w/ 9809160245

SAFETY EVALUATION REPORT

Model No. NUHOMS®-MP187
Multi-Purpose Cask
Certificate of Compliance No. 9255
Revision No. 0

Summary

By application dated October 8, 1993, as supplemented, VECTRA Technologies, Inc. (VECTRA) (previously Pacific Nuclear Systems, Inc. (PNSI)) now Transnuclear West Inc. (TN West)) requested approval of the Model No. NUHOMS®-MP187 multi-purpose cask as a Type B(U)F package. Based on the statements and representations in the application as supplemented, and the conditions listed in the Certificate of Compliance (CoC), the staff has concluded that the Model No. NUHOMS®-MP187 package meets the requirements of 10 CFR Part 71.

References

Transnuclear West Inc., consolidated Safety Analysis Report (SAR) for the NUHOMS® -MP187 Multi-Purpose Cask, dated August 28, 1998.

Background

Pacific Nuclear Systems Inc. application dated October 8, 1993.

VECTRA Technologies, Inc. supplements dated February 28, 1995, April 10, 1996, July 24, 1997, and November 11, 1997.

Transnuclear West Inc. supplements dated January 19, 1998, March 20, 1998, May 21, 1998, and September 4, 1998.

NOTE: The section/paragraph numbering in this SER follows the Standard Review Plan format.

1 General Information Review

REVIEW OBJECTIVE

The objective of this chapter is to document that the application contains sufficient depth for consideration by staff in the licensing process by (1) an overview of relevant package information, including intended use and (2) a summary description of the packaging, operational features, and contents adequate to provide reasonable assurance that the package can meet the regulations and operating objectives.

1.5.1 General SAR Format

The application was prepared in accordance with Regulatory Guide (RG) 7.9 "Standard Format and Content for Part 71 Applications for Approval of Packaging for Radioactive Material."

1.5.2 Package Design Information

Following the receipt of the initial application for a CoC, dated October 8, 1993, an initial acceptance review was conducted. The staff determined that the application contained sufficient information to begin review.

1.5.2.1 Purpose of Application

The application was for the approval of an exclusive use spent fuel transportation package. The initial application was restricted to the use of B&W fuel from the Rancho Seco Nuclear Station.

1.5.2.2 Quality Assurance (QA) Program

As documented in NUREG-0383 Volume 3 Revision 17, VECTRA (now TN West) has an NRC-approved program. The approval covered both design, fabrication, assembly, testing, procurement, maintenance, repair, modification, and use. The approval was issued September 10, 1980, and the current expiration date is September 30, 2000.

1.5.2.3 Proposed Use/Contents

Proposed Use

The NUHOMS®-MP187 Multi-Purpose Cask (package) will be used for both on-site transfer and off-site transportation of NUHOMS® Dry Shielded Canisters (DSCs), in accordance with 10 CFR Part 72 for on-site movement and 10 CFR 71 and 49 CFR 173 for off-site transportation.

(1) Type and Form of Material:

a. Intact fuel assemblies.

b. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

c. The fuel authorized for shipment in the NUHOMS®-MP187 package is B&W 15X15 uranium oxide pressurized water reactor (PWR) fuel assemblies with a maximum initial enrichment of 3.43% by weight of ^{235}U, and a total uranium content not to exceed 466 Kg per assembly.

d. Fuel assemblies without control components shall only be shipped in the fuel-only dry shielded canister (FO-DSC).

e. Fuel assemblies with control components shall be only shipped in the fuel with control components dry shielded canister (FC-DSC).

f. The maximum burnup and minimum cooling times for the individual assemblies shall meet the requirements of Table 1. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of Section 1.5.2.3(1) (g) and (h). The maximum total allowable cask heat load is 13.5 kW.

g. The maximum assembly decay heat (including control components when present) of an individual assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

h. Control components shall be cooled for at least 8 years.

Table 1

Burnup (MWD/MTIHM)*	Required Type I Cooling Time (years)	Required Type II Cooling Time (years)	Burnup (MWD/MTIHM)*	Required Type I Cooling Time (years)	Required Type II Cooling Time (years)
0	5	5	32,000	6	10
23,200	5	5	33,000	7	10
24,000	5	6	34,000	7	11
25,000	5	6	35,000	7	11
26,000	5	7	36,000	8	13
27,000	5	7	37,000	8	14
28,000	5	8	38,000	9	15
29,000	5	8	39,000	9	16
30,000	5	8	40,000	9	17
31,000	6	9	* Megawatt Days per Metric Ton of Initial Heavy Metal		

(2) Maximum Quantity of Material per Package

a. For material described in Section 1.5.2.3(1). Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight and geometric configuration as a standard fuel assembly shall be installed in the unoccupied spaces.

b. For material described in Section 1.5.2.3(1), the approximate maximum payload (including control components when present) is 81,100 lbs.

1.5.2.4 Package Type and Model Number

USA/9255/B(U)F-85

1.5.2.5 Package Category and Maximum Activity

Package Category: Category 1

Maximum Activity:

The maximum activity of the package is controlled by the fuel authorized for shipment (see Section1.5.2.3), the maximum burnup, minimum cooling time, and maximum heat load. The fuel will have: (1) a maximum burnup of 40,000 megawatt days per metric ton of initial heavy metal (MWd/MTIHM); (2) been stored in an approved facility for a length of time sufficient to meet the thermal criteria defined below, but not less than 5 years; and (3) a cask heat load, under any conditions of use, of no more than 13.5 kW, with a maximum fuel assembly decay heat of 0.764 kW (designated as Type I) or 0.563 kW (designated as Type II).

1.5.2.6 Fabrication and Welding Criteria

In general, the licensee proposed to both design and construct the NUHOMS®-MP187 in strict compliance with the ASME Code, Section III, Division 1. However, as discussed in Section 3.3 of NUREG/CR-3019, "Recommended Welding Criteria for use in the Fabrication of Shipping Containers for Radioactive Materials" access limitations often hinder the ability of the fabricator to inspect multi-wall vessels in strict compliance with the ASME Code requirements. This is the case for two of the welds on the NUHOMS®-MP187 Multi-Purpose Cask the specifics of which are described in Section 8.2.4.2 of this document.

1.5.2.7 Transport Index and Maximum Number of Packages

Transport Index for Nuclear Criticality Control

Any number of undamaged or damaged (10 CFR 71.73) packages will remain subcritical in any arrangement with close full-water reflection and optimum interspersed hydrogenous moderation. Therefore, the transport index for the package is 0 (10 CFR 71.59b).

Maximum Number of Packages: N/A

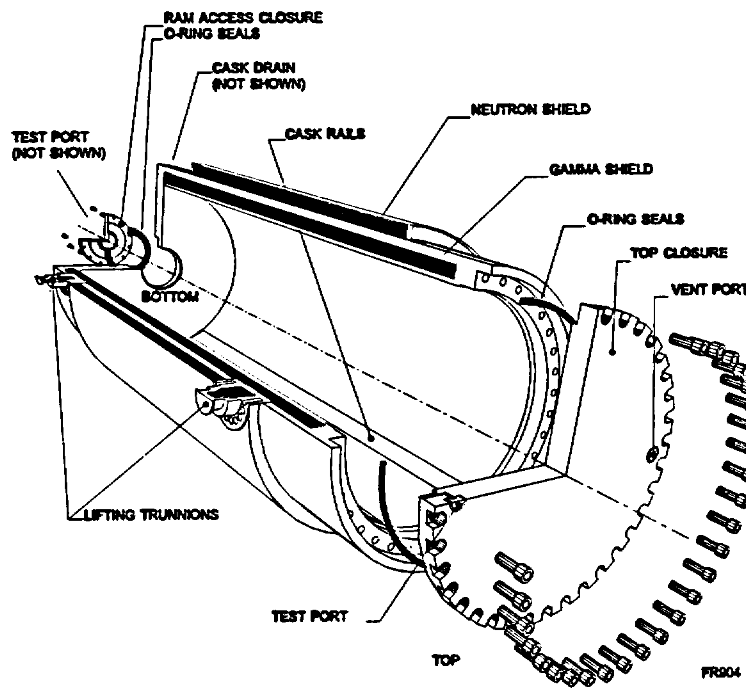
1.5.3 Package Description

1.5.3.1 Packaging

Cask

The purpose of the cask is to provide containment and shielding of the radioactive materials contained within the DSC during shipment. The cask is constructed of stainless steel and lead with a neutron shield of cementitious material. The inside cavity of the cask is a nominal 68 inches in diameter and 187-inches long. The bottom access closure is approximately 5-inches thick and 17 inches in diameter, secured by 12 1-inch diameter bolts. The top closure is approximately 6.5 inches thick and is secured by 36 2-inch diameter bolts. Both closures are sealed by redundant O-rings.

Containment is provided by a stainless steel closure lid bolted to the stainless steel cask. The containment system of the NUHOMS®-MP187 transportation cask consists of (1) the inner shell, (2) the bottom end closure plate, (3) the top closure plate, (4) the top closure inner O-ring seal, (5) the ram closure plate, (6) the ram closure inner O-ring seal, (7) the vent port screw, (8) the vent port O-ring seal, (9) the drain port screw, and (10) the drain port O-ring seal. No credit was given to the DSC as a containment boundary in the transportation safety analysis.

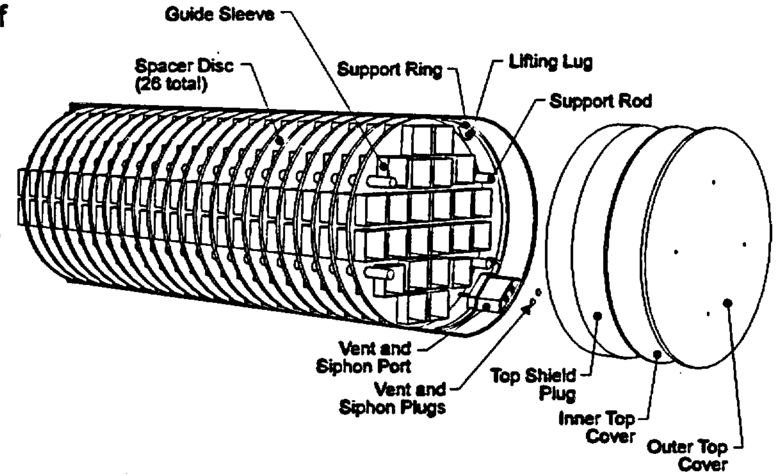


MP-187 Transport Cask

Shielding is provided by 4-inches of stainless steel, 4 inches of lead, and approximately 4.3 inches of neutron shielding. The overall length of the cask is approximately 200 inches; the outer diameter is approximately 93 inches. The maximum gross weight of the package, with impact limiters, is approximately 282,000 lbs. The total length of the package with the impact limiters attached is approximately 308 inches. Four removable trunnions (two upper and two lower) are provided for handling and lifting.

Dry Shielded Canisters (DSCs)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The approximately 5/8-inch thick shell has an outside diameter of about 67 inches and an external length of about 186 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. The basket is composed of circular spacer discs machined from thick carbon steel plates. Axial support for the DSC basket is provided by four high strength steel support rod assemblies. Carbon steel components of each DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion.



Dry Storage Canister (DSC)

On the bottom of each DSC is a grapple ring, which is used to transfer a DSC horizontally from the cask into and out of dry storage modules. Because of the nature of the fuel that is to be transported, two different types of DSCs are designed for the package. Variations in the DSC configurations are summarized below:

- Fuel-Only Dry Shielded Canister (FO-DSC)
  The FO-DSC has a cavity length of approximately 167 inches and has solid carbon steel shield plugs at each end. The FO-DSC is designed to contain up to 24 intact pressurized water reactor (PWR) spent fuel assemblies. The FO-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies.

- Fuel/Control Components Dry Shielded Canister (FC-DSC)
  The FC-DSC has an internal cavity length of approximately 173 inches to accommodate fuel with the B&W control components installed. To obtain the increased cavity length, the shield plugs are fabricated from a composite of lead and steel. The FC basket is similar to the FO-DSC except that the support rod assemblies and guide sleeves are approximately 6-inches longer. The FC-DSC is also designed to contain up to 24 intact PWR spent fuel assemblies with control components.

Impact Limiters
The impact limiter shells are fabricated from stainless steel. Within that shell are closed-cell polyurethane foam and aluminum honeycomb material. The impact limiter is attached to the cask by carbon steel impact limiter attachment bolts. Each impact limiter is bolted to the cask body through the neutron shield top and bottom support rings. The weight of each impact limiter is approximately 15,800 lbs.

Drawings
The package shall be constructed and assembled in accordance with the following TN West Drawing Numbers:

NUH-05-4000NP, Revision 7
NUHOMS®-MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 9
NUHOMS®-MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 4
NUHOMS®-MP187 Multi-Purpose Cask
Impact Limiters

NUH-05-4003, Revision 7
NUHOMS®-MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 10
NUHOMS FO-DSC & FC-DSC
Main Assembly

NUH-05-4006NP, Revision 6
NUHOMS®-MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

1.5.3.2 Operational Features

The NUHOMS®-MP187 package is not considered to be operationally complex and all operational features are readily apparent from inspection of the General Arrangement Drawings.

1.5.3.3 Contents

See Section 1.5.2.3.

1.5.4 Compliance with 10 CFR Part 71

1.5.4.1 General Requirements of 10 CFR 71.43

The applicant provided summary statements indicating that the NUHOMS®-MP187 was in compliance with the general standards for all packages. These statements were verified during the review process of the specific chapters of the SAR and found to be accurate.

Minimum Package Size

Because no dimension of the package is less than 4 inches, the package meets the requirement of 10 CFR 71.43(a) for minimum size.

Positive Closure

The package containment system is positively closed by bolted lids. With the containment closure and the presence of the impact limiters, inadvertent opening of the cask cannot occur.

Valves or Other Devices

All ports are protected against unauthorized operation and are sealed by a steel plug and an O-ring seal to retain any leakage. Thus, the package design meets the requirements of 10 CFR 71.43(e).

Continuous Venting

The package has no feature that would allow venting during transport.

1.5.4.2 Condition of Package After 10 CFR 71.71 and 10 CFR 71.73 Testing

Summary descriptions were provided within the SAR and the references were verified for the physical condition of the package subsequent to the tests specified in 10 CFR 71.71 (normal conditions of transport), and 10 CFR 71.73 (hypothetical accident conditions). These statements were verified by staff and that verification was documented within the applicable sections of this Safety Evaluation Report (SER).

1.5.4.3 Structural, Thermal, Containment, Shielding, and Criticality

Summary statements in the SAR attested to the adequacy of the package design to meet the structural, thermal, containment, shielding, and criticality requirements of 10 CFR Part 71. These statements were verified by staff and that verification was documented within the applicable sections of this SER.

1.5.4.4 Operational Procedures, Acceptance Tests, and Maintenance

A summary statement in the SAR, attesting to the adequacy of the development of the operational procedures and acceptance tests and maintenance program to ensure compliance with the requirements of 10 CFR Part 71 was made by the applicant. These statements were verified by staff and that verification was documented within the applicable sections of this SER.

1.6 EVALUATION FINDINGS

1.6.1 General SAR Format

The package has been described in sufficient detail to provide an adequate basis for its evaluation.

1.6.2 Package Design Information

Drawings provided, in the SAR, contained adequate detail allowing their evaluation by staff against the requirements of 10 CFR Part 71. Each drawing was reviewed and was found to be consistent with the text of the SAR. Further each drawing contains keys or annotation to explain and clarify information on the drawing.

1.6.3 Package Description

The application for package approval includes a reference to an approved QA program.

1.6.4 Compliance with 10 CFR Part 71

The application for package approval committed to the use of acceptable codes and standards for the package design, fabrication, assembly, testing, maintenance, and use.

The package meets the general requirements of 10 CFR 71.43(a) and 10 CFR 71.43(b).

Drawings submitted with the application (as supplemented) are adequately detailed descriptions of the package to be evaluated for compliance with 10 CFR Part 71.

2 Structural Review

REVIEW OBJECTIVE

Structural reviews are performed to ensure that the packaging design meets the acceptance criteria and requirements of 10 CFR Part 71. Loads and loading combinations are reviewed for the normal transport conditions and the hypothetical accident conditions specified in 10 CFR Part 71. Structural materials and material specifications are reviewed and compared with acceptable codes and standards. Design assumptions, analyses, fabrications, examinations, and testing are evaluated to ensure the packaging meets the acceptance criteria of 10 CFR Part 71. Critical stresses and strains are verified by confirmatory analysis or calculations to be within allowable values of acceptable design codes and standards.

2.5.1 Description of Structural Design

2.5.1.1 Descriptive Information Including Weights and Centers of Gravity

To demonstrate that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71, the applicant performed various structural analyses, engineering evaluations, including ¼-scale drop tests of the impact limiters. The impact limiters were modeled to scale, however, the cask body and internals were simulated for size and mass only. The applicant's analyses were examined by the staff and confirmed to demonstrate that the applicant properly considered load combinations as described in RG 7.8, "Load Combinations for the Structural Analysis of Shipping Casks."

2.5.1.2 Codes and Standards

The package containment boundary components are designed in accordance with RG 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NB, "Class 1 Components," and Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits." These design criteria are consistent with the ASME Section III, Division 3 Code, and thus they are acceptable. The applicant also performed engineering evaluations to show that the package containment boundary components are not subject to brittle fracture or buckling under the test conditions specified in 10 CFR Part 71. The staff reviewed the applicant's calculations and confirmed that they demonstrated, with reasonable assurance, that they meet the requirements of the regulations.

The outer (structural) shell of the cask is not part of the containment boundary but is conservatively evaluated and constructed to the requirements of Subsection NB of the ASME Code. The neutron shield jacket assembly and cask trunnion assemblies are designed to meet the requirements of Subsection NF of the ASME Code for Class 1 components.

The cask basket structure consists of spacer discs, support rods/support plates, and guide sleeves and is designed to Subsection NG of the ASME Code. Buckling of the components of the basket structure was evaluated in accordance with Subsection NF and Appendix F of the

ASME Code. The staff reviewed the applicant's calculations and confirmed that they demonstrated, with reasonable assurance, that they meet the requirements of the regulations.

The applicant performed crush tests on ¼-scale models of the impact limiter to measure their force-deflection characteristics. A simulated cask with ¼-scale impact limiters was subjected to 30-ft drop and 40-inch puncture tests to confirm the performances of the impact limiter. The results of the drop tests were analyzed and verified to be consistent with the assumptions made by the applicant in engineering analyses supporting the design of the impact limiter.

2.5.2 Material Properties

2.5.2.1 Materials and Material Specifications

The NUHOMS®-MP187 cask shell and end closures are made from both mild and high strength austenitic stainless steels. All of these steels are approved by the ASME B&PV Code for use in Class 1 components and are highly resistant to corrosion. The material specifications for the cask structural components are shown in Table 2.3-1 of the SAR. The material properties used in the structural evaluations are contained in Table 2.3-3 and the material properties for the closure bolts are contained in Table 2.3-4 of the SAR.

The DSC basket structures are fabricated primarily from mild austenitic stainless steel and low alloy carbon steel. The DSC major structural components and material specifications are listed in SAR Table 2.3-2. The material properties used in the structural analysis of the DSC are given in SAR Table 2.3-3.

The impact limiters are composed of two types of energy absorbing materials: (1) aluminum honeycomb and (2) polyurethane foam. The energy absorbing material is encased by a stainless steel shell. The crush strengths of the energy materials are to be established by bench tests, with both the acceptance tests and acceptance criteria for the materials specified in Chapter 8 of the SAR. The NUHOMS®-MP187 cask gamma shield and DSC top and bottom shield plugs are made of lead. The structural evaluations of the cask conservatively ignore any support provided by the lead; however, lead slump effects are included in the cask inner shell buckling analysis.

The solid neutron absorbing material and the neutron absorber sheets in the guide sleeves are not considered as structural materials. However, their weights are included in the NUHOMS®-MP187 cask and DSC analysis models. The mechanical properties of solid neutron absorbing material and neutron absorber sheet material are listed in Table 2.3-5 of the SAR.

2.5.2.2 Prevention of Chemical, Galvanic, or Other Reactions

The cask is constructed from stainless steel and lead shielding materials. Experience has shown that there are no significant chemical or galvanic reactions between stainless steel and lead. The DSC shell is also fabricated from stainless steel and the carbon steel components making up the DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion. Thus the staff has concluded that the materials of construction of the package are such that there will be no significant chemical, galvanic, or other reaction between the individual package components or the package components and package contents in either dry or wet conditions.

2.5.2.3 Effects of Radiation on Materials

Radiation has no known damaging effects on the packaging material properties. To ensure the package performance throughout its service life, Chapter 8 of this document describes the acceptance tests and maintenance program for the package.

2.5.3 Lifting and Tie-Down Standards for All Packages

2.5.3.1 Lifting Devices

The loaded package is designed to be lifted vertically by two removable upper trunnions. The trunnions are located at 90° and 270° azimuths of the cask. The two trunnion sleeves are welded to the cask structural shell by full-penetration weld near the top end of the cask. The removable trunnion body is then fastened to each trunnion sleeve by 18 cap screws placed at a radius of 7.5 inches. The loaded package weight for the vertical lift from the fuel pool is conservatively assumed to be 250,000 lbs distributed evenly to the two trunnions. The analysis has shown that the lifting devices can support three times the weight of the loaded package without yielding, as required by 10 CFR 71.45(a).

The lifting devices were analyzed to show that the trunnion fasteners would fail first in an overload condition, leaving the cask body intact and, therefore, not impairing the ability of the package to meet the other requirements of 10 CFR Part 71. The package is also equipped with two removable bottom trunnions for placement and tie-down to the transporter; however, they are not used for lifting.

2.5.3.2 Tie-Down Devices

The package is secured during transport by the transportation skid. The longitudinal cask transport load is transferred to the skid by a shear key, which is welded to the cask structural shell. The shear key is a welded-box structure consisting of bearing blocks and tie bars. The vertical and transverse cask transport loads are transferred to the transportation cradle by bearing on the 2-inch wide end rings of the neutron shield jacket. The transportation loads were calculated based on the heaviest configuration of the package weight of approximately 282,000 lbs.

The applicant performed analyses to show the that package design meets the tie-down device requirements of 10 CFR 71.45(b). Under excessive overloads, the welds between the shear key and the cask structural shell would fail in shears, leaving the cask body intact without impairing the ability of the package to meet the other requirements of 10 CFR Part 71.

2.5.4 General Considerations for Structural Evaluation of Packaging

2.5.4.1 Evaluation by Analysis

The structural components of the packaging (i.e., cask, neutron shield, and basket) were evaluated by hand calculations using well-developed theory or by finite element analysis using the ANSYS computer code. The SAR described clearly the assumptions, analytical models, and methods of analysis. The specific analysis performed by staff as well as their results are discussed in Sections 2.5.5 and 2.5.6 for both normal conditions of transportation and hypothetical accident conditions respectively. The staff concluded that the applicants analysis results were presented clearly and accurately and demonstrated, to the satisfaction of staff, adequate margins of safety for the structural design.

2.5.4.2 Evaluation by Test

The only packaging component evaluated by test was the impact limiter. The impact limiters are composed of aluminum honeycomb and closed-cell polyurethane foam materials and encased in a mild stainless steel shell. The applicant performed both static and dynamic tests on ¼-scale models of the impact limiter. This testing was required to demonstrate, as specified in 10 CFR Part 71, that the impact limiter will perform its intended function of absorbing the impact energy thus reducing dynamic loads on the package during the free drop conditions. The ¼-scale model impact limiter utilized for the static and dynamic test programs simulated the dimensions, fabrication details, and features of the full-size limiter required for the cask. The force-deflection curves of the impact limiter obtained by the static tests are used to validate the applicant's analytical model. The impact limiter attachments and its performance under the hypothetical 30-ft drop conditions are confirmed by the 30-ft drop tests. The results of the dynamic tests are discussed in Sections 2.5.5.7 and 2.5.6.1.

2.5.5 Normal Conditions of Transport

2.5.5.1 Heat

The thermal stress analysis of the cask for the normal heat condition was performed using an axisymmetric finite element model. The temperature distribution in the cask due to normal heat condition, evaluated in Chapter 3, was applied to the analytical model and a linear-elastic static analysis was performed with the ANSYS computer program. The same axisymmetric finite element model for thermal analysis was used for the cask internal pressure analysis. A design pressure of 50 psig, which bounds internal pressures under normal conditions (7.8 psig) and accident conditions (42.0 psig), was conservatively assumed for the cask for normal heat conditions. The 50 psig pressure loads were applied to the inner surface of the cask and a linear-elastic analysis was performed using the ANSYS program. The analysis results showed that the stresses in the cask for thermal and internal pressure loads were well below the allowable values specified in RG 7.6 and that the normal condition heat test will not adversely affect the package. The applicant also analyzed the rupture discs for the neutron shield jacket of the cask. The staff concluded that the rupture discs will maintain pressure within the neutron shield during normal operating conditions and prevent rupture of the neutron shield jacket under excessive pressure loading.

For differential thermal expansion, a design temperature of 300° F is used for the DSC inside the cask for the normal heat conditions. The analysis has shown the thermal growth of the DSC shell in the longitudinal and radial directions is less than the nominal gaps between the outer surface of the DSC and the cask inner surface. Therefore, the cask and DSC will expand freely relative to each other and no interference or thermal stresses will occur due to differential expansion under the normal heat condition.

2.5.5.2 Cold

The stresses in the cask were calculated for an ambient temperature of -40° F in still air and shade and with maximum decay heat. The analysis was performed using the ANSYS computer program and was similar to the analysis performed for the normal heat condition above. The applicant also evaluated the condition for an ambient temperature of - 40° F and no decay heat. Under this condition, the temperature for the whole cask body will be - 40° F with no thermal gradients resulting in lower thermal stresses in the cask body. The internal pressure is lower for the cold condition but the design internal pressure of 50 psig was conservatively used for the cold condition. The analysis shows that the stresses in the cask are within allowable limits and the cold condition will not adversely affect the structural performance of the package. Differential thermal contractions of the cask components were considered based on a design temperature of 150° F for the DSC inside the cask. The staff concluded that differential thermal expansion between the cask and DSC shell is controlled by the normal heat condition discussed above.

2.5.5.3 Reduced External Pressure

A decrease in external pressure to 3.5 psia will result in a net internal pressure of 11.2 psig. The cask has been evaluated for a bounding design internal pressure of 50 psig and the resulting stresses in the cask are well within allowable values.

2.5.5.4 Increased External Pressure

As in 2.5.5.3 above, an increase of external pressure to 20 psia will result in a net external pressure of 5.3 psig. The cask is analyzed for external pressure using the same finite element model as for the internal pressure analysis. The analysis shows that the cask stresses due to the 5.3 psig external pressure loads are insignificant.

2.5.5.5 Vibration

The applicant considered a 2 g vertical vibration load imposed on the cask at the location of neutron shield end support rings. With this loading, the analysis shows that the maximum stress intensity in the cask structural shell at the location of neutron shield end support rings is 6.0 ksi. The allowable primary membrane stress intensity for the cask structural shell is 20 ksi for a design temperature of 300° F. Therefore, the cask meets the normal condition stress acceptance criteria for the vibration load condition.

2.5.5.6 Water Spray

All exterior surfaces of the package are stainless steel. The water spray test will have no effect on the package performance.

2.5.5.7 Free Drop

The package is transported solely in a horizontal orientation by approved rail car, barge, or trailer. Because of the weight and size, once the package is secured on the conveyance, it will not be moved or lifted again during transport. Consequently, the package is only analyzed for a 1-ft drop in the horizontal orientation for the normal conditions of transport.

The peak g-loads resulting from a 1-ft side drop for hot (100° F) and cold (-20° F) conditions are 13.9 g and 21.0 g, respectively (SAR Table 2.10.9-3). The equivalent static loads used for the package 1-ft drop analyses are adjusted for dynamic effects by multiplying the g-loads by the appropriate maximum dynamic load factors (DLF) for the various package components (SAR Tables 2.10.10.2 through 2.10.10.4). Thus, the cask design loads are 22.9 g (hot) and 34.2 g (cold). Likewise, the static design loads for the basket structure are 16.8 g (hot) and 24.8 g (cold). The application contained no analysis for the DSC shell because the DSC shell is not considered a containment boundary and, as such, is not considered a structural component for the transportation package.

The applicant performed finite element analyses of the cask and the basket structures using the ANSYS finite element computer code and the g-loads described above. Detailed descriptions of the impact limiters and DSC basket structures are provided in the 30-ft free drop under the hypothetical accident conditions below. The impact stresses were combined with stresses from other loadings such as internal pressure loads and thermal loads as required by RG 7.8. The results showed that the combined stresses are within the allowable limits specified by the ASME Division 3 Code and RG 7.6 for normal conditions of transport. Thus, the applicant has shown that the cask will provide containment of the contents and there will be no substantial reduction in the effectiveness of the packaging under normal transport conditions.

2.5.5.8 Corner Drop

The corner drop test is not applicable because the package weight exceeds 100 kg (220 lbs) and neither wood nor fiberboard is used as a material of construction.

2.5.5.9 Compression

The compression test is not applicable because the weight of the package exceeds 500 kg (11,000 lbs).

2.5.5.10 Penetration

The exterior shells and surfaces of the package are capable of withstanding the impact forces imposed by the normal condition penetration test. No valves or relief devices could be impacted by the test.

2.5.6 Hypothetical Accident Conditions

2.5.6.1 Free Drop

The applicant's evaluation of the package for the 30-ft drop under hypothetical accident conditions included both finite element analyses and ¼-scale model tests. To minimize damage to the package, the cask is equipped with top and bottom impact limiters to absorb the impact energy of a 30-ft drop. The impact limiters are composed of aluminum honeycomb and closed-cell polyurethane foam materials encased in a stainless steel shell. The applicant performed crush tests of the aluminum honeycomb and polyurethane foam materials to establish the stress-strain characteristics during crushing. The crush tests took into account both temperature and loading-rate effects on these materials. Based on the impact limiter geometric configurations and the stress-strain characteristics of the energy absorbing materials, force-deflection relationships (curves) of the impact limiter for the various drop orientations were predicted and adjusted for temperature and dynamic loading rate effects. Static tests on the ¼-scale model of the impact limiter were performed by the applicant to confirm the validity of the analytically derived force-deflection curves.

The predicted force-deflection curves for each drop orientation were used to predict analytically the rigid-body response of the package to impact loading via the use of the SLAPDOWN computer code. The drop analyses were performed for the extreme initial ambient heat (100° F) and cold (-20° F) conditions, assuming the worst case impact limiter crush strength properties to provide bounding results. The orientations of the drop analysis performed for the package includes vertical end drop, horizontal side drop, center of gravity over the corner (CGOC) drop, and oblique drop including secondary (slap down) impacts. Since the impact limiter has square outside dimensions, the side, CGOC, and oblique drops were performed for both "flat side" and "diagonal corner" impact on the impact limiter. CGOC drops were performed with the cask axis making an angle of 72° with the horizontal target surface and oblique drops were performed for drop angles of 30° and 60° with the horizontal target surface. Primary impact results for 30-ft drops are summarized in Table 2.10.9-1 and secondary impact results are summarized in Table 2.10.9-2 of the SAR. The resulting deceleration and displacement responses predicted by analysis were then validated by 30-ft drop impact tests. The drop tests were performed on a ¼-scale model of the impact limiters simulating the dimensions, fabrication details, and features of the full size impact limiter. The simulated "dummy" cask was fabricated from 22-inch diameter, 2-inch thick pipe, which was internally reinforced with welded steel plates. The pipe was then filled, with tightly packed lead shot, to ensure that the scaled size, weight, and mass moment of inertia were correctly modeled based upon the full size package. The dynamic impact test used two ¼-scale prototypical impact limiters and the test sequence was as follows:

- A 30-ft drop with the cask axis inclined at 30° to the horizontal. Primary impact was on the large flat facet on impact limiter "a" with a secondary impact (slap down) onto impact limiter "b".

- A 30-ft drop with the cask center of gravity over the corner and the cask axis inclined at 72° to the horizontal and rotated in plane so that impact would occur on a diagonal corner of impact limiter "a".

In addition to the two 30-ft drop tests, the impact limiter ¼-scale models were subjected to a side puncture (limiter "b") and an end puncture (limiter "a") event to establish the behavior of the honeycomb and foam materials and the integrity of the impact limiter.

The experimentally measured and analytically predicted displacements and decelerations for the 30-ft drop tests are summarized in Table 2.10.11-3 of the SAR. The agreement between the test results and the analytical predictions was excellent. For secondary impact, due to the cask rigid-body rotational effects to transverse acceleration at the slap down end, the data confirmed that a DLF of approximately 1.6 exists. The test results also confirmed that the impact limiters will remain attached to the cask and that they will perform their intended function, which is to absorb impact energy during the 30-ft drop hypothetical accident condition.

The cask was evaluated for each of the postulated accident drop conditions using finite element methods and hand calculations. The stresses in the cask body due to the drop loads were reported at selected stress points. A total of 52 cask stress points were selected to encompass all of the critical stress locations. Stress points were located on sections of the top cover plate, the bottom plate, the RAM closure, and the cask inner and outer shells. The locations covered all stress transition points, quarter points, midpoints, and on both the inner and outer face at each section. The equivalent static g-loads used in the analysis are equal to, or greater than, the product of the peak accelerations (SAR Table 2.10.9-1) and the DLFs (SAR Table 2.10.10-4) and are summarized in Table 2.7.1-1 of the SAR. To determine stresses in the cask, the applicant performed linear-elastic, static analysis using the ANSYS computer code. For the end drop orientation, the cask was represented by a two-dimensional, axisymmetric finite element model. A thee-dimensional, ½-symmetry finite element model was used for other impact orientations. Impact stresses were combined with stresses produced by other loads such as internal pressure and temperature. The combined stresses were within the allowable values specified in RG 7.6 and ASME Code Section III, Division 3. Stresses in the closure bolts were evaluated in accordance with NUREG/CR-6007 and were shown to be less than the yield strength of the material. The cask shells were evaluated using ASME Code Case N-284 and shown not to buckle under 30-ft drop test conditions. The applicant performed a separate analysis to show that the maximum lead slump (1.57 inches for cold temperature end drop condition) would be less than the conservatively assumed upper bound lead slump of 2.5 inches used in the post-drop shielding evaluation in Chapter 5.

The cask contents (i.e., spent fuel assemblies) are placed inside a DSC. Each DSC consists of a shell assembly and a basket assembly. For transportation conditions, the DSC shell assembly provides biological shielding in the axial direction but no credit is taken for containment. Thus, structural evaluation of the DSC shell assembly is not required. The basket assembly is designed to support and to position the fuel assemblies and neutron absorbing materials to maintain criticality control under the accident drop conditions. There are two different basket assemblies for the package. FO and FC basket assemblies are essentially identical except that the FC support rods extended 6.0 inches further above the top spacer disc than the FO basket. Also, the FC guide sleeves have 6.5-inch long angles welded to the top corners of the guide sleeves. The extension is necessary to accommodate the additional length of the control components and to limit axial movement of the guide sleeves. Both FO and FC basket assemblies consist of 26 spacer discs, 4 support rod assemblies, and 24 guide sleeve assemblies.

The fuel weight carried by the FC basket is slightly heavier than that of the FO basket due to the added weight of the control components. Since the FO and FC basket assembly configurations are identical with the exception of the support rod length and guide sleeve extensions, stress calculations are performed for the FC basket components only. The stresses in the FC basket assembly components will bound those in the FO basket assembly components. The free-drop stress analyses are performed for the extreme initial ambient heat (100° F) and cold (-20° F) conditions, assuming worst-case impact limiter crush strength properties to provide bounding results.

The structural evaluation of the FO-DSC and FC-DSC basket spacer discs was performed using a combination of classical solutions and finite element analysis. A ¼-symmetry finite element model of the spacer disc is used for the 30-ft end drop analysis. For 30-ft drops that impacted on the large flat side of the impact limiter, the spacer disc ½-symmetry finite element model was used for stress analyses. For 30-ft drop impacts on the diagonal corner of the impact limiter, the full spacer disc finite element model is used for the stress analyses. Equivalent static loads, calculated as the peak g-load times the corresponding DLFs, were used to calculate the stresses in the spacer disc. The analyses provided in the application were linear-elastic static analysis performed using the ANSYS computer code. The results showed that the stresses in the spacer discs for the 30-ft drop impact were less than the allowable values specified in Subsection NG of the ASME Code. The applicant performed a bifurcation buckling analysis for the bounding flat side drop condition using the ANSYS computer program (the result of which was reported as an eigenvalue). The resulting eigenvalue, which equals the factor of safety against elastic buckling, was 1.64. The factor of safety specified in ASME Code, F-1331.5(a), against buckling for compressive loads is equal to Pcr / P = 1.5. Thus, the FO-DSC and FC-DSC spacer discs meet the elastic buckling acceptance criteria of the ASME Code. In addition to spacer disc buckling analysis, the staff analyzed the most heavily loaded spacer disc ligament as a beam-column for both elastic and inelastic buckling under the combined effects of axial, shears, and bending loads. The results showed that the most heavily loaded spacer disc ligament satisfied the interaction equations per ASME Code NF-3322.1(e) for elastic analysis and NF-3342.2(b) for inelastic analysis.

The basket support rod assembly consists of a 2-inch diameter rod and twenty-six 3-inch O.D. x 2.08-inch I.D. sleeves. The support rod assemblies are tensioned with a nominal 80 kip preload during basket fabrication to apply a clamping force and provide resistance to out-of-plane bending moment at each spacer disc. The support rod assembly stress analysis was shown to meet the stress acceptance criteria of F-1334.5 of the ASME Code for compression members subjected to combined axial compression and bending.

The FO-DSC and FC-DSC guide sleeves are analyzed for the 30-ft drop conditions using equivalent static loads as presented in Table 2.7.6-40 of the SAR. The connections between the guide sleeves and bottom spacer disc are not designed to withstand the 30-ft end drop loading. Thus, for the postulated 30-ft end drop, the welded connections at the bottom spacer disc will fail and the guide sleeves will bear directly on the inner surface of the DSC end plug. The applicant-analyzed stresses in the guide sleeve, resulting from a bounding equivalent static loading of 43 g in an end drop, were small compared to the allowable stress. The analysis demonstrated to the satisfaction of staff that the most severe loading for the guide sleeves was the cold flat side slap down drop for which the maximum tangential equivalent static acceleration is 91.7 g at 80 inches from the package center of gravity. Therefore, a design basis tangential acceleration load of 95 g was conservatively used for the guide sleeve slap down drop analysis. The applicant performed an elastic-plastic analyses of the guide sleeve finite element model in accordance with NG-3228.1(b) of the ASME Code. The analysis showed that maximum elastic-plus-plastic deformation of the guide sleeve was 0.168 inches at the center of the guide sleeve bottom panel. Accordingly, this resulted in a maximum predicted permanent deformation of approximately 0.125 inches at the center of the guide sleeve bottom panel. The permanent deformation was considered in the criticality evaluation of Chapter 6.

2.5.6.2 Crush

Because this package has a mass greater than 500 kg (1100 lbs) and a density greater than water, this test is not applicable.

2.5.6.3 Puncture

The package was evaluated for the 40-inch puncture test under the hypothetical accident conditions. The puncture pin was conservatively assumed to impinge directly on (1) cask bottom end, either at the center of the ram closure plate or near the ram closure plate seal region; (2) cask top end, either at the center of the top closure plate or near the top closure plate seal region; and (3) cask outer shell, at the center of gravity of the cask.

The test, vent, and drain ports are small and located in solid steel forgings. They are protected by the impact limiter and a solid steel plug. Closure of the ports is accomplished by a ¾-inch diameter bolt with seals under the bolt head and tightened to a prescribed preload to maintain the seal. The steel plug is also equipped with an O-ring seal to retain any leakage. Thus, the ports are adequately protected from the puncture bar and no structural evaluation of puncture was performed.

The cask end puncture analysis is was performed by the finite element method using the ANSYS computer code. Top and bottom closure bolts are included in the finite element model as beam elements and the bolt preloads are represented by prescribing an initial strain in the beam elements. The applicant performed both elastic and plastic analyses for the top and bottom end puncture conditions. The analyses were performed for both cold and hot conditions and consisted of (1) linear-elastic static analysis to determine the maximum stresses in the cask, and (2) plastic analysis, assuming classical bi-linear kinematic hardening, to determine the maximum deformations in the cask seal regions. The puncture analyses conservatively ignored the presence of the impact limiter and used the dynamic flow stress of 50,000 psi for the A36 steel puncture bar for loading. The results of the analyses showed that the stresses in the cask were within the ASME Code stress allowable for Level D service limits and that the total elastic-plus-plastic deformation of the end forgings would not result in damage to the DSC or breach of containment in the seal regions.

The cask outer shell was analyzed for a puncture side drop event in which the package center of gravity was directly above the point of impact and the neutron shield was conservatively neglected. The drop orientation was expected to cause maximum damage to the cask outer shell. The required shell thickness to prevent a through puncture was calculated using the Nelms equation for a lead-backed shell. The required thickness for puncture integrity was shown to be 1.38 inches, which is considerably less than the 2.5-inch thick outer shell of the cask. Thus, the cask shell thickness is sufficient to provide puncture integrity for the postulated puncture test condition. The primary membrane and membrane plus bending stresses in the cask structural shells due to the side puncture load were calculated assuming the cask acts as a simple beam. The cask inner and outer shells were treated as parallel beams. The resulting stresses are small compared with the ASME Code, Level D stress limits.

Cask internals, DSC, and the basket structures are protected by the cask. The equivalent static g-loads from the puncture event were much smaller than those of the 30-ft drop. Therefore, no structural evaluations are necessary for cask internals for the puncture event.

2.5.6.4 Thermal

The applicant demonstrated that the package has adequate structural integrity to withstand the 30-minute fire test. The results from the accident fire thermal analysis showed that due to the insulating properties of the neutron shielding material and the impact limiters the temperature of the DSC would remain relatively constant during a 30-minute accident fire test. Consequently, the relative thermal growth of the various package components is more uniform. Because of the gaps and clearances provided in the design, differential thermal expansion between the package components for the fire test is not a concern. The maximum cask internal pressure due to the accident fire condition is 42.0 psig. A bounding design pressure of 50 psig is used for cask stress analysis. This pressure load would produce relatively small stresses in the cask body.

The stresses in the cask due to the accident fire condition were determined by finite element analysis. The thermal stresses in the cask body result both from differential thermal expansion between the cask shells and the lead shielding and from local thermal gradients. Stresses resulting from the accident fire thermal condition are classified as secondary stresses and as such need only be evaluated in accordance with RG 7.6 for low-cycle fatigue. The maximum stress intensity in the cask body due to the fire event is much less than the stress range limits for 10 cycles from the design fatigue curves given in the ASME Code. Thus, the accident thermal requirements of RG 7.6 are satisfied.

2.5.6.5 Immersion – Fissile Material

Water in leakage is assumed for the criticality analysis of the package. Therefore, the immersion test for fissile materials is not applicable (10 CFR 71.73(c)(5)). However, immersion under 3-feet of water is equivalent to an external pressure of 1.3 psig which has no effect on cask integrity.

2.5.6.6 Immersion – All Material

The applicant performed stress analysis for an external pressure load of 284.3 psig, equivalent to a 200-meter head of water. The maximum primary membrane stress intensity in the cask due to the 200-meter immersion load is 3.1 ksi in the inner shell. The maximum membrane plus bending stress intensity in the cask due to the 200-meter immersion load is 11.7 ksi at the center of the top closure plate. These stress intensities are within allowable ASME Code limits. Consequently, an external pressure of 21 psig, equivalent to immersion under 50 feet of water, would have no significant effects on the cask body.

2.5.7 Special Requirement for Irradiated Nuclear Fuel Shipments

The applicant performed stress analysis for an external pressure load of 284.3 psig, equivalent to a 200-meter head of water. The maximum primary membrane stress intensity in the cask due to the 200-meter immersion load is 3.1 ksi in the inner shell. The maximum membrane plus bending stress intensity in the cask due to the 200-meter immersion load is 11.7 ksi at the center of the top closure plate. These stress intensities are within allowable ASME Code limits. Consequently, an external pressure of 21 psig, equivalent to immersion under 50 feet of water, would have no significant effects on the cask body.

2.5.8 Internal Pressure Test

The package maximum normal operating pressure (MNOP) is 31.5 psig. The cask was analyzed for a bounding internal pressure load of 50 psig. The resulting stress intensities in the cask body for this pressure load are small. Therefore, it can be concluded that the package containment will not yield under the 150% MNOP test pressure load and the stresses are within the allowable stress limits set by the design ASME Code.

2.6 EVALUATION FINDINGS

2.6.1 Description of Structural Design

The staff has reviewed the package structural design description and found reasonable assurance that the contents of the application meet the requirements of 10 CFR 71.31.

2.6.2 Material Properties

To the maximum credible extent, there are no significant chemical, galvanic or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry or wet environment conditions. The effects of radiation on materials were considered and package containment is constructed from materials meeting the guidelines of RGs 7.11 and 7.12. Therefore, the material properties were found to meet the requirements of 10 CFR 71.43(d).

2.6.3 Lifting and Tie-Down Standards for Package

The staff has reviewed the lifting and tie-down systems for the package and found reasonable assurance that they meet 10 CFR 71.45 standards.

2.6.4 General Considerations for Structural Evaluation of Packaging

The staff has reviewed the packaging structural evaluation and found reasonable assurance that the application meets the requirements of 10 CFR 71.35.

2.6.5 Normal Conditions of Transport

The staff has reviewed the packaging structural performance under the normal conditions of transport and found reasonable assurance that there will be no substantial reduction in the effectiveness of the packaging.

2.6.6 Hypothetical Accident Conditions

The staff has reviewed the packaging structural performance under the hypothetical accident conditions and found reasonable assurance the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

2.6.7 Special Requirement for Irradiated Nuclear Fuel Shipments

The staff has reviewed the containment structure and found reasonable assurance that it will meet the 10 CFR 71.61 requirements for irradiated nuclear fuel shipments.

2.6.8 Internal Pressure Test

The staff has reviewed the containment structure and found reasonable assurance that it will meet the 10 CFR 71.85(b) requirements for a pressure test without yielding.

3 Thermal Review

REVIEW OBJECTIVE

The objective of this review is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.5.1 Description of the Thermal Design

3.5.1.1 Packaging Design Features

The NUHOMS®-MP187 package is designed to transport up to 24 PWR fuel assemblies. The package is composed of two major components, a cask which provides for the containment of the radioactive materials under 10 CFR Part 71 and one of two DSCs which maintain the transport configuration of the spent fuel. The FC-DSC is for fuel assemblies with control components, and the FO-DSC is for fuel assemblies without control components. The DSC is a high-integrity stainless steel, welded pressure vessel that provides confinement of radioactive materials for storage under 10 CFR Part 72. Within the DSC, structural support for the PWR fuel and basket guide sleeves is provided by circular spacer plates which also act as fins. This allows for enhanced transfer of decay heat from the fuel assemblies to the inner shell of the DSC.

3.5.1.2 Codes and Standard

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant.

3.5.1.3 Content Heat Load Specification

The applicant analyzed the transportation cask for two different assembly types: Type I fuel, which has a larger radiological source term and may only be placed in the innermost four fuel cells of an FO-DSC or FC-DSC, and Type II fuel, which has a smaller radiological source term and therefore, may be placed in any fuel cell of any DSC. Each Type I fuel assembly in the package is allowed a maximum decay heat of 764 watts. Each Type II fuel assembly is allowed a maximum decay heat of 563 watts. Both fuel types are bound by a maximum 40GWd/MTU burnup and minimum 5-year cooling time. The design basis decay heat for the entire package containing 24 PWR assemblies with or without control rods is 13.5 kW. The ORIGEN-2 code was used to determine the assembly decay heat load using burnup, enrichment, and cooling time of the fuel. The method in determining heat load was reviewed and confirmed to be correct.

3.5.1.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, Tables 3.1-1, 3.4-1, and 3.4-2 of the SAR, were verified to include the impact limiters, containment vessel, seals, shielding, and neutron absorbers and were consistent with the temperatures presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions. The staff also confirmed that the summary tables contained the design temperature limits for each of the critical components for both the normal conditions of transport and hypothetical accident conditions. For the hypothetical accident conditions, the applicant accounted for the pre-fire, during-fire and post-fire component temperatures. With the exception of the impact limiters which are not critical to containment during the fire, all components remain below their material property limits. The temperatures and design temperature limit criteria for the package components were reviewed and found to be consistent throughout the SAR.

3.5.1.5 Summary Tables of Pressures in the Containment System

Summary tables of the pressure in the containment system under the normal conditions of transport and hypothetical accident conditions, Tables 3.6-6 and 3.6-7 of the SAR, were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation sections of the SAR. The design basis pressure was reported along with the MNOP and the accident condition pressure.

3.5.2 Material Properties and Component Specifications

3.5.2.1 Material Properties

The package application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. Conservative thermal emissivities were used to model the radiative heat transfer to and away from the transportation cask. The thermal properties used for the analysis of the package were appropriate for the materials specified. Additionally, the fluid properties of the surrounding air were provided in the evaluation of thermal convection parameters. These properties were appropriate for the conditions of the cask required by 10 CFR Part 71 during normal and accident conditions.

3.5.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant. All components were shown to perform without fail under normal conditions with an ambient temperature of -40^{0}F.

3.5.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified. The staff verified that the maximum allowable fuel cladding temperature of 1058^{0} F was justified and supported by the Pacific Northwest National Laboratory report, PNL-4835, which is currently a methodology accepted by the NRC staff.

3.5.3 Thermal Evaluation Methods

3.5.3.1 Evaluation by Analyses

The staff confirmed the methods used for the thermal analysis were identified and sufficiently described to permit a complete and independent verification.

The applicant used the HEATING v7.2 finite-difference code to perform the thermal evaluation of the cask. For the normal operating conditions, the applicant performed a steady-state evaluation of the entire cask. This analysis produced a maximum cladding temperature of 669°F that remains below the limit of 1058°F. The maximum seal temperature under normal conditions is 283°F, which is substantially below the extended exposure limit of 700°F. For the accident conditions, the model for normal operating conditions was modified to account for the various degrees of hypothesized damage to the impact limiter. This analysis produced a maximum cladding temperature of 790°F, which is below the limit of 1058° F. Under these conditions, the maximum seal temperature was shown to be 445°F. This seal temperature for the 30-minute fire accident is below the limit of 700°F. The HEATING v7.2 code is currently a heat transfer code accepted by the staff for spent fuel cask modeling. The input files used to generate the results of the thermal analyses are appended to the Thermal Chapter of the SAR.

3.5.3.2 Evaluation by Tests

The thermal review acceptance test required prior to the first use of the cask is described in Section 8.2.4.8.

3.5.3.3 Temperatures

See Section 3.5.6.3.

3.5.3.4 Pressures

See Section 3.5.6.3.

3.5.3.5 Thermal Stresses

The applicant demonstrated that the package has adequate structural integrity to withstand the 30-minute fire test. The results from the accident fire thermal analysis show that the temperature of the DSC remains relatively constant during the 30-minute accident fire test due to the insulating properties of the neutron shielding material and the impact limiters. Consequently, the relative thermal growth of the various package components is more uniform. Because of the gaps and clearances provided in the design, differential thermal expansion between the package components for the fire test is not a concern. The maximum cask internal pressure due to the accident fire condition is 42 psig. A bounding design pressure of 50 psig is used for cask stress analysis. This pressure load would produce relatively small stresses in the cask body.

3.5.3.6 Confirmatory Analyses

Confirmatory analyses were performed by the staff using the SCANS code and the ANSYS finite element code. Modeling the cask as a series of cylindrical shells encasing a homogenized fuel/basket mass, the staff predicted cask temperatures were in agreement with the applicant's results.

3.5.3.7 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and in analytical methods. Because of significant design margins, the staff found reasonable assurance that the applicant used appropriate considerations throughout the application.

3.5.4 Evaluation of Accessible Surface Temperature

Under normal conditions, the transportation package is enclosed by a protective screen to ensure that the accessible surface remains below a temperature of 185° F. No solar insolation was applied to the package in making this determination.

3.5.5 Thermal Evaluation under Normal Conditions of Transport

3.5.5.1 Heat

Under normal conditions, all of the materials used remain below their respective failure temperatures. The applicant performed three steady-state calculations under normal conditions of transport. These calculations provided steady-state temperature distributions for the following combined boundary conditions: (1) an ambient temperature of 100° F with solar insolation and maximum decay heat, (2) an ambient temperature of -40° F with no solar insolation and maximum decay heat, and (3) an ambient temperature of -40° F with no solar insolation and no decay heat.

The applicant modeled the cask in a cylindrical geometry having different radial regions. A longitudinal cross-section was modeled symmetrically about the axis and employed an ambient temperature of 100° F with an adiabatic boundary along the axis.

The analysis to determine the peak cladding temperatures within the DSC was performed in two steps. First, the design basis decay heat was averaged over the entire DSC as a volumetric heat density. Using a 100° F ambient condition, a steady-state thermal analysis was performed to determine the temperature distribution in the cask and DSC shell. Using the DSC shell temperatures as a boundary condition, the temperature distribution of DSC shell internals was back-calculated with a greater-than-design-basis heat load. This allowed for a conservative estimation of the maximum cladding temperatures in the package. In the case of detailed cask components and composite-type materials, effective thermal conductivities were used to simplify the modeling and analysis. The design basis decay heat used was 764 W per assembly.

3.5.5.2 Cold

With no decay heat and an ambient temperature of -40° F, the entire package will maintain a steady-state temperature of -40 F. Cask components, including the containment system seals, would not be adversely affected by this low temperature.

3.5.6 Thermal Evaluation under Hypothetical Accident Conditions

3.5.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package under hypothetical accident conditions. The applicant used the HEATING v7.2 finite-difference code to develop the model. Three different scenarios were modeled to determine the worst possible fire accident. The variations were based on the amount of damage to the impact limiter. In the first case, the impact limiter was crushed and penetrated exposing the impact absorbing material. The second case modeled the impact limiter without the enclosed impact absorbing material. The final case analyzed the package with undamaged impact limiters.

The model of the damaged impact limiter with the exposed impact absorbing material produced the highest cask temperatures. However, many of these peak temperatures were not realized until the package reached steady-state conditions after the fire was extinguished. Therefore, the accident temperatures in Table 2 below reflect the peak temperature of a specified component from the time the fire was extinguished to the time the package reached steady-state conditions. The most limiting thermal conditions experienced by the cask internals are in the period following the 30-minute fire. The post-fire transient was evaluated for a period of approximately 10 hours to observe the cooling of the package to post-fire steady-state temperatures. The peak cladding temperatures were determined using the same method as before with the normal operating conditions.

Several key assumptions were made during the accident conditions to facilitate the thermal analysis: (1) the cask was separated from the skid, (2) the neutron shield jacket surrounding the neutron shielding material was in place but punctured due to the cask drop accident, (3) the cask inner cavity remained sealed due to the integrity of the closure, (4) the neutron shield material was decomposed but still present in the neutron shield cavity, and (5) no credit was taken for the aluminum stiffeners in the neutron shield cavity. The above assumptions resulted in increased component temperatures throughout the package.

3.5.6.2 Fire Test

See Section 3.5.6.1.

3.5.6.3 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are given in Table 2. As before, the accident temperatures in the table reflect the peak temperature of a specified component from the time the fire was extinguished to the time the package reached steady-state conditions.

Table 2
Maximum Calculated Temperatures (°F)

Location	Normal Conditions	Accident Conditions
Impact Limiter	191	1375
Cask Outer Surface	207	1291
Neutron Shield	258	1284
Cavity Gas	351	536
Lead Shielding	300	506
DSC Shell	400	564
Fuel Support Structure	258	1284
Fuel Cladding	669	790

Under normal conditions, all of the materials used in the fabrication of the cask and internals remain below their respective failure temperatures. None of the temperatures except for the neutron shielding material, the aluminum stiffeners, and the impact absorbing materials exceeded the failure temperatures. There was no lead melting and the containment seals were not compromised.

The applicant calculated the MNOP, assuming that 100% of the fuel rods fail and that 30% of the gaseous fission products are available for release. The total gas volume considered the gaseous fission products, the helium fill gas, and the cavity back-fill gas. The gaseous fission products were based upon a fuel burnup of 40 GWd/MTU.

The average gas temperature was calculated to be 351°F. Based on this gas temperature, the MNOP was determined to be 31.5 psig. The maximum pressure under hypothetical accident conditions is 42 psig, based on the average cavity gas temperature of 536°F.

3.5.6.4 Maximum Thermal Stresses

The stresses in the cask due to the accident fire condition are determined by finite element analysis. The thermal stresses in the cask body results both from differential thermal expansion between the cask shells and the lead shielding and from local thermal gradients. Stresses resulting from the accident fire thermal condition are classified as secondary stresses and as such need only be evaluated in accordance with RG 7.6 for low-cycle fatigue. The maximum stress intensity in the cask body due to the fire event is much less than the stress range limits for

10 cycles from the design fatigue curves given in the ASME Code. Thus, the accident thermal criteria of RG 7.6 are satisfied.

3.6 EVALUATION FINDINGS

3.6.1 Description of the Thermal Design

The staff has reviewed the package description and evaluation and found reasonable assurance that they satisfy the thermal requirements of 10 CFR Part 71.

3.6.2 Material Properties and Component Specifications

The staff has reviewed the material properties and component specifications used in the thermal evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

3.6.3 Thermal Evaluation Methods

The staff has reviewed the methods used in the thermal evaluation and found reasonable assurance that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.

3.6.4 Evaluation of Accessible Surface Temperature

The staff has reviewed the accessible surface temperatures of the package as it will be prepared for shipment and found reasonable assurance that they satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle.

3.6.5 Evaluation under Normal Conditions of Transport

The staff has reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during normal conditions of transport consistent with the tests specified in 10 CFR 71.71.

3.6.6 Evaluation under Hypothetical Accident Conditions

The staff has reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-time limits during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73.

4 Containment Review

REVIEW OBJECTIVE

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

4.5.1 Description of the Containment System

4.5.1.1 Containment Boundary

The containment system of the NUHOMS®-MP187 spent fuel transportation cask consists of the following components: (1) the inner shell, (2) the bottom end closure plate, (3) the top closure plate, (4) the top closure inner O-ring seal, (5) the ram closure plate, (6) the ram closure inner O-ring seal, (7) the vent port screw, (8) the vent port O-ring seal, (9) the drain port screw, and (10) the drain port O-ring seal. Table 3 lists all containment boundary components and their material of construction. No credit is given to the DSC as a containment boundary.

Table 3: NUHOMS®-MP187 Containment System Components

COMPONENT	MATERIAL	Item No. from NUH-05-4001, REV. 9
Inner Shell	ASME SA-240, Type 304 (middle section) and Type XM-19 (upper and lower sections)	1
Bottom End Closure	ASME SA-240, Type 304	2
Top Closure Plate	ASME SA-240, Type 304	24
Top Closure Inner O-Ring Seal	Metallic - Helicoflex HN series w/ aluminum jacket, Inconel liner, Inconel spring	25
Ram Closure Plate	ASME SA-240, Type XM-19	23
Ram Closure Inner O-Ring Seal	Metallic - Helicoflex HN series w/ aluminum jacket, Inconel liner, Inconel spring	29
Vent Port Screw	ASTM A320 GR L43, cadmium plated	36
Vent Port O-Ring Seal	Metallic - silver plated stainless steel tubular O-ring precompressed	39
Drain Port Screw	ASTM A320 GR L43, cadmium plated	36
Drain Port O-Ring Seal	Metallic - silver plated stainless steel tubular O-ring precompressed	39

The containment system is designed to be leaktight as defined in ANSI N14.5-1987 (i.e., a leakage rate of 1 x 10^{-7} std-cm^3/s or less).

All containment seals are metallic, static face O-ring seals. The top closure and ram closure plates are equipped with dual O-ring seals. The inner O-rings are the containment seals and are made with an aluminum jacket, an Inconel liner, and an Inconel spring. The outer O-rings facilitate leak testing of the inner containment O-rings and are either metallic or elastomeric (ram closure only). The seals on the vent and drain ports are silver plated, stainless steel, precompressed, tubular O-rings. All containment seals are leak tested in accordance with ANSI-N14.5 and replaced after each use.

The top closure plate is closed with 36 2-inch diameter bolts. The ram closure plate is closed with 12 1-inch diameter bolts. The vent and drain ports are each closed with a single ¾-inch diameter bolt. All containment closure bolts are cadmium-plated, SA-320, Grade L43 alloy steel. Bolt torque or preload values are specified in TN West Drawing No. NUH-05-4000, Rev. No. 7.

4.5.1.2 Codes and Standards

All containment welds are full-penetration bevel or groove welds. All containment welds are radiographically inspected in accordance with ASME Code Section III, Division 1, Subsection NB, except for the circumferential weld of the inner shell to the top forging. The inspection requirements for this weld are described in Section 8.2.4.2.

The staff has reviewed the description of the containment system, as given in Chapters 1 and 4 of the SAR. The staff found reasonable assurance that: (1) the SAR describes the containment system in sufficient detail to provide an adequate basis for its evaluation; (2) the SAR identifies established codes and standards for the containment system; (3) the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package; and (4) the containment system is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction.

4.5.1.3 Special Requirements for Damaged Spent Nuclear Fuel

Failed fuel is not considered in this review, therefore, this section is not applicable.

4.5.2 Containment Under Normal Conditions of Transport

4.5.2.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed that the pressure under both normal conditions of transport and hypothetical accident conditions would not exceed the package design pressure of 50 psig.

4.5.2.2 Containment Criteria

The containment system is designed to be leaktight (i.e., a leak rate of 1×10^{-7} std-cm^3/sec or less). Therefore, it was not necessary for the applicant to calculate the releaseable radiological source term or the maximum allowable leak rate. In accordance with ANSI 14.5, fabrication verification, periodic verification, and assembly verification leak tests will be performed to verify the leaktightness of the containment system.

4.5.2.3 Compliance with Containment Criteria

Results of the applicant's structural and thermal analyses show that the containment system remains leaktight under the tests specified in 10 CFR 71.71. Therefore, the loss or dispersal of radioactive material from the cask will be less than 10^{-6} A_2 per hour under normal conditions of transport, as required in 10 CFR 71.51(a)(1).

4.5.3 Containment Under Hypothetical Accident Conditions

4.5.3.1 Pressurization of Containment Vessel

See Section 4.5.2.1.

4.5.3.2 Containment Criteria

See Section 4.5.2.2.

4.5.3.3 Compliance with Containment Criteria

Results of the thermal analysis show that seal temperatures will remain below the seal material temperature limits during and after the 30-minute fire. Results of the structural analysis show that the cask inner shell will not buckle under accident loading conditions.

In the closure bolt analysis (Section 2.10.5 of the SAR), the applicant identified the potential for a momentary unloading of the inner seals on the top and ram closure plates during a 30-ft drop. At the instant of impact, a gap occurs at the inner seal location of the interface between the closure plate and cask body. However, the gap recloses immediately following impact. Seal integrity is maintained because each closure bolt continues to provide approximately the full preload, which is higher than the required minimum seal force. In addition, the stresses in the closure bolts remain below yield and the bolts are not over stressed.

The applicant also claimed that outer O-ring seals on the top and ram closure plates would serve as secondary containment seals during hypothetical accident conditions. The staff did not review the validity of this claim. For the reasons stated above, the staff believes that the inner O-rings provide adequate containment under hypothetical accident conditions and that it is unnecessary to give credit to the outer O-rings as a secondary containment boundary.

Overall, results of the structural and thermal analyses also showed that the containment system remained intact under the tests specified in 10 CFR 71.73. Therefore, as required in 10 CFR 71.51(a)(2), the escape of krypton would not exceed 10 A_2 in 1 week, and the escape of other radioactive materials would not exceed A_2 in 1 week under hypothetical accident conditions.

4.6 EVALUATION FINDINGS

4.6.1 Description of Containment System

The staff has reviewed the description and evaluation of the containment system and found reasonable assurance that: (1) the SAR identifies established codes and standards for the containment system; (2) the package includes 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; (3) the package is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction; (4) the package vent and drain ports are protected against unauthorized operation and are provided with an enclosure to retain any leakage (there are no valves on the containment system).

4.6.2 Containment Under Normal Conditions of Transport

The staff has reviewed the evaluation of the containment system under normal conditions of transport and found reasonable assurance that the package is designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71 (normal conditions of transport) the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system.

4.6.3 Containment Under Hypothetical Accident Conditions

The staff has reviewed the evaluation of the containment system under hypothetical accident conditions and found reasonable assurance that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for hypothetical accident conditions, with no dependence on filters or a mechanical cooling system.

In summary, the staff has reviewed the Containment Evaluation Section of the SAR and found reasonable assurance that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71 and that the package meets the containment criteria of ANSI N14.5.

5 Shielding Review

REVIEW OBJECTIVE

The objective of this review is to verify that the package design satisfies the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.5.1 Description of Shielding Design

The primary gamma ray shielding on the side of the cask is provided by 4 inches of lead (minimum tolerance thickness is 3.9 inches). The lead is sandwiched between an inner and an outer stainless steel shell, 1.25- and 2.5-inches thick, respectively. The neutron shield on the cask side consists of 4.31 inches of NS-3 with a minimum B_4C content of 2 weight percent. The neutron shield is encased in a stainless steel shell that is supported by angle braces of a steel-aluminum laminate. Rupture plugs are placed in the neutron shield casing to prevent over-pressure from gaseous decomposition of the shield material during the hypothetical fire accident. Radiation streaming on the package side can occur at the trunnion mounts, shear key way, and gap between the neutron shield and impact limiter. Special plugs are bolted on to the upper and lower trunnion mounts to shield against streaming.

Shielding at the cask ends includes 8 inches of stainless steel on the bottom and 6.5 inches of stainless steel in the cask lid. In addition, the DSC has end shield plugs made of steel (FO-DSC) or steel and lead layers (FC-DSC).

5.5.1.1 Packaging Design Features

See Section 5.5.1.3.

5.5.1.2 Codes and Standards

See Section 5.5.1.3.

5.5.1.3 Summary Table of Maximum Radiation Levels

The General Information Chapter, the Shielding Chapter, and the Drawings in the application were reviewed for completeness of information and consistency. The information, parameters, and dimensions provided are sufficient to perform a review and were consistent among the chapters and drawings. Where appropriate, standards are identified and used. The summary table of maximum radiation levels for both normal and accident conditions outside the cask shows values within the regulatory limits for an exclusive-use shipment.

5.5.2 Source Specification

The cask can contain up to 24 B&W 15x15 spent fuel assemblies with a maximum burnup of 40,000 MWd/MTU. There are two types of fuel as shown in Table 4 below. Type II fuel may be loaded anywhere in the basket. Type I fuel, which has the greater radiological source term, may only be loaded into the four interior cells of the basket, where it will be shielded by the Type II fuel in the outer cells of the basket. To prevent misloading of fuel, a loading plan is prepared and independently verified. Then the actual loading is then performed with specific verification procedures, videotaped, independently verified, and then verified a third time.

The applicant calculated neutron and gamma ray source terms for the design basis fuel assembly using the ORIGEN2 computer code. Gamma ray source strengths were also calculated for the design basis control components to be shipped with fuel in the FC-DSC. Source terms for five different parameter combinations for both Type I and Type II fuel were calculated as shown in Table 4:

Table 4
ORIGEN2 Input Cases for Shielding Evaluation

Cases	Burnup (MWd/MTU)	Initial Enrichment (w/o ^{235}U)	Type I Minimum Cooling Time (years)	Type II Minimum Cooling Time (years)
I	23,200	2.38	5	5
II	25,000	2.49	5	6
III	30,000	2.76	5	8
IV	35,000	2.99	7	11
V	40,000	3.19	9	17

The ORIGEN2 33,000 MWd/MTU PWR data library was used for burnups less than 33,000 MWd/MTU and the 50,000 MWd/MTU PWR data library for burnup cases greater than 33,000 MWd/MTU.

The source terms include radioactive isotopes in both the active fuel and the activated hardware. Source terms were developed for the four distinct regions of in-core fuel, plenum, top nozzle, and bottom nozzle. Subsequently, the plenum and top nozzle regions were combined. Due to its significant mass and in core presence during irradiation, the axial power shaping rod assembly (APSRA) was selected as the design basis control component. Gamma ray source terms for both black and gray APSRA control components with an 8-year cooling time and irradiation cycle identical to the fuel source cases were calculated. Based on simple 1-D ANISN dose rate calculations at the cask surface, the gray APSRA components were found to be the most limiting. The neutron source term included contributions from spontaneous fission and (α,n) reactions. The spectral distribution of the neutron source was assumed to be the same as that coming from the spontaneous fission of ^{244}Cm.

The energy group structure used in the dose rate calculations was that of the CASK-81 cross-section library. The ORIGEN2 gamma-ray output was converted to the CASK-81 spectral structure by assuming a logarithmic particle distribution with energy.

5.5.2.1 Gamma Source

See Section 5.5.2.2.

5.5.2.2 Neutron Source

The applicant's methods for calculating the radiation source terms were reviewed. Staff used the SAS2H computer module with the 27-group cross-section set to perform an independent calculation of the bounding fuel assembly and found acceptable agreement with the applicant's reported values.

5.5.3 Model Specification

The model of the package included irregularities in the lead shielding to represent the tapered ends of the lead column. The spacer discs were ignored in the homogenized fuel volume but explicitly modeled in the gap between the fuel region and the DSC shell. Nominal dimensions were used for the basic dose rate calculations. The effect of minimum tolerance dimensions for the shielding materials was analyzed as an incremental change to the values calculated for nominal shield dimensions. The material composition of the neutron shield was reduced by 10% in hydrogen weight and 50% in boron weight to conservatively bound the effects of any hydrogen disassociation due to aging or the hypothetical fire accident and any boron depletion during the life of the package.

The applicant calculated dose rates for both the normal conditions of transport and hypothetical accident conditions. The neutron shielding, shield jacket, and impact limiters were assumed to be lost during the hypothetical accident. The cask lead shield was assumed to slump 2.5 inches during the drop, versus the calculated value of 1.57 inches, and the accident analysis considered the effect of radiation streaming through the gap. Also, during the accident, the DSC is assumed to slide to the top of the cask cavity and the fuel is assumed to slide to the top of the DSC cavity.

The radiation source term concentration in the fuel region was assumed to have an axial distribution that follows the burnup profile to the first power for gamma rays and to the fourth power for neutrons.

The models for normal and accident conditions were reviewed and found to be consistent with the drawings and appropriate or bounding for the analyses presented in the structural and thermal analyses. The assumption of a burnup profile for the source term in the fuel was found acceptable because of the low dose rates at the package ends. Dose rate profiles along the axial length of the package were provided by the applicant and the major radiation streaming paths were included in the analysis. The package will be normally shipped by rail or barge, therefore, the occupancy requirement of 10 CFR 71.47(b)(4) was not specifically evaluated. However, due to the low dose rates (0.3 mrem/hour at 2 meters) at the package ends, the staff has reasonable assurance that the applicant will be able to demonstrate compliance with 10 CFR 71.47(b)(4) in the event that a tractor trailer or similar conveyance is used to transport the NUHOMS®-MP187 on public roads.

5.5.3.1 Configuration of Source and Shielding

See Section 5.5.3.

5.5.3.2 Material Properties

The reported material properties were reviewed and a sample of mass and atom densities was checked and found to be correct.

5.5.4 Evaluation

5.5.4.1 Methods

The applicant calculated the gamma ray and neutron dose rates using the 2-D discrete ordinates code DORT-PC, supplemented by the 3-D Monte Carlo MCNP code for selected cases. Subcritical multiplication of the neutron source by ^{235}U, ^{239}Pu, and ^{241}Pu was accounted for in the dose rate calculations. The production of secondary gamma rays from neutron capture was included in the calculations.

The DORT dose rate calculations included contributions through neutrons and gamma rays in the fuel region and gamma rays in the combined top nozzle plus plenum region and the bottom nozzle region. The basic DORT calculations used a radiological source term for all Type II fuel assemblies. The incremental contribution from a mixed load of Type I and Type II assemblies was determined by comparing full 3-D MCNP runs with Type I or Type II fuel assemblies in the center four fuel cells. The ratio of the two MCNP calculations was applied as a scaling factor to the 2-D DORT calculations

The applicant performed special calculations to treat radiation streaming around the trunnion mounts and the shear key. These two components penetrate through the neutron shield, and the trunnion mounts also penetrate through part of the lead shield. A combination of ANISN and DORT calculations was used to treat the complex configuration of these streaming paths.

After completing its basic analysis, the applicant performed an analysis to determine the effect of assuming the minimum tolerance thicknesses allowed by the drawings for the shielding materials in the cask. The applicant's calculations showed dose rates within the regulatory limits at the package surface, the vehicle surface, 2 meters from the vehicle, and at the underside of the vehicle.

5.5.4.2 Key Input and Output Data

Not applicable.

5.5.4.3 Flux-to-Dose Rate Conversion

Flux-to-dose conversion factors were taken from ANSI/ANS 6.1.1-1977.

5.5.4.4 Radiation Levels

The applicant reported the following dose rate values (maximum values are given separately for each category of gamma rays, neutrons, and total, and may not add up):

Table 5
Maximum Dose Rate
(mrem/hr)

	Location			Limit
	Side	Top End	Bottom End	
Normal Conditions:				
Package Surface				
Gamma Ray Neutron Total	18.7 193 198	0.218 0.679 0.847	0.640 1.31 1.62	1000 Exclusive Use
Vehicle Outer Surface				
Gamma Ray Neutron Total	10.7 51.3 55.6	0.218 0.679 0.847	0.640 1.31 1.62	200
2 meters				
Gamma Ray Neutron Total	3.14 7.04 9.94	0.065 0.302 0.337	0.216 0.539 0.672	10
Accident Conditions:				
1 meter				
Gamma Ray Neutron Total	180 440 480	1.7 36 36	1.8 58 59	1000

Staff reviewed the analyses, methods, and calculations reported by the applicant and performed independent calculations for selected cases. For its calculations, staff used the shielding codes in the SCALE 4.2 computer code system provided by Oak Ridge National Laboratory for an IBM work station. The SAS4 computer module with the 27n-18 couple cross-section set was used to perform independent calculations of the external dose rate on the side of the cask under normal and accident conditions. The staff found the values to be within regulatory limits.

Based on (1) its review of the information and analyses reported by the applicant, (2) its own calculations, and (3) the limits placed on maximum fuel burnup, staff believes that there is reasonable assurance that the package, with approved contents, will meet the requirements for shielding safety in 10 CFR Part 71.

5.6 EVALUATION FINDINGS

5.6.1 Description of the Shielding Design

The staff has reviewed the description of the packaging design and found reasonable assurance that it provides an adequate basis for the shielding evaluation.

5.6.2 Source Specification

The staff has reviewed the source specifications used in the shielding evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against 10 CFR Part 71 shielding requirements.

5.6.3 Model Specification

The staff has reviewed the models used in the shielding evaluation and found reasonable assurance that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package shielding design.

5.6.4 Evaluation

The staff has reviewed the external radiation levels of the package and vehicle as it will be prepared for shipment and found reasonable assurance that they satisfy 10 CFR 71.47(b) for packages transported by exclusive-use vehicle.

The staff has reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the external radiation levels will not significantly increase during normal conditions of transport consistent with the tests specified in 10 CFR 71.71.

The staff has reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the maximum external radiation level at 1 meter from the external surface of the package will not exceed 10 mSv/hr (1 rem/hr) during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73.

6 Criticality Review

REVIEW OBJECTIVE

The objective of this review is to verify that the package design satisfies the criticality safety requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The applicant performed a criticality analysis to show that the package remains subcritical under normal conditions of transport and the hypothetical accident conditions. The analysis shows that the package meets the requirements of 10 CFR Part 71.

6.5.1 Description of the Criticality Design

Criticality control is provided by mechanical and neutronic isolation of the fuel assemblies. The fuel is placed in a basket with square fuel sleeves and support disks. Adjacent fuel assemblies are neutronically separated by BORAL absorber plates held in place with a stainless steel cladding. The BORAL plates have a minimum areal poison density of 0.025 gm/cm^2 of ^{10}B. The basket maintains a spacing of 0.675 to 1.66 inches between fuel assembly sleeves. This spacing acts as a flux trap when water floods the cavity. The fuel sleeves have cutouts at the bottom to allow the volume inside and outside the fuel sleeves to flood and drain at the same rate. The structural analysis shows that the configuration of the fuel basket will be maintained under normal conditions of transport and the hypothetical accident conditions.

6.5.1.1 Packaging Design Features

See Section 6.5.1.4.

6.5.1.2 Codes and Standards

See Section 6.5.1.4.

6.5.1.3 Summary Table of Criticality Evaluations

See Section 6.5.1.4

6.5.1.4 Transport Index

The General Information Chapter, the Criticality Chapter, and the Drawings in the application were reviewed for completeness of information and consistency. The information, parameters, and dimensions provided are sufficient to perform a review and are consistent among the chapters and drawings. Where appropriate, standards are identified and used. The summary table of the criticality evaluation shows values of k_{eff} after adjustment that were below 0.95 and that support a Transport Index of 0.

6.5.2 Spent Nuclear Fuel Contents

The package may be used to transport up to 24 irradiated B & W 15x15 PWR fuel assemblies. A basket design variation allows the fuel assemblies to have control components inserted. The fuel consists of solid UO_2 pellets in fuel rods clad with zircaloy. The uranium has a maximum initial enrichment of 3.43% in ^{235}U. The cladding may not contain defects greater than pin holes and hairline cracks. The detailed contents description is given in Section 1.5.2.3.

6.5.3 General Considerations for Evaluations

6.5.3.1 Model Configuration

The applicant used a finite cask model with water reflection on the top and bottom. The applicant also assumed the most reactive fabrication dimensions for the steel guide tube thickness, poison cladding thickness, and neutron poison plate width along with the most reactive fuel assembly basket position (i.e., toward the cask center). Fuel rods were modeled explicitly and were assumed to be fully flooded with water in the fuel-to-cladding gap. Control components, if present, were ignored in the modeling. All fuel rods were modeled intact with no rods missing and the maximum enrichment throughout. No credit was taken for fuel burnup.

The models for normal and accident conditions were reviewed and found to be consistent with the drawings and in keeping with the structural and thermal analyses.

6.5.3.2 Material Properties

The reported material properties were reviewed and a sampling of mass and atom densities was checked and found to be correct. Within the applicant's analysis, the poison content of the neutron absorber plates was assumed to be 75% of the minimum acceptable boron load. This was reviewed by staff and found to be both a conservative and acceptable assumption.

6.5.3.3 Computer Codes and Cross-Section Libraries

The applicant's criticality calculations were performed using a microcomputer version of KENO-5A supplied by Oak Ridge National Laboratory and the Hansen-Roach 16-group cross-section library. Corrections for resonance and heterogeneous effects were made to the cross-section library using the VECTRA proprietary program PN-HET.

Sample input files for the KENO runs were provided and reviewed. In addition, benchmarking of the analytic method by the applicant and independent calculations by the staff provide reasonable assurance that the package meets regulatory requirements.

6.5.3.4 Demonstration of Maximum Reactivity

The applicant performed an analysis to determine the optimum moderation conditions by separately varying the moderator density inside and outside the casks. A new value of effective resonance cross-section (σ_{peff}) was calculated for ^{235}U and ^{238}U for each different value of moderator density in the cask cavity. The optimum water densities occurred with full density water inside the cask and 0.7 density water between the casks in the array for both normal and accident conditions. The highest value of k_{eff} for optimum moderation was 0.94280 after adjusting for statistical uncertainty. This maximum occurred for the accident case. An overall higher value of k_{eff} was calculated for the single-package analysis (see Subsection 6.5.4).

6.5.3.5 Confirmatory Analyses

See Evaluation Findings below.

6.5.4 Single Package Evaluation

The applicant performed an analysis to determine the effect of removing the outer shell components of the cask as required in 10 CFR 71.55(b)(3). This analysis found a maximum k_{eff} of 0.94311 when adjusted for uncertainty. This value was the highest reactivity calculated by the applicant.

6.5.4.1 Configuration

Not applicable.

6.5.4.2 Results

Not applicable.

6.5.5 Evaluation of Package Arrays Under Normal Conditions of Transport

The applicant performed calculations for normal conditions (neutron shield assumed intact) and hypothetical accident conditions (neutron shield not present). The applicant requested a Transport Index of 0 for criticality control, which requires analysis of an infinite array of packages under both normal and accident conditions. The applicant's array model placed an infinite number of casks side by side but assumed infinite water reflection at each end of the planar array of casks. The applicant did not provide justification that water reflection at the cask ends is more conservative than an infinite array of casks in the axial direction. However, this assumption was bounded by the staff's calculations as discussed below.

6.5.5.1 Configuration

Not applicable.

6.5.5.2 Results

Not applicable.

6.5.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

To assess the effect of the hypothetical accident conditions, the applicant performed an analysis to study the change in k_{eff} resulting from deformation of the fuel sleeve. The maximum value of

k_{eff} was 0.94280 (after adjustment for uncertainty) at a deflection distance of 0.18 inches.

6.5.6.1 Configuration

Not applicable.

6.5.6.2 Results

Not applicable.

6.5.7 Benchmark Evaluations

The applicant benchmarked its calculational procedure using an initial set of 134 critical experiments. In almost all cases, the code over predicted k_{eff}. However, the code consistently under predicted k_{eff} when a uranium reflector was present. Cases with a uranium reflector were determined not to be applicable to the calculations on the NUHOMS®-MP187 and were discarded. Nineteen benchmarks with borated or stainless steel absorber plates were the closest to the NUHOMS®-MP187 design. The criticality code over predicted k_{eff} for all 19 absorber benchmarks. Since negative benchmarks are not applied, the bias was set to zero and is acceptable.

6.5.7.1 Experiments and Applicability

Not applicable.

6.5.7.2 Bias Determination

Not applicable.

6.6 EVALUATION FINDINGS

Staff reviewed the analysis methods and calculations reported by the applicant and performed independent calculations of selected cases. For its calculations, staff used the CSAS/KENO Va version of the SCALE 4.2 code provided by Oak Ridge National Laboratory for an IBM work station. The new 44-group neutron cross-section set was used. The staff's model included flooding of the fuel-to-cladding gap and only 75% credit for the boron poison. To assess the effect of the applicant's array model with infinite reflection at the ends of a planar array of casks instead of an infinite array of casks in the axial direction, staff performed calculations using a model with an infinitely long fuel section and an infinite number of casks side by side. Staff also performed calculations to check for the presence of a peak in k_{eff} at very low internal moderator densities and found full-density moderation in the cask cavity maximizes k_{eff}. Finally staff performed calculations that removed the outer shell components of the cask to check compliance with 10 CFR 71.55(b)(3). The results of all staff's calculations gave an adjusted value of k_{eff} below 0.95.

Based on its review of the information and analyses reported by the applicant and its own calculations, staff has determined that there is reasonable assurance that the package design meets the criticality safety requirements in 10 CFR Part 71.

6.6.1 Description of Criticality Design

The staff has reviewed the description of the packaging design and found reasonable assurance that it provides an adequate basis for the criticality evaluation.

The staff has reviewed the summary information of the criticality design and found reasonable assurance that it indicates the package is in compliance with the requirements of 10 CFR Part 71.

6.6.2 Spent Nuclear Fuel Contents

The staff has reviewed the description of the spent nuclear fuel contents and found reasonable assurance that it provides an adequate basis for the criticality evaluation.

6.6.3 General Considerations for Evaluations

The staff has reviewed the criticality description and evaluation of the package and found reasonable assurance that it addresses the criticality safety requirements of 10 CFR Part 71.

6.6.4 Single Package Evaluation

The staff has reviewed the criticality evaluation of a single package and found reasonable assurance that it is subcritical under the most reactive credible conditions.

6.6.5 Evaluation of Package Arrays under Normal Conditions of Transport

The staff has reviewed the criticality evaluation of the most reactive array, which is an infinite number of packages, and found reasonable assurance that it is subcritical under normal conditions of transport.

6.6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The staff has reviewed the criticality evaluation of the most reactive array, which is an infinite number of packages, and found reasonable assurance that it is subcritical under hypothetical accident conditions.

6.6.7 Benchmark Evaluations

The staff has reviewed the benchmark evaluation of the calculations and found reasonable assurance that the calculations are sufficient to determine an appropriate bias and uncertainties for the criticality evaluation of the package.

7 Operating Procedures Review

REVIEW OBJECTIVE

The objective of this review is to verify that the operating procedures comply with the requirements of 10 CFR Part 71 and ensure that the package will be operated in a manner consistent with the conditions assumed in its evaluation for approval.

The CoC has been conditioned to specify that the package shall be both prepared for shipment and operated in accordance with detailed written operating procedures to be prepared by the licensee. Procedures for preparation and operation, shall be developed in accordance the guidance presented within the application and shall include those tests and inspections detailed within the CoC.

7.5.1 Package Loading

7.5.1.1 Preparation for Loading

Because the NUHOMS®-MP187 allows for preferential loading, the greatest concern was to ensure that the applicant addressed the potential for misloading. The applicant states, "The potential of fuel misloading is essentially eliminated through the implementation of multiple procedural and administrative barriers." The staff reviewed the recommendations and agrees that the requirement for three independent verifications of each of the individual fuel assemblies prior to closure is adequate. The controls, described with the application, to ensure that each fuel assembly is loaded into a known cell location within a DSC, will contain the following steps:

- The loading plan shall be independently verified and approved.
- A fuel movement schedule shall be based upon a written loading plan. The plan shall be independently verified and approved. All fuel movements from any rack location are to be performed under controls that will ensure strict verbatim compliance with the fuel movement schedule.
- All fuel assemblies are to be videotaped and independently verified, by ID number, to match the movement schedule, prior to the placement of the shield plug.
- A third independent verification shall be performed by a senior manager. This third verification verifies that fuel in the DSC is placed per the original cask loading plan.

7.5.1.2 Loading

The loading procedures were reviewed by staff and found to contain sufficient detail to allow the licensee, as required by the CoC, to develop detailed loading procedures. However, critical requirements both for the welding and inspection of the cask body and for the leak testing of the NUHOMS®-MP187 have been made a part of the CoC. Those requirements include the following:

1) a loading plan which has been independently verified and approved by a qualified individual other than the plan developer(s) which shall include:
   (a) hold points to verify that all fuel movements are performed under strict verbatim compliance with the fuel movement schedule;
   (b) videotaping and independent verification by ID number of each fuel assembly loaded; and
   (c) a final independent verification of the fuel placement.

2) procedures requiring that before shipment the licensee shall:
   a) perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron measurement instruments are calibrated for the energy spectrum of neutrons being emitted from the package;
   b) verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87; and
   c) leak test containment vessel seals to verify a leak rate of less than 1 x10-7 standard cubic centimeters per second of helium (std-cc/sec). The leak test shall have a test sensitivity of at least 5 x 10-8 std-cc/sec and shall be conducted:
      1) before first use of each package
      2) within the 12-month period prior to each shipment
      3) after seal replacement.

3) procedures that require that the package metallic seals be replaced after each use.

4) procedures requiring that tThe DSC outer top cover plate weld be verified by either volumetric or multi-layer PT examination. If PT is used, at a minimum, it must include the root, each successive 1/4 inch weld thickness, and the final layer. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PVC Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing a retrievable record of weld integrity.

7.5.2 Package Unloading

7.5.2.1 Receipt of a Package from Carrier

Procedures for the unloading of the cask were reviewed and found to have sufficient detail to allow a licensee a basis for the development of a detailed site specific procedure for the receipt of a cask.

7.5.2.2 Preparation for Unloading

The unloading procedures were reviewed by the staff and found to be essentially the reverse of the loading procedures. They contain sufficient detail, as required by the CoC, to provide the basis for the development of detailed unloading procedures by the licensee.7.5.2.3 Contents Removal

The design of the NUHOMS®-MP187 allows for the DSC to be removed intact and if necessary transferred for interim storage or to a hot cell or pool for handling of the individual fuel assemblies. The procedures were reviewed by staff and found to contain both adequate depth and sufficient detail to allow the licensee, as required by the CoC, to develop detailed operating procedures.

7.5.3 Preparation of Empty Package for Transport

The applicant committed to prepare previously used and empty NUHOMS®-MP187 casks for shipment per the requirements of 49 CFR 173.427. This was found acceptable by staff.

7.6 EVALUATION FINDINGS

The operating procedures review resulted in the following findings:

7.6.1 Package Loading

The staff has reviewed the proposed special controls and precautions for transport, loading, and handling and any proposed special controls in case of accident or delay, and found reasonable assurance that they satisfy 10 CFR 71.35(c).

The staff has reviewed the description of the radiation survey requirements of the package exterior and found reasonable assurance that the limits specified in 10 CFR 71.47 will be met.

The staff has reviewed the description of the temperature survey requirements of the package exterior and found reasonable assurance that the limits specified in 10 CFR 71.43(g) will be met.

The staff has reviewed the description of the routine determinations for package use prior to transport, and found reasonable assurance that the requirements of 10 CFR 71.87 will be met.

The staff has reviewed the description of the special instructions (if applicable) needed to safely open a package and found reasonable assurance that the procedures for providing the special instruction to the consignee are in accordance with the requirements of 10 CFR 71.89.

7.6.2 Package Unloading

The staff has reviewed the proposed special controls and precautions for unloading and handling and found reasonable assurance that they satisfy 10 CFR 71.35(c).

7.6.3 Preparation of Empty Package for Transport

The staff has reviewed the description of the routine determinations for package use prior to transport and found reasonable assurance that the requirements of 10 CFR 71.87 will be met.

7.6.4 Other Procedures

The staff made no findings with respect to other procedures.

8 Acceptance Tests and Maintenance Program Review

REVIEW OBJECTIVE

The objectives of this review are to verify that the acceptance tests for the packaging comply with the requirements of 10 CFR Part 71 for the package design and that a maintenance program will ensure acceptable packaging performance throughout its service life.

Acceptance Tests

Section 8.1 of the application specifies a maintenance program for the package. The acceptance tests and inspections considered critical to the safe operation of the NUHOMS®-MP187 were captured within the CoC.

8.2.4.1 Visual Inspections and Measurements

The licensee has committed that the NUHOMS®-MP187 Multi-Purpose Cask materials of construction and welds shall be examined in accordance with the specifications delineated on the Packaging General Arrangement Drawing. Staff has reviewed the commitments and has reasonable assurance that the packaging will be fabricated and assembled in accordance with drawings and other requirements specified in the SAR.

8.2.4.2 Weld Inspections

In general, the licensee proposed to both design and construct the NUHOMS®-MP187 in strict compliance with the ASME Code, Section III, Division 1. However, as discussed in Section 3.3 of NUREG/CR-3019, "Recommended Welding Criteria for use in the Fabrication of Shipping Containers for Radioactive Materials" access limitations often hinder the ability of the fabricator to inspect multi-wall vessels in strict compliance with the ASME Code requirements. This is the case for two of the welds on the NUHOMS®-MP187 Multi-Purpose Cask.

The first weld is the circumferential weld between the inner shell and top forging of the cask. Because this is a containment boundary weld, the ASME Code requires a radiographic examination (RT); however, because of the weld configuration and fabrication sequence, an RT does not appear to be possible. Therefore, the applicant proposed an alternative inspection process which would allow the acceptability of the weld to be verified using a combination of ultrasonic (UT) and liquid penetrant (PT) examination. These examinations are to be done in accordance with the requirements of the ASME Code, Section III, Division 1, using the acceptance standards of NB-5330 and NB-5350 respectively. A UT in combination with a multi-layer PT, although less desirable than RT, gives reasonable assurance of the weld quality and, therefore, adequate confidence in the acceptability of the weld.

The second weld requiring relief from the ASME Code, Section III, Division 1 inspection requirements is the DSC outer top cover plate weld. As is the case with the weld discussed above, it cannot be inspected using RT. Therefore, the applicant proposed that it be verified by either UT or a multi-layer PT examination. The staff's review of the applicant's proposal resulted in the following findings: (1) because the DSC is made of a very ductile material (stainless steel), the weld can tolerate relatively large flaws; (2) if PT is used for both the root weld, each successive 1/4 inch thickness, and the final layer the maximum potential flaw size would be limited; and (3) a design stress-reduction-factor 0.8, for the weld, is prudent to address remaining uncertainties about potential effects of any potential flaw. These three findings, in conjunction with a commitment by the applicant to meet the acceptance requirements of ASME B&PVC Section III, NB-5350 for the weld provided reasonable assurance of its acceptability.

Although the DSC is not a containment boundary for transportation, the DSC final closure weld is a confinement boundary weld for storage under 10 CFR Part 72. Therefore, because PT does not generally leave a permanent record, the CoC was conditioned to require the inspection process of the final closure weld, including findings (indications) be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

8.2.4.3 Structural and Pressure Tests

The cask containment boundary shall be pressure tested prior to first use to 150% of the MNOP per the requirements of 10 CFR 71.85(b) to verify structural integrity. The cask containment maximum design pressure is 50 psig, which is greater than the MNOP of 31.5 psig calculated in Section 3.1.3. The containment vessel will be tested to 75 psig which is 150% of the design pressure. Accessible weld and material inspections will be performed after the pressure test to verify maintenance of structural integrity and absence of any permanent deformations.

8.2.4.4 Leakage Tests

The fabrication verification leak test for the inner shell shall be performed after initial fabrication, but prior to lead pour, to verify that the leak rate from the cylindrical containment shell is less than 1×10^{-7} std-cc/sec of He. A second fabrication verification leak test shall be performed on the finished cask to demonstrate a leak rate of less than 1×10^{-7} std-cc/sec of He. The results of both tests shall have a sensitivity of 5×10^{-8} std-cc/sec.

8.2.4.5 Component Tests

The maximum working load for each of the lifting trunnions (the two trunnions closest to the top of the cask) is one-half the weight of the NUHOMS®-MP187 cask, canister, and fuel, without impact limiters and lids, and including the weight of the cask cavity full of water. Each of the lifting trunnions will be load tested to 150% of this maximum working load per ANSI N14.6 - 1986, as specified in the design drawings provided in Appendix 1.3.2. Per the drawings in Appendix 1.3.2, all welds and material in the lifting load path for the trunnions shall be visually inspected for plastic deformation or cracking, visually inspected, and liquid penetrant inspected or magnetic particle inspected per ASME Boiler and Pressure Vessel Code, Section V, Article 6 and Section III, Division 1, Subsection NB, Article NB-5000, as called for in ANSI N14.6-1986. Any indication of cracking or distortion shall be recorded in a Nonconformance Report and dispositioned prior to final acceptance in accordance with the Transnuclear West QA Program.

8.2.4.6 Shielding Tests

The NUHOMS®-MP187 cask body poured-lead shielding integrity will be confirmed via gamma scanning prior to installation of the neutron shield. The scan shall use, at a maximum, a 6x6-inch grid. The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches. The neutron shield shall have a minimum thickness of 4.31 inches. Its integrity shall be confirmed through a stringent combination of fabrication process control and verification by measurement. This may be done either at first use or with a check source using, at a maximum, a 6x6-inch grid.

8.2.4.7 Neutron Absorber Tests

The neutron absorber plate's minimum acceptable areal boron content loading is 0.025 g/cm^2 B10. The minimum B10 content per unit area and the uniformity of dispersion within the sandwiched material shall be verified by testing each sheet with a sufficient sensitivity (at least to the 95/95 confidence level) to assure compliance with the drawings.

8.2.4.8 Thermal Tests

The complete cask shall be subjected to a thermal heat rejection test to demonstrate satisfactory operation of the as-built shells, top lid and shielding materials. This test may be performed without the ram closure installed. The staff reviewed the proposed methods and acceptance criteria and has reasonable assurance that they can be carried out in a satisfactory manner.

Maintenance Tests

Section 8.2 of the application specifies a maintenance program for the package. The maintenance program includes: (1) verification leak testing of the package per ANSI 14.5, (2) replacement of metallic O-rings after each use, and (3) visual inspection of various package components prior to loading and shipment.

8.3.4.1 Structural and Pressure Tests

Other than the tests required prior to first use, no structural or pressure tests are necessary to ensure continued performance of the packaging.

8.3.4.2 Leakage Tests

The metallic containment seals are to be replaced after each use and shall be tested to show a leak rate of less than 1 $\times 10^{-7}$ standard cubic centimeters per second of helium. The leak test shall have a test sensitivity of at least 5 x 10^{-8} standard cubic centimeters per second.

8.3.4.3 Component Tests

All threaded parts will be inspected after each use and annually for deformed or stripped threads. Damaged parts shall be evaluated for continued use and replaced as required. The metallic containment seals are to be replaced after each use. The impact limiters shall be visually inspected within 1 year of use for water absorption or degradation. Each impact limiter shall also be weighed at the time of inspection. If there is more than a 3% weight increase, the impact limiter shall be repaired or replaced.

8.3.4.4 Neutron Absorber Tests

After initial fabrication inspection, no further special maintenance is required.

8.3.4.5 Thermal Tests

Prior to first use, each package will undergo a thermal acceptance test to verify that its heat rejection capabilities are consistent with the thermal analysis. The thermal acceptance test will compare measured temperatures and gradients with the values calculated for normal conditions.

Evaluation Findings

The staff has reviewed the identification of the codes, standards, and provisions of the QA program applicable to maintenance of the packaging and found reasonable assurance that the requirements specified in 10 CFR 71.31(c) and 10 CFR 71.37 (b) will be met.

The staff has reviewed the description of the routine determinations for package use prior to transport and found reasonable assurance that the requirements of 10 CFR 71.87(b) and 10 CFR 71.87(g) will be met.

CONCLUSIONS

Based upon the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. NUHOMS®-MP187 package meets the requirements of 10 CFR Part 71.

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