned by the DOE (including naval fuel), and canisters of solidified high-level radioactive waste from prior commercial and defense fuel reprocessing operations.

Section 114(d) of the NWPA (42 U.S.C. 10134(d)) (NWPA 1982 [DIRS 101681]) limits the first repository's capacity to no more than 70,000 Metric Tons Heavy Metal (MTHM) "...until such time as a second repository is in operation." The types of waste that will be accepted at the repository have been allocated as follows (DOE 2002 [DIRS 155970], Chapter 2):

- 63,000 MTHM of commercial spent nuclear fuel
- 7,000 MTHM of DOE high-level radioactive waste, commercial high-level radioactive waste, and DOE spent nuclear fuel.

The waste forms received at a repository will be in solid form. Materials that could ignite or react chemically at a level that would compromise containment or isolation will not be accepted by the repository. Neither the waste forms nor the waste packages will contain free liquids that could compromise waste containment. Materials that are regulated as hazardous waste under the Resource Conservation and Recovery Act of 1976 (42 U.S.C. 6901 et seq. [DIRS 103936]) will not be disposed in the repository.

Waste Package Overview—The design of a waste package is based on the characteristics of the waste forms that it will hold. Because commercial and DOE high-level radioactive waste forms have similar characteristics, both may be placed into a waste package of the same design. This has allowed the DOE to design waste packages capable of accommodating all the types of spent nuclear fuel and high-level radioactive waste currently generated or anticipated in the United States, whether commercial or governmental.

The waste package has been designed, in conjunction with the natural and other engineered barriers, to ensure compliance with applicable U.S. Nuclear Regulatory Commission (NRC) regulations, to contribute to safe operations during the preclosure phase, to make efficient use of the repository area, and to preserve the option of retrieving the waste. To perform its containment and isolation functions, the waste package described in this report has been designed to take advantage of a location in the unsaturated zone.

All the waste package design configurations consist of two concentric cylinders in which the waste forms will be placed. The inner cylinder will be composed of stainless steel type 316 (Unified Numbering System [UNS] S31600). The outer cylinder will be made of a corrosion-resistant nickel-based alloy, Alloy 22 (UNS N06022). The waste package design configurations for naval spent nuclear fuel are larger in diameter and thicker than those for commercial spent nuclear fuel. The outer layer of corrosion-resistant material protects the underlying layer of structural material from corrosion, and the structural material supports the thinner material of the outer layer.

Each waste package design configuration has outer and inner lids. The outer (closure) lids will be made of Alloy 22 (UNS N06022). The inner lids will be made of stainless steel type 316 (UNS S31600), and their thickness will vary, depending on the waste package design configuration. The final closure weld of the Alloy 22 (UNS N06022) outer lid undergoes stress-mitigation to inhibit stress corrosion cracking. Since this mitigation cannot easily be performed through the entire thickness of the outer lid, an Alloy 22 (UNS N06022) lid on the closure end of the waste package (middle lid) provides additional protection against stress corrosion cracking in the closure weld area.

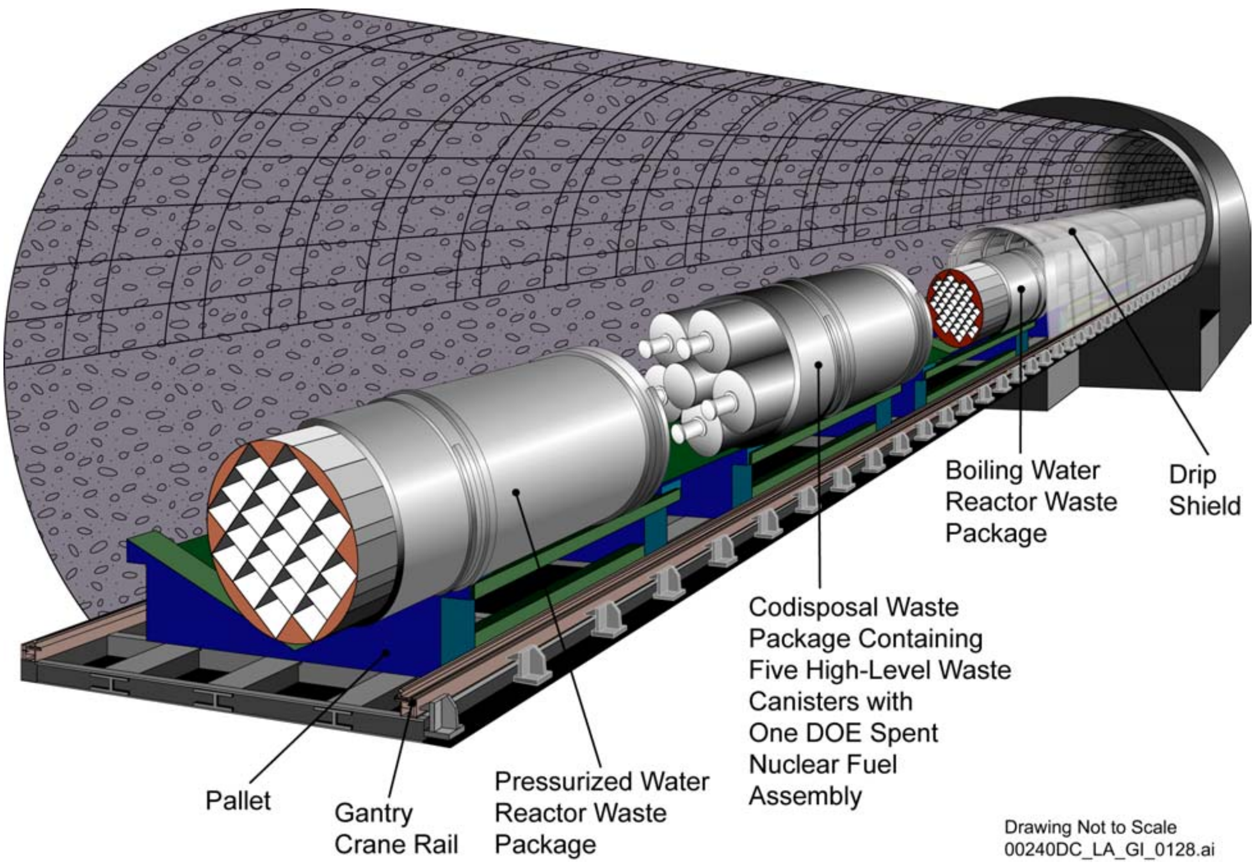
Before the double-walled waste package is sealed, helium will be added as a fill gas. The helium will prevent oxidation of the waste form (e.g., naval SNF canister, uncanistered commercial fuel, etc.) and help transfer heat from the waste form to the wall of the inner vessel of the waste package. Transferring heat away from the waste form is an important means of controlling waste form temperatures and preserving the integrity of the metal cladding on the fuel elements, thus extending the life of an existing barrier to water infiltration.

All waste package design configurations will use a remote lifting-and-handling mechanism. The collar-sleeve-and-trunnion joint apparatus will allow the necessary handling of the waste package before it is placed on an emplacement pallet and transferred to the designated drift. Each waste package will also have a unique permanent label (BSC 2003 [DIRS 166611], Section 6.4.2).

Although they share the features described previously, the waste package design configurations have different internal components to accommodate the different waste forms. For example, the waste package for uncanistered commercial spent nuclear fuel has an internal basket assembly to support fuel assemblies. In other waste packages (e.g., the high-level radioactive waste and DOE spent nuclear fuel waste packages), the internal basket has a different design, or, as is the case with naval spent nuclear fuel, the basket is contained inside the canister.

5.1.1 General Design Basis for the Waste Package

The waste isolation system is an important element of a repository. The primary component of the system is the waste package. As defined in 10 CFR 63.2 (66 FR 55732 [DIRS 156671], a waste package includes the waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding it. The invert material does not immediately surround the waste package, so it is not considered part of the waste package. Figure 1 illustrates the waste package within the emplacement drift of the waste isolation system.



Source: Mecham 2004 [DIRS 166168], Figure 2

Figure 1. Schematic Illustration of the Emplacement Drift with Cutaway Views of Different Waste Packages

The waste package has been designed to use materials that perform well under the anticipated conditions at Yucca Mountain. The design analyses performed on the waste package include evaluations of structural integrity, thermal performance, criticality safety, corrosion allowance, and shielding properties.

5.1.2 Preclosure Design Performance Specifications

The performance specifications for the functionality of the waste package during the repository's preclosure phase are consistent with 10 CFR 63.112(b) (10 CFR 63 [DIRS 158535]). This regulation provides for the DOE's analysis of the ability of the waste package's structures, systems, and components to perform their intended safety functions during an accident or event sequences. For the waste package, event sequences are determined by identifying the functions of the waste package and evaluating the effects on its performance of given events that could occur during normal handling of the waste package or during a credible accident scenario (i.e., events that have at least 1 chance in 10,000 of occurring before permanent closure of the repository) 10 CFR 63.2 (66 FR 55732 [DIRS 156671]).

These event sequences and their effects on performance were defined by reviewing the results of *Preliminary Categorization of Event Sequences for License Application* (BSC 2003 [DIRS 164128], Section 6) constituting a bounding list of preclosure event sequences that could affect the waste packages. Using this list, engineers performed structural, thermal, and criticality analyses of the impacts such events could have on waste package performance.

5.1.3 Postclosure Performance Specification

10 CFR 63.113(b) (10 CFR 63 [DIRS 158535]) requires the entire repository system to meet specific dose limits for 10,000 years. The waste package is one of many barriers relied upon to meet this limit. The DOE's objective is to design a waste package that works in concert with the natural environment to meet performance standards while reducing the uncertainty associated with the current understanding of natural processes at the site.

5.1.4 Design Descriptions

An analysis was undertaken to determine the number of design configurations needed to handle the different waste forms that will constitute the anticipated waste stream in the most economical manner (CRWMS M&O 1997 [DIRS 100224]). The objective of the evaluation was to determine:

- The number of different waste package design configurations needed
- The capacity of each waste package design configuration (i.e., the amount of waste it will hold)
- The limits on spent nuclear fuel properties (e.g., age, thermal characteristics) that might apply to each waste package design configuration.

The complete system of waste package design configurations is intended to allow reliable disposal of those waste forms that a repository will accept while still enhancing overall efficiencies.

To determine the most efficient set of waste package design configurations for commercial spent nuclear fuel, the DOE designed waste packages of various assembly-holding capacities and incorporated into the design methods for removing decay heat and preventing criticality. This resulted in the selection of a set of five waste package design configurations as the most efficient means of accommodating the anticipated waste stream of commercial spent nuclear fuel. A similar process led to three configurations for DOE non-naval spent nuclear fuel and DOE and commercial high-level radioactive waste. Two other configurations are specific to naval spent nuclear fuel, which will arrive presealed in canisters (Macheret 2001 [DIRS 154624], Sections

4.2 and 4.3). Some DOE non-naval spent nuclear fuel will be loaded into waste packages with high-level radioactive waste; this DOE spent nuclear fuel and high-level radioactive waste will also arrive in presealed canisters.

5.1.5 U.S. Department of Energy Spent Nuclear Fuel and High-level Radioactive Waste

Nine types of canisters of DOE spent nuclear fuel and high-level radioactive waste may be received at the repository (Macheret 2001 [DIRS 154624], Sections 4.2 and 4.3):

1. Naval spent nuclear fuel canisters, short
2. Naval spent nuclear fuel canisters, long
3. DOE spent nuclear fuel canisters, short
4. DOE spent nuclear fuel canisters, long
5. Larger-diameter DOE spent nuclear fuel canisters, short
6. Larger-diameter DOE spent nuclear fuel canisters, long
7. Solidified high-level radioactive waste canisters, short
8. Solidified high-level radioactive waste canisters, long
9. Multicanister overpacks containing spent nuclear fuel from the Hanford N Reactor.

The number of canisters of solidified high-level radioactive waste will greatly exceed the number of canisters of DOE spent nuclear fuel. Therefore, the DOE has developed an efficient arrangement for packing them together (Macheret 2001 [DIRS 154624], Section 4.2). This mixing of DOE spent nuclear fuel and high-level radioactive waste is called "co-disposal." Co-disposal also helps maintain criticality control for DOE spent nuclear fuel that contains highly enriched uranium. Naval spent nuclear fuel canisters, which are larger in diameter, will not be placed in co-disposal waste packages; they will be placed one canister per waste package.

5.1.6 U.S. Department of Energy Spent Nuclear Fuel

DOE spent nuclear fuel has a wide variety of physical, chemical, and nuclear characteristics and represents an inventory of approximately 2,500 MTHM; 2,333 MTHM of this is included in the waste allocation for disposal in the first repository (DOE 2002 [DIRS 158405], Section 8.1). The waste packages designed for DOE spent nuclear fuel will accept fuel irradiated at DOE facilities, naval spent nuclear fuel, and certain types of material irradiated at commercial nuclear reactors, including debris from the Three Mile Island-2 reactor and fuel from the Fort Saint Vrain reactor. All DOE waste canisters will be sealed before they are transported to the repository.

Naval nuclear fuel is designed to operate in a high-temperature, high-pressure environment for many years. Naval fuel is highly enriched. To ensure it can withstand battle-shock loads, naval fuel is surrounded by large amounts of structural material made of Zircaloy. There are two canister designs for naval fuel; both use similar materials and mechanical arrangements. The DOE plans to emplace about 65 MTHM of naval spent nuclear fuel in the repository. This fuel will be contained within about 300 sealed canisters, which will be transferred directly from transport casks into waste packages (DOE 2002 [DIRS 158405], Appendix C, Section 5.1, Table 1.

5.2 NAVAL WASTE PACKAGE CONFIGURATIONS

Naval spent nuclear fuel arrives at the repository in canisters suitable for long-term disposal. The canisters fit one to a waste package. Because the naval fuel arrives in canisters of two sizes (one short and one long), the DOE has developed two waste package design configurations for it. The larger of these two types is the heaviest and longest of all the waste packages design configurations. No additional features are necessary for structural support, heat transfer, and criticality control. The two waste package configurations and dimensions are provided by the following drawings:

- *Naval Long Waste Package Configuration* (BSC 2003 [DIRS 164854]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165158]) and BSC (2003 [DIRS 165159]), respectively.

- *Naval Short Waste Package Configuration* (BSC 2003 [DIRS 164855]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165160]) and BSC (2003 [DIRS 165161]), respectively.

There are a number of major components that comprise the waste package. A standard nomenclature has been established for referring to these components. This nomenclature is shown in Table 1.

Table 1. Standard Nomenclature for Waste Package Components

Preferred Terminology	Acceptable for Clarity or Brevity	Description
Trunnion Sleeve	Trunnion Collar Sleeve	The welded attachment that accepts the trunnion collar
Trunnion Collar		The removable ring that mates with the trunnion sleeve
Outer Corrosion Barrier	Outer Barrier Alloy 22 Shell	The Alloy 22 (UNS N06022) shell (sides and the outer corrosion barrier bottom lid)
Outer Lid	Final Alloy 22 Lid	The outermost lid, Alloy 22 (UNS N06022)
Middle Lid		An Alloy 22 (UNS N06022) lid, the middle of three lids
Spread Rings		The four-part ring that, when spread into position, mechanically holds the inner vessel lid in place
Inner Vessel Lid	Inner Lid	The stainless steel lid that seals the Inner Vessel
Inner Vessel	Stainless Steel Vessel	The inner vessel that is the ASME B&PV code-stamped pressure vessel
Shell Interface Ring		The stainless steel ring that sits between the support ring and the inner vessel

Preferred Terminology	Acceptable for Clarity or Brevity	Description
Inner Vessel Support Ring		The Alloy 22 (UNS N06022) ring that keeps the inner vessel off of the bottom of the outer corrosion barrier

Source: BSC 2003 [DIRS 167167], Appendix D

5.3 JUSTIFICATION OF DESIGN FEATURES

The outer lid is designed with a flat top. This is a result of the value engineering study in *Value Study Report—Waste Package Reevaluation* (BSC 2003 [DIRS 163185], Attachment III]). The outer lid weld is laser peened or burnished to reduce residual stresses (BSC 2003 [DIRS 167278], Section 4.1.1.6). The middle lid, with its own separate weld, provides another level of protection, serving as a defense in depth feature.

The bottom trunnion sleeve is extended past the outer corrosion barrier to act as an energy absorber in case of an accident. The part that extends has a tapered surface to allow runoff when the waste package is horizontal.

For ease of assembly, the inner vessel and outer corrosion barrier has a gap in between, both radially and axially. The axial gap is at least 10 mm (BSC 2003 [DIRS 161691], Section 7, p. 13), and the radial gap will be at least 1 mm (BSC 2001 [DIRS 152655], Tables 4 and 5, p. 13). These distances account for differences in thermal expansion values for Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600).

The shell interface ring is added as a measure to absorb energy during the corner drop load case. Its placement alleviates high stresses from occurring in the inner vessel bottom corner (CRWMS M&O 2000 [DIRS 157822], Section 6, p. 15).

The support ring is added to prevent the weight of the fuel from creating a force in the middle of the bottom lid of the outer corrosion barrier when the waste package is in the vertical position. The support ring elevates the inner vessel and prevents it from contacting the outer corrosion barrier.

5.3.1 Dimensions

The cavity lengths and diameters for the naval long and short waste packages are determined from the overall dimensions of the naval SNF canisters. Since there are two canister lengths (Naples 1999 [DIRS 109988], Enclosure 3, p. 2) there are two waste package configurations to accommodate them. The cavity length of the waste packages is approximately 12.7 mm (0.5 in.) greater than the length of the naval canisters. For the naval long waste package the cavity length is 5.398 m (212.5 in.), and for the naval short waste package the cavity length is 4.763 m (187.5 in.). Since both canisters share a common overall diameter, the waste package cavity diameter for both configurations is 1.702 m (67.0 in.), allowing 12.7 mm (0.5 in.) of diameter clearance. Dimensions of the two waste packages can be found in the following drawings:

- *Naval Long Waste Package Configuration* (BSC 2003 [DIRS 164854]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165158]) and BSC (2003 [DIRS 165159]), respectively.

- *Naval Short Waste Package Configuration* (BSC 2003 [DIRS 164855]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165160]) and BSC (2003 [DIRS 165161]), respectively.

5.3.2 Material Selection

The selection of materials from which reliable waste packages could be fabricated followed a multistep analysis and design process. It began by analyzing the critical functions of a particular waste package and its various components. In selecting a material for a component, the designers considered both the material's availability and the critical functions the component will serve as part of the waste package. Eight major components and eight performance criteria were identified for selecting fabricating materials (CRWMS M&O 1997 [DIRS 100259], Section 3). The eight major components are:

- Structural vessel
- Corrosion-resistant barrier
- Fill gas
- Interlocking plates for commercial design configurations
- Fuel tubes for commercial design configurations
- Structural guides for commercial design configurations
- Guide tube for co-disposal design configurations
- Thermal shunts for commercial design configurations.

Not every waste package design configuration requires all of these components; it varies according to the waste form each will hold. However, all eight of these components cover the major requirements of all ten waste package design configurations.

The eight criteria that contribute to performance are:

- Mechanical performance (strength)
- Chemical performance (resistance to corrosion and microbial attack)
- Predictability of performance (understanding the behavior of materials)
- Compatibility with materials of the waste package and waste form
- Ease of fabrication using the material
- Previous experience (proven performance record)
- Thermal performance (heat distribution characteristics)
- Neutronic performance (criticality and shielding).

Reasonableness of cost was considered as a discriminator.

Corrosion-Resistant Materials—Corrosion performance has been determined to be the most important criterion for a long waste package lifetime. Essential performance qualities therefore include a material's resistance to general and localized corrosion, stress corrosion cracking, and hydrogen-assisted cracking and embrittlement. The effects of long-term thermal aging are also important. To address the performance requirements for the waste package, the DOE has initiated studies to gain a better understanding of the processes involved in predicting the rate of waste package material corrosion over the 10,000-year regulatory period.

Combinations and arrangements of materials as containment barriers were carefully considered from several perspectives. In the process, analysts considered such criteria as (1) material compatibility (e.g., galvanic/crevice corrosion effects); (2) the material's ability to contribute to defense in depth (e.g., because it has a different failure mode from other barriers); (3) the material's ease of fabrication; and (4) the potential impact of thin, corrosion-resistant materials used as containment barriers on a repository's essential operations, such as waste package loading, handling, and emplacement.

The major objectives centered on understanding the temperature and humidity conditions that exist at different times for a range of thermal operating modes in a particular unsaturated zone, then designing the waste packages accordingly. Since the properties of any material selected for a corrosion barrier will inevitably be influenced by the temperature and humidity conditions in a repository of a particular design at a particular site, selecting the right corrosion-resistant material became one of the most important priorities.

After assessing potential materials available for waste package corrosion barriers, analysts selected nickel- and titanium-based alloys as the most promising candidate materials for corrosion resistance in an oxidizing environment. Using a corrosion-resistant material as the outer corrosion barrier of the waste package significantly lowers the risk of waste package failure from corrosion. Alloy 22 (UNS N06022) was selected as the preferred material for the outer corrosion barrier because it has excellent resistance to corrosion in the environment expected at Yucca Mountain; it is easier to weld than titanium; and it has a better thermal expansion coefficient match to stainless steel type 316 (UNS S31600) than titanium. A structurally strong material (stainless steel) was chosen for the inner layer of the waste package (CRWMS M&O 2000 [DIRS 138173], Section 7.6).

Alloy 22 (UNS N06022) also offers benefits in the areas of program and operating flexibility. It is extremely corrosion-resistant under conditions of high temperature and low humidity, such as those that will prevail for hundreds to thousands of years in a repository designed to allow a relatively high thermal output from the waste packages.

Structural Materials—The major functional requirement of the structural material for the inner layer of the waste package is to support the corrosion-resistant outer material. Stainless steel type 316 (UNS S31600) was selected for the structural layer (CRWMS M&O 2000 [DIRS 138173], Section 5.2). This material provides the required strength; has a better compatibility with Alloy 22 (UNS N06022) than carbon steel; and provides an economical solution to functional requirements. Table 2 presents the yield and tensile strengths of Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600).

Table 2. Yield and Tensile Strengths of Alloy 22 (UNS N06022) and Stainless Steel Type 316 (UNS S31600)

		Alloy 22 (UNS N06022) (MPa)	**Stainless Steel Type 316 (UNS S31600) (MPa)**
Yield Strength (σ_y)	RT [e]	310	207
	100°C [a]	273	177
	204°C [e]	236	148
	300°C [c]	214	132
	316°C [e]	211	130
Engineering Tensile Strength (S_u)	RT [e]	689	517
	100°C [a]	688	515
	204°C [e]	657	496
	300°C [c]	632	495
	316°C [e]	628	495
True Tensile Strength (σ_u)	RT [b]	971	703
	100°C [b]	977	664
	204°C [f]	926	675
	300°C [d]	910	619
	316°C [f]	885	673
	316°C [g]	911	619

NOTES: RT = room temperature.
[a] BSC 2002 [DIRS 162346], Section 5.1.
[b] BSC 2003 [DIRS 162346], Section 5.1.2.
[c] BSC 2003 [DIRS 165895], Section 5.1.
[d] BSC 2003 [DIRS 165895], Section 5.1.2.
[e] ASME 2003 [DIRS 158115], Section II, Part D, Tables Y-1 and U.
[f] BSC 2002 [DIRS 161880], Section 5.2.2.
[g] With Modified Elongation, BSC 2002 [DIRS 161880], Section 5.2.2.

The fill gas can be a significant conductor of heat from the waste form to the internal basket, so thermal performance was deemed one of the most important criteria in choosing a gas. The fill gas should not degrade other components of the waste package, so compatibility with other materials was another important criterion. Helium is inert and is routinely used as the fill gas for fuel rods, which indicates that helium will have an excellent compatibility with spent nuclear fuel. Based on a review of data on thermal conductivity, it was chosen over other candidate gases, such as nitrogen, argon, and krypton (CRWMS M&O 2000 [DIRS 138192], Sections 3.3.1 through 3.3.3).

5.3.3 ASME Code Position

The basis for the selection and application of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code to the waste package is documented in the document entitled, *BSC Position on the Use of the ASME Boiler and Pressure Vessel Code for the Yucca Mountain Waste Packages* (BSC 2003 [DIRS 165058]). This section summarizes the salient points of that document with regard to the design of the waste packages.

Yucca Mountain Review Plan, Final Report (NRC 2003 [DIRS 163274]) provides specific guidance on the appropriateness of using the ASME B&PV Code (ASME 2001 [DIRS 158115]) in the design of waste packages (e.g., Section 2.1.1.7.2.3 (1)); however, it does not prescribe the exact implementation of the ASME B&PV Code.

In any discussion of the ASME B&PV Code, it is important to first note that it is a pressure vessel safety code and that its primary mission is to assure structural adequacy for pressure loading. Any other use of the ASME B&PV Code, such as the use of the conservative material properties contained in it or failure limits for non-pressure loading, must be justified on insight into the structural phenomena that are postulated to occur. For the waste packages, component sizing and thickness are not determined by pressure loads but rather by dynamic events that the waste package might experience. Therefore, the application of the ASME B&PV Code design rules for dynamic loading of the waste packages must be carefully scrutinized to ensure that the rules are properly applied. The preparation of that document is described in *BSC Position on the Use of the ASME Boiler and Pressure Vessel Code for the Yucca Mountain Waste Packages* (BSC 2003 [DIRS 165058]).

For the application of the ASME B&PV code, Section III, Division I, Subsection NC (ASME 2001 [DIRS 158115]), has been selected by Bechtel SAIC Company (BSC) for the code-compliant design and fabrication of the waste packages. It is important to differentiate the parts of the waste package to which the code apply. There are four major assembled components of the waste package. These are (1) the stainless steel type 316 inner vessel, (2) the Alloy 22 (UNS N06022) outer corrosion barrier, (3) the internal basket assemblies, and (4) the removable trunnion collar that is used for lifting and handling purposes. With regard to the code design, the only one of these parts that is considered a pressure vessel is the stainless steel type 316 inner vessel.

With regard to the hermeticity of the inner vessel and integrity of the same against pressure loads, no currently postulated dynamic structural event involves simultaneous over-pressurization of the inner vessel. For over-pressurization, the capability of the spread ring and seal weld combination to retain the design pressure is assured by a helium leak check. While the seal welds are anticipated to be sound welds, no credit for resistance against dynamic events is taken, as these are partial-penetration welds. Therefore, for dynamic structural events where the inner vessel in the vicinity of the seal welds may be reasonably anticipated to experience significant loads, these welds are not credited to maintain the hermeticity of the inner vessel. In such cases, it must be shown that the outer corrosion barrier does not breach to maintain containment of the waste form.

For the other components of the waste package, the ASME B&PV code is only used as guidance, either through the use of conservative material properties or conservative stress limits. For credible preclosure event sequences and the assessment of those event sequences, the code and supporting code interpretations are used to formulate layered defensible material failure criteria. The basis for these failure criteria is discussed in Section 7.1.2.3.1.2.

It should be noted that if a waste package suffers a nontrivial dynamic event (i.e., drop, tip-over, etc.), the waste form would be repackaged in a new waste package and the original waste package permanently removed from service.

6. SUMMARY OF DESIGN REQUIREMENTS

Preclosure and postclosure requirements are discussed in this section. Functional requirements are taken from *Naval Spent Nuclear Fuel Waste Package System Description Document* (BSC 2003 [DIRS 165427]). The applicability of the functional requirement for compliance demonstration by Waste Package and Components is also noted.

6.1 PRECLOSURE

6.1.1 Normal Operations

Functional Requirement Number: 3.1.3.1

Functional Requirement Title: Waste Package Handling Limits

Functional Requirement Text: Waste package handling shall not introduce any surface defect in the corrosion barrier exceeding those identified by performance assessment and on interface exchange drawings. Surface defects include, but are not limited to, scratches, nicks, dents, and permanent changes to the surface stress condition (Table 3).

Table 3. Waste Package Handling Limits Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	This issue is under investigation and will be resolved prior to construction authorization.	Yes

Functional Requirement Number: 3.1.3.2

Functional Requirement Title: Waste Package Closure

Functional Requirement Text: Sealing operations shall be performed on the waste package (Table 4).

Table 4. Waste Package Closure Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	Waste package sealing operations shall meet the requirements for the waste package as specified in the SDD for the waste package closure system.	Yes

Functional Requirement Number: 3.2.1.1

Functional Requirement Title: Allowance for Decontamination of Surfaces

Functional Requirement Text: The waste package shall have surfaces that can be reasonably decontaminated (Table 5).

Table 5. Allowance for Decontamination of Surfaces Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	Performance requirements will be developed prior to license application.	Yes

Functional Requirement Number: 3.3.1.1

Functional Requirement Title: Primary Configuration and Materials

Functional Requirement Text: The waste package shall be constructed in two primary components consisting of an inner vessel and an outer corrosion barrier (Table 6).

Table 6. Primary Configuration and Materials Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	The waste package inner vessel shall have one lid and be made of stainless steel type 316 (UNS S31600).	Yes
2	The waste package outer corrosion barrier shall have two lids and be made of Alloy 22 (UNS N06022).	Yes

Functional Requirement Number: 3.3.2.1

Functional Requirement Title: Nuclear Material Control and Accountability

Functional Requirement Text: The waste package shall have a means to maintain nuclear material control and accountability (Table 7).

Table 7. Nuclear Material Control and Accountability Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	Performance requirements will be developed prior to license application.	Yes

6.1.1.1 Thermal

Functional Requirement Number: 3.1.1.5

Functional Requirement Title: Postclosure Primary Performance

Functional Requirement Text: The waste package shall be designed so that, working in combination with natural barriers and other engineered barriers, the radiological exposures to the reasonably maximally exposed individuals are within the limits established through 10 CFR 63.113(b) [DIRS 158535], and the release of radionuclides into the accessible environments are within the limits established through 10 CFR 63.113(c) [DIRS 158535] (Table 8).

Table 8. Postclosure Primary Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	The maximum waste package power at emplacement is 11.8 kW.	Yes

The thermal calculations for normal operations are performed continuously through preclosure and postclosure times. The maximum heat expected in a naval waste package is 8.51 kW (McKenzie 2001 [DIRS 158051], Table 2), much lower than the general limit placed on all waste packages of 11.8 kW. Maximum surface temperatures of the naval canisters are dependent on heat flux as shown below (Naples 1999 [DIRS 109988], Table 6). These maximum canister surface temperatures are not requirements placed on the naval spent fuel canister.

Heat Flux (kW/m^2)	Maximum Canister Surface Temperature (°C)
0.600	50
0.535	131
0.491	197
0.224	216

6.1.1.2 Structural

Functional Requirement Number: 3.1.1.1

Functional Requirement Title: Preclosure Containment

Functional Requirement Text: The waste package contains the waste form within its boundary for the preclosure period (Table 9).

Table 9. Preclosure Containment Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	The sealed waste package shall not breach during normal operations or during credible preclosure event sequences.	Yes
2	The waste package shall be designed and constructed to the codes and standards specified in (Minwalla 2003 [DIRS 161362], Section 5.1.1.)	Yes
3	Normal operations and credible event sequence load combinations are defined in (Minwalla 2003 [DIRS 161362], Table 5.1.1-1).	Yes
4	The waste package shall be designed to permit retrieval during the preclosure period until the completion of a performance confirmation program and commission review of the information obtained from such a program.	Yes
5	The waste package shall be designed to permit retrieval during the preclosure period so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the commission.	Yes
6	The waste package shall be designed to meet the full range of preclosure operating conditions for up to 300 years after the final waste emplacement.	Yes

The waste package and ancillary components shall withstand forces resulting from the normal conditions of a vertical lift and a horizontal lift (Minwalla 2003 [DIRS 161362], Table 5.1.1-1).

The waste package shall be designed to account for residual and differential thermal expansion stresses (Mecham 2004 [DIRS 166168], Section 6.2.2.2).

The waste package shall be designed to account for internal pressure resulting from differential thermal expansion and increase in temperature.

The waste package shall sustain its own self-weight while resting on the emplacement pallet (Minwalla 2003 [DIRS 161362], Table 5.1.1-2).

6.1.1.3 Shielding

Shielding analyses evaluate the effects of ionizing radiation on personnel, equipment, and materials. The primary sources for waste package radiation are gamma rays and neutrons emitted from spent nuclear fuel and high-level radioactive waste. Loading, handling, and transporting of waste packages would be carried out remotely to keep personnel exposure as low as is reasonably achievable (e.g., having the human operators behind radiation shield walls, using remote manipulators, viewing operations with video cameras). The general shielding requirements are stated in Section 4.9.1.3 of Minwalla (2003 [DIRS 161362]). Table 4.9.1-2 of *Project Design Criteria Document* (Minwalla 2003 [DIRS 161362]) does not list any shielding requirements on the waste package. The transporter and building provide shielding.

6.1.1.4 Waste Form Accommodation

Functional Requirement Number: 3.1.2.1

Functional Requirement Title: Naval SNF Quantities and Characteristics

Functional Requirement Text: The waste package shall accommodate naval SNF canisters (Table 10).

Table 10. Naval SNF Quantities and Characteristics Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	The naval short waste package shall accommodate a short (maximum 4.750 m [187.00 inches]) naval SNF canister with a maximum diameter of 1.689 m (66.5 inches) and made of stainless steel type 316L (UNS S31603).	Yes
2	The naval long waste package shall accommodate a long (maximum 5.385 m [212.00 inches]) naval SNF canister, with a maximum diameter of 1.689 m (66.5 inches) and made of stainless steel type 316L (UNS S31603).	Yes
3	The waste package shall accommodate a naval SNF canister with a maximum weight of 49 tons.	Yes
4	The waste package shall accommodate a naval SNF canister with an axial and radial center-of-gravity range to be established by the Naval Nuclear Propulsion Program.	Yes

6.1.1.5 Criticality

Functional Requirement Number: 3.1.1.3

Functional Requirement Title: Criticality Control

Functional Requirement Text: The sealed waste package shall provide criticality control (Table 11).

Table 11. Criticality Control Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	The methodology defined in the Naval Nuclear Propulsion Program addendum (Mowbray 1999 [DIRS 149585]) *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) shall be used to demonstrate acceptable postclosure criticality control for canisters and the waste packages in which they are disposed.	No. The Naval Nuclear Propulsion Program will address this requirement.
2	The methodology defined in the Naval Nuclear Propulsion Program letter (Griffith 2003 [DIRS 165175]) shall be used to demonstrate acceptable preclosure criticality control for canisters and the waste packages in which they are disposed.	No. The Naval Nuclear Propulsion Program will address this requirement.
3	The Naval Nuclear Propulsion Program will verify meeting the criticality criteria.	No. The Naval Nuclear Propulsion Program will address this requirement.

Due to the confidential nature of naval reactor fuel, no criticality calculations will be performed by the Waste Package and Components group. The U.S. Navy has provided an addendum to the *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]), which outlines the criticality methodology used by the Naval Nuclear Propulsion Program. Any assumptions concerning criticality are beyond the scope of this document.

6.1.2 Event Sequence Evaluation

6.1.2.1 Thermal

The only event sequence affecting thermal performance is the fire accident (see Table 5.1.1-1 of Minwalla (2003 [DIRS 161362]). Thermal performance requirements during a fire are addressed by the Naval Nuclear Propulsion Program.

6.1.2.2 Structural

The waste package shall not breach during normal operation or during credible preclosure event sequences (BSC 2003 [DIRS 165427], Section 3.1.1). The events considered include:

> **Rock Fall on Waste Package**–The waste package is at rest on the emplacement pallet in the drift without a drip shield, when rock(s) fall and impact the waste package surface (Mecham 2004 [DIRS 166168], Section 6.2.2.4).

Object Drop on Waste Package–The waste package is at rest in a vertical position and a equipment failure (i.e., gantry crane) falls and impacts the top of the waste package (Mecham 2004 [DIRS 166168], Section 6.2.2.4).

Missile Impact on Waste Package–The waste package is at rest and a small object at high velocity impacts the waste package surface (Mecham 2004 [DIRS 166168], Section 6.2.2.4).

Waste Package Vertical Drop–The waste package is being lifted in a vertical orientation at a height of 2.0 m when the lifting device inadvertently drops it. The waste package impacts the ground squarely on its base (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Waste Package Tip-Over–The waste package is at rest on the ground in a vertical position and an external force (such as a seismic event) causes the waste package to tip over and impact the ground. A tip-over from an elevated surface is also considered (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Waste Package Horizontal Drop–The waste package is being lifted in a horizontal orientation at a height of 2.4 m when the lifting device inadvertently drops it. The waste package impacts the ground squarely on its side (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Horizontal Drop with Emplacement Pallet–The emplacement pallet with waste package is being lifted in a horizontal orientation when the lifting device inadvertently drops it. The emplacement pallet with waste package impacts the ground along its horizontal axis. This is also done as a horizontal drop onto the emplacement pallet. The emplacement pallet is the object considered that may puncture the waste package (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Waste Package Corner Drop–The waste package is being lifted in a vertical orientation at a height of 2.0 m when the lifting device inadvertently drops it. A corner of the waste package impacts the ground first (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Waste Package 10-Degree Oblique Drop with Slap Down–The waste package is being lifted in a horizontal orientation at a height of 2.4 m when the lifting device inadvertently releases one end. After the bottom end has rotated 10 degrees, the lifting device holding the top of the waste package fails and the entire waste package falls due to gravity and impacts the ground.

Waste Package Swing Down–The waste package is being lifted in a horizontal orientation at a height of 2.4 m when the lifting device inadvertently releases one end. One end of the waste package remains held by the lifting device while the other end swings down and impacts the ground (Mecham 2004 [DIRS 166168], Section 6.2.2.5).

Waste Package Exposed to Vibratory Ground Motion–The waste package is subjected to vibratory ground motion in the underground for a seismic evaluation for an annual frequency of exceedance of $5x10^{-4}$ per year (Mecham 2004 [DIRS 166168], Section 6.2.2.6).

The drop analyses described previously are performed with the trunnion collars attached to the fully loaded waste package, since that is the normal condition for moving the waste package. The trunnion collar adds a small amount of mass and loading, so in general, the drops analyzed with the attached trunnion collar are conservative

6.2 POSTCLOSURE

6.2.1 Structural

Functional Requirement Number: 3.1.1.2

Functional Requirement Title: Postclosure Confinement

Functional Requirement Text: The sealed waste package shall restrict the transport of radionuclides to the outside of the waste package boundary after repository closure (Table 12).

Table 12. Postclosure Confinement Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	In conjunction with natural barriers and other engineered barriers, the sealed waste package shall limit transport of radionuclides in a manner sufficient to meet long-term repository performance requirements.	Yes
2	The waste package shall be designed and constructed to the codes and standards specified in the (Minwalla 2003 [DIRS 161362], Section 5.1.1).	Yes
3	Normal operations and event load combinations are defined in the (Minwalla 2003 [DIRS 161362], Table 5.1.1-2).	Yes

The waste package shall not breach during a postclosure seismic event (Minwalla 2003 [DIRS 161362], Table 5.1.1-2). This postclosure seismic event is addressed in *Commercial SNF Waste Package Design Report* (BSC 2004 [DIRS 166876], Appendix A).

6.2.2 Thermal

Thermal requirements are the same for preclosure as for postclosure. See Section 6.1.1.1 for thermal requirements.

7. SATISFACTION OF DESIGN REQUIREMENTS

7.1 PRECLOSURE

The waste package must satisfy defined performance specifications to protect the public and workers and to meet the performance objectives of a repository. An example of a performance specification is the ability of a waste package to withstand a tip-over event without breaching. Performance specifications are discussed in the following sections, where they are categorized by relevant engineering discipline (i.e., thermal, criticality, structural, and shielding). Detailed discussions of performance specifications are available in system description documents (e.g., BSC 2003 [DIRS 165427]).

Some of the performance specifications and supporting evaluations depend on temperature. In these cases, the evaluation is based on the higher-temperature operating mode. Further evaluations of lower-temperature operating modes are part of ongoing engineering studies.

7.1.1 Normal Operations

7.1.1.1 Thermal

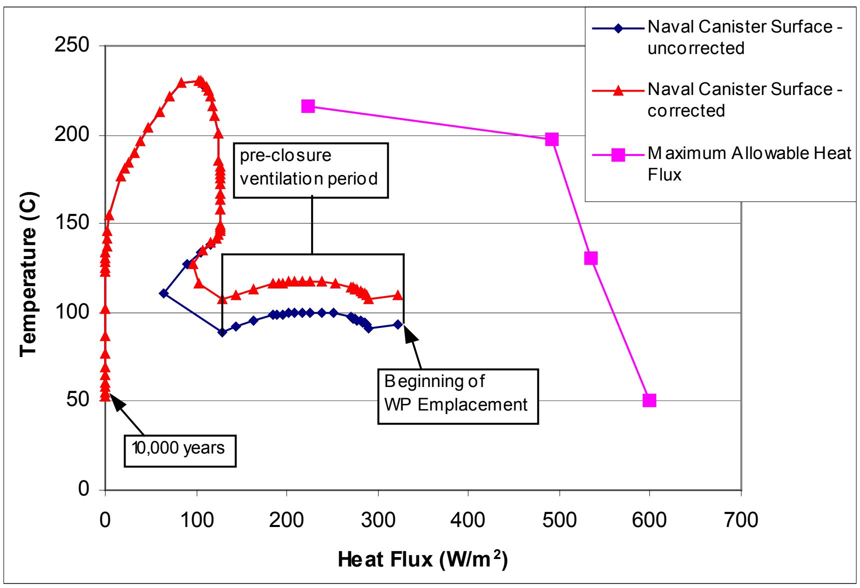
The thermal calculations for normal operations are performed continuously through preclosure and postclosure times. Two-dimensional, thermal calculations for a naval long waste package are reported in *Thermal Evaluation of the Naval Long Waste Package* (BSC 2004 [DIRS 167080]).

The temperature boundary conditions applied to the waste package outer surface for the two-dimensional calculations are taken from a three-dimensional (pillar) calculation of a representative drift segment (BSC 2003 [DIRS 164726], Section 6).

The thermal condition for the naval waste package inside of the trolley in the surface facility weld cell has not been evaluated; however, this condition was addressed for the 21-PWR waste package, which is a more bounding case (BSC 2003 [DIRS 164075], Section 6). Most of the cases in this calculation were for a waste package in a shielded transporter, except for one case, which determined results without a transporter. These calculations show that without shielding, waste form temperatures near the center of the waste package remain below 350°C at all times, even if steady-state thermal conditions are achieved. If shielding is used, temperatures remain below 350°C for several days, but will eventually exceed this temperature. For this reason the duration the waste packages with high heat loads can remain in a shielded transporter must be limited.

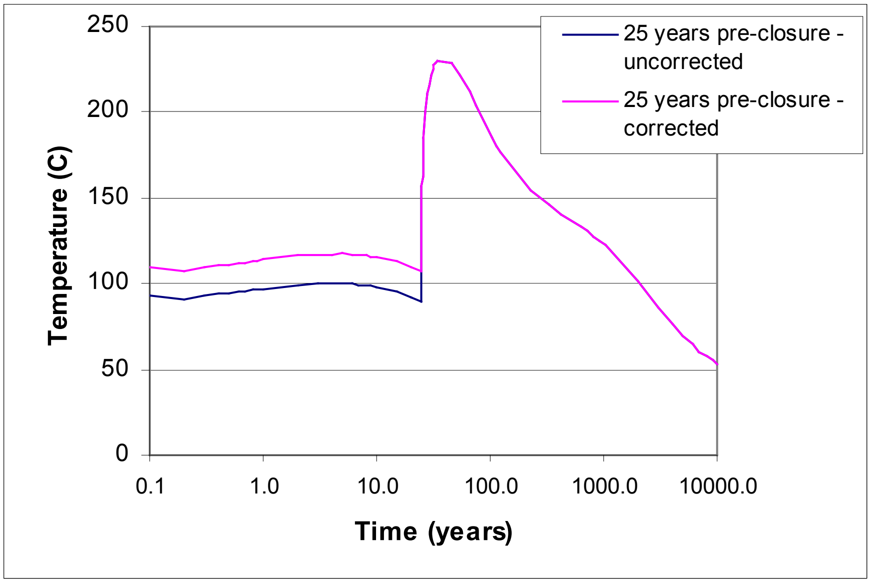
7.1.1.1.1 Thermal Results

Figure 2 and Figure 3 from *Thermal Evaluation of the Naval Long Waste Package* (BSC 2004 [DIRS 167080], Figures 5 and 8, respectively) are shown below. The maximum canister surface temperature is about 230°C and occurs a few decades after closure. All temperatures are below their respective heat flux limits. The "corrected" calculation, shown in Figure 3, investigated the sensitivity of peak cladding temperature to the method used to calculate heat removal by ventilation. In the base case, all the heat removed by ventilation (80% heat removal efficiency) was considered to convect directly from the waste package surface to the air stream. More realistically, most of the heat is radiated from the waste package to the drift wall and then convected to the air stream. This increases the waste package surface temperature by about 35°C during the ventilation period but has negligible effect on the peak cladding temperature that occurs during post-closure. Further thermal calculations internal to the naval canister are performed by the Naval Nuclear Propulsion Program independently from BSC waste package design.



Source: BSC 2004 [DIRS 167080], Figure 8

Figure 2. Temperature as a Function of Heat Flux



Source: BSC 2004 [DIRS 167080], Figure 5

Figure 3. Temperature of the Outer Surface of the Naval SNF Canister

Several conservative assumptions are used in these calculations, including calculating heat transfer through the rock as conduction only and ignoring the heat transfer due to gross water movement. The peak surface temperature of a 21-PWR waste package using conduction only is about 230°C (BSC 2004 [DIRS 166695], Section 6) and this temperature is bounding for all waste packages including the naval waste package. Additional conservative assumptions include using low ventilation efficiency, maximum gap between the stainless steel and Alloy 22 (UNS N06022), and no natural convection during postclosure.

7.1.1.1.2 Two-Dimensional Repository Calculations

Calculations have been performed with a two dimensional representation of the repository (BSC 2003 [DIRS 165093]). These calculations use line heat loads, and waste packages are considered as a continuous infinite cylindrical heat source. Such calculations can be performed rapidly, and the numerous results are used to generate response surfaces, i.e., surfaces of constant peak waste package surface temperature as a function of waste package spacing, ventilation efficiency, and ventilation time. Different sets of response surfaces are generated for varied line heat loads. By holding all but one variable constant on a given response surface, operating curves can be generated, which show the variation of waste package temperature due to variation in the remaining variable. The repository two-dimensional temperatures cannot be strictly applied to waste packages, but the change in temperature in a small locus of points provides the designer with a good method to determine the impact that changes in design have on waste package temperatures. Hence, the results of bounding calculations presented in this report can be used with the operating curves from *Two-Dimensional Repository Thermal Design Calculations* (BSC 2003 [DIRS 165093]) to estimate thermal margins resulting from design variations.

7.1.1.2 Structural

7.1.1.2.1 Lifting

The waste package must be able to be lifted using the twist-on trunnion collars for normal operations. The waste package will be lifted by the top trunnion collar when in the vertical orientation and by both the top and bottom trunnion collars when in the horizontal orientation. Since the top trunnion collar lifts the entire waste package in the vertical orientation and the Naval SNF Long waste package has the greatest mass, this scenario was analyzed. The resulting maximum stress intensities for various waste package components at room temperature and 300°C are presented in Tables 13 and 14 from *Vertical Lift of the Naval SNF Long Waste Package* (BSC 2003 [DIRS 166827], Table 6-3 and 6-4, respectively).

Table 13. Maximum Stress Intensities at Room Temperature

	σ_{int} (MPa)	σ_y (MPa)	σ_u (MPa)	σ_{int}/σ_y	σ_{int}/σ_u	1/3 σ_y (MPa)	1/5 σ_u (MPa)
Outer Corrosion Barrier	56	310	689	0.18	0.08	103	138
Trunnion Sleeve	280	310	689	0.90	0.41	103	138
Trunnion Sleeve Bottom Weld	44	310	689	0.14	0.06	103	138
Trunnion Collar	320	1170	1310	0.27	0.24	390	262
Trunnion	158	1170	1310	0.14	0.12	390	262

Source: BSC 2003 [DIRS 166827], Table 6-3

Table 14. Maximum Stress Intensities at 300°C

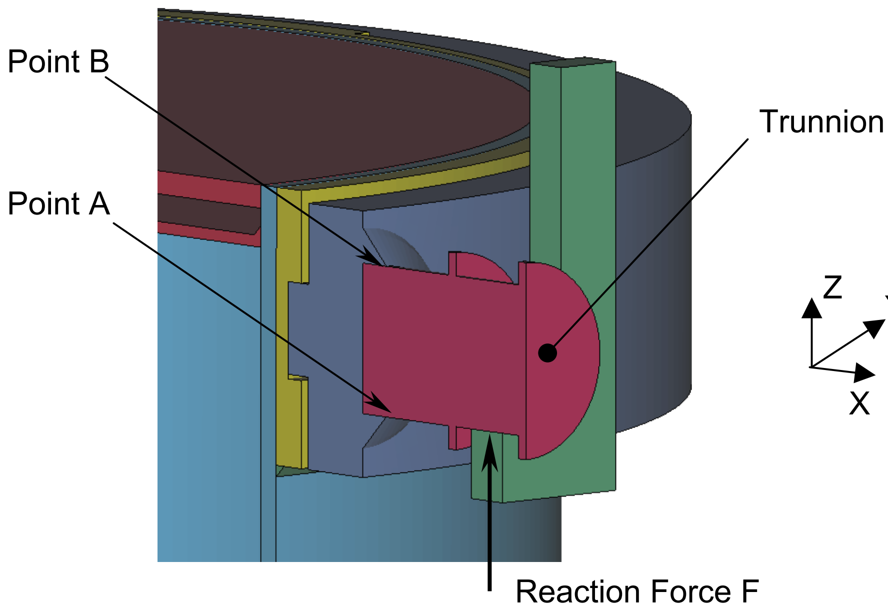
	σ_{int} (MPa)	σ_y (MPa)	σ_u (MPa)	σ_{int}/σ_y	σ_{int}/σ_u	1/3 σ_y (MPa)	1/5 σ_u (MPa)
Outer Corrosion Barrier	54	214	688	0.25	0.08	71	138
Trunnion Sleeve	360	214	688	1.68	0.52	71	138
Trunnion Sleeve Bottom Weld	44	214	688	0.21	0.06	71	138
Trunnion Collar	320	965	1100	0.33	0.29	322	220
Trunnion	156	965	1100	0.16	0.14	322	220

Source: BSC 2003 [DIRS 166827], Table 6-4

Tables 13 and 14 show that the maximum stresses in the components of the waste package are less than 1/3 the yield strength and 1/5 the tensile strength (ANSI N14.6-1993 [DIRS 102016], Section 4.2.1.1). However, the trunnion sleeve and the lifting collar have maximum stress intensities above those limits. BSC (2003 [DIRS 166827], Section 6) shows the maximum stress in the trunnion sleeve is a localized contact stress between the trunnion sleeve and the trunnion collar. Furthermore, BSC (2003 [DIRS 166827], Section 6) shows that the stresses are far below the requirements in the surrounding areas and through the thickness of the engagement. In addition the lifting collars and trunnion sleeve have filleted corners and chamfered corners that would alleviate stresses in the corners and edges.

The trunnion undergoes repeated bending stress from the engagement of the hooks. From Table 13 and Table 14 the tensile stress at Point A will cycle from zero to approximately 160 MPa. Since fatigue failure occurs faster in compression than in tension (ASM 1980 [DIRS 104317], Figure 10), Point B of Figure 4 will be considered. Since Point B lies on the exact opposite surface of the direction of bending, the stress will be exactly the same, only it will be in compression instead of tension.

Therefore, Point B will undergo cyclic compression from 0 to 160 MPa. Meaning the mean stress is 80 MPa and the alternating stress is also 80 MPa.



Source: BSC 2003 [DIRS 166827], Figure 6-1

Figure 4. Location of Stress on the Trunnion

It is seen that the stress is approximately 7 times less than the fatigue limit for 10^7 cycles (ASM 1980 [DIRS 104317], Figure 10). Although the yield and tensile strength of the material for this Constant-life diagram is slightly higher, considering the trunnion collar will never undergo 10^7 cycles and its cycling is not constant, the design of the lifting collar is adequately designed for any possible fatigue. Therefore the trunnion collars are appropriately designed for normal handling operations.

7.1.1.2.2 Radial Thermal Expansion

Seven different potential waste package design configurations are evaluated in *Waste Package Outer Barrier Stress Due to Thermal Expansion with Various Barrier Gap Sizes* (BSC 2001

[DIRS 152655]): 21-PWR, 44-BWR, 24-BWR, 12-PWR Long, 5 DHLW/DOE SNF Short, 2-MCO/2-DHLW, and Naval SNF Long. For each one of these potential waste package design configurations, a parametric study is performed by calculating the interference produced by the thermal expansion of the inner vessel and outer corrosion barrier. The interference between the two components causes a pressure at the interface of the two components. This pressure is used to calculate the outer corrosion barrier tangential stresses at the inner and outer surfaces. The temperature range for this calculation is 20°C to 239°C.

The outer corrosion barrier maximum tangential stresses at the outer and inner surfaces for a corresponding gap size are shown in Tables 15 and 16.

Table 15. Outer Corrosion Barrier Maximum Tangential Stress at the Outer Surface

	Maximum Tangential Stress at the Outer Surface, σ_{os} (MPa)										
Waste Package Type	**Gap Size (mm)**										
	0.0	**0.1**	**0.2**	**0.3**	**0.4**	**0.5**	**0.6**	**0.7**	**0.8**	**0.9**	**1.0**
21-PWR	140.9	122.1	103.2	84.4	65.6	46.8	27.9	9.1	0.0	0.0	0.0
44-BWR	140.9	122.4	103.9	85.5	67.0	48.5	30.1	11.6	0.0	0.0	0.0
24-BWR	141.3	117.4	93.5	69.6	45.8	21.9	0.0	0.0	0.0	0.0	0.0
12-PWR Long	140.8	117.2	93.6	69.9	46.3	22.7	0.0	0.0	0.0	0.0	0.0
5 DHLW/DOE SNF Short	131.4	117.9	104.4	90.9	77.4	63.9	50.4	36.9	23.4	9.9	0.0
2-MCO/2-DHLW	130.9	115.0	99.2	83.4	67.5	51.7	35.8	20.0	4.2	0.0	0.0
Naval SNF - Long	130.4	115.7	101.1	86.4	71.7	57.0	42.4	27.7	13.0	0.0	0.0

Source: BSC 2001 [DIRS 152655], Table 4

Table 16. Outer Corrosion Barrier Maximum Tangential Stress at the Inner Surface

	Maximum Tangential Stress at the Inner Surface, σ_{is} (MPa)										
Waste Package Type	**Gap Size (mm)**										
	0.0	**0.1**	**0.2**	**0.3**	**0.4**	**0.5**	**0.6**	**0.7**	**0.8**	**0.9**	**1.0**
21-PWR	144.6	125.3	106.0	86.6	67.3	48.0	28.7	9.4	0.0	0.0	0.0
44-BWR	144.5	125.6	106.6	87.7	68.7	49.8	30.8	11.9	0.0	0.0	0.0
24-BWR	146.1	121.4	96.7	72.0	47.3	22.7	0.0	0.0	0.0	0.0	0.0
12-PWR Long	145.6	121.1	96.7	72.3	47.8	23.4	0.0	0.0	0.0	0.0	0.0
5 DHLW/DOE SNF - Short	134.8	120.9	107.1	93.2	79.4	65.5	51.7	37.9	24.0	10.2	0.0
2-MCO/2-DHLW	134.8	118.5	102.2	85.9	69.5	53.2	36.9	20.6	4.3	0.0	0.0
Naval SNF - Long	134.1	119.0	103.9	88.8	73.7	58.6	43.5	28.5	13.4	0.0	0.0

Source: BSC 2001 [DIRS 152655], Table 5

As a result of this calculation the minimum radial gap is determined to be 1.0 mm. The naval configuration drawings, referenced in Section 5.2, incorporate this minimum radial gap.

7.1.1.2.3 Axial Thermal Expansion

Four different potential waste package design configurations are evaluated in *Waste Package Axial Thermal Expansion Calculation* (BSC 2003 [DIRS 161691]): the 21-PWR, Naval SNF

Long, 44-BWR, and 5 DHLW/DOE SNF Long. For each one of these potential waste package design configurations, a parametric study is performed to calculate the interference produced by the thermal expansion of the inner vessel and outer corrosion barrier to determine the required axial gap. Because the inner vessel undergoes a greater change in temperature and has a larger coefficient of thermal expansion as compared to those of the outer corrosion barrier, this interference is calculated as the inner vessel length minus the outer corrosion barrier cavity length subsequent to thermal expansion. The length of this interference is equal to the required axial gap created during fabrication (i.e., at room temperature and prior to fuel loading) to avoid contact between the inner vessel and outer corrosion barrier during thermal expansion. These minimum axial gaps are presented in Table 17.

Table 17. Minimum Required Axial Gap Between the Inner Vessel and Outer Corrosion Barrier

Waste Package Type	**Maximum Interference**	
	(mm)	**(in.)**
21-PWR	8.1	0.32
Naval SNF - Long	7.6	0.30
44-BWR	8.0	0.32
5 DHLW/DOE SNF- Long	6.8	0.27

Source: BSC 2003 [DIRS 161691], Table 4

As a result of this calculation and for added conservatism the minimum axial gap is determined to be 10.0 mm. The naval configuration drawings, referenced in Section 5.2, incorporate this minimum axial gap.

7.1.1.2.4 Internal Pressurization Due to Thermal Expansion

This calculation determines the resulting tangential (hoop) and longitudinal stresses in the outer corrosion barrier produced by an internal pressure increase due to elevated temperatures and a decreasing volume from thermal expansion. From *Waste Package Outer Barrier Stress Due to Thermal Expansion with Various Barrier Gap Sizes* (BSC 2001 [DIRS 152655], Tables 4 and 5, p. 13) the required radial gap between the waste package inner vessel and outer corrosion barrier to avoid contact is 1 mm. This calculation assumes that the waste package inner vessel and outer corrosion barrier have a 1-mm gap between them, and this gap collapses completely due to uneven thermal expansion; consequently, the gas volume between the inner vessel and outer corrosion barrier decreases, increasing the internal pressure. Temperature is also taken into consideration for calculating the internal pressure, and for added conservatism the only mode of heat transfer is solely through radiation to determine the temperature difference between the inner vessel and outer corrosion barrier.

Table 18 below provides the resulting gage pressure with respect to ambient pressure for each waste package. The results are summarized in Table 19 and the non-dimensional results in Table 20, comparing the tangential (hoop) and longitudinal stress to the yield stress (BSC 2003 [DIRS 167005]).

Table 18. Resulting Gage Pressure with Respect to Ambient Pressure

Waste Package	Gage Pressure with Respect to Ambient, p_{gage} (atm)	(KPa)	(psi)
21-PWR	2.65	268	38.9
Naval SNF - Long	2.24	227	32.9
44-BWR	2.59	263	38.1
5 DHLW/DOE SNF - Long	2.09	212	30.7

Source: BSC 2003 [DIRS 167005], Table 2

Table 19. Calculation Results

Waste Package	Tangential Stress, σ_h		Longitudinal Stress, σ_l	
	(MPa)	(ksi)	(MPa)	(ksi)
21-PWR	10.2	1.48	5.12	0.742
Naval SNF - Long	10.3	1.50	5.17	0.750
44-BWR	10.2	1.48	5.12	0.742
5 DHLW/DOE SNF - Long	10.5	1.52	5.26	0.762

Source: BSC 2003 [DIRS 167005], Table 3

Table 20. Non-dimensional Results

Waste Package	σ_h / σ_y (%)	σ_l / σ_y (%)
21-PWR	4.51	2.26
Naval SNF - Long	4.55	2.28
44-BWR	4.51	2.25
5 DHLW/DOE SNF - Long	4.63	2.32

Source: BSC 2003 [DIRS 167005], Table 4

Based on the results of Table 20, the outer corrosion barrier is subjected to a stress that is less than 5 percent of its yield strength in the hoop direction and less than 3 percent in the axial direction.

7.1.1.2.5 Static Waste Package on Pallet

Static Waste Packages on Emplacement Pallet (BSC 2002 [DIRS 165492]) reports static stresses for each of the four flagship waste packages resting on the emplacement pallet with three different variations in radial gap between the inner vessel and outer corrosion barrier, producing parametric results. This was done to create a solution that can be used for later modifications to the design. The radial gap sizes evaluated were 4 mm, 10 mm, and 15 mm. The outer corrosion barrier thickness was reduced to conservatively show 10,000 years of corrosion degradation. Table 21 shows that the Naval Long waste package is capable of supporting its own weight when on the emplacement pallet.

Table 21. Maximum Stresses Intensities in Outer Corrosion Barrier

Waste Package	4 mm Radial Gap (MPa)	10 mm Radial Gap (MPa)	15 mm Radial Gap (MPa)
21-PWR	90	80	90
44-BWR	86	80	116
Naval SNF - Long	74	84	76
5 DHLW/DOE SNF-Short	20	42	52

Source: BSC 2003 [DIRS 165492], Table 6-2

The stresses reported are less than the yield stress of Alloy 22 (UNS N06022). The results indicate significant margin to failure for a range of gap sizes. The yield stress of Alloy 22 (UNS N06022) may be found in Section 5.3.2, Table 2 of this document. Therefore, the waste package is able to withstand the stresses of its own weight even after 10,000 years of degradation. Since this is a bounding case, the results show that the non-degraded waste package is also capable of withstanding the stresses of its own weight.

7.1.1.3 Shielding

The following dose rate calculations are from *Dose Rate Calculation for the Naval Long Waste Package* (BSC 2004 [DIRS 167082]).

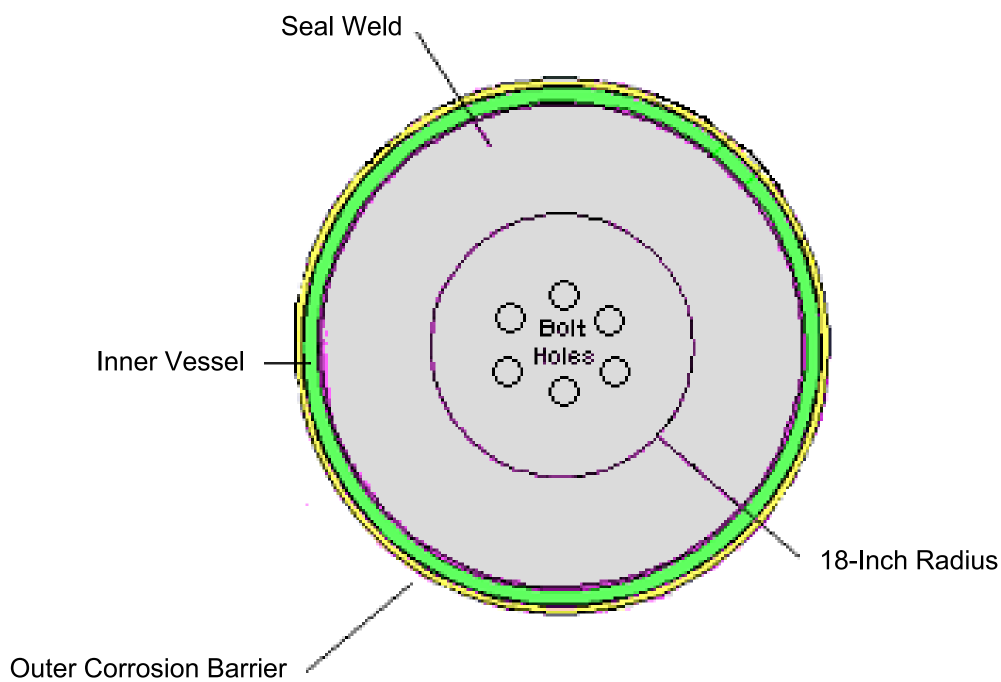
7.1.1.3.1 Description of Shielding Calculations

7.1.1.3.1.1 Source Sampling

Calculations are performed to determine the dose rates at selected surfaces on and near the waste package exterior at 2 and 5 years after reactor shutdown as a result of photon and neutron currents exiting the naval long SNF canister filled with naval SNF. The currents exiting the naval canister surfaces are the radiation sources emitting from the canister. Since the neutron sources from the top, bottom, and side surfaces do not vary significantly, only one calculation is necessary for the neutron surface dose rates. However, for the photon dose rates, three separate calculations are required for the top, bottom, and side surfaces. The side source of $4.7069 \cdot 10^{15}$ photon/sec is several orders of magnitude larger than the top and bottom sources of $3.5102 \cdot 10^{7}$ and $7.5892 \cdot 10^{11}$ photon/sec, respectively. These large differences in the relative magnitude of the source intensities result in under sampling of the top and bottom source regions, therefore making a single calculation insufficient.

Furthermore, the naval canister top surface includes three different sources corresponding to the six bolt holes, the region outside the bolt holes and inside an 18-in. radius, and the region above the seal weld as illustrated in Figure 5. When multiple source regions are present, corresponding source distribution numbers are used to specify their locations. Spatially, a degenerate cylindrical volume distribution is used to specify particle sampling from the top, bottom, and radial surfaces of the naval canister. The outer dimensions of the degenerate cylinder are

0.0001 cm beyond the outer surfaces of the naval long SNF canister. This representation of the surface sources does not make any detectable difference in the answers. Also, a cylinder with reflective surfaces is used to represent the naval canister. This representation is advantageous because the material and specific geometric details of the naval SNF canister are not needed. Furthermore, reflective surfaces transform the default isotropic source for a cylindrical volume distribution to an isotropic distribution only in the outward direction of the naval canister, which corresponds to the physical reality.



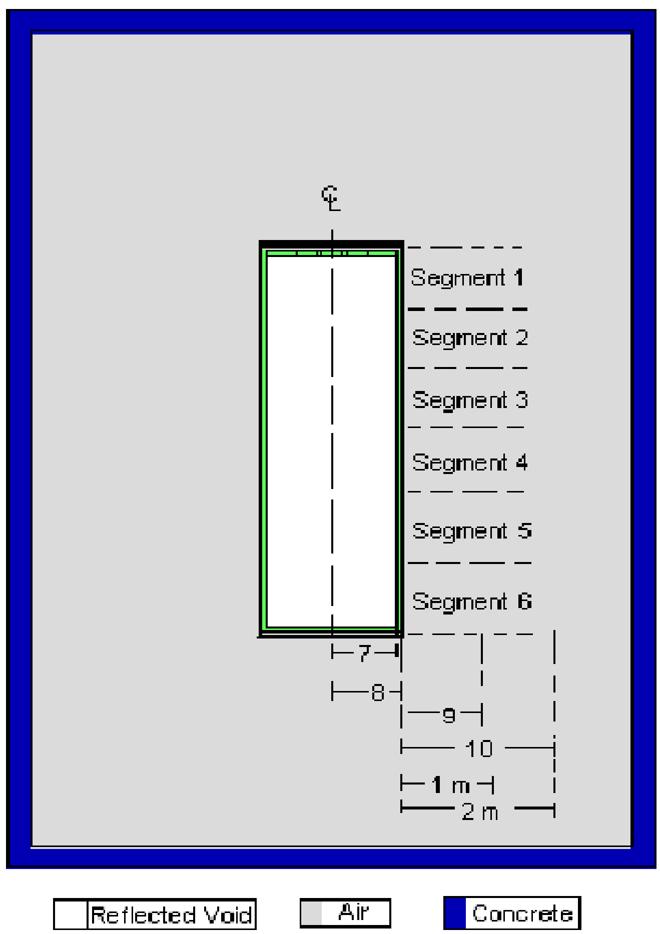
NOTE: Drawing not to scale.
Source: BSC 2004 [DIRS 167082], Figure 5.2-1

Figure 5. Source Region Representation of the Naval SNF Canister Top Surface

7.1.1.3.1.2 Dose Rate Calculation

In this calculation, particle crossings over the surfaces of interest are estimated to determine the particle flux. Therefore, the external radial and axial surfaces of the waste package are divided into surface segments. The average dose rate over each segment area is tabulated to examine the spatial distribution of the dose rate. The waste package outer radial surface was divided into six equivalent segments, as shown in Figure 6, corresponding to the naval waste package’s inner height to examine the axial dose variance. Segments 1 through 6 are 89.96 cm in height.

Segment 7 corresponds to the radial surface area between the waste package centerline and the waste package outer corrosion barrier outer radius of 90.645 cm. Segment 8 corresponds to the radial surface area between the waste package centerline and the waste package outer corrosion barrier outer radius of 93.185 cm. Segment 9 corresponds to the radial surface area between the waste package outer corrosion barrier outer radius of 93.185 cm and the radial surface 1 m from the waste package outer corrosion barrier outer radius of 193.185 cm. Segment 10 corresponds to the radial surface area between the waste package outer corrosion barrier outer radius of 93.185 cm and the radial surface 2 m from the waste package outer corrosion barrier outer radius of 293.185 cm. Figure 6 also displays the concrete structure (30 cm thick) surrounding the waste package.



NOTE: Drawing not to scale.
Source: BSC 2004 [DIRS 167082], Figure 5.2-2

Figure 6. Waste Package Radial Surfaces Segments used in Dose Rate Calculation

The photon calculations, for a given time of reference (i.e., 2 and 5 years), were conducted in three separate simulations: top, bottom, and side. The neutron dose calculation was performed in a single calculation. Furthermore, the effects of photon backscattering was quantified by examining the difference in the exterior dose rates with and without the 30 cm thick concrete cylindrical cell present.

7.1.1.3.2 Waste Package Surface Dose Rates

Table 22 through Table 33 from BSC (2004 [DIRS 167082]) present the gamma, neutron, and total dose rates averaged over the surface segments of the naval long waste package and at 1 and 2 meters away from the waste package external radial surfaces. Table 22 shows that the waste package surface dose rates due to secondary gamma rays are negligible as compared to the gamma dose rates. Consequently, this dose component was not listed in the following tables.

Table 22. Dose Rates on the Waste Package Inner Vessel Inner Surface T = 2 Years

	Primary Gamma		Neutron		Secondary Gamma		Total	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	93042.8	0.0011	0.662	0.002	0.0013	0.012	93043.5	0.0011
Segment 2	98600.0	0.0011	0.711	0.001	0.0014	0.011	98600.7	0.0011
Segment 3	98380.2	0.0011	0.710	0.001	0.0014	0.012	98380.9	0.0011
Segment 4	98393.8	0.0011	0.711	0.001	0.0014	0.012	98394.5	0.0011
Segment 5	98548.8	0.0011	0.707	0.001	0.0014	0.011	98549.5	0.0011
Segment 6	98467.2	0.0011	0.694	0.002	0.0013	0.011	98467.9	0.0011

Source: BSC 2004 [DIRS 167082], Table 6.1-1

Table 23. Dose Rates on the Waste Package Inner Vessel Outer Surface T = 2 Years

	Primary Gamma		Neutron		Total	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	1985.6	0.0019	0.231	0.002	1985.8	0.0019
Segment 2	2106.2	0.0018	0.252	0.002	2106.4	0.0018
Segment 3	2108.5	0.0018	0.251	0.002	2108.7	0.0018
Segment 4	2101.9	0.0018	0.252	0.002	2102.1	0.0018
Segment 5	2108.1	0.0018	0.251	0.002	2108.4	0.0018
Segment 6	2072.7	0.0018	0.239	0.002	2073.0	0.0018

Source: BSC 2004 [DIRS 167082], Table 6.1-2

Table 24. Dose Rates on the Waste Package Outer Corrosion Barrier Outer Surface T = 2 Years

	Primary Gamma		Neutron		Total	
Axial Location	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Segment 1	368.0	0.0025	0.095	0.002	368.1	0.0025
Segment 2	391.4	0.0024	0.104	0.001	391.5	0.0024
Segment 3	392.1	0.0024	0.104	0.001	392.2	0.0024
Segment 4	390.7	0.0024	0.104	0.001	390.8	0.0024
Segment 5	391.8	0.0024	0.104	0.001	391.9	0.0024
Segment 6	383.3	0.0025	0.099	0.002	383.4	0.0025

Source: BSC 2004 [DIRS 167082], Table 6.1-3

Table 25. Dose Rates 1m from the Waste Package Outer Corrosion Barrier Outer Surface T = 2 Years

	Primary Gamma		Neutron		Total	
Axial Location	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Segment 1	124.4	0.0022	0.033	0.002	124.5	0.0022
Segment 2	160.3	0.0019	0.041	0.001	160.4	0.0019
Segment 3	166.9	0.0019	0.044	0.001	166.9	0.0019
Segment 4	166.8	0.0019	0.044	0.001	166.8	0.0019
Segment 5	161.8	0.0019	0.041	0.001	161.8	0.0019
Segment 6	128.0	0.0022	0.034	0.002	128.0	0.0022

Source: BSC 2004 [DIRS 167082], Table 6.1-4

Table 26. Dose Rates 2m from the Waste Package Outer Corrosion Barrier Outer Surface T = 2 Years

	Primary Gamma		Neutron		Total	
Axial Location	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Segment 1	76.3	0.0021	0.023	0.002	76.4	0.0021
Segment 2	99.5	0.0018	0.028	0.001	99.5	0.0018
Segment 3	108.3	0.0018	0.031	0.001	108.3	0.0018
Segment 4	108.7	0.0018	0.031	0.001	108.7	0.0018
Segment 5	100.4	0.0018	0.029	0.001	100.4	0.0018
Segment 6	77.9	0.0021	0.024	0.002	78.0	0.0021

Source: BSC 2004 [DIRS 167082], Table 6.1-5

Table 27. Dose Rates on the Waste Package Surface T = 2 Years

		Gamma		Neutron		Total	
Axial Location	Segment	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Lower Inner Vessel Bottom Surface	Segment 7	4.072E+01	0.0027	7.384E-02	0.0035	4.079E+01	0.0027
Outer Corrosion Barrier Bottom Surface	Segment 8	9.392E+00	0.0163	3.070E-02	0.0036	9.423E+00	0.0162
	Segment 9	7.228E+01	0.0042	2.623E-02	0.0021	7.231E+01	0.0042
1 m from Outer Corrosion Barrier Bottom	Segment 8	1.034E+01	0.0091	1.554E-02	0.0041	1.036E+01	0.0091
	Segment 9	1.416E+01	0.0044	1.382E-02	0.0025	1.417E+01	0.0044
2 m from Outer Corrosion Barrier Bottom	Segment 8	8.945E+00	0.0087	1.120E-02	0.0046	8.957E+00	0.0087
	Segment 10	1.110E+01	0.0030	1.090E-02	0.0018	1.111E+01	0.0030
Top of Waste Package Cavity	Hole # 1	2.694E-02	0.0111	2.763E-02	0.0946	5.457E-02	0.0482
	Hole # 2	2.716E-02	0.0112	2.772E-02	0.1110	5.488E-02	0.0563
	Hole # 3	2.662E-02	0.0113	3.278E-02	0.1002	5.940E-02	0.0555
	Hole # 4	2.679E-02	0.0112	3.084E-02	0.0987	5.763E-02	0.0531
	Hole # 5	2.687E-02	0.0112	3.239E-02	0.0995	5.926E-02	0.0546
	Hole # 6	2.656E-02	0.0112	3.314E-02	0.1091	5.970E-02	0.0608
	18 in. Radius	1.967E-02	0.0013	3.117E-02	0.0144	5.083E-02	0.0088
	Inner Vessel ID	3.773E+01	0.0407	3.116E-02	0.0100	3.776E+01	0.0406
Inner Vessel Top Exterior Surface	Hole # 1	1.282E-03	0.0111	1.618E-02	0.1406	1.747E-02	0.1303
	Hole # 2	1.280E-03	0.0112	1.435E-02	0.1259	1.563E-02	0.1156
	Hole # 3	1.276E-03	0.0111	1.415E-02	0.1379	1.542E-02	0.1265
	Hole # 4	1.296E-03	0.0111	1.194E-02	0.1180	1.324E-02	0.1065
	Hole # 5	1.304E-03	0.0111	1.495E-02	0.1020	1.626E-02	0.0938
	Hole # 6	1.305E-03	0.0110	1.234E-02	0.1416	1.364E-02	0.1281
	18 in. Radius	1.073E-01	0.7270	1.366E-02	0.0154	1.209E-01	0.6449
	Outer Corrosion Barrier OD	5.192E+00	0.0517	1.795E-02	0.0079	5.210E+00	0.0515
Outer Lid Exterior Surface	Hole # 1	4.38E+00	0.7181	7.760E-03	0.1539	4.388E+00	0.7168
	Hole # 2	3.16E+00	0.3634	1.101E-02	0.2029	3.173E+00	0.3622
	Hole # 3	1.55E+00	0.4346	8.702E-03	0.1921	1.560E+00	0.4322
	Hole # 4	1.10E+00	0.6111	8.338E-03	0.1338	1.105E+00	0.6065
	Hole # 5	7.35E+00	0.5832	1.011E-02	0.2375	7.364E+00	0.5824
	Hole # 6	9.524E+00	0.4934	7.469E-03	0.1411	9.531E+00	0.4930
	18 in. Radius	2.775E+00	0.0467	7.964E-03	0.0158	2.783E+00	0.0466
	Outer Corrosion Barrier OD	3.177E+00	0.0257	9.332E-03	0.0081	3.186E+00	0.0256
	Segment 9	5.77E+01	0.0041	2.264E-02	0.0021	5.773E+01	0.0041
1 m from Outer Lid Exterior Surface	Segment 8	8.15E+00	0.0097	9.05E-03	0.0059	8.154E+00	0.0097
	Segment 9	1.25E+01	0.0044	1.08E-02	0.0029	1.248E+01	0.0044
2 m from Outer Lid Exterior Surface	Segment 8	7.84E+00	0.009	8.21E-03	0.0058	7.850E+00	0.0090
	Segment 10	1.02E+01	0.0029	9.20E-03	0.0019	1.023E+01	0.0029

Source: BSC 2004 [DIRS 167082], Table 6.1-6

Table 28. Dose Rate on the Waste Package Inner Vessel Inner Surface T = 5 Years

	Gamma		Neutron		Total	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	45607.5	0.0011	0.662	0.002	45608.2	0.0011
Segment 2	48324.7	0.0011	0.711	0.001	48325.4	0.0011
Segment 3	48210.8	0.0011	0.710	0.001	48211.5	0.0011
Segment 4	48214.9	0.0011	0.711	0.001	48215.6	0.0011
Segment 5	48286.5	0.0011	0.707	0.001	48287.2	0.0011
Segment 6	48252.1	0.0011	0.694	0.002	48252.8	0.0011

Source: BSC 2004 [DIRS 167082], Table 6.1-7

Table 29. Dose Rate on the Waste Package Inner Vessel Outer Surface T = 5 Years

	Gamma		Neutron		Total	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	941.7	0.0019	0.231	0.002	941.9	0.0019
Segment 2	1000.1	0.0019	0.252	0.002	1000.3	0.0019
Segment 3	1000.1	0.0019	0.251	0.002	1000.4	0.0019
Segment 4	997.7	0.0019	0.252	0.002	998.0	0.0019
Segment 5	1001.9	0.0019	0.251	0.002	1002.2	0.0019
Segment 6	981.8	0.0019	0.239	0.002	982.0	0.0019

Source: BSC 2004 [DIRS 167082], Table 6.1-8

Table 30. Dose Rate on the Waste Package Outer Corrosion Barrier Outer Surface T = 5 Years

	Gamma		Neutron		Total	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	167.8	0.0025	0.095	0.002	167.9	0.0025
Segment 2	179.0	0.0024	0.104	0.001	179.1	0.0024
Segment 3	179.3	0.0024	0.104	0.001	179.4	0.0024
Segment 4	178.5	0.0024	0.104	0.001	178.6	0.0024
Segment 5	178.8	0.0024	0.104	0.001	178.9	0.0024
Segment 6	174.6	0.0024	0.099	0.002	174.7	0.0024

Source: BSC 2004 [DIRS 167082], Table 6.1-9

Table 31. Dose Rate 1 m from Waste Package Outer Corrosion Barrier Outer Surface T = 5 years

	Gamma		**Neutron**		**Total**	
Axial Location	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Segment 1	57.1	0.0022	0.033	0.002	57.1	0.0022
Segment 2	73.6	0.0019	0.041	0.001	73.6	0.0019
Segment 3	76.4	0.0019	0.044	0.001	76.5	0.0019
Segment 4	76.3	0.0019	0.044	0.001	76.4	0.0019
Segment 5	74.1	0.0019	0.041	0.001	74.1	0.0019
Segment 6	58.6	0.0022	0.034	0.002	58.6	0.0022

Source: BSC 2004 [DIRS 167082], Table 6.1-10

Table 32. Dose Rates 2 m from the Waste Package Outer Corrosion Barrier Outer Surface T = 5 years

	Gamma		**Neutron**		**Total**	
Axial Location	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Segment 1	35.1	0.0021	0.023	0.002	35.2	0.0021
Segment 2	45.9	0.0018	0.028	0.001	45.9	0.0018
Segment 3	49.7	0.0017	0.031	0.001	49.8	0.0017
Segment 4	49.8	0.0017	0.031	0.001	49.8	0.0017
Segment 5	46.1	0.0018	0.029	0.001	46.1	0.0018
Segment 6	35.8	0.0021	0.024	0.002	35.8	0.0021

Source: BSC 2004 [DIRS 167082], Table 6.1-11

Table 33. Dose Rates on the Waste Package Surface T = 5 years

		Primary Gamma		Neutron		Total	
Axial Location	**Segment**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**	**Dose Rate rem/hr**	**Relative Error**
Lower Inner Vessel Bottom Surface	Segment 7	1.820E+01	0.0170	7.384E-02	0.0035	1.820E+01	0.0170
Outer Corrosion Barrier Bottom Surface	Segment 8	4.077E+00	0.0173	3.070E-02	0.0036	4.109E+00	0.0172
	Segment 9	3.294E+01	0.0041	2.623E-02	0.0021	3.294E+01	0.0041
1 m from Outer Corrosion Barrier Bottom	Segment 8	4.729E+00	0.0095	1.554E-02	0.0041	4.745E+00	0.0095
	Segment 9	6.556E+00	0.0046	1.382E-02	0.0025	6.561E+00	0.0046
2 m from Outer Corrosion Barrier Bottom	Segment 8	4.178E+00	0.0091	1.120E-02	0.0046	4.191E+00	0.0091
	Segment 10	5.177E+00	0.0031	1.090E-02	0.0018	5.181E+00	0.0031
Top of Waste Package Cavity	Hole # 1	4.515E-03	0.0101	2.763E-02	0.0946	1.562E-02	0.0813
	Hole # 2	4.467E-03	0.0102	2.772E-02	0.1110	1.567E-02	0.0956
	Hole # 3	4.484E-03	0.0102	3.278E-02	0.1002	1.578E-02	0.0882
	Hole # 4	4.508E-03	0.0102	3.084E-02	0.0987	1.571E-02	0.0861
	Hole # 5	4.473E-03	0.0102	3.239E-02	0.0995	1.567E-02	0.0874
	Hole # 6	4.499E-03	0.0102	3.314E-02	0.1091	1.570E-02	0.0961
	18 in. Radius	3.174E-03	0.0012	3.117E-02	0.0144	4.474E-03	0.0131
	Inner Vessel ID	1.788E+01	0.0412	3.116E-02	0.0100	1.788E+01	0.0411
Inner Vessel Top Exterior Surface	Hole # 1	1.982E-04	0.0106	1.618E-02	0.1406	1.130E-02	0.1389
	Hole # 2	1.948E-04	0.0107	1.435E-02	0.1259	1.139E-02	0.1242
	Hole # 3	1.966E-04	0.0108	1.415E-02	0.1379	1.130E-02	0.1360
	Hole # 4	1.999E-04	0.0107	1.194E-02	0.1180	1.130E-02	0.1161
	Hole # 5	2.001E-04	0.0107	1.495E-02	0.1020	1.130E-02	0.1007
	Hole # 6	1.954E-04	0.0108	1.234E-02	0.1416	1.120E-02	0.1394
	18 in. Radius	7.766E-02	0.6881	1.366E-02	0.0154	7.916E-02	0.5852
	Outer Corrosion Barrier OD	2.278E+00	0.0529	1.795E-02	0.0079	2.279E+00	0.0525
Outer Lid Exterior Surface	Hole # 1	7.075E-01	0.3810	7.760E-03	0.1539	7.231E-01	0.3768
	Hole # 2	2.821E+00	0.6027	1.101E-02	0.2029	2.837E+00	0.6004
	Hole # 3	6.720E-01	0.3730	8.702E-03	0.1921	6.877E-01	0.3682
	Hole # 4	5.528E-01	0.3754	8.338E-03	0.1338	5.684E-01	0.3698
	Hole # 5	1.983E+00	0.7514	1.011E-02	0.2375	1.999E+00	0.7476
	Hole # 6	2.250E+00	0.5052	7.469E-03	0.1411	2.266E+00	0.5035
	18 in. Radius	1.318E+00	0.0492	7.964E-03	0.0158	1.320E+00	0.0489
	Outer Corrosion Barrier OD	1.464E+00	0.0255	9.332E-03	0.0081	1.465E+00	0.0253
	Segment 9	2.630E+01	0.0041	2.264E-02	0.0021	2.630E+01	0.0041
1 m from Outer Lid Exterior Surface	Segment 8	3.870E+00	0.0101	9.047E-03	0.0059	3.871E+00	0.0101
	Segment 9	5.829E+00	0.0046	1.079E-02	0.0029	5.830E+00	0.0046
2 m from Outer Lid Exterior Surface	Segment 8	3.691E+00	0.0093	8.211E-03	0.0058	3.692E+00	0.0093
	Segment 10	4.793E+00	0.0030	9.201E-03	0.0019	4.794E+00	0.0030

Source: BSC 2004 [DIRS 167082], Table 6.1-12

Table 34 and Table 35 from BSC (2004 [DIRS 167082]) highlight the magnitude of the increase in the dose on the top and radial exterior waste package surfaces resulting from photon backscatter, from the concrete structure at T = 2 years. The waste package surface dose rates due to secondary gamma rays are negligible as compared to the gamma dose rates; consequently, this dose component was not listed in the following tables.

Table 34. Photon Backscatter Effects on the Outer Lid Exterior Surface (Top)

Total Gamma Dose (Scatter)			Total Gamma Dose (No Scatter)	
	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 9	57.7	0.0041	47.9	0.0044

Source: BSC 2004 [DIRS 167082], Tables 6.2-1

Table 35. Photon Backscatter Effects on the Outer Corrosion Barrier Exterior Surface (Sides)

Total Gamma Dose (Scatter)			Total Gamma Dose (No Scatter)	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Segment 1	368.1	0.0025	360.8	0.0024
Segment 2	391.5	0.0024	383.8	0.0023
Segment 3	392.2	0.0024	383.6	0.0023
Segment 4	390.8	0.0024	381.8	0.0023
Segment 5	391.9	0.0024	384.8	0.0023
Segment 6	383.4	0.0025	376.0	0.0023

Source: BSC 2004 [DIRS 167082], Tables 6.2-2

Table 36 shows the photon backscattering effects on the waste package bottom exterior surface.

Table 36. Photon Backscatter Effects on the WP Bottom Exterior Surface

Total Gamma Dose (Scatter)			Total Gamma Dose (No Scatter)	
Axial Location	Dose Rate rem/hr	Relative Error	Dose Rate rem/hr	Relative Error
Outer Corrosion Barrier OD	1.522	0.0008	1.517	0.0007
Segment 8	7.870	0.0194	4.961	0.0255

Source: BSC 2004 [DIRS 167082], Tables 6.2-3

7.1.1.3.3 Dose Rate Conclusions

In this section BSC (2004 [DIRS 167082], Section 6.2) draws the following conclusions regarding the naval waste package.

The primary gamma and neutron components of the total dose rates have been calculated for a number of surfaces on the interior and exterior of the waste package, as well as at 1 and 2 meters away from the waste package exterior. The tabulated dose rate results are suitable to support the waste package and repository facility designs.

The maximum external dose rate is 392.2 rem/hr, at 2 years after reactor shutdown. This occurs at segment 3 of the outer corrosion barrier exterior radial surface as presented in Table 24. The exterior radial surface dose rate varies from 392.2 to 368.1 rem/hr across the height of the naval waste package interior cavity (see Table 24). The dose rates on the waste package bottom and top contact surfaces are 9.423 rem/hr (see Table 27) and 3.186 rem/hr, respectively (see Table 27).

The maximum external dose rate is 179.4 rem/hr, at 5 years after reactor shutdown. This occurs at segment 3 of the outer corrosion barrier exterior radial surface as presented in Table 30. The exterior radial surface dose rate varies from 179.4 to 167.9 rem/hr across the height of the naval long waste package interior cavity (see Table 30). The dose rates on the waste package bottom and top contact surfaces are 4.109 rem/hr and 1.465 rem/hr, respectively (see Table 33).

The presence of concrete augments the top, bottom, and the radial exterior surface dose rates due to photon backscattering from the surrounding concrete walls.

The top and bottom waste package exterior surface dose rates are influenced the greatest by photon backscatter, as seen in Table 34 and Table 36, mainly as a result of back-scattering from the side-walls around the waste package.

As a result of photon back-scatter, the radial exterior surface dose rates increase from 384.8 – 360.8 rem/hr across the height of the naval long waste package interior cavity (see Table 35) to 392.2 – 368.1 rem/hr (see Table 35).

The secondary gamma component is negligible with respect to the primary photon component of the total dose rates.

7.1.1.4 Waste Form Accommodation

There are two naval SNF canisters that must be accommodated by waste package design configurations. The naval canisters have the same 1.689-m (66.5-in) diameter with different lengths. The short canister is 4.750 m (187.0 in.) in length and the long canister 5.385 m (212.0 in.) in length (BSC 2003 [DIRS 165427], Section 3.1.2, p. 15, Performance Requirements 1 and 2). Two waste package configurations have been designed and developed. The naval SNF short waste package accommodates the naval short canister and the naval long SNF long waste package accommodates the long canister. The two packages, differing only in length, are identical in materials, diameter, lid, and weld details.

These dimensions may be verified on the following configuration drawings:

- *Naval Long Waste Package Configuration* (BSC 2003 [DIRS 164854]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165158]) and BSC (2003 [DIRS 165159]), respectively.

- *Naval Short Waste Package Configuration* (BSC 2003 [DIRS 164855]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165160]) and BSC (2003 [DIRS 165161]), respectively.

7.1.1.5 Criticality

Due to the classified nature of naval reactor fuel, no criticality calculations will be performed by the Waste Package and Components group. The addendum to *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) outlines the criticality methodology used by the Naval Nuclear Propulsion Program. Any assumptions concerning criticality are beyond the scope of this document

7.1.2 Preclosure Event Sequences

7.1.2.1 Thermal

Thermal performance of a naval package during a fire is addressed by the Naval Nuclear Propulsion Program independently from Waste Package and Components.

7.1.2.2 Structural

The waste package is a component identified as important to safety (BSC 2003 [DIRS 165179], Table A-2, p. A-4) since it provides containment for the waste forms. The waste package is credited to prevent a release, in terms of dose to workers and the public during the preclosure period. Therefore, the waste package is designed to a set of criteria to ensure that the waste package will not breach as a result of credible event sequences.

The waste package design is evaluated using a finite element analysis based on numerical simulations of waste package dynamic events including, but not limited to, vertical and horizontal drops, slapdowns, drops onto objects, collisions, and equipment drops onto the waste package.

The failure criterion used is explained in detail in Section 7.1.2.3.1.2 and is broken into tiered screening criteria shown below. The easiest to apply and most conservative criteria are applied initially. If these can not be met, less conservative screening criteria are imposed that require more calculations. These screening criteria in decreasing order of conservatism are listed below (an element's total stress intensity, σ_{int}, is equal to twice the element's maximum shear stress (ASME 2001 [DIRS 158115], Section III, Division 1, NB-3000)):

Maximum $\sigma_{int} < 0.7\sigma_u$?	Yes: Meets P_m and P_L limits without the need for wall averaging.
No:	
Maximum $\sigma_{int} < 0.77\sigma_u$?	Yes: Meets P_L limit without the need for wall averaging but the stress field must not be uniform

	around the entire circumference (only a concern for vertical drop events).
No:	
Maximum wall-averaged $\sigma_{int} < 0.7\ \sigma_u$?	Yes: Meets P_m and P_L limits.
No:	
Maximum wall-averaged $\sigma_{int} < 0.77\sigma_u$?	Yes: Meets P_L limit if the stress fields are not uniform around the entire circumference (only a concern for vertical drop events).
No:	
Maximum wall-averaged $\sigma_{int} < 0.84\ \sigma_u$ <u>and</u> wall-averaged $\sigma_{int} < 0.77\ \sigma_u$ at $\sqrt{R \cdot t}$ surrounding maximum location?	Yes: Meets P_L and average primary shear limit
No:	
Maximum wall-averaged $\sigma_{int} < 0.9\ \sigma_u$ <u>and</u> wall-averaged $\sigma_{int} < 0.77\ \sigma_u$ at $\sqrt{R \cdot t}$ surrounding maximum location? <u>and</u> wall-average of each shear stress on the stress classification line (τ_{xy}, τ_{yz} and τ_{xz}) $< 0.42\sigma_u$? (x,y,z are element (not global) directions orthogonal to the stress classification line)	Yes: Meets P_L and average primary shear limit

No: Fails simplified screening criterion.

Note: P_m is the general primary membrane stress
P_L is the local primary membrane
P_b is the primary bending
R is the median wall radius
t is the wall thickness.

If the wall-averaged σ_{int} limits can not be met, perform a less conservative rigorous Code evaluation using all six stress components (and solve a cubic equation for principle stress direction values) and/or use multiple stress classification lines to extrapolate to governing wall locations when they have significant non-membrane primary stress intensity contributions.

If the average primary shear limit can not be met, then review appropriateness of using a stress classification plane rather than a stress classification line.

7.1.2.2.1 Preclosure Rock Fall Evaluations

Rock falls may occur both in the preclosure and postclosure periods. For the preclosure period, the drip shields have not yet been emplaced, so rocks may fall onto the emplaced waste packages. *Rock Fall on Waste Packages* (BSC 2004 [DIRS 167182]) investigates four waste package configurations for license application (21-PWR waste package, 44-BWR waste package, 5-DHLW/DOE SNF codisposal short waste package, and Naval Canistered SNF long waste package) to determine their structural response to rock fall dynamic loads (Minwalla 2003 [DIRS 161362], Table 5.1.1-1. Furthermore, four different axial locations of impact are evaluated. The first impact location is selected at the waste package bottom-end, directly above the trunnion-to-bottom lid fillet-weld region; second, at mid-length; third, directly above the emplacement pallet support; and fourth, at the waste package upper-end, directly above the closure-weld region. These locations are critical to the waste package.

One waste package configuration is used to determine the most damaging location of impact among the four impact locations. The results of these calculations suggest that the most damaging results are obtained when the axial location of the initial impact zone is directly above the trunnion collar sleeve, at the bottom lid fillet-weld region. Consequently, the rock fall calculations for all waste package configurations are performed for this impact location.

Single rock fall evaluations have been performed for the waste package, resting on the emplacement pallet, with one bounding rock size (6.8 MT [15,000 lbs.]) and initial impact velocity (6 m/s [20 ft/s]). An additional scenario is performed to determine the effect of the greatest initial impact velocity (1.3 MT [2900 lbs.], 14 m/s [46 ft/s]). The results show that the rock with the greatest mass causes higher stresses than the rock with the greatest initial impact velocity.

BSC (2004 [DIRS 167182]) also determines the response of waste package components to multiple rock falls onto the same location. The main purpose of this study is to determine the effect of two rocks on the stress intensity history when they impact the same location on the waste package. For this purpose, a 21-PWR waste package configuration is selected. This waste package configuration is subjected to the highest stress intensity for the bounding impact location; therefore, the results of this case are bounding for all waste package configurations. Similar to the single rock fall evaluations, the bottom end of the waste package is impacted by two identical rocks (3.0 MT [6600 lbs.], 5.9 m/s [19 ft/s]).

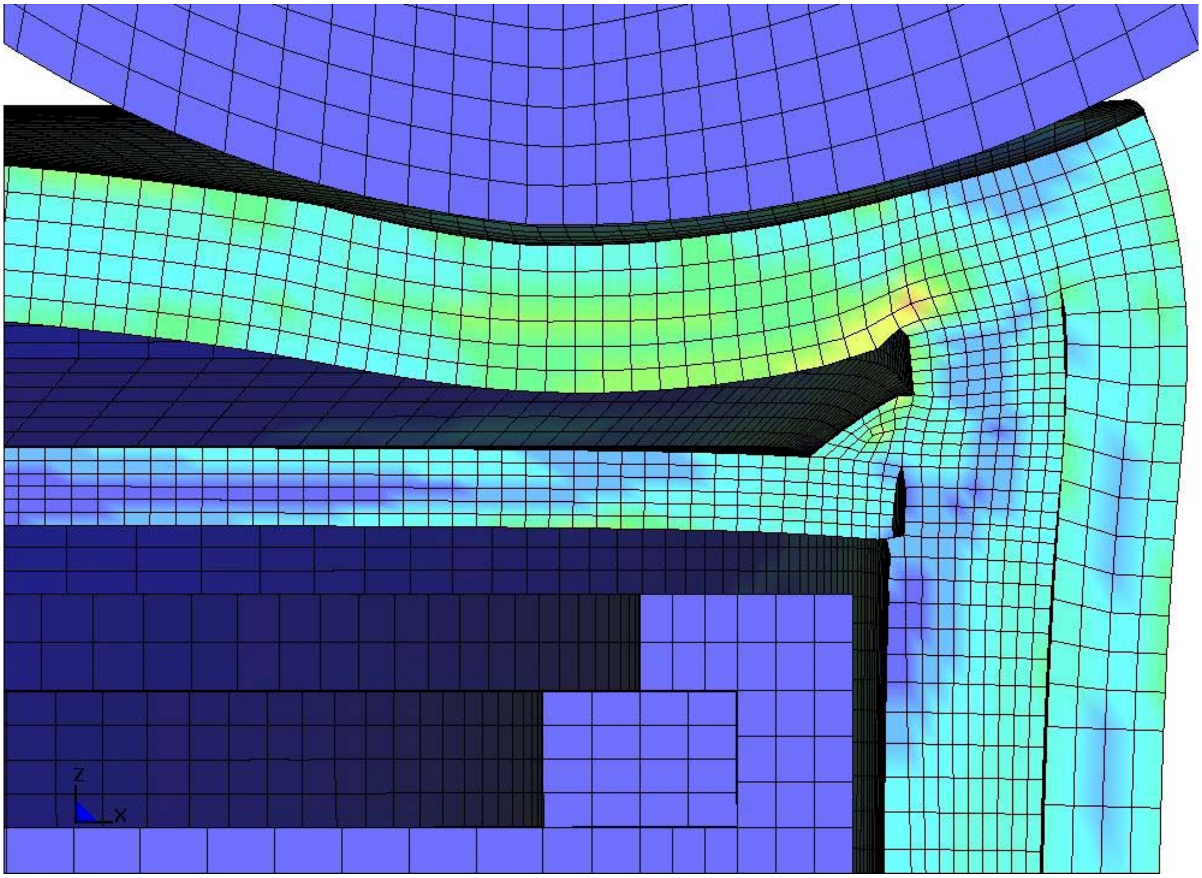
For the simulation of the rock fall onto a corroded waste package, the thickness of the Alloy 22 (UNS N06022) components is appropriately reduced, based on the calculation of the depth of corroded the layer presented in BSC (2004 [DIRS 167182], Section 5.3).

The results of the rock fall evaluations indicate that for all rock impact simulations, the maximum stress intensity in the outer corrosion barrier and outer lids is less than 70% of the ultimate tensile strength of Alloy 22 (UNS N06022) at maximum temperatures during their presence in the repository (BSC 2004 [DIRS 167182], Section 6). Therefore, no breach of the waste package is expected from preclosure rock fall.

7.1.2.2.2 Object Drop on Naval Long Waste Package

The object drop (see Section 6.1.2.2 of this document) consists of raising a hook that is used to lift the waste package directly above a vertically standing waste package. The hook is raised to a maximum height of 9.1 m (30 ft). The lifting device with the hook attached, fails, and the hook falls due to gravity. The hook then impacts the waste package top surface. *Object Drop on Naval Long Waste Package* (BSC 2003 [DIRS 165895]) performs this calculation at room temperature and 300°C to bound potential waste package operational temperatures.

Figure 7 outlines the shape of the impact region at the end of the simulation for a 1,361-kg (3,000-lb.) hook drop from 9.1 m (30 ft) at two different temperatures. Note that the middle lid remains to be directly supported by the outer corrosion barrier. Consequently, there is no contact between the middle lid and the inner vessel.



Source: (BSC 2003 [DIRS 165895], Figure 11, p. 30)

Figure 7. Detail of the Waste Package and Hook at the End of the Drop

It is important to emphasize that there are no contacts between the Alloy 22 (UNS N06022) and the stainless steel type 316 (UNS S31600) waste package components. Therefore, it can be concluded that the hook drop does not affect the containment capabilities of the inner vessel.

7.1.2.2.3 Pressurized System Missile Impact on Waste Package

Four different waste package design configurations are evaluated in *Pressurized System Missile Impact on Waste Packages* (CRWMS M&O 2000 [DIRS 149351]): 21-PWR, 44-BWR, 5-DHLW/DOE SNF Short, and Naval SNF Long waste packages. For each one of these waste package design configurations, a parametric study is performed by reporting the results for different missile diameter, mass, and velocities. These parameters are provided in Table 37.

Table 37. Missile Impact Parameters for Three Different Case Studies

	Case 1	Case 2	Case 3
Missile diameter	10 mm	20 mm	30 mm
Missile mass	0.5 kg	1.0 kg	1.5 kg
Missile velocity	5.7 m/s	6.0 m/s	6.3 m/s

Source: CRWMS M&O 2000 [DIRS 149351], Table 5.2-1, p. 4

The structural response of the waste package to dynamic impact of a pressurized system missile is reported in terms of the minimum velocities required for a pressurized system missile to cause perforation of the waste package barriers. The calculation results are summarized in Table 38.

Table 38. Pressurized System Missile Impact Results for Different Waste Packages

Waste Package	Minimum Required Velocity of Projectile to Cause Perforation (m/s)		
	Case1	Case 2	Case 3
21-PWR	322	383	424
44-BWR	322	383	424
5-DHLW/DOE SNF Short	339	403	446
Naval SNF Long	339	403	446

Source: CRWMS M&O 2000 [DIRS 149351], Table 6-1, p. 7.

7.1.2.2.4 Vertical Drop of Naval Long Waste Package

The vertical drop consists of raising the waste package vertically to a maximum height of 2.0 m (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). The lifting device carrying the waste package fails and the waste package falls due to gravity. The waste package then impacts an unyielding surface. *Vertical Drop of Naval SNF Long Waste Package* (BSC 2003 [DIRS 161690]) performs this calculation at room temperature and 100°C to bound potential waste package operational temperatures. The results for the vertical drop of the naval long waste package are shown in Table 39.

Table 39 lists the maximum stress intensities, σ_{int}, in the inner vessel and outer corrosion barriers of the Naval SNF Long waste package at room temperature and 100°C.

Table 39. Maximum Stress Intensities in the Inner Vessel and Outer Corrosion Barrier

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Inner Vessel	RT	461	0.66
Outer Corrosion Barrier	RT	1084	1.12
Inner Vessel	100°C	456	0.69
Outer Corrosion Barrier	100°C	1002	1.02

NOTE: RT = room temperature.

Source: BSC 2003 [DIRS 161690], Table 8, p. 24

The preceding table shows that the inner vessel's maximum stress intensities were less than 70 percent of the tensile strengths at both temperatures, indicating that the inner vessel remained intact during this drop scenario. For added containment the naval SNF canister provides an additional level of protection. Therefore, it can be concluded that the waste package containment capabilities remain effective.

7.1.2.2.5 Naval Long Tip-over from Elevated Surface

The tip-over from an elevated surface consists of the waste package toppling from a 0.5-m (1.6-ft) pedestal (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). The waste package tips about its bottom edge to the point at which the center of gravity is directly above the rotation point. The waste package then continues to tip over due to gravity. The waste package then impacts an unyielding surface with the top edge. *Naval SNF Long Waste Package Tip-Over from an Elevated Surface* (BSC 2004 [DIRS 167079]) performs this calculation at room temperature and 300°C to bound potential waste package operational temperatures. The results for the tip-over from an elevated surface of the naval long waste package are shown in Table 40.

Table 40. Maximum Stress Intensity in the Outer Corrosion Barrier

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Outer Corrosion Barrier	RT	852	0.88
Outer Corrosion Barrier	300°C	726	0.80

NOTE: RT = room temperature.
Source: BSC 2004 [DIRS 167079], Section 6

According to Table 40, the maximum wall-averaged stress intensities in the outer corrosion barrier are below nine-tenths of the true tensile strength of Alloy 22 (UNS N06022). For both temperatures the maximum wall-averaged stress intensities exceed $0.77\sigma_u$ and a study is needed of the size of the region where the wall-averaged stress intensities exceed $0.77\sigma_u$. The distance over which wall-averaged stress intensities exceed $0.77\sigma_u$ is 0.131, which is less than $\sqrt{R \cdot t}$ (where R is the average radius of the outer corrosion barrier and t is the wall thickness). Therefore, for both temperature scenarios the outer corrosion barrier meets the criteria set forth in Section 7.1.2.2 of this document. Therefore no breach of the outer corrosion barrier is expected for a tip-over from a 0.5-m (1.6-ft) pedestal. For a tip-over from an elevated surface, there is greater potential energy associated with this scenario; consequently, this case is bounding for the event sequence involving a waste package tip-over while at rest on a flat surface.

7.1.2.2.6 Naval Long Waste Package Horizontal Drop on Flat Surface

The horizontal drop consists of raising the waste package horizontally to a maximum height of 2.4 m (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). The lifting device carrying the waste package fails and the waste package falls due to gravity. The waste package then impacts an unyielding surface. *Horizontal Drop of Naval SNF Long Waste Package* (BSC 2004 [DIRS 167078]) performs the calculation at room temperature and 300°C to bound potential waste package operational temperatures. The results for the horizontal drop of the naval long waste package are shown in Table 41. This table contains maximum wall-averaged stress intensities in the outer corrosion barrier at room temperature and 300°C and a comparison of these stresses to the true tensile strength of Alloy 22 (UNS N06022).

Table 41. Wall-averaged Stress Intensity in the Outer Corrosion Barrier

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Outer Corrosion Barrier	RT	802	0.83
Outer Corrosion Barrier	300°C	700	0.77

NOTE: RT = room temperature.
Source: BSC 2004 [DIRS 167078], Table 6-4

From Table 41 the maximum wall-averaged stress intensities in the outer corrosion barrier are below nine-tenths of the true tensile strength of Alloy 22 (UNS N06022). Furthermore, at 300°C maximum wall-averaged stress intensities do not exceed 0.77σ_u. Therefore, at 300°C the outer corrosion barrier meets the plastic analysis criteria (see Section 7.1.2.2). However, at RT the maximum wall-averaged stress intensity exceeds 0.77σ_u and a study is needed to determine the size of the region where the wall-averaged stress intensities exceed 0.77σ_u. The distance over which wall-averaged stress intensities exceed 0.77σ_u is 0.152 which is less than $\sqrt{R \cdot t}$ (where *R* is the average radius of the outer corrosion barrier and *t* is the wall thickness). Therefore, for the room temperature case, the outer corrosion barrier meets the criteria set forth in Section 7.1.2.2 of this document. Consequently, no breach of the outer corrosion barrier is expected.

7.1.2.2.7 Naval Long Waste Package Horizontal Drop with Emplacement Pallet

The horizontal drop with emplacement pallet consists of raising the waste package horizontally while resting on the emplacement pallet to a maximum height of 0.782 m (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). The lifting device carrying the waste package and emplacement pallet fails and the two fall due to gravity. The waste package then impacts an unyielding surface with the emplacement pallet first. This scenario is also performed as a waste package horizontal drop onto the emplacement pallet from a maximum height of 2.4 m. *Naval Waste Package Drop with Emplacement Pallet* (BSC 2002 [DIRS 162346]) performs these calculations at room temperature and 100°C to bound potential waste package operational temperatures. The results for the naval long waste package drop with the emplacement pallet are shown below.

Table 42 presents the maximum stress intensity in the waste package outer corrosion barrier and inner vessel for each of the temperature conditions while loaded with the naval long waste package from a 0.782-m drop.

Table 43 shows the maximum stress intensity in the waste package outer corrosion barrier and inner vessel from a 2.4-m drop onto the emplacement pallet.

Table 42. Maximum Stress Intensity for a Drop from 0.782 m while Loaded on the Emplacement Pallet

Part	**Temperature**	**Max Stress Intensity, σ_{int} (MPa)**	**σ_{int} / σ_u**
Inner Vessel	RT	244	0.35
Outer Corrosion Barrier	RT	308	0.32
Inner Vessel	100°C	212	0.32
Outer Corrosion Barrier	100°C	302	0.31

NOTE: RT = room temperature.
Source: BSC 2002 [DIRS 162346], Table 6-3, p. 18

Table 43. Maximum Stress Intensity for a Drop from 2.4 m onto the Emplacement Pallet

Part	**Temperature**	**Max Stress Intensity, σ_{int} (MPa)**	**σ_{int} / σ_u**
Inner Vessel	RT	234	0.33
Outer Corrosion Barrier	RT	340	0.35
Inner Vessel	100°C	202	0.30
Outer Corrosion Barrier	100°C	302	0.31

NOTE: RT = room temperature.
Source: BSC 2002 [DIRS 162346], Table 6-4, p. 18

As a result of the two horizontal drop scenarios the emplacement pallet experiences stresses in excess of its yielding point in all cases. These stresses range from 692 MPa (0.782-m drop at 100°C) to 934 MPa (2.4-m drop at room temperature) (BSC 2002 [DIRS 162346], Section 6, p. 18). This yielding is unimportant, as it does not cause breaching in the waste package, and the damaged emplacement pallet would be replaced before emplacement in the drift. The deformation of the emplacement pallet absorbs a tremendous amount of energy and lowers the stresses in the waste package considerably by this yielding during the impact. According to Table 42 and Table 43 the stresses in the inner vessel and outer corrosion barrier are below seven tenths of the corresponding material ultimate tensile strengths in all scenarios. Therefore, no breach of the waste package is expected.

7.1.2.2.8 Corner Drop of the Naval Long Waste Package

The corner drop consists of raising the waste package vertically to a maximum height of 2.0 m (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). At this time the waste package swings possibly due to seismic occurrences. The lifting device carrying the waste package fails when the waste package has swung to the point where the center of gravity is vertically aligned with the bottom corner of the waste package. The waste package then falls due to gravity and the waste package impacts an unyielding surface. *Corner Drop of Naval Canistered SNF Long Waste Package* (BSC 2004 [DIRS 167165]) performs this calculation at room temperature and

300°C to bound potential waste package operational temperatures. The maximum wall-averaged stress intensities for the corner drop of the naval long waste package from 2.0 m are shown in Table 44.

Table 44. Wall-averaged Stress Intensities for a 2-m Drop

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Inner Vessel	RT	443	0.63
Outer Corrosion Barrier	RT	941	0.97
Inner Vessel	300°C	364	0.59
Outer Corrosion Barrier	300°C	871	0.96

NOTE: RT = room temperature.
Source: BSC 2004 [DIRS 167165], Table 6-3, p. 26

From Table 44, the wall-averaged stress intensity values in the inner vessel are below seven tenths of the true ultimate tensile strength of stainless steel type 316 (UNS S31600). The wall-averaged stress intensity values in the outer corrosion barrier are above nine tenths of the ultimate tensile strength for Alloy 22 (UNS N06022). Therefore, the outer corrosion barrier does not meet the plastic analysis criteria for a 2-m corner drop (see Section 7.1.2.2).

Since the outer corrosion barrier does not meet the plastic analysis criteria defined in Section 7.1.2.2, a 1-m drop at room temperature is performed to determine an allowable lifting height of the naval waste package. For a 1-m drop, only a room temperature corner drop is performed, since the room temperature case from Table 44 has proven to be the more bounding case of the two temperature conditions for 2-m drop. The 1-m drop results are presented in Table 45.

Table 45. Wall-averaged Stress Intensities for a 1-m Drop

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Outer Corrosion Barrier	RT	388	0.40

NOTE: RT = room temperature.
Source: BSC 2004 [DIRS 167165], Table 6-7, p. 28

From Table 45, the wall-averaged stress intensity of the outer corrosion barrier is significantly below 70 percent of the ultimate tensile strength for Alloy 22 (UNS N06022). Therefore, for a 1-m corner drop of the naval waste package, no breach is expected.

7.1.2.2.9 Naval Long Waste Package 10-Degree Oblique Drop with Slap Down

The 10-degree oblique drop with slap down consists of raising the waste package horizontally to a maximum height of 2.4 m (see Section 6.1.1.2 of this document). The lifting device carrying the bottom half of the waste package fails, and the bottom half of the waste package begins to fall due to gravity. After the bottom end has rotated 10 degrees, the lifting device holding the top of the waste package fails and the entire waste package falls due to gravity. The waste package then impacts an unyielding surface with the bottom edge first followed by the top end. *Naval SNF Long Waste Package 10 Degree Oblique Drop with Slap Down* (BSC 2003 [DIRS 167081]) performs the calculation at room temperature and 300°C to bound potential waste package operational temperatures. The results for the 10-degree oblique drop with slap down of the naval long waste package are shown in Table 46, presenting the maximum stress intensity at room temperature and 300°C.

Table 46. Maximum Stress Intensity

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Outer Corrosion Barrier	RT	715	0.74
Inner Vessel	RT	532	0.76
Inner Lid	RT	396	0.56
Spread Ring	RT	279	0.40
Outer Corrosion Barrier	300°C	572	0.63
Inner Vessel	300°C	475	0.76
Inner Lid	300°C	370	0.59
Spread Ring	300°C	204	0.33

NOTE: RT = room temperature.
Source: BSC 2003 [DIRS 167081], p. 19, Table 6-2

For both temperature conditions the stresses in the outer corrosion barrier and inner vessel are less than 77 percent of the corresponding material ultimate tensile strength. Therefore no breach of the waste package is expected.

7.1.2.2.10 Naval Long Waste Package Swing Down

The swing down consists of raising the waste package horizontally to a maximum height of 2.4 m (Mecham 2004 [DIRS 166168], Section 5.2.8, p. 54). The lifting device carrying the top half of the waste package fails and the top of the waste package falls due to gravity. The waste package then impacts an unyielding surface with the top edge. *Swing-Down of Naval Long Waste Package* (BSC 2002 [DIRS 161880]) performs this calculation at room temperature, 204°C, and 316°C to bound potential waste package operational temperatures. The results for the swing down of the naval long waste package are shown in Table 47.

Table 47. Maximum Stress Intensity

Part	Temperature	Max Stress Intensity, σ_{int} (MPa)	σ_{int} / σ_u
Inner Vessel	RT	487	0.69
Inner Vessel Upper Lid	RT	270	0.38
Spread Ring	RT	329	0.46
Outer Corrosion Barrier	RT	541	0.59
Inner Vessel	204°C	480	0.71
Inner Vessel Upper Lid	204°C	217	0.32
Spread Ring	204°C	300	0.44
Outer Corrosion Barrier	204°C	479	0.52
Inner Vessel	316°C	466	0.75
Inner Vessel Upper Lid	316°C	203	0.33
Spread Ring	316°C	273	0.44
Outer Corrosion Barrier	316°C	459	0.52
Inner Vessel	316°C [a]	516	0.86
Inner Vessel Upper Lid	316°C [a]	270	0.45
Spread Ring	316°C [a]	305	0.51
Outer Corrosion Barrier	316°C [a]	452	0.50

NOTE: RT = room temperature.
[a] With modified elongation.
Source: BSC 2002 [DIRS 161880], Table 6-3, p. 43

In all temperature conditions the outer corrosion barrier maximum stress intensity is below 70 percent of the ultimate tensile strength. Therefore no breach of the waste package is expected.

7.1.2.2.11 Waste Package Exposed to Vibratory Ground Motion

The objective of *Structural Calculations of Waste Package Exposed to Vibratory Ground Motion* (BSC 2004 [DIRS 167083], Section 1) is to determine the residual stress distribution in the outer corrosion barrier of a waste package exposed to vibratory ground motion and estimate the area of the waste package outer corrosion barrier for which the residual stress exceeds threshold limits. This calculation has been performed for the 21-PWR waste package. Currently the same calculation has not been performed for any of the naval waste packages. However, the naval waste package design is similar. Given the nature and conservatism of the results, no significant change of results is anticipated for the naval waste package (see BSC 2004 [DIRS 166876], Appendix A).

7.1.2.3 Uncertainties of Structural Analyses

In the past interactions with the NRC (Kelmenson 2000 [DIRS 154350]) sources of uncertainty and variability affecting structural analyses were discussed. This particularly dealt with finite element analysis representations and the failure criterion for waste package structural analyses. Six other areas considered were:

1. Residual and differential thermal expansion stresses
2. Strain-rate effects
3. Dimensional and material variability
4. Seismic effect on ground motion
5. Initial tip-over velocities
6. Sliding and inertial effect of waste package contents.

At this time, additional uncertainties have not been identified. As the design progresses, any additional uncertainties that are identified will be addressed as part of the design process. These identified uncertainties will be documented within the documents supporting the license application.

Finite Element Analysis Discretization and Failure Criterion—With regard to the adequacy of finite element analysis representations, a process has been developed to ensure that the mesh density is computationally adequate, and this process is followed for all structural calculations. The failure criterion is an application of the Tresca (strength of materials) failure criterion based on the implementation of ASME B&PV Code (ASME 2001 [DIRS 158115]) design-by-analysis primary stress intensity limits. A tiered evaluation approach was implemented that used increasingly less simplified, and increasing less conservative screening criterion whose satisfaction will assure meeting the ASME B&PV Code (ASME 2001 [DIRS 158115]) primary stress intensity limits.

For the six specific areas of uncertainty concern, the responses may be summarized as:

> **Residual and Differential Thermal Expansion Stresses**—Differential thermal expansion is accommodated by providing adequate gaps between the two shells that comprise the waste package to ensure that there is no mutual loading due to thermal expansion. For residual stresses purposefully imposed on the outer corrosion barrier, the effects on structural analysis results are found to be negligible.
>
> **Strain-rate Effects**—While material-specific strain-rate dependent properties are not currently available for Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600), parametric studies of such effects based on stainless steel type 304 (UNS S30400) strain-rate dependent properties have shown that the use of static properties has negligible effect on the safety assessment.
>
> **Dimensional and Material Variability**—Dimensional variability is addressed by assuming minimum dimensions for those parameters that are important to component performance. Material variability is accommodated by the use of ASME B&PV Code (ASME 2001 [DIRS 158115])—and other codes as necessary—structural properties, which provide for minimum structural performance margins.

Seismic Effect on Ground Motion—In the surface facility, in the transporter, and on the emplacement gantry, it is assumed that the fixturing is provided to restrain the waste package during evolutions in that facility, and these devices are sufficient to provide restraint during vibratory ground motion. For vibratory ground motion in the underground, results are provided for a seismic evaluation for an annual frequency of exceedance of $5x10^{-4}$ per year. These results show a very modest waste package movement and large margin to breach.

Initial Tip-Over Velocities—A study has been performed to demonstrate that the increase in tip-over velocity due to credible vibratory ground motion causes a negligible increase in impact velocity.

Sliding and Inertial Effect of Waste Package Contents—The waste form contents are represented in dynamic structural analyses for which such motion is anticipated to be important. Examples of the loads and boundary conditions used in calculations and analyses can be found in the supporting calculations (BSC 2001 [DIRS 152655]; BSC 2003 [DIRS 161691]; BSC 2004 [DIRS 167083]; BSC 2003 [DIRS 166795]; BSC 2003 [DIRS 165497]). In addition, the technical bases and or rationale for the loads and boundary conditions used in calculations supporting the license application will be based on the preclosure safety analysis and derivative design constraints.

7.1.2.3.1 Response to General Issue of Adequacy

The section addresses the adequacy of the finite element analysis mesh discretization and the failure criterion.

7.1.2.3.1.1 Mesh Discretization

A set process is followed in the development of the mesh for finite element analysis that provides confidence that the results are stationary in a numerical sense (Mecham 2004 [DIRS 166168], Section 6.2.3).

The purpose of mesh refinement is to ensure the mesh objectivity of the finite element analyses, i.e., that the results obtained are not mesh-sensitive. The basis for the validity of this process of successive refinement is that it has been found to produce convergent stress fields in a systematic manner. The acceptable variations in the stress fields are well within the benchmarking basis for the LS-DYNA code (BSC 2003 [DIRS 166918]). A mesh-refinement study consists of the development of an optimum mesh that yields mesh-objective (mesh-insensitive) results. That mesh is then refined again, and computational results for the two mesh sizes are compared. The finite element representation is considered mesh-objective if the relative difference in results (e.g., stresses) between the two meshes is approximately an order of magnitude smaller than the relative difference in mesh size in the region of interest; otherwise further mesh refinement is needed. The mesh size, as used throughout this section, refers to the volume or the area of the representative element (three-dimensional or two-dimensional, respectively) in the region of interest (for example, the element characterized by the highest stresses or strains).

The optimum mesh is created by the following sequence of steps:

- The initial mesh is created by following the customary engineering practices: the element type is appropriately chosen; the mesh is refined in the regions of interest (the highest stress/strain regions, initial impact regions, stress concentration regions, etc.); the mesh is mapped whenever possible; and the aspect ratio of elements is kept reasonable.

- The initial mesh is—in the region of interest—refined in one direction while the element size in the other two directions is kept unchanged (for example, the mesh is refined across the thickness while kept unchanged in the hoop and axial directions). The mesh-refinement procedure is repeated in this manner until the relative difference in results between the two successive meshes is acceptable (i.e., approximately an order of magnitude smaller than the relative difference in the mesh size). The mesh dimension in this direction is then fixed at the largest value that satisfied the previously mentioned criterion.

- The intention of this one-direction-at-a-time mesh refinement is to create, in a consistent and systematic manner, a mesh that is objective.

- The same procedure is consecutively repeated in the remaining two directions.

- Whether the created mesh meets the requirement is verified by the final step: the simultaneous mesh refinement in all three directions. The level of this mesh refinement should be similar in all three directions. In this final step, the same mesh-acceptance criterion is invoked: the mesh is considered objective if the relative difference in results (e.g., stresses) between the two meshes is approximately an order of magnitude smaller than the relative difference in mesh size in the region of interest.

It should be emphasized that the mesh objectivity is verified by the final step regardless of whether the final mesh is arrived at by the described one-direction-at-a-time mesh refinement or not. The one-direction-at-a-time mesh refinement is optional since its only purpose is to develop an optimum mesh (that satisfies the mesh-objectivity requirement) in a systematic way.

An example of the implementation of this mesh discretization approach may be found in the calculation entitled *44-BWR Waste Package Tip-Over from an Elevated Surface* (BSC 2003 [DIRS 166795]). While all calculations perform such discretization studies, this calculation is selected because it is the vehicle cited in the balance of this section to assess the importance of strain rates (Section 7.1.2.3.2.2) and initial tip-over velocities (Section 7.1.2.3.2.5).

7.1.2.3.1.2 Selection of the Failure Criterion

For structural analyses of preliminary designs that consider material nonlinear behavior, the maximum-shear-stress or Tresca (strength of materials) criterion is used in determining stress limits. In general terms, this criterion assumes that the design is safe as long as stress intensity (defined as the difference between the maximum and minimum principal stress) remains below a certain limit. In particular, the failure criterion chosen was the acceptance criteria for plastic analysis outlined in Appendix F, F-1341.2 of the ASME B&PV Code (ASME 2001 [DIRS 158115] Section III, Division 1, Appendix F). This is an acceptable vessel designer choice of ASME B&PV Code acceptance criteria for service loadings with Level D Service Limits for vessel designs in accordance with NC-3200 (Safety Class 2 Vessels) when a complete stress analysis is performed. (See ASME 2001 [DIRS 158115], NC-3211.1(c), Appendix XIII and Note (4) to Table NC-3217-1).

The ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix F, F-1341.2) suggests the following primary stress intensity limits for plastic analyses:

- The general primary membrane stress intensity shall not exceed 0.7 S_u for ferritic steel materials included in Section II, Part D, Subpart 1, Table 2A and the greater of 0.7 S_u and $S_y + {}^1/_3\,(S_u - S_y)$ for austenitic steel, high-nickel alloy, and copper-nickel alloy materials included in Section II, Part D, Subpart 1, Table 2A, where S_u and S_y are tensile strength and yield strength, respectively.

- The maximum primary stress intensity at any location shall not exceed 0.9 S_u.

The Pressure Vessel Research Council of the Welding Research Council has provided recommended guidelines (Hechmer 1998 [DIRS 166147]) to the ASME B&PV Code rule committees for assessing stress results from three-dimensional finite element analysis in terms of ASME B&PV Code (ASME 2001 [DIRS 158115], stress limits in the design-by-analysis rules of Section III (Class 1, NB) and Section VIII, Division 2). These guidelines were developed for linear analyses and Pressure Vessel Research Council recommends that future research work should be conducted to generate state-of-the-art guidelines for applying inelastic, large-deformation analyses. Therefore, a cautious use of the Pressure Vessel Research Council recommendations was made in developing methodologies for post-processing LS-DYNA nonlinear plastic simulations to assure conservative representations of the general primary membrane stress intensity and maximum primary stress intensity.

The Pressure Vessel Research Council recommendations also refer to an earlier Pressure Vessel Research Council (Phase 1) report that recommended ASME B&PV Code (ASME 2001 [DIRS 158115]), Appendix F "should be revised to provide a limit on effective plastic strain which is more appropriate for events that are energy controlled, rather than load controlled, which is all that was considered when ASME B&PV Code Appendix F was written." The Yucca Mountain Project recognizes that strain-based or deformation-based criterion may be more appropriate than stress-based limits for evaluation of the credible preclosure sequence events (see Mecham 2003 [DIRS 166168], Section 4.1.4.1). However, the project is also committed to applying the ASME B&PV Code for structural analyses, and until the ASME B&PV Code rule committees prepare rules in ASME B&PV Code Appendix F for using strain limits, primary stress intensity limits will be used.

The ASME B&PV Code (ASME 2001 [DIRS 158115]) design-by-analysis guidance recognizes the differences in importance of different types of stresses and provides guidance on their correct assignment to the different categories of stress intensity used to evaluate different types of failure modes. The three types of stresses are membrane, bending and peak stresses. The three categories of stress intensity are primary (P_m, P_L and P_b [general primary membrane, local primary membrane, and primary bending, respectively]), secondary (Q), and peak (F).

A primary stress is defined in ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) XIII-1123(h): "Primary stress is a normal stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium of external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. Primary stresses which considerably exceed the yield strength will result in failure or, at least, in gross distortion."

A secondary stress is defined in ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) XIII-1123(i): "Secondary stress is a normal or a shear stress developed by the constraint of adjacent parts or by self-constraint of the structure. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the conditions which cause the stress to occur and failure from one application of the stress is not expected." A cited example of a secondary stress is "bending stress at a gross structural discontinuity." A gross structural discontinuity is defined in ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) XIII-1123(b): "Gross structural discontinuity is a source of stress or strain intensification which affects a relatively large portion of a structure and has a significant effect on the overall stress or strain pattern or on the structure as a whole." Cited examples of gross structural discontinuities are head-to-shell junctions and junctions between shells of different thickness.

A local primary membrane stress is also defined in ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) XIII-1123(j): "Cases arise in which a membrane stress produced by pressure or other mechanical loading and associated with a discontinuity would, if not limited, produce excessive distortion in the transfer of load to other portions of the structure. Conservatism requires that such a stress be classified as a local primary-membrane stress even though it has some characteristics of a secondary stress." The other differentiating feature of a local primary membrane stress is that it is localized, and ASME B&PV Code (ASME 2001 [DIRS 158115]) guidance is provided for evaluating if membrane stress fields are adequately "local" to be assigned a P_L classification rather than a more restrictive P_m classification.

Per Pressure Vessel Research Council recommendations (Hechmer 1998 [DIRS 166147], Guideline 1) the failure mode being addressed by the general primary membrane stress intensity (P_m) limit is "collapse" in the sense that collapse includes tensile instability and ductile rupture under short term loading. The principle failure mode being addressed by the maximum primary stress intensity (P_L + P_b) is excessive plastic deformation. However, it also relates to tensile instability due to the nature of P_b .

The sequence events considered in this report are not repetitive where fatigue cracking or incremental collapse might be an issue. It follows that evaluation of secondary stress intensities (Q) or maximum total stress intensities (P_L + P_b + Q + F) are not appropriate. Brittle fracture is also precluded by the high ductility of the outer boundary material, Alloy 22 (UNS N06022), at the temperatures experienced after waste form loading. Although the high-stress areas are comprised of primary, secondary and peak stresses, only the primary stress intensities (P_m, P_L and P_b) contribute to plastic instability (tensile tearing) or excessive plastic deformation, and therefore, only the primary stress intensities are evaluated for the sequence events.

Use was also made of the stress classification guidance in the ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) Table XIII-1130-1 to determine which stress fields should be classified as primary and which should be classified as secondary when evaluating the sequence events. It was decided to conservatively classify all membrane stress fields as primary. Classification of the bending stresses was more involved.

Review of representative analyses for the sequence events indicated that the most significant wall-bending stresses in the outer corrosion barrier were occurring near gross structural discontinuities. Some of these gross structural discontinuities were integral to the outer boundary and some were introduced by the constraint of adjacent parts or impact surfaces.

The integral gross discontinuities in the outer corrosion barrier are similar to Code vessel details such as shell-to-lid junctures and step-changes in wall thickness. The bending stresses are being created by self-constraint, and Table XIII-1130-1 (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII) classifies these bending stresses as secondary. The only exception is at the shell-lid junction where concern about the predictability of the lid's central stresses leads the Code to caution the designer (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII, Note (4) of Table XIII-1130-1) to consider classifying the bending stresses as P_b. However, this is not appropriate guidance for inelastic analyses because the increased flexibility of the juncture due to inelastic behavior is correctly captured and the lid's central stresses are accurately predicted.

The bending stresses created by the constraint of adjacent parts or impact surfaces (which can be considered (temporary) "adjacent parts") were reviewed on individual cases with attention to the amount and type of constraint introduced. In the design analyses to date, the constraint of the adjacent part (e.g., trunnion sleeve) or impact surface (e.g., emplacement pallet, crane hook or rock) created local yielding and minor distortions in the outer barrier. The outer corrosion barrier distorted shape reduced the outer corrosion barrier bending stresses while increasing the outer corrosion barrier membrane stresses. The bending stresses in these locally yielded regions are therefore self-limiting and satisfy the basic characteristic of a secondary stress.

The structural criterion developed for the outer boundary for the sequence events was to directly address the dominant failure mode, tensile instability, and limit the membrane stresses to acceptable limits. The use of inelastic analyses assures that local thinning or shape changes that could increase membrane stresses are properly accounted for.

The inelastic analyses were conducted using true stress (σ_u) and true strain based load/deformation relationships, therefore, per ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix F, F-1322.3(b) and F-1341.2), for Alloy 22 (UNS N06022):

The limit on P_m is $0.7\sigma_u$, and

the limit on P_L is $0.9\sigma_u$,where $P_b = 0$, and

σ_u is the true tensile strength at temperature.

As stated earlier, the ASME B&PV Code (ASME 2001 [DIRS 158115], Section III, Division 1, Appendix XIII, XIII-1123(j)) provides guidance on how "local" P_L must be to not be classified as a more restrictive general primary membrane stress intensity, P_m . Interpretation of this guidance with respect to the Appendix F limits results in requiring P_L values exceeding $0.77\sigma_u$ to not extend for greater than $\sqrt{R \cdot t}$ in any direction (not just the meridional direction), where R is the midsurface radius and t is the thickness of the outer barrier.

Rigorously performed, the calculation of the primary membrane stress intensities requires the following steps:

- Identification of the governing wall location (stress classification plane normal to the mid-plane of the shell or lid thickness) which may not necessarily contain the maximum stressed point (Hechmer 1998 [DIRS 166147], Guidelines 3 and 4)
- Identification of the orientation of the stress classification line (SCL), typically normal to the mid-plane of the shell or lid thickness (Hechmer 1998, Guideline 4d).
- Identification of stress component ($\sigma_x, \sigma_y, \sigma_z,\ \tau_{xy},\ \tau_{yz},\ \tau_{zx}$) fields across the wall of the outer corrosion barrier
- Averaging of the stress component fields to create wall-averaged stress components
- Translation of these wall-averaged stresses to principle stress directions by solving a cubic equation

- Calculation of the difference between the maximum (σ_1) and minimum (σ_3) principle stress direction values.

To simplify the calculation, the wall-average of the element total stress intensity (twice the maximum shear stress) values through the outer corrosion barrier is used to define the primary membrane stress intensities. This is a conservative representation because it ignores the possible changing of the principle stress planes through the wall and includes the secondary and peak stress contributions.

To further simplify the calculation, tiered screening criteria are applied to the outer corrosion barrier finite element analysis results. The easiest to apply and most conservative criteria are applied initially. If these can not be met, less conservative screening criteria are imposed that require more calculations.

In the case of lifting analyses, the acceptance criteria are outlined in *American National Standard for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More* (ANSI N14.6-1993 [DIRS 102016], Section 4.2.1.1). The load-bearing members of the lifting device shall be capable of lifting three times the combined weight of the shipping container, plus the weight of the intervening components of the lifting device, without generating a combined shear stress or maximum principal stress at any point in the device in excess of S_y and shall also be capable of lifting five times the weight without exceeding S_u.

7.1.2.3.2 Responses to Specific Issues

The following sections address the specific issues enumerated in Section 7.1.2.3.

7.1.2.3.2.1 Residual and Differential Thermal Expansion Stresses

7.1.2.3.2.1.1 Differential Thermal Expansion

Differential thermal expansion is accommodated by providing adequate gaps between the inner vessel and outer corrosion barrier to ensure that there is no mutual loading due to thermal expansion. The required radial gap between the inner vessel and the outer corrosion barrier of the waste package is documented in a calculation entitled *Waste Package Outer Barrier Stress Due to Thermal Expansion with Various Barrier Gap Sizes* (BSC 2001 [DIRS 152655]). This calculation resulted in a minimum gap spacing between the inner vessel and outer corrosion barrier to accommodate radial expansion to be set at 1 mm (BSC 2001 [DIRS 152655], Tables 4 and 5, p. 13). The axial gap between the inner vessel and outer corrosion barrier and the lids of each is documented in a calculation entitled *Waste Package Axial Thermal Expansion Calculation* (BSC 2003 [DIRS 161691]). This calculation established a minimum axial gap of 1 cm between the inner vessel and outer corrosion barrier (BSC 2003 [DIRS 161691], Section 7, p. 13). A similar approach will be used to ensure clearance between the inner vessel of the waste package and the internals. These clearances are addressed in the two naval configuration drawings:

- *Naval Long Waste Package Configuration* (BSC 2003 [DIRS 164854]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165158]) and BSC (2003 [DIRS 165159]), respectively.

- *Naval Short Waste Package Configuration* (BSC 2003 [DIRS 164855]). Sheets 2 and 3 are found in BSC (2003 [DIRS 165160]) and BSC (2003 [DIRS 165161]), respectively.

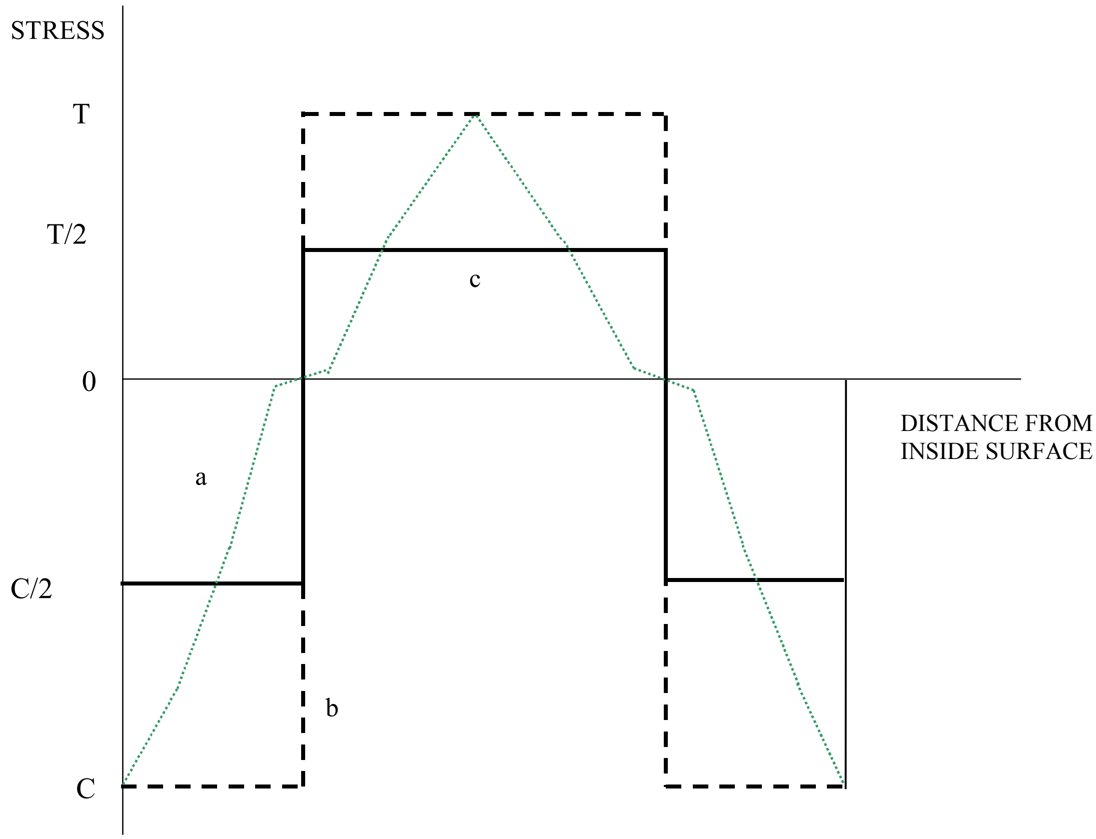
7.1.2.3.2.1.2 Effect of Residual Stresses

The waste package outer corrosion barrier is not in a stress-free condition at the beginning of service life due to residual stresses purposefully induced by solution annealing and quenching. The purpose of these residual stress fields is to create compressive residual stresses at the outside surface, and perhaps the inside surface as well (depending on the quenching techniques) of the outer corrosion barrier to help mitigate corrosion. The effect of this stress profile on the waste package during dynamic events is documented in a calculation entitled *Drop of Waste Package on Emplacement Pallet-A Mesh Study* (BSC 2003 [DIRS 165497]). While this calculation was prepared for a postclosure evaluation, it illustrates the basic physics of the phenomenon, and the conclusions are equally appropriate for preclosure evaluations of preclosure dynamic structural calculations.

The residual stresses due to the solution annealing and quenching are analyzed for a mockup waste package outer corrosion barrier in *Residual Stress Analyses on the 21 PWR Mockup Waste Package Outer Shell Due to Quenching and General corrosion Using a Side-wall Thickness of 20mm* (Herrera et al. 2002 [DIRS 166799]). The residual stress analyses are performed for two different quenching techniques: (1) the outside quench (on the outside surface only) and (2) the double-sided quench (on both the inside and outside surfaces). The results reported herein correspond only to the residual stress distribution due to the double-sided quenching.

It must be recognized that the accuracy of this study is limited by the through-wall discretization of the outer corrosion barrier. Since only four layers of solid (brick) elements are used for the finite element analysis representation of the outer corrosion barrier in this calculation, the residual stress distribution is necessarily rather coarse. Furthermore, the one-point-integration solid elements used in this calculation are not best suited for the representation of the initial stress distribution. Nonetheless, no change has been made in the finite element analysis representation for the residual stress calculations since it was important to make a comparison between the results obtained by using the same representation, which was defined (representation) by the objective of the source calculation (BSC 2003 [DIRS 165497]).

Two different magnitudes of the initial stress distribution are used in this study to explore a sensitivity of results to the details of the stress distribution. (Note the schematic representation of the residual stress distribution—generic for both hoop and axial direction—presented as the dotted green line [a] in Figure 8). In the first approximation, the initial stress (i.e., the residual stress caused by the annealing and quenching) in each layer of elements is defined by using the maximum stress value reached anywhere within the element layer (the dashed line [b] in Figure 8; see also row "Full" in Table 48). In the second approximation, the initial stress in each layer of elements is obtained by averaging the actual stress distribution (the green dotted line [a] in Figure 8) over the element layer. Keeping in mind the actual residual stress distribution, the averaging is performed by assigning to the approximated initial stress distribution one half of the maximum stress value reached anywhere within each element layer (solid line [c] in Figure 8; see also row "Half" in Table 48). The approximated initial stress distributions are presented in Figure 8. The actual stress values are obtained from Herrera et al. (2002 [DIRS 166799], Figures 48 and 52). For the axial stress distribution the maximum compressive stress at both the inside and outside surface is $C = -300MPa$; the maximum tensile stress at the middle surface is $T = 150MPa$. For the hoop stress profile the maximum compressive stress at both inside and outside surface is $C = -260MPa$; the maximum tensile stress at the middle surface is $T = 190MPa$.

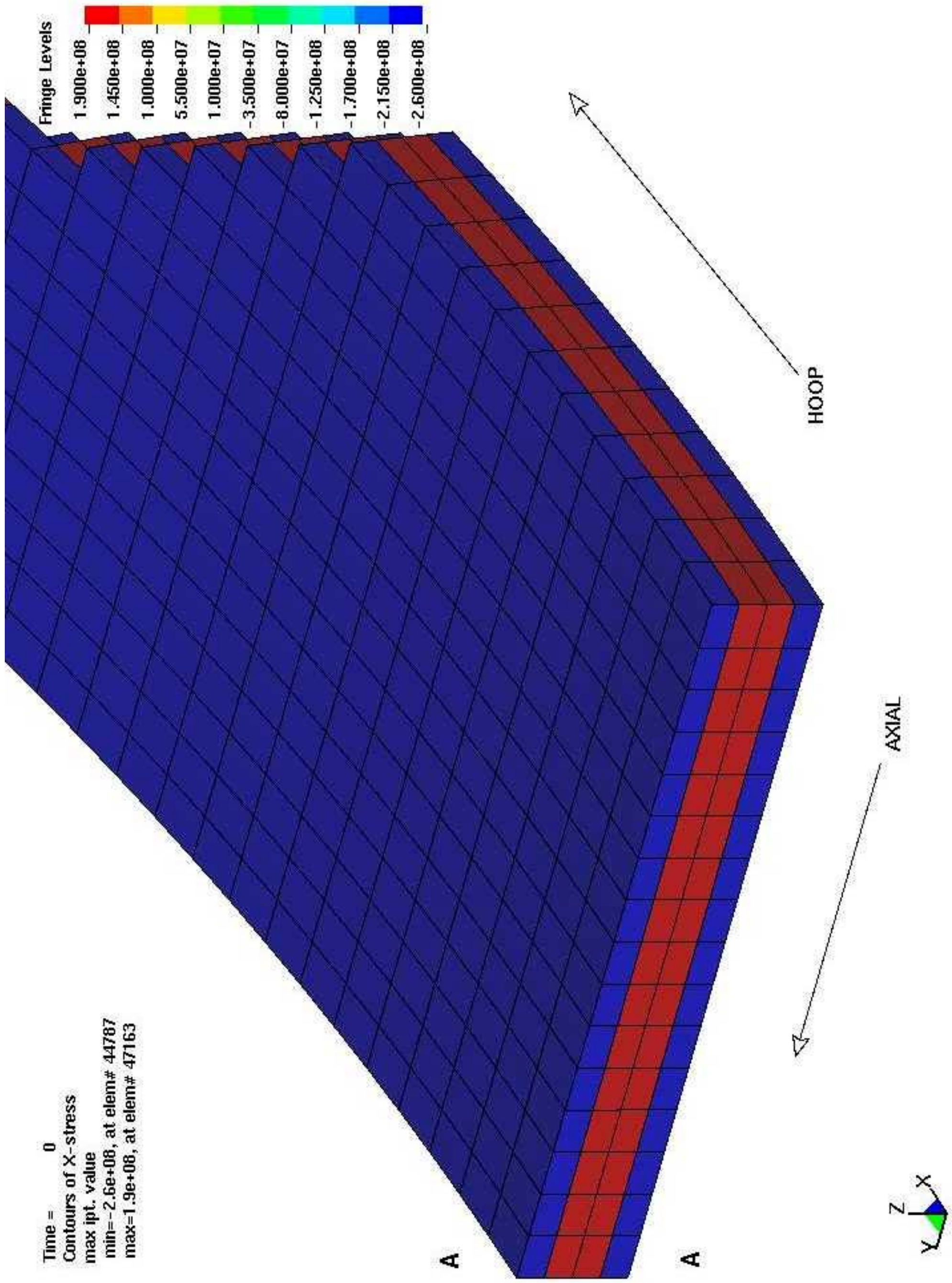


NOTES: (a) Schematic representation of axial and hoop stress distribution from Herrera et al. (2002 [DIRS 166799], Figures 48 and 52) (green dotted line), (b) first ("full") approximation (dashed line), and (c) second ("half") approximation (solid line).

Source: BSC 2003 [DIRS 165497], Figure VII-1

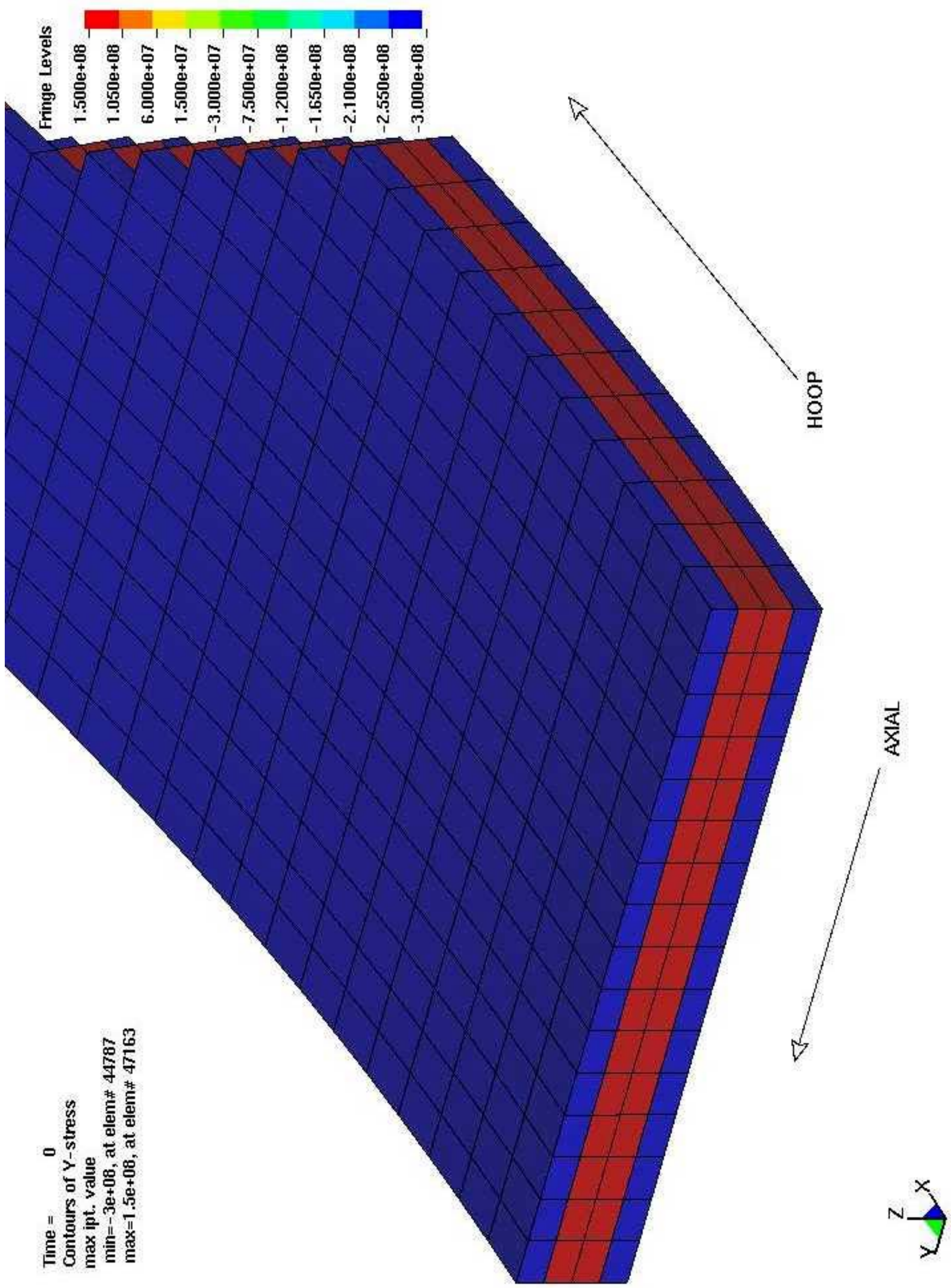
Figure 8. Initial Stress Distribution across the Outer Corrosion Barrier Wall

The resulting initial stress distributions in hoop and axial directions are, for the first approximation ("Full"), presented in Figure 9 and Figure 10, respectively. (Note that LS-DYNA finite element analysis code requires the initial stresses to be specified in the global Cartesian coordinate system. Thus, the initial stress distribution in x direction, presented in Figure 9, corresponds to the hoop stress distribution only at the symmetry plane.) The initial effective plastic strain, used for both approximations, is zero.



NOTE: Normal stress in x-direction is identical to hoop stress at symmetry plane designated as A-A section.
Source: BSC 2003 [DIRS 165497], Figure VII-2

Figure 9. Initial Stress Distribution in X Direction in Outer Corrosion Barrier Caused by Annealing and Double-Sided Quenching



Source: BSC 2003 [DIRS 165497], Figure VII-3

Figure 10. Initial Axial (Y-) Stress Distribution in Outer Corrosion Barrier Caused by Annealing and Double-Sided Quenching

The calculations are performed for the horizontal drop of the waste package on the pallet with impact speed of $8\,m/s$.

The results are presented in Table 48. The row designated with "No" represents the initially stress-free case (i.e., without the initial stress). The results obtained by using the first and second initial stress approximations are presented in rows "Full" and "Half," respectively.

Table 48. Results for Three Different Initial Stress Approximations

Magnitude of Residual Stress	Maximum Stress Intensity (MPa)	Maximum Effective Plastic Strain (%)	Damaged Area (80% criterion/90% criterion) ($\times 10^{-3}$ m^2)[a]
No	630	30.3	7.47 / 2.46
Half	632	30.4	6.41 / 2.29
Full	631	30.7	5.82 / 2.21

NOTE: [a] This is the percentage of yield stress and is used in postclosure seismic analyses as a measure of susceptibility to accelerated corrosion.

Source: BSC2003 [DIRS 165497], Table VII-1

According to results presented, the maximum stress intensity and the maximum effective plastic strain are not significantly affected by presence of the initial stress (i.e., the residual stress caused by the solution annealing and double-sided quenching). The damaged area is moderately sensitive to the initial stresses. The damaged area is used in postclosure analyses to assess the susceptibility to accelerated corrosion, which is not important for preclosure safety.

7.1.2.3.2.2 Strain-Rate Effects

The plastic behavior of materials is sensitive to strain rate, which is known as material strain-rate sensitivity. The strain-rate data for Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600) (the stress-strain curves for different strain rates or the change of a characteristic stress with strain rate) are not available in literature at present. Thus, the effect of strain rate on the mechanical strengths of Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600) was studied parametrically by using as a guidance the strain-rate data for stainless steel type 304 (UNS S30400) (Nicholas 1980 [DIRS 154072], Figures 10 and 27) for both materials. Stainless steel type 304 (UNS S30400) is used as an analogue for stainless steel type 316 (UNS S31600) and Alloy 22 (UNS N06022) insofar as strain rate effects are concerned. The tangent (hardening) moduli for Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600) are assumed to be unaffected by the rate of loading. The rationale is that according to the document, *Dynamic Tensile Testing of Structural Materials Using A Split Hopkinson Bar Apparatus* (Nicholas 1980 [DIRS 154072], Figure 10), the tangent modulus for stainless steel type 304 (UNS S30400) is not significantly affected by the strain rate. This evaluation is documented in a calculation entitled *44-BWR Waste Package Tip-Over from an Elevated Surface* (BSC 2003 [DIRS 166795]).

Strain rate is accounted for in this study by using Cowper and Symonds approach that scales the yield strength with the factor:

$$\beta = 1 + \left(\frac{\dot{\varepsilon}}{C}\right)^{1/p} \qquad \text{(Eq. 1)}$$

Here $\dot{\varepsilon}$ is the strain rate, and C and p are input parameters obtained by fitting the experimental data (Hallquist 1998 [DIRS 155373], p. 16.37).

The test results provided for stainless steel type 304 (UNS S30400) are used to establish reasonable limits for strain-rate factor β. The results obtained at strain rates of 20 s^{-1} and 900 s^{-1} are selected (Nicholas 1980 [DIRS 154072], Figures 10 and 27) for fitting of the strain-rate parameters, since those two values adequately span the strain-rate range relevant for this calculation. From that data (Nicholas 1980 [DIRS 154072], Figure 27, curve 304, $\varepsilon = 0.10$)

$$\beta\left(\dot{\varepsilon} = 20\,s^{-1}\right) = 1.135 \qquad \text{(Eq. 2)}$$

$$\beta\left(\dot{\varepsilon} = 900\,s^{-1}\right) = 1.37 \qquad \text{(Eq. 3)}$$

To establish the upper bound for strain-rate effects, the change of stress of 13.5 percent at strain rate of 20 s^{-1} (compared to the static test) is increased to 20 percent (corresponding to relative increase of 50 percent). Thus, for the upper bound, $\beta\left(\dot{\varepsilon} = 20\,s^{-1}\right) = 1.20$. Similarly, the change of stress of 37 percent at strain rate of 900 s^{-1} (compared to the static test) is increased to 55 percent (corresponding to relative increase of 50 percent); this value is then rounded to 60 percent. Thus, for the upper bound, $\beta\left(\dot{\varepsilon} = 900\,s^{-1}\right) = 1.60$.

Results for stainless steel type 304 (UNS S30400) from two additional sources are also presented in the source document for this data (Nicholas 1980 [DIRS 154072], Figure 27). All three test results from this source document are used to establish the lower bound for the strain-rate factor β, $\beta\left(\dot{\varepsilon} = 20\,s^{-1}\right) = 1.05$ and $\beta\left(\dot{\varepsilon} = 900\,s^{-1}\right) = 1.15$. The purpose of this lower bound is to explore sensitivity of results with regards to the amount of the strain-rate strengthening of material.

In summary, the scale factor β corresponding to strain rate of 20 s^{-1} is 1.05 and 1.20 for the lower and upper bounds, respectively (see Table 49). The scale factor β corresponding to strain rate of 900 s^{-1} is 1.15 and 1.60 for the lower and upper bounds, respectively (Table 49). Note that at both strain rates the increase of stress (expressed as percent increase compared to the static value) from the lower to the upper bound is four times. Also, for both the upper and lower bound the increase of stress (expressed as percent increase compared to the static value) from 20 s^{-1} to 900 s^{-1} is three times.

Table 49. Strain-Rate Parameters

	Lower Bound	Upper Bound
$\beta(20\ s^{-1})$	1.05	1.20
$\beta(900\ s^{-1})$	1.15	1.60
p	3.465	3.465
C	644,300	5,284

Source: BSC 2003 [DIRS 166795], Table 5

These values can be used as boundary conditions for determination of strain-rate parameters in Table 49. For example for the lower bound, the expression,

$$1.05 = 1 + \left(\frac{20}{C}\right)^{1/p} \quad \Rightarrow \quad C = \frac{20}{0.05^{p}} \qquad \text{(Eq. 4)}$$

is obtained by substituting the first boundary condition ($\beta(\dot{\varepsilon} = 20\ s^{-1}) = 1.05$) in Equation 1.

Similarly, by substituting ($\beta(\dot{\varepsilon} = 900\ s^{-1}) = 1.15$) in Equation 1,

$$1.15 = 1 + \left(\frac{900}{C}\right)^{1/p} \qquad \text{(Eq. 5)}$$

and adding Equation 4, the parameter p can be readily calculated:

$$0.15 = \left(\frac{900}{20/0.05^{p}}\right)^{1/p} \quad \Rightarrow \quad p = \frac{\ln(45)}{\ln(0.15) - \ln(0.05)} = 3.465 \qquad \text{(Eq. 6)}$$

From Equation 4 it follows directly that $C = 644{,}300\,s^{-1}$.

By repeating the same calculation for the upper-bound values of β the following parameters can be readily obtained, $p = 3.465$ and $C = 5{,}284\,s^{-1}$ (see Table 49).

Three calculations are performed to explore the strain-rate sensitivity of results presented in this calculation (see Table 50 and Table 52). The first calculation is performed with static material properties without strain-rate effects accounted for (row “No” in Table 50 and Table 52). The second calculation corresponds to the lower-bound strain-rate sensitivity (row “Low” in Table 50 and Table 52). Finally, the third calculation is performed with highly rate-sensitive material strengths (row “High” in Table 50 and Table 52, corresponding to the upper-bound strain-rate parameters in Table 49).

Table 50. Maximum Stress Intensity in Outer Corrosion Barrier and Inner Vessel for Three Different Levels of Strain-Rate Sensitivity

Strain-rate Sensitivity	Maximum Stress Intensity (MPa)	
	Inner Vessel	Outer Corrosion Barrier
No	518	902
Low	528	942
High	601	1,037

Source: BSC 2003 [DIRS 166795], Table V-2

Maximum stress intensity, as expected, increases with increased strain-rate sensitivity of the material strengths (see Table 50). The strain-rate strengthening of material implies increase of the true tensile strength, which must be quantified in order to make a meaningful assessment of the material condition upon deformation.

The strain rates encountered in the inner vessel and outer corrosion barrier, at the time when the maximum stress intensities occur, are determined from Figure 11 and presented in Table 51. Note that the effective-strain time histories presented in Figure 11 correspond to elements characterized by the maximum stress intensity (presented in Table 50), i.e., elements 27077 and 27078 (inner vessel) and element 10174 (outer corrosion barrier). Strain-rate factor β is then calculated using Equation 1 for the strain-rate parameters (presented in Table 49) and the strain rate (presented in Table 51). Finally, the true tensile strengths of Alloy 22 (UNS N06022) and stainless steel type 316 (UNS S31600) are scaled by the factor β.

Table 51. Parameters Defining Strain-Rate Sensitivity for Inner Vessel and Outer Corrosion Barrier at the Time Characterized by Maximum Stress Intensity

Strain-rate Sensitivity	Strain Rate (1/s)	Strain-Rate Factor β (-)	True Tensile Strength (MPa)
		Inner Vessel	
No	N/A	1	703
Low	11	1.042	733
High	11	1.168	821
		Outer Corrosion Barrier	
No	N/A	1	971
Low	8	1.038	1,008
High	8	1.154	1,121

Source: BSC 2003 [DIRS 166795], Table V-3

The ratio of the maximum stress intensity and true tensile strength is calculated for the inner vessel and outer corrosion barrier for all three strain-rate sensitivity cases. In other words, the maximum stress intensity (Table 50) is divided by the strain-rate-scaled true tensile strength (Table 51). The calculation results are presented in Table 52.

Table 52. Ratio of Maximum Stress Intensity and True Tensile Strength in Outer Corrosion Barrier and Inner Vessel for Three Different Levels of Strain-Rate Sensitivity

Strain-rate Sensitivity	σ_{int} / σ_u	
	Inner Vessel	Outer Corrosion Barrier
No	0.74	0.93
Low	0.72	0.94
High	0.73	0.93

Source: BSC 2003 [DIRS 166795], Table V-4

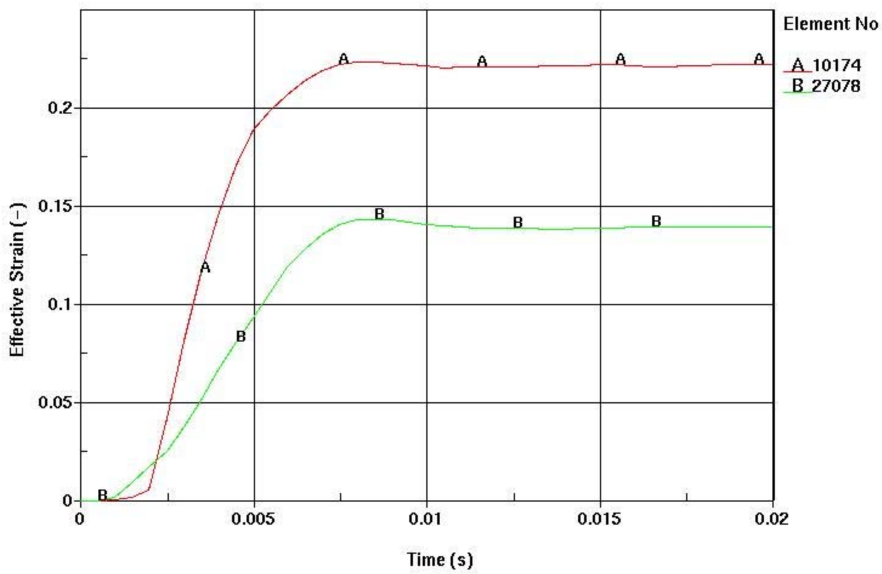
Based on the results presented in Table 52, it can be concluded that:

1. The level of strain-rate sensitivity (i.e., "Low" vs. "High") does not have a significant effect on the ratio of the maximum stress intensity and true tensile strength.

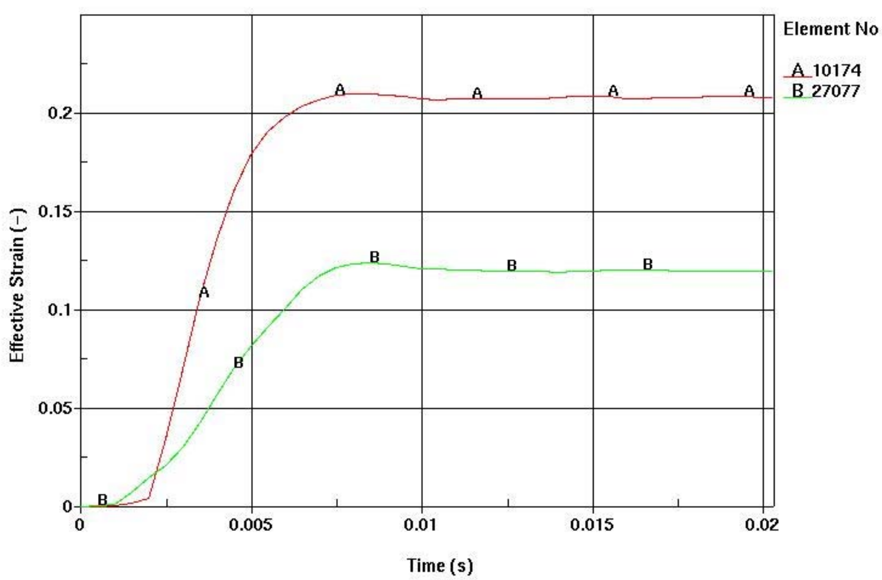
2. The use of the static material properties for the tip-over calculation does not have a significant effect on the ratio of the maximum stress intensity and true tensile strength.

Finally, it is important to note that the strain rates reported in Table 51 are the strain rates corresponding to times when the maximum stress intensities are recorded (as an example, for the outer corrosion barrier it is 0.007 s). At that time, the strain rate in the outer corrosion barrier is in rapid decline. Specifically, for the element characterized by the maximum stress intensity (element 10174; see Figure 11) it is reduced from 70 s^{-1} to 8 s^{-1}. This raises fundamental questions. If a material is strengthened by elevated-strain-rate loading and then the rate of loading is reduced, is material strength going to reduce as well? If that is so, what is the characteristic time related to that strength reduction? Can it possibly happen "instantaneously"? These important questions are not addressed in available literature at present. Answering these, and similar, questions would require a detailed insight into mechanical and metallurgical aspects of the strain-rate strengthening of material. However, this is not necessary because the effect of strain-rate strengthening of the material is conservatively accounted for in this calculation by scaling the true tensile strength with the strain-rate factor β corresponding to the instantaneous strain rate at the time when the maximum stress intensity occurs. (As an example, if the strain rate of 70 s^{-1} could be used instead of 8 s^{-1} to scale the true tensile strength for the "High" outer corrosion barrier bound, the increase of the true tensile strength would be from $\sigma_u\left(\dot{\varepsilon} = 8\ s^{-1}\right) = 1{,}121\ MPa$ to $\sigma_u\left(\dot{\varepsilon} = 70\ s^{-1}\right) = 1{,}250\ MPa$, which would imply the reduction of the stress ratio from 0.93 to 0.90.)

Therefore, based on the parametric study for strain-rate effects using stainless steel type 304 (UNS S30400) strain-rate dependent properties, it has been demonstrated that the use of static properties for stainless steel type 316 (UNS S31600) and Alloy 22 (UNS N06022) in lieu of material specific strain-rate effects is appropriate.



(a)



(b)

Source: BSC 2003 [DIRS 166795], Figure V-1

NOTE: (a) Low Strain-Rate Sensitivity and (b) High Strain-Rate Sensitivity.

Figure 11. Effective-Strain Time History for Elements Characterized by the Peak Maximum Stress Intensity in the Inner Vessel (Elements 27077 and 27078) and Outer Corrosion Barrier (Element 10174)

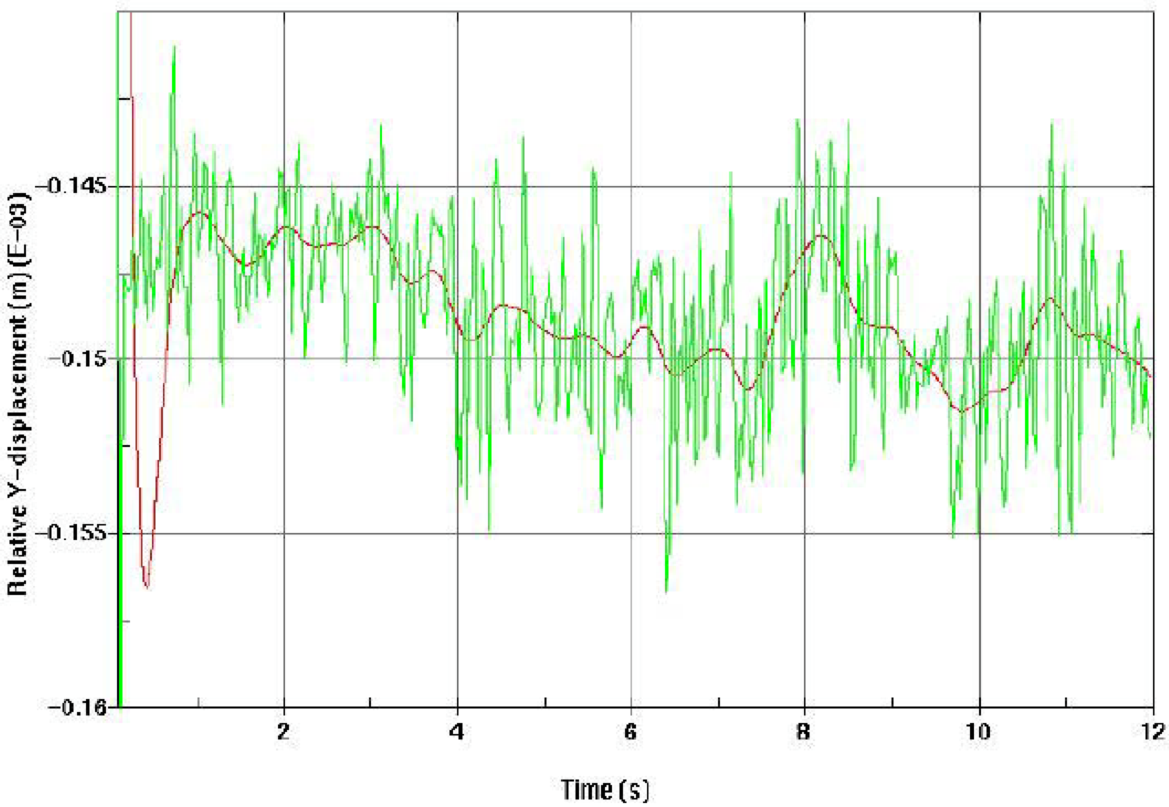
7.1.2.3.2.3 Dimensional and Material Variability

All structural calculations assume the thicknesses for the inner vessel and outer corrosion barrier are the minimum material thicknesses. Future drawings will indicate tolerances that show these dimensions as minimum values. This assures structural design requirements will be achieved.

Maintaining conservative answers due to material variability is managed by using the minimum material-property strengths available (e.g., from the ASME B&PV code and other codes). When available, material properties that are temperature dependent are used for variable-temperature environment calculations. In general, when a range of values is given for material properties, the values that ensure conservative results are used.

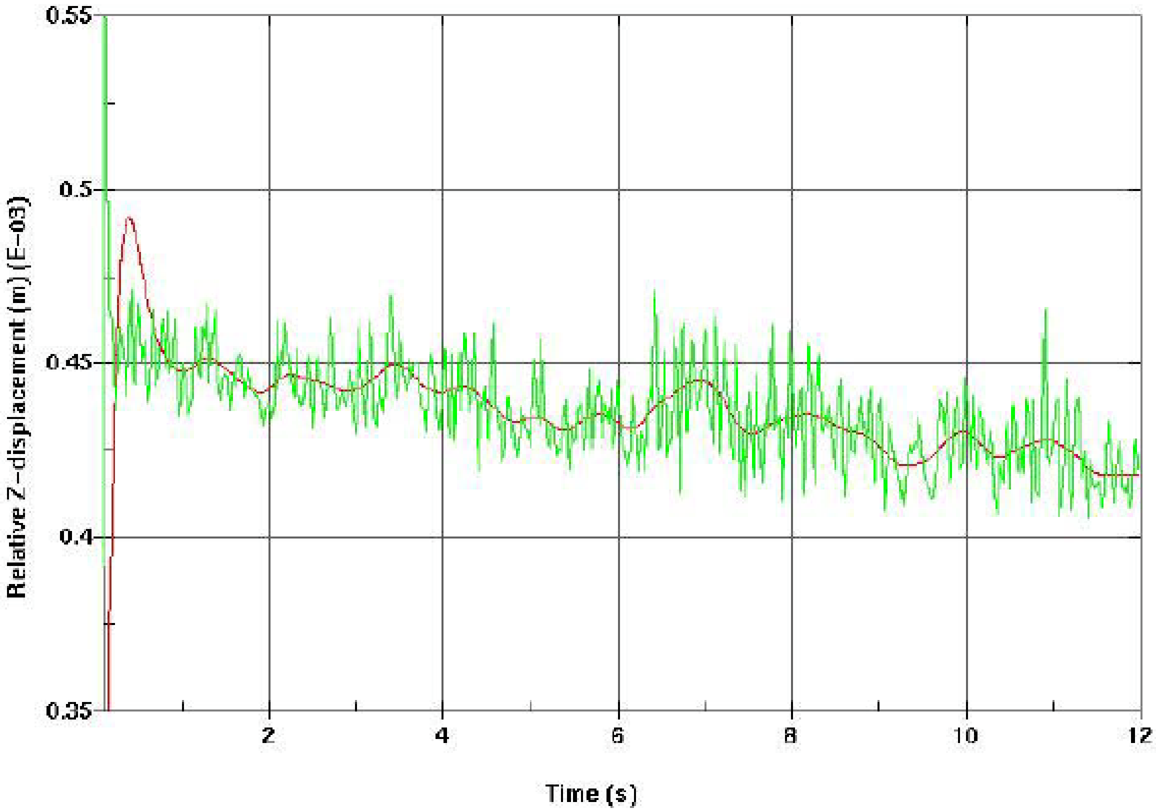
7.1.2.3.2.4 Seismic Effect on Ground Motion

In the surface facility, it is assumed that the fixtures are provided to restrain the waste package during evolutions in that facility, and these devices are sufficient to provide restraint during vibratory ground motion. For vibratory ground motion in the underground, margin to the breach of the waste package has been calculated for vibratory ground motion with an annual exceedance frequency (annual frequency of occurrence) of $5x10^{-4}$ per year. For this calculation, the motion of the waste package was very small, on the order of fractions of millimeters as illustrated in Figure 12 and Figure 13 (BSC 2004 [DIRS 167083], Section 6.3, pp. 63 to 64).



Source: BSC 2003 [DIRS 167083], Figure 10

Figure 12. Relative Longitudinal (Y) Displacement (Raw–green and Filtered–red) of Waste Package with Respect to Pallet for Annual Frequency of Occurrence 5 x 10-4 per year



Source: BSC 2003 [DIRS 167083], Figure 11

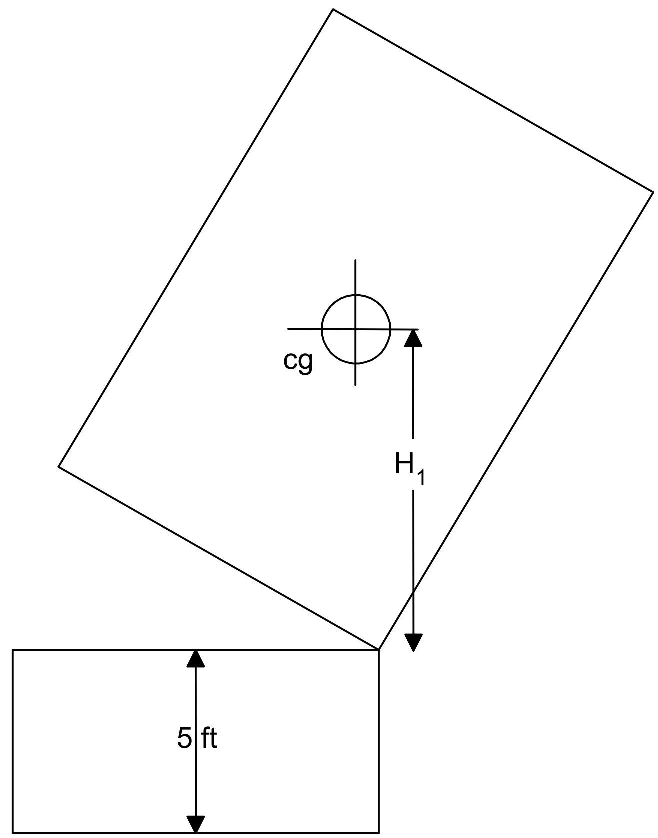
Figure 13. Relative Vertical (Z) Displacement (Raw–Green and Filtered–Red) of Waste Package with Respect to Pallet for Annual Frequency of Occurrence 5 x 10-4 per year

7.1.2.3.2.5 Initial Tip-Over Velocities

A sensitivity study was performed where a range of tip-over velocities were considered and bound those expected in the surface facilities. This evaluation is documented in a calculation entitled *44-BWR Waste Package Tip-Over from an Elevated Surface* (BSC 2003 [DIRS 166795]). The point of incipient toppling is illustrated in Figure 14. Using the energy method, the rotational velocity of the waste package is calculated at the point just before impact. Table 53 shows a possible range of initial velocities. The peak ground velocity (PGV) is multiplied by values of 0, 1, 5, and 10 to span the parameter space.

$$mg \cdot \Delta h = ½ \cdot I \cdot \Delta(\omega^2) \quad \textbf{(}\text{Eq. 7)}$$

Here, "m" is the mass of the waste package, "g" is the gravitational acceleration constant, "Δh" is the change in the height of the center of gravity of the waste package from the moment of toppling to impact, "I" is the moment of inertia of the waste package, and "ω" is the angular velocity.



Source: BSC 2003 [DIRS 166795], Figure 5-1

Figure 14. Waste Package Position at Maximum Potential Energy

Evaluating this expression,

$$(43{,}400\ \text{kg})\cdot(9.81\text{m/s}^2)\cdot(2.587\ \text{m}) = \tfrac{1}{2}\cdot(0.4276\text{e}06\ \text{kg-m}^2)\cdot(\omega^2-\omega_0^2)$$

Here, "ω_0" is the initial angular velocity.

The PGV has a value of 0.4378 m/s (DTN: MO0306SDSAVDTH.000 [DIRS 164033]) on the repository horizon, yielding:

$$\text{PGV } (10^{-4}\ \text{event}) = 0.4378\ \text{m/s} = V_0$$

(The only ground motions available for this frequency of exceedance are for the repository horizon. Subsequent to the performance of this work, the PGV for an annual frequency of exceedance of 1×10^{-4} per year at the surface became available (DTN: MO0312WHBDE104.001 [DIRS 167126]). This PGV is 1.17 m/s, which is about three times the velocity at the repository horizon. The corresponding PGVs at the surface are higher and are covered by the sensitivity study range.)

Finally,

$$\omega_0 = V_0/H_1 \tag{Eq. 8}$$

In this equation, "H_1" is the distance from the center of gravity of the waste package to the bottom edge of the waste package at the point of toppling (see Figure 14).

Note that predicted PGV—albeit at the repository horizon—results in a negligible change in the rotational velocity at impact.

Table 53. Resultant Impact Velocities by Parameter

Parameter	V_0 (m/s)	ω_0 (rad/s)	ω (rad/s)
PGV*0	0	0	2.27
PGV*1	0.438	0.161	2.27
PGV*5	2.19	0.812	2.41
PGV*10	4.38	1.62	2.79

Source: BSC 2003 [DIRS 166795], Table IV-1

The resulting maximum stress intensities for this sensitivity study are shown in Table 54. While substantial increases in initial tip-over velocity result in higher stress levels, the effect is modest and is clearly a second-order effect. Further, for the PGV to be a significant contributor to the angular velocity at impact, the fixturing must fail; the waste package must reach the imminent-toppling configuration at the time of PGV; and the PGV must be applied in the proper direction. These considerations support the conclusion that the current treatment of initial velocity for tip-over calculations is appropriate.

Table 54. Resultant Maximum Stress Intensity by Parameter

	Part	σ_{int} (MPa)	σ_{int} / σ_u
PGV*0	Outer Corrosion Barrier	902	0.93
	Inner Vessel	518	0.74
	Inner Lid	426	0.61
	Spread Ring	286	0.41
PGV*5	Outer Corrosion Barrier	944	0.97
	Inner Vessel	558	0.79
	Inner Lid	442	0.63
	Spread Ring	292	0.42
PGV*10	Outer Corrosion Barrier	1079	1.1
	Inner Vessel	644	0.92
	Inner Lid	478	0.68
	Spread Ring	302	0.43

Source: BSC 2003 [DIRS 166795], Table IV-2

7.1.2.3.2.6 Sliding and Inertial Effect of Waste Package Contents

Inertial effects of waste package contents are an intrinsic part of dynamic structural calculations performed explicitly by finite element analysis codes. Sliding effects of waste package contents during impacts are evaluated in calculations where the movement of such contents is reasonably anticipated to affect the kinematics and the resulting stress fields. Coefficients of friction are used based on the materials and situation. An example of the treatment of the waste package contents is the calculation entitled *44-BWR Waste Package Tip-Over from an Elevated Surface* (BSC 2003 [DIRS 166795]). In this calculation, the internals of the waste package and the commercial spent nuclear fuel assemblies are represented (BSC 2003 [DIRS 166795], Section 5.3, p. 17).

When the waste package contents are not considered as important to the resulting measures of waste package performance, those contents are often simplified so that the mass and inertial effects are maintained but geometry is simplified.

7.2 POSTCLOSURE

7.2.1.1 Thermal

The thermal calculations for normal operations are performed continuously through preclosure and postclosure times. Details are given in Section 7.1.1.1 above.

7.2.1.2 Structural

7.2.1.2.1 Static Waste Package on Pallet

For each of the waste packages three different variations in radial gap between the inner vessel and outer corrosion barrier were executed to produce a parametric result. This was done to create a solution that can be used for later modifications to the design. The radial gap sizes evaluated were 4 mm, 10 mm, and 15 mm. Table 55 from *Static Waste Packages on Emplacement Pallet* (BSC 2002 [DIRS 165492], Table 6-2, p. 11) shows that the Naval Long waste package is capable of supporting its own weight when on the emplacement pallet.

Table 55. Maximum Stresses Intensities in Outer Corrosion Barrier

Waste Package	4 mm Radial Gap (MPa)	10 mm Radial Gap (MPa)	15 mm Radial Gap (MPa)
21-PWR	90	80	90
44-BWR	86	80	116
Naval Long	74	84	76
5 DHLW/DOE SNF-Short	20	42	52

Source: BSC 2002 [DIRS 165492], Table 6-2, p. 11

The stresses reported are less than the yield stress of Alloy 22 (UNS N06022). The yield stress of Alloy 22 (UNS N06022) may be found in Section 5.3.2, Table 2 of this document. Therefore, the waste package is able to withstand the stresses of its own weight even after 10,000 years of degradation.

7.2.1.2.2 Postclosure Seismic Activity

For waste package postclosure seismic related issues refer to Appendix A of *Commercial SNF Waste Package Design Report* (BSC 2004 [DIRS 166876], Appendix A).

8. OPERATIONAL CONSIDERATIONS

Operational requirements are discussed in this section. Functional requirements are taken from *Naval Spent Nuclear Fuel Waste Package System Description Document* (BSC 2003 [DIRS 165427]). The applicability of the functional requirement for compliance demonstration by Waste Package and Components is also noted.

8.1 INFERFACE REQUIREMENTS

Functional Requirement Number: 3.1.3.1

Functional Requirement Title: Waste Package Handling Limits

Functional Requirement Text: Waste package handling shall not introduce any surface defect in the corrosion barrier exceeding those identified by performance assessment and on interface exchange drawings. Surface defects include, but are not limited to, scratches, nicks, dents, and permanent changes to the surface stress condition (Table 56).

Table 56. Waste Package Handling Limits Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	This issue is under investigation and will be resolved prior to construction authorization.	Yes

Functional Requirement Number: 3.1.3.2

Functional Requirement Title: Waste Package Closure

Functional Requirement Text: Sealing operations shall be performed on the waste package (Table 57).

Table 57. Waste Package Closure Performance Requirements

Performance Requirement Number	Performance Requirement Text	Applicability
1	Waste package sealing operations shall meet the requirements for the waste package as specified in the SDD for the waste package closure system.	Yes

8.2 INTERFACE WITH OTHER SYSTEMS

The loaded waste package has its final closure performed by the waste package closure system in accordance with Section 3.1.3, Functional Requirement Number 3.1.3.2 of BSC (2003 [DIRS 165427]), at which time it assumes its preclosure and postclosure functions.

During receiving, loading, sealing, and emplacement the waste package is handled by or interfaces with non-nuclear handling system, SNF/HLW transfer system, emplacement and retrieval system, remediation system, and emplacement drift system in addition to the waste package closure system. These systems must comply with Section 3.1.3, Functional Requirement Number 3.1.3.1 of BSC (2003 [DIRS 165427]). The waste package passes through the Warehouse and Non-Nuclear Receipt, Dry Transfer, Canister Handling, Remediation, and Subsurface Facilities.

The waste package is handled initially by the trunnions on the trunnion collars. The trunnion collars are installed upon receipt and removed after the waste package is returned to the horizontal position on the pallet. The waste package is loaded and under goes closure in the vertical position. After the waste package is placed on an emplacement pallet it is transported to the designated drift for emplacement and the trunnion collar is returned for reuse.

9. SUMMARY

The report describes the physical configuration of the naval waste package, describes the waste forms that it accommodates, and demonstrates how it responds to event sequences and prevents release of radionuclides. Also included are summaries of the assessments of ionizing dose rates from the enclosed waste forms and postclosure closure performance assessments that provide information to Performance Assessment. The results are reasonable compared to the inputs and are suitable for the intended use of this analysis. Sources of uncertainty in the sources analyses are described and the effect on the results discussed.

The design requirements and the supporting calculations are provided as justification for meeting each criterion in Section 7.1.2. An assessment of applicable design requirements for the naval waste package is summarized in Table 58 and are taken from BSC (2003 [DIRS 165427]).

Table 58. Summary of Design Performance Requirements

Functional Requirement Number	Performance Requirement Number	Performance Requirement	Comment
3.1.1.1	1	The sealed waste package shall not breach during normal operations or during credible preclosure event sequences.	Compliance demonstrated.
3.1.1.1	2	The waste package shall be designed and constructed to the codes and standards specified in the (Minwalla 2003 [DIRS 161362], Section 5.1.1.)	In progress.
3.1.1.1	3	Normal operations and credible event sequence load combinations are defined in the (Minwalla 2003 [DIRS 161362], Table 5.1.1-1).	Compliance demonstrated.
3.1.1.1	4	The waste package shall be designed to permit retrieval during the preclosure period until the completion of a performance confirmation program and Commission review of the information obtained from such a program.	Compliance demonstrated.
3.1.1.1	5	The waste package shall be designed to permit retrieval during the preclosure period so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission.	Compliance demonstrated.
3.1.1.1	6	The waste package shall be designed to meet the full range of preclosure operating conditions for up to 300 years after the final waste emplacement.	Compliance demonstrated.
3.1.1.2	1	In conjunction with natural barriers and other engineered barriers, the sealed waste package shall limit transport of radionuclides in a manner sufficient to meet long-term repository performance requirements.	Compliance demonstrated.
3.1.1.2	2	The waste package shall be designed and constructed to the codes and standards specified in Minwalla (2003 [DIRS 161362], Section 5.1.1).	In progress.
3.1.1.2	3	Normal operations and event load combinations are defined in Minwalla (2003 [DIRS 161362], Table 5.1.1-2).	Compliance demonstrated.

Functional Requirement Number	Performance Requirement Number	Performance Requirement	Comment
3.1.1.3	1	The methodology defined in the Naval Nuclear Propulsion Program addendum (Mowbray 1999 [DIRS 149585]). *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) shall be used to demonstrate acceptable postclosure criticality control for canisters and the waste packages in which they are disposed.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.3	2	The methodology defined in the Naval Nuclear Propulsion Program letter (Griffith 2003 [DIRS 165175]) shall be used to demonstrate acceptable preclosure criticality control for canisters and the waste packages in which they are disposed.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.3	3	The Naval Nuclear Propulsion Program will verify meeting the criticality criteria.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.4	1	The sealed waste package environment shall provide conditions that maintain waste form characteristics that restrict transport of radionuclides.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.4	2	The Naval Nuclear Propulsion Program will establish and verify meeting the temperature limits for packages containing naval fuel.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.4	3	The Naval Nuclear Propulsion Program will establish and verify meeting short-term accident condition temperature limits (e.g., fire exposure) for packages containing naval fuel.	Addressed by the Naval Nuclear Propulsion Program.
3.1.1.4	4	The waste form region of the sealed waste package shall have an inert atmosphere with limited oxidizing agents.	Compliance demonstrated external to the naval SNF canister and addressed by the Naval Nuclear Propulsion Program internal to the naval SNF canister.
3.1.1.4	5	The waste package shall be designed to preclude chemical, electrochemical, or other reaction (such as internal corrosion) of the waste form so that there will be no adverse effect on normal handling, storage, emplacement, containment, isolation, or on abnormal occurrences such as a canister drop accident and premature failure in the repository.	Compliance demonstrated external to the naval SNF canister and addressed by the Naval Nuclear Propulsion Program internal to the naval SNF canister.
3.1.1.5	1	The maximum waste package power at emplacement is 11.8 kW.	Compliance demonstrated.

Functional Requirement Number	Performance Requirement Number	Performance Requirement	Comment
3.1.2.1	1	The naval short waste package shall accommodate a short (maximum 4.750 m [187.00 inches]) naval SNF canister with a maximum diameter of 1.689 m (66.5 inches) and made of stainless steel type 316L (UNS S31603).	Compliance demonstrated.
3.1.2.1	2	The naval long waste package shall accommodate a long (maximum 5.385 m [212.00 inches]) naval SNF canister, with a maximum diameter of 1.689 m (66.5 inches) and made of stainless steel type 316L (UNS S31603).	Compliance demonstrated.
3.1.2.1	3	The waste package shall accommodate a naval SNF canister with a maximum weight of 49 tons.	Compliance demonstrated.
3.1.2.1	4	The waste package shall accommodate a naval SNF canister with an axial and radial center-of-gravity range to be established by the Naval Nuclear Propulsion Program.	Compliance demonstrated.
3.1.3.1	1	This issue is under investigation and will be resolved prior to construction authorization.	Under investigation.
3.1.3.2	1	Waste package sealing operations shall meet the requirements for the waste package as specified in the SDD for the waste package closure system.	Under investigation.
3.2.1.1	1	Performance requirements will be developed prior to license application.	Under investigation.
3.3.1.1	1	The waste package inner vessel shall have one lid and be made of stainless steel type 316 (UNS S31600).	Compliance demonstrated.
3.3.1.1	2	The waste package outer corrosion barrier shall have two lids and be made of Alloy 22 (UNS N06022).	Compliance demonstrated.
3.3.2	1	Performance requirements will be developed prior to license application.	Under investigation.

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BSC

Engineering Change Notice

Complete only applicable items.

1. QA: QA
2. Page 1 of 1

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3. Document Identifier:	4. Rev.:	5. Title:	6. ECN:
000-00C-DNF0-00800-000-00A	00A	Naval Waste Package Design Report	1

7. Reason for Change:
Per LP-3.12Q-BSC Design Calculations and Analyses Section 5.1 [2]c,

"The decision of the DEM, PCSA Manager, Criticality Manager, or PCA Manager to issue calcualtions or analyses with a "committed" status will be based on an experienced assessment of the likelihood that the results of the calculation or analysis will change, and the degree of impact those changes will have on designs that support the regulatory submittals or procurement activities, based on the design bounding conservatism."

The status designation of *Naval Waste Package Design Report* (000-00C-DNF0-00800-000-00A) can be changed to "Committed" as the results are not expected to change in such a manner that will affect support of regulatory submittals.

8. Supersedes Change Document: ☐ Yes If, Yes, Change Doc.: ______ ☒ No

9. Change Impact:

Inputs Changed: ☐ Yes ☒ No	Results Impacted: ☐ Yes ☒ No
Assumptions Changed: ☐ Yes ☒ No	Design Impacted: ☐ Yes ☒ No

10. Description of Change: (Address any "Yes" answers)
Add a "Committed" option in Block 7. On the cover sheet and change the "Document Status Designation" from Preliminary to "Committed". Block 7 on the cover sheet should read as follows

7. Document Status Designation:
☐ Preliminary ☒ Committed ☐ Final ☐ Cancelled

11. Originator: (Print/Sign/Date) Tim Schmitt	[signature]	8/12/05
Checker: (Print/Sign/Date) Michael Mullin	[signature]	8/12/05
Approved: (Print/Sign/Date) Michael J. Anderson	[signature]	8/12/05