BNFL
Inc.

Big Rock Point
Reactor Vessel Package
Safety Analysis Report

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1.0 GENERAL INFORMATION

This chapter of the Big Rock Point (BRP) Reactor Vessel Package (RVP) Safety Analysis Report (SAR) presents a general description of the package including its contents, and the application basis for package approval.

1.1 INTRODUCTION / PURPOSE OF APPLICATION

Consumers Energy's BRP Nuclear Power Plant located in Charlevoix, Michigan ceased operation in August of 1997. BRP began commercial operation in September of 1962, with a 67 Megawatt electrical (MWe) capacity, and was shut down after 35 years of power generation. The reactor was a General Electric Boiling Water Reactor (BWR) rated at 240 Megawatt thermal (MWt). Consumers Energy elected to decommission the unit in accordance with the DECON option in the NRC's Final Generic Environmental Impact Statement, NUREG-0586. The plan for decommissioning was submitted to the NRC in the "Post Shutdown Decommissioning Activities Report (PSDAR) for Big Rock Point," in September 1997.

Disposal of the Reactor Vessel (RV) is part of BRP's decommissioning activities. The BRP RV will be packaged and shipped for burial as low level radioactive waste. This SAR serves as the basis of the application for the BRP RV package[1] approval for shipping. The BRP package is a newly designed exclusive use, Type B(U)-85, Category II (See Section 1.2.6) package in accordance with the requirements of 10 CFR 71, "Packaging and Transportation of Radioactive Material." The model number assigned to the package is "BRP RVP SAR-5339." The package shipment will be a one-time transportation for burial at the licensed low level radioactive waste disposal facility of Chem-Nuclear Systems at Barnwell, South Carolina. The application basis is described in Section 1.1.1.

The Quality Assurance (QA) requirements of 10 CFR 71, Subpart H, applicable to the design, fabrication, assembly, and use of packaging for radioactive materials are covered by BNFL Inc.'s Quality Assurance Program which complies with 10 CFR 71, Subpart H, Quality Assurance. This program was approved and docketed by the NRC via Docket Number 71-0912, issued to BNFL Inc. on March 27, 2001.

1.1.1 Application Basis

The package presented in this application complies with the 10 CFR 71 criteria for a Type B package. The criteria of 10 CFR 71.71 for Normal Conditions of Transport (NCT) are used in

[1] The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

the package design. Chapters 2, 3, 4, 5, and 6 demonstrate the package design adequacy under the NCT by analysis.

This package is also evaluated for the 10 CFR 71.73 Hypothetical Accident Conditions (HAC) by analysis. In compliance with 10 CFR 71.73 criteria, the package structural performance is evaluated in Chapter 2 under the HAC to determine the post-accident condition of the package. The results of this structural evaluation are then considered in Chapter 4 in order to determine whether the post-accident condition of the package is within the acceptance criteria of 10 CFR 71. The results of the Chapter 4 evaluation verify compliance with the 10 CFR 71.51(a)(2) HAC radioactivity dose rate and release limits.

A brittle fracture evaluation is performed in Chapter 2 based on the criteria of NUREG/CR-1815. As discussed in Section 2.8, the results of this evaluation show that in order to protect the package material from brittle fracture, transportation must be halted if the ambient temperature falls to -5°F. To provide margin, a minimum temperature limit of 0°F will be enforced during package transportation. Therefore, operating controls will be put into place to ensure compliance with this transportation temperature condition as stated in Chapter 7.

In summary, the package design complies with all of the 10 CFR 71 criteria for a Type B package, with a temperature limit imposed on the transportation through appropriate operating controls. This SAR constitutes the basis for the application of the package design approval from the NRC as a Type B package under the provisions of 10 CFR 71.41(c).

1.2 PACKAGE DESCRIPTION

1.2.1 Package

The BRP package is designed as a 10 CFR 71 Type B package. This package is a cylindrical steel shell with welded steel flat top and bottom covers. The overall dimensions of the package are approximately 25' long and 13' in diameter. The package shell is 3" thick with an additional 4" thick shield plate welded to the interior surface of the 3" thick cylinder shell. This interior shield plate is approximately 8' long covering the former reactor core region of the RV. The top and bottom covers are 4" thick steel plates. The package material of construction is ASME SA-516, Grade 70. The package configuration is presented in Figure 2-1.

The design details satisfying structural, thermal, containment, shielding, and criticality criteria of 10 CFR 71 are presented in Chapters 2 through 6 of this SAR. Chapter 8 of this SAR, "Acceptance Tests and Maintenance Program," addresses all the tests and inspections required for compliance with the design of the package per the criteria of 10 CFR 71. Codes and standards applicable to design, fabrication, assembly, and testing, of the package are specified throughout this SAR.

The fabricated package will be delivered to BRP as one section, with the top plate to be welded to the cylindrical shell after the package has been loaded. The loaded package will contain the RV and its remaining internals (Section 1.2.5), as well as Low Density Cellular Concrete (LDCC) filling the void space in the RV and in the annulus between the RV and the package. The maximum total weight of the loaded package is 565,000 lbs. This total weight does not include the lifting devices and tie-down cables, as they are not integral parts of the package. The package component weights are presented in Chapter 2.

LDCC with a density range of 30-36 lb/ft^3 will be injected into the RV through penetration holes drilled in the top plate previously used for lifting attachments. Penetration holes in the body of the package will be used to fill the annulus between the RV and the package with LDCC with a density range of 50-60 lb/ft^3. The penetration holes will be plugged and seal welded after the process of placing the LDCC has been completed. The general configuration of the package is illustrated in Figure 2-1.

Once the loaded package is ready to be shipped, it will be hydraulically moved onto the transporter and tied down to ensure it remains in place during transportation. Details of the package lifting, loading, and transportation to the disposal facility are summarized in Chapter 7, "Operating Procedures."

Containment of the radioactive material is provided by the package cylindrical shell, top and bottom plates, and welded penetrations on the top plate and the body of the package. The RV internals are fixed in place by the existing hardware. The LDCC will fill the open space in the package and will fix any RV surface contamination. Therefore, the LDCC prevents surface contamination migration, component shifting, and dose rate changes during transportation. Once the top plate is welded to the cylindrical shell of the package, and the penetration holes have been seal welded, the package will be an integral unit with no features that allow the package to be opened. Containment is further detailed in Chapter 4.

Shielding of the radioactive material, as discussed in Chapter 5, is provided by the package shell, the top and bottom plates, and the additional 4" thick shield plate at the interior surface of the shell at the former reactor core region. The LDCC in the package will provide minor additional shielding.

1.2.2 Package Lifting Devices

A lifting lug and trunnions will be provided for the up-ending and down-ending process before and after the RV is loaded into the package as discussed in Chapter 7. The lifting devices are bolted attachments and are not integral parts of the package. These devices will be removed prior to package transport, and are not part of the package safety analysis for Part 71 considerations.

1.2.3 Package Tie-down Devices

The package tie-down system consists of saddle assemblies between the package and the transport vehicle, wire rope hold-down cables, and two end-stop assemblies. These components will not be mechanically attached to the package, and are not part of the package safety analysis for Part 71 considerations. Package tie-down is discussed in Chapter 7.

1.2.4 Operational Features

The package is a welded integral unit providing containment and shielding of the radioactive components. The penetration holes will be plugged and seal welded prior to the shipment, and there are no vents, valves, connections, etc. Therefore, there are no operational requirements for the package.

1.2.5 Package Contents

The radioactive content of the package is normal form radioactive material composed of the RV without the vessel top head, some of the RV stainless steel mirror insulation, and selected fixed RV internals as presented in Figure 1-1. These components are described in detail below. The package component weights are presented in Chapter 2. The radioactive material consists entirely of irradiated (activated) steel with a small quantity of surface contamination.

- **Reactor Vessel (RV)**

  The RV is a cylindrical unit with a hemispherical lower head integral with the vessel shell plates and a removable closure top head. The top head will not be included in the package and will be shipped and disposed of separately. The RV is approximately 24'-0" long (without the top head) with an outside diameter of 11'-5½" at the top head flange. The RV wall is 5 ¼" thick carbon steel with minimum 5/32" thick stainless steel cladding.

  The RV is principally fabricated from ASME SA-302 Grade B steel. The RV has steel nozzle penetrations ranging from ½" to 20" in diameter. Before placement in the package, the RV nozzles will be cut and closed within the envelope defined by the RV upper flange diameter. All the RV internal surfaces that were in contact with the coolant are clad with 304 stainless steel.

- **RV Insulation**

  The outside of the RV is covered with 3" thick 304 stainless steel mirror insulation. A portion of the insulation at the bottom of the RV has been removed and disposed of separately. This has been done to allow the RV to fit securely inside the donut support at the bottom of the package.

- **RV Internals**

  Of the RV internals, the Neutron Windows and the Top Guide Grid Bars with the exception of the Grid Bar End Pieces (GBEP) have been removed. These components will not be included in the package, and will be disposed of separately since they are Greater Than Class C (GTCC) waste.

  The remaining RV internals, as depicted in Figure 1-1, are comprised of stainless steel components, and will be left in place to be shipped as part of the package. These components are the Top Guide Plate, GBEP, Steam Baffle, Emergency Cooling Sparger, Seal Housing, Thermal Shield, Thermal Shield Retainer, Seal Weights, Core Support Plate, Inlet Diffuser, and Inlet Baffle (Flow Distributor).

Details on the radioactive source term are presented in Section 5.2. The total radionuclide inventory of the package used in the package evaluations is presented in Table 5-4. The table provides radionuclide activities for both internal surface contamination and neutron-activated material. The activities are decay corrected to September 2002. Based on measured dose rates, isotopic analyses of samples, and conservative calculations, the total activity in the package will be less than 13,100 curies on September 1, 2002 (Appendix 1-2). Of this, 2.90 curies are contamination on surfaces inside the RV, and the rest are activation products immobilized in the matrix of large steel components. There is less than 0.05 gram of fissile material and less than 0.05 curies of plutonium (Appendix 1-2). There are no gaseous radionuclides (i.e., Xe and Kr), no liquid radioactivity, and the entire radioactive inventory is immobile and fixed in place. The principal gamma emitter is Co-60 which accounts for 54% of the total activity and > 99% of the total dose rate. The Co-60 content of each component is summarized in Table 5-5.

The RV internals are fixed in place by the existing hardware. The LDCC emplacement program, as discussed in Chapter 7, will ensure that the LDCC will fill the open space including RV penetrations, and that there will be no free standing water after the concrete has cured. The final package is a single, solid, integrated unit consisting of steel and LDCC.

Potential for reaction amongst the various materials of the package and its contents is discussed in detail in Sections 2.4.4 and 4.2.2. In summary, the package materials are such that no significant galvanic or chemical reactions will occur. The combustible gas (hydrogen) generated due to radiolytic decomposition of the LDCC inside the package is conservatively calculated to be 3.7 % of the free gas volume in any confined region of the package in one year. Combustion effect of the hydrogen is addressed in Chapter 4, and its effect on the internal pressure of the package is considered in Chapter 3.

The heat generated from radioactive decay has been conservatively calculated to be 485 BTU/hr (Appendix 3-1). This heat has been taken into consideration in the thermal evaluation of the package in Chapter 3. As concluded in that chapter, the maximum normal operating pressure of the package is 15.0 psig at elevation 591 feet and 17.9 psig at elevation 6684 feet (See Section 3.4.4 for details.).

1.2.6 Determination of Package Category

10 CFR 71 requires that a Type B package be designed for ambient temperatures as low as -20°F. At this temperature, several types of ferritic steels are brittle and subject to fracture. Regulatory Guide 7.11 addresses fracture toughness criteria and test methods acceptable to the NRC for use in evaluation of Type B packages with a maximum 4 inch thickness. Regulatory Guide 7.11 recommends test methods based on various package categories defined in that guide's Table 1.

The package categories in Regulatory Guide 7.11 depend on the radioactivity of the package contents. The calculation in Appendix 1-1 examines the package content's material type and total activity against the limits of Regulatory Guide 7.11, and concludes that the package is Category II. Fracture toughness criteria appropriate for this category are addressed in Chapter 2 of this SAR.

1.3 REFERENCES

1.3-1 NUREG-0586, "Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities," August 1988, U. S. NRC.

1.3-2 "Post Shutdown Decommissioning Activities Report (PSDAR) for Big Rock Point," Revision 1, September 1997, Consumers Energy Company.

1.3-3 10 CFR 71, "Packaging and Transportation of Radioactive Material", March 31, 1999, U. S. NRC.

1.3-4 Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels With a Maximum Wall Thickness of 4 Inches," June 1991, U. S. NRC.

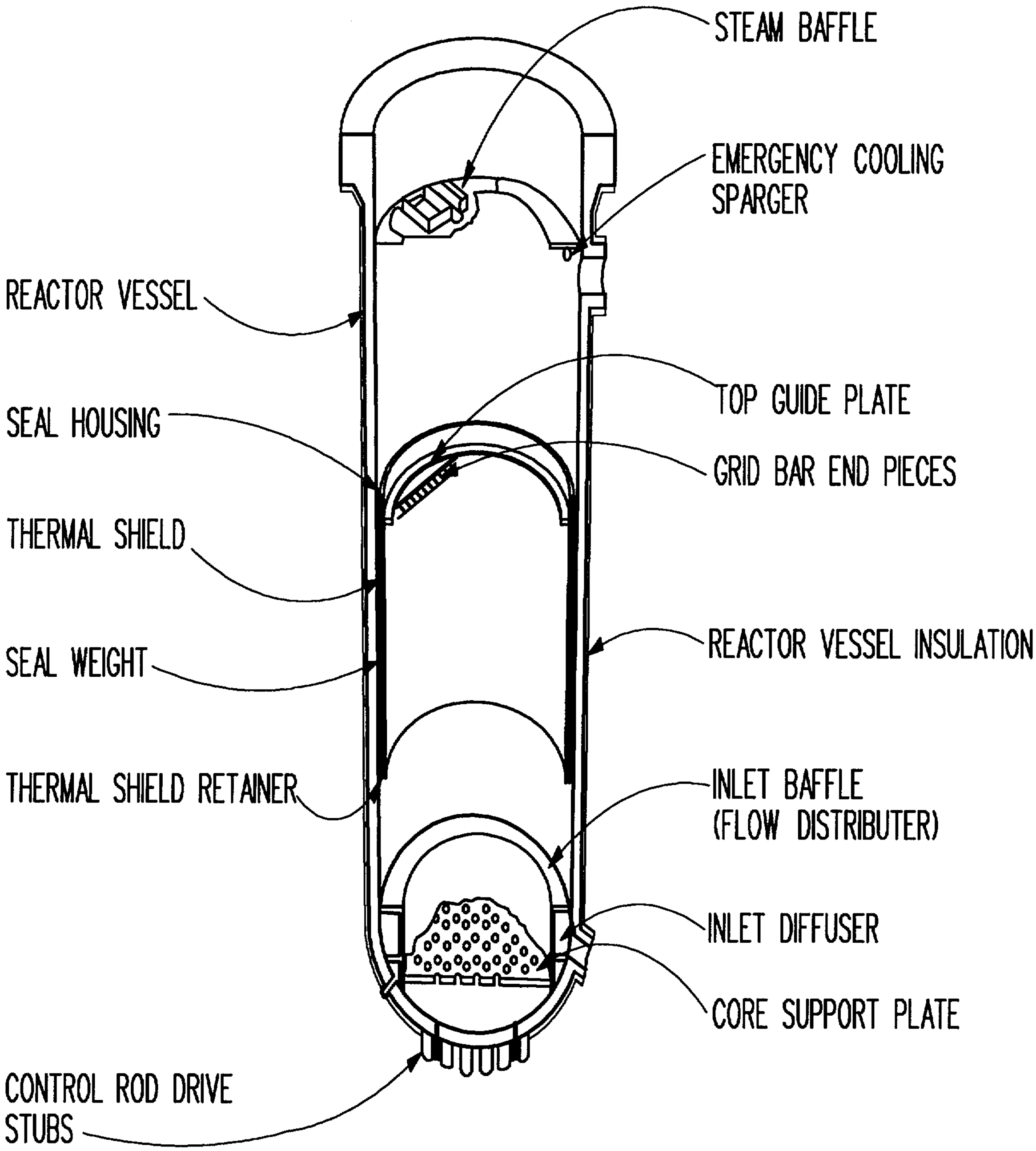

Figure 1-1: General Arrangement of Remaining RV Internals

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2.0 STRUCTURAL EVALUATION

This chapter presents the structural evaluation of the Big Rock Point (BRP) Reactor Vessel (RV) package[1]. The evaluations are performed in accordance with the requirements of 10 CFR 71 for an exclusive use Type B package.

2.1 STRUCTURAL DESIGN

The package is designed to meet the requirements of 10 CFR 71.71 for the Normal Conditions of Transport (NCT). The package performance is also evaluated under the Hypothetical Accident Conditions (HAC) outlined in 10 CFR 71.73. The post-HAC performance of the package is found to be within the acceptance criteria of 10 CFR 71.51(a)(2).

A brittle fracture evaluation of the package based on the criteria of NUREG/CR-1815 is performed in Section 2.8. To satisfy these criteria, a minimum temperature is imposed on the transportation as discussed in that section.

The structural design and acceptance criteria are discussed in Section 2.1.2. The structural evaluations summarized in this chapter are documented in Appendix 2-1.

2.1.1 Structural Description

The package is a cylindrical steel shell, approximately 13 feet in diameter and 25 feet long, with welded steel flat top and bottom plates. Material of construction of the package is ASME SA-516, Grade 70 steel. This package will be used for transportation and burial of the RV (SA-302, Grade B) and its internals detailed in Section 1.2.5. The void space in the package will be filled with Low Density Cellular Concrete (LDCC) as discussed in Section 1.2.1. The weight of the package components and contents, as well as the package center of gravity are discussed in Section 2.2. The package configuration and dimensions are provided in Figure 2-1. Fabrication will be in accordance with fabrication specifications satisfying the design requirements described in this SAR. Chapter 8 addresses the inspections and examinations that will be performed on the package for compliance with applicable design and regulatory requirements.

The package shell thickness varies along the length of the package to satisfy radiological shielding and structural design requirements. The cylindrical package shell is 3" thick with an additional 4" thick plate welded to its interior surface, providing 7" thick shielding for a length of approximately 8' in the former reactor core region of the RV. This extra internal shielding is

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positioned starting at a distance of approximately 8'-6" from the bottom of the package, and is secured by 1" fillet welds at both ends. Also, a 3" thick, 40" long reinforcing ring plate is attached to the interior surface of the package shell by ½" fillet welds at both ends. This ring plate is designed for use with the trunnion attachments which are part of the package lifting devices. The lifting devices will be removed from the package prior to shipment as stated in Section 2.5, but the ring plate will be included in the package configuration during transport. Therefore this plate is modeled in the computer structural analyses, but is not credited for shielding.

The package 4" thick bottom plate is shop welded to the shell with a full penetration weld. A donut shape support structure at the bottom of the package is designed to support the bottom of the RV when the package is in both horizontal and vertical positions.

The package top cover is a 4" thick circular plate. The RV flange is attached to the top plate[2] with fourteen of the forty-two A-193 RV flange existing studs re-threaded, and new SA-105 or SA-266, Grade 2 cap nuts. These studs support the RV during loading, NCT, and HAC. After the RV has been loaded into the package, the top plate will be field welded to the package shell with a full penetration weld, as discussed in Section 7.1.2.

The remaining twenty-eight RV studs will be cut off as close as possible to the RV flange. However, there may be a potential for some of these studs to extend beyond the RV flange and cause an interference with the top plate. To facilitate this potential interference, twenty-eight dimples are specified on the top plate inner surface, along the stud circle as shown in Figure 2-1.

The penetration holes in the body of the package as well as the holes in the top plate used for injecting the LDCC as discussed in Sections 1.2.1 and 7.1.2 will be plugged and seal welded after use and prior to shipment. The RV attachment cap nuts will also be seal welded.

2.1.2 Design and Acceptance Criteria

The package is designed for the NCT of 10 CFR 71.71. The performance of the package is also evaluated under the HAC of 10 CFR 71.73. The load combinations used in the analyses are in compliance with the guidance set forth in Regulatory Guide 7.8 as shown in Table 2-1.

Compliance with the "General Standards for All Packages" specified in 10 CFR 71.43 and the "Lifting and Tie-down Standards" specified in 10 CFR 71.45 are discussed in Sections 2.4 and 2.5 respectively.

[2] A lifting lug will be bolted to the top plate for loading the RV into the package and positioning the package during up-ending and down-ending operations discussed in Chapter 7. The lifting lug is not a structural part of the package (Section 2.5) and will be removed prior to shipment (Section 7.1.2.1). The lifting lug bolt holes will be plugged and seal welded after use and prior to shipment.

Additional details on the evaluation methodology and acceptance criteria under NCT and HAC are presented in Sections 2.1.2.1 and 2.1.2.2, with buckling and fracture toughness criteria addressed in Sections 2.1.2.3 and 2.1.2.4.

2.1.2.1 Normal Conditions of Transport

The design of the package under the NCT is in compliance with the guidance set forth in Regulatory Guide 7.6. Linear elastic analyses are used in the structural evaluations under the NCT. For vibration analysis, the package is designed to sustain the vertical, lateral, and longitudinal accelerations specified by:

- The Association of American Railroads (AAR) set forth in Part 1 of the "Open Top Loading Rules Manual," and
- ANSI N14.2, "Tie-down for Truck Transport of Radioactive Materials, (Draft)".

Material properties and design stress intensities (Sm) for the package shell are taken from the ASME Boiler and Pressure Vessel Code (ASME BPV), Section III, Division 1, Subsection NB. For the NCT load cases, service level A applies with the following stress limits:

Package Shell, Top and Bottom Plates, SA-516 Gr. 70

Allowable primary membrane stress per Subsection NB-3221.1:	1Sm
Allowable primary membrane plus bending stress per Subsection NB-3221.3:	1.5Sm
Allowable primary plus secondary stress per Subsection NB-3222.2:	3Sm

At a temperature of less than 200°F, 1Sm = 23.1 ksi

RV Studs at Top Plate[3]

Allowable average stress per Subsection NB-3232.1:	2Sm
Allowable maximum stress per Subsection NB-3232.2:	3Sm

At a temperature of less than 200°F, 1Sm = 23.3ksi

Package Shell, Top and Bottom Plates Full Penetration Welds

The allowable stress value for the full penetration weld is the same as the base material.

[3] The existing RV flange studs, which are used to secure the RV to the top plate, are A-193 with a minimum yield strength of 120 ksi and a minimum tensile strength of 140 ksi. The structural evaluation documented in Appendix 2-1 uses the material properties for SA-193 Grade B7 studs. This is conservative since the strength of A-193 studs with special properties is much higher than the SA-193 Grade B7 studs.

Fillet Welds

The allowable shear stress on the effective area is 0.3 times nominal tensile strength of the weld material per ASME BPV, Section III, Article NF-3226.2.

Fatigue stress analysis for stresses under NCT is performed in accordance with the cumulative damage criteria given in ASME BPV, Section III, Article NB-3222.4 and the guidelines given in Regulatory Guide 7.8.

2.1.2.2 Hypothetical Accident Conditions

Material properties for the package shell are taken from the ASME BPV, Section III, Division 1, Subsection NB.

The package structural performance evaluations under the HAC are performed using non-linear elastic-plastic dynamic impact Finite Element Analysis (FEA) in order to predict any permanent deformation and potential local structural failure caused by the subject accidents. In these analyses, the element maximum principal stress is monitored interactively throughout the transient. If this stress at any element exceeds the ASME BPV specified ultimate tensile strength, Su, a local failure is assumed at that element. The structural stiffness of the failed element is effectively removed from the analysis so that load is transferred to the adjacent elements. The process is continued until the impact velocity diminishes without a gross structural collapse. The cumulative local damage, or the total failed element area, is then considered to be the total opening area of the package after the HAC.

The total damaged area of the package due to the HAC must be less than any of the following non-mechanistic scenarios discussed in Section 4.3.1 in order to demonstrate compliance with the radiation and radioactivity release limits under the HAC set forth in 10 CFR 71.51(a)(2):

- A 1" wide circumferential gap forming in the 7" thick package shell, exposing a 1" wide band of the RV and its insulation along the package beltline, in the former reactor core (Figure 2-1) region.
- A 1" wide by 48" long longitudinal gap forming in the package, exposing a 1" by 48" longitudinal band of the RV and its insulation, centered on the package beltline, in the former reactor core region.
- A 6" diameter hole through the 7" thick package shell, centered on the package beltline in the former reactor core region, exposing a 6" diameter area of the RV and its insulation.
- The top plate completely separates from the package.

Consideration of creep and thermal ratcheting is not required in the package design for all of the loading conditions, as discussed in Regulatory Guide 7.8. However, for the HAC fire condition, temperature dependent material properties with reduced strength values at elevated temperatures are considered to assess the cumulative damage. The material properties at elevated temperatures are provided in Tables 2-3 through 2-7.

Under impact, local stresses at contact areas are primarily compressive. Compressive contact yielding will increase the local contact area but will not cause a gross structural failure. Potential buckling of the structure under a compressive load is included in the analysis by utilizing the large deformation feature of the FEA as discussed in Section 2.1.2.3. Therefore, the failure criterion based on the maximum principal stress is justified for all HAC analyses. The ASME BPV minimum tensile strength of the plate material defines the point at which the element is no longer effective and is removed from the model.

The following stress limits apply for the welds and studs:

RV Studs at Top Plate

Tension and shear are the loading on the RV studs; therefore, the failure criterion for the RV studs under the HAC is based on the maximum stress intensity. The ASME BPV minimum tensile strength of the stud material is the basis for the failure criterion under impact loading.

Package Shell, Top and Bottom Plates Full Penetration Welds

The allowable stress value for the full penetration weld is the same as the base material.

Fillet Welds

The allowable shear stress on the effective area of the internal shield plate and reinforcing ring plate fillet welds is 0.45 times the nominal tensile strength of the weld material per NF-3226.2

Stresses due to impact loads on the donut support are predominantly compressive. Compressive forces are transmitted directly by contact between the support plates and the package. Therefore, fillet welds are not directly in the main load paths for any loading condition. The double fillet welds of the donut support plates are designed to resist moment loading resulting from plate distortions under the impact load. The effective section modulus of the double fillet welds per unit length is greater than that of the connecting plates. Therefore, the double fillet weld is as strong as a full penetration weld which has the same allowable stress as the base material. Consequently, a separate weld stress calculation is not required.

2.1.2.3 Buckling

Structural buckling of the package under NCT and HAC is evaluated. Buckling of the package shell under external pressure is evaluated using the maximum allowable external pressure per ASME BPV, Section III, Article NB-3133. Buckling of the package shell under the HAC impact loading is evaluated using large deformation FEA. With large deformation FEA, load is applied

and updated gradually in the deformed configuration to ensure that the deformed structure is stable and can sustain the maximum applied load. Converged solutions of the package drop analyses demonstrate the adequacy of the package for HAC buckling loads.

2.1.2.4 Fracture Toughness

10 CFR Part 71 requires that Type B packages used to transport radioactive material be designed with consideration of NCT and HAC that might occur at –20°F, the Lowest Service Temperature (LST). The provisions of Regulatory Guide 7.11 and NUREG/CR-1815 are used in the measurement and determination of the Nil Ductility Transition (NDT) temperature for the package plate material to meet the 10 CFR 71 criteria.

2.2 WEIGHTS AND CENTERS OF GRAVITY

The maximum total weight of the package including the RV, remaining RV internals and LDCC, is 565,000 lbs. This total weight does not include the lifting devices and tie-down cables, as they are not integral parts of the package (see Section 2.5). Itemized weights of various components are provided in Table 2-2. The package Center of Gravity (CG) axial location is approximately 12'-8" from the bottom of the package. The package radial CG is centered along the longitudinal axis. See Figure 2-1.

2.3 MECHANICAL PROPERTIES OF MATERIALS

The temperature dependent mechanical properties available from ASME BPV, Section II, Part D, Subpart 1, Tables TM-1, 2A, U and Y-1 are presented in Tables 2-3 and 2-4. Table 2-5 presents the mechanical properties obtained from ASME BPV, Section II, Part D, Subpart 1, Tables 4 and Y-3 for SA-193 B7 (1Cr-1/5Mo) with bolt diameter greater than 4".

For HAC fire accident analysis, Table 2-6 presents the material yield strength and tensile strength at elevated temperatures, calculated from the temperature trend curve in "European Convention for Constructional Steelwork" given in ASCE Manual No. 78, "Structural Fire Protection". The same temperature trend curve is assumed for the material tangent modulus. This assumption is based on the plastic slopes of the stress strain curves of ASTM A36 material at various temperatures as shown in Figure 2.8 of the ASCE Manual. Table 2-7 presents the material temperature-dependent tangent modulus used in the HAC fire accident.

2.4 GENERAL STANDARDS FOR ALL PACKAGES

10 CFR 71.43 establishes the general standards for packages. This section identifies these standards and provides the bases that demonstrate compliance.

2.4.1 Minimum Package Size

10 CFR 71.43(a) requires that:

> "The smallest overall dimension of a package must not be less than 10 cm (4 in)."

The smallest overall dimension of the package is the 13'-0" outside diameter, therefore, the minimum package size requirement is met.

2.4.2 Tamper-Proof Feature

10 CFR 71.43(b) requires that:

> "The outside of a package must incorporate a feature, such as a seal, which is not readily breakable, and which, while intact, would be evidence that the package has not been opened by unauthorized persons."

The package will have welded closures, seal welds on all threaded plugs, and seal welds on the RV attachment cap nuts. Therefore, the requirements for tamper-proof features are met.

2.4.3 Positive Closure

10 CFR 71.43(c) requires that:

> "Each package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package."

The package is a welded containment system. All threaded plugs and the RV attachment cap nuts are seal welded. The package integrity under internal pressure is demonstrated in various applicable load combinations as described in Sections 2.6.1 through 2.6.11. Thus, unintentional opening will be precluded, thereby satisfying the positive closure requirements.

2.4.4 Chemical and Galvanic Reactions

10 CFR 71.43(d) states:

> "A package must be of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent. Account must be taken of the

behavior of materials under irradiation."

The containment boundary and internal support structure are constructed of welded carbon steel. The package contents (RV, remaining RV internals, insulation, and LDCC) are comprised of carbon steel, stainless steel, and concrete as discussed in Chapter 1. While the stainless steel insulation and the carbon steel package are dissimilar metals, the LDCC serves to ensure an essentially dry and benign environment inside the RV and between the insulation and the package shell. The LDCC is a Portland cement-based mixture. There are no significant chemical reactions between Portland cement mixtures and ferrous metals. Any free-standing water will be removed from the RV as stated in Section 7.1.1, and the controlled process used for injecting the LDCC will ensure that no excess water remains in the LDCC mixture (Section 7.1.2). The LDCC also acts as an insulator between the dissimilar materials, thus effectively eliminating galvanic or chemical reactions through direct contact or coupling.

The package boundary is a welded steel containment with seal welded penetrations. Inleakage of water is prevented since there are no openings in the package boundary as described in Section 4.1, and the structural integrity of the package will preclude inleakage of water to the package interior.

10 CFR 71.43(d) also requires that the "behavior of materials under irradiation" be considered. All materials of construction in the package are steel or cured concrete (LDCC), and no organic/polymeric materials are used. These materials do not degrade with the gamma radiation dose rates associated with this package.

The potential for generation of combustible gases due to decomposition of water entrapped in the LDCC during package transport is addressed in Sections 4.2 and 4.2.2. The discussions presented in this section and Chapter 4 demonstrate compliance with the criteria of 10 CFR 71.43(d).

2.4.5 Package Valves

10 CFR 71.43(e) requires that:

> "A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operation and, except for a pressure relief device, must be provided with an enclosure to retain any leakage."

The package does not contain valves or other pressure relief devices. Containment penetrations will be closed and seal welded, thus precluding escape of the contents. The design of the package meets the requirements of 10 CFR 71.43(e)

2.4.6 Effectiveness of Package Containment

10 CFR 71.43(f) requires that:

> "A package must be designed, constructed, and prepared for shipment so that under the tests specified in § 71.71 ("Normal conditions of transport") there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging."

The package is designed in accordance with the design criteria for the structural analysis of Type B packages provided in Regulatory Guide 7.6, thus meeting the NCT as specified in 10 CFR 71.71. The package is constructed in accordance with the requirements of the ASME BPV, Section III, Division 1, Subsection NB and complies with the applicable Standards for shipping (ANSI N-14.2 and AAR). Therefore, there will be no loss or dispersal of radioactive contents, and no substantial reduction in the effectiveness of the packaging under NCT. Compliance with "no significant increase in external surface radiation levels" is demonstrated by compliance with the same requirement stated in 10 CFR 71.51(a)(1) as addressed in Section 5.1.2. The requirements of 10 CFR 71.43(f) are therefore satisfied.

2.4.7 Accessible Surface Temperature

Compliance with the accessible surface temperature limit of 10 CFR 71.43(g) is demonstrated by the thermal evaluation of the package as discussed in Section 3.1.3.1.

2.4.8 Venting

10 CFR 71.43(h) requires that:

> "A package may not incorporate a feature intended to allow continuous venting during transport."

The package is a welded container with no feature for continuous venting during transport. Thus, the package design meets the requirement of 10 CFR 71.43(h).

2.5 LIFTING AND TIE-DOWN DEVICES

10 CFR 71.45 specifies the requirements for the lifting and tie-down devices that are "structural parts of the package". The lifting and tie-down devices for the BRP RV package are designed such that they are not "structural parts of the package". Therefore, their design is not part of the package safety analysis for Part 71 considerations, and the criteria of 10 CFR 71.45 do not apply.

The configuration details of these devices are presented in Sections 7.1.2.1 and 7.2.3.1.

2.6 NORMAL CONDITIONS OF TRANSPORT

This Section demonstrates that the package is structurally adequate to meet the performance requirements of Subpart E of 10 CFR 71 when subjected to NCT as defined in 10 CFR 71.71(c)(1) through 71.71(c)(10) with the initial conditions defined in 10 CFR 71.71(b). Compliance with these requirements is demonstrated by analysis in lieu of testing as allowed by 10 CFR 71.41(a) and Regulatory Guide 7.6.

The FEA Program ANSYS, Version 5.6, is used to analyze the package for the loading conditions of NCT. The description for each analysis is provided in this Section. The ANSYS element type SHELL181, a 3-D four-node shell element with both bending and membrane capabilities, is used to model the package cylindrical shell, package top and bottom plates, RV, donut support, internal shield plate, and ring plate. The package cylindrical shell nodal spacing matches the saddle support locations. This can be observed in Figures 2-3 and 2-4. Figure 2-2 shows the element mesh of the model. One half of the package is modeled by symmetry. A brief description of the model is presented below:

- The bolted connection between the RV flange and the package top plate is modeled as rigid links at the RV stud circle.
- The dimples on the top plate described in Section 2.1.1 are excluded from the model. As discussed in Appendix 2-1, the dimples have negligible effect on the top plate stresses and displacements, and do not impact the package stress analyses presented in this chapter.
- The fillet welds of the internal shield plate and the ring plate are modeled by translational Degree Of Freedom (DOF) couplings with the connecting package shell nodes.
- The double fillet welds at the junction of the donut support and the package are modeled as rigid links in all six DOF's.
- For weight and shock analysis of the package during transportation, saddle supports are modeled as contact elements in the radial direction using ANSYS CONTAC52 elements.
- Tie-down cables are modeled using tension only link elements, ANSYS LINK10.
- The weight of LDCC is included in the model. However, the LDCC structural stiffness is negligible in comparison with steel, and is not considered in the analysis.
- The model global Y-axis is the package longitudinal direction, the Z-axis is the vertical gravity direction of the package position during transportation, and the X-axis is the package lateral direction.

Initial conditions per 10 CFR 71.71(b) associated with the various load combinations applicable to the analysis for NCT as outlined in Regulatory Guide 7.8 are:

- Minimum and maximum ambient temperatures of -20°F and 100°F for the evaluation of each NCT except for the 10 CFR 71.71(c)(2) cold load condition which requires a temperature of -40°F.
- Maximum insolation at 400 g cal/cm^2 on horizontal curved surfaces, and 200 g cal/cm^2 on the vertical flat surfaces for a duration of 12 hours.

- Maximum decay heat generated by the radioactive material. For the BRP package, a total of 485 BTU/hr maximum decay heat was considered in the thermal heat transfer analysis in Chapter 3.
- Maximum and minimum normal operating pressures of 20 psig and –3 psig respectively, enveloping the pressures calculated in Section 3.4.4. These pressures are consistent with the ambient temperature, insolation and decay heat.
- Minimum internal pressure of –3.3 psig as calculated in Section 3.4.4 for the evaluation of 10 CFR 71.71(c)(2) cold load condition at -40°F.
- Fabrication stresses caused by interference fits and the effects of concrete shrinkage are negligible for the BRP package design. Bolt load is the only assembly stress which is required to be evaluated. The flange bolts are snug tight. Therefore, the package bolt assembly stresses are negligible. Note that the residual stresses due to plate formation and welding are not considered to be fabrication stress as noted in Regulatory Guide 7.8. Post-weld heat treatment of welded joints per ASME BVP, Section III, Subsection NB, provides stress relief for the package assembly.

2.6.1 Heat

The package is evaluated for an ambient temperature of 100°F with a total insolation of 400 g cal/cm^2 on the horizontal curved surfaces, and 200 g cal/cm^2 on the vertical flat surfaces for a duration of 12 hours (10 CFR 71.71(c)(1)). Primary weight load (1g inertial load in the Z direction) and maximum internal pressure (20 psig) are applied on the package, which is positioned on saddles during transport. This load case is designated as load case N1, and the results are shown in Table 2-8. The minimum safety margin against the Regulatory Guide 7.8 acceptance criteria is 6.4 for load case N1.

2.6.2 Cold

The ambient temperature is taken as -40°F with no insolation or decay heat to evaluate the package for the minimum temperature condition (10 CFR 71.71(c)(2)). Primary weight load (1g inertial load in the Z direction) and the minimum internal pressure at -40°F (-3.3 psig) are applied on the package, which is positioned on saddles during transport. This load case is designated as load case N2, and the results are shown in Table 2-8. The minimum safety margin against the Regulatory Guide 7.8 acceptance criteria is 5.4 for load case N2.

2.6.3 Increased External Pressure

An increased external pressure of 20 psia (10 CFR 71.71(c)(4)) is combined with the package minimum internal pressure (-3 psig) corresponding to an initial ambient temperature of -20°F, not including the effects of insolation or decay heat. Primary weight load (1g inertial load in the Z direction) is applied on the package, which is positioned on saddles during transport. This load case is designated as load case N3, and the results are shown in Table 2-8. The minimum safety margin against the Regulatory Guide 7.8 acceptance criteria is 5.6 for load case N3.

2.6.4 Reduced External Pressure

A reduced external pressure of 3.5 psia (10 CFR 71.71(c)(3)) is combined with the package maximum internal pressure (20 psig) corresponding to an initial ambient temperature of 100°F with the maximum effects of insolation and decay heat included. Primary weight load (1g inertial load in the Z direction) is applied on the package, which is positioned on saddles during transport. This load case is designated as load case N4, and the results are shown in Table 2-8. The minimum safety margin against the Regulatory Guide 7.8 acceptance criteria is 4.6 for load case N4.

2.6.5 Vibration

The package is evaluated for vibration normally incident to transport (10 CFR 71.71(c)(5)) under two different conditions. First, the package is evaluated for the effects of vibration at an ambient temperature of 100°F, considering maximum insolation and decay heat along with the corresponding maximum internal pressure (20 psig). Second, the package is evaluated for the effects of vibration at an ambient temperature of -20°F, considering zero insolation, zero decay heat, and the corresponding minimum pressure (-3 psig).

The tie-down design criteria of AAR and ANSI N14.2 are used in the vibration evaluations of the package. The enveloping vertical, lateral, and longitudinal accelerations are:

- Vertical direction: 2.0g
- Lateral direction: 2.0g
- Longitudinal direction: 3.0g

Shock loads produced by coupling, switching, etc., in rail transport and by bumps, potholes, etc., in truck transport are considered to be bounded by the above loads. Each shock load is analyzed independently. Shock acceleration is applied to the package, which is secured on saddles with cable ties and axial bumpers. A total of six load cases are applied by combining the three acceleration directions with the hot and cold environments, designated as N5, N6, N7, N8, N9 and N10, and the results are shown in Table 2-8. The minimum safety margin against the Regulatory Guide 7.8 acceptance criteria is 1.9 corresponding to load case N7, the lateral shock in the hot environment.

The normal condition shock fatigue evaluation of the package is conservatively based on the enveloped stress intensity range of the normal condition load cases N1 through N10. Longitudinal shock load cases N9 and N10 are the two highest stress load cases for the normal transport shock loading. Stress range, $2S_{alt}$, for load ranges F1 through F5 (Table 2-9), are calculated based on the alternating stress intensity S_{alt} definition given in Regulatory Guide 7.6 (combination rule based on principle stress). The results of the five greatest alternating stress ranges are shown in Table 2-9.

The maximum stress intensity range is $2S_{alt}$ = 17.3 ksi, including local stress at the saddle support. This is less than the acceptance limit of 3Sm = 69.3 ksi. The package shell has four 3" OD threaded holes for LDCC filling and several bolt holes at the lifting trunnions. These holes are not located near the high fatigue stress area. However, a local stress concentration factor, K = 4 (the maximum concentration factor suggested by Regulatory Guide 7.6) is used as a bounding peak stress index for fatigue evaluation.

The enveloped peak stress range, which is calculated from the maximum local stress range and the bounding stress concentration, is 4 x 17.3 ksi = 69.2 ksi. The maximum alternating stress, S_{alt} = 0.5 x 69.2 = 34.6 ksi, from which the allowable number of cycles is 1.38E+04, determined from the fatigue curve in ASME BPV, Section III, Appendix I, Figure I.9.1 for Su < 80 ksi, and E = 30E+06 psi. Therefore, the cumulative fatigue usage factor based on an estimated number of 1500 cycles of shock loads is 0.109. Note that shock load is produced by coupling, switching, etc. in rail transport and by bumps, potholes, etc. in truck transport. Thus, the number of full magnitude shocks is not expected to exceed one per mile during NCT. Based on the travel distance of approximately 1500 miles from BRP to the disposal site in Barnwell, SC, the estimated total number of full magnitude shocks is 1500. The estimated travel distance for the fatigue evaluation is not a critical factor, and may vary +/-100% without affecting the fatigue life of the package because the calculated fatigue usage factor has a margin of 9 to the ASME BPV allowable fatigue usage factor.

2.6.6 Water Spray

10 CFR 71.71(c)(6) requires an evaluation of the package for water spray that simulates exposure to a rainfall of approximately two inches per hour for at least one hour. As previously described, the package is fabricated from thick welded steel plate. All joints and openings are welded; therefore, water spray will have no effect on the package performance.

2.6.7 Free Drop

10 CFR 71.71(c)(7) calls for a free drop test/evaluation for packages in excess of 33,100 lbs through a distance of 1' onto a flat, essentially unyielding, horizontal surface. A linear elastic analysis is performed to evaluate the structural adequacy of the package under this condition.

During transport, the package rests horizontally in cradles and is secured by tie-down cables and axial bumpers. The package will be transferred by jacking from the road transporter to the rail car and will not be lifted during transport to the Barnwell rail siding. Considering the package orientation and position during transportation, loading operations, the size of the package, and based on the evaluations performed in Appendix 2-1, the governing drop orientation for the 1' drop during the NCT is horizontal. The flat, essentially unyielding, horizontal surface in the 1' drop is considered to be a flat surface of an infinite thick concrete slab.

The following two cases of horizontal drop are evaluated using hot and cold initial conditions:

- Load Case N11: A 1' horizontal drop with maximum internal pressure. This is a hot environment with an ambient temperature of 100°F, considering maximum insolation and decay heat along with the corresponding maximum internal pressure.
- Load Case N12: A 1' horizontal drop with minimum internal pressure. This is a cold environment with an ambient temperature of -20°F, considering zero insolation, zero decay heat, and the corresponding minimum internal pressure.

The lumped parameter dynamic analysis methodology per NUREG/CR–3966, "Methods for Impact Analysis of Shipping Containers," November 1987, is used to calculate the maximum dynamic responses due to the 1' drop. ANSYS dynamic transient large deflection analysis is used to calculate the responses of the lumped parameter model and to check buckling of the package. The calculated peak impact acceleration of 12g is then applied to the three-dimensional FEA model of the package for stress recovery and structural buckling analysis. The analysis results show that the package remains stable, and no buckling occurs as indicated by the convergence of the solution under the impact load.

The primary stress intensities for the 1' drop load cases N11 and N12 are shown in Table 2-8. The minimum stress margin is 1.2. The localized compressive stress at the impact surface is checked against the Regulatory Guide 7.6 acceptance criteria for the extreme total stress range. The results of the five extreme alternating stress ranges are shown in Table 2-10, with a minimum stress margin of 2.35.

2.6.8 Corner Drop

The corner drop in 10 CFR 71.71(c)(8) is not applicable to the package because the package is not constructed of either fiberboard or wood and does not contain fissile material.

2.6.9 Compression

The compression test in 10 CFR 71.71(c)(9) is not applicable to the package because the package weighs more than 11,000 lbs.

2.6.10 Penetration

The package is evaluated for the impact of the hemispherical end of a vertical steel cylinder of 1¼" diameter and 13 lb mass, dropped from a height of 40" onto the exposed surface of the package. The package is evaluated for this penetration condition utilizing the analytical method developed by the Ballistics Research Laboratories (BRL), and presented in ASCE Manual No. 58, "Structural Analysis and Design of Nuclear Plant Facilities." The recommended minimum design thickness calculated from the BRL formulation is 0.55". Since the minimum thickness of the package is 3", which is well over this recommended design thickness, the package shell thickness is adequate for resisting the required puncture loading.

2.6.11 Test Pressure

Since the maximum normal operating pressure in the package is greater than 5 psig as calculated in Section 3.4.4, a pressure test of the package is required to be conducted at an internal pressure of at least 50% higher than the maximum normal operating pressure in accordance with 10 CFR 71.85(b). A pressure test of the package will be conducted at an internal pressure of 30 psig as described in Section 8.1.3.

A structural evaluation of the package for this loading condition is also performed. The primary weight load and test pressure of 30 psig are applied on the package combined with hot (100°F ambient) and cold (-40°F ambient) environments, forming load cases TP1 and TP2 respectively. The stress intensity values for these load cases are summarized in Table 2-8. The minimum stress margin is 4.8.

2.6.12 Weld and Stud Evaluations

2.6.12.1 Package Welds

All load carrying welds such as the top and bottom plate corner welds and the package seam welds are full penetration welds made to ASME Section III, Subsection NB, and therefore, are equivalent to the base metal. The weld stress is the same as the stress in the plate at the junctures. Plate maximum stresses as discussed in the previous sections are all within the allowables with sufficient margin.

2.6.12.2 Internal Shield Plate and Ring Plate Welds

The fillet welds of the shield plate and the ring plate are modeled by translational DOF couplings with the connecting package shell nodes. The total forces of the DOF coupled nodes in these plates are used to calculate the average shear stress in their fillet welds. The NCT maximum shear stress in the shield plate and ring plate welds is 3 ksi and 2.2 ksi, respectively. These weld stresses are well below the NCT shear stress allowable of 21 ksi.

2.6.12.3 Donut Support Welds

The junctures of the donut support with the package shell and bottom plate are modeled as rigid links in all six DOFs. This means that the double fillet welds are transmitting all forces and moments across the junctions. However, loads on the donut support are predominantly in compression. Compressive forces are transmitted directly by contact between the plates. Therefore, fillet welds are not directly on the main load paths for all loading conditions. Moment loading resulting from plate distortion can be resisted by the fillet welds as discussed in Appendix 2-1. Therefore, the double fillet weld is considered to be as strong as the connecting plates, and the weld stress is approximately the same as the stress in the plate at the junctures.

2.6.12.4 RV Studs at Top Plate

The RV studs are modeled by translational DOF couplings between the RV flange and the package top plate. The axial and the resultant shear forces of the DOF coupled nodes are used to calculate the tensile stress and the average shear stress in the RV studs in accordance with ASME BPV Section III, NB-3232. The studs' resulting tensile and shear stresses are 11.3 ksi and 11.5 ksi respectively, which are less than the allowable stress of 46.6 ksi.

2.6.13 Conclusion of NCT Evaluations

The discussions presented in Sections 2.6.1 through 2.6.12 demonstrate that the package maintains its structural integrity under the NCT of 10 CFR 71.71. By demonstration of the package structural integrity, the containment and shielding integrity during the NCT are also confirmed. Specific requirements regarding the containment and shielding design are discussed in Chapters 4 and 5 respectively.

2.7 HYPOTHETICAL ACCIDENT CONDITIONS

This section presents the package performance evaluations under the HAC defined in 10 CFR 71.73(c)(1) through 71.73(c)(6). 10 CFR 71.73(a) outlines the required sequential application of the HAC to determine the cumulative effect on the package. The initial temperature and pressure conditions to be considered during the HAC are defined in 10 CFR 71.73(b). The package performance under these conditions is evaluated by analysis in lieu of testing as allowed by 10 CFR 71.41(a) and Regulatory Guide 7.6.

The FEA Program ANSYS, Version 5.6, is used to analyze the package for the HAC loading conditions. The description for each analysis is provided in this section. The ANSYS element type SHELL181, a 3-D, 4-node shell element, with both bending and membrane capabilities, is used to model the package cylindrical shell, package top and bottom plates, RV, donut support, internal shield plate, and ring plate. The element is well suited for elastic-plastic materials, large strain, and large deformation applications. The element has 5 integration points through the element thickness, and has both in-plane and out-of-plane bending and membrane loading capabilities. The SHELL181 element with full integration option is highly accurate even with coarse meshes. A benchmark analysis documented in Appendix 2-1 demonstrates high performance of SHELL181 in an analysis of a bar impacting a rigid wall.

The initial conditions defined in 10 CFR 71.73(b) and a brief description of the model are presented in Section 2.6. Note that for HAC loading, the studs between the RV flange and the package top plate are modeled using ANSYS plastic pipe element PIPE20. Stud stress intensity is calculated by ANSYS POST26 processor and is a part of the ANSYS output. Figure 2-5 presents the key locations of the package for the HAC analyses.

2.7.1 Free Drop

10 CFR 71.73(c)(1) requires a free drop test of the package specimen through a distance of 30 feet onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected. Assessment of the package damages resulting from the 30' drop is performed by non-linear elastic-plastic dynamic impact analyses. Several contact surfaces between the package and the unyielding, horizontal surface defined in Section 2.6.7 are modeled to be consistent with the various drop orientations. Also, potential contacts between the RV and the package interior surface are modeled at some local locations to correctly represent the internal effects of each drop orientation.

During analysis, the package stresses are monitored for element failure. Failure is considered to occur at any element for which the maximum principal stress exceeds the package material tensile strength. The failed elements are removed from the FEA model, and the analysis process is continued until the package displacements reach their maximum values. The following drop scenarios are considered:

- Vertical drop
- Horizontal drop
- Corner drop through the package center of gravity (CG)
- Oblique drop (slap down drop)

The results of the 30' free drop dynamic impact analyses are verified using the principle of energy conservation. The total kinetic energy (KE) and the cumulative plastic work should be approximately equal to the initial impact energy, $KE = \frac{1}{2} mv^2$, where "m" is the package mass and "v" is the impact velocity. Analysis result verification was performed to verify that the FEA model and the analyses are adequate for assessing the structural plastic deformation under the 30' drop impact loading. The verification shows that the total impact energy of $\frac{1}{2} mv^2$ is transformed into the total plastic work done at the final load step of the analyses, thereby demonstrating that the input energy is fully dissipated through plastic work in the package.

2.7.1.1 Vertical Drop

Since the RV support configurations within the package at the top and bottom ends are different as shown in Figure 2-1, two different vertical drop scenarios could occur. In a drop on the top plate scenario, since the RV flange is directly attached to the top plate by studs, the load transfer would occur through direct bearing. In a drop on the bottom plate scenario, since the RV's hemispherical end rests on the donut support, the distance through which it must travel to come into direct contact with the package bottom plate is large by comparison. This configuration is more susceptible to buckling in the package shell and to large shell deformations.

Therefore, a vertical drop on the bottom plate of the package is evaluated. In this analysis, the FEA model considers the contact between the bottom plate and the unyielding, horizontal surface along with the local contact between the RV and the bottom plate.

The high stress locations in this analysis occur at the center of the top plate, the top plate rim, and the lower shell junction at the donut support plate. During the impact, the peak value of the maximum principal stress is approximately 50 ksi, which is less than the material tensile strength of 70 ksi. The package gross structure remains intact after the impact; therefore, no containment failure and no gross buckling failure occur in the package. The maximum stud stress intensity is 93.7 ksi, which is less than the material tensile strength of 100 ksi. Figure 2-6 presents the horizontal deformation of the package due to this drop.

2.7.1.2 Horizontal Drop

A horizontal drop of the package causes concentrated damage at the two ends of the package, where the top plate and the bottom plate contact the unyielding, horizontal surface. The bottom plate and the donut support plate share the impact load at the bottom end while the impact load at the opposite end is taken only by the top plate. The FEA model considers contact between the package shell and the unyielding surface along with contact between the RV flange and the package shell.

The high stress location in this analysis occurs at the rim of the top plate near the point of impact. During the impact, the peak value of the maximum principal stress is approximately 68 ksi, which is less than the material tensile strength of 70 ksi. The package gross structure remains intact after the impact; therefore, no containment failure and no gross buckling failure occur in the package. The maximum stud stress intensity is 92 ksi, which is less than the material tensile strength of 100 ksi. Figure 2-7 presents the vertical deformation of the package due to this drop.

2.7.1.3 Corner Drop through Package Center of Gravity

The maximum potential damage to the package during a corner drop through the package CG occurs at the impact corner due to local crushing effects. The damage to the package from a CG drop on the bottom corner is greater than the damage to the package from a CG drop on the top corner. This is due to the larger distance through which the RV must travel to come into contact with the package bottom plate and transfer the load through direct bearing. Therefore, a CG drop on the bottom corner is considered in the analysis. The FEA model considers contact between a corner of the bottom plate and the unyielding, horizontal surface.

The high stress location in this analysis occurs at the shell junction with the donut support plate. During the impact, the peak value of the maximum principal stress reaches 71 ksi, exceeding the material tensile strength of 70 ksi for a short duration before reducing to below the material tensile strength. This may cause a circumferential crack to be initiated at the outer fiber of the package shell near the junction of the donut support plate, but breach of containment will not occur due to the short duration of this stress and the thickness of the package shell. The package gross structure remains intact after the impact, and no gross buckling failure occurs in the package. The maximum stud stress intensity is 95.36 ksi, which is less than the material tensile

strength of 100 ksi. Figures 2-8 and 2-9 show the vertical deformation and high stress locations on the package due to this drop.

2.7.1.4 Oblique (Slap Down) Drops

The dynamic responses of the package in several oblique drop scenarios are investigated using the lumped parameter dynamic analysis methodology per NUREG/CR-3966. The FEA lumped parameter model is a beam that has the same distributed mass, lumped masses, and cross-sectional moment of inertia as the package.

The lumped parameter beam is dropped onto a spring that represents the package stiffness at the impact corner. The spring force deflection curve is obtained from the impact force and deflection time histories of the corner drop analysis. Large deformation analysis is considered to capture the effect of rigid body motion and rotation. The 30' drop dynamic impact analyses for the lumped parameter model are performed for eight oblique drop angles ranging from 60 degrees to 5 degrees off the horizontal plane.

The maximum vertical slap down velocity at the top plate corresponds to a primary 10 degree oblique drop. The maximum horizontal slap down velocity at the top plate corresponds to a primary 50 degree oblique drop. In these limiting cases, the package slaps down onto the unyielding, horizontal surface with a linearly distributed velocity in both vertical and horizontal directions. The damage caused by the secondary slap down drop is the most critical among all drop positions due to a higher vertical impact velocity combined with a horizontal impact velocity at the top plate.

2.7.1.4.1 Maximum Vertical Slap Down Velocity

The analysis shows that during the impact, the peak value of the maximum principal stress reaches the material tensile strength of 70 ksi at several locations along a 45 degree arc of the package top plate rim. Therefore, the element stiffness at these locations is removed, the interactive analysis continues, and the adjacent elements continue to take load without additional failure until the end of the impact duration.

In this analysis, the package gross structure remains intact after the impact. The maximum stud stress intensity is 97.73 ksi, which is less than the material tensile strength of 100 ksi. Figure 2-10 presents the vertical deformation of the package due to this drop.

2.7.1.4.2 Maximum Horizontal Slap Down Velocity

The analysis shows that during the impact, the peak value of the maximum principal stress reaches the material tensile strength of 70 ksi at several locations along a 30 degree arc of the package top plate rim. Therefore, the element stiffness at these locations is removed, the interactive analysis continues, and the adjacent elements continue to take load without additional failure until the end of the impact duration.

This analysis shows that the package gross structure remains intact after the impact. The maximum stud stress intensity is 96.37 ksi, which is less than the material tensile strength of 100 ksi. Figure 2-11 presents the horizontal deformation of the package due to this drop.

2.7.2 Crush

10 CFR 71.73(c)(2) requires a crushing assessment for packages with a mass of not greater than 1,100 lbs. The package weighs far in excess of this value. Therefore, a crush test is not required.

2.7.3 Puncture

2.7.3.1 General Puncture Conditions

The puncture event specified in 10 CFR 71.73(c)(3) involves dropping the package from 40" onto the upper end of a solid, vertical, cylindrical, 8" long (minimum) mild steel bar with a diameter of 6" that is mounted on an essentially unyielding, horizontal surface. This drop must occur in a position for which maximum damage is expected.

For a conservative determination of potential radiological consequences, as discussed in Section 4.3.1, it is assumed that the impact results in a 6" diameter puncture hole. In consideration of cumulative effects required by 10 CFR 71.73(a), the analyses for the puncture drop are performed with the package modeled with the maximum damage resulting from the 30' drop. This maximum damage to the package extends along a 45 degree arc of the top plate rim resulting from the oblique drop as discussed in Section 2.7.1.4. For conservatism, damage along a 90 degree arc is modeled in the puncture analyses. This is achieved by removing the elements along a 90 degree arc of the top plate rim prior to initializing the puncture drop scenarios. The greatest effect on the existing damage on the package is expected to occur from a horizontal package drop onto the 6" diameter steel bar near the damaged location at the corner of the top plate.

2.7.3.2 Package Puncture Stress Analysis

The package shell is analyzed for a 40" puncture drop onto a 6" diameter steel rod as discussed in Section 2.7.3.1. The rod is assumed, conservatively, to be rigid and to remain in the vertical position throughout the puncture analyses. The two following puncture locations are considered to be the most critical: 1) near the mid-shell, between the top plate and the internal shield plate, and 2) on the top plate next to the damaged section of the rim.

2.7.3.2.1 Puncture Drop Near Mid-Shell

Contact between the package shell and the unyielding, horizontal surface is modeled along with the contact between the RV flange and the package shell. After impact, the rims of the top and bottom plates contact the ground, and the mid-shell has deformed at the puncture location of the

steel rod. The maximum principal stresses anywhere in the model, except the puncture location, remain within the material tensile strength of 70 ksi. Therefore, the 40" puncture drop on the steel rod at the mid-shell does not increase the existing damage resulting from the worst case 30' drop scenario.

2.7.3.2.2 Puncture Drop Near Top Shell

Contact between the package shell and the unyielding, horizontal surface is modeled along with the contact between the RV flange and the package shell. After impact, the RV flange contacts the package shell at the damaged rim. The top shell has deformed at the puncture location of the steel rod, but the maximum principal stress in the undamaged elements along the boundary of the damaged arc remains within the material tensile strength of 70 ksi. Therefore, the 40" puncture drop on the steel rod at the top shell does not increase the existing damage resulting from the worst case 30' drop scenario.

Figure 2-12 presents a typical stress distribution due to the puncture drop.

2.7.4 Thermal Incident

Thermal response of the package when exposed to the HAC fire for 30 minutes (10 CFR 71.73(c)(4)) is evaluated in Chapter 3. The resulting temperature distribution of the package is then used for the structural thermal analyses. The thermal structural analyses are performed with two initial ambient conditions: 100°F and –20°F. Figure 2-13 shows the temperature distribution in the package after the 30 minute HAC fire accident.

During the 30 minute HAC fire, differential thermal expansion occurs between the package and the RV in both the longitudinal and radial directions. Since the RV is enclosed inside the package, and the package expansion is greater than the RV's, there is no binding due to thermal expansion. The differential thermal expansion, however, may cause a gap between the base of the RV and the internal donut support. In this case, the RV would become a cantilevered beam supported only by the package top plate through the RV attachment studs. Although the LDCC in the annulus between the package and the RV could provide compressive support during this scenario, the effect of the LDCC is conservatively neglected in the structural analysis. This ensures that the RV attachment studs and the package top plate can adequately support the weight of the RV during and after the 30-minute thermal exposure.

To evaluate the performance of the package due to the fire accident, the impact of the 30' drop and the puncture on the package is considered in order to include the cumulative effects. Since it was concluded that the puncture scenarios do not increase the damage resulting from the worst case 30' drop scenario, the damage included in the fire accident analysis model is also along a 90 degree arc on the top plate rim as described in Section 2.7.3.1.

A nonlinear weight analysis is performed without support of the RV hemispherical bottom at the donut plate. The analysis includes the maximum HAC pressure of 95 psi applied with the two initial fire accident conditions, using the thermal material properties provided in Appendix 2-1. To model the damage to the package discussed earlier, the elements along the 90 degree arc of the top plate rim are removed from the model prior to initializing the fire analysis.

The analysis shows that the maximum stress intensity in the package is 31.5 ksi. This occurs at a location near the damaged top plate rim as the temperature rises to about 450° F. This stress intensity is less than the material tensile strength of 70 ksi corresponding to temperatures below 700° F. The maximum stress intensity at the damaged top plate rim is then reduced to 18 ksi as it reaches its highest temperature of 1400° F. This stress intensity is less than the material tensile strength of 23.46 ksi at the corresponding temperature (Table 2-6).

This analysis also shows that the stud stress intensity reaches a maximum of 40 ksi as the temperature rises to 950° F. The stud tensile strength at 950° F is greater than 52.42 ksi which corresponds to a higher temperature of 1200° F as shown in Table 2-6. The maximum stud stress of 40 ksi is therefore less than the tensile strength. As the analysis continues, the load is redistributed to the other studs. At the highest stud temperature of 1250° F, the stress intensity in all the studs is 31.7 ksi, which is less than the stud tensile strength of 46.25 ksi at 1250° F.

Based on the above discussion of results, the stresses due to the HAC fire accident do not cause any additional damage to the package. Therefore, the worst case cumulative effect of the HAC accidents is the same as the worst case effect resulting from the HAC 30' drop.

2.7.5 Water Immersion

The package is subjected to an external pressure equivalent to immersion under 50' of water (21.7 psig) per 10 CFR 71.73(c)(6). This analysis is performed on an undamaged package as allowed by 10 CFR 71.73(a). The package buckling effects are checked by combining the minimum internal pressure of –3 psig (resulting from the initial temperature condition of -20°F) with the above immersion pressure to maximize the total pressure effect. Therefore, an equivalent external pressure of 21.7 + 3 = 24.7 psig is used for water immersion analysis.

The maximum allowable external pressure is determined to be 385 psi as calculated in Appendix 2-1 in accordance with the criteria of ASME BPV, Section III, Article NB-3133.3. Water immersion external pressure is 24.7 psig which is less than the allowable external pressure. Therefore, the package meets the 10 CFR 71.73(c)(6) requirement for water immersion condition with a safety margin of 15.

2.7.6 Weld and Stud Evaluations

The methodology for weld evaluation is described in Section 2.6.12, and the results are summarized as follows.

2.7.6.1 Package Welds

Stresses in the package full penetration welds are the same as the stresses in the base metal at the weld junctions. Therefore, no separate calculation is performed for these welds. As discussed in the HAC analyses earlier, the package experiences a local damage along a 45 degree arc of the top plate rim. This indicates that the weld at that location may fail. However, the effect of this damage on the performance of the package is within the acceptance criteria of 10 CFR 71 discussed in Section 2.1.2.2.

2.7.6.2 Internal Shield Plate and Ring Plate Welds

The 30' vertical drop causes the highest load on the internal shield plate welds along the package longitudinal axis because of the drop direction and the impact acceleration. This load causes a shear stress of 16.7 ksi in the shield plate weld and 10.5 ksi in the ring plate weld. These weld stresses are less than the shear stress allowable of 31.5 ksi. Therefore, the internal shield plate and the ring plate remain intact under the HAC.

2.7.6.3 Donut Support Welds

Stresses in the double fillet weld of the donut support are the same as the stresses in the base metal at the weld junctions since the double fillet weld is as strong as a full penetration weld as discussed in Section 2.1.2.2. Therefore no separate weld stress calculation is required. The HAC analyses discussed earlier show no failure of the package shell plate at the donut support location. Therefore, no weld failure at this location will occur.

2.7.6.4 RV Studs at Top Plate

The studs between the RV flange and the package top plate are modeled in the computer analysis as described in Section 2.7. Stud stresses are addressed for each accident case in the preceding sections.

2.7.7 Conclusion of HAC Evaluations

The limiting load case among the 30' drop HAC analyses is the secondary slap down drop resulting from a 10 degree oblique corner drop. The nonlinear dynamic impact analysis results show a potential local damage that extends over a 45 degree arc along the package top plate rim. This failure could be either at the weld line or on the top plate close to the weld. Local damage is also expected for the other slap down drops, especially the 50 degree oblique drop which has the maximum horizontal velocity. This slap down drop causes local damage along a 30 degree arc of the top plate rim.

No local failure is predicted in the vertical and horizontal drops. In the corner drop through the package CG, a circumferential crack may initiate at the outer fiber of the package shell near the

junction of the donut support plate but breach of containment will not occur. In all 30' drops, the package sustains no structural failure that exceeds the damage limits set forth in Section 2.1.2.2.

The fillet welds of the internal shield plate are well within the weld shear stress allowable. Stud stress intensity remains less than the ASME BPV minimum specified material tensile strength. No stud failure is predicted.

Based on the results of the 30' drop analyses, it is concluded that the maximum damage would not extend more than 1/8 of the top plate circumference (45 degree arc). Thus, the damage limits set forth in Section 2.1.2.2 are met. This demonstrates compliance with the radiation and radioactivity release limits under the HAC requirements per 10 CFR 71.51(a)(2).

In the subsequent HAC analyses, failure in a quarter of the top plate circumference is assumed conservatively prior to the puncture and fire accident analyses. The puncture drop onto a 6" diameter steel rod results in local deformation (or a hole in worst case) on the package shell but does not extend the existing package damage due to the 30' drop. The fire accident analysis demonstrates that the damaged package remains a stable structure under the weight load and HAC fire accident pressure at the elevated temperature. The package also meets the 10 CFR 71 criteria under the HAC water immersion.

The results of the HAC analyses performed in accordance with the criteria of 10 CFR 71.73 summarized above indicate that the potential damages to the package are acceptable per the criteria set forth in Section 2.1.2.2. The post-HAC radiological consequences due to the package boundary damages predicted in Section 2.7 are bounded by the evaluation discussed in Section 4.3.1, and are therefore in compliance with the acceptance criteria of 10 CFR 71.51(a)(2).

2.8 FRACTURE TOUGHNESS CONSIDERATIONS

For compliance with the 10 CFR 71 Lowest Service Temperature (LST) of –20°F during the package transport, the provisions of Regulatory Guide 7.11 and NUREG/CR-1815 are used in the measurement and determination of the NDT temperature for the package plate material. The BRP package is a Type B, Category II package. Therefore, the required Nil Ductility Transition (NDT) temperature is determined by using a value of β=0.6 in accordance with the methodology provided in Section 5.2 of NUREG/CR-1815.

For a package shell thickness of 3", SA-516 Grade 70 carbon steel plate with a yield strength less than 60 ksi, the NDT temperature requirement determined from Section 5.2.1, Figure 6 of NUREG/CR-1815 is:

NDT temperature = LST - A°F = -20°F - 0°F = -20°F.

For the top and bottom plates, fabricated from 4", SA-516 Grade 70 carbon steel plate with a yield strength less than 60 ksi, the shifted temperature, A°F, from Figure 6 of NUREG/CR-1815 is 15°F. Therefore, the required NDT temperature is:

NDT temperature = LST - A°F = -20°F - 15°F = -35°F.

Therefore, to protect the package material from brittle fracture under the 10 CFR 71 LST of -20°F, the required NDT temperature for the package material is -35°F. However, the lowest NDT temperature achievable in manufacturing SA-516, Grade 70 steel is -20°F. To ensure the package material remains ductile during transportation, the LST must be limited to:

LST = NDT + A°F = -20°F +15°F = -5°F

This means that the package must not be subjected to dynamic loads and transportation must be stopped once the ambient temperature falls to -5°F. After the ambient temperature increases to -5°F or above, the package surface temperature must be measured. If the package temperature is also -5°F or above, transportation can be resumed. Otherwise, sufficient time must be allowed for the package surface temperature to warm up to at least -5°F before transportation can be continued.

However, in order to provide a margin to account for instrument accuracy and/or error in reading the instrument, the lowest ambient temperature for stopping the transportation and the minimum package surface temperature for resuming the transportation are set at 0°F (zero). The package surface temperature measurement is required on the top and bottom plates only. Compliance with the minimum temperature requirement is ensured through operating controls imposed on the transportation activities as stated in Section 7.3.1.

2.9 REFERENCES

2.9.1 Regulations:

2.9.1-1 United States Nuclear Regulatory Commission Rules and Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Materials," March 31, 1999.

2.9.1-2 United States Nuclear Regulatory Commission Rules and Regulations, Title 10, Part 20, "Standards for Protection Against Radiation," October 29, 1999.

2.9.1-3 Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," Rev. 1, March 1978.

2.9.1-4 Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," Rev. 1, March 1989.

2.9.1-5 Regulatory Guide 7.11, "Fracture Toughness Criteria of Basic Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches," Rev. 0, June 1991.

2.9.1-6 IE Information Notice 84-72,"Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation," USNRC, September 10, 1984.

2.9.2 Codes and Standards:

2.9.2-1 ASME Boiler and Pressure Vessel Code, Section II, Part D, "Properties", and Section III, Subsection NA, NB, NF and Appendices, 1995 Edition with the 1996 Addenda as modified by 10 CFR 50.55a(b)(1).
2.9.2-2 ASME Boiler and Pressure Vessel Code, Section VIII, 1998 Edition.
2.9.2-3 ANSI/AWS D1.1-98, "Structural Welding Code-Steel," 1998.
2.9.2-4 ANSI N14.5, "Leakage Tests on Packages for Shipment," 1997.
2.9.2-5 Association of American Railroads (AAR), "Open Top Loading Rules Manual," Part 1, dated March 1, 2000.
2.9.2-6 ANSI N14.2, "Tie-down for Truck Transport of Radioactive Materials," Draft.

2.9.3 Computer Codes:

2.9.3-1 Finite Element Program ANSYS Version 5.6, Sargent and Lundy Validated Program Number ANSYS 03.7.596-5.62.
2.9.3-2 HEATING 7.2f, "Multidimensional, Finite-Difference Heat Conduction Analysis," Sargent & Lundy Program No. 03.7.564-7.2.
2.9.3-3 APLAN, Sargent and Lundy Validated Program No. 03.7.282-1.10.
2.9.3-4 STAAD-III, Sargent & Lundy Validated Program No. 03.7.065-2.23.

2.9.4 Publications:

2.9.4-1 NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick," August 1981.
2.9.4-2 NUREG/CR-3854, "Fabrication Criteria Shipping Containers," March 1985.
2.9.4-3 NUREG/CR-3966, "Methods for Impact Analysis of Shipping Containers," November 1987.
2.9.4-4 NUREG/CR-3019, "Recommended Welding Criteria For used in the Fabrication of Shipping Containers for Radioactive Materials," March 1985.
2.9.4-5 NUREG/CR-0128, "Shock and Vibration Environments For A Large Shipping Container During Truck Transport (Part II)," May, 1978.
2.9.4-6 ASCE Manual, "Structural Analysis and Design of Nuclear Plant Facilities," 1980 Edition.
2.9.4-7 Roark J. R. and Young W.C., "Formulas for Stress and Strain," 5th Edition, McGraw-Hill Book Company.
2.9.4-8 ASCE Manual and Reports on Engineering Practice No. 78, "Structural Fire Protection," American Society of Civil Engineers.

Table 2-1
Load Cases and Combinations with Initial Conditions

Conditions		Load Case No.	Applicable Initial Conditions								
			Ambient Temp.		Insolation		Decay Heat		Internal Pressure		Fabrication Stress
			100° F	-20°F	Max	Zero	Max	Zero	Max	Min	
NCT	Hot 100°F ambient temp.	N1	√		√		√		√		√
	Cold -40°F ambient temp.	N2				√		√		√	√
	Increased 20 psia external pressure	N3		√		√		√		√	√
	3.5 psia external pressure	N4	√		√		√		√		√
	Shock normally incident to transport in the vertical, transverse and longitudinal directions	N5	√		√		√		√		√
		N6		√		√		√		√	√
		N7	√		√		√		√		√
		N8		√		√		√		√	√
		N9	√		√		√		√		√
		N10		√		√		√		√	√
	1' drop	N11	√		√		√		√		√
		N12		√		√		√		√	√
	Steel cylinder drop onto package (See Note below.)	Hand Calc.	√		√		√		√		√
				√		√		√		√	√
HAC	Horizontal free drop	HD1	√		√		√		√		√
		HD2		√		√		√		√	√
	Vertical free drop	VD1	√		√		√		√		√
		VD2		√		√		√		√	√
	Corner free drop	CD1	√		√		√		√		√
		CD2		√		√		√		√	√
	Slap down drop	FD11, FD21	√		√		√		√		√
		FD12, FD22		√		√		√		√	√
	Puncture drop onto steel bar	PTD1,PMD1	√		√		√		√		√
		PTD2,PMD2		√		√		√		√	√
	Thermal fire accident	THACC1	√		√		√		√		√
		THACC2		√		√		√		√	√

Note: ANSYS analyses results for the load cases in this table are presented in Tables 2-8 through 2-10 except for the NCT steel cylinder drop. This load condition is evaluated by a different methodology described in Section 2.6.10.

Table 2-2
Package Component Weights

Package Components	**Weight (kips)**
Core Plate	2.54
Inlet Baffle	2.32
Inlet Diffuser	0.20
Steam Baffle	1.64
Sparger	0.13
Seal Housing	1.07
Seal Weights	5.08
Thermal Shield	13.00
Thermal Shield Retainers	0.55
Top Guide Plate	0.82
RV Shell and Nozzles	183.67
Remaining RV Insulation	3.30
30-36* lb/ft^3 LDCC	46.70*
Donut Support	16.40
Package Shell	122.45
Internal Shield Plate	49.93
Trunnion Reinforcing Ring Plate	15.73
Top Plate	19.75
Bottom Plate	20.04
14 RV Attachment Studs	1.50
50-60* lb/ft^3 LDCC	57.33*
Total	**565**

* The upper bound LDCC weights are used in the structural evaluations in order to obtain bounding results.

Table 2-3
Temperature Dependent Modulus of Elasticity and Design Stress Intensity
(Ref. 2.9.2.1)

Temperature °F	Modulus of Elasticity, E X10^6 (psi)		Design Stress Intensity, Sm X10^3 (psi)	
	SA-516 Gr. 70 C-Mn-Si	SA-302 Gr. B Mn-½Mo	SA-516 Gr. 70 C-Mn-Si	SA-302 Gr. B Mn-½Mo
-100	30.2	29.9	--	--
-20	--	--	23.3	26.7
70	29.5	29.2	23.3	26.7
100	29.34*	29.0*	23.3	26.7
200	28.8	28.5	23.1	26.7
300	28.3	28.0	22.5	26.7
400	27.7	27.4	21.7	26.7
500	27.3	27.0	20.5	26.7
600	26.7	26.4	18.7	26.7
650	--	--	18.4	26.7
700	25.5	25.3	18.3	26.7
800	24.2	23.9	--	--
900	22.4	22.2	--	--

* Interpolated

Table 2-4
Temperature Dependent Tensile and Yield Strengths
(Ref. 2.9.2.1)

Temperature °F	Tensile Strength, Su x10³ (psi)		Yield Strength, Sy x10³ (psi)	
	SA-516 Gr. 70 C-Mn-Si	SA-302 Gr. B Mn-½Mo	SA-516 Gr. 70 C-Mn-Si	SA-302 Gr. B Mn-½Mo
70	70.0	80.0	38.0	50.0
100	70.0	80.0	38.0	50.0
150	--	--	35.7	48.6
200	70.0	80.0	34.6	47.2
250	--	--	34.2	46.2
300	70.0	80.0	33.7	45.3
400	70.0	80.0	32.6	44.5
500	70.0	80.0	31.7	43.2
600	70.0	80.0	28.1	42.0
650	70.0	80.0	27.6	41.4
700	70.0	80.0	27.4	40.6
750	69.3	80.0	26.5	40.0
800	64.3	80.0	25.3	38.8
850	58.6	76.6	24.4	37.2
900	52.0	72.7	24.1	34.9
950	46.2	67.3	23.2	31.9
1000	40.3	62.2	21.1	28.4

Table 2-5
Temperature Dependent Design Stress Intensity and
Yield Strength for Bolting Material
(Ref. 2.9.2.1)

Temperature ^{o}F	**Design Stress Intensity, Sm $x10^3$ (psi)**	**Yield Strength, Sy $x10^3$ (psi)**
100	25.0	75.0
200	23.3	69.9
300	22.4	67.2
400	21.8	65.4
500	21.0	63.2
600	20.3	60.9
650	19.7	59.2
700	19.2	57.5
750	18.5	55.4
800	17.5	52.7

Table 2-6
Yield Strength, Sy, and
Tensile Strength, Su,
At Elevated Temperatures
(See Section 2.3 for reference.)

Temperature °F	SA-516 Gr. 70 C-Mn-Si		SA-302 Gr. B Mn-½Mo		SA-193 B7 (1Cr-1/5Mo)	
	Sy (ksi)	Su (ksi)	Sy (ksi)	Su (ksi)	Sy (ksi)	Su (ksi)
1200	19.529	35.974	25.696	41.113	38.544	52.419
1250	17.232	31.744	22.674	36.278	34.011	46.255
1300	15.418	28.402	20.287	32.459	30.430	41.385
1350	13.949	25.695	18.354	29.366	27.531	37.442
1400	12.735	23.459	16.756	26.810	25.134	34.183
1450	11.715	21.579	15.414	24.662	23.121	31.444

Table 2-7
Temperature Dependent Material Tangent Modulus
(See Section 2.3 for reference.)

Temperature, °F	SA-516 Gr. 70 C-Mn-Si	SA-302 Gr. B Mn-½Mo	SA-193 B7 (1Cr-1/5Mo)
-20 to 200	160E+03	166E+03	139E+03
400	140 E+03	145 E+03	122 E+03
600	121 E+03	126 E+03	106 E+03
800	97E+03	100E+03	84E+03
1000	65E+03	67E+03	56E+03
1450	49E+03	51E+03	43E+03

Table 2-8
Stress Intensity Summary for NCT

Normal Conditions of Transport	Load Case No.		Primary Membrane Stress, Pm			Primary Membrane plus Bending Stress, Pm+Pb			Primary Plus Secondary Stress, Pm+Pb+Q			
			Node No.	ksi	Sm ksi	Node No.	ksi	1.5 Sm	Node No.	ksi	3m Sm	Min. Stress Margin
Hot 100°F ambient temp.	N1		10304	1.85	23.1	966	5.41	34.65	968	7.29	69.3	6.4
Cold -40°F ambient temp.	N2		10304	2.0	23.1	3082	6.52	34.65	3082	6.4	69.3	5.4
Increased 20 psia external pressure	N3		10304	1.81	23.1	3082	6.20	34.65	3082	6.11	69.3	5.6
Minimum 3.5 psia external pressure	N4		3802	2.77	23.1	966	7.51	34.65	968	9.70	69.3	4.6
Vertical Shock	Hot	N5	10304	4.62	23.1	3082	9.82	34.65	3082	9.81	69.3	3.5
	Cold	N6	10304	4.11	23.1	3082	9.36	34.65	3082	9.28	69.3	3.7
Lateral Shock	Hot	N7	3838	4.84	23.1	3302	18.3	34.65	3302	18.3	69.3	1.9
	Cold	N8	3838	5.20	23.1	3302	17.6	34.65	3302	17.5	69.3	2.0
Long. Shock	Hot	N9	10304	9.93	23.1	3082	14.9	34.65	1085	16.0	69.3	2.3
	Cold	N10	10304	9.96	23.1	3082	15.5	34.65	3082	15.4	69.3	2.2
1' Drop	Hot	N11	3853	19.3	23.1	3853	19.7	34.65	3853	22.4	69.3	1.2
	Cold	N12	3853	19.2	23.1	3853	19.7	34.65	3853	18.5	69.3	1.2
Test Pressure	Hot	TP1	3802	2.67	23.1	966	7.27	34.65	968	9.44	69.3	4.8
	Cold	TP2	3802	2.67	23.1	966	7.27	34.65	10446	6.97	69.3	4.8

Table 2-9
Stress Intensity Range for NCT Vibration

Load Range (I-J)	**Load Case I**	**Load Case J**	**Stress Range $2S_{alt}$ (ksi)**	**Node No.(*)**
F1	Longitudinal shock, hot, Pmax, (N9)	Inc. ext. pressure, cold, (N3)	13.5	942
F2	Longitudinal shock, cold, Pmin (N10)	Min. ext. pressure, hot, (N4)	12.3	3082
F3	Longitudinal shock, hot, Pmax, (N9)	Vert. shock, cold, Pmin, (N8)	13	942
F4	Longitudinal shock, cold, Pmin (N10)	Vert. shock, hot, Pmax, (N7)	12.6	10304
F5	Longitudinal shock, hot, Pmax, (N9)	Zero stress state	17.3	3082

(*) Saddle Support Nodes

Table 2-10
Extreme Stress Intensity Range for NCT

Load Range (I-J)	Load Case I	Load Case J	2 x Alternating Stress $2S_{alt}$ (ksi)	Half Peak Stress Intensity (*) KS_{alt} (ksi)	Allowable Stress, Sa (ksi)	Stress Margin
F6	Horizontal drop, hot, Pmax, (N11)	Cold –40°F, (N2)	121.9	243.80	580	2.38
F7	Horizontal drop, cold, Pmin (N12)	Hot 100°F, (N1)	119.14	238.28	580	2.43
F8	Horizontal drop, hot, Pmax, (N11)	Longitudinal shock, cold, (N10)	114.59	229.18	580	2.53
F9	Horizontal drop, hot, Pmin (N11)	Zero stress state	123.24	246.48	580	2.35

(*) Conservatively includes a peak stress index, K = 4.

FIGURE WITHHELD UNDER 10 CFR 2.390

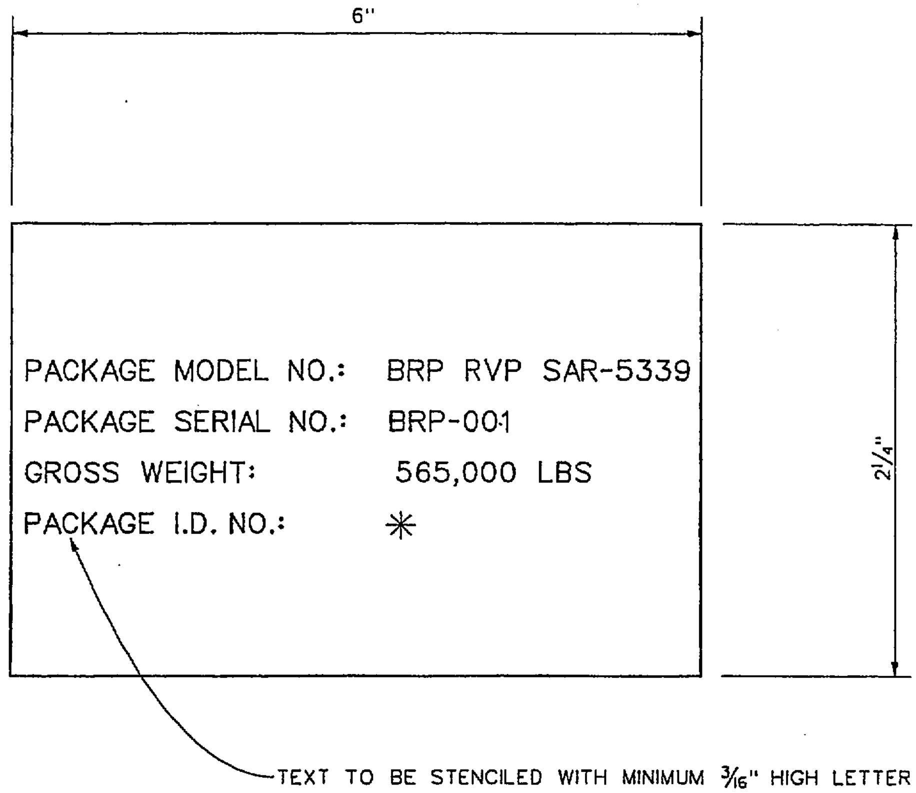
FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390



✳ INDICATES THE PACKAGE I.D. NO. WILL BE ASSIGNED BY NRC.

NAME PLATE
PACKAGE CONFIGURATION AND DIMENSIONS

Figure 2-1, Sheet 7 of 7

BNFL Inc.
Big Rock Point
10269 US 31 North
Charlevoix, MI
49720-9436

Sargent & Lundy

FIGURE RELEASE RECORD

REV	DATE	PREPARED	REVIEWED	APPROVED	PURPOSE
0	9/15/01				INITIAL ISSUE

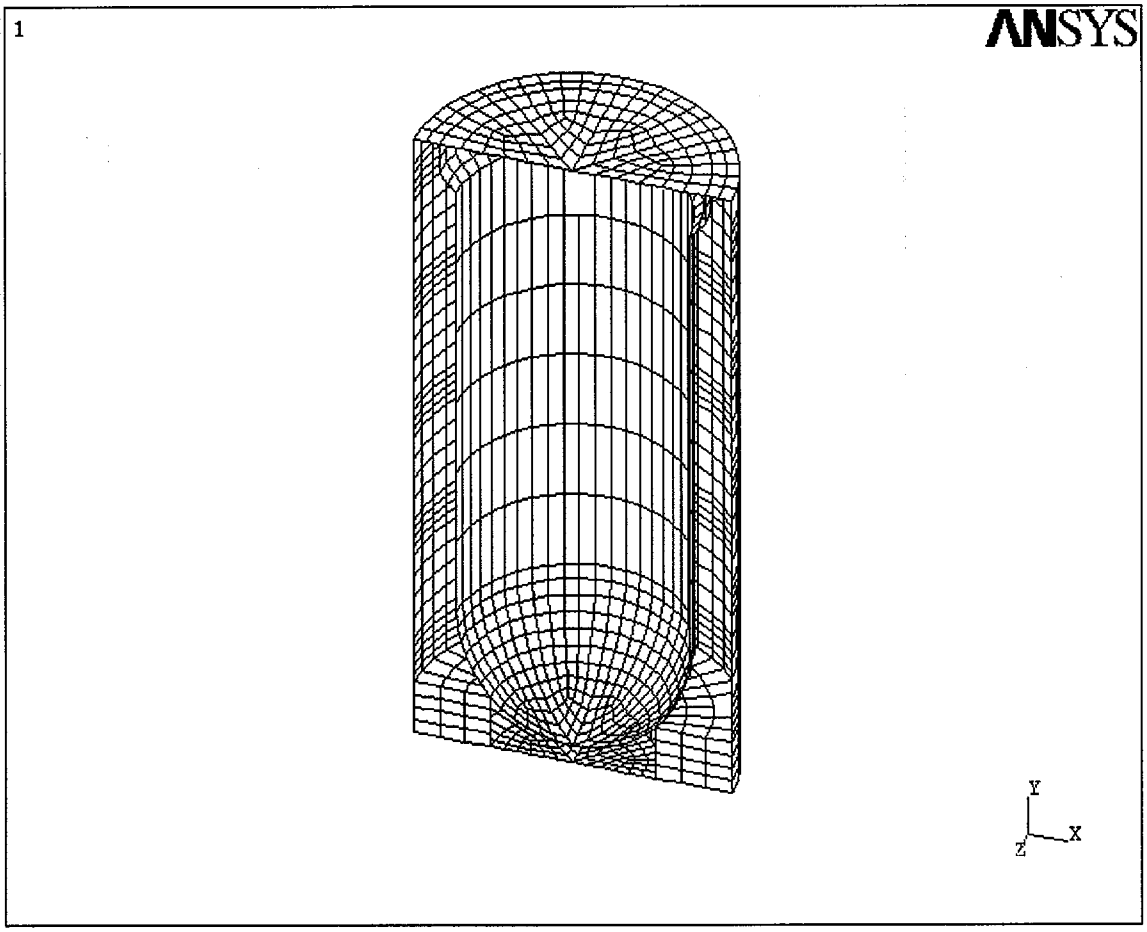


Figure 2-2: Package FEA Model in Vertical Position

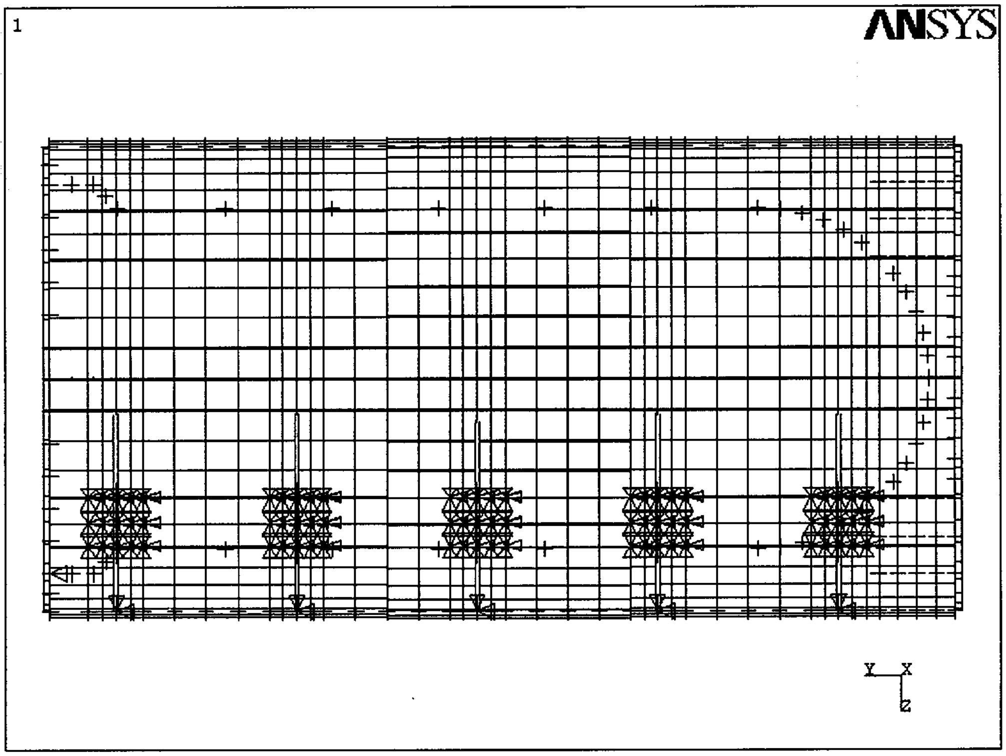


Figure 2-3: Package FEA Model in Horizontal Position with Tie-down System

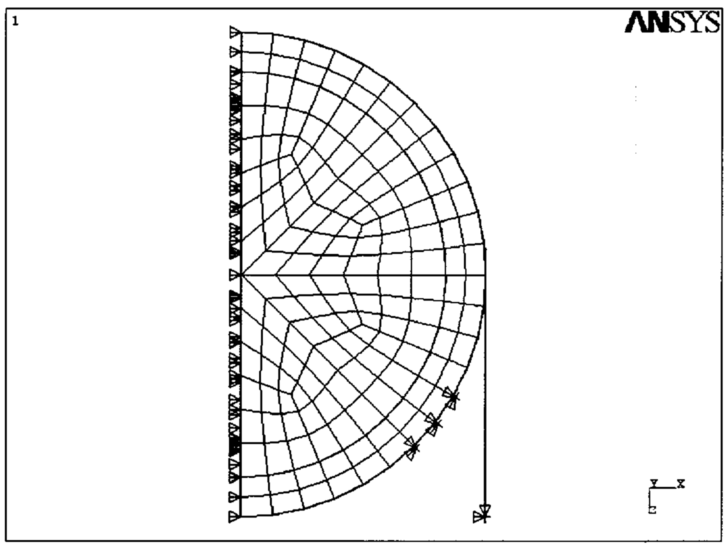


Figure 2-4: Package FEA Side View with Tie-down System

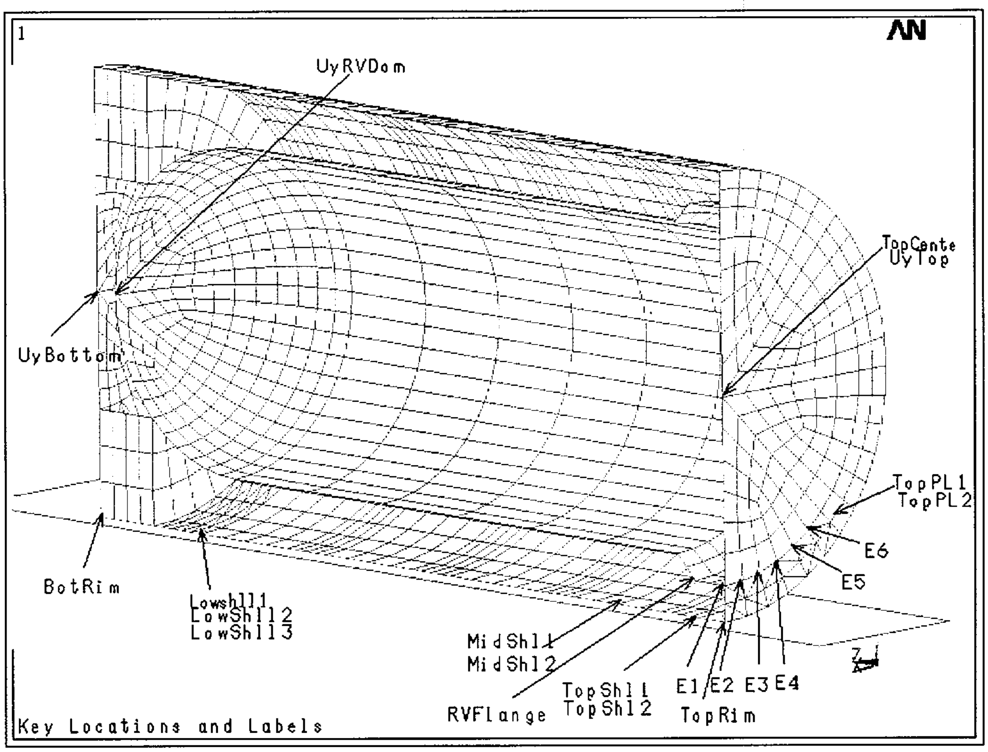


Figure 2-5: Key Locations of Package Model for HAC Analyses

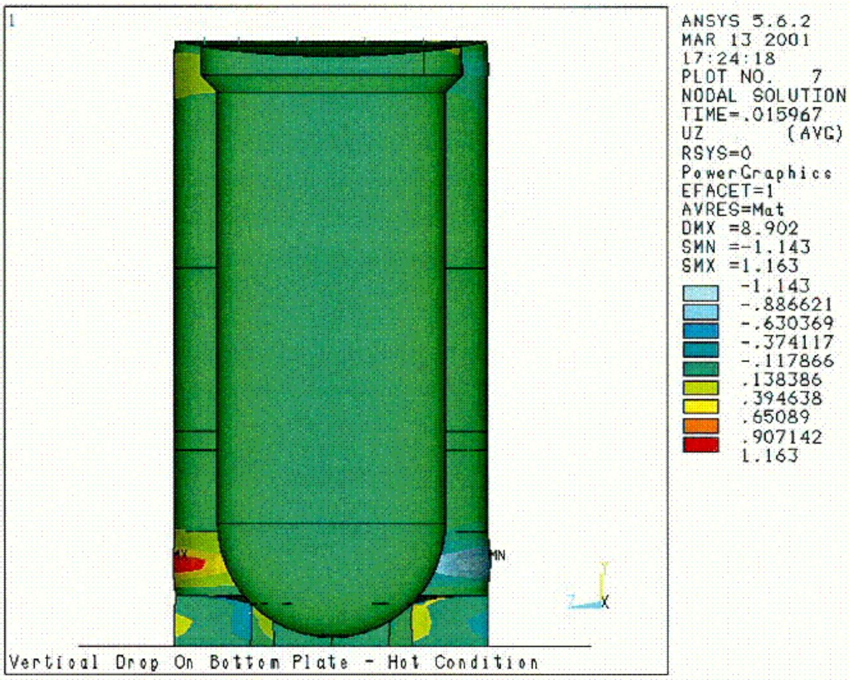


Figure 2-6: Vertical Drop Horizontal Deformation
(Unit: inches)

C-1

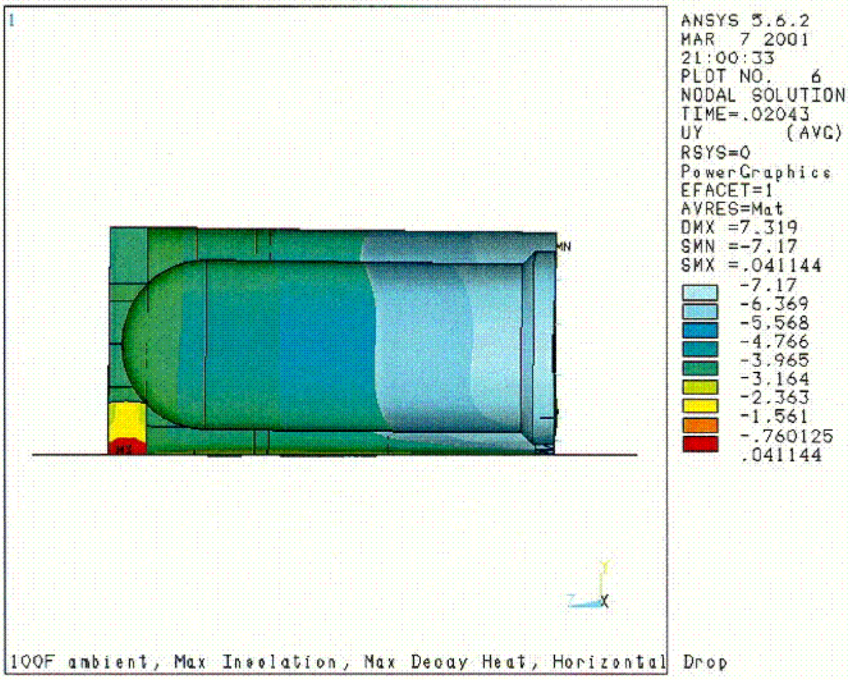


Figure 2-7: Horizontal Drop Vertical Deformation
(Unit: inches)

C-2

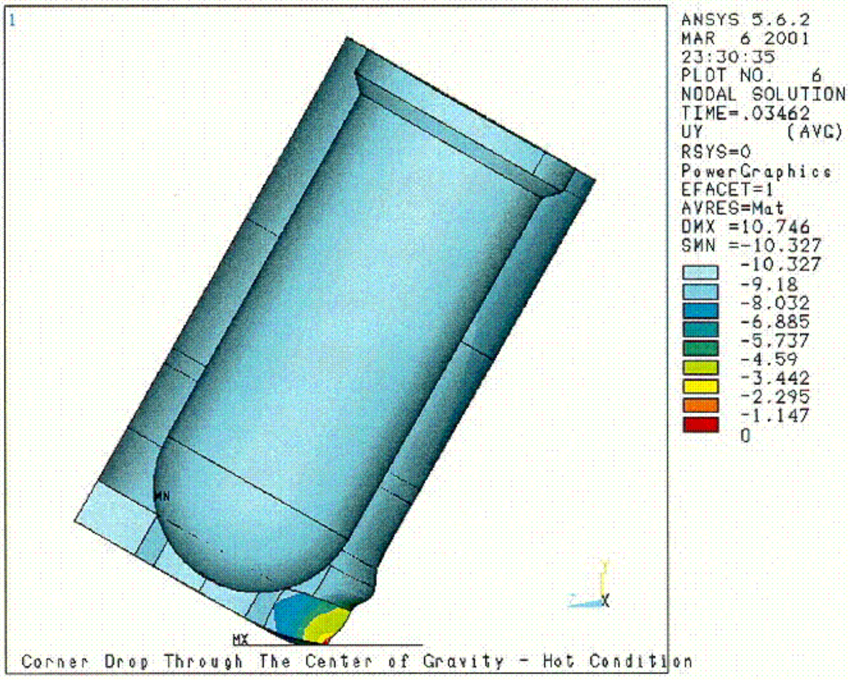


Figure 2-8: Corner Drop Vertical Deformation
(Unit: inches)

C-3

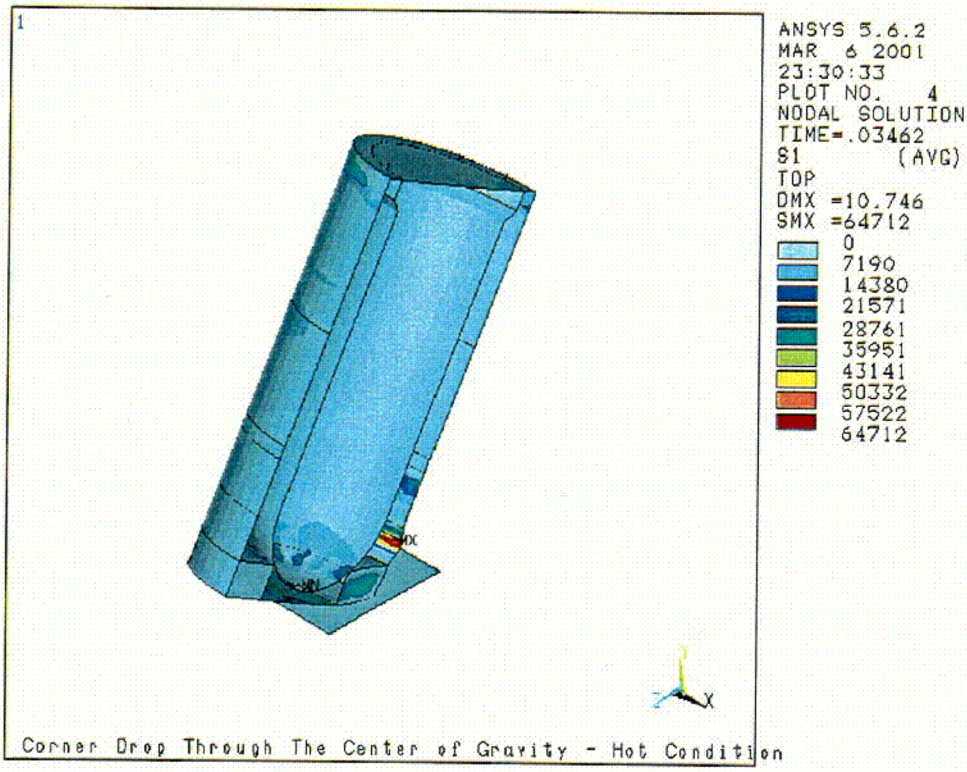


Figure 2-9: Corner Drop High Stress Location
(Unit: psi)

C-4

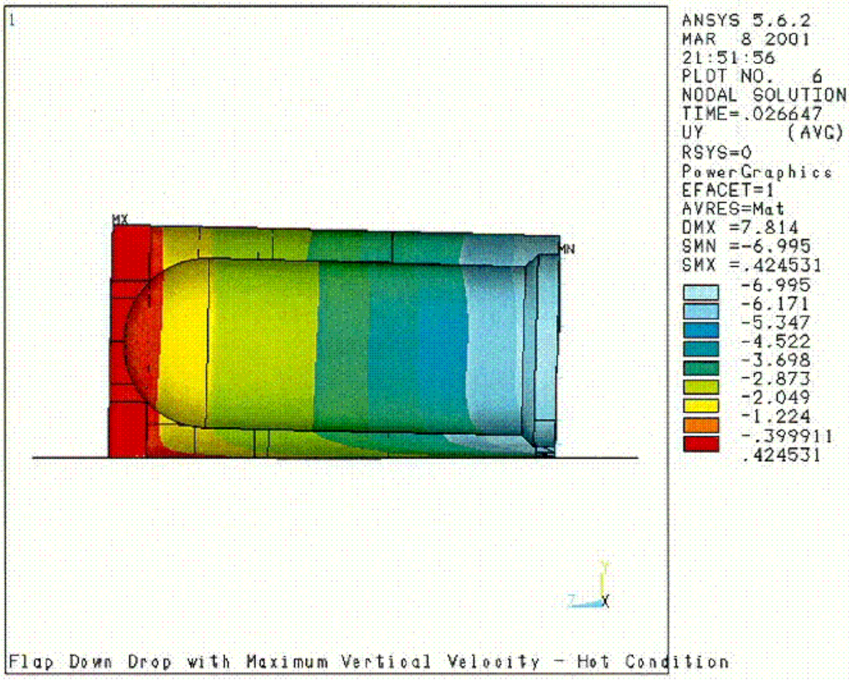


Figure2-10: Slap Down Drop Vertical Deformation
(Unit: inches)

C.5

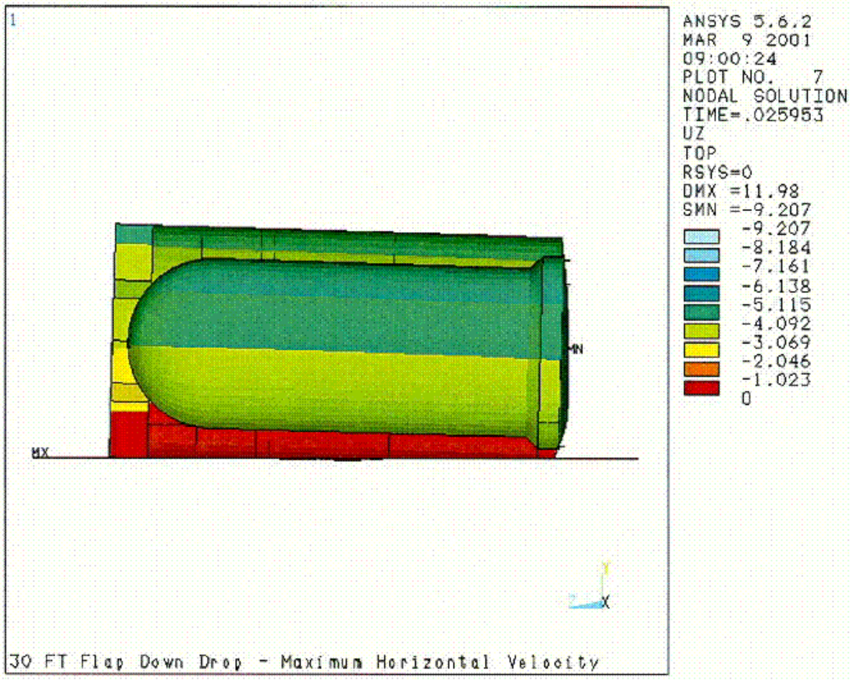


Figure 2-11: Slap Down Drop Horizontal Deformation
(Unit: inches)

C-6

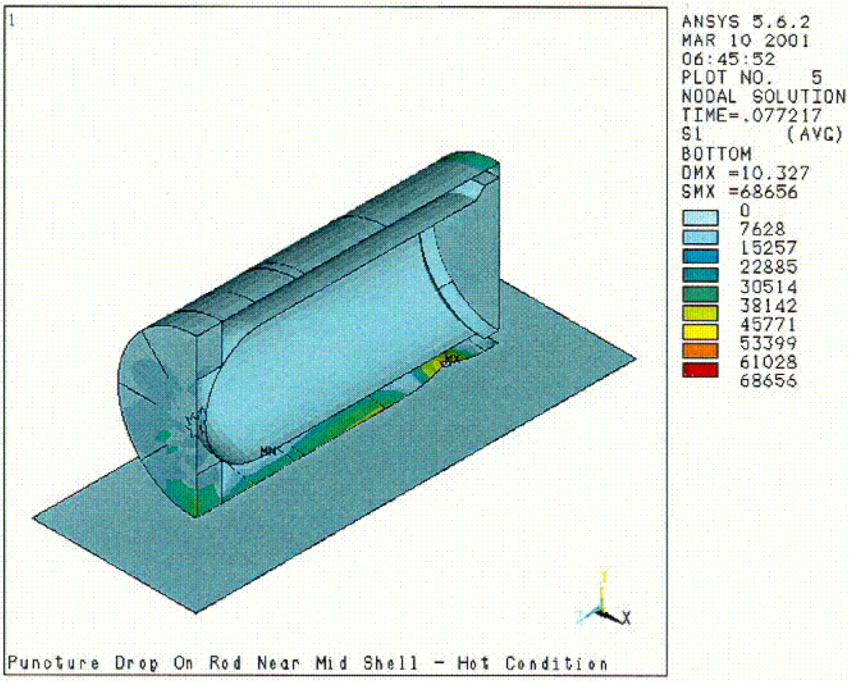


Figure 2-12: Typical Puncture Drop Stress Distribution
(Unit: psi)

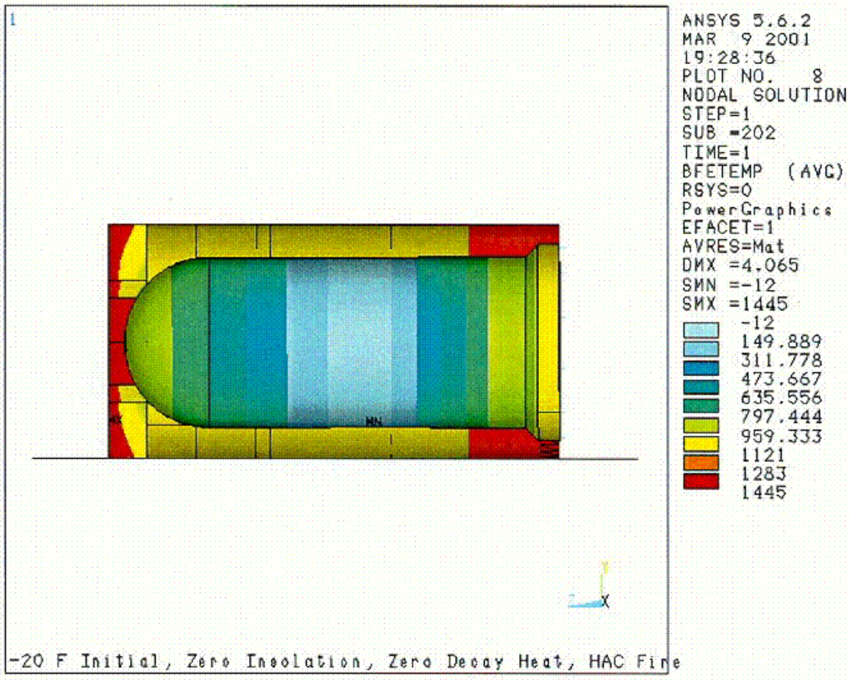


Figure 2-13: HAC Fire Temperature Distribution 30 Minutes after Event Initiation (Unit: °F)

C-8

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3.0 THERMAL EVALUATION

This chapter evaluates thermal performance of the package design under the thermal conditions specified in 10 CFR 71.71 for Normal Conditions of Transport (NCT) and 10 CFR 71.73 for Hypothetical Accident Conditions (HAC). The objective of this evaluation is to demonstrate that the design of the package is such that:

- There will be no loss of radioactive contents, no significant increase in external surface radiation levels, and no significant decrease in package effectiveness under the NCT test conditions (10 CFR 71.71) as stated in 10 CFR 71.43(f) and 71.51(a)(1);
- Its accessible surface temperature in still air at 100°F and in the shade does not exceed the exclusive use shipment limit of 185°F as specified in 10 CFR 71.43(g);
- Its performance under the HAC thermal test conditions (10 CFR 71.73) will not result in exceeding the activity release limits of 10 CFR 71.51(a)(2).

Section 3.1.2 states how this objective is met. Detailed analyses supporting the discussions in this chapter are provided in Appendices 3-1 through 3-3.

3.1 DISCUSSION AND RESULTS

3.1.1 Design Features

The physical characteristics of the package contributing to its thermal performance are described below. Additional details on these components are presented in Chapter 1, Section 1.2.

- The pressure-containing boundary is the cylindrical package[1] including its top and bottom plates, and welded closures that seal the penetration openings. The dimples in the top plate described in Section 2.1.1 are not included in the thermal model. These dimples have negligible effect on the thermal evaluations presented herein.

- The package contains the Reactor Vessel (RV), some remaining internals, and a portion of its insulation. The void space in the RV is filled with Low Density Cellular Concrete (LDCC) with a density range of 30-36 lb/ft^3. The annulus between the RV and the package shell is filled with LDCC with a density range of 50-60 lb/ft^3.

[1] The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

- The package provides shielding of the radioactive material. The nominal thickness of the package shell is 3". In addition, a 4" thick band of steel plate is welded to the interior surface of the package shell in the former reactor core region of the RV. The former reactor core region is taken to be approximately 8' in axial length, starting at a distance of approximately 8'-6" measured from the bottom of the package.

Materials of these components are shown in Table 3-1.

The above package configuration leads to the development of a three-dimensional heat transfer model. The following heat transfer path is considered in the thermal evaluation of the package under NCT and HAC:

- Normal Conditions of Transport:

    a. Decay heat produced by the internal components of the RV is transferred to the inner surface of the RV by radial and axial conduction through the 30-36 lb/ft^3 LDCC. The total decay heat of the RV assembly and internals is 485 BTU/hour as calculated in Appendix 3-1.

    b. Heat is transferred through the RV wall by radial and axial conduction, and by axial conduction to the top plate.

    c. Heat conducted through the RV wall and decay heat produced by the RV shell are transferred to the outer annular region of LDCC by radial conduction through the RV metallic insulation.

    d. Heat transferred through the insulation to the outer annular region of LDCC and decay heat produced by the RV activated metallic insulation are transferred to the inner surface of the package shell by radial conduction through the outer layer of LDCC.

    e. Heat is then transferred by conduction to the outer surface of the package shell, top plate, and bottom plate. Since the package is filled with LDCC, no significant air gaps will be present within the package. Therefore, natural convection and radiation heat transfer mechanisms within the package are negligible compared to conduction heat transfer mechanisms.

    f. Heat is transferred from the outer surface of the package to the surrounding air by natural convection and radiation.

- Hypothetical Accident Conditions:

    a. The heat of a hypothetical fire is imposed on the exterior of the entire package by both radiant heat transfer and forced convection heat transfer from a flame temperature of 1475°F for a duration of 30 minutes as specified in 10 CFR 71.73(c)(4).

b. Transient conduction conveys the heat inward from the package shell and top and bottom plate into the outer annular region of LDCC.

c. The heat from the hypothetical fire is absorbed in the outer annular region of LDCC, resulting in large temperature gradients in the outer annular LDCC region.

d. The heat is transferred through the RV wall into the inner LDCC by conduction.

e. The heat stored in the package due to the fire is then dissipated to the ambient atmosphere using the same heat transfer mechanisms as described for the NCT.

3.1.2 Thermal Acceptance Criteria

Acceptance criteria for the package design is set by the 10 CFR requirements referenced in Section 3.0. The objective of the thermal evaluation as described in that section is met through demonstration of compliance with the:

- Package surface temperature limit of 10 CFR 71.43(g), and
- Structural integrity affected by the thermal stresses under the thermal NCT and HAC.

Compliance with the above is satisfied as stated in Sections 3.1.3.1, and 3.1.3.2.

3.1.3 Analysis Results

The thermal analysis for the NCT and HAC conditions is documented in Appendix 3-2. Details of the thermal model and loading conditions considered in this analysis are described in Sections 3.4 and 3.5 of this SAR. A summary of the analysis conclusions for the maximum package temperatures is presented below.

3.1.3.1 Normal Conditions of Transport

10 CFR 71.43(g) states that the maximum accessible surface temperature of the package in still air at 38°C (100°F) and in the shade shall not exceed a temperature of 85°C (185°F) in an exclusive use shipment. Evaluation of the package thermal design (Appendix 3-2) performed for the 10 CFR 71 thermal loading conditions is described in Section 3.4.

The first loading condition accounts for internal heat generation from the decay heat calculated in Appendix 3-1. The calculated package maximum surface temperature is 103.8°F for this loading condition. This temperature satisfies the surface temperature limit of 185°F in still air and in the shade as specified in 10 CFR 71.43(g).

The second loading condition considers maximum surface temperature of the package in still air at 38°C (100°F) including the effects of insolation and internal heat generation from the decay

heat. This condition is considered for use in stress calculations in order to satisfy the requirements of 10 CFR 71.71(c)(1) in the package containment and structural integrity confirmation. A total insolation of 400 g cal/cm^2 (1475 BTU/ft^2) for a 12-hour period was specified on the cylindrical shell of the package. For the flat top and bottom plates, a total insolation of 200 g cal/cm^2 was specified. The calculated maximum surface temperature for this loading condition is 191.2°F.

The maximum temperatures of the various package components under the above NCT loading conditions are also included in Table 3-2. The package structural integrity under the NCT maximum temperatures is maintained as demonstrated in Chapter 2.

3.1.3.2 Hypothetical Accident Conditions

The maximum temperatures of the package components under the HAC thermal condition of 10 CFR 71.73(c)(4) are obtained from Appendix 3-2 and are summarized in Table 3-3. These temperatures are used in calculating the thermal stresses to confirm the containment and structural integrity of the package. The package performance under the HAC maximum temperatures is in compliance with the 10 CFR 71 criteria as discussed in Chapter 2.

3.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

The material properties used in the thermal analysis are shown in Table 3-1, as referenced in Appendix 3-2. The materials of the package and its contents included in the thermal model are SA-302 Grade B steel, SA-516 Grade 70 carbon steel, Type 304 stainless steel, stainless steel corrugated foil insulation, 30 lb/ft^3 LDCC, and 50 lb/ft^3 LDCC.

The lower bound value of thermal conductivity for the LDCC is used in the thermal analyses in order to obtain bounding thermal gradients. Use of the upper bound thermal conductivity would be conservative for the determination of the maximum surface temperature. However, the difference in results would be negligible compared to the available margin of 81.2°F (See Section 3.4.2). Therefore the use of the lower bound values is acceptable for the determination of the maximum surface temperature.

3.3 TECHNICAL SPECIFICATIONS FOR COMPONENTS

The package contains no active components such as pressure relief valves or fusible plugs. There are no special requirements for the package coating. Therefore, there are no material temperature limits or technical specifications applicable to the package.

3.4 THERMAL EVALUATION FOR NCT

As previously stated in Section 3.1.2, a thermal evaluation is performed in order to confirm compliance with the package surface temperature limit of 10 CFR 71.43(g) and to obtain the maximum and minimum temperatures for calculation of the package thermal stresses. The following sections describe the details of this evaluation (Appendix 3-2) under the NCT conditions.

3.4.1 Thermal Model

The maximum and minimum package temperatures are determined under the thermal loading conditions of 10 CFR 71.71(b), 71.71(c)(1), and 71.71(c)(2). The maximum accessible package surface temperature is evaluated using the criteria specified in 10 CFR 71.43(g).

For determination of the accessible surface temperature, the environmental condition of a 100°F ambient temperature with no solar insolation is evaluated at steady-state. To determine the performance of the package during the NCT, the package is evaluated at a minimum ambient temperature of -40°F without solar insolation and a maximum ambient temperature of 100°F with solar insolation. The initial condition of –20°F ambient temperature and no insolation is evaluated to provide the minimum temperatures for evaluation of the NCT. A total decay heat load of 485 BTU/hour is employed in the thermal model.

For the NCT, heat transfer from the package outer surface to the ambient is modeled with parallel heat transfer mechanisms of laminar natural convection and radiation. The laminar natural convection heat transfer coefficient is computed based on correlations for natural convection of horizontal cylinders. The package end plates are treated as flat vertical plates for the purposes of computing the laminar natural convection heat transfer coefficient. Radiation heat transfer is governed by the emissivity of the package outer surface, taken as $\varepsilon = 0.66$, which is representative for rolled steel. The emissivity of typical epoxy coatings that may be applied to the outer surface of the package is typically greater than the analyzed value of $\varepsilon = 0.66$. The use of lower emissivity value is conservative for the analysis of Appendix 3-2.

A total solar insolation of 400 g cal/cm^2 on the curved package surfaces, for a 12-hour period, is specified. For the flat, vertical end plates (as positioned during shipment) of the package, a total insolation of 200 g cal/cm^2 is specified per 10 CFR 71.71(c)(1).

The thermal analysis of the package is performed using the HEATING 7.2f computer code. HEATING is a finite difference thermal analysis code capable of solving steady-state and transient thermal analysis problems in one, two, or three dimensions in a rectangular, cylindrical, or spherical coordinate system. HEATING was developed by Oak Ridge National Laboratory, and the solution algorithms have been verified against a large number of representative problems presented in the literature having analytical solutions. The code is capable of modeling conduction, radiation, and natural or forced convection heat transfer mechanisms. A flexible

input structure allows the user to model a wide variety of boundary conditions, encompassing all of the postulated scenarios required per 10 CFR 71.

The three-dimensional package model consists of the following regions listed in the direction of increasing radius: RV internal compartment filled with LDCC interior filler, stainless steel cladding, RV wall, stainless steel insulation, LDCC in the annulus between the RV and the package exterior filler, and the package steel shell. The LDCC acts to stabilize the RV internal components, immobilize any potential loose radioactive contamination, and to fill any gaps or voids in the package. In the former reactor core region, the thermal shield is also modeled. The former reactor core region inside the thermal shield is modeled as a homogeneous region of 30 lb/ft^3 LDCC, neglecting the RV internal components. This conservatively lowers the thermal conductivity of the former reactor core region, maximizing internal temperatures for the analysis.

The axial dimensions of the package are modeled to encompass edge effects for the upper and lower package end plates. The RV is constructed of SA-302 Grade B steel, and the package is made of SA-516 Grade 70 carbon steel.

3.4.2 Maximum Temperatures

10 CFR 71.43(g) requires the maximum accessible surface temperature in still air at 100°F and no insolation to be less than 185°F for an exclusive use package. The maximum accessible surface temperature of the package for this ambient condition is evaluated to be 103.8°F. Therefore, the 10 CFR 71.43(g) requirement is satisfied with a margin of 81.2°F.

The thermal evaluation also considers the 10 CFR 71.71(c)(1) requirement of an ambient condition of 100°F in still air with full insolation in order to determine the maximum temperatures of the package components. These temperatures are provided in Table 3-2.

3.4.3 Minimum Temperatures

To determine the minimum package temperatures, the thermal evaluation considers the 10 CFR 71.71(c)(2) ambient temperature of -40°F without insolation. The minimum temperatures of the package components are presented in Table 3-2.

3.4.4 Maximum and Minimum Normal Operating Pressures

The package internal pressures are computed in Appendix 3-3 and are summarized in this section. The package maximum normal operating pressure is defined in 10 CFR 71.4 as:

> "Maximum normal operating pressure" means the maximum gauge pressure that would develop in the containment system in a period of 1 year under the heat condition specified in 10 CFR 71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport.

The maximum normal operating pressure is determined assuming that the air trapped in the package is heated by virtue of the difference between the ambient conditions at the time of package closure and the maximum package temperature. The trapped air is treated as an ideal gas with a constant volume.

Other factors considered in the determination of the maximum normal operating pressure are the vapor pressure from the moisture within the LDCC and the gas pressure resulting from hydrogen and oxygen gas generation due to the radiolytic decomposition of the LDCC water content for a period of one year.

The maximum pressure is calculated using the lowest initial LDCC temperature and the highest temperature attained in the package when subjected to 100°F still air and 12 hours of 400 g-cal/cm^2 insolation. The minimum initial temperature is the lower bound of the LDCC pour temperature, which is 60°F. The calculated maximum temperature within the package after 12 hours of the prescribed insolation is 191.17°F. All calculated gauge pressures are with regard to the atmospheric pressure at 591 feet above sea level, which is 14.37 psia. This is the elevation at Big Rock Point, where the package will be sealed.

The hydrogen concentration after one year is conservatively taken to be 4% by volume (See Chapter 4 for hydrogen generation in the package.). The partial pressures calculated in Appendix 3-3 are as follows:

$P_{air} = 18.00$ psia
$P_{vapor} = 9.58$ psia
$P_{H2} = 1.17$ psia
$P_{O2} = 0.59$ psia

The total pressure, calculated as the summation of the above partial pressures, is:

$P_{total} = 29.34$ psia $= 15.0$ psig

Thus, the maximum normal operating pressure within the package is 15.0 psig at elevation 591 feet. However, it should be noted that in transport from Big Rock Point to the disposal facility in South Carolina, higher elevations might be encountered in the Appalachian Mountains. Mount Mitchell in North Carolina is the highest point in the Appalachians with an elevation of 6684 feet. The atmospheric pressure at this elevation is 11.47 psia. Thus, at these high elevations, the gauge pressure will be higher than when using 14.37 psia as a basis. Lower elevations may also be encountered, but they will not adversely affect maximum pressure calculations. At elevation 6684 feet, the maximum normal operating pressure within the package is 17.9 psig.

Since the maximum normal operating pressure in the package is greater than 5 psig as calculated above, a pressure test of the package is required to be conducted at an internal pressure of at least 50% higher per 10 CFR 71.85(b). This pressure test will be performed as described in Chapter 8.

The minimum normal operating pressure is calculated using the highest initial temperature and the lowest temperature attained in the package when subjected to -20°F (10 CFR 71.71(b)) still air and shade. The maximum initial temperature is the upper bound of the LDCC pour temperature, which is 85°F, and the minimum temperature within the package is –19.45°F. At this temperature, water vapor will not exist, and to be conservative, radiolysis is neglected. Therefore, the total pressure is only dependent on the air component. Appendix 3-3 shows that the minimum normal operating pressure within the package at elevation 591 feet is:

$$P_{total} = P_{air} = 11.61 \text{ psia} = -2.8 \text{ psig}$$

The minimum pressure is also calculated under the cold condition as specified in 10 CFR 71.71(c)(2). This temperature is calculated using the highest initial temperature and the lowest temperature attained in the package when subjected to -40°F still air and shade. The maximum initial temperature is the upper bound of the LDCC pour temperature, which is 85°F, and the minimum temperature within the package is –39.4°F. At this temperature, water vapor will not exist, and to be conservative, radiolysis is neglected. Therefore, the total pressure is only dependent on the air component. The minimum pressure within the package at -40°F, and at elevation 591 feet is:

$$P_{total} = P_{air} = 11.09 \text{ psia} = -3.3 \text{ psig}$$

3.4.5 Package Performance under NCT Differential Temperatures and Pressures

As stated in Section 3.4.2, the 10 CFR 71.43(g) requirement is satisfied since the maximum accessible surface temperature of the package, 103.8°F, is less than 185°F. To determine whether the package design is capable of withstanding the NCT thermal changes, the effect of the maximum and minimum temperature and pressure values discussed herein were considered in the structural evaluations detailed in Chapter 2. As concluded in Chapter 2, the package structural design under the subject conditions is adequate. Therefore, the package containment and shielding integrity are not impacted by the thermal NCT specified in 10 CFR 71.

3.5 THERMAL EVALUATION FOR HAC

Thermal response of the package under 10 CFR 71.73 HAC is documented in detail in Appendix 3-2. A summary of that evaluation is presented in the following sections.

3.5.1 Thermal Model

Consistent with the 10 CFR 71.73(c)(4) requirement, the package is fully engulfed in a hydrocarbon fuel/air fire with an average flame temperature of 1475°F for a period of 30 minutes. The model used in the HAC analysis is identical to that discussed previously for the NCT scenarios, although the boundary conditions are chosen to match the 10 CFR 71.73(c)(4)

requirements. The emissivity for the radiation heat transfer mechanism is increased to 0.9, consistent with 10 CFR 71.73(c)(4). Two HAC scenarios are evaluated corresponding to initial ambient temperatures of 100°F and -20°F, consistent with the temperature range cited in 10 CFR 71.73(b).

In parallel with the radiation heat transfer mechanism, a turbulent forced convection heat transfer coefficient of 2.6 BTU/hr-ft^2-°F is applied at the package outer surface for the 30-minute duration of the fire. Details regarding the computation of this value are provided in Appendix 3-2. Following the 30-minute duration of the fire, the ambient temperature returns to the initial value, the package surface emissivity is returned to a value of 0.66, and the forced convective heat transfer mechanism is replaced by the natural convective heat transfer mechanism discussed in Section 3.4.1. No significant material or geometrical deformation is expected to occur as a result of the HAC, therefore the steady-state temperature distribution of the package following removal of the applied heat sources is essentially the same as the initial temperature profile.

3.5.2 Package Condition and Environment

10 CFR 71.73(a) specifies that the hypothetical accident fire analysis be performed after the package has been subjected to free drop and puncture tests. Potential damage resulting from these tests is considered minimal to the package and of no significant effect on the thermal performance.

Puncture damage does not measurably alter the thermal behavior of the package. Puncture deformations imposed upon the package shell only affect the package configuration in the immediate vicinity of the mild steel bar strike. This area represents a very small fraction of the total package surface area, although this type of damage could result in localized increased temperatures. Conduction heat transfer to cooler adjacent areas in the radial, axial, and longitudinal directions would act to mitigate these localized maximum temperatures.

3.5.3 Maximum Temperatures

The highest package temperatures under the HAC occur when the pre-fire and post-fire ambient conditions are 100°F. The peak package temperature distribution occurs at the end of the 30-minute fire. Results of the HAC analysis are shown in Table 3-3.

3.5.4 Maximum Internal Pressures

The maximum pressure for the HAC is calculated using the lowest initial temperature and the package volume average temperature as computed in Appendix 3-3. The minimum initial temperature is the lower bound of the LDCC pour temperature, which is 60°F, and the volume "average" temperature is 311.7°F. It should be noted that the "average" temperature is used since using the "maximum" temperature would lead to artificially high pressures within the package. This is due to the conservatism used in the thermal analysis of Appendix 3-2, which does not account for vaporization of water within the package.

The factors contributing to the package internal pressure are addressed in Section 3.4.4. As calculated in Appendix 3-3, the maximum pressure within the package due to the HAC is 93.0 psig at elevation 591 feet. At elevation 6684 feet, the maximum pressure within the package is 95.9 psig. The partial pressures are summarized below.

$P_{air} = 21.33$ psia
$P_{vapor} = 79.6$ psia
$P_{H2} = 4.29$ psia
$P_{O2} = 2.15$ psia

3.5.5 Package Performance under HAC Differential Temperatures and Pressures

To evaluate the package performance under the HAC thermal changes, the effect of the maximum and minimum temperature and pressure values discussed herein were considered in the structural evaluations detailed in Chapter 2. As concluded in Chapter 2, the package performance under the subject conditions is in compliance with the acceptance criteria of 10 CFR 71.

3.6 REFERENCES

3.6-1 HEATING 7.2f, "Multidimensional Finite-Difference Heat Conduction Analysis," Sargent and Lundy Program No. 03.7.564-7.2.

Table 3-1
Thermal Material Properties

Material	**Density (lb/ft^3)**	**Thermal Conductivity (BTU/hr-ft-°F)**	**Specific Heat (BTU/lb_m-°F)**
LDCC	30.0	0.0917	0.15
LDCC	50.0	0.146	0.15
SA-302 Grade B Alloy Steel (RV Material)	489.0	23.3 @ 70°F	0.10 @ 70°F
SA-516 Grade 70 Carbon Steel (Package Material)	490.8	23.6 @ 70°F	0.11 @ 70°F
Type 304 Stainless Steel (Cladding)	501.1	8.6 @ 70°F	0.11 @ 70°F
Stainless Steel Foil Insulation	12.88	0.0232 @ 221°F	0.11 @ 70°F

Table 3-2
Package Temperatures Due to NCT

	100 °F No Solar	**100 °F Solar**	**-40 °F No Solar**	**-20 °F No Solar**
Component				
30 lb/ft^3 LDCC Max	117.5 °F	190.5 °F	-21.7 °F	-1.8 °F
30 lb/ft^3 LDCC Min	102.2 °F	156.2 °F	-38.7 °F	-18.8 °F
RV Max	109.9 °F	181.4 °F	-31.8 °F	-11.9 °F
RV Min	102.4 °F	169.9 °F	-38.2 °F	-17.1 °F
50 lb/ft^3 LDCC Max	104.7 °F	191.2 °F	-37.0 °F	-17.1 °F
50 lb/ft^3 LDCC Min	100.3 °F	156.6 °F	-39.4 °F	-19.4 °F
Package Max	103.8 °F	191.2 °F	-38.2 °F	-18.3 °F
Package Min	100.3 °F	156.1 °F	-39.4 °F	-19.4 °F
Maximum Surface	103.8 °F	191.2 °F	-38.2 °F	-18.3 °F

Table 3-3
Package Temperatures Due to HAC (at 100°F ambient)

	Fire Initiation t = 0 minutes	**Fire End** t = 30 minutes	**Post-Fire** t = 60 minutes	**Post-Fire** t = 90 minutes
Component				
30 lb/ft^3 LDCC	117.5°F	1137°F	1149°F	1050°F
RV	107.7°F	705°F	829°F	819°F
50 lb/ft^3 LDCC	102.6°F	1355°F	1187°F	1058°F
Top Lifting Plate	101.2°F	1444°F	1187°F	1050°F
Bottom Plate	100.8°F	1458°F	1195°F	1058°F
Maximum Surface	101.2°F	1458°F	1140°F	1004°F

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4.0 CONTAINMENT

Containment of the radioactive material is provided by the package[1] boundary. The package performance is evaluated under the Normal Conditions of Transport (NCT) and the Hypothetical Accident Conditions (HAC) of 10 CFR 71.71 and 71.73 as described in Chapters 2, 3, and 5. The package contains solid radioactive material with the vast majority of it bound in the neutron activated Reactor Vessel (RV) wall, insulation, and in the remaining RV internals. Only a very small percentage of radioactive material is surface contamination on the RV which will be fixed in place by the LDCC in the RV and in the annulus between the RV and the package. There is no gaseous or liquid radioactive material in the package. For more details on the contents of the package, see Chapter 1.

10 CFR 71.43 addresses general requirements for all package types. Specific containment requirements for Type B packages are stated in 10 CFR 71.51 addressing both NCT and HAC criteria of 10 CFR 71.71 and 71.73.

The following sections describe the containment boundary components and the regulatory requirements in more detail, and demonstrate compliance with these requirements.

4.1 CONTAINMENT BOUNDARY

The containment boundary consists of the body of the package (cylindrical shell plus the top and bottom plates), penetration seal plugs, and welds. The general requirements for the containment boundary are addressed in 10 CFR 71.43(b), (c), (e), (f), and (h). Description of the containment boundary presented in Sections 4.1.1 through 4.1.4, and the discussions presented in Section 2.4 demonstrate compliance with these general requirements.

4.1.1 Package

The package is designed in accordance with the criteria described in Chapter 2 taking into consideration the thermal and shielding requirements addressed in Chapters 3 and 5. As addressed in Chapter 2, the package will be fabricated with ASME SA-516, Grade 70 steel. The package shell is 3" thick with an additional 4" thick plate welded to its interior surface for a total length of approximately 8', in the former reactor core region of the RV. The top and bottom covers are 4" thick steel plates. The bottom plate will be welded to the cylinder during shop fabrication, and the top plate will be welded in place at Big Rock Point (BRP) after the RV is placed inside the package.

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1 The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

Fabrication and welding of the package will be performed in accordance with NUREG/CR-3854, "Fabrication Criteria for Shipping Containers" and NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials" to ensure construction compliance with the design.

4.1.2 Penetrations

The package has openings for LDCC filling, bolt holes on the top plate for studs attaching the RV flange to the top plate, and bolt holes for the lifting lug. The holes for LDCC filling and lifting lug bolts will be plugged and seal welded after use and prior to shipment. The RV attachment studs will be covered by cap nuts and sealed. These features will prevent opening of the package, fulfilling the tamper-proof requirement.

4.1.3 Seals and Welds

There are no gasketed closures on the package. All openings will be plugged and seal welded. Shop and field seal welding will be performed in accordance with the requirements of NUREG/CR-3019 and BNFL Inc.'s NRC Approved QA Program.

4.1.4 Closures

There are no closure devices, no vents, and no valves in this package. The package top plate will be welded to the body prior to transportation, and there will be no possibility of accidental opening of the package.

4.2 REQUIREMENTS FOR NORMAL CONDITIONS OF TRANSPORT

The requirements under NCT are as follows:

- 10 CFR 71.71 states: "Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in this section . . ." In summary, 10 CFR 71.71 requires the package to be designed to withstand the effects of the temperature, pressure, vibration, water spray, free drop, corner drop, compression, and penetration as defined in that section.

- 10 CFR 71.43(d) states: "A package must be made of materials and construction that assure there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent. Account must be taken of the behavior of materials under irradiation." Also, NRC Information Notice (IN) 84-72 addresses a concern regarding the potential for generation of combustible gases due to radiolysis during the package transport.

The gases generated during a period of one year should not exceed 5% of the free gas volume in any confined region of the package.

- 10 CFR 71.51(a)(1) states: ". . . a Type B package . . . must be designed, constructed, and prepared for shipment so that under the tests specified in Section 71.71, there would be no loss or dispersal of radioactive contents-as demonstrated to a sensitivity of 10^{-6} A_2 per hour, . . . and no substantial reduction in the effectiveness of the packaging."

- 10 CFR 71.51(c) states: "Compliance with the permitted activity release limits of paragraph (a) of this section may not depend on filters or on a mechanical cooling system."

Sections 4.2.1 through 4.2.3 demonstrate how compliance with the above criteria is satisfied.

4.2.1 Containment of Radioactive Material

As stated earlier, the package, including its welds and penetration seal welds, provides the mechanism for containment of the radioactive material. The LDCC placed in the RV and in the RV/package annulus will fix any loose surface contamination in place. Therefore, to ensure the radioactive material will remain contained, the integrity of the containment boundary under the NCT defined in 10 CFR 71.71 needs to be demonstrated.

Chapters 2, 3, and 5 describe the 10 CFR 71 criteria for which the package is designed. The evaluations described in Chapter 2 consider the conditions and criteria specified in Chapters 3 and 5. As concluded in Chapter 2, under all the applicable NCT specified in 10 CFR 71.71, the package structural integrity, and therefore, its containment boundary and shielding effectiveness will be maintained. This conclusion demonstrates that the package design is adequate for the NCT. Chapter 8 specifies the required acceptance tests and/or inspections to ensure the package construction is in compliance with its design.

By demonstrating that the package design and construction are adequate for NCT, compliance with the 10 CFR 71.51(a)(1) requirement for "no substantial reduction in the effectiveness of the packaging" is shown. Compliance with the requirement for "no loss or dispersal of radioactive contents-as demonstrated to a sensitivity of 10^{-6} A_2 per hour" stated in 10 CFR 71.51(a)(1) is addressed in Section 4.2.3.

4.2.2 Pressurization of Containment Vessel

10 CFR 71.43(d) requires demonstration of no significant chemical, galvanic, or other reaction among the package components, among package contents, or between the package components and the package contents, including possible reaction resulting from inleakage of water. The reactions "between the package components, or between the package components and the package contents" are addressed in Chapter 2. As concluded in that chapter, the materials of the subject components are such that there will be no significant reaction among them. Compliance

with the requirements regarding the "inleakage of water," and "behavior of materials under irradiation" is also demonstrated in Chapter 2.

Package pressurization due to the "reaction among the package contents" (10 CFR 71.43(d)) is addressed by: 1) evaluation of the % combustible gases generated in the package (IN 84-72), and 2) the internal pressure considerations in the package design:

1) The potential for combustible gas generation exists due to the radiolytic decomposition of the LDCC in the package. If a sufficient amount of hydrogen is generated, it could become explosive when combined with oxygen in the package. NRC IN 84-72 specifies the combustible gases generated during a period of one year should not exceed 5% of the free gas volume in any confined region of the package.

   To address the flammability concern, an evaluation is performed (Appendix 4-1) using the methodology in EPRI Report NP-4938. This evaluation demonstrates that the maximum hydrogen gas generated in the package over a period of one year is 3.7% of the free gas volume in any confined region of the package. This satisfies the maximum allowable limit of 5% as specified in IN 84-72.

2) Other than the flammability concern, the hydrogen generated in the package also contributes to the changes in the internal pressure of the package. The effect of hydrogen gas and other contributing factors are considered in calculating the maximum internal pressure of the package as described in Chapter 3. Demonstration of the containment integrity under the resulting internal pressure is accomplished by evaluation of the package's structural integrity as detailed in Chapter 2. As concluded in that chapter, the package design is adequate for the effect of this maximum internal pressure.

The above discussions demonstrate containment integrity in compliance with the requirements of 10 CFR 71.43(d).

4.2.3 Containment Criterion

As discussed in Section 4.2.1, analytical demonstration of the containment integrity under NCT of 10 CFR 71.71 provides assurance that the radioactive materials will remain contained in the package. To demonstrate compliance with the 10 CFR 71.51(a)(1) requirement for the maximum allowable radioactivity release limit of 10^{-6} A_2 per hour, a leak test will be conducted as described in Chapter 8. The test criteria are determined based on the leak rate evaluation per ANSI N14.5-1997 as documented in Appendix 4-2. This evaluation takes into account both the NCT and HAC (Section 4.3) radioactivity release limits of 10 CFR 71.51(a)(1) and (2) in calculating the acceptance criterion for the leak test. This criterion is the reference air leakage rate, L_R, calculated as 0.1 cm^3/sec, satisfying both NCT and HAC limits of 10 CFR 71.51.

Since no filters or mechanical cooling system are used in the package design to satisfy the permitted activity release limits of paragraph (a) of 10 CFR 71.51, compliance with 10 CFR 71.51(c) is ensured.

4.3 REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The requirements under HAC are as follows:

- 10 CFR 71.73 states: "Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section . . ." 10 CFR 71.73 requires the package to be evaluated for the effects of free drop (30 ft), crush, puncture, fire, and immersion as defined in that section.

- 10 CFR 71.51(a)(2) states: " . . . Type B package must be designed, constructed, and prepared for shipment so that under the tests specified in Section 71.73, there would be no escape of Krypton-85 exceeding $10A_2$ in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 1 rem/hour at 1 meter from the external surface of the package."

Sections 4.3.1 and 4.3.2 describe how compliance with the above criteria is demonstrated.

4.3.1 Containment of Radioactive Material

As stated earlier, the package including its welds and penetration seal welds provides the mechanism for containment of the radioactive material. The LDCC placed in the RV and in the annulus between the RV and the package will fix any loose surface contamination in place. To ensure compliance with the HAC containment criteria, the package structural performance under the 10 CFR 71.73 conditions is evaluated to determine its condition after the hypothetical accidents as detailed in Section 2.7. The resulting post-accident package condition provides the input needed for evaluating the potential external radiation and radioactivity release due to these accidents. The latter evaluation demonstrates compliance with the acceptance criteria of 10 CFR 71.51(a)(2) as described below.

In order to evaluate the potential external radiation and release levels due to damages to the package boundary under the HAC, an evaluation is performed in Appendix 4-3. This evaluation conservatively assumes four non-mechanistic package damage scenarios such that the highest radioactivity areas of the package are exposed. These non-mechanistic scenarios bound the damages to the package predicted by the structural evaluation discussed in Section 2.7. The bounding, non-mechanistic damage scenarios considered in Appendix 4-3 are:

- A 1" wide circumferential gap is formed in the package, exposing a 1" wide band of the RV and its insulation along the package beltline, in the former reactor core region.

- A 1" wide by 48" long longitudinal gap is formed in the package, exposing a 1" by 48" longitudinal band of the RV and its insulation, centered on the package beltline in the former reactor core region.

- A 6" diameter hole is punched through the 7" thick package shell, centered on the package beltline in the former reactor core region, exposing a 6" diameter area of the RV and its insulation.

- The top plate of the package completely separates from the package shell.

The results of the evaluations in Appendix 4-3 show that the bounding radiological consequences for the above four non-mechanistic cases of containment breach correspond to the complete loss of the top plate. The results of this evaluation related to the 10 CFR 71.51(a)(2) criteria are as follows:

The release limit for Krypton-85 is met because there is no gaseous radioactive material in the package. This is because all fuel has been removed from the package, and there are no sources or reservoirs for noble gas in the package. The release limit of 10 CFR 71.51(a)(2) for other airborne radioactive material is A_2 in one week which is calculated as 3.59 curies/week for the BRP package. This release limit is satisfied because the total quantity of radioactivity available for airborne release in the package is only 2.90 curies.

A conservative estimate of the release via a liquid[2] pathway including potentially leachable material is only 1.76 curies as calculated in Appendix 4-3. The water release limit is A_2/week which is 19.72 curies for this package. This limit is greater than the calculated potential release of 1.76 curies, and is therefore satisfied.

Appendix 4-3 also evaluates the worst case dose rate due to the HAC. This dose rate is 0.79 rem/hr at 1 meter, which also corresponds to the non-mechanistic scenario of a complete loss of the top plate.

The foregoing discussion demonstrates that the 10 CFR 71.51(a)(2) radioactivity release limit of A_2 in one week and the external radiation limit of 1 rem/hour at 1 meter from the package surface are satisfied.

4.3.2 Containment Criterion

The discussion presented in Section 4.3.1 demonstrates that a breach of containment would result in a maximum external radiation and radioactivity release less than that allowed by 10 CFR 71.51(a)(2). Therefore, due to the low releasable radioactivity level of the package contents, no reference air leakage rate limit needs to be specified as the HAC containment criterion.

[2] Note that the air and water releases and release limits differ because the water release includes radioactivity bound in the steel that can be removed only by corrosion of the steel.

4.4 REFERENCES

4.4-1 NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," Lawrence Livermore National Lab., Prepared for NRC, March 1985.

4.4-2 NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," Lawrence Livermore National Lab., Prepared for NRC, March 1985.

4.4-3 NRC Information Notice 84-72, "Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation," September 10, 1984.

4.4-4 EPRI Report NP-4938, "Methodology for Calculating Combustible Gas Concentration in Radwaste Containers," Electric Power Research Institute, March 1987.

ANSI N14.5-1997, "Leakage Tests on Packages for Shipment," American National Standard for Radioactive Materials, Approved February 5, 1998.

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5.0 SHIELDING EVALUATION

The Big Rock Point (BRP) Reactor Vessel (RV) and its contents will be transported in a Type B package[1] as an exclusive use shipment. Shielding design of the package is evaluated for compliance with the following regulatory criteria.

10 CFR 71.47(b) specifies the following limits for an exclusive use package during Normal Conditions of Transport (NCT):

1) 200 mrem/hour on the accessible external surface of the package;

2) 200 mrem/hour at any point on the outer surface of the transport vehicle. In the case of an open vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the package, and on the lower external surface of the vehicle;

3) 10 mrem/hour at any point 2 meters from, in the case of an open vehicle, the vertical planes projected from the outer edges of the conveyance;

4) 2 mrem/hour in any normally occupied space. (This provision does not apply to private carriers if exposed personnel wear dosimetry in conformance with 10 CFR 20.1502. The BRP package will be transported as an exclusive use shipment. This criterion is not a design requirement, and compliance with it will be ensured through a requirement for the transportation personnel to wear dosimetry as stated in Section 7.2.2.j.)

Also under NCT, 10 CFR 71.51(a)(1) requires that there will be "no significant increase in external surface radiation levels." Hypothetical Accident Conditions (HAC) criteria as stated in 10 CFR 71.51(a)(2) limit the dose rate at 1 meter from the external surface of the package to 1 rem/hour.

The following sections describe the shielding design and demonstrate compliance with the regulatory requirements.

[1] The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

5.1 DISCUSSION AND RESULTS

5.1.1 Design Features

Shielding of the radioactive material is provided by the cylindrical steel package with welded steel flat top and bottom covers described in Chapter 1. The package is approximately 25' long and 13' in diameter. The package shell is 3" thick with an additional 4" thick plate welded to its interior surface for a length of approximately 8', in the former reactor core region of the RV. This extra shielding is positioned starting at a distance of approximately 8'-6" from the bottom of the package. The top[2] and bottom covers are 4" thick steel plates.

The Low Density Cellular Concrete (LDCC) in the annulus between the package and the RV (50-60 lb/ft^3) and the LDCC in the RV (30-36 lb/ft^3) provide minor additional shielding, immobilize the surface contamination inside the RV, and prevent the contents of the package from shifting during transportation. The package design therefore precludes the possibility of components shifting and causing dose rate changes.

After the package has been loaded with its contents, and prior to transportation, field surveys will be performed to verify that the dose rates do not exceed the 10 CFR 71 limits as stated in Sections 5.0 and 7.2.2.j. As described throughout Chapter 5, the package is designed using conservative analyses and sufficient safety margins to ensure that no additional shielding will be required after the field surveys. However, in the unlikely event that the field survey identifies localized elevated dose rates exceeding the 10 CFR 71 limits, additional ½" thick steel plate may be added to those localized areas of the package.

The package configuration is presented in Figure 2-1. Nominal dimensional values are used in the shielding analysis because the effect of manufacturing tolerances on the gamma shield design is insignificant, and the LDCC ensures that the contents will not shift.

5.1.2 Summary of Maximum Radiation Levels

Figure 5-1 shows the computer model used in the shielding evaluation documented in Appendix 5-1. The model and the methodology used for the shielding evaluation are described in the later sections of this chapter.

The results of the shielding evaluation for NCT are listed in Table 5-1. As shown in that table, the 10 CFR 71.47(b) limits are satisfied. It should be noted that the package will be loaded onto a flat-bed transporter, and the diameter of the package is larger than the width of the bed.

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[2] The dimples in the top plate described in Section 2.1.1 are excluded from the shielding evaluation model. The lack of the dimples in the model is more than compensated by omitting the RV flange from the model. Since the RV flange thickness is much greater than the missing shielding area due to the dimples, the results of the shielding analyses presented here are still conservative.

Therefore, the dose rates reported for the "surface of the package" address the 10 CFR 71.47(b) criteria with regard to the definition of surfaces and distances.

Also, under NCT, 10 CFR 71.51(a)(1) requires that there will be "no significant increase in external surface radiation levels." This criteria is satisfied since the LDCC in the package, as discussed in Section 5.1.1, will prevent the radioactive contents from migrating and causing dose rate changes.

Hypothetical Accident Conditions (HAC) criteria as stated in 10 CFR 71.51(a)(2) limit the dose rate at 1 meter from the external surface of the package to 1 rem/hour. The limits of 10 CFR 71.51(a)(2) are considered containment integrity criteria. Compliance with these criteria is addressed in Chapter 4.

5.2 SOURCE SPECIFICATION

The radioactive material is composed of the reactor vessel without the vessel head, selected activated internal steel components, and some of the vessel stainless steel insulation as detailed in Section 1.2.5. The radioactive material consists of irradiated (activated) steel with a small quantity of surface contamination. There is no gaseous or liquid radioactive material in the package. There is less than 0.05 gram of fissile material and less than 0.05 curies of plutonium.

Measured data is combined with analytical models to determine the source term and shield design. The source term/analytical model combination is benchmarked by comparing the results with measured dose rates and isotopic analyses (Tables 5-10, 5-12 and 5-13). These comparisons confirm that; (a) the calculated dose rates at locations on the outside of the RV bound measured dose rates at the same locations, and (b) the calculated inventories used for the RV wall and reflective insulation bound the measured inventories. These dose rate measurements and isotopic inventories were made at locations of expected peak activity as determined by the structure of the RV internals. The comparisons demonstrate that the source term/analytical model combination bounds the final package.

For the purposes of discussion, the source term is divided into two components; (a) surface contamination and, (b) activation. Both source term components are decay corrected to the shipment date of September 1, 2002.

The total activity in the package does not exceed 1.31E+04 curies. The total activity is presented in Table 5-4. The total surface contamination activity distribution by nuclide is presented in Table 5-9. The total activation activity distribution by nuclide is presented in Table 5-11.

5.2.1 Surface Contamination

The RV wall inner surfaces and RV internals were chemically decontaminated in 1998 (Appendix 5-4). The remaining quantity of surface contamination is small. It is characterized

based on a 0.52-gram sample of material that was taken from the wetted surface of the core spray nozzle, and analyzed by Teledyne Brown Engineering Environmental Services (Appendix 5-2). The sample isotopic analysis is presented in Table 5-12. This isotopic mix was increased by about 50% to envelop potential variations in H-3 and transuranic radionuclides (Appendix 1-2). The final isotopic mix is multiplied by the total wetted surface area to obtain the total contamination inventory. The total contamination inventory that is used in the analyses is calculated to be 2.9 curies. The total surface contamination is presented in Table 5-9.

5.2.2 Activation

The quantity and distribution of the activation products in the RV wall and internals were determined using the same methodology that has been used to support decommissioning activities at Shoreham, Yankee Rowe, Maine Yankee, Trojan, and Saxton (Appendix 5-2). One additional analysis was performed to determine the quantity of activation products that remained in the Grid Bar End Pieces (GBEP) after the bulk of the upper grid bars had been removed (Appendix 1-2).

The ANISN computer program was used to estimate neutron fluxes at several radial and axial locations in the RV using the SAILOR cross section library. The resultant fluxes were then used as inputs to the ORIGEN2 computer program to perform activation analysis on individual components. The ORIGEN2 results were normalized to radiation measurements taken on the internal components.

The ANISN model explicitly represented the reactor core, neutron windows, thermal shield, seal weights, RV cladding, RV wall, RV insulation, cooling jacket tank, and biological shield. The neutron windows, which are unique to the BRP RV, were specifically included in the ANISN model because they generated azimuthal flux peaks. There are four neutron windows located in the former reactor core region, between the thermal shield and the reactor core, on the east and west side of the RV. The neutron windows are 6" diameter schedule 160 pipe, capped and filled with helium. Since the neutron windows displace reactor coolant in the periphery of the core, they reduce the neutron attenuation and produce localized flux peaks at the RV wall. The effects of the neutron windows were included in the determination of the total quantity of activation products. (Because they generated flux peaks, one of the azimuthal neutron window locations was used when dose rates were measured for comparison with the analytical model.)

The reactor core was broken into three subsections to model the out-in fuel management strategy. Variation in water density in the axial direction was explicitly addressed. Core inlet water density was used for the lower internals regions. The former reactor core region was broken into nine axial regions based on water quality and void fraction. The upper region of the RV used a water/steam mixture corresponding to the nominal reactor exit quality.

The ORIGEN2 model used the BWRUS cross section library for the core activation, and cross sections from the thermal cross section library were used for the thermal activation. The core and thermal ORIGEN2 input fluxes were weighted to obtain a location-specific neutron spectrum

for each component or region of interest. The thermal values were adjusted for the local area temperatures.

Five sets of surveys were conducted inside the water filled RV to provide the empirical data used to normalize source terms. They are discussed in detail in Appendices 1-2 and 5-2. (These radiation measurements are distinct from those taken on the outside of the RV that were used for comparison with the combined source term/shielding analysis model.) The detailed radiation profiles of the internals that were used to normalize the radionuclide concentrations are listed below:

- Axial Distribution (surveys taken on the south side of the RV)
- Azimuthal Distribution (surveys taken every 18° from 108° to 180° at the 604' and 606' elevations)
- Radial Distribution (surveys taken on the top guide bars)
- Neutron Windows Axial Distribution (survey taken on the neutron windows every 6" axially)
- Grid Bar Longitudinal Distribution (surveys taken every 12" along the removed section of the grid bar)

These radiation profiles were used in conjunction with QAD-CGGP, MegaShield™ , and MicroShield™ models to estimate the Co-60 content in the components surveyed. The Co-60 activities estimated using the radiation profiles were compared to the Co-60 activities estimated using ORIGEN2 to determine scaling factors used to normalize the activity in each component. Activities for other components, and for nuclides other than Co-60, were calculated from the ORIGEN2 results using the appropriate scaling factors. These scaling factors are discussed in detail in Appendices 1-2 and 5-2.

One separate and additional calculation was performed to determine the Co-60 content of the GBEP. The GBEP are small segments of the top guide bars that remained in the RV after the bulk of the guide bars were cut and removed. These segments are at the ends of the guide bars, in relatively lower flux regions (compared to the core planar average) and thus require a more detailed estimate than that for the upper guide as a single unit. A separate survey was conducted on each guide bar section that was removed. These surveys were used in conjunction with MicroShield™ models to estimate the Co-60 content. The methodology is discussed in detail in Appendix 1-2.

The activation estimates for the RV, internals and GBEP were further refined, and decay corrected to 9/1/2002. The activation inventories were benchmarked by comparison with isotopic analyses from the RV wall and external reflective insulation. As discussed below, the activation inventories bound the inventories at the locations of the expected peaks.

The total activation source term is provided in Table 5-11. The Co-60 activity for each component is provided in Table 5-5.

5.2.3 Gamma Source

The principal gamma emitter is Co-60 which accounts for 54% of the total activity and >99% of the total dose rate. The input to the analyses consists of radionuclide concentrations. The gamma and beta decay spectra are calculated from nuclide specific data contained in DRALIST (RSIC Data Library DLC-80) and TPASGAM (RSIC Data Library DLC-88-C).

Two gamma spectra are presented in Table 5-6, one for the RV wall, which is carbon steel, and one for the reflective insulation which is stainless steel. The gamma decay source strength (photons/sec) as a function of energy is provided for information only. It is not used as input to the analyses.

5.2.4 Neutron Source

There is no fuel or other neutron source material included in the package. The total quantity of fissile material is less than 0.05g (Appendix 1-2). Under these conditions, a neutron source is not applicable to this package.

5.2.5 Flux and Dose Rate Variations

The source term development accounts for spatial variations in neutron flux. Three radial ANISN models were used to account for the azimuthal effect of the neutron windows and seal weights on the flux in the RV wall and components (e.g., thermal shield, mirror insulation, etc.). A separate axial model with ten core regions was used to determine the axial variation in the flux. The dose rate surveys that were used for normalizing the source term included the areas where the flux peaks were expected to be located. This is discussed in detail in Appendix 5-2.

The shielding analytical model ensures that dose rate peaks at contact on the external surface of the package, and at 2 meters from the package, are captured. The model uses 16 side, 4 top and 4 bottom dose point locations to ensure that the package dose rate peaks are identified. Key dose point locations are shown in Figure 5-1. The peak dose rates are listed in Table 5-1. The peak contact dose rate is 52.8 mrem/hr. It is located at the upper transition from 7" thick steel to 3" thick steel. The peak 2-meter dose rate is 8.67 mr/hr. It is located at the top of the package.

5.2.6 Source Term/Analytical Model Benchmark

The source term and analytical model benchmarking are considered together because the analytical model is used to calculate the dose rates that are compared with the measured dose rates. The source term, measured isotopic composition, and dose rates are all decay corrected to the reference date of September 1, 2002. As discussed below, the methodology used to generate the source term and shield design has produced results that are conservative, bound measured data, and have sufficient margin to accommodate the peak dose rates and activities.

Samples and dose rates were obtained at two locations. These locations are outside the north and west RV walls, at the midsection of the former reactor core region. Due to the core configuration, these locations are where the fuel was closest to the RV wall, and are the locations where the flux is expected to peak. The flux at the west wall is also influenced by the neutron windows, as discussed in Section 5.2.2. The highest activation peak will be at one of these two locations.

The activation source term was benchmarked by; (a) comparing the RV wall source term with a sample obtained from the north RV wall, and (b) comparing the reflective insulation source term with samples obtained from the north and west RV mirror insulation. The data is compared in Table 5-13. As can be seen from the data, the activation source term Co-60 content, which is responsible for 99% of the external package dose rate, envelopes the data measured at the peak locations for both the RV wall and the reflective insulation.

The source term/analytical model combination was benchmarked by comparing calculated dose rates with dose rates measured at the locations on the RV wall where the peak dose rates are expected. These locations were reached via core bores through the concrete shield around the RV. The measured dose rates at these locations include the effects of the activated steel and concrete adjacent to the RV as well as the RV itself, so they are higher than if they were due solely to the RV and internals. This introduces some conservatism into the comparison, although the magnitude has not been determined. The comparison is presented in Table 5-10. As can be seen, the source term/analytical model produces dose rates that bound the measured dose rates.

5.3 MODEL SPECIFICATION

The components described in Section 5.1.1 are used in the computer model for shielding evaluation[3]. All shielding is for gamma radiation because, as discussed above, there is no neutron source in the package. Point-kernel methodology is appropriate for analysis of gamma shielding and dose rates. The computer code used is ISOSHLD-PC, which contains predefined geometry models, and a generalized geometry package commonly referred to as the QAD P5A model. The analyses were performed using the QAD model. A total of 41 zones, 34 boundaries, and 27 energy groups are used to model the RV, internals and shielding[4]. Each shield transition and source component is explicitly modeled. Dose rates are calculated near transitions, and along axial and radial traverses, to ensure that dose rates are calculated at locations that resulted in the highest values. Activity is distributed above and below the former reactor core region based on thermal flux profiles. However, for conservatism, where appropriate (e.g., RV wall),

[3] In order to accommodate design flexibility, the bottom plate was modeled as both 3" and 4" thick steel. The final design utilizes a 4" thick bottom plate. The dose rates presented in Table 5-1 are for the more conservative case of a 3" plate model which bounds the dose rate for the actual 4" bottom plate.

[4] Penetrations, including penetration holes for injection of LDCC, are modeled separately to ensure streaming dose rates are properly analyzed. The penetration analyses are discussed in Section 5.3.3.

the entire activity is also placed in the former reactor core region. This artificially increases the total source term but is judged to be acceptably conservative. An additional 1 Ci source was added to the bottom of the RV to conservatively account for dross[5] remaining in the package. The shielding materials are LDCC and steel; no hydrogenous material or neutron absorption material is required. The geometries are based on nominal dimensions because (a) the RV and internals are fixed in place with LDCC, and (b) fit-up and manufacturing tolerances have no effect on gamma shielding calculations.

5.3.1 Description of Radial and Axial Model Configuration

A sketch showing the source locations, dose point locations and shielding is provided in Figure 5-1. Shielding is provided by the package and LDCC described in Section 5.1.1. The shielding is azimuthly symmetric.

5.3.2 Material Densities and Compositions

The materials in the package are steel and LDCC. The compositions for steel are provided in Table 5-7. The compositions are essentially the same for all the point-kernel models, although the region numbering differs.

The minimum LDCC densities were used for the shielding evaluation, i.e. 30 lb/ft^3 in the RV and 50 lb/ft^3 in the annulus between the package and the RV. Use of the minimum LDCC densities maximizes calculated dose rates since gamma dose rates are inversely proportional to shielding material density.

5.3.3 Streaming Path Analysis

There are no true radiation streaming paths in the package. This is because all RV openings are capped with steel and filled with LDCC, and there are no openings in the package shielding. Specific analyses were performed to evaluate locations that could be affected by RV nozzle openings.

Potential streaming paths in the RV were analyzed separately to ensure that these highly localized dose rates were not masked by the dose rates from the package in its entirety. Streaming was calculated for the 14" diameter steam nozzle in the upper section of the vessel and the 20" diameter inlet nozzle in the lower section of the vessel. These calculations were performed using the ISOSHLD-PC computer program. These are point-kernel calculations similar to the QAD P5A model, the only difference being that the geometries are predefined in ISOSHLD-PC. The calculated streaming dose rates are summarized in Table 5-3.

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[5] Dross at the bottom of the RV will be removed prior to removing the RV from the cavity as stated in Section 7.1.1.

The sources used for the streaming analyses were obtained by using the fraction of the component (thermal shield, etc.) that could contribute to dose rates outside the penetration. These are discussed in detail in Appendix 5-1.

The geometrical relationship between the GBEP source and the 3" diameter RV nozzle penetrations located directly above the GBEP was evaluated. The geometrical relationship between the GBEP and RV nozzle precludes a direct streaming path (See Appendix 5-1).

The ports used to inject LDCC into the space between the package and the RV were evaluated for potential streaming paths. After the LDCC is injected, each port will be filled with a steel plug that is seal welded to the steel shield. In order to allow for the weld process and other possible interferences, the plug is assumed to be 1" thinner than the steel shell in which it is placed. The ports themselves are in regions of the shield that are near low dose rate zones. These ports are located below the bottom of the former reactor core region and approximately 3' above the core region. The location of the ports is such that any reduction in shielding effectiveness due to a reduction in the steel thickness by 1" is compensated for by the reduced source term near the port. The result is that the steel plugs may be 1" thinner than the steel shield and neither the contact dose rates or 2 meter dose rates will be above the limits set forth in 10CFR71.47. The supporting evaluation is contained in Appendix 5-1.

5.4 EVALUATION

The package is evaluated against the dose rate criteria specified in 10 CFR 71.47(b). Dose rates were calculated at several locations, including streaming paths and shield thickness transitions, to ensure that the locations of the maximum radiation levels were captured.

The gamma dose rates were calculated using standard point-kernel methodology. Gamma flux-to-dose-rate conversion factors are from ANSI/ANS-6.1.1-1977. The buildup factors are those used in QAD-CGGP, and gamma attenuation data is from Mass Attenuation Coefficients from Storm and Israel Nuclear Data Tables. The neutron flux is negligible because there is less than 0.05 Ci of fissile material and there is no neutron source material in the package. Therefore, neutron dose rates are not calculated. Dose points are located based on the outer package dimensions.

The QAD model of ISOSHLD-PC is used to explicitly model the sources and shielding, including shield thickness transitions, RV and internals, and the LDCC densities. The QAD model utilizes 41 zones and 34 boundaries. Figure 5-2 shows a sectional view of the QAD geometry model. Each QAD zone is identified in Figure 5-2 by a number enclosed in a box. A description of the zones is contained in Table 5-8. The origin for the coordinate system is at the midplane of the former reactor core region, along the former fuel centerline. The QAD model used to determine dose rates for the GBEP source region varies slightly from the model depicted in Figure 5-2. A detailed description of QAD model differences is provided in Appendix 5-1.

For source components that extend above and below the former reactor core region, the source outside the core region is estimated by applying a factor that is proportional to the ratio of the thermal flux in the region of interest to the thermal flux in the core region. Streaming path analyses to evaluate the dose rate impact due to radiation streaming out of the nozzle penetrations and through the plugged LDCC injection ports were performed using the predefined geometries in ISOSHLD-PC. A sketch of the streaming path model used in the RV nozzle penetration streaming evaluations is provided in Figure 5-3. The model used to evaluate radiation streaming through the plugged LDCC injection ports is similar to that shown in Figure 5-3 except for differences in number and types of shield materials.

5.5 REFERENCES

5.5-1 ORNL CCC-254, ANISN-ORNL "One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," Oak Ridge National Laboratory, April 1991.

5.5-2 DLC-76, SAILOR-Coupled, Self-Shielded, 47-Neutron, 20 Gamma-Ray P3, Cross Section Library for Light Water Reactors, Oak Ridge National Laboratory Radiation Shielding Information Center, April 1991.

5.5-3 ORNL/TM-7175, A User's Manual for the ORIGEN Computer Code, Croff, A.G., 1980.

5.5-4 ORNL CCC-493, QAD-CGGP, A Combinatorial Geometry Version of QAD-P5A, A Point Kernel System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor, July 1990.

5.5-5 ORNL/TM-11018, Standard and Extended Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code, Ludwig, S. B., and Renier, J. P., December 1989.

5.5-6 MegaShield™, Proprietary point-kernel computer code, WMG, Inc.

5.5-7 "Trojan Nuclear Plant Reactor Vessel and Internals Removal Project Safety Analysis Report," PGE-1076, Revision 1, July 1999, Portland General Electric Company.

5.5-8 ISOSHLD-PC, "A Computer Program for General Purpose Isotope Shielding Analysis Using Gaussian Quadrature Integration of Point And Line Kernel Dose Rate Equations for Predefined and Ad Hoc Source and Shield Geometries," Sargent & Lundy Program Number 03.7.508-4.10, June 1, 1999.

5.5-9 American Nuclear Society, "American National Standard for Neutron and Gamma-Ray Fluence to Dose Factors," ANSI/ANS 6.1.1-1977, La Grange Park, Illinois.

5.5-10 Mass attenuation coefficients from Storm, E. and Israel, H., Nuclear Data Tables, A7, 56-681 (1970), Radiation Shielding and Information Center (RSIC) Data Library Collection DLC-15/STORM-ISREAL.

5.5-11 Kocher, D.C., "DRALIST – Radioactive Decay Data for Application to Radiation Dosimetry and Radiological Assessments," RSIC Data Library DLC-80, 1/15/81.

5.5-12 Kocher, D.C., "Radioactive Decay Data Tables," DOE/TIC-11026, 1981.

5.5-13 Dickens, J.K. and Perdue, P.T., "TPASGAM Radioactive Decay Library of Gamma-Ray Energies,Branching Ratios and Cross Sections," RSIC Data Library DLC-88-C, 10/26/84.

5.5-14 MicroShield User's Manual, Version 5, Grove Engineering, 1998.

Table 5-1
Peak Dose Rates For NCT (mrem/hr)

	Package Surface			2 Meters From Package Surface		
	Side	Top	Bottom (Note 1)	Side	Top	Bottom (Note 1)
Gamma	52.8	16.2	1.16	7.31	8.67	0.44
Neutron	N/A	N/A	N/A	N/A	N/A	N/A
Total	**52.8**	**16.2**	**1.16**	**7.31**	**8.67**	**0.44**
10 CFR 71.47(b) Limit	200	200	200	10	10	10

Note 1: The dose rate presented is for a 3 inch plate. See Section 5.3 for details.

Table 5-2
Maximum Dose Rates For HAC

1 Meter From Package Surface	
10 CFR Part 71.51(a)(2) Limit N/A (See Note 2 below.)	1000 mrem/hr

Note 2: This limit is not applicable to the shielding design of the BRP RV package. The limits of 10 CFR 71.51(a)(2) are considered containment integrity criteria and are addressed in Chapter 4. See Section 5.1.2 for details.

Table 5-3
Streaming Path Dose Rates (mrem/hr)

	Shielded Transport Package Intact	
Distance	**Contact**	**2 meters**
20 in. Dia. Inlet Nozzle	1.09	0.079
14 in. Dia. Steam Nozzle	12.3	0.862

Table 5-4
Total Package Radiation Source Term as of 9/1/2002

Nuclide	Activity (Curies)	Weighting Fraction	Nuclide	Activity (Curies)	Weighting Fraction
C14	1.42E+00	1.09E-04	Ce144	3.03E-04	2.31E-08
Fe55	4.96E+03	3.78E-01	Eu152	1.66E-01	1.26E-05
I129	1.53E-03	1.17E-07	H3	4.28E+00	3.26E-04
Ni59	6.23E+00	4.74E-04	Am241	8.81E-04	6.71E-08
Ni63	1.03E+03	7.82E-02	Cm242	3.79E-07	2.89E-11
Sr90	6.04E-03	4.60E-07	Cm243	2.35E-04	1.79E-08
Tc99	5.22E-03	3.98E-07	Cm244	2.23E-04	1.70E-08
Mn54	1.41E+01	1.08E-03	Pu238	5.83E-04	4.44E-08
Co60	7.11E+03	5.42E-01	Pu239/240	6.93E-04	5.28E-08
Zn65	1.36E-03	1.04E-07	Pu241	3.31E-02	2.52E-06
Nb94	2.15E-02	1.64E-06	Pu242	3.07E-06	2.34E-10
Sb125	1.35E-02	1.03E-06			
			Total	1.31E+04	1.00E+00

Table 5-5
Component Co-60 Activity as of 9/1/2002

Component	**Co-60 Activity (Curies)**
Steam Baffle	6.17E+00
Sparger	4.93E-01
Top Guide Plate	6.55E+02
Seal Housing	7.44E+01
Thermal Shield	4.24E+03
TS Retainer	6.56E+01
Seal Weights	6.30E+02
Core Support Plate	8.02E-02
Inlet Diffuser	1.28E-02
Inlet Baffle	9.34E-02
RV Wall	2.31E+02
RV Clad	1.40E+02
Grid Bar End Pieces (GBEP)	1.03E+03*
RV Insulation	3.33E+01
Total	7.11E+03

* Includes 40% conservative margin.

Table 5-6
Gamma Spectra As of 9/1/2002
(This table is for information only. Analyses used nuclide input. See Section 5.2.3)

Group	Ebar (Avg Energy) (MeV)	RV Wall (Fraction)	RV Insulation (Fraction)
1	1.50E-02	3.24E-01	8.41E-02
2	2.50E-02	6.62E-04	8.74E-04
3	3.50E-02	4.07E-04	5.38E-04
4	4.50E-02	2.48E-04	3.21E-04
5	5.50E-02	1.49E-04	1.99E-04
6	6.50E-02	1.05E-04	1.41E-04
7	7.50E-02	7.20E-05	9.69E-05
8	8.50E-02	5.12E-05	6.90E-05
9	9.50E-02	1.21E-04	1.64E-04
10	1.50E-01	5.50E-05	6.97E-05
11	2.50E-01	2.62E-06	2.19E-06
12	3.50E-01	5.53E-06	2.77E-06
13	4.75E-01	1.33E-06	6.68E-07
14	6.50E-01	5.85E-05	7.62E-05
15	8.25E-01	1.72E-03	8.57E-04
16	1.00E+00	5.58E-06	2.79E-06
17	1.23E+00	6.72E-01	9.12E-01
18	1.48E+00	4.32E-06	2.16E-06
19	1.70E+00	0.00E+00	0.00E+00
20	1.90E+00	0.00E+00	0.00E+00
21	2.10E+00	0.00E+00	0.00E+00
22	2.30E+00	0.00E+00	0.00E+00
23	2.50E+00	0.00E+00	0.00E+00
24	2.70E+00	0.00E+00	0.00E+00
25	3.00E+00	0.00E+00	0.00E+00
26	6.14E+00	0.00E+00	0.00E+00
27	7.11E+00	0.00E+00	0.00E+00

Table 5-7
Material Compositions (wt/%)

	304 Stainless Steel	**Carbon Steel**
Nitrogen	0.045	0.008
Chromium	18.400	0.000
Manganese	1.530	1.350
Iron	70.600	97.570
Nickel	10.000	0.610
Molybdenum	0.260	0.580
Niobium	0.009	0.002
Cobalt	0.141	0.012

Note: These compositions are from NUREG/CR 3474.

Table 5-8
QAD Model Zone Description

Zone Number	Zone Description	Model Discussion
1	Internal Vessel LDCC	Cylindrical area inside thermal shield filled with LDCC
2	Thermal Shield	Annular area, 0 to 360 degrees
3	Internal Vessel LDCC	Annular area (0 to 360 degrees) between thermal shield and RV wall filled with LDCC
4	Reactor Vessel Cladding	Annular area (0 to 360 degrees)
5	Reactor Vessel Wall	Annular area (0 to 360 degrees)
6	Reactor Vessel Insulation	Annular area (0 to 360 degrees)
7	External Vessel LDCC	Annular area (0 to 360 degrees) between RV wall and Package
8	Package Shell	Annular area (0 to 360 degrees) for 7" thick iron shield band over former reactor core region
9	Air Outside and to Side of Package	Annular area (0 to 360 degrees) of air
10	Internal Vessel LDCC	Cylindrical geometry of internal vessel LDCC, just below thermal shield
11	Thermal Shield Retainer	10" wide by about 30" high segment of stainless steel, extending from 84.4 degrees to 95.6 degrees
12	Internal Vessel LDCC	Annular area (0 to 360 degrees) between thermal shield retainer and RV wall
13	External Vessel LDCC	Annular area (0 to 360 degrees) between RV wall and Package
14	Package Shell	Annular area (0 to 360 degrees) for 3" thick iron shield band below former reactor core region
15	Internal Vessel LDCC	Cylindrical geometry of internal vessel LDCC just below thermal shield retainer
16	Core Support Plate	Cylindrical geometry near bottom of RV
17	Internal Vessel LDCC	Annular area (0 to 360 degrees) adjacent to core support plate
18	Reactor Vessel Bottom	Cylindrical geometry. The model for Reactor Vessel bottom is conservatively located directly below the core support plate.
19	External Vessel LDCC	Cylindrical geometry directly below RV bottom
20	Package Bottom Plate	Cylindrical geometry, 3" thick iron
21	Internal Vessel LDCC	Cylindrical geometry adjacent to top guide plate
22	Top Guide Plate	Annular area (0 to 360 degrees) near top of thermal shield

Zone Number	Zone Description	Model Discussion
23	External Vessel LDCC	Annular area (0 to 360 degrees) between RV wall and 3" thick package shell above former reactor core region
24	Package Shell	Annular area (0 to 360 degrees) for 3" thick iron above former reactor core region
25	Internal Vessel LDCC	Cylindrical geometry above top guide plate
26	Internal Vessel LDCC	Cylindrical geometry above thermal shield
27	Thermal Shield Seal	Annular area (0 to 360 degrees) above thermal shield
28	Internal Vessel LDCC	Cylindrical geometry above thermal shield seal
29	Internal Vessel LDCC	Cylindrical geometry adjacent to sparger
30	Sparger	Annular area (0 to 360 degrees) under steam baffle near top of RV
31	Internal Vessel LDCC	Annular geometry between sparger and RV wall
32	Steam Baffle	Cylindrical geometry
33	Internal Vessel LDCC	Annular geometry between steam baffle and RV wall
34	Internal Vessel LDCC	Cylindrical geometry above steam baffle
35	Package Top Plate	Cylindrical geometry. Top plate is at RV flange elevation.
36	Air	Above package
37	Air	Below package
38	Package Shell	7" thick iron shell segment, extending 16" up from top of former reactor core region
39	Package Shell	7" thick iron shell segment, extending 10" down from bottom of former reactor core region
40	External Vessel LDCC	Annular geometry between RV wall and Zone 38
41	External Vessel LDCC	Annular geometry between RV wall and Zone 39

Table 5-8 (Continued)

Table 5-9
Total Surface Contamination

Nuclide	(Curies)	Nuclide	(Curies)
C14	7.51E-04	CE144	3.03E-04
FE55	1.99E-01	EU152	0.00E+00
I129	1.53E-03	H3	5.30E-04
NI59	0.00E+00	AM241	8.81E-04
NI63	1.42E-01	CM242	3.79E-07
SR90	6.04E-03	CM243	2.35E-04
TC99	6.07E-04	CM244	2.23E-04
MN54	8.22E-03	PU238	5.83E-04
CO60	2.49E+00	PU239/240	6.93E-04
ZN65	1.36E-03	PU241	3.31E-02
NB94	0.00E+00	PU242	3.07E-06
SB125	1.09E-02	Total	2.90E+00

Table 5-10
Comparison of Measured and Calculated Dose Rates

Description and Measurement Date	Measured Dose Rate Decay Corrected to 9/1/2002 (R/hr)	Calculated Dose Rate Decay Corrected to 9/1/2002 (R/hr)
North RV – 6/7/00	17.1	65.29
West RV – 6/7/00	29.8	65.29
North RV – 7/28/00	12.1	65.29
West RV – 7/28/00	34.2	65.29
North RV – 1/17/01	14.5	65.29

Table 5-11
Total Activation Source Term

Nuclide	Design Basis Activity (Curies)
C14	1.42E+00
FE55	4.96E+03
NI59	6.23E+00
NI63	1.03E+03
TC99	4.61E-03
MN54	1.41E+01
CO60	7.11E+03
NB94	2.15E-02
SB125	2.64E-03
EU152	1.66E-01
H3	4.28E+00
Total	1.31E+04

Table 5-12
Comparison of Measured and Design Basis Surface Contamination

Nuclide	Measured (µCi/gm)	Design Basis (µCi/gm)
C14		2.35E-04
FE55	5.67E-03	6.22E-02
I129	4.80E-04	4.78E-04
NI63	3.71E-03	4.44E-02
SR90		1.89E-03
TC99	1.90E-04	1.90E-04
MN54	2.40E-03	2.57E-03
CO60	5.76E-01	7.78E-01
ZN65	4.26E-04	4.25E-04
SB125		3.41E-03
CE144		9.47E-05
H3		1.66E-04
AM241	5.07E-05	2.75E-04
CM242	8.69E-08	1.18E-07
CM243		7.35E-05
CM244		6.97E-05
PU238	1.17E-05	1.82E-04
PU239/240	2.60E-05	2.17E-04
PU241		1.03E-02
PU242		9.60E-07

Table 5-13
Comparison of Measured and Calculated Activation

Nuclide	Measured North RV Wall (μCi/gm)	RV Wall Source (μCi/gm)	Measured North Insulation (μCi/gm)	Measured West Insulation (μCi/gm)	RV Insulation Source (μCi/gm)
C14	4.91E-03	7.29E-03	3.01E-03	1.34E-03	1.08E-02
FE55	2.11E+01	4.76E+01	7.08E+01	3.55E+01	3.45E+01
NI59	5.65E-03	3.17E-02	1.97E-01	1.48E-01	4.72E-02
NI63	6.05E-01	5.23E+00	2.05E+01	2.23E+01	7.77E+00
TC99	3.51E-04	2.35E-05			3.50E-05
MN54	1.17E-01	7.16E-02	5.72E-02	7.11E-02	1.07E-01
CO60	5.21E+00	1.39E+01	2.20E+01	5.60E+01	5.70E+01
ZN65	5.83E-03			2.98E-01	
NB94		1.09E-04			1.63E-04
SB125		1.34E-05			2.00E-05
H3	5.71E-03	2.18E-02	3.58E-03	2.78E-03	3.24E-02
PU239/240	8.61E-07		8.54E-07	1.46E-06	
U233/234	6.85E-06		4.04E-06	2.13E-06	
U238	4.98E-07		9.46E-07	0.00E+00	
EU152		8.43E-04			1.25E-03
Total	2.70E+01	6.68E+01	1.14E+02	1.14E+02	9.94E+01

FIGURE WITHHELD UNDER 10 CFR 2.390

*

Figure 5-1: Source, Shielding, and Dose Point Locations

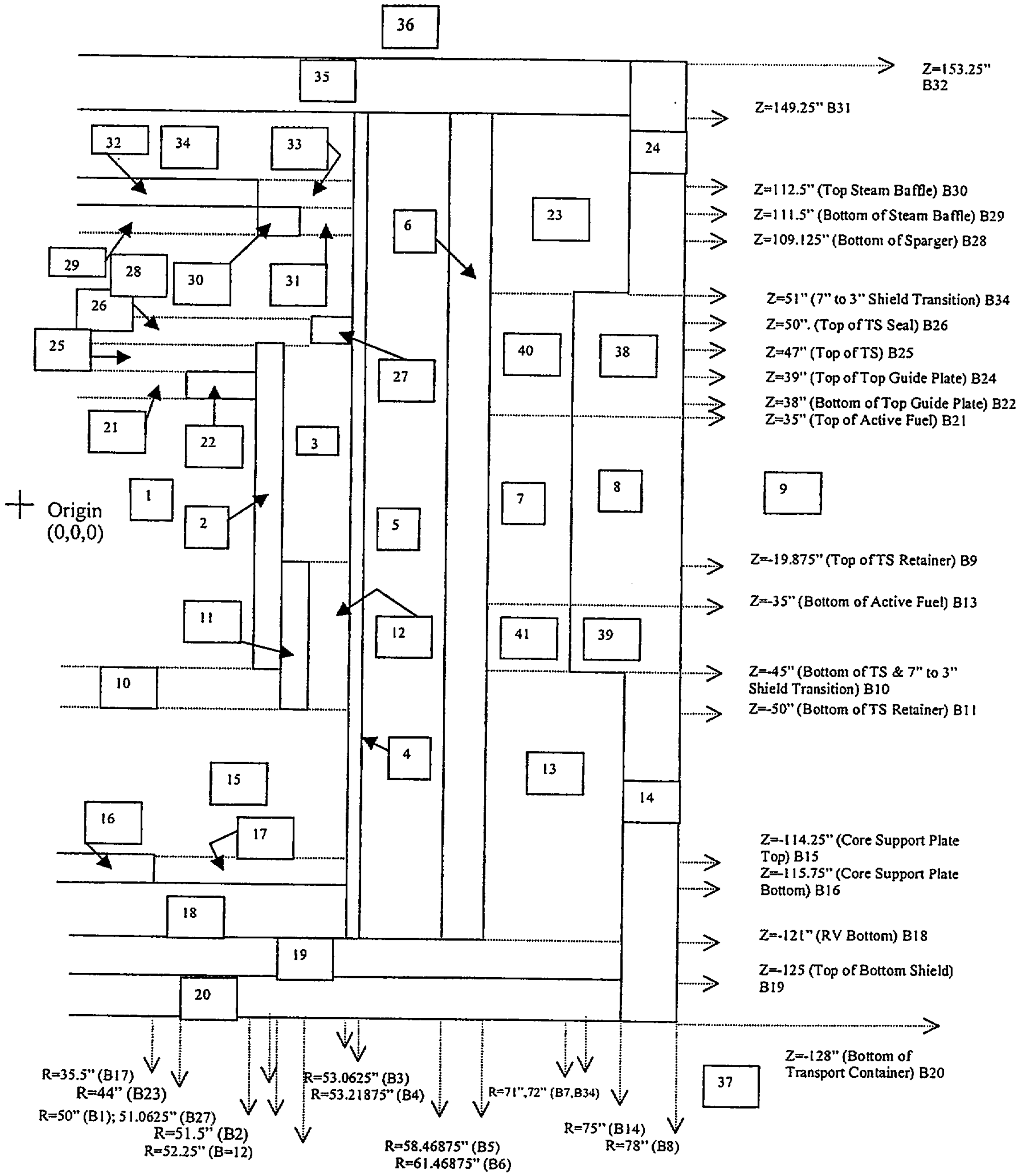

Figure 5-2: QAD Geometry Model

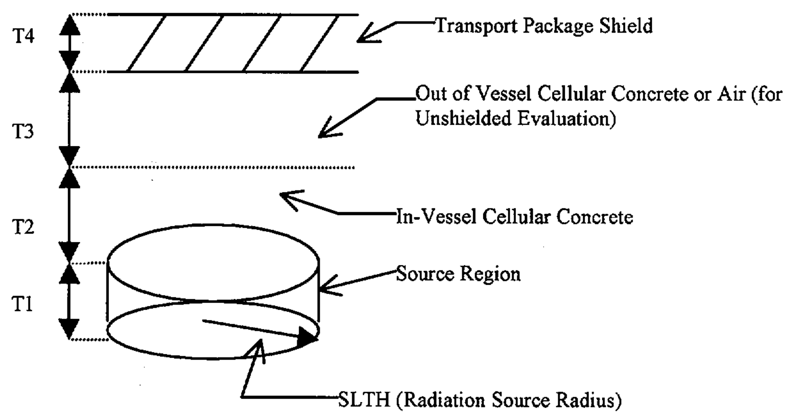

Figure 5-3: Streaming Path Model

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6.0 CRITICALITY EVALUATION

The total quantity of fissile material in the package is less than 0.1 g as calculated in Appendix 1-2. This is less than the 15 g limit stated in 10 CFR 71.53(a)(1). Therefore, the package is exempt from the fissile material criteria of 10 CFR 71.55 and 71.59, and no criticality evaluation is required.

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7.0 OPERATING PROCEDURES

This chapter describes the operating procedures to be used for loading and transportation of the Big Rock Point (BRP) Reactor Vessel (RV) package[1] in compliance with the requirements of 10 CFR 71, Subpart G. These requirements are stated in 10 CFR 71.85, "Preliminary Determinations", and 10 CFR 71.87, "Routine determinations". Also, 10 CFR 71.47(c) and (d) require written instructions to be used during shipment. Compliance with these requirements will be demonstrated in this chapter and Chapter 8 where indicated.

In accordance with Regulatory Guide 7.9, this chapter describes the procedures for loading and unloading the package. The BRP package is a 10 CFR 71, exclusive use Type B package to be used for a one-time shipment and disposal of the RV at the licensed low level radioactive waste disposal facility of Chem-Nuclear Systems, Barnwell, South Carolina. Since the package is permanently sealed and will be buried with its contents, the "Unloading and Transportation of Empty Package" as defined in Regulatory Guide 7.9 do not apply here. For the same reason package opening instructions as stated in 10 CFR 71.89 are not applicable.

This chapter describes the plan for loading and transportation of the package in order to ensure safe operations in compliance with the regulations and the package evaluation in this SAR. All the required operations discussed in this chapter will be performed in accordance with written procedures approved under BNFL Inc.'s Quality Assurance (QA) Program which complies with 10 CFR 71, Subpart H, Quality Assurance. All the applicable record keeping, inspections, reporting, and advance notification requirements addressed in 10 CFR 71.91, 93, 95, and 97 will be complied with under BNFL Inc.'s QA Program. All of the appropriate procedures will comply with the ALARA requirements of 10 CFR 20.

Implementation of the above as detailed through the remaining sections of this chapter demonstrates compliance with the requirements of 10 CFR 71.81, "Applicability of Operating Controls and Procedures."

[1] The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

7.1 PACKAGE LOADING / PRELIMINARY DETERMINATIONS

The package and its radioactive contents are described in detail in Chapter 1. As discussed in that chapter, the loaded package will contain the RV, some of the RV stainless steel insulation and limited internals as described in Chapter 1, as well as Low Density Cellular Concrete (LDCC) filling the void space in the RV and in the annulus between the RV and the package. 10 CFR 71.85, "Preliminary Determinations" requires certain tests and inspections to be performed prior to the first use of all package types. Compliance with these requirements is demonstrated by the operations described in Section 8.1.1, "Visual Inspections", Section 8.1.2, "Weld Examinations," Section 8.1.3, "Structural and Pressure Tests", and Section 7.1.3, "Package Closure and Marking".

The following sections describe the process of loading the package with its contents in the sequential order of operations.

7.1.1 Preparation for Loading

Prior to RV removal, certain actions will be taken in preparation for the package loading process. These operations are:

- Removing the dross and any free-standing water
- Cutting and capping the nozzles
- Bolting the package top plate to the RV
- Removing the RV stabilizer arms and support brackets.

The above preparations will be described in detail in approved work packages, and the operations will be performed in accordance with written procedures. Prior to placing the RV into the package, the following will be confirmed:

- The contents to be loaded into the package are those authorized by the NRC in the Certificate of Compliance.
- The use of the package complies with the NRC's conditions of approval in that certificate.

7.1.2 Loading RV Into Package, and Injecting LDCC

After the initial preparations described in Section 7.1.1 are complete, the RV is ready to be lifted out of its cavity and loaded into the package. The package will be fabricated as one section, with a top plate to be welded to the cylindrical shell after the RV is loaded into the package. The fabricated package will initially be delivered to BRP in a horizontal position on a road transporter. The package will then be moved into a vertical position by means of two A-frame pivot devices supporting two trunnions on opposite sides of the package. The RV will be lifted out of the cavity by means of the package top plate and lifting lug assembly, and placed into the package in a vertical position. The lifting devices are described in Section 7.1.2.1.

After the RV has been loaded into the package, the top plate will be welded to the package and the lifting lug will be removed. At this time, Low Density Cellular Concrete (LDCC) will be injected into the RV through the holes in the top plate which were previously used for the lifting lug attachment. The void space inside the RV will be filled with LDCC. The design characteristic of the LDCC critical for this application is its density. A density range of 30-36 lb/ft^3 has been considered for the LDCC in the RV in the design of the package.

To ensure that no voids are left within the RV, the LDCC density range is achieved, and no water remains in the LDCC, a controlled LDCC injection process will be followed. The LDCC injection process will be accomplished using written procedures approved under BNFL Inc.'s QA Program. The process of LDCC injection including confirmatory tests performed for demonstration of compliance with the requirements will be documented.

After this LDCC injection phase has been completed and the LDCC has cured, the lifting lug, which was previously removed after loading the RV into the package, will be refitted on the top plate. The package will then be down-ended into a horizontal position using the Reactor Building 125 ton capacity crane, the lifting lug, the trunnions, and the A-frame pivot device.

Once the package loaded with the RV is in a horizontal position, the lifting lug will be removed for the last time, and the penetration holes on the top plate will be plugged and seal welded. The pressure and leak tests described in Sections 8.1.3 and 8.1.4 will be conducted after this step.

Following successful completion of these tests, the second phase of LDCC injection will begin while the loaded package is still in a horizontal position. The penetration holes in the body of the package will be used to fill the annulus between the RV and the package with 50-60 lb/ft^3 LDCC. Once the annulus has been filled and the LDCC has cured, these holes will also be plugged and seal welded.

The foregoing operations will be performed in accordance with written procedures approved under BNFL Inc.'s QA Program.

7.1.2.1 Lifting Devices

The package lifting devices include a lifting lug assembly and two trunnion assemblies. The lifting lug and the package top plate will be used to lift the RV from the cavity. The lifting lug and the trunnions will be used for the package up-ending and down-ending operations discussed in Section 7.1.2. The lifting lug and trunnions are not structural parts of the package and will be removed prior to shipment.

There are no other structural parts of the package that could be used for lifting the package during transport. Therefore, the 10 CFR 71.45(a) requirement regarding inoperability of these devices during transport is satisfied.

As stated in Section 2.5, since these devices are not structural parts of the package, their design is not part of the package safety analysis for Part 71 considerations. These devices, however, are

designed to withstand the applied loads during the planned operations. This design has been performed under BNFL Inc.'s QA Program. General information on the lifting device components is provided below.

7.1.2.1.1 Lifting Lug Assembly

The lifting lug assembly will be fabricated from nominal 2" thick steel plates, with a nominal 2-1/2" thick base plate of the same material. The lug base plate will be secured to the package top plate with high strength fasteners. The lifting lug will be used for lifting the RV (without LDCC fill) and placing it in the package in the vertical orientation. The lug will also be used for down-ending the package to the horizontal position after field welding of the top plate. The lug will be removed, and the bolt holes will be plugged and seal welded prior to shipment.

Down-ending will be performed after the top plate is field welded to the package and the LDCC has been injected into the RV, but before the LDCC is injected into the annular space between the RV and the package.

7.1.2.1.2 Trunnion Assembly

Each trunnion assembly is machined such that the portion that bears on the A-frame saddle is 15" in diameter. The trunnion is machined with a bolted flange for attachment to the package. The trunnion will rest in a nominal 4" wide A-frame saddle, which will provide a pivoting support for the package during the up-ending and down-ending operations. The trunnion will be bolted to the package shell through a steel beveled plate. The beveled plates will be welded to the package shell and provide a flat mounting surface for the flange. The trunnion assembly will be removed, and all bolt holes will be plugged and seal welded prior to shipment.

7.1.3 Package Closure and Marking

The penetration holes will be plugged and seal welded in accordance with the details provided on approved design drawings and written procedures, after the process of LDCC injection has been completed. The general configuration of the loaded package is illustrated in Chapter 2, Figure 2-1. The package welds will be visually inspected, and weld examination will be performed as described in Sections 8.1.1 and 8.1.2.

10 CFR 71.85(c) states:

> "The licensee shall conspicuously and durably mark the packaging with its model number, serial number, gross weight, and a package identification number assigned by NRC . . ."[2]

To comply with the above, the package will be marked per the above requirements as indicated in Figure 2-1.

[2] Compliance with the remainder of this paragraph is addressed in Section 8.1.1.

7.2 PREPARATION FOR TRANSPORT

7.2.1 Evaluation of Transport Route and Conveyance

Clearances will be sought from the County and State road authorities and railroad operators for the shipment of the package. These clearances will be based upon evaluations completed by these authorities. Should any modifications be required as a result of these evaluations, they will be completed prior to the start of the transportation.

7.2.2 Routine Determinations

Prior to shipment of the package, the licensee is required to ensure that the package and its contents satisfy the requirements of 10 CFR 71.87, "Routine Determinations" Items “a” through “k” below address each of the determinations enumerated in 10 CFR 71.87, and state how compliance with these requirements will be satisfied.

a) Proper Package

10 CFR 71.87(a) requires the licensee to determine that:

> "The package is proper for the contents to be shipped."

The package described in this Safety Analysis Report (SAR) is an exclusive use Type B package complying with the criteria of 10 CFR 71 with no exemption. The package will be fabricated in accordance with the design requirements contained within this SAR and the NRC issued Certificate of Compliance for the package. Performing the tests/inspections/examinations specified in Chapter 8 will assure fabrication compliance with the package design presented in this SAR and in accordance with the Certificate of Compliance issued by the NRC. This confirms compliance with 10 CFR 71.87(a).

b) Unimpaired Package

10 CFR 71.87(b) requires the licensee to determine that:

> "The package is in unimpaired physical condition except for superficial defects such as marks or dents."

As described in Section 8.1.1, the package will be inspected visually and the pressure retaining base material and fabrication welds nondestructively examined before transportation from BRP. These activities will ensure compliance with this requirement.

c) Closure Devices

10 CFR 71.87(c) requires the licensee to determine that:

> "Each closure device of the packaging, including any required gasket, is properly installed and secured and free of defects."

There are no closure devices on this package. Therefore, this requirement does not apply.

d) Liquid Containment

10 CFR 71.87(d) requires the licensee to determine that:

> "Any system for containing liquid is adequately sealed and has adequate space or other specified provision for expansion of the liquid."

The contents of the package are solid radioactive material and LDCC. Any free-standing water existing in the RV will be removed in accordance with approved written procedures as stated in Section 7.1.1. Therefore, this requirement does not apply.

e) Pressure Relief Devices

10 CFR 71.87(e) requires the licensee to determine that:

> "Any pressure relief device is operable and set in accordance with written procedures."

This requirement is not applicable to the package since it does not contain any pressure relief devices.

f) Loading and Closure

10 CFR 71.87(f) requires the licensee to determine that:

> "The package has been loaded and closed in accordance with written procedures."

The package loading and closure processes will be performed in accordance with written procedures as stated in Sections 7.1.1 through 7.1.3. Therefore, this requirement is satisfied.

g) Neutron Absorber

10 CFR 71.87(g) requires the licensee determine that:

> "For fissile material, any moderator or neutron absorber, if required, is present and in proper condition."

The package contains a negligible amount of fissile material as stated in Chapter 6. Therefore, no moderator or neutron absorber material is required.

h) Lifting or Tie-down Attachments

10 CFR 71.87(h) requires the licensee to determine that:

> "Any structural part of the package which could be used to lift or tie-down the package during transport is rendered inoperable for that purpose unless it satisfies the design requirements of §71.45."

As described in Sections 7.1.2.1 and 7.2.3.1, the lifting devices and tie-down system are not structural parts of the package. The lifting devices, i.e. the trunnions and the lifting lug, will be removed prior to shipment. The studs that penetrate the top plate will be covered by cap nuts that will be seal welded prior to shipment. Therefore, there is no structural part of the package that could be used to lift or tie-down the package during transport.

i) Surface Contamination

10 CFR 71.87(i) requires the licensee to determine that:

> "The level of non-fixed (removable) radioactive contamination on the external surfaces of each package offered for shipment is as low as reasonably achievable, and within the limits specified in DOT regulations in 49 CFR 173.443."

Prior to shipment, the package surface contamination levels will be measured to ensure the 49 CFR 173.443 limits are satisfied. Since the package will be completely sealed, and its design and fabrication will be confirmed through the acceptance testing and inspections specified in Chapter 8, there will be no increase in the external contamination levels during transportation.

j) External Radiation Levels

10 CFR 71.87(j) requires the licensee to determine that:

> "External radiation levels around the package and around the vehicle, if applicable, will not exceed the limits specified in §71.47 at any time during transportation."

Applicable to exclusive use shipment, the 10 CFR 71.47(b) limits are:

1. 200 mrem/hr on the external surface of the package,
2. 200 mrem/hr at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load, and on the lower external surface of the vehicle,
3. 10 mrem/hr at any point 2 meters from the vertical planes projected from the outer edges of the vehicle, and,

4. 2 mrem/hr in any normally occupied positions of the vehicle, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry in conformance with 10 CFR 20.1502.

The BRP package is designed to comply with the limits in items 1 through 3 above, under the 10 CFR 71.71 Normal Conditions of Transport (NCT) as demonstrated in Chapter 5. The exclusive use shipment of the package will be accomplished by carriers contracted by BNFL Inc. BNFL Inc. will provide the carriers with appropriate written instructions for this shipment as stated in Section 7.3. These instructions will require that the transportation personnel wear radiation dosimetry in conformance with 10 CFR 20.1502. Therefore, the requirement in item 4 above is also satisfied.

Prior to shipment, radiation surveys of the package will be performed to ensure that the external dose rates satisfy the requirements of 10 CFR 71.47. Even though some loose surface contamination exists in the RV, the LDCC filling the void space in the package, as described in Section 7.1.2, will fix this internal contamination in place. Therefore, the contamination will not migrate and cause radiation level changes during the shipment.

k) Package Surface Temperature

10 CFR 71.87(k) requires the licensee to determine that:

> "Accessible package surface temperatures will not exceed the limits specified in §71.43(g) at any time during transportation."

As determined by analysis in Section 3.4.2, the maximum accessible package surface temperature does not exceed the 10 CFR 71.43(g) limit.

7.2.3 Loading Package on Road Transporter

Once all the preliminary and routine determinations discussed in Sections 7.1 and 7.2 have been completed, the loaded package is ready for shipment. While the package is in a horizontal position as described at the end of Section 7.1.2, it will be moved through the containment construction access opening on a slide beam system. The package will be moved out of BRP by a road transporter. Hydraulic cylinders will be used to provide the motive force in moving the package onto the transporter.

After the package is placed on the road transporter, it will be secured in place utilizing the tie-down system designed for this specific application. Prior to the transporter leaving the plant, the tie-down system will be inspected by qualified personnel to ensure the package is properly and securely tied to the transporter. Loading and securing the package on the transporter will be performed in accordance with approved written procedures. The tie-down system is described as follows:

7.2.3.1 Tie-down Devices

The package does not include attachments for tie-down devices. The package will be down-ended from the vertical position, moved out of the BRP Containment Building, and placed horizontally on the saddle support system mounted on the road transporter. After the down-ending but prior to shipping, all handling devices will be removed, and all bolt holes and penetrations will be plugged and seal welded such that they can not be used as a tie-down attachment. The saddle support system includes five pairs of saddles, which fit the package cylindrical curvature, five pairs of tie-down cables that encircle the package to restrain its upward motion, and longitudinal bumpers at each end of the package.

As stated in Section 2.5, since these devices are not structural parts of the package, their design is not part of the package safety analysis for Part 71 considerations. The tie-down system components however, are designed to withstand the applied loads during transportation in accordance with the load provisions of ANSI N14.2 "Tie-down for Truck Transport of Radioactive Materials," and Association of American Railroads (AAR), "Open Top Loading Rules Manual". This design has been performed under BNFL Inc.'s QA Program.

7.3 TRANSPORTATION

Transportation of the package will be accomplished through a combination of road and rail. After the package has been tied down to the road transporter, it will be driven out of the plant to the prepared rail siding. The package will then be transferred to the rail car using hydraulic jacks and a slide beam system. The package will be secured in place using the tie-down system described in Section 7.2.3.1, above. Prior to the rail car leaving the rail siding, the tie-down system will be inspected by qualified personnel to ensure that the package is properly and securely tied to the rail car. The last portion of the transportation will take place using a road transporter to the disposal site in Barnwell, South Carolina. The same operations discussed above will take place for the package transfer from the rail car to the road transporter, tie-down, and inspections.

To organize and coordinate all of the transportation activities, a Transportation Safety Plan (TSP) will be developed by BNFL Inc. This document will be utilized as the transportation operations controlling document throughout the entire shipment from BRP to the disposal site. The TSP will identify appropriate operating controls such as the minimum required temperature during transportation discussed in Section 7.3.1. The TSP will also include details such as the transportation route, mode of transportation and transfer locations, distances, processes, and equipment, and identifies responsibilities and interfaces for the transportation activities. These activities include package transfer from one conveyance to the next, tie-down instructions and inspections, radiological controls, package delivery to the disposal site, etc. Guidance on the required activities to coordinate with appropriate federal and local agencies in response to emergencies will also be provided in the TSP.

10 CFR 71.47(c) states:

> "For shipments made under the provisions of paragraph (b) of this section, the shipper shall provide specific written instructions to the carrier for maintenance of the exclusive use shipment controls. The instructions must be included with the shipping paper information."

10 CFR 71.47(d) states:

> "The written instructions required for exclusive use shipments must be sufficient so that, when followed, they will cause the carrier to avoid actions that will unnecessarily delay delivery or unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public."

Implementation of the transportation activities described in the TSP will be in accordance with written procedures. This will satisfy the 10 CFR 71.47(c) and (d) requirement for written instructions to be used during the shipment. These instructions will be included with the shipping paper information provided to the carrier.

7.3.1 Operating Controls for Minimum Required Temperature

Section 2.8 sets specific minimum temperature limits during the transportation in order to protect the package top and bottom plates from brittle fracture. To comply with the criteria discussed in that section, operating controls will be specified in the TSP as summarized below:

Prior to leaving BRP, weather reports along the transportation route will be reviewed, and the transportation will not begin if an ambient temperature below 0°F is predicted. The ambient temperature will be monitored throughout the transportation duration. If this temperature falls to 0°F while the transporter is in motion, transportation will be stopped. The criteria for resuming the transportation will be based on the package surface temperature. Measurements of this temperature will be taken at a few locations on the top and bottom plates. Once the minimum measured surface temperature reaches 0°F or above, the transportation will resume. The ambient temperature monitoring will continue, and the foregoing operations will take place again if the minimum ambient temperature condition is reached.

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8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The Big Rock Point (BRP) Reactor Vessel (RV) package[1] design requirements and operating activities are discussed in Chapters 1 through 7. The package is designed as an exclusive use 10 CFR 71 Type B package to be used for one-time transportation and disposal of the RV at the low level radioactive waste disposal facility of Chem-Nuclear Systems at Barnwell, South Carolina. This chapter describes the acceptance tests and inspections that will be performed on the package to ensure compliance with its design requirements, and the requirements of 10 CFR 71.85, "Preliminary Determinations" and 10 CFR 71.87, "Routine Determinations."

8.1 ACCEPTANCE TESTS / PRELIMINARY DETERMINATIONS

Acceptance tests and inspections will be performed prior to the transportation of the package in compliance with 10 CFR 71.85. The sequential order of these inspections and tests will be coordinated with other operations as detailed in Chapter 7. All the tests and inspections on the package described in this chapter will be conducted and documented in accordance with written procedures approved under BNFL Inc.'s NRC Approved Quality Assurance (QA) Program.

8.1.1 Visual Inspections

10 CFR 71.85(a) states:

> "The licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging."

10 CFR 71.87(b) requires the licensee to determine (also addressed in Section 7.2.2.b) that:

> "The package is in unimpaired physical condition except for superficial defects such as marks or dents."

10 CFR 71.85(c) states:

> "... the licensee shall determine that the packaging has been fabricated in accordance with the design approved by the Commission."[2]

[1] The term "package" in this SAR is used for the shipping container loaded with its radiological contents. For simplicity, the term "package" has also been used throughout this SAR when describing the configuration of the container alone. The approval requested in this application is for the loaded package for a one-time-only use. This package will be permanently sealed and buried with its radiological contents as specified in this SAR.

[2] The 10 CFR 71.85(c) requirement regarding the package marking is addressed in Section 7.1.3.

The package will be fabricated under BNFL Inc.'s NRC Approved QA Program and in accordance with the design presented in this SAR. Inspections and examinations of the fabricated material and shop welds will be done at the shop, per the criteria identified in the fabrication specifications and under BNFL Inc.'s NRC Approved QA Program, prior to shipping the package to BRP. The fabricated package will be delivered to BRP in one section as a cylindrical body (including the bottom plate) with a top plate which will be field welded to the body to form one integral unit after the package has been loaded, as discussed in Section 7.1.

Upon arrival on site, the fabricated sections will be examined visually to assure no damage has occurred during transport. The top plate field weld to the package will be nondestructively examined in accordance with subsection NB of the ASME Boiler & Pressure Vessel Code (BPV) as explained in Section 8.1.2. If the examinations reveal any defects, the defects will be evaluated to ascertain whether remedial actions may be warranted. Inspections and repairs, if required, will be appropriately documented.

The above inspections will be performed using approved written procedures under BNFL Inc.'s NRC Approved QA Program. Fabrication of the package under BNFL Inc.'s NRC Approved QA Program and accomplishment of the inspections described above will satisfy the 10 CFR 71.85 and 10 CFR 71.87 requirements stated Section 8.1.1. Compliance with the requirements of 10 CFR 71.85(a) is further assured with the weld examinations described in Section 8.1.2.

8.1.2 Weld Examinations

Compliance with the requirements of 10 CFR 71.85(a) is also confirmed by the package weld examinations. Examinations of the shop welds will be done at the shop, per the criteria identified in the fabrication specifications and under BNFL Inc.'s NRC Approved QA Program, prior to shipping the package to BRP.

Nondestructive examination of containment-related welds will be performed. These nondestructive examinations will be performed in accordance with conservatively interpreted guidance contained in NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials." Stress intensity values for Class 1 of Section III of the ASME BPV have been used in the design of the package. These values contain assumptions about materials, fabrication, and nondestructive examination; consequently, the package pressure retaining materials, fabrication and examination will be in accordance with Subsection NB of this code with an exception. The exception is that the girth weld joining the top plate to the cylindrical shell will be volumetrically examined. The version of Section III that will be used is the 1995 Edition with the 1996 Addenda as modified by 10 CFR 50.55a(b)(1).

Circumferential butt welds will be radiographed and either liquid penetrant or magnetic particle examined per NB-5221 and Figure NB-3351-1. The pressure retaining base material plate, including the weld ends, will be ultrasonically examined per NB-2531.

The trunnion assemblies on either side of the package are not structural parts of the package and will be removed prior to shipment as discussed in Sections 2.5 and 7.1.2. Therefore, their design, fabrication, and inspection are not part of this SAR for application of the package design approval from the NRC.

Prior to shipment, the trunnions, i.e. the assemblies and bolts, will be removed, and SA-516, Grade 70, plugs will be placed in the 3" diameter bolt holes. The SA-516 material will have been ultrasonically inspected per NB-2531 prior to machining. In addition, after machining, the plugs will be visually inspected in the shop per NB-2582, and either magnetic particle or liquid penetrant examined per NB-2583 or NB-2584. Similarly, SA-516, Grade 70, plugs will replace the 1-1/2" bolts on the top plate after down-ending. These will have the same ultrasonic, visual, and surface examination as the trunnion bolt hole plugs. Finally, SA-516, Grade 70, plugs will be placed into the 3" diameter holes, used for placing the Low Density Cellular Concrete (LDCC) into the package after curing is complete. These plugs will also be ultrasonic, visual, and surface examined. Seal welds will be placed on the top of the plugs connecting them to the package. The seal welds will be magnetic particle or liquid penetrant examined per NB-5261.

The above operations will be performed using approved written procedures under BNFL Inc.'s NRC Approved QA Program. Accomplishment of the inspections and examinations described in Sections 8.1.1 and 8.1.2 will satisfy the requirement of 10 CFR 71.85(a).

8.1.3 Structural and Pressure Tests

10 CFR 71.85(b) states:

> "Where the maximum normal operating pressure will exceed 35 kpa (5 lb/in^2) gauge, the licensee shall test the containment system at an internal pressure at least 50 percent higher than the maximum normal operating pressure, to verify the capability of that system to maintain its structural integrity at that pressure."

Per the analysis presented in Section 3.4.4, the maximum normal operating pressure in the package is 15.0 psig at elevation 591 feet, and 17.9 psig at elevation 6684 feet which is the highest elevation throughout the proposed transportation route. This pressure is greater than the 5 psig limit stated in 10 CFR 71.85(b). Therefore, a pressure test of the package will be conducted at an internal pressure of 30 psig (greater than 1.5 x 17.9 = 26.85 psig) to verify its ability to maintain structural integrity at that pressure in compliance with 10 CFR 71.85(b).

The structural integrity of the package at 30 psig is analytically demonstrated in Appendix 2-1 as an assurance prior to performing the pressure test. The pressure test will be conducted after the package, loaded with the RV is in a horizontal position, the penetration holes on the top plate are plugged and seal welded, and prior to the second process of LDCC injection, as discussed in Section 7.1.2. To prepare the package for pressure testing, the openings at the side of the package (used for the second LDCC injection process discussed in Section 7.1.2) will be capped.

A pressure test header consisting of valves and gauges will be installed in one of the openings to regulate the pressure and to protect the package from over-pressurization. The gauges will be qualified to an accuracy of 1% or less and will be in the range of 1.5 to 4 times the test pressure of 30 psig.

Once the package is pressurized, the valves on the pressure test header will be closed, and the test will begin. The pressure will be held for a minimum of 10 minutes. Witnessing that the structural integrity of the package is maintained at 30 psig, demonstrates compliance with the requirement of 10 CFR 71.85(b).

8.1.4 Leak Tests

As discussed in Section 4.2.3, to demonstrate compliance with the 10 CFR 71.51(a)(1) requirement for the maximum allowable radioactivity release limit of 10^{-6} A_2 per hour, a leak test will be conducted. The leak rate to be used as the acceptance criterion during the test is determined based on ANSI N14.5-1997 guidelines. This leak rate is calculated to be 0.1 cm^3/sec as documented in Appendix 4-2.

A helium sniff test will be conducted on the package to determine if there is any leakage. The measured leakage shall not exceed 0.1 cm^3/sec. If the leakage exceeds this value, then the source of the leak will be determined and repaired. After the repair, the package will be leak tested again.

8.1.5 Component and Material Tests

The package is a welded steel enclosure used for the transportation and disposal of the RV. This package does not perform any active function. Tests to be performed prior to fabrication of the package (e.g., ultrasonic examination of the plates) will be specified in the fabrication specification. Since fabrication of the package will be accomplished in accordance with BNFL Inc.'s NRC Approved QA Program, verification of the materials of construction against the design requirements is also covered under that program. The containment integrity will be verified through the weld examinations, and pressure and leak tests discussed in Sections 8.1.2 and 8.1.3. Therefore, no component and materials tests are required.

8.1.6 Tests for Shielding Integrity

As discussed below, dedicated shielding tests are not required for this package. Fabrication of the package, performed in accordance with BNFL Inc.'s NRC Approved QA Program, will provide assurance that the package is constructed in compliance with its design requirements described in this SAR. The controlled process for loading the package described in Sections 7.1.1 and 7.1.2, the base material and weld examinations of Sections 8.1.1 and 8.1.2, and the pre-shipment dose rate surveys discussed in Section 7.2.2.j, will confirm the adequacy of the package design and construction for the required shielding. Because this is a single use package, the dose rate and shielding requirements are applicable only to the final configuration, and dedicated

shielding tests are not applicable to intermediate configurations (i.e., prior to emplacement of the LDCC).

8.1.7 Thermal Acceptance Tests

The analyses performed for thermal evaluation of the package in Chapter 3 have used conservative thermal properties for the materials present in the package. The package materials are capable of withstanding temperatures within its design envelope as shown in Appendix 2-1. Therefore, thermal acceptance tests are not required.

8.2 MAINTENANCE PROGRAM

The package is a single use steel container which will be used for transportation and disposal of the BRP RV. This package is a sealed enclosure with no instrumentation or operating control devices which are relied upon for maintaining and monitoring its integrity during the shipment. The initial acceptance tests and inspections described in Section 8.1, and the pre-shipment routine determinations performed in accordance with 10 CFR 71.87 criteria as detailed in Section 7.2.2 will ensure that the package complies with all applicable requirements. The procedures and instructions provided for the transportation operations as discussed in Section 7.3 will ensure safe transportation of the package. Therefore, no maintenance program is required for this package.

8.3 REFERENCES

8.3-1 NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," Lawrence Livermore National Lab., Prepared for NRC, March 1985.

8.3-2 ANSI N14.5-1997, "Leakage Tests on Packages for Shipment," American National Standard for Radioactive Materials, Approved February 5, 1998.

8.3-3 ASME Boiler and Pressure Vessel, Section III, Subsection NB, and Section V, 1995 Edition with the 1996 Addenda as modified by 10 CFR 50.55a(b)(1).