SAFETY EVALUATION REPORT

Docket No. 71-9302
Model No. NUHOMS®-MP197 Transportation Package
Certificate of Compliance No. 9302
Revision No. 0

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SUMMARY

By application dated May 2, 2001, as supplemented by letters dated November 20, 2001, January 29, 2002, January 31, 2002, March 1, 2002, March 20, 2002, April 29, 2002, and May 16, 2002, Transnuclear Inc., (TN) requested review and approval of the NUHOMS®-MP197 Transport Packaging and its associated payload utilizing the NUHOMS®-61BT Dry Shielded Canisters (DSC's). The NUHOMS®-MP197 packaging design is very similar to the previously approved NUHOMS®-MP187 packaging, Docket No. 71-9255. The staff used NUREG-1617 "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" to conduct its review. This application, as supplemented, satisfies 10 CFR Part 71 and is therefore approved.

References

Transnuclear Inc., application dated May 2, 2001.

Transnuclear Inc., November 20, 2001, January 29, 2002, January 31, 2002, March 1, 2002, March 20, 2002, April 29, 2002, and May 16, 2002.

1.0 GENERAL INFORMATION

The NUHOMS®-MP197 Transport Package will be used to transport the NUHOMS®-61BT DSC. The -61BT DSC is designed to contain 61 intact standard boiling water reactor (BWR) fuel assemblies with or without fuel channels. The NUHOMS®-MP197 transportation package is designed for a maximum heat load of 15.86kW or 260 W/assembly. The maximum gross weight of the loaded package is 132.5 tons including a maximum payload of 21.5 tons. During normal operating conditions the maximum pressure within the DSC is 1.67 atm (9.8psig). Within the cask body the maximum normal operating pressure is 1.37 atm (5.4psig). The NUHOMS®-MP197 packaging is transported in the horizontal orientation, on a specially designed shipping frame, with the lid end facing the direction of travel. There is no forced cooling or cooling fins utilized.

The NUHOMS®-MP197 packaging consists of the following components:

> A NUHOMS®-61BT Dry Shielded Canister (DSC) consisting of a cylindrical shell with top and bottom shield plugs, inner and outer bottom closure plates, and inner and outer top cover plates. After loading, the DSC is vacuum dried and back-filled with an inert gas.
>
> A fuel basket assembly, located inside the DSC, which locates and supports the fuel assemblies, transfers heat to the DSC wall, and provides neutron absorption to satisfy nuclear criticality requirements. A basket hold down ring is installed on top of the basket, after fuel loading, to prevent axial motion of the basket within the canister.
>
> A NUHOMS®-MP197 transport cask consisting of a containment boundary, structural shell, gamma shielding material, and solid neutron shield. The containment boundary consists of a cylindrical shell, bottom end (closure) plate with a ram access penetration, top end forging ring, bottom and top cover plates (lids) with associated seals and bolts, and a vent and drain port with associated closure bolts and seals. The transport cask cavity also contains an inert gas atmosphere.
>
> Sets of removable upper and lower trunnions, bolted to the outer shell of the cask that provide support, lifting, and rotation capability for the NUHOMS®-MP197 cask.
>
> Impact limiters consisting of balsa and redwood, encased in stainless steel shells, are attached to each end of the NUHOMS®-MP197 cask during shipment. A thermal shield is provided between the bottom impact limiter and the cask to minimize heat transfer to the bottom limiter. Each limiter is held in place by twelve (12) attachment bolts.
>
> A personnel barrier is also mounted to the transport frame to prevent unauthorized access to the cask body.

Engineering drawings included in Chapter 1 of the SAR show all component parts, fabrication procedures, and applicable codes and standards. Selected materials properties are presented in Chapter 2. The staff determined that the drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the -61BT DSC and -MP197 transportation packaging.

The fabrication is largely conducted in compliance with ASME Code, Section III, Subsections NB and NG. Exceptions, to the ASME Code, for components covered by the code are listed for the NUHOMS®-MP197, the NUHOMS®-61BT DSC and DSC fuel basket. For aspects that are not in compliance with the code, justification and compensatory measures are appropriately described. Materials not covered under the code have been discussed in the SAR and in responses to staff's specific questions related to materials and methods of fabrication. Descriptions have been given for the anisotropic wood used in the impact limiters, including the purpose of bonding agents and the results of performance tests and fabrication methods that ensure adequate and reproducible performance.

The staff reviewed the materials specifications, fuel specifications, environmental conditions, and operating conditions. The staff reviewed the components of the total system, including the materials specified, their properties under applicable service conditions and their methods of fabrication, and exceptions taken to the appropriate ASME code requirements. The staff found that the components of the system are described in adequate detail, their methods of fabrication are in accordance with acceptable codes and standards, with exceptions that have been described and justified, and that the design and materials selection for all components reviewed are acceptable, appropriate, and suitable for satisfying the applicable safety requirements of this application.

The staff reviewed the environmental effects on material properties deemed important to ensure performance of the intended functions for the components of the NUHOMS®-MP197. These include the effects of radiation on properties of materials, including sealing materials and shielding materials. They also include the potential for brittle fracture of structural and other components, as appropriate to the application. The staff reviewed the locations of all elastomeric/fluorocarbon seals for potential thermal or radiation degradation due to exposure in service. The staff had no safety concerns for any of the applications of elastomerics, in relation to brittle fracture, thermal, and radiation exposure. The staff concluded that the material selections are adequate for satisfying both the requirements of the design features of the -MP197 transportation package and the applicable safety requirements associated with the transport of the-61BT DSC.

2.0 STRUCTURAL

2.1 Review Scope and Objective

The objective of this review is to verify that the structural aspects of the NUHOMS®-MP197 Transport Package design meets the acceptance criteria and requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions. Loads and loading combinations were reviewed for normal transport conditions and the hypothetical accident conditions specified in 10 CFR Part 71. Structural materials and material specifications were reviewed and compared with acceptable codes and standards for materials used in similar environments. Design assumptions, analyses, fabrication details and methods, examination and testing results were evaluated to ensure the packaging meets the acceptance criteria of 10 CFR Part 71. Critical stresses and strains resulting from the analyses were reviewed and evaluated by comparison to the allowable values of acceptable design codes and standards. Confirmatory calculations were performed for certain aspects of the design.

2.2 Description of Structural Design

2.2.1 Descriptive Information Including Weights and Centers of Gravity

The NUHOMS®-MP197 transport packaging consists of three major structural components: the -MP197 transport cask, the impact limiters, and the -61BT dry shielded canister (DSC).

The NUHOMS®-MP197 transport cask consists of a containment boundary, a structural shell, gamma shielding material, and a solid neutron shield of borated polyester resin contained in aluminum modules, all concentrically arranged around the cask. The cask containment boundary is made up of a cylindrical shell with a bottom end closure plate, top end forged ring, bottom and top cover plates (lids) with the associated seals and bolts along with the vent and drain port seal and closure bolts. The transport cask is filled with an inert gas (helium) at a pressure above atmospheric pressure. The cask interior cavity volume is a cylinder 68 inches in diameter and 197 inches long with the inner shell forming the boundary acting as the containment boundary. The outer transport cask structural body dimensions are 82 inches in diameter and 208 inches in length. With the radial neutron shield around the cask body the outer diameter becomes 91½ inches. The inner stainless steel shell of the cask which is the containment boundary has a radial thickness of 1¼ inches with a 3¼ inch radial thickness of cast lead gamma shielding against that outer surface of the containment shell. The outer stainless steel shell has a radial thickness of 2½ inches and serves as the structural shell. The cask bottom closure has a thickness of 6½ inches with a central hole of 23.88 inches in diameter for the unloading/loading ram which is then closed by a 2½ inch thick closure plate. The top closure lid has a thickness of 4½ inches. The normal internal design pressure for the cask cavity is 50 psig (maximum calculated pressure of 5.4 psig) with the normal design external pressure of 25 psig (maximum calculated pressure of 20 psig). Sets of removable upper and lower trunnions, bolted to the outer structural shell of the cask provide the support points for lifting and rotating the -MP197 cask. The trunnions are stainless steel forgings and are removed prior to the transport of the -MP197 and replaced with non-protruding plugs to provide the required crush clearance distance for the impact limiter. The cask is transported in the horizontal position.

The maximum payload of spent fuel or other permitted materials is 43,000 pounds in the transport cask and the empty weight of the transport cask with all the accessories and attachments, excluding the transport skid is 222,100 pounds. Thus the total maximum loaded transport cask weight is 265,100 pounds (266,290 pounds used in structural calculations).

The center of gravity of the loaded transport cask is 102.85 inches from the base and on the axial centerline.

During transport, impact limiters consisting of steel, balsa wood and predominantly redwood, encased in ¼ inch thick stainless steel shells, are bolted to each end of the MP197. The impact limiters on each end of the cask increase the overall length to 281¼ inches and the impact limiters diameter is 122 inches. The impact limiters are each approximately 60 inches in length and fit over approximately 24 inches of each end of the transport cask. The cylindrical cap-shaped impact limiters are subdivided into 12 cells separated by radial steel rib plates with the cell volume filled with wood materials acting as the cushion. The wood materials are layered and oriented with respect to the wood grain in the radial and longitudinal directions. The purpose of the top and bottom impact limiters is to absorb the kinetic energy from the 1 foot normal free drop and the hypothetical 30 foot accidental free drop. A thermal shield is provided between the bottom impact limiter and the cask to minimize heat transfer to the bottom limiter. For transport, a personnel barrier is mounted to the transport frame to prevent unauthorized access to the exterior of the cask body.

The NUHOMS®-61BT Dry Shielded Canister (DSC) is a welded, stainless steel pressure vessel, and is housed and transported inside the NUHOMS®-MP197 transport cask and is normally supported on all four transfer support rails that are attached to the inside of the transport cask. The -61BT DSC contains the spent fuel basket that in turn contains the spent fuel or other permitted material for the canister. The canister is also pressurized with helium. The canister is capable of accepting 61 intact standard BWR fuel assemblies; with or without fuel channels. The external diameter of the canister is 67¼ inches and has an overall length of 199.67 inches. The normal internal design pressure is 50 psig. The spent fuel basket is a welded assembly of stainless steel boxes or tubes that are separated by poison plates and are open on both ends. The fuel assemblies are supported longitudinally by the canister ends and body, not the spent fuel basket. The lateral support for the fuel assemblies is from the canister shell which is supported by the canister inner support rails. The tubes are assembled in modules of 2 by 2 and 3 by 3 units. Each of the tubes, or fuel compartment cells, has an opening of a nominal 6 inches by 6 inches cross-section. Attached to the cell walls are the poison and aluminum plates that provide for heat conduction and necessary criticality control. The only structural function they perform is to transmit compressive stress through the thickness, but they are accounted for as loads on the basket structure. A basket hold-down ring is used between the top of the fuel basket and the inside surface of the canister top shield plug to restrain the basket assembly from axial movement.

To demonstrate that the packaging had adequate structural integrity to meet the requirements of 10 CFR Part 71, the applicant performed a series of structural analyses and evaluations of the design. In addition to the analytical methods used in the design and evaluation of the transport cask system, testing was performed on the impact limiters for specific loading conditions. A series of dynamic tests were conducted on four (4) one-third scale models of the -MP197 impact limiters to evaluate the effect of the 30 foot free drop hypothetical accident loading condition and to verify the analyses that have been performed for the -MP197 and to study the behavior of the impact limiters. Also one of these test specimens was used to evaluate the effects of the puncture requirements under accident conditions.

2.2.2 Codes and Standards

The design, analysis, materials, fabrication, and inspection/testing used for the three major structural components were based almost exclusively on existing codes and standards, and where deviations from the reference documents were necessary, these have been identified and reviewed for acceptability.

The NUHOMS®-MP197 transport package components that function as a containment boundary utilize the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, for "Class 1 Components," and Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits," except as noted in the defined Code exceptions presented in Section 2.11 of the Safety Analysis Report (SAR). In addition, the containment design meets the requirements of the ASME B&PV Code, Section III, Division 3, Subsection WB which addresses "Class TP (Type B) Containment," as part of "Containment System for Storage and Transport Packagings of Spent Nuclear Fuel and High Level Radioactive Material and Waste." The edition of the ASME B&PV Code utilized for the NUHOMS®-MP197 is the 1998 Edition along with the addenda through the 1999 Addenda. The stresses from each load are categorized by the type of stress induced and the classification of the stresses as primary, secondary, etc., is completed. The allowable stresses for elements that are part of containment other than bolts are considered Level A for normal conditions and as Level D for accident conditions and are shown in Table 2-2 of the SAR. The containment bolting stress limits are shown in Table 2-3 of the SAR. The applicant's analyses were examined by the staff and confirmed to demonstrate that the applicant properly considered appropriate design criteria and load combinations as described in Regulatory Guides 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessel," and 7.8, "Load Combinations for the Structural Analysis of Shipping Casks."

The outer concentric structural stainless steel shell, the trunnions and the other structural elements not part of the containment boundary of the transport cask also utilize reference codes and standards. The outer structural shell uses the same stress limits that were used for the containment shell with the same Code exceptions. The trunnions are designed and fabricated in accordance with the requirements of ANS N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." The neutron shield shell of ¼ inch thick stainless steel is designed, fabricated, and inspected in accordance with the ASME B&PV Code, Section III, Division 1, Subsection NF, "Supports," to the maximum practical extent. This neutron shield shell is not designed to withstand all of the hypothetical accident loads. For example, the ¼ inch thick stainless steel shell may degrade during the accidental fire or from the 40 inch drop onto the puncture bar, but a bounding shielding analysis has been performed considering the complete loss of the neutron shield.

The welding procedures, welders, and weld operators are qualified in accordance with the ASME B&PV Code, Section IX, "Welding and Brazing Qualifications." Electrodes, wire, and fluxes used for welded fabrication comply with the applicable requirements of the ASME B&PV Code, Section II, Part C, "Specifications for Welding Rods, Electrodes and Filler Metals. Seal welds were examined visually or by liquid penetrant method in accordance with the ASME

B&PV Code, Section V, “Nondestructive Examination,” with the liquid penetrant examination acceptance standards being in accordance with Section III, Subsection NF, Paragraph NF-5350.

The impact limiters were designed based on engineering principles and the knowledge of the material properties of those materials that are part of the impact limiters. The design criteria established for the impact limiters was that with an impact limiter on each end of the cask body, the containment vessel and the associated non-containment elements would not exceed their respective design criteria under a 1-foot high normal drop or a 30-foot accidental drop. The impact limiters are to remain attached to the transport cask after such events. As previously noted, the design of the impact limiters was verified by ⅓ scale testing.

The NUHOMS®-61BT DSC shell, the inner top cover plate, the inner bottom cover plate, the siphon/vent block, and the siphon/vent port cover plate and associated welds are designed, fabricated, and inspected in accordance with the ASME B&PV Code, Section III, Division 1, Subsection NB, to the maximum practical extent with the exemptions noted as listed in Section 2.11 of the SAR. This is the same design basis used for the -MP197 containment boundary. The allowable stresses are based on Article NB-3200 for normal condition loads (Level A), and Appendix F for the accident condition loads (Level D). The stress limits for these conditions were provided in the SAR in Table 2-2 and were found to be acceptable.

The internal spent fuel basket and basket hold down ring are designed, fabricated, and inspected in accordance with the ASME B&PV Code, Section III, Division 1, Subsection NG, to the maximum practical extent for the normal condition loads. Under the accident load conditions, Appendix F of the ASME B&PV Code, Section III controls the design. Exceptions to the Code are as noted in Section 2.11 of the SAR. The allowable stresses for the basket structural materials are provided in Table 2-4 of the SAR.

2.3 Material Properties

2.3.1 Materials and Material Specifications

The NUHOMS®-MP197 transport cask has the inner and outer shells constructed of stainless steel and the NUHOMS®-61BT Dry Shielded Canister (DSC), including the spent fuel basket which is contained inside the canister are also constructed of stainless steel. The impact limiters enclosing shell structure as well as the neutron shield shell are constructed of stainless steel. Other significant materials include the lead fill material used in the cask composite wall, the various wood materials used inside the impact limiter steel shells, and the bolting materials. Table 2-5 of the FSAR provides the specifications of the major metallic materials used in the fabrication and testing of the -MP197. These materials are identified with a material specification number and are identified in the ASME B&PV Code, Section II, “Materials,” Part D, “Properties,” or are taken from other acceptable reference documents or sources. Table 2-5 of the SAR identifies the numerically relevant values for the key parameters as a function of temperature since the -MP197 will experience a range of temperatures. Table 2-6 of the SAR defines the calculated maximum service temperatures for the various components of the -MP197 system and also identifies the design value used for hot temperatures. Considerations for fracture toughness in cold conditions have been addressed in the design of the -MP197. All structural cask and canister components are fabricated from stainless steel and these materials do not undergo a ductile to brittle transition in the temperature range of interest, greater than -40 degrees F, that is the lower service temperature for the system. The cask lid bolts, that are carbon steel, have been evaluated for fracture toughness against the criteria of the ASME B&PV Code, Section III, Division 3, Subsection WB, Paragraph WB-2333. The testing was conducted by the applicant and lid bolts were found to comply with the Division 3 requirements.

2.3.2 Prevention of Chemical, Galvanic, or Other Reactions

The cask is constructed from stainless steel and internal lead shielding materials. Experience has shown that there are no significant chemical or galvanic reactions between stainless steel and lead in air, helium, or water environments to which the system may be exposed during fabrication, testing, loading, transport, storage, and unloading operations. Portions of the NUHOMS®-MP197 system are sealed in an air environment prior to the use of the system. These include the radial neutron shielding materials and the aluminum resin-filled boxes that are sealed between stainless steel shells, the lead shielding sealed between the stainless steel shells, wood materials sealed inside the stainless steel impact limiter, and a carbon steel shield plug sealed between the stainless steel inner and outer bottom covers of the -61BT DSC. The free volume in the sealed spaces is small so that the amount of oxygen or moisture is not sufficient to cause significant corrosion or galvanic reactions between these materials. Additional information is provided in Section 2.4.4 of the SER.

In general, materials and fabrication procedures used in the fabrication of the cask, the gamma shield and the outer neutron shield are adequately described by the applicant. In light of regulatory requirements, these were reviewed by staff. No safety significant findings related to components of the system were found for chemical, galvanic, or other reactions including the effects of radiation on materials. The materials proposed for use in NUHOMS®-MP197 are found to be acceptable for the service requirements of this system.

2.3.3 Effects of Radiation on Materials

Radiation has no significant effect on the structural materials being used in the NUHOMS®-MP197 for the design life of the system. To ensure package performance throughout its service life, Chapter 8 of this SER describes the acceptance tests and maintenance program for the package.

2.4 General Standards for All Packages

The NUHOMS®-MP197 transportation package is designed to meet the requirements of 10 CFR 71.43.

2.4.1 Minimum Dimension

The dimensions of the NUHOMS®-MP197 demonstrate that the minimum dimension of 10 cm is more than exceeded.

2.4.2 Tamper-Proof Access Features

The major access into the NUHOMS®-MP197 is through the bolted top closure lid, that during transport, is entirely covered by the impact limiter that is also bolted in place. This impact limiter also covers the test port and vent port penetrations. The impact limiter attachment bolt is fitted with a security wire seal which will indicate whether unauthorized tampering or opening of the cask has occurred. The impact limiter on the bottom of the -MP197 covers the drain and test ports as well as the cover closure plate for the ram unloading/loading access opening. NRC staff review concludes that an adequate tamper-proof feature is present.

2.4.3 Positive Closure

All access openings have positive closures in the form of bolts that prevent unintentional opening, which combined with the covered locations beneath the impact limiters make unintentional opening not possible. NRC staff review concludes there are adequate positive closure provisions for the NUHOMS®-MP197.

2.4.4 Chemical and Galvanic Reactions

In addition to presentations of selected mechanical properties, Chapter 2 presents an analysis of the expected corrosion behavior for the materials of the NUHOMS®-MP197 cask. The discussion includes the chemical, galvanic, or other reactions among the materials, the contents and the environmental conditions for loading, unloading, handling, and transport conditions. It was concluded that the wetted materials will not experience significant chemical, galvanic, or other reactions in air, helium, or water (pool or clean deionized) environment. The interior of the canister and the space between the canister and the cask is exposed to inert helium environment which precludes general or galvanic corrosion on the interior surfaces.

During transportation, service conditions are discussed for various components. The exteriors of the cask and impact limiters are exposed to ambient environmental conditions. As the exterior surfaces (with the exception of bolts and fusible plugs) are fabricated from stainless steel, the cask exterior is protected from chemical, galvanic, or other reactions during transportation. Various materials are sealed under air at the fabricator. Materials sealed between stainless steel shells are the radial neuton shielding aluminum resin boxes. Wood of the impact limiters is sealed within a stainless steel shell. A carbon steel shield plug is sealed between the stainless steel inner and outer bottom covers of the canister. There is insufficient oxygen within the small volume of available free space to promote any significant corrosion or galvanic effects. An inert neutron shielding material will not affect the aluminum boxes in which it is sealed.

The staff has reviewed the discussions, presented in the SAR, for selected mechanical properties as well as the analysis of the expected corrosion behavior and the staff agrees with the analysis presented therein and finds that materials selection are appropriate for the mechanical properties and chemical requirements for this application.

2.5 Lifting and Tie-Down Standards

2.5.1 Lifting Devices

While there are four trunnion sockets or anchor points on the NUHOMS®-MP197 (two front/top and two back/bottom), all lifting of a loaded -MP197 transport package in a vertical orientation is accomplished by lifting the package by the front/top trunnions. All trunnions are removable. The trunnions are bolted to the trunnion anchor blocks that are welded to the outer structural shell of the cask. A total of twelve 1¼ inch diameter bolts on a circular pattern with a diameter of 21 inches are used. The back/bottom trunnion pair is used in handling as a point of rotation when moving the cask between a horizontal and a vertical position or the reverse. In the horizontal position the front/top and the back/bottom trunnions all support the loaded cask. There are two designs for the front/top trunnions. One set of front/top trunnions has double shoulders and is designed as non-single failure proof for a load capability against the yield strength of the material three times that of the lifted load. These trunnions are also designed to carry five times the weight of the lifted load without exceeding the ultimate strength of the material. The other set of front/top trunnions has a single shoulder and is designated as single failure proof for a load capability of six times the lifted load without exceeding the yield strength of the material and for a load capability of ten times the lifted load without exceeding the ultimate strength of the material. The design of the cask system also provides that the failure of any lifting device under excessive load would not impair the ability of the package to meet the other package requirements. The trunnions were evaluated on the basis of a cask design weight of 260,000 pounds with a dynamic load factor of 1.1 due to the lifted loads. The design to be used at a specific site is a function of the site and the specific transfer operation details. The back/bottom trunnion is of the same design as the double-shouldered front/top trunnion, but never carries more than one-half the load of a front/top trunnion. The key structural elements relative to the adequate functioning of the trunnions include the trunnion, the trunnion bolts, the trunnion anchor block, the welds between the trunnion anchor block and the cask body outer structural shell, and the cask body outer shell itself.

Stresses were evaluated in the trunnions at the various critical sections for both trunnion designs under the design lifted loads for shear and bending with the conclusion that the stresses were within the allowable stress values of either yield or ultimate stress as appropriate. The criteria associated with the yielding of the material had the smallest acceptable stress margin to the criteria for both trunnion configurations.

The bolted connection between the trunnion and the trunnion anchor block is designed for shear, bending, and the thermal conditions that may exist at the interface. All shear loads are transmitted by bearing between the machined surfaces of the trunnion flange and the trunnion anchor block recess or socket. No shear loads are transferred to the bolts since the bolt to bolt hole clearance is set to be greater than the fit-up space between the trunnion anchor block and the trunnion flange. The bolts are designed to carry the tension resulting from the trunnion moment or bending loads as well as the thermal loads. The resulting bolt stresses for both trunnion designs under both loading conditions for each are within the yield and ultimate stress allowables.

The local stresses induced in the cask shell from the trunnions were calculated for both types of front (top) trunnions using an accepted calculational technique based on classical shell theory.

The method is outlined in a Welding Research Council Bulletin titled, “Local Stresses in Spherical and Cylindrical Shells Due to External Loadings.” Both shell membrane and shell bending stresses were calculated using this method. The results show that the computed stresses are within the allowable stresses. For example, under the loading of three times the lifted weight the ratio of calculated stress to allowable stress for the two designs are 0.74 and 0.93, both clearly less than 1.0, which is acceptable.

The weld between the trunnion anchor block and the cask body is considered a welded structural attachment under the requirements of NB-4433 of the ASME B&PV Code. The weld was considered by the staff to be a continuous partial penetration bevel weld with ¼ inch reinforcing fillets on an inside and outside weld location. The bevel on the outside weld joint is 1½ inches and 1 inch on the inside weld location. Based on the summary of calculated stresses and the associated allowable stresses, the minimum margins of safety occur at the trunnion shoulders which would dictate that if there were to be excessive loads, the damage would be to the trunnion and not to the cask itself.

The staff concludes that the requirements of 10 CFR 71.45(a) have been met.

2.5.2 Tie-Down Devices

The NUHOMS®-MP197 package during transport is secured to the transportation skid with a combination of devices. A cask shear keyway is provided for transferring the largest transportation load of 10g as required, in the longitudinal direction from the cask body to the transportation skid. The required vertical transportation load of 2g and the transverse transportation load of 5g were provided for by the support saddles and the tie-down straps.

The ANSYS® structural analysis software package was used to analyze the effect of these three load components on the loaded -MP197. The loading condition used for this case included loads from the cask and lid inertial loads, the axial and radial loads from the rear impact limiter, the loads on the inside of the cask lid from the internals, meaning the canister and its contents of spent fuel assemblies, and the loads in the form of pressures from the transport skid saddles and tie-down straps. The weight of the loaded -MP197 used in the calculations was 280,000 pounds. The results of this analysis on the shell body of the cask were defined in stresses within the shell of the cask. The results are provided in Table 2.10.1-46 of the SAR and show that all of the representative stresses are less than the yield strength of the materials.

The shear keyway is welded to a reinforcing pad plate that is welded to the cask outer shell. The keyway is formed by bearing plates and tie plates that form a box structure for the key of the transport skid to interlock into. Through this shear key system the large longitudinal loads can be transferred. The vertical and transverse cask transport loads are transferred to the transportation skid via the cradles/saddles that provide the bearing of the neutron jacket on them.

The staff review has concluded that the requirements of 10 CFR 71.45(b) have been met.

2.6 General Considerations for Structural Evaluation of Packaging

Table 2-1 of the SAR summarizes the evaluation methods used for the NUHOMS®-MP197 transportation package to demonstrate compliance with the regulatory requirements. These are grouped under the categories of numerical analysis, material tests, and model tests for the various loading conditions.

2.6.1 Evaluation by Analysis

The major structural components of the packaging such as the cask and its major elements, the canister and the spent fuel basket, were evaluated by finite element analyses (FEA) using the well documented ANSYS® (Rev. 5.6) computer code, supplemented with other specific calculations based on classical elastic or plastic theory and accepted calculational methodologies. Two separate FEA models were developed to represent the cask shell body under the various loading conditions. One was a 2-dimensional (2D) model using planar elements and the other was a 3-dimensional (3D) model using brick elements. Actual boundary conditions were considered and provisions were incorporated to reflect these conditions when the type of element was selected, as well as defining the nodal point conditions. The 2D model contained more than 2000 elements and 2200 nodes, while the 3D model contained more than 38,000 elements and 9500 nodes. A separate FEA model was developed for the bolted closures of the cask lid and the ram bolted cover plate in the cask bottom. The 3D model represented the detail of the bolts by a series of elements and linkages for the bolt head, the bolt shank and the bolt threads. Other specific areas of the cask were modeled in detail using finite elements to represent the continuum. These included areas such as the shear key bearing block, the cask top flange with the bolt holes and counterbores, as well as the ram cover plate. For the analytical evaluation of the 30-foot free drop accident condition for the packaging on an unyielding surface, a computer code known as ADOC (Acceleration due to Drop On Covers) was developed and used to predict the response of the packaging during impact. The various assumptions in the software are identified such as the cask body is assumed to be rigid and axisymmetric and all the energy absorption is in the impact limiters. The solution is based on the computer solution to the equations of motion by completing many individual problems, each solved at a different time step in the history of the motion of the packaging. Solutions are generally executed using time increments of not more than a millisecond and provide a time-history of forces, accelerations, and deformations on the packaging. Predicted results from this computer software application were also compared to scale model test results.

The spent fuel basket and canister were evaluated for the lateral impact loadings from transport conditions using four finite element models for analyses. A 3D cross-sectional finite element model of the basket and canister shell was used, a 3D finite element model was used to evaluate a section of the fuel basket under buckling loads from the lateral impact loading, a simple 3D finite element model was used to evaluate the hold down ring alignment legs, and a 2D axisymmetric finite element model of the canister was used to evaluate axial impact loads as well as the internal and external pressures. In addition, more detailed breakout finite element models were utilized from these larger models to evaluate specific loadings or effects. For example, the 3D basket and canister cross-section model detailed models were broken out into support rails Types 1 and 2 for buckling evaluations.

The SAR identified the analytical techniques being used and provided adequate description and representations of the models used in the calculations, including the necessary assumptions in the model and the evaluation method. It is clear from the descriptions of the evaluations and a review of the results and comparison with sample calculations that have been performed, that the bounding conditions were identified and used to assess the adequacy of the packaging design. The applicant fully utilized the capabilities of the ANSYS® software to review the computer models for any of the computer generated warnings that may arise from transition areas with a finite element representation of the component/structure/continuum under evaluation. These warnings usually highlight areas in the model where the resulting strains and stresses computed may not be as accurate as would be expected in other areas of the model. For this application these warnings were identified in the models and a disposition was made for each. The information provided in Appendix 2.10.10 of the SAR provides additional detail of this capability and its use in clarifying and qualifying the results from the analytical solutions.

Based on the staff review of the information provided in the SAR and specific checks of the design evaluation, the staff has concluded that the structural evaluation presented in the SAR is clearly presented and represents the expected behavior of the NUHOMS®-MP197 transport system under design conditions. The results demonstrate that there are adequate structural margins for the -MP197 to operate under the defined design conditions and is adequate to demonstrate compliance with 10 CFR Part 71.

2.6.2 Evaluation by Test

As indicated in Table 2-1 of the SAR, the only packaging component that was evaluated by test was the packaging impact limiters that were tested for the 30-foot free fall drop. However, the testing was not the exclusive method used to evaluate the impact limiters under these conditions. Analytical evaluations of the response of the impact limiters to the 30-foot drop were also performed as noted in Section 2.6.1 above for this accident condition. The dynamic test program consisted of four ⅓ scale impact limiters (two pair) that were used to conduct a series of four tests, three of which were performed for the 30-foot free fall drop test in three orientations of the test model (0-degree side drop, 20-degree slap-down, and a 90-degree end drop). The fourth test was a 40 inch free fall drop test model as an end drop on a model puncture bar. One of the three 30 foot drop tests also included testing a model impact limiter at a lower temperature after being chilled at -20°F for 48 hours. The results of the tests, comparisons to analytical results, and the final evaluation of the design are addressed in Sections 2.8.1 and 2.8.3 of this SER.

2.6.3 Fatigue Analysis of Containment Boundary

The fatigue analysis is based on the procedure described in Regulatory Guide 7.6 and the ASME Code Section III, assuming 1000 shipments and expected conditions of operation for the NUHOMS®-MP197 Transportation Package The staff reviewed the analysis presented and agrees with the conclusion that fatigue effects on the containment vessel are acceptable.

Appendix 2.10.2 presents the cask lid bolt analysis and Appendix Section 2.10.2.6 details a fatigue analysis for these bolts. The analysis assumes cask lid bolt replacement after 85 round trip shipments. The analysis assumes various loadings with the principal load being the operating preload. The staff finds no faults in the fatigue analysis for these bolts. This analysis shows that the fatigue damage to the bolts during normal conditions of transport is acceptable.

2.7 Normal Conditions of Transport

For all loading conditions the NUHOMS®-MP197 will have certain loads imposed, i.e., the dead load of the cask and the contents when in a horizontal transport orientation and the vertical handling condition. In addition, the preload conditions on the lid bolts and body when the transport cask is in normal operation represent a consistent imposed load. The effects of fabrication arising from the placement of molten lead for the gamma shielding inside the annular volume between the cask inner containment shell and the outer structural shell were considered in the design evaluation. The stress summaries in the SAR for each calculation provide a representative stress for a defined region of the cask structure. Each of these loads was analyzed as an independent load condition. Under the gravity load the maximum stress was approximately 15% of the allowable stress and occurred in the upper cask wall on the inside face. Under the preload of the lid bolts which was a condition analyzed using both the 2D and 3D models, the most significant representative stress calculated was approximately 15% of the allowable stress based on the 3D results. Since the 2D model did not account for the bolt holes and the counterbores, the 3D stress analysis results produced a higher stress, as expected. In the following discussions, unless noted, the stress results are from an ANSYS® 3D model and subsequent analysis. With respect to the induced stresses and strains from the molten lead, the cool down rate of the molten lead is controlled so that cool down occurs over approximately a one week period which results in a tensile hoop stress of about 300 psi in the lead based on the consideration of the inelastic behavior and the stress relaxation and creep properties of lead. This hoop tension causes a corresponding compressive hoop stress in the inner containment shell of approximately 800 psi. which is essentially a negligible stress condition for the shell.

2.7.1 Heat

The thermal stress analysis of the cask for the normal conditions of heat was performed using a 3D axisymmetric finite element model. The temperature distributions were determined from the thermal analysis performed and described in Section 3 of the SAR. This analysis also utilized finite elements for the thermal analysis. The thermal conditions used to define the temperature distributions in the cask materials included the maximum decay heat from the spent fuel, a steady state ambient external air temperature of 100°F, and the maximum solar heat loading. Under these conditions the highest computed stress was approximately 33% of the allowable stress of the structural material. This stress occurred on the inside face of the structural shell near the weld to the cask body flange.

Under the hot environment conditions described above, the internal pressure in the cask will rise. The thermal conditions from the hot conditions along with other assumptions were used in the computation of the maximum normal operating pressure described in Chapter 4 of the SAR. The maximum internal pressure generated in the cask under the analyzed conditions was 5.4 psig. The design pressure of the cask was 50 psig. The results from the analysis show that the highest computed stress in the cask is in the upper portion of the cask in the cask body flange of the structural shell adjacent to the periphery of the bolting region. The calculated stress was only 10% of the allowable stress.

Since the above individual loads do not exist alone for any length of time, it was necessary to analyze for a combined load case that represented all the individual loads that would be acting. These included gravity, the bolt preload of the closure lid, the thermal hot condition, and the design internal pressure. The results of this analysis indicates that the cask body stresses are within the allowable limits and that the high stress for this loading combination is in the area around the lower trunnion where the stress is approximately 34% of the allowable. All other stresses are a lower percentage of their allowable stress limit.

2.7.2 Cold

The stresses in the cask were calculated for a ambient temperature of -20°F in still air with the cask at a constant temperature with minimum decay heat and minimum solar heating. This condition was analyzed with the 3D model as an individual load case. The representative stress results indicate that the maximum stress is in the mid section of the cask wall with a stress of approximately 30% of the allowable. Another cold temperature condition analyzed was for an ambient condition of -40°F with all cask components at that temperature with an external pressure of 25 psi. For this case the highest representative stress was at the cask mid section where the stresses were approximately 26% of the allowable.

2.7.3 Reduced External Pressure

An analysis was performed for the condition of an external pressure of 3.5 psia ambient under normal transport conditions. Using the cask maximum normal operating pressure of 7.9 psig the net cask pressure under this condition is 19.1 psig. The internal pressure value used in the structural calculations was conservatively taken as 50 psig. The representative stress that was identified as the highest, under this loading combination, was located on the inner surface of the cask at the mid section with a value approximately 31% of the allowable stress.

2.7.4 Increased External Pressure

In this loading condition it is assumed there is no internal cask pressure resulting in a net external pressure load of 20 psi. A value of 25 psi, however, was conservatively used for these calculations. The thermal conditions are taken at the -20°F value. The maximum representative stress computed from these loads are approximately 27% of the allowable at the mid section of the cask.

2.7.5 Shock and Vibration

The NUHOMS®-MP197 transport cask can be utilized as a rail transport cask or as a truck transport cask and therefore must be analyzed or considered for both conditions.

For the rail shock load two cases were analyzed, one under the hot thermal conditions of 100°F with internal pressure and the other under the -20°F thermal condition with external pressure. The shock load was taken as 4.7g in each of the coordinate directions, based on information from “Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks,” SAND 76-0427, June 1977. As expected, the resulting stresses were found to be highest in the region where the trunnions attached to the cask body and in the mid section of the cask. The stresses range from 35% to 40% of the allowable stresses under the hot condition and 25% to 35% of the allowable stresses in the cold condition.

The large truck shock loads, based on recommendations from Draft 1980 ANSI Standard N14.23, were defined as 3.5g vertical, 2.3g longitudinal, and 1.6g lateral. These shock loads represent a smaller load than the rail shock loads, therefore, the rail shock load is the bounding condition. No further computations were necessary.

For the rail transport vibrational loads, the peak inertia values were based on information from SAND 76-0427 that resulted in a vertical acceleration of 0.37g and longitudinal and lateral accelerations of 0.19g. Two cases were analyzed in the same manner completed for the shock loads. One considering the hot thermal condition of 100°F with internal pressure and the other at the cold condition of -20°F with external pressure. The highest representative stress in the cold condition is in the wall, mid-height, and represents about 26% of the allowable stress. In the hot condition the maximum representative stress is in the wall, mi-height, and is 30% of the allowable stress.

The large truck vibrational loads were based on truck bed accelerations taken from the same 1980 draft ANSI standard (N14.23) used for the shock loadings. The directional loads were 0.6g in the vertical direction and 0.3g in the longitudinal and lateral directions. A scale factor of 1.6 was used since the truck vibrational loads are larger than the rail vibrational loads. The results for the same two conditions of 100°F with internal pressure and -20°F with external pressure were determined. The highest representative stress in the cold condition was in the wall, mid-height, and represents 45% to 50% of the allowable stress, and in the hot condition the maximum representative stress is in the wall, mid-height, and represents 50% to 60% of the allowable stress.

2.7.6 Water Spray

All exterior surfaces of the NUHOMS®-MP197 are metal and are therefore not subject to soaking, water infiltration, or structural degradation from the effects of water spray as long as the metal exterior surfaces remain intact.

2.7.7 Free Drop

The NUHOMS®-MP197 package is only transported in the horizontal position. Due to its size and weight, once the package is secured to the transport skid it will not be moved or lifted during transport. With the package weight greater than 33,100 pounds 10 CFR 71.71 requires maintaining structural integrity through a free drop distance of one (1) foot. In order to address other orientations of the cask that may arise during handling, the -MP197 has been analyzed for three separate orientations for the 1 foot free fall drop. These include a 1 foot drop on the lid end of the cask, the bottom end of the cask, and the side of the cask. The evaluations were made with normal conditions present including the hot temperature environment of 100°F with internal pressure (50psi) and the cold temperature environment of -20°F with an external pressure (25psi), both with impact loads considered. The maximum deceleration value for the end drops of 1 foot onto an unyielding surface was calculated to be 10g, however for a conservative stress analysis for the cask it was assumed to be a deceleration of 30g. For the side drop of 1 foot onto an unyielding surface the deceleration value was calculated to be 24g, however for a conservative stress analysis for the cask it was assumed to be a value of 30g. For analyses of these loading combinations, the inertia loads of the impact limiters and the internals of the loaded 61BT-DSC were applied as pressure loads scaled for the deceleration values. The analyses were performed using ANSYS® FEA similar to the other loading combinations previously described.

From the series of six(6) loading combinations evaluated the most highly stressed conditions in the -MP197 arose from the side drop loading under the hot environmental conditions. The highest stresses were found in the mid-wall of the cask and the stresses are on the order of 61% of the allowable stresses. The upper and lower cask wall regions have stresses in the range of 55% to 60% of the allowable stresses. The maximum stresses in the base were found to be approximately 27% of the allowable stress and the lid was found to be stressed to about 18% of the allowable stress. Tables 2.10.1-24 through 2.10.1-29 of the SAR present the representative stresses for these load combinations and the evaluations.

The free drop effects on the -61BT Dry Shielded Canister (DSC) contained in the -MP197 transport cask under the normal loading condition have also been considered. The same deceleration loads of 30g for end drops and for side drops were used for the evaluation of the -61BT DSC as were used for the -MP197 evaluation. The analyses for the evaluation were performed on the spent fuel basket, the hold down ring, the rails, the rail studs, and other relevant components as well as the surrounding -61BT DSC shell.

2.7.8 Corner Drop

The corner drop test or evaluation is not required for the NUHOMS®-MP197 under 10 CFR 71.71 since the weight of the package exceeds 220 pounds and neither wood nor fiberboard is used as a material of construction in an exposed condition. Wood is only used inside the steel shell of the impact limiters and these components have undergone separate testing and analysis.

2.7.9 Compression

Under 10 CFR 71.71, the compression test or evaluation is not required since the weight of the NUHOMS®-MP197 exceeds 11,000 pounds.

2.7.10 Penetration

The standard object for penetration testing or evaluation under normal conditions is a 13 pound steel cylinder that is 1¼ inches in diameter with a hemispherical head that free falls through a distance of 40 inches onto the transport package at the most vulnerable location. The exposed surfaces of the NUHOMS®-MP197 are stainless steel and include the neutron shield shell (0.19 inches thick) and the outer shell (0.25 inches thick) of the impact limiters. The significant materials beneath this outer skin are at least 2½ inches of the stainless steel structural shell. The design and fabrication of the -MP197 will adequately resist this penetration event and is acceptable without test in meeting the requirements of 10 CFR 71.71.

2.7.11 Summary of Evaluation Under Normal Conditions

Sections 2.6.15 and 2.6.16 of the SAR provide an overview of the state of stress in the major portions of the NUHOMS®-MP197 cask and its contained -61BT Dry Shielded Canister under the normal loading conditions by identifying the maximum stress intensities in each of those major portions. These major portions are listed below along with the computed maximum stress expressed as a % of the allowable for that particular loading case, material, location, or stress component.

COMPONENT	MAXIMUM STRESS AS % OF ALLOWABLE STRESS
CASK	
Lid	26%
Upper Flange	59%
Inner Containment Shell	38%
Outer shell	61%
Bottom	55%
CANISTER	
Primary Membrane Stress	67%
Primary Membrane + Primary Bending Stress	88%
BASKET	
Basket	92%
Rails	99%
Hold Down Ring	18%
Plate Insert Weld	46%

Based on the staff review, the NUHOMS®-MP197 with the NUHOMS®-61BT DSC contained therein have been shown to meet the normal conditions of transport under the requirements of 10 CFR 71.71 and will respond to the imposed loadings in a predictable manner and within the allowable stress limits.

2.8 Hypothetical Accident Conditions

For the accident loading conditions specified in 10 CFR 71.73, the design evaluation was accomplished in essentially the same manner as was done for the normal loading conditions, however different loads, loading combinations, and stress allowables were used. The analytical methodology employed was the same except under some conditions where a non-linear elastic-plastic analysis technique was used.

2.8.1 Free Drop

The NUHOMS®-MP197 transportation package was evaluated for a free drop from a height of 30 feet onto an unyielding surface with several orientations, including on the lid end, on the bottom end, and on the side. The design of the -MP197 packaging includes the impact limiters on each end of the cylindrical cask which are designed to absorb the impact energy from a 30 foot free drop. A total of seven (7) orientations were analytically evaluated. The inertial loads were determined from the dynamic analyses and then applied to the -MP197 components by analyzing sixteen (16) load combinations to evaluate the resulting stresses. The seven (7) orientations included an end drop on the lid, an end drop on the bottom, a side drop, a 20 degree slap-down on both the lid and the bottom ends, and the center of gravity over the corner drop on both the lid and the bottom ends. A one-third scale model test program was also performed to obtain data on the dynamic response of the impact limiters and the cask body. Comparisons were made between the analytical results and the test data. The inertial loadings developed from these analytical studies were applied in all cases to calculate the stresses in the cask and its components. These analytical stress values always exceeded the predicated values from the test data and the dynamic analyses. The resulting computed stresses can therefore be considered conservative.

The analyses for the free drop cases were executed using the same cask finite element models used for the normal conditions. The loading conditions, however, were revised to reflect the specific accident cases as described in the loading combinations identified in Tables 2-11 and 2.10.10-5 of the SAR. The resulting stresses were compared to the material allowables at the appropriate temperature under accident conditions shown in Table 2-14 of the FSAR. As in the case of the normal load conditions, the key areas of the cask were identified on a regional basis and the maximum stresses were reported for each of the free drop loading conditions. Provided below is a summary of the maximum stresses by region and loading condition as a % of the allowable stress for that identified loading case, material, location, and stress component for the 30 foot drop conditions. For this data the follow abbreviations are used: for impact surface identification, B=bottom, T=top; for thermal condition, H=hot, C=cold.

COMPONENT	**MAXIMUM STRESS AS % OF ALLOWABLE STRESS**
CASK	
<u>30' End Drop</u>	
Lid	8% B, C
Upper Wall	30% T, C
Upper Trunnion	31% T, H
Mid Wall	26% B, H
Lower Trunnion	19% B, C
Lower Wall	18% B, C
Bottom	9% B, C
<u>30' Side Drop</u>	
Lid	14% C
Upper Wall	74% C
Upper Trunnion	41% H
Mid Wall	50% H
Lower Trunnion	44% H
Lower Wall	67% H
Bottom	29% H

<u>20-Degree Slap Down</u>	
Lid	35% T, C
Upper Wall	77% T, C
Upper Trunnion	52% T, H
Mid Wall	53% B, C
Lower Trunnion	55% B, H
Lower Wall	63% B, H
Bottom	33% B, C

<u>CG Over Corner</u>	
Lid	77% T, H
Upper Wall	51% T, H
Upper Trunnion	23% T, C
Mid Wall	25% T, H
Lower Trunnion	27% B, C
Lower Wall	55% B, C
Bottom	66% B, C

In addition to the analyses of the cask shells (inner and outer), a series of analyses were performed to address the phenomenon of lead slumping under vertical loads such as generated from the 30 foot drop tests/analyses. This condition was evaluated to determine if the gamma shielding would be adversely affected by lead slump. A two-dimensional finite element model was used to evaluate the condition. The model allowed for the use of gap elements to model the interaction between the lead and the inner and outer shells of the cask. An elastic-plastic analysis approach was used due to the nature of the lead's material behavior. The loading cases analyzed showed that the maximum gap that could be developed by the load under the 30 foot drop was 0.235-inches. This gap was used in subsequent shielding calculations in Chapter 6 of the SAR.

The analyses executed for the canister contained in the -MP197 cask were completed using a two-dimensional axisymmetric finite element model for the axisymmetric loads supplemented with a three-dimensional finite element model for the analyses of side drop loads. An elastic analysis was used for all loading conditions except those of the side accident drop loads for which an elasto-plastic analysis was used. The axisymmetric model detailed the boundary conditions and coupling of the structural elements of the multi-layer top and bottom lid assemblies to the shell body as illustrated in Figures 2.10.3-64 and 2.10.3-65 of the SAR. The three-dimensional model used incorporated both the canister and the basket as shown in Figure 2.10.3-1 of the SAR. Table 2.10.3-7 of the SAR presented a summary of the canister maximum stresses as a result of the accident loading conditions for the 30-foot drop in the various orientations. Provided below is the summary of the maximum primary membrane plus primary bending stresses for the various orientations/azimuths expressed as a percentage of the allowable stress.

CANISTER

30' End Drop

24% T, H	43% C
30% T, C	52% H
31% B, H	
30% B, C	

30' Side Drop

@45-degree az.	@60-degree az.	@90-degree az.	@180-degree az.
43% C	43% C	38% C	43% C
52% H	52% H	47% H	52% H

The analyses for the basket structure and hold down ring, including the transition rails, were accomplished using the previously described finite element analyses with additional hand calculations. The analyses include elastic and elasto-plastic conditions. The summary maximum stresses are provided in Tables 2.10.3-3 and 2.10.3-4 of the SAR. Provided below is a summary of the maximum stresses for the various structural elements expressed as a % of the allowable stress for the specific conditions.

BASKET ASSEMBLY

BASKET

30' End Drop

0%-Not Loaded

30' Side Drop

@45-degree az.	@ 60-degree az.	@ 90-degree az.	@ 180-degree az.
70%	70%	63%	62%

RAILS

30' End Drop

0%-Not Loaded

30' Side Drop

@45-degree az.	@60-degree az.	@90-degree az.	@180-degree az.
47%	53%	61%	52%

HOLD DOWN RING

30' End Drop

17%

30' Side Drop

@45-degree az.	@90-degree az.
61%	81%

The test program consisted of four (4) impact limiters, or two (2) pair, that were used to create separate tests for the cask evaluation program. Each impact limiter was identified as a unique specimen. Previous as well as concurrent analytical work and past experience helped to guide the development of the test program specifics. The side drop generates the highest transverse decceleration as well as greatest potential deformation and the highest potential for the neutron shield to impact a target and suffer potential damage. The 20-degree slap-down was used since this orientation will put the highest load on the impact limiter attachment bolts and the stainless steel shell around the wood core material to study the effects of cold temperatures on the behavior. The end drop is part of the test program since this orientation results in the highest axial deceleration. In addition, this test specimen was chilled at -20°F for 24 hours prior to the test.

The scaling relationships necessary to carry out the test program were adequately developed and defined. This facilitated development of the physical properties of the model impact limiters and cask body test specimens. The impact limiter attachment block as well as the impact limiter attachment bolts were scaled for the test program using the same materials as in the prototype. The scaled impact limiters were constructed in the exact same manner as the prototypes, with radial plates creating closed cells that were filled with the wood materials oriented in the same manner as the prototype and using the same glues and fabrication techniques. The test facility had a drop pad base that weighed in excess of 250,000 pounds (the test specimen weight was 9750 pounds). The drop pad base had a steel plate securely attached to the concrete. This assembly constituted an unyielding surface.

The instrumentation and data collected were key elements of the test program. The inertial g-loads were measured during the 30 foot drop test by a series of accelerometers. The accelerometers could be located in 12 positions on three circumferences of the specimens, spaced at 90-degrees apart. For tests where the destruction of an instrument could be assured because of its vulnerable position or location, that instrument was omitted, but at least ten (10) accelerometers were used in each test. The accelerometers measured g-loads in only one direction so they were oriented in the desired location for each test configuration to collect the necessary data. Other data were also obtained that defined the initial conditions at the start of the test including temperature, wind speed, bolt torque on impact limiter bolts, as well as data obtained during and after the tests. These data included a video tape of the test, observations of damage observed on all relevant elements of the test body such as the depth of external crushing on the impact limiter, the overall thickness of the limiter after the test, the size of the impact footprint, and the impact acceleration time histories and frequency response from the drop.

For the side drop test (zero degrees) the impact duration on an impact limiter is projected to be 0.036 seconds with an average transverse acceleration of 61g compared to the predicted range of 53-60g based on the analytical results. The damage from the drop consisted of two of the impact limiter bolts of the top impact limiter failing in shear. The total maximum crush depth developed on the top limiter was 2.69 inches while the bottom limiter had a total maximum crush depth of 2.75 inches. The analytical prediction was approximately 3.35 to 4.05 inches. A single tear at a weld joint was observed after the test that measured approximately 0.25 inches wide and 6 inches long, however none of the inner crushable wood core products were displaced out of the split stainless steel shell.

For the 20 degree slap-down test the cask was oriented so that the bottom end made the first impact contact with the unyielding surface. Since this test was conducted with the specimen at an angle with the target surface there were two sets of accelerations from the two impacts (top and bottom) and with the location of instruments on the specimen it was possible to record the response of the center of gravity of the package as well as the response at other locations of devices. The duration of the first impact was approximately 0.048 seconds with the second impact having a duration of approximately 0.027 seconds. The recorded accelerations for the first impact for the transverse acceleration of the center of gravity was 17g with the predicted value ranging from 40-53g. For the rotational acceleration at the bottom limiter the recorded value was 19g and the predicted range was 62-80g. For the second impact the recorded transverse accelerations were 32g for the center of gravity with the predicted range being 36-44g. For the rotational acceleration of the top impact limiter the recorded value was 53g with the predicted range being 69-83g. The crush depths were evaluated after the test was completed and the impact limiters had been removed from the specimen. In this test there was evidence of crushing on the inside of the impact limiter at the surface interface with the cask as well as on the outside where the impact limiter surface contacted the target. For the top impact limiter the inside maximum crush depth was approximately 2.4 inches and the maximum outside crush depth was approximately 1.8 inches. With an allowance of 0.5 inches for spring back based on observations in previous tests, the total maximum crush depth observed for the top limiter was approximately 4.7 inches. The predicted crush depth was approximately 7.5 inches. For the bottom impact limiter the inside maximum crush depth was approximately 0.4 inches and the maximum outside crush depth was approximately 4.0 inches. With the 0.5 inch allowance for spring back, the maximum crush depth observed for the bottom impact limiter was 4.9 inches. The predicted crush depth was approximately 6.5 inches. Four of the twelve impact limiter attachment bolts for the top impact limiter failed meaning these failures were associated with the second impact limiter to contact the unyielding surface. The four failed bolts were oriented at 90-degrees so that with a 30-degree arc between bolts, no failures of an adjacent bolt occurred. There were no observed ruptures in the stainless steel shell surrounding the wood core of the impact limiters, consequently no core material was lost.

For the end drop (90 degree) test the drop was performed so that the bottom impact limiter was the surface contacting the target. The impact duration based on the test data was approximately 0.03 seconds. Accelerations were measured in the axial direction at several locations on the test specimen with the range of values being 62-70g with the average value of 65g. This observed value exceeded the predicted value of 44-50g. This was apparently caused by the fact that the bottom impact limiter had been cooled to -20°F prior to the test. This cold temperature causes an increase in the crush strength of the wood core materials. Increasing the crush strength of the wood core materials results in a higher g-loading from the more abrupt restraint against crushing. This phenomenon is evident in the crush depths observed from this cold end drop test. The maximum crush depth on the inside of the bottom impact limiter was found to be approximately 1.75 inches with the maximum outside crush depth found to be approximately 0.25 inches. With the 0.5 inch allowance for spring back, the total maximum crush depth observed as a result of this drop test was 2.5 inches. The predicted maximum crush depth was expected to be in the range of 3.5 to 4.5 inches. No damage was observed such as failed attachment bolts or ruptures in the stainless steel shell of the impact limiters other than the deformations characterized by the crush depths.

As a result of the tests, it can be concluded that the analytical results are conservative in the prediction of actual behavior. The testing program established that predictions of crush depth indicate that there will be no damage to the cask from the accidental drops, no exposure of the neutron shielding to impacts with the target surface, impact limiters will remain intact and anchored to the cask body, and that low temperatures of down to -20°F have little effect on performance of the -MP197 packaging.

As shown, all maximum stresses are well with the allowable stress levels.

2.8.2 Crush

Since the NUHOMS®-MP197 package has a mass greater than 500 kg and a density greater than water, a test under these conditions is not required.

2.8.3 Puncture

The NUHOMS®-MP197 was evaluated for puncture considering a 40 inch drop onto a 6 inch diameter bar of at least 8 inches in length. In addition, a puncture test was performed on the one-third scale specimen and puncture target in conjunction with the 30 foot drop test program.

The puncture analysis was performed using the ASME Boiler and Pressure Vessel Code, Section III, Division 3, 1998 with1999 Addenda. That analysis showed that the minimum wall thickness required was less than that provided by the 2.5-inch outer shell wall of the cask. The analysis conservatively neglected the thickness of the neutron shielding and the stainless steel thin shell acting as the outer cover for the neutron shielding. The impinging force was also calculated and considered as a loading on the cask body as a lateral force. The puncture load was applied, by analyses, directly to the cask lid and bottom ram cover by utilizing a two-dimension finite element model even though the thickness and capability of the impact limiters will prevent direct cask outer shell body contact with an 8-inch long puncture rod. The calculated stresses were within the allowable stresses as summarized in Table 2-12 and 2-13 of the SAR. Other exterior surfaces at penetrations of the cask such as the test port and vent port lids were also evaluated for the effects of puncture. It was concluded there is adequate protection.

The puncture test was performed to meet the requirements of 10 CFR 71.73 as well as to verify the analytical methods being utilized to calculate the response of the cask assembly to the puncture loading. This test was conducted so the puncture impact would be on the damaged bottom impact limiter resulting from the previous 30 foot end drop test. The target location to represent the most vulnerable area was identified as the center of the bottom circular surface of the impact limiter. The results produced a test specimen impaled on the test puncture rod with a hole sheared through the stainless steel impact limiter shell in a nearly exact circular pattern with the diameter essentially matching that of the test puncture rod. No other damage was observed relative to the impact limiter attachment bolts or the stainless steel shell. The wood core of the impact limiter was damaged as the puncture occurred. This resulted in the sweeping of wood pieces, fibers, and wood debris in front of the puncture bar as it penetrated into the wood core. An examination of the test specimen showed there was no loss of wood core materials and there were no observable damage to the stainless steel impact limiter shell welds and no penetration of the inner shell. Consequently, the cask was not impacted by the puncture bar. There were also no failed attachment bolts of the impact limiters as a result of the puncture test.

The applicant described the properties and acceptance testing of the redwood and balsa wood to be used in the impact limiters in which wood pieces are contained inside a stainless steel shell designed to both distribute the load to the wood during impact events and to maintain the wood free of external sources of water during the service period. Scale model testing of the impact limiters confirm that the limiters completely maintain and confine the wood segments during all required free drop events. The function of the adhesive material for the wood is to position the tightly packed wood within the space of the impact limiter.

2.8.4 Thermal

The accident conditions arising from the exposure of the NUHOMS®-MP197 packaging to the fire environment requirements of 10 CFR 71.73 produce increased temperatures and pressure that impose loading on the structure of the cask system. The calculated pressure as a result of these conditions is 1.64 atm (9.4 psig), however the design pressure was assumed to be 4.4 atm (50 psig). Table 3-3 of the SAR presents the maximum calculated temperatures as a result of this design basis fire under the described conditions. Table 2.10.1-45 of the SAR provides a summary of the resulting calculated stresses for various structural elements of the cask system. Stresses are not of major significance, in fact, the table compares the primary membrane stress plus the primary bending stress to the allowable stress for only the primary membrane stress. The maximum values for the various structural elements are provided below in terms of the % of the permitted primary membrane stress. Note that these values represent the loading combination of thermal and pressure acting on the -MP197.

COMPONENT	MAXIMUM STRESS AS % OF ALLOWABLE PRIMARY MEMBRANE STRESS
Lid	4%
Upper Wall	36%
Upper Trunnion	49%
Mid Wall	43%
Lower Trunnion	45%
Lower Wall	27%
Bottom	9%

As shown, maximum stresses are well within the allowable stress levels.

2.8.5 Immersion

2.8.5.1 Fissile Material

Water in-leakage is assumed for the criticality analysis of the package. Therefore, the immersion test for fissile materials is not applicable and the requirements of 10 CFR 71.73(c)(5) are met. The cask body stresses for this immersion condition under 3 feet of water (1.3 psi external pressure) are however bounded by the stresses generated under the immersion condition for all packages to a head of 50 feet of water that is equivalent to an external pressure of 36.4 psi addressed below.

2.8.5.2 All Packages

The immersion test requirement is that the applicant subject the package to water pressure equivalent to a head of water of at least 50 feet. This is equivalent to an external pressure of 36.4 psi. The cask body stresses for this immersion condition are however bounded by the stresses generated under the requirements for a package of irradiated nuclear fuel under an external water pressure of 290 psi addressed below.

2.9 Special Requirements for Irradiated Nuclear Fuel Shipments

The requirement of the package to resist an external pressure of 290 psi under 10 CFR 71.61 for this type of material has been evaluated using the axisymmetric finite element model with a loading combination including the cold environment. The results of that analysis are summarized in Table 2.10.1-44 of the SAR. The maximum primary membrane plus primary bending stresses expressed as a % of the allowable primary membrane stress for specific regions of the cask are listed below.

COMPONENT	MAXIMUM STRESS AS % OF ALLOWABLE PRIMARY MEMBRANE STRESS
Lid	13%
Upper Wall	12 %
Upper Trunnion	14%
Mid Wall	27%
Lower Trunnion	18%
Lower Wall	10%
Bottom	10%

As shown, maximum stresses are well within the allowable stress levels.

The buckling mode of the -61BT DSC has also been evaluated for the loading condition of an external pressure of 290 psi and when subjected to the accidental 30 foot end drop conditions. The buckling stress calculations determined a buckling stress of 31.5 ksi compared to the imposed stress of 29.8 ksi. Consequently there is no buckling potential for the shell under the condition of 290 psi external pressure. The buckling considerations under the accidental drop conditions are provided in Section 2.8.1 herein.

2.10 Internal Pressure Test

A pressure test is performed at a pressure of 62.5 psig which exceeds1.5 times the maximum normal operating pressure. This condition has also been analyzed as an internal pressure loading using an axisymmetric model. Stress allowables are well above the computed stresses on the containment boundary.

2.11 Evaluation Findings

F2.1 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.

F2.2 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 are considered and package containment is constructed from materials that meet the requirement of Regulatory Guides 7.11 and 7.12. The material properties and fabrication procedures for the impact limiters were reviewed. The staff found reasonable assurance that the package material properties will be maintained for the service life of the system.

F2.3 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.

F2.4 The staff has reviewed the packaging structural evaluation and found reasonable assurance that the application meets the requirements of 10 CFR 71.35.

F2.5 The staff has reviewed the packaging structural performance under the normal conditions of transport and found reasonable assurance that the packaging has adequate structural integrity to satisfy subcriticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

F2.6 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.

F2.7 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.

F2.8 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.

3.0 THERMAL

3.1 Review Scope and Objective

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

The staff reviewed the thermal aspects of the NUHOMS®-MP197 Transport Package to verify that package performance 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. Confirmatory calculations were performed for certain aspects of the design.

3.2 Description of the Thermal Design

3.2.1 Packaging Design Features

The NUHOMS®-MP197 package is designed to transport 61 intact standard Boiling Water Reactor (BWR) fuel assemblies. The package is composed of the following components:

- A transport cask consisting of a containment boundary, structural shell, gamma shielding material, and solid neutron shield.
- The NUHOMS®-61BT Dry Shielded Canister (DSC), a high-integrity stainless steel welded pressure vessel that provides confinement of radioactive materials for storage under 10 CFR Part 72.
- A fuel basket assembly located inside the DSC, which provides structural support for the BWR fuel and basket guide sleeves.
- Impact limiters consisting of balsa and redwood, encased in stainless steel shells, attached to the ends of the transport cask.
- Sets of removable upper and lower trunnions, bolted to the outer shell of the cask.

3.2.2 Codes and Standards

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

3.2.3 Content Heat Load Specification

The NUHOMS®-61BT DSC (and therefore the -MP197 transport cask) is designed to transport 61 intact BWR fuel assemblies. The total decay heat of the contents will not exceed 15.86 kW or 260 W/assembly.

3.2.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, Tables 3-1, 3-2, and 3-3 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 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 reported the maximum transient temperatures for essential components, as well as the approximate time at which the maximum temperatures were reached. For the hypothetical fire accident, all components remained below their material property limits with the exception of the cask impact limiters and the neutron shielding resin, neither of which are essential for maintaining the containment function of the cask. The temperatures and design temperature limit criteria for the package components were reviewed and found to be consistent throughout the SAR.

3.2.5 Summary Tables of Pressures in the Containment System

A discussion of the pressure in the containment system (canister) and cask body under the normal conditions of transport and hypothetical accident conditions is presented in Appendix 3.7.3 of the SAR. This section was reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation sections of the SAR. The maximum pressure reported for normal conditions was 1.67 atm (9.8 psig) for the canister and 1.37 atm (5.4 psig) for the cask body. The maximum pressure reported for the accident condition was 3.87 atm (42.2 psig) for the canister and 1.65 atm (9.6 psig) for the cask body[1]. The design basis accident pressure for the canister is 4.42 atm (65 psig).

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

The applicant provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. The applicant used surface absorptivity values to model solar insolation into the transportation cask and emmissivity values to model radiative heat transfer interaction between the environment and the transportation cask. The staff reviewed the thermal properties used for the analysis of the package and determined that they were appropriate for the materials specified, with the exception of the emmissivity value for stainless steel, as discussed in Section 3.4.7 of this SER. The approach used by the applicant for applying solar insolation loads was consistent with the SRP (NUREG 1617).

The applicant listed the properties of air in the SAR and these properties were utilized to analyze for the conditions of the cask required by 10 CFR Part 71 during normal conditions, off-normal and accident conditions. In Appendix 3.7.2 of the SAR, the fluid properties of the

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[1] The value reported in the SAR was incorrect, the corrected value is provided in this SER. The incorrect value was due to a calculation error and does not affect the analysis.

surrounding air were used in the evaluation of a thermal convection coefficient for fire accident conditions.

3.3.2 Technical Specifications of Components

The applicant provided references (in Section 3.6 of the SAR) for the technical specifications of pre-fabricated package components including O-rings, impact limiters (wood), and neutron absorber materials (polyester resin). All components are rated to perform under cold normal conditions for an ambient temperature of -40°F.

3.3.3 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 maximum allowable fuel cladding temperature of 1058°F is used by the applicant as a limit for the fuel cladding. This limit is justified and supported by the Pacific Northwest National Laboratory report, PNL-4835, which is a methodology accepted by the NRC staff.

3.4 Thermal Evaluation Methods

3.4.1 Evaluation by Analyses

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

The applicant used the ANSYS® finite element analysis code to perform the thermal evaluation of the cask. The applicant assembled several analysis models of the -MP197 cask and -61BT DSC to determine the temperatures that the cask, canister, and fuel would experience during normal and accident conditions. The models are described below.

3.4.2 Cask Body Model

The cask body model is a 3 dimensional model which represents a 90° symmetrical section of the packaging. The model includes the geometry and material properties of the impact limiters, thermal shield, the cask body, lead, neutron shield (resin in aluminum channels), and the outer shell. All gaps used in the model were specified in the SAR.

The neutron shielding consists of 60 slender resin-filled aluminum containers placed between the cask body and outer stainless steel shell. The applicant added an air gap of 0.01 inches at the thermal equilibrium condition between the resin boxes and the adjacent shells. No radiation heat transfer is accounted for across the gaps.

Redwood and balsa within the impact limiters are modeled as an isotropic material with material properties as described in Section 3.2 of the SAR. Figure 3.1 shows the cask body finite element model.

3.4.3 Fuel Basket Model

To determine the temperatures of components within the canister itself, a three dimensional 90° symmetrical section of the packaging was developed to include the geometry and material properties of the canister, basket, active lengths of the fuel assemblies, basket peripheral inserts, and the helium backfill between the canister and the cask body.

This model also includes fuel assemblies modeled as solid rectangular regions with an effective conductivity. The decay heat of the fuel was applied directly to the fuel elements and included a peaking factor of 1.2. As with the cask model, all gaps specified in the basket model are documented in the SAR. Figure 3.2 shows the fuel basket finite element model.

3.4.4 Thermal Analysis Results

For the normal operating conditions, the applicant performed a steady-state evaluation of the entire cask, using both the cask body model and the fuel basket model. These analyses produced a maximum fuel cladding temperature of 598°F, which is below the limit of 1058°F. The maximum seal temperature under normal conditions is 217°F for the Ram Plate closure, which is below the limit of 400°F.

The applicant again utilized the cask body model for the hypothetical accident condition analysis, and developed three additional finite element models (discussed in Section 3.6 of this SER) to properly assess the effects of the accident conditions on the cask. These analyses produced a maximum fuel cladding temperature of 680°F, which is below the limit of 1058° F. Under these conditions, the maximum seal temperature was shown to be 279°F. This seal temperature for the 30-minute fire accident is below the limit of 400°F.

3.4.5 Confirmatory Analysis

The staff, along with Pacific Northwest National Laboratory (PNNL), reviewed the finite element models that were submitted by the applicant. The review of the applicant's models revealed two issues which were subsequently communicated to the applicant as a request for additional information. The findings by the staff and the resolution of these issues are discussed below.

3.4.6 Canister Model Deficiencies

The applicant's approach to modeling the canister excluded all heat transfer mechanisms between the top of the fuel and the upper lid of the canister. While conservative for determining peak fuel cladding temperatures, the staff did not consider this a conservative approach for determining the maximum temperature of the cask seals. The applicant revised their thermal model to account for radiation in the gap between the top and bottom of the basket and the shield plugs of the canister. The revised analysis is presented in Appendix 3.7.5 to the SAR.

3.4.7 Surface Emmissivity Values

The applicant used an emmissivity value of 0.8 for weathered stainless steel from a heat transfer textbook to represent the surface emmissivity of the cask. The staff considered this value inappropriate given that the procedures for handling the cask would prevent the surface from seeing significant weathering, and that in credible references familiar to the staff, values given for emmissivity of weathered stainless steel range from 0.3 to 0.5.

3.4.8 Revised Analysis Results

The applicant revised their thermal models to include conduction paths and to account for radiation in the gap between the top and bottom of the basket and the shield plugs of the canister. The applicant also used a value of 0.5 for stainless steel emmissivity. In addition, the applicant chose to lower the absorptivity value of the stainless steel external cask surface to match a credible published value of 0.5. While this is less conservative than in the original analysis, the lower absorptivity value for stainless steel is considered to be realistic and is therefore, acceptable. The applicant determined that there was a maximum 14°F increase in seal temperatures. The maximum seal temperature according to the revised analysis was 231°F for normal conditions, which is below the seal temperature limit of 400°F. Given the margin between seal temperatures calculated for normal conditions and the seal temperature limits, the staff determined that it was not necessary for the applicant to re-analyze the seals for accident conditions. The staff reviewed the applicant's revised analysis and found that the analysis was acceptable.

3.4.9 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 and in analytical methods. Because of significant design margins, the staff found reasonable assurance that the applicant used appropriate considerations throughout the application.

3.4.10 Evaluation of Accessible Surface Temperature

The accessible surfaces of the NUHOMS®-MP197 package include the personnel barrier and the outermost vertical and radial surfaces of the impact limiters. The applicant analyzed the impact limiter surfaces under normal conditions in shade and determined that the accessible impact limiter surfaces would not exceed 110°F.

The applicant described the personnel barrier that surrounds the cask body as having an open area of 80%. The applicant states that heat transfer between the cask and the personnel barrier will be minimal, and therefore, the temperature of the personnel barrier will not be significantly elevated over ambient temperatures. The positioning of the personnel barrier is such that no surfaces of the cask body are accessible. The only areas of the package that are accessible are the impact limiters.

The staff identified that the applicants stated value of 80% open area for the personnel barrier is high, and that typically personnel barrier meshes are more on the order of 60% to 70% open area. However, the staff verified that the surface temperature of the personnel barrier would not exceed 185°F by means of a simple calculation based on the surface temperature of the cask. In addition, the staff determined that the personnel barrier is in place on the cask when prepared for transport from it's point of origin, as an integral piece to the transport skid. Because all accessible surfaces of the package, as prepared for transport, will remain below the regulatory limit of 185°F, the staff finds this acceptable.

3.5 Thermal Evaluation under Normal Conditions of Transport

3.5.1 Heat

The applicant performed steady-state calculations for an ambient temperature of 100°F with solar insolation and a maximum decay heat of 260 W/assembly utilizing the models described in Section 3.4 of this SER.

The fuel basket model was used to determine the maximum fuel cladding temperatures for normal conditions of transport. Each fuel assembly in the model is represented as a three dimensional rectangular solid and is given an effective thermal conductivity according to the values calculated by the applicant in Appendix 3.7.1 of the SAR. Each homogenized fuel assembly was also divided into twelve intervals and a volumetric heat generation rate multiplied by the average peaking factor for that interval was applied to each interval. This process is described in more detail in Section 3.4.1.3 of the SAR.

The staff reviewed the applicant's models, the procedures used to analyze normal conditions of transport, and the procedures used to apply the decay heat to the fuel regions and found them to be acceptable.

3.5.2 Cold

At an ambient temperature of -40°F and no applied decay heat, the entire package will approach a temperature of -40°F. The applicant reported temperatures based on an analysis of the models described in Section 3.4 of this SER for ambient temperatures of -20°F and -40°F in Table 3.2 of the SAR. The applicant concluded that cask components, including the containment system seals, would not be adversely affected by this low temperature. The staff reviewed the information provided by the applicant and agrees with the conclusion.

3.5.3 Maximum Normal Operating Pressure (MNOP)

The applicant calculates the MNOP within the cask body and DSC for normal conditions of transport in Appendix 3.7.3. The maximum pressure reported for normal conditions was 1.67 atm (9.8 psig) for the canister and 1.37 atm (5.4 psig) for the cask body. The design basis pressure for the cask body is 4.4 atm (50 psig) and for the -61BT DSC (canister) the normal design pressure is 1.68 atm (10 psig). The MNOP for both the cask body and canister are within the limits set by the applicant.

3.5.4 Maximum Thermal Stresses

The applicant reports maximum thermal stresses for normal conditions of transport in Section 2.6.1.3 of the SAR. All thermal stresses are below the allowable stresses for critical cask components.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

3.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package under hypothetical accident conditions. The pre-fire condition was a 100°F ambient with radiation and convection from the surface of the cask, based on normal conditions. The applicant stated that the impact limiters on the cask were not damaged sufficiently during the free drop and puncture tests to warrant a reduction in thermal effectiveness during the fire accident condition. The staff found this assumption acceptable given the reported margin in the thermal design of the package.

The applicant developed additional ANSYS® models for the accident condition analysis. These models are described below.

3.6.2 Cask Cross Section Model

A three dimensional, quarter symmetry model of the cask cross section was developed by the applicant. The model includes the cask body, canister, basket, and fuel assemblies. The same gaps applied to the normal condition cask body and basket models were applied to this model. The applicant stated that the normal condition gaps were used to bound the heat conductance uncertainty between adjacent packaging components. The staff does not agree with this assumption. Gaps will decrease as cask component temperatures increase and therefore the heat transfer rate across the gaps will increase. The effect of decreased gaps, during the hypothetical accident fire, would be a slight rise in fuel cladding temperature. Given the margin in this package design, the staff finds the assumptions made by the applicant to be acceptable. Figure 3.3 shows the cask cross section finite element model.

3.6.3 Trunnion Region Model

The applicant developed a model of the cask lifting trunnions to determine the peak transient lead temperature in the trunnion regions. The model is a 2 dimensional axisymmetric finite element model that includes the geometry and material properties of the trunnion block, trunnion plug, and cask body in the region of the trunnion. Gaps for this model are described in Section 3.5.4 of the SAR. The gaps in the model were removed for the fire transient, but were present for the pre- and post-fire transient. The staff reviewed this model and found it to be an accurate representation of the actual cask trunnion regions.

3.6.4 Bearing Block Region Model

The applicant also developed a model of the cask bearing block to determine the peak transient lead temperature in the bearing block region. The model is a 3 dimensional quarter-symmetry finite element model that includes the geometry and material properties of the adjacent neutron shielding and the corresponding portion of the cask body. Gaps for this model are described in Section 3.5.5 of the SAR. The gaps in the model were removed for the fire transient, but were present for the pre- and post-fire transient. Both the Trunnion Region Model and the Bearing Block Model are depicted in Figure 3.4.

3.6.5 Fire Test

For the fire accident, the applicant subjected the analysis models to an ambient temperature of 1475°F for 30 minutes, as defined by 10 CFR 71.73. A convective coefficient of 2.75 BTU/hr-ft^2-°F is used to model the turbulent nature of the fire environment. The staff finds this value acceptable. The model surfaces were given an absorptivity of 0.8 for pre- and post fire accident periods, as per the regulations.

The applicant removed the gaps between the materials for the fire transient, to maximize heat input into each of the models. Following the fire transient, the models were then returned to normal condition for the duration of the post fire transient.

3.6.6 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are given in Table 3-1. 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.

Under hypothetical accident 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 resin and the impact absorbing materials exceeded the failure temperatures. The applicant's analysis indicates that there will be no melting of the lead gamma shield as the calculated lead temperature of 478°F is less than the melting point of lead (620°F). In addition the analysis indicates that the containment seals would not be compromised, as their maximum calculated temperature of 279°F is less than the maximum service temperature of 400°F.

The applicant calculated the maximum internal operating pressure, considering only the helium fill gas. The average gas temperature in the cask body was calculated to be 504°F, while the average gas temperature in the canister was calculated to be 583°F. Based on these gas temperatures, the maximum internal operating pressure was determined to be 3.87 atm (42.2 psig) and 1.65 atm (9.6 psig) for the canister and cask body, respectively. For the -61BT DSC (canister) the accident design pressure is 5.38 atm (65 psig), while for the cask body, the design pressure is 4.35 atm (50 psig). Therefore, the values calculated by the applicant are below the pressure limits specified by the applicant.

The staff reviewed the calculations submitted by the applicant and found them to be adequate. One minor error was found in the pressure calculations for the cask body (see footnote 1, Section 3.2.5). Therefore, the staff agrees that the cask and canister meet the design requirements for maximum pressures.

Table 3-1
Maximum Calculated Temperatures (°F)

Location or Cask Component	Normal Conditions	Accident Conditions	Maximum Allowable
Impact Limiter	195	1456	N/A
Thermal Shield	186	1172	1
Cask Body	302	535	1
Seals, Ram Plate	231[2]	270	400
Seals, Cask Lid	206[2]	279	400
Neutron Shield (Resin)	249	N/A	N/A
Average Cavity Gas (Cask)	345	504	N/A
Average Cavity Gas (Canister)	493	583	N/A
Lead Shielding	299	478	620
Canister Shell	388	485	1
Basket	578	661	1
Fuel Cladding	598	680	1058
1) Component performs intended safety function within the operating range 2) Maximum seal values calculated from revised analysis in Appendix 3.7.5 to the SAR			

3.6.7 Maximum Thermal Stresses

The applicant reports maximum thermal stresses for the hypothetical accident condition fire in Section 2.7.3.2 of the SAR. All thermal stresses are below the allowable stresses for critical cask components.

3.7 Appendix

The applicant provided four (4) Appendices to Chapter 3 of the SAR, which are listed below.

Appendix 3.7.1 Effective Thermal Conductivity for the Fuel Assembly
Appendix 3.7.2 Average Heat Transfer Coefficient for Fire Accident Conditions
Appendix 3.7.3 Maximum Internal Operating Pressures
Appendix 3.7.4 Thermal Evaluation for Vacuum Drying Conditions

These appendices provided the information necessary for the staff to make its safety findings regarding the adequacy of the design when compared to the requirements in 10 CFR Part 71.

3.7.1 Justification for Assumptions or Analytical Procedures

The applicant included general discussion throughout Chapter 3 of the SAR to justify the approaches taken in modeling the thermal performance of the NUHOMS®-MP197. The staff was in agreement with the justifications presented by the applicant. While the staff found the homogenized fuel model used in this analysis to be acceptable in this instance for conditions that result in peak clad temperatures near the limit, the applicant should validate it's thermal methods against available spent fuel cask temperature data (e.g., INEEL/EPRI dry cask storage data).

3.7.2 Computer Program Description

The applicant provided a brief description of the ANSYS® finite element analysis code in Section 3.1.3.1. Because ANSYS® is a widely used and accepted general purpose code, and one the staff is familiar with, the description provided by the applicant was adequate.

3.7.3 Computer Input and Output Files

Upon the staff's request, the applicant provided ANSYS® database files containing the models used in the thermal analysis.

3.8 Evaluation Findings

F3.1 The staff has reviewed the package description and evaluation and has reasonable assurance that the information provided satisfies the thermal requirements of 10 CFR Part 71.

F3.2 The staff has reviewed the material properties and component specifications used in the thermal evaluation and has reasonable assurance that the information provides sufficient basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

F3.3 The staff has reviewed the methods used in the thermal evaluation and has reasonable assurance that the models are described in sufficient detail to permit an independent review of the package thermal design. The application of the analysis methods, presented in the SAR, to this cask design has been found to be adequate.

F3.4 The staff has reviewed the accessible surface temperatures of the package, as it will be prepared for shipment, and has reasonable assurance that the requirements of 10 CFR 71.43(g) for packages transported by exclusive-use vehicle have been satisfied.

F3.5 The staff has reviewed the package design, construction, and preparations for shipment and has 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.

F3.6 The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-term limits during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73.

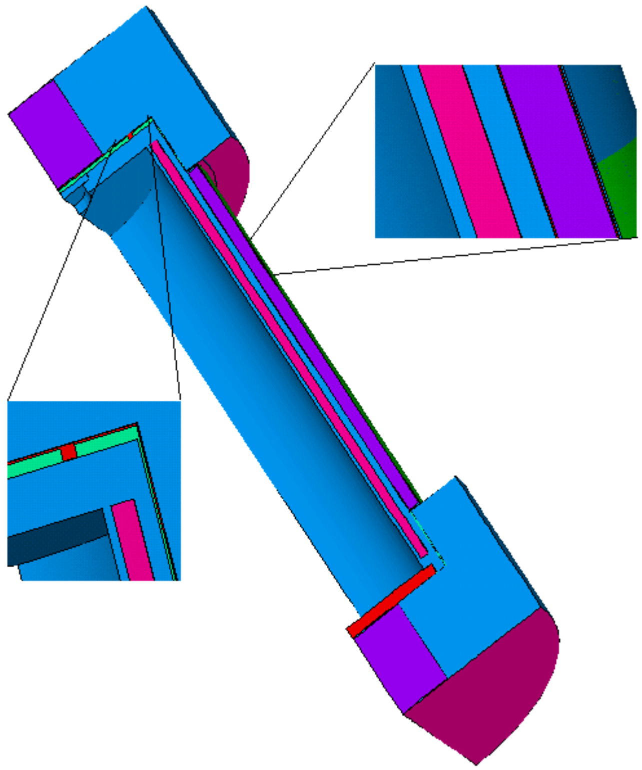

Figure 3.1 MP-197 Cask Body ANSYS® Model

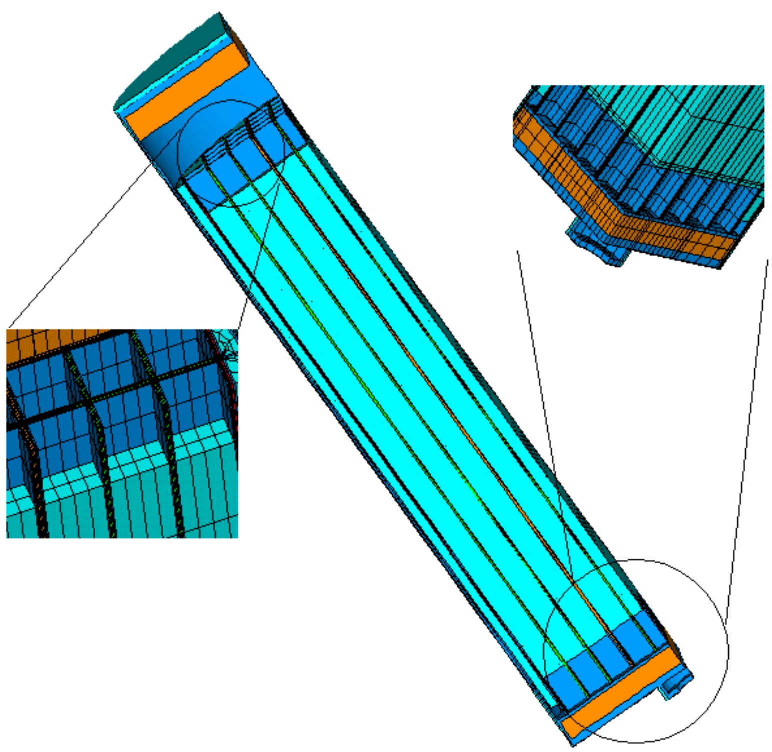

Figure 3.2 Canister and Fuel Basket ANSYS Model

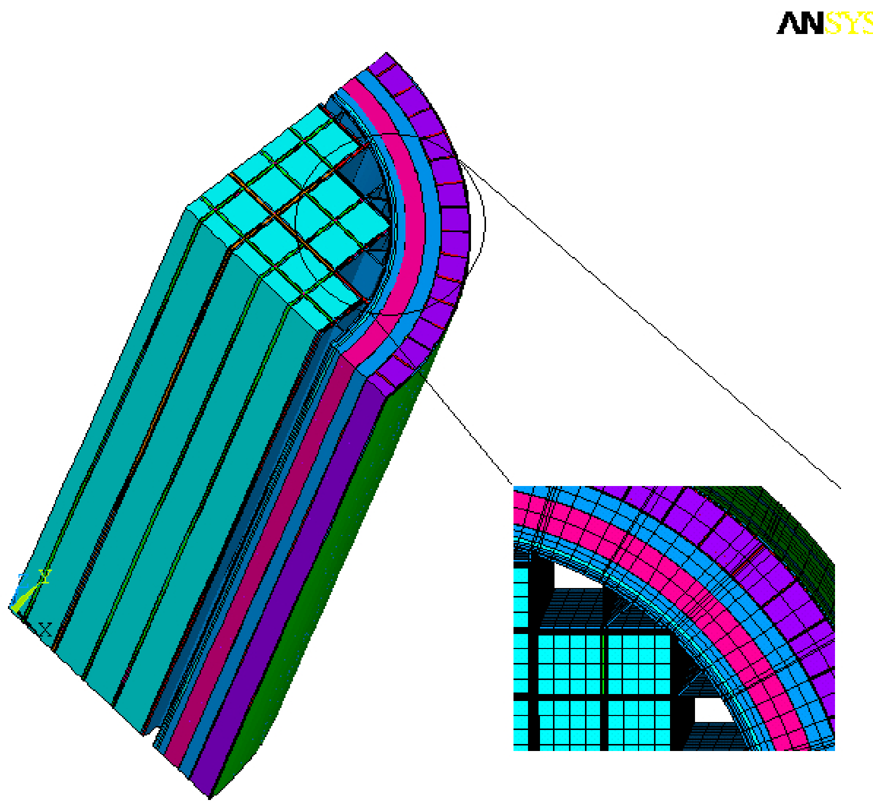

Figure 3.3 MP-197 Cask Cross Section ANSYS® Model

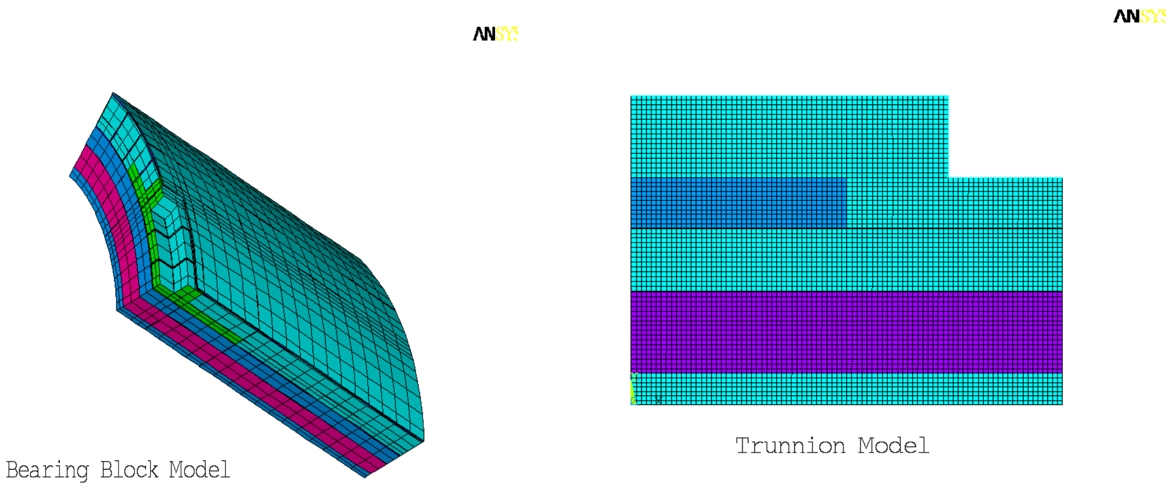

Figure 3.4 MP-197 Bearing Block and Trunnion ANSYS® Models

4.0 CONTAINMENT

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.

4.1 Description of the Containment System

4.1.1 Containment Boundary

The containment system of the package consists of the following components: (1) the inner shell, (2) the 6½ inch bottom end closure plate with a ram access penetration with seal, (3) cask body flange, (4) the 4½ inch top lid with seal, (5) the lid bolts, (6) the vent and drain cover plates, (7) the vent and drain port covers, and (8) the vent and drain port cover plate bolts. Table 4-1 lists containment system components and their material of construction.

Table 4-1
MP-197 Containment System Components

COMPONENT		
MATERIAL	**Item No. from General Arrangement Parts List 1093-71-3 , REV. 1**	**Part**
SA-240, Type XM-19	2	Inner Shell
SA-240, Type XM-19 or SA-182, Type FXM19	33	RAM Closure Plate
SA-540, Grade B24	34	RAM Closure Bolts
Fluorocarbon per AMS-R-83485 Class 1	37	RAM Closure Seal
SA-182, Type FXM19	5	Bottom End Closure
SA-182, Type FXM19	12	Cask Body Flange
SA-705, Type 630, H1100	20	Top Lid
SA-540, Grade B24	21	Closure Lid Bolts
Fluorocarbon per AMS-R-83485 Class 1	24	Top Closure O-Ring Seal
SA-564, Type 630 H1150	39	Drain and Vent Port Closures
Fluorocarbon per AMS-R-83485 Class 1	42	Drain and Vent Port O-Ring Seal
SA-540, Grade B24	40	Drain and Vent Port Bolts

The containment system is designed to a leakage rate of 1×10^{-7} ref-cm^3 /s or less.

Each penetration has double O-ring seals. The inner O-rings are the containment seals and the outer O-rings facilitate leak testing of the inner containment O-rings. All containment seals are leak tested in accordance with ANSI-N14.5 and replaced after each use. The seal areas are designed for no significant plastic deformation under normal and accident loads.

The top closure plate is closed with 48, 1½ inch diameter bolts. The ram closure plate is closed with 12 one-inch diameter bolts. The vent and drain ports are each closed with a single ¾ inch diameter bolt with seals under the head. Bolt torque values are specified in TN Drawings No. 1093-71-3, Rev. No. 1. The staff reviewed the containment system description and concludes that the description of the containment boundary is sufficient in detail to provide an adequate basis for its evaluation, per the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(4). The staff also finds that 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, as required by 10 CFR 71.43(c).

4.1.2 Codes and Standards

The containment vessel is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code Section III, Subsection NB to the maximum extent practicable. Exceptions to the ASME Code are listed in TN drawing No. 1093-71-22, Rev. No. 1. All containment welds are full penetration bevel or groove welds and are designed in accordance with ASME Code Section III, Division 1, Subsection NB.

The staff has reviewed the description of the containment system, as described in Chapters 1 and 4 of the SAR. The staff concludes that the established codes and standards applicable to the containment design have been identified per the requirements of 10 CFR 71.31(c).

4.1.3 Special Requirements for Damaged Spent Nuclear Fuel

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

4.2 Containment Under Normal Conditions of Transport

4.2.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed that the pressure under normal conditions of transport would not exceed the package design pressure of 1.37 atm (5.4 psig).

4.2.2 Containment Criteria

The containment system is designed to a leakage rate of 1×10^{-7} ref-cm^3 /sec or less. The applicant assumed a rod failure rate of 3% with all free gases within the rods released. In accordance with ANSI 14.5, fabrication verification, periodic verification, and assembly verification leak tests will be performed to verify the containment capability of the containment system.

4.2.3 Compliance with Containment Criteria

Results of the applicant's structural and thermal analyses show that the containment system retains the capability to maintain a seal of 1×10^{-7} ref-cm^3/sec or less under the conditions specified in 10 CFR 71.71 which is considered leak tight per ANS/ANSI N14.5. Therefore, the staff concludes that 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).

4.3 Containment Under Hypothetical Accident Conditions

4.3.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed that the pressure under hypothetical accident conditions would not exceed the package design pressure of 4.35 atm (50 psig).

4.3.2 Containment Criteria

The containment system is designed to a leakage rate of 1×10^{-7} ref-cm^3/sec or less under hypothetical accident conditions assuming a rod failure rate of 100%.

4.3.3 Compliance with Containment Criteria

Results of the thermal analysis show that seal temperatures will remain below the seal material temperature limits of 400°F during and after the 30-minute fire. Results of the structural analysis show that the cask inner shell will not buckle under hypothetical accident conditions.

Results of the structural and thermal analyses in Chapters 2 and 3 of the SAR showed that the containment system remained leak tight under the tests specified in 10 CFR 71.73. Since the containment vessel is designed, fabricated, and tested to meet the leak tight criteria of "American National Standard for Radioactive Materials," ANSI N14.5-1997, there is no contribution to the radiological consequences due to a potential release of canister contents. The staff agrees with the applicant's conclusion that the containment system meets the requirements of 10 CFR 71.51(a)(2).

4.4 Evaluation Findings

F4.1 Based on review of the statements and representations in the application, the staff has reasonable assurance that the package containment design has been adequately described and evaluated and that the package meets the containment performance requirements of 10 CFR Part 71.

5.0 SHIELDING

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

5.1 Shielding Design Description

Gamma-ray shielding is provided in the radial direction by the lead and stainless steel shells of the cask wall which are 3¼ inches and 3¾ inches thick, respectively. In addition, gamma shielding of the top and bottom of the system is provided by the steel assemblies of the NUHOMS®-61BT canister, which is then covered (when the impact limiters are in place) by an assembly of wood encased in stainless steel. The neutron shielding is provided by a borated polyester resin compound, approximately 4½ inches thick, which surrounds the cask body radially. The NUHOMS®-MP197 shielding details are provided in drawings 1093-71-2, -3, -4, and -5.

5.2 Radiation Source

The contents for the package are limited to 61 intact General Electric (GE) Boiling Water Reactor (BWR) fuel assemblies, or equivalent reload fuel, with zircaloy cladding. An intact fuel assembly is a spent nuclear fuel (SNF) assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. The fuel may be transported with or without channels. The permissible fuel assembly types are presented in Section 1.2.3 of the SAR.

The source specification is presented in Section 5.2 of the SAR. The gamma and neutron source term calculations for the seven basic GE designs were performed with the SAS2H/ORIGEN-S computer code. The GE 7x7 was chosen as the design basis fuel assembly as it has the highest initial heavy metal loading (0.198 MTU). Source terms were then calculated for the design basis assembly for the four proposed burnup/enrichment combinations listed in Section 1.2.3 of the SAR. Group 2 was identified as having the bounding source term.

To correct for changes in the neutron flux outside the fuel zone during irradiation, the masses of the materials in the bottom end fitting, plenum, and top end fitting were multiplied by scaling factors of 0.15, 0.2, and 0.1, respectively. These are the scaling factors recommended in “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” PNL-6906, June 1989, and are considered to provide bounding values.

Axial peaking factors are taken from the Transnuclear TN-68 SAR. These peaking factors were determined based on typical axial burnup distributions for BWR assemblies using axial water density distributions found during core operations. The data provided burnup and moderator density for 25 axial locations along the assembly, which the applicant collapsed into 12 axial zones. SAS2H was used to calculate the source terms for each zone of the design basis fuel with the power and water density varied in each zone. The relative source distributions are shown in Figure 5.2-1 of the SAR. The staff notes that the neutron source term in the upper half of the fuel may be underestimated due to the water densities used in the neutron source calculation. However, this is offset by the applicant’s conservative gamma source term.

The hardware activation analysis considered the cobalt impurities in the assembly hardware. The cobalt content is listed in Table 5.2-3. Although cobalt impurities can vary, the applicant's assumed values are reasonable and acceptable. The gamma source also includes the contribution from the fuel channels.

The applicant's methods for calculating the radiation source terms were reviewed. The staff also performed confirmatory calculations with the SCALE-4.4, SAS2H, and ORIGEN-S computer modules and found acceptable agreement with the applicant's reported values.

5.3 Shielding Model

The applicant demonstrates compliance with the external radiation requirements specified in 10 CFR 71.47 and 71.51 by using dose rate modeling. The MCNP-4B computer code was utilized by the applicant to calculate dose rates at various locations around the package. The applicant calculated dose rates for various combinations of burnup, initial enrichment, and cooling time. The dose rate results are presented in Table 5.1-2 of the applicant's SAR. All of the applicant's reported dose rate results are below the applicable 10 CFR Part 71 regulatory limits for exclusive-use transport.

5.4 Shielding Evaluation

The exterior dose rates are adequately controlled by limits specified in the Certificate of Compliance. Utilizing parameters provided in the SAR, the staff performed confirmatory shielding analyses using the SCALE-4.4a computer code. These analyses were performed to ensure that the applicant's source term and dose rate calculation methodologies were satisfactory. The staff's calculated dose rates were in reasonable agreement with the SAR values. The staff found that the SAR has adequately demonstrated that the NUHOMS®-MP197 package is designed to meet the applicable regulatory limits specified in 10 CFR 71.47 and 71.51.

Materials used in this application were either those already approved by staff for use in the -61BT canister or are materials whose properties are adequately described in the SAR for the -MP197. The staff reviewed the information presented in the SAR. The staff has no safety concerns on the materials previously approved for use in the -61BT canister. The staff also has not found any safety concerns in relation to materials selections, as presented in the SAR for the -MP197, for the functions of either gamma shielding or neutron shielding. The staff concludes that the materials selections for the -MP197 are adequate for satisfying the applicable shielding safety requirements for this application.

5.5 Evaluation Findings

F5.1 Based on review of the statements and representations in the application, the staff has reasonable assurance that the design has been adequately described and evaluated and that the package meets the shielding performance requirements of 10 CFR Part 71.

6.0 CRITICALITY

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

The staff reviewed the NUHOMS®-MP197 Transport Package criticality analysis to ensure that the transportation package meets the criticality safety requirements of 10 CFR Part 71[1], and in particular, to ensure that the fissile contents remain adequately subcritical for all postulated configurations under the normal conditions of transport specified in 10 CFR 71.71[1] and the hypothetical accident conditions specified in 10 CFR 71.73[1].

6.1 Description of the Criticality Design

The package design criterion for criticality safety is that the sum of the effective neutron multiplication factor (k_{eff}), two standard deviations of the statistical uncertainty and the bias uncertainty (95% confidence), and any bias which increases k_{eff} should not exceed 0.95.

6.1.1 Packaging Design Features

The NUHOMS®-MP197 Transport Package consists of the NUHOMS®-MP197 Transport Cask (outer packaging) and an inner container designated the NUHOMS®-61BT Dry Shielded Canister (DSC). The DSC relies on basket geometry and fixed neutron poisons to maintain criticality safety.

Each fuel compartment in the basket is made of a stainless steel square tube which is nominally 6.00 inches by 6.00 inches with a minimum passable opening of 5.80 inches. A neutron absorber plate is placed between each fuel compartment and is made of an aluminum matrix composite containing boron carbide particles or a wrought aluminum alloy containing enriched boron. The minimum acceptable areal density of ^{10}B in the absorber plate is specified in Drawing No. 1093-71-3 and varies with enrichment and material type.

The design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71.71 and 71.73.

6.1.2 Codes and Standards

The applicant’s criticality analysis is consistent with the appropriate codes and standards for nuclear criticality safety. Also, the criticality analysis is consistent with the recommendations provided in NUREG/CR-5661[2].

6.1.3 Summary Table of Criticality Evaluations

The applicant used the methods recommended in NUREG/CR-6361[3] to establish Upper Subcritical Limits (USLs) for the calculated k_{eff} values of the package as adjusted. The USL value includes the calculational bias and the administrative margin. The final values reported for k_{eff} include a factor of two times the standard deviation of the statistical uncertainty of the calculated value. The reported values of k_{eff} were then compared against the minimum USL value of 0.9414.

Tables summarizing the final results of the criticality safety analysis are provided in Section 6.4 of the SAR. The table in Section 6.4.4 of the SAR reports results for a single package analysis as required in 10 CFR 71.55. Table 6-7 of the SAR reports results for an array of dry packages and also gives results for an array of damaged flooded packages under the hypothetical accident conditions. While an array of packages under the normal conditions of transport was not specifically modeled, the applicant modeled an infinite array of damaged packages, without any internal flooding, to search for the optimum flooding conditions. This array of dry packages has conditions which are closest to an array of packages under the normal conditions of transport. Both array results address the requirements in 10 CFR 71.59.

The maximum k_{eff} for each condition, as calculated by the applicant, is summarized in Table 6-1 below. The results are less than the associated minimum USL value of 0.9414 and these results illustrate that the package design meets the criticality safety requirements of 10 CFR Part 71.

Table 6-1
Maximum k_{eff} Results in MP197
SCALE 4.4 Calculations

Condition	$k_{eff} + 2\sigma$
Single Package, Flooded and Reflected 10 CFR 71.55(b), (d) and (e)	0.9371
Infinite Array of Damaged Packages, Dry 10 CFR 71.59(a)(1)	0.4372
Infinite Array of Damaged Packages, Flooded 10 CFR71.59(a)(2)	0.9387

6.1.4 Transport Index

The applicant modeled an infinite array of packages for both the normal conditions of transport and the hypothetical accident conditions. Therefore, pursuant to 10 CFR 71.59(b), the transport index based on criticality safety for the package is 0.

6.2 Spent Nuclear Fuel Contents

The DSC is designed to contain up to 61 intact Boiling Water Reactor (BWR) fuel assemblies with or without channels. The range of fuel assemblies includes the following General Electric Corporation (GE) designs: 7x7 array assemblies (GE2 and GE3); 8x8 array assemblies (GE4, GE5 GE-Pres, GE-Barrier, GE8-Type I, GE8-Type II, GE9 and GE10); 9x9 array assemblies (GE11 and GE13); and 10x10 array assemblies (GE12). Reload fuel from other manufactures with parameters bounded by those described in the CoC are also allowed. The maximum average lattice enrichment for each fuel assembly is 3.7, 4.1, or 4.4 wt. % ^{235}U depending on the ^{10}B content in the poison plates.

Only intact fuel assemblies may be shipped in the NUHOMS®-MP197 Transport Package. Any missing fuel rods in an assembly must be replaced by dummy rods that displace an amount of water at least equal to the original rod for the assembly to be considered intact.

The staff reviewed the description of the spent nuclear fuel contents and agrees that all relevant specifications have been provided.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The basic model for the NUHOMS®-MP197 Transport Package analysis had the following assumptions and features: (1) no credit was taken for fissile depletion or fission product poison buildup during fuel irradiation, (2) no burnable poisons were included, (3) a fuel density at 95% of theoretical (neglecting dishing) was assumed, (4) grid plates, spacers and hardware in the fuel assembly were omitted, (5) the ends of the active fuel were either reflected by water or had a reflective boundary making the fuel appear infinitely long, (6) flooding was with pure water at 68°F (20°C) including the pellet-clad gap, (7) the least material condition for the fuel clad OD, fuel compartment, poison plates and wrappers was assumed, (8) the active fuel region was assumed to start level with the bottom of the poison plates, (9) the maximum possible gap between poison plates in the worst case position was assumed, and (10) the stainless steel rails that hold the basket together were modeled as water. For fuel with variable enrichments, the lattice average enrichment for each axial zone was computed and the maximum average value was used throughout the fuel length including the natural uranium blankets.

In addition to the above assumptions, the neutron shield and stainless steel skin of the cask were replaced with moderator in the hypothetical accident mode. The analytical results reported in Section 2 of the SAR demonstrate that the containment boundary and DSC basket structure do not experience any significant distortion under the hypothetical accident conditions.

Two model variations were used in the analysis. The first model was a full-active fuel height model and full-radial cross section of the DSC with water boundary conditions at the ends and reflective boundary conditions on the sides. This model did not include gaps in the poison plates and was used to determine only the most reactive fuel assembly/channel combination and to justify the use of the lattice average enrichment for the fuel bundle enrichment. The second model was used for the subsequent analyses and had: (1) a fully reflective boundary at the fuel ends, (2) maximum gaps between the poison plates, and (3) minimum material conditions for the basket internals.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the package and contents given in SAR Sections 1 and 6, including the engineering drawings. Staff also reviewed the applicant's modeling assumptions and found them to be consistent with the NRC's acceptance criteria[4] and consistent with the conditions of the package as determined in the other analyses in the SAR.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety models are provided in Table 6-5 of the SAR. The standard material data libraries in the SCALE code package were used to model these materials. No Credit was taken for burnable poisons in the fuel.

The applicant's calculations take credit for only 75% of the minimum acceptable ^{10}B areal density in the BORAL® basket absorber material and 90% of the minimum acceptable ^{10}B areal density in the borated aluminum alloy and in each of two approved boron-carbide/aluminum-metal-matrix composite basket absorber materials. The percentage credit was commensurate with the degree of boron content verification provided in the acceptance testing program as described in Section 8.1.6 of the SAR.

The basket materials are not expected to degrade during storage such that there is any impact on criticality safety. The neutron flux in the package is very low such that depletion of the ^{10}B is negligible.

6.3.3 Computer Codes and Cross Section Libraries

The applicant utilized the CSAS25 (with Keno Va) control module in SCALE[5], Version 4.4, for the criticality safety analysis. The 44-group cross section set available in the SCALE system was used. This cross section set is derived from ENDF/B-V data and is designed to handle a wide variety of thermal systems. The calculations of k_{eff} used approximately 500,000 histories and produced standard deviations of less than 0.2%.

The staff agrees that the code and cross-section set used are appropriate for this particular cask design and contents. Representative input files for the calculations were reviewed for consistency with the content specifications and the engineering drawings.

6.3.4 Demonstration of Maximum Reactivity

The fuel lattice for each assembly type was evaluated to determine the most reactive type. These models had the fuel centered in the basket cells. The GE12, 10x10 array, was found to be the most reactive fuel type. Also, the four cases of no channel and channel thicknesses of 0.065, 0.080, and 0.120 inches were calculated. The results found that the difference between channel thicknesses was insignificant, i.e, within the statistical variation. Calculations for the GE2, GE5, and GE9 were made to compare the results for a lattice average enrichment model versus an explicit model with variable enrichment. The lattice average results were bounding in all three cases. Thus, subsequent analyses used a lattice average model of the GE12, 10x10 array, fuel without channels.

The applicant analyzed a number of model parameter variations to determine the conditions that produce the maximum value of k_{eff}. An analysis of the variation of the fuel assembly position in the basket cells showed the maximum k_{eff} occurs when the assemblies are shifted toward the DSC center. This configuration was used in the subsequent analyses. An analysis of the effect of canister shell thickness found no statistical difference and the nominal thickness was use subsequently. An analysis of the effect of poison plate thickness with constant areal density showed that the minimum plate thickness was most reactive and this thickness was used subsequently. A sensitivity analysis of the tolerance on the cladding thickness found k_{eff} to be maximized when the GE12 fuel rod had a minimum OD of 0.394 inches (0.010 inches less than the design value). This minimum OD was used subsequently. The next analysis varied the fuel cell size from 5.8 inches to 6.1 inches and found the minimum opening to be most bounding. This minimum dimension was used subsequently. An analysis of the effect of internal moderator density showed full density to maximize k_{eff} and this assumption was used subsequently. An analysis of the effect of moderator density between packages found variations which were within the statistical uncertainty but the apparent maximum of 70% density was used for the damaged array calculations.

Finally, the applicant performed calculations to determine the minimum areal density of ^{10}B needed in the basket as a function of the lattice average enrichment. This model represented the most reactive configuration with a full load of intact GE12 fuel all shifted toward the center with nominal DSC shell thickness, minimum poison plate thickness, minimum fuel clad OD, minimum fuel cell width, and full internal and external moderator density. Table 6-2 below presents the minimum areal densities of ^{10}B as found in the analysis.

Table 6-2
Minimum ^{10}B Content

Maximum Lattice Average Enrichment (wt. % ^{235}U)	Value Used in Calculations (mg of $^{10}B/cm^2$)	Borated Aluminum and Approved Aluminum Metal Matrix Composites (mg of $^{10}B/cm^2$)	BORAL® (mg of $^{10}B/cm^2$)
3.7	19	21	25
4.1	29	32	38
4.4	36	40	48

The methods and calculations used to determine the optimum conditions that maximize k_{eff} were reviewed and found acceptable.

6.3.5 Confirmatory Analyses

The staff used the CSAS25/KENO-Va code in the SCALE[5] suite of analysis codes, Version 4.4a, to perform confirmatory analyses. These calculations used both the 27-group and 44-group cross sections in SCALE. The CSAS code was developed by the Oak Ridge National Laboratory for performing criticality analyses and is appropriate for this particular application and fuel system.

The staff's calculations were based on the information provided in the SAR. The GE12, 10x10 array, fuel assembly was used in the calculations. The staff used two different models in its analysis. The first model simulated infinitely long fuel rods. This model was used to compare staff's calculations against those of the applicant. Next the model was used to confirm that shifting all fuel toward the center of the cask was most reactive and found this configuration to be about 0.5% higher in k_{eff} than when the fuel assemblies are centered in their individual fuel cells. The staff also evaluated a case where a 2x2 array of fuel assemblies in the middle of the basket are shifted toward their common center and found this to be slightly less reactive than shifting all assemblies toward the basket center.

During the review, a question was raised on how the user can determine whether the basket poison plates contain the appropriate level of ^{10}B material. The minimum ^{10}B content is indicated by a letter designation (A, B, or C) in the package Identification Number stamped on the bottom of the cask. To evaluate the adequacy of this labeling, the staff used the first model to determine the potential consequence of misloading fuel assemblies enriched in ^{235}U to 4.4% into a cask designed to hold only 3.7% enriched fuel. The staff found that the cask remained subcritical under this misloading and determined that the labeling approach was acceptable.

Finally, the staff used the first model to evaluate the adequacy of the acceptance testing program for the basket poison plates. In this evaluation, the staff simulated a series of cases where one to four of the poison plates around the center assembly in the basket have a ^{10}B content that is one half of the value credited in the applicant's analysis. This evaluation found an incremental increase in k_{eff} that accumulated to 0.65% when the portion of the poison plates on all four sides around the center assembly had the lower ^{10}B content. Based on these results the acceptance testing method for the poison plates was found adequate and this is discussed further in Section 8.

The second model included the ends of the cask and the space above the poison plates. This model was used to assess the situation where an upright cask is flooded just enough to cover the poison plates but there is air above the plates that could allow neutrons to scatter off of the lid and thus bypass the poison plates. The value of k_{eff} when air or water filled the space above the poison plates was below 0.95 and maintained a sufficient subcritical margin.

Overall, the staff's confirmatory analyses showed acceptable agreement with the applicant's results for the NUHOMS®-MP197 Transport Package and support the conclusion that there is reasonable assurance the package will remain subcritical under the normal conditions of transport and the hypothetical accident conditions.

6.4 Single Package Evaluation

The applicant performed a single package analysis to satisfy the requirements of 10 CFR 71.55(b). The single package model is based on the most reactive case for an infinite array of packages. To satisfy 10 CFR 71.55(b)(3), the applicant modeled the three cases of the DSC surrounded by only the inner steel (containment) shell; the DSC surrounded by the inner shell and the lead gamma shield; and the DSC surrounded by the inner shell, the gamma shield, and the outer steel shell. In all three cases, the canister was flooded with full density water and the package components were reflected by full density water on all sides.

The results of the single package calculations were very close to each other. In addition, the results were compared with the results from the package arrays described below and it was found that the array results equaled or bounded the single package in all cases.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.55(b), (d), and (e).

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

In order to achieve a transport index of 0, an infinite array of packages must be analyzed under the normal conditions of transport and the hypothetical accident conditions. The structural analysis of the package determined that there is insignificant difference in the condition of the fuel basket under normal conditions versus accident conditions. The only change in the applicant's model for the accident conditions calculation was the replacement of the neutron shield and its shell by water moderator. Thus, the applicant performed an analysis of an infinite array of packages under the hypothetical accidents conditions and used those results to bound the package under the normal conditions of transport.

The staff accepts the accident array calculations as bounding the normal conditions case.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

6.6.1 Standard Analysis

The modeling assumptions for the hypothetical accident conditions are described in Section 6.3.1. An accident analysis was performed for an infinite array of packages containing fuel enriched to 4.4 wt. % in ^{235}U with full density water inside the packages and 70% density water between packages. The applicant also performed calculations of k_{eff} for an infinite array of packages containing the maximum enrichment allowed for each of the three levels of boron in the poison plates. These last three calculations assumed full density water inside and between the packages.

In all cases, the results for an array of packages under hypothetical accident conditions gave a value of k_{eff} which is less than the corresponding USL.

6.6.2 Special Considerations

The applicant's model used for the analysis of an array of packages under hypothetical accident conditions assumes an infinitely long active fuel section without any cask ends. This model assumes that the fuel in each assembly is always shielded from the other assemblies by a poison plate. However, there is as much as 19 inches of space above the poison region of the basket plates and the applicant was asked to show that the fuel assemblies and basket could not shift sufficiently to allow the active fuel to be uncovered by the poison plates. The applicant provided an evaluation considering all basic fuel types and showed that under normal conditions the neutron absorber would overlap the fuel by at least 0.96 inches for the worst case assembly dimensions. The staff reviewed and accepts the evaluation for normal conditions.

The applicant also evaluated the configuration under hypothetical accident conditions. The accident evaluation considered a top drop with a shift of the fuel rods as well as a shift of the basket. In this evaluation, the applicant determined that the minimum overlap would be 1.79 inches. The staff accepts the shift of the fuel rods but does not agree that a similar shift of the basket will always occur. In the applicant's top drop evaluation, the fuel pins shift toward the upper tie plate until bottoming out the compression springs or contacting the upper tie plate, depending on design. The applicant also calculated a deflection of the handling bail of no more than 0.071 inches. The staff applied the sum of these two movements to the results for the normal conditions and determined that the poison plates continue to overlap the fuel even if the basket does not shift as assumed by the applicant.

The staff reviewed the applicant's evaluation, and although it disagrees with the assumption that the basket moves in the accident analysis, the staff accepts the applicant's overall conclusion that the package meets the regulatory requirements of 10 CFR 71.59(a)(1), 71.59(a)(2), and 71.59(b). The staff agrees that the Transportation Index based on criticality safety is shown to be 0.

6.7 Benchmark Evaluations

6.7.1 Experiments and Applicability

The applicant performed benchmark comparisons on 125 selected critical experiments that were chosen to model a wide range of uranium enrichments, fuel pin pitches, water/fuel ratios, assembly separation, and control elements. The staff reviewed the benchmark experiments chosen and found that the benchmark parameters provided adequate coverage of the range of parameters for the fuel assemblies and the cask design.

6.7.2 Bias Determination

The applicant used the method recommended in NUREG/CR-6361[3] to calculate a USL. Separate USL ranges were calculated for the five parameters of enrichment, fuel rod pitch, water/fuel ratio, assembly separation, and average energy group causing fission which matched the values of the fuel assemblies to be transported in the NUHOMS®-MP197. All parameters were evaluated for trends and to determine the minimum USL for each parameter. The applicant then selected the minimum value among these five parameters which resulted in an overall USL of 0.9414. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations.

The staff reviewed the applicant's method for determining the USL and concluded that it is sufficient to provide a basis for the criticality evaluation of the package.

6.8 Evaluation Findings

F6.1 Based on review of the statements and representations in the application, supplemental information supplied by the applicant, and the analyses performed by staff, the staff has reasonable assurance that the package design meets the criticality safety requirements of 10 CFR Part 71.

6.9 References

1. U.S. Code of Federal Regulations, "Packaging and Transportation of Radioactive Material," Title 10, Part 71, January 1, 2001.

2. U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, April, 1997.

3. U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," NUREG/CR-6361, March, 1997.

4. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,” NUREG-1617, March, 2000.

5. U.S. Nuclear Regulatory Commission, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” NUREG/CR-0200, Vol. 1-5, Rev. 6, May, 2000.

7.0 OPERATING PROCEDURES

The objective of this review is to verify that the operating controls and procedures meet the requirements of 10 CFR Part 71 and that the operating procedures are adequate to assume the package will be operated in a manner consistent with it's evaluation for approval.

The staff reviewed the NUHOMS®-MP197 Transport Package operating procedures to ensure that the transportation package will be operated in accordance with 10 CFR Part 71, and in particular, to ensure that the preparation for loading, the loading of contents, preparation for transport, receipt of package from the carrier and removal of the contents are done in accordance with regulatory requirements.

F7.1 The staff has reviewed the proposed special controls and precautions for transport, loading, unloading, handling, and any proposed special controls in case of accident or delay, and found reasonable assurance that they satisfy 10 CFR 71.35(c).

F7.2 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.

F7.3 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.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The objective of this review is to verify that the acceptance tests for the packaging comply with the requirements of 10 CFR Part 71 for the package design and that the maintenance program will ensure acceptable packaging performance throughout it's service life.

The staff reviewed the NUHOMS®-MP197 Transport Package acceptance tests to ensure that the transportation package will be tested and constructed in accordance with the design described in the SAR, and in particular, to ensure that the sequence established to evaluate the package against applicable 10 CFR Part 71 requirements has been maintained.

8.1 Acceptance Tests

Section 8.1 of the application specifies review, inspection, and testing of the package. The acceptance tests and inspections considered critical to the safe operation of the NUHOMS®-MP197 were captured within the CoC.

8.1.2. Visual Inspections and Measurements

The licensee has committed that the NUHOMS®-MP197 packaging welds shall be examined, the sealing surfaces inspected, and dimensions inspected to ensure conformance with the drawings contained in Chapter 1 of the SAR. The 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.

8.1.3 Weld Inspections

In general, the licensee proposed to both design and construct the NUHOMS®-MP197, and the NUHOMS®-61PT DSC in strict compliance with the ASME Boiler and Pressure Vessel (B&PV) Code. Non destructive examination requirements for the welds are delineated in Chapter 1 of the SAR. The licensee committed to have all NDE performed in accordance with written procedures and inspection personnel qualified in accordance with SNT-TC-1A.

The NUHOMS®-MP197, and the NUHOMS®-61PT DSC containment welds are designed fabricated, tested, and inspected as in accordance with ASME B&PV Code, Section III, Subsection NB. The NUHOMS®-61PT DSC fuel basket is designed, fabricated, and inspected as in accordance with ASME B&PV Code, Subsection NG. Exceptions to the Code have been described in Chapter 2.11 of the SAR. Welds of the non-containment structure are inspected as per the NDE acceptance criteria of ASME B&PV Code, Subsection NF.

8.1.4 Structural and Pressure Tests

The NUHOMS®-MP197 cask containment boundary design pressure is 2.4 atm (50 psig), which is greater than the MNOP of 1.37 atm (5.4 psig). The containment vessel will be tested to 62.5 psig. Accessible weld and material inspections will be performed after the pressure test to verify that structural integrity has been maintained.

8.1.5 Leakage Tests

The fabrication verification leak tests performed on the NUHOMS®-MP197 cask are divided into five areas 1) cask leakage integrity, 2) cask vent port closure seal integrity, 3) cask drain port closure integrity, 4) cask top closure (lid) seal integrity, and 5) ram closure plate seal integrity. Testing is performed in accordance with ANSI N14.5, and personnel performing the leakage test are qualified in accordance with SNT-TC-1A.

The testing of the cask containment integrity 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 $x10^{-7}$ ref cm^3/s. The test will be performed in conjunction with the non-destructive examination of the inner shell welds in accordance with ASME B&PV Code, Section III, Subsection NB, a Penetrant Test (PT) examination of every weld layer in the shell to top forging closure weld, and a PT examination of all the final machined weld surfaces of the inner shell per the ASME Code.

Fabrication leak tests will include the following:

- Testing the cask vent port closure bolt seal integrity,
- Testing the cask drain port closure integrity,
- Testing the cask top closure (lid) seal integrity, and
- Testing the ram closure plate seal integrity.

Testing is performed in accordance with ANSI N14.5 and acceptance testing criteria shall be that each component must be individually leak tight, i.e., 1 $x10^{-7}$ ref cm^3/s.

8.1.6 Component Tests

There are no valves or couplings in the NUHOMS®-MP197 packaging.

Lid and other containment penetrations are sealed using double elastomeric seals. Leak testing of these components is as described above in Section 8.1.5 of this SER and Chapter 8.1.3 of the SAR.

Each impact limiter container is tested to a pressure of approximately three-psig. All weld seams and surfaces are tested for leakage using a soap bubble test. This will ensure that the impact limiter wood has been protected from any moisture exchange with the environment.

8.1.7 Shielding Tests

8.1.7.1 Neutron Shield Tests

The applicant described the properties of the radiation neutron shield. This includes the polyester-resin shielding material, the procedures for mixing and pouring the polyester resin as well as the methodology of fabrication and testing of these materials and the requirement for qualification testing of personnel. For the radial neutron shielding, a polyester resin is used. The nominal values of hydrogen and boron contents, as tabulated in Chapter 8.1.5.1 of the SAR, are ensured using qualification tests of the personnel and procedures used for mixing and pouring the shield material, so as to ensure that the shield materials are poured in a manner that prevents void formation. The staff reviewed the information presented and found it to satisfy the minimum specification. The staff concluded that the neutron shielding will maintain its properties during service.

8.1.7.2 Gamma Shield Test

In the SAR and in details furnished in responses to staff questions, the applicant described the fabrication procedures used to ensure the integrity of the poured lead shielding. For the gamma shield, in addition to process control on the pouring of the lead, measurements are taken using appropriate acceptance criteria to ensure quality and function. The integrity of the shield is confirmed via gamma scanning prior to installation of the neutron shield. The staff reviewed the information provided and has no safety concerns related to the adequacy of this material in performing its safety function throughout the life of the system.

8.1.7.3 Neutron Absorber Tests

In general, materials used in this application were previously approved by staff for use in the -61BT DSC. However, potential non-uniformities in the neutron absorber materials (poison plates) and the acceptance testing may exist, as developed for application in the NUHOMS® -MP197 package. The approved acceptance tests did not provide adequate assurance that each plate being placed into service had been checked for uniformity of the distribution of ^{10}B within the plate. The potential safety-related effects of non-uniformities within the neutron absorber plate materials were reviewed.

The neutron absorber materials are of two types, which receive differing credit for the specified minimum boron content of the absorber plate materials, as listed in Table 8-1 of the SAR. While statistical data is compiled from the plates of a heat or lot during production of absorber plate materials, this data serves principally to ensure that the minimum required ^{10}B content is present in the production plates. It is not practical to directly measure the uniformity of ^{10}B within each of the plates in a heat or lot of material during or after production.

The staff conducted additional calculations, which assumed (for the most reactive fuel approved for use in the basket of the -61BT) that from one to four plates of the neutron absorber material contained only 50% of ^{10}B content credited in the applicant's criticality safety analysis. These selected calculations were performed for plates made of Boral®, an absorber material, for which credit is taken for only 75% of the ^{10}B shown to be present. Other approved materials are given credit for 90% of the ^{10}B shown to be present. The results of calculations using the 75% or 90% credit would be the same whether the assumed absorber material is the Boral® or one of the other absorber materials. In the SAR the minimum specified boron content is adjusted by the applicant so that each material has essentially the same credited amount of ^{10}B.

The applicant's standard calculation, gave an effective neutron multiplication factor, K_{eff}, of 0.93637, with a standard deviation of 0.00118. Staff calculations showed that the potential non-uniformities do not pose an unacceptable risk. To show this the following calculations were made: The level of credit assumed for the amount of boron present was halved (decreased from 75% to 37.5%), and four sets of calculations were made using the assumptions that (a) one, two, three, or four full absorber plates contained the decreased level of credited ^{10}B content, and (b) the affected plates are those that surround an assembly at the center of the basket. The calculated results yielded K_{eff} values of: 0.93846, 0.93938, 0.93981, and 0.94291, respectively. Each of these calculated results exceed those of the standard calculation and, as expected, the values increase with increasing numbers of affected plates, but K_{eff} does not exceed the allowable neutron multiplication factor limited to 0.95.

The results of these criticality calculations resolved staff concerns regarding the significance of the potential non-uniformity of the ^{10}B content in absorber plate materials. In the approval of absorber plates for the -61BT, for materials to be given 90% credit, for the ^{10}B, a combination of results of qualification tests and acceptance tests were used. These qualification tests included direct tests of uniformity of the boron distribution throughout an entire plate of production material. In acceptance tests, the results of transmission measurements, which were conducted at one or more spots (1 cm diameter) on coupons taken from production plate materials, were used to demonstrate efficacy (absorptivity) of the ^{10}B in the plate materials. As an additional acceptance test, the boron variability over the entire area of the coupon was made by radiographic or other means. The transmission results for each production batch of plates as well as the results of each spot test must pass vigorous statistical requirements to ensure that the minimum specified content will be present in each plate and throughout the batch.

Acceptance test measurements are performed on coupons taken from strips located between adjacent absorber plates of a production run. When these measured values for a batch of plates have little statistical variability, it supports the conclusion that large areas can not contain less than the minimum specified value. This conclusion is also supported when, as is often the case, the mean value is also much higher than the minimum required value. The statistical analysis ensures, with 95% confidence, that no more than 5% of the area is below the minimum specified value. The concern is whether or not there is a significant risk associated with such hypothetical non-uniformities. This area, with low values representing only five percent of a plate, can potentially be in clusters (in linear streaks, or in discrete blocks) due to production effects on homogeneity. These concerns relate to both the B_4C particles dispersed in an aluminum alloy matrix of a metal matrix composite and to the boron-containing precipitates present in an enriched aluminum alloy.

The staff finds that questions relating to the potential effects of non-uniformities have been resolved by the results of the criticality calculations described above. The analytical results indicate that, even with changes in the credit taken for the boron within a critical location in the basket assembly, the effect of this potential variability on computed values of k_{eff} are acceptable for the configuration and fuel approved for use with the NUHOMS®-MP197. The assumed variability is found to be acceptable assuming three unlikely conditions or events: (1) a level of the boron much lower than that expected, (2) a distribution that more highly localized than could be expected from the statistical data taken on coupons, and (3) an area (with low boron content) that is much larger than what the statistical requirements of the acceptance tests would permit.

8.1.8 Thermal Tests

The staff determined that the cask need not be subjected to a thermal heat rejection test to demonstrate satisfactory operation of the as-built shells, top lid, and shielding materials due to the following reasons:

(a) The maximum decay heat load of the cask is relatively low (15.86 kW).
(b) The cask design has appreciable margin as indicated in the thermal analysis presented in Chapter 4 of the SAR. None of the temperature limits of the materials of

the cask are approached according to the analysis presented in the SAR, which has been confirmed by the staff.

(c) The analysis as presented in the SAR made conservative assumptions that would tend to over predict the temperatures of cask components and fuel cladding.

(d) The cask design does not employ new fabrication or heat rejection techniques that have not been proven by previous cask designs.

In the future, the staff may request a thermal heat rejection test of this cask design if changes are made to the design or the contents.

8.1.9 Evaluation Findings

F8.1.1 The staff has reviewed the identification of the codes, standards, and provisions of the QA program applicable to testing 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.

F8.1.2 The staff has reviewed the description of the preliminary determinations for package use prior to transport and found reasonable assurance that the requirements of 10 CFR 71.85 and 10 CFR 71.87(g) will be met.

8.2 Maintenance Tests

Section 8.2 of the SAR specifies a maintenance program for the package.

8.2.1 Structural and Pressure Tests

Within 14 months prior to any lift, each of the NUHOMS®-MP197 lifting (top) trunnions will be either:

(e) Load tested to 150% of the maximum working load per ANSI N14.6, paragraph 7.3.1(b) for non-single failure proof trunnions or load tested equal to 300% of the maximum service load per ANSI N14.6, paragraph 7.3.1(a) for single failure proof trunnions. The trunnions shall be visually inspected for defects, and all components shall be inspected for permanent deformation; or

(f) Dimensionally tested, visually tested, and nondestructive examinations performed of critical areas of the trunnions including the bearing surfaces in accordance with ANSI 14.6, paragraph 6.3.1. The examinations shall be performed for subsequent shipments.

8.2.2 Leakage Tests

The vent port, drain port, top closure plate, and ram port shall be subject to periodic maintenance and preshipment leakage testing in accordance with ANSI 14.5. Maintenance intervals and testing of the elastomeric seals are delineated in Chapter 8.2.2 of the SAR.

8.2.3 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 impact limiters shall be visually inspected before each shipment for weld cracking which could indicate water absorption or degradation of the wood. If there is visual damage the impact limiter will be removed from service, repaired if possible, and inspected for damage to the wood. Impact limiters shall be leak tested once every 5 years to ensure water has not entered the impact limiters. If the leak test indicates that the impact limiters have a leak, a humidity test shall be performed to verify that there is no free water in the impact limiters.

If a ram cover or lid is removed, the seals shall be replaced prior to spent fuel transport. The seals shall be leak tested after retorquing the bolts.

The elastomeric seals may be reused for transport of an empty NUHOMS®-MP197packaging.

There are no valves or rupture discs on the NUHOMS®-MP197 packaging containment.

8.2.4 Neutron Absorber Tests

After initial fabrication inspection, no further special maintenance is required. Radiation surveys shall be performed on the package exterior to ensure that the limits specified in 10 CFR 71.47 are met prior to shipment.

8.2.5 Thermal Tests

There are no periodic tests or inspections required for the NUHOMS®-MP197 packaging heat transfer components.

8.2.6 Evaluation Findings

F8.2.1 The staff has reviewed the identification of the codes, standards, and provisions of the quality assurance 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.

F8.2.2 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.

CONCLUSION

The staff concludes that the NUHOMS®-MP197 packaging will meet the requirements of 10 CFR Part 71. Pursuant to 10 CFR Part 71, Certificate of Compliance No. 9302 for the NUHOMS®-MP197 transportation package is approved. Several conditions have been included to reflect the use of the NUHOMS®-61PT DSC.

Issued with Certificate of Compliance No. 9302, Revision No. 0,
on _______________, 2002.