SAFETY EVALUATION REPORT
Docket No. 71-9302
Model No. NUHOMS®-MP197HB Package
Certificate of Compliance No. 9302
Revision No. 7

TABLE OF CONTENTS

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SUMMARY

By letter dated March 2, 2012, Transnuclear, Inc. (TN) submitted an amendment request to certificate of compliance (CoC) No. 9302 for the Model No. NUHOMS®-MP197HB package by adding high burnup fuel assemblies, as authorized contents, and including the 32PT, 24PTH, 32PTH, 32PTH1, and 37PTH dry storage canisters (DSCs) to transport high burnup fuel. The application was supplemented on August 8, 2013, January 10, March 12 and April 22, 2014. Revision No. 15 of the package application, dated January 10, 2014, as supplemented, supersedes in its entirety the application dated March 2, 2012.

The packaging is used to transport several types of boiling *w*ater *r*eactor (BWR) fuel assemblies with or without fuel channels, or pressurized *w*ater *r*eactor (PWR) fuel assemblies with or without control components, in DSCs, with a burnup not to exceed 62 GWd/MTU. In addition, the packaging is also used to transport a secondary container with dry irradiated and/or contaminated non-fuel bearing solid materials. The packaging is designed for a maximum heat load of 32 kW depending (i) on the DSC being transported and (ii) the package configuration.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). To address uncertainties on high burnup fuel cladding material properties, a multi-pronged licensing approach was used:

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NRC staff reviewed the application using the guidance in "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617, and associated Interim Staff Guidance (ISG). Based on the statements and representations in the application, and the conditions listed in the certificate of compliance, the staff concludes that the package meets the requirements of 10 CFR Part 71.

References

NUHOMS®-MP197 Transportation Package, Safety Analysis Report, Revision 15, Document No. NUH09.0101, dated January 10, 2014, as supplemented March 12 and April 22, 2014.

1.0 GENERAL INFORMATION

1.1 Packaging

The Model No. NUHOMS®-MP197HB packaging consists of a (i) containment boundary, (ii) structural shell, (iii) gamma shielding materials, and (iv) solid neutron shielding materials.

The containment boundary consists of the cylindrical inner shell, a 6.50 inch thick bottom plate with a 28.88 inch diameter, a 2.50 inch thick ram access closure plate with seal and bolts, a packaging body flange, a 4.50 inch thick lid with seal and bolts, vent and drain ports with closure bolts and seals, and all containment welds. The cavity of the packaging has a 70.50 inch diameter and is 199.25 inch long. To preclude air in-leakage, the cavity is pressurized with helium to above atmospheric pressure. The inner containment shell is SA-203, Grade E, and the bottom and top flange materials are SA-350-LF3. The lid is constructed from SA-350-LF3 or SA-203, Grade E.

The packaging is fabricated primarily of nickel-alloy steel (NAS). The body of the packaging consists of a 1.25 inch thick, 70.50 inch inside diameter NAS inner (containment) shell and a 2.75 inch thick, 84.50 inch outside diameter NAS structural shell which "sandwich" the 3-inch thick cast lead.

The lead and steel shells of the packaging provide gamma shielding while neutron shielding is provided by a borated resin compound, cast into slender aluminum tubes, surrounding the outer shell. The total thickness of the resin and aluminum is 6.25 inches. In addition to serving as resin containers, the aluminum provides a heat conduction path from the body of the package to the neutron shield shell.

Removable front and rear trunnions, which are bolted to the outer shell of the packaging, provide support, lifting, and rotation capability for the package. Impact limiters, consisting of balsa and redwood encased in stainless steel shells, are attached with 12 bolts to each end of the package during shipment. Each has an outside diameter of 126 inches and a height of 58 inches. A thermal shield is also provided between each impact limiter and the package to minimize heat transfer to the impact limiters. Each impact limiter is provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. A personnel barrier is mounted to the transport frame to prevent unauthorized access to the package body.

All of the nine DSCs consist of a cylindrical shell, top and bottom shielded cover plate closures, and a fuel basket assembly. For some DSC designs, a basket "hold down" ring is installed on top of the basket, after fuel loading, to prevent axial motion of the basket within the DSC. Dry irradiated and/or contaminated non-fuel bearing solid materials are contained in a radioactive waste canister (RWC).

For a NUHOMS® -69BTH DSC with a heat load greater than 26 kW, the use of removable external fins is required. The Model No. NUHOMS® -MP197HB packaging is transported in the horizontal orientation, on a specially designed shipping frame, with the lid end facing the direction of travel. DSCs with a spent fuel payload are shipped dry in a helium atmosphere.

Table 1 below shows nominal dimensions (in) and weights (lb) of the Model No. NUHOMS®-MP197HB packaging:

Table 1

Packaging overall length with impact limiters and thermal shield	271.25
Impact limiter outside diameter	126.0
Packaging outside diameter (w/o impact limiters and fins)	97.75
Packaging outside diameter with fins (w/o impact limiters)	104.25
Cavity inner diameter	70.50
Cavity length (minimum)	199.25
Lead gamma shield radial thickness	3.0
Packaging lid thickness	4.5
Packaging bottom thickness	6.5
Weight of Contents (maximum)	112,000
Empty weight of packaging without lid or impact limiters	157,500
Impact limiters, thermal shield, and attachments	25,000
Total loaded weight of MP197HB Packaging	303,600

1.2 Contents

The contents of the Model No. NUHOMS®-MP197HB packaging are (i) fuel assemblies stored inside anyone of the nine DSCs, and (ii) dry irradiated and/or contaminated nonfuel bearing solid materials in an RWC.

The types of the fuel assemblies and their maximum quantity to be loaded in any of the nine DSCs are specified below:

DSC Type	Applicable Fuel Specifications from the application
NUHOMS® -24PT4	Tables A.1.4.1-1 and A.1.4.1-2
NUHOMS® -32PT	Table A.1.4.2-2
NUHOMS® -24PTH	Table A.1.4.3-2
NUHOMS® -32PTH	Table A.1.4.4-2
NUHOMS® -32PTH1	Table A.1.4.5-2
NUHOMS® -37PTH	Table A.1.4.6-2
NUHOMS® -61BT	Table A.1.4.7-2
NUHOMS® -61BTH	Table A.1.4.8-2
NUHOMS® -69BTH	Table A.1.4.9-1

The maximum peaking factor of the fuel assembly average burnup in all fuel contents shall not exceed 1.212 and 1.152 for BWR and PWR fuel, respectively, for burnups greater than 45 GWd/MTU.

For both the Model Nos. NUHOMS®-MP197 and the NUHOMS®-MP197HB packages, fuel assemblies with missing fuel rods shall not be shipped as intact fuel unless the missing fuel rods are replaced with dummy rods, or fuel rods with an enrichment lower or equal to the original assembly enrichment and 2/3 of the assembly burnup, with the same nominal weight as a standard fuel assembly, that displace an equal or greater amount of water.

The NUHOMS®-24PT4 DSC basket is designed to accommodate up to 24 intact or a combination of intact and damaged PWR fuel assemblies. The 24PT4 DSC can hold up to 12 damaged Westinghouse-CENP 16x16 (CE 16x16) fuel assemblies in specially designed Failed Fuel Cans (FFCs) with the balance being loaded with intact fuel. The fuel has a maximum initial enrichment of 4.85 wt. % U-235, a maximum allowable assembly average burnup of 60,000 MWd/MTU, and a minimum cooling time of 15 years. Reconstituted assemblies are authorized contents provided that these fuel assemblies contain no more than eight replacement stainless steel rods or replacement Zircaloy clad uranium rods in places of damaged fuel rods.

The NUHOMS®-24PTH DSC baskets are designed to accommodate 24 intact or a combination of intact and up to 12 damaged fuel assemblies. The intact fuel assemblies may also contain control components. The fuel is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235, and a maximum allowable assembly average burnup of 62 GWd/MTU. Minimum cooling time requirements are given in Table A.1.4.3-2 of the application but additional cooling time is required for fuel loaded at the peripheral locations and with burnup greater than 51 GWd/MTU. Intact fuel assemblies transported in the 24PTH DSC may include reconstituted assemblies containing replacement rods up to 10 stainless steel rods per assembly. There is no restriction on the number of lower enrichment UO_2 replacement rods per assembly.

The NUHOMS®-32PTH DSC baskets are designed to accommodate 32 intact or up to 16 damaged with the remainder for intact PWR fuel assemblies with or without control components. There are two designs for the NUHOMS®-32PTH DSC, namely 32PTH and 32PTH1 or 32PTH Type 1, the difference between the 32PTH and 32PTH Type 1 design being the length of the basket. The maximum allowable assembly average burnup is 62 GWd/MTU.

The NUHOMS®-37PTH DSC baskets are designed to hold 37 all intact, or up to 4 damaged plus the remainder intact, PWR fuel assemblies with or without control components. There are two designs for the NUHOMS®-32PTH DSC, namely 37PTH-S and 37PTH-M, to accommodate PWR fuel assemblies with different lengths. Spacers, instrument tube tie rods, and anchors that were used to facilitate handle of fuel assemblies may also be loaded as part of the fuel assemblies. Reconstituted assemblies may contain up to 10 replacement irradiated stainless steel rods or stainless steel clad rods per assembly.

The NUHOMS®-61BTH DSC baskets are designed to accommodate 61 intact, or up to 16 damaged with up to four (4) Failed Fuel Cans (FFCs) loaded with damaged fuel with the remainder intact BWR fuel assemblies with or without fuel channels. The maximum burnup of the allowable BWR fuel assemblies for the 61BTH cask is 62 GWd/MTU. The 61BTH DSC system includes three design configurations, 61BTH Type 1, 61BTH Type 2, and 61BTHF, to accommodate fuel assemblies with different lengths. The 61BTH DSC allows for reconstituted fuel assemblies that contain up to four replacement irradiated stainless steel rods per assembly or 61 non-zircaloy clad fuel with lower enrichment UO_2. The 61BTH DSC uses zoned loading. For the fuel assemblies in the peripheral zones, additional cooling time is required.

The NUHOMS®-69BTH DSC basket is designed to accommodate 69 intact, or up to 24 damaged plus 45 intact BWR fuel assemblies with or without fuel channels. The maximum burnup of the allowable BWR fuel assemblies for the 69BTH cask is 62 GWd/MTU. The 69BTH basket has a variety of complicated loading patterns, some requiring extremely low decay heat or dummy fuel assemblies in some zones.

The RWC is used to transport dry irradiated and/or contaminated non-fuel-bearing solid materials. Each RWC system includes an outer cylindrical shell assembly. The RWC system consists of two design configurations, a Welded Top Shield Plug Design (RWC-W) and a Bolted Top Shield Plug Design (RWC-B). The type and form of contents and their maximum quantity to be loaded in an RWC are specified below:

Applicable Content Specification for RWC

Type and Form of Material	BWR Control Rod Blades, BWR Local Power Range Monitors, BWR Fuel Channels, BWR Poison Curtains, PWR Burnable Poison Rod Assemblies, PWR and BWR Reactor Vessel and Internals.
Decay Heat Load	≤ 5kW
Loading	RWC is normally filled to capacity with high specific activity components placed in the center of the RWC. Shoring or spacers may be used.
Maximum Quantity of Material per Package	Maximum of 8,182A_2 (90,000 Ci of Co-60); surface contamination may be present. Maximum payload of 56 tons of dry irradiated or contaminated non-fuel bearing solid materials in the RWC.

1.3 Criticality Safety Index

The Criticality Safety Index (CSI) for the package is zero, as an unlimited number of packages will remain subcritical under the procedures specified in 10 CFR 71.59(a).

1.4 Drawings

The ninety four (94) licensing drawings for the Model No. NUHOMS®-MP197HB packaging and its associated nine (9) DSCs are listed in Chapter 1 of the application as well as in Condition No. 5(a)(5) of the CoC.

Licensing drawings NUHRWC-71-1001, Rev. 1, NUHRWC-71-1002, Rev. 1, and Figure A.1.1 in the application provide a general sketch of the MP197HB packaging system.

Engineering drawings MP197HB-71-1001 to 1014 provide details of the geometric dimensions and the structural materials of the packaging system.

NUH24PT4-71-1001 to NUH24PT4-71-1003, NUH24PTH4-71-1001 to NUH24PTH4-71-1009, NUH61BT-71-1000 to NUH61BT-71-1002, NUH61BTH-71-1100, NUH61BTH-71-1102 to NUH61BTH-71-1106, NUH69BTH-71-1000, NUH69BTH-71-1002, NUH69BTH-71-1002, NUH69BTH-71-1011, and NUH69BTH-71-1012 provide details of the fuel basket structures. NUHRWC-71-1003, Rev. 0 provides detailed information on the RWC design and geometric dimensions.

1.5 Evaluation Findings

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. NUHOMS®-MP197HB package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL REVIEW

The objective of the structural review is to verify that the structural performance of the package meets the requirements of 10 CFR Part 71, including performance under the tests and conditions for both NCT and HAC.

2.1 Structural Design

The applicant presented several analyses to address technical issues including (i) the demonstration of fuel integrity under NCT to satisfy 10 CFR 71.55(d)(2), and (ii) the demonstration of moderator exclusion.

2.1.1 Normal Conditions of Transport

2.1.1.1 Baskets

The applicant evaluated the following DSC basket configurations for a one foot drop under NCT: 24PTH, 32PT, 32PTH, 32PTH Type 1, 32PTH1 (Type 1 and Type 2), 37PTH, and 24PT4.

In the case of side drops, equivalent fuel weight with the addition of a dynamic load factor is the primary means of loading the basket components whereas, in the end drop case, only basket self-weight is considered. Thermal loads were also included in the analyses.

The structural analysis of the side drops was carried out using a quasi-static finite element analysis with a commercial finite element analysis (FEA) code, ANSYS, and the structural analysis of the end drops consisted of hand calculations. Stress evaluations, including comparisons with allowable stresses, were performed either using a non-linear elastic analysis or an alternate stress analysis, as described in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, and Appendix A.2.13.8 of the application.

Based on the results of analyses exhibited in Appendix A.2.13.8 of the application, all identified DSC basket designs meet the ASME Code Subsection NG requirements. Therefore, the baskets are structurally adequate for NCT.

2.1.1.2 DSC Shell

The MP197HB DSC shell assemblies are comprised of a cylindrical shell, cover plates, and shield plugs. Each DSC assembly was designed to support a basket assembly and provide containment for spent fuel assemblies.

Multiple DSC shell assembly designs were evaluated by the applicant. Each design was categorized into one of four groups based on geometric and loading similarities. Each group used a bounding payload weight, i.e., basket plus fuel assembly.

The four canister groups are the following: (1) 69BTH*, 37PTH, 32PTH, 32PTH Type 1, 32PTH1; (2) 61BT*, 61BTH Type 1*; (3) 61BTH Type 2*, 32PT, 24PTH; and (4) 24PT4, 24PTH-S-LC. The asterisk denotes previously approved DSCs.

The loading conditions considered in the evaluation of the DSC shell consist of inertial loads resulting from normal condition inertial loading (1 foot drop), internal /external pressures, and thermal loads.

Bounding acceleration loadings of 25 g and 30 g were used for the side drop and end drop with the exception of the Group 1 DSCs, where a normal side drop acceleration of 30 g was conservatively used. These accelerations are considered bounding because they are formulated by amplifying a baseline g load, using a dynamic load factor, and subsequently increased a final bounding value.

Three dimensional finite element analyses were performed using ANSYS in order to determine stresses in the DSCs due to transport loads. These included half symmetry models for the side drops, while 2D axisymmetric models are used for end drop and thermal expansion analyses.

Based on the results of analyses demonstrated in Appendix A.2.13.7 of the application, all the identified DSC shell assembly designs meet the ASME B&PV Code Subsection NB requirements.

2.1.2 Hypothetical Accident Conditions

2.1.2.1 Baskets

The applicant evaluated the following basket configurations for a one foot drop under HAC: 24PTH, 32PT, 32PTH, 32PTH Type 1, 32PTH1 (Type 1 and Type 2), 37PTH, 24PT4.

In the case of side drops, equivalent fuel weight with the addition of a dynamic load factor is the primary means of loading the basket components whereas in the end drop case, only basket self-weight is considered. Thermal loads were also included in the analyses.

The structural analysis of the side drops was carried out using a quasi-static finite element analysis in the commercial FEA code, ANSYS, and the structural analysis of the end drops consisted of hand calculations. Stress evaluations, including comparisons with allowable stresses, were performed using an elastic-plastic analysis for the side drop load cases and hand calculations were used for the end drop load cases. Non-linear elastic plastic finite element analyses were performed to calculate the critical loads for buckling.

The stress analysis of the baskets due to inertial and thermal loads were described in detail in Appendix A.2.13.8 and stress results and associated allowable stresses were reported.

Based on the results of analyses shown in the application, all identified basket designs meet the ASME Code Subsection NG requirements; therefore, the baskets are structurally adequate for HAC.

2.1.2.2 DSC Shell

The DSCs were evaluated for HAC in a similar fashion than for NCT. A notable exception was that HAC side drops and end drops were both performed using a bounding 75 g load. A buckling analysis was also presented by the applicant.

Based on the results of analyses shown in Appendix A.2.13.7 and A.2.13.15 of the application, including certain cases utilizing limit analysis, all the identified DSC shell assembly designs meet the ASME B&PV Code Subsection NB and Appendix F requirements.

2.1.2.3 Moderator Exclusion

Compliance with 10 CFR 71.55(e)(2) can be achieved using alternatives in ISG-19, which provides criteria for demonstrating that the worst case damaged condition of the package does not result in water in-leakage.

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2.1.2.4 MP197HB Packaging

The MP197HB packaging was evaluated for all regulatory requirements and found to be adequate. Specifically, containment functions were shown to be adequate by evaluating the packaging body, closure lids, seals, and bolts against loadings developed in NCT and HAC. Specific calculations were presented in Appendices A.2.13.1, A.2.13.2, A.2.13.3, A.2.13.6, and A.2.13.14 of the application.

2.1.2.5 DSC

The DSCs identified in this amendment request were evaluated for compliance to 10 CFR Part 71 requirements and found to be adequate. Specifically, containment functions were shown to be adequate by evaluating the canister body, closure lids, and welds against loadings developed in NCT and HAC. Due to the reliance on a second independent water barrier, the applicant specifically referenced the ability of the DSCs to successfully meet the requirements of 10 CFR 71.61 which specifies a 290 psi external pressure applied to the structure of interest. Specific calculations were presented in Appendices A.2.13.7, A.2.13.9, A.2.13.10, and A.2.13.15 of the application.

2.2 Materials

The DSCs, previously approved for storage of high burnup fuels (HBF) in one or more overpacks, are now to be used for transport either after having been loaded and stored for some undetermined period of time, or within a year after having been directly loaded wet from a pool. The MP197HB packaging was reviewed in a previous amendment for use with low burnup fuel (LBF). The only concern with high burnup fuel is the effect of any increased temperature or dose on the polymer seals, and the behavior of the lead in the gamma shield and resin in the neutron shield. This review only focuses on the effect of the additional radiation field and possible higher temperatures due to the HBF on these components (see Section 2.2.2.4).

Due to the dearth of mechanical properties of the high burnup cladding with reoriented hydrides after drying, the applicant presented a licensing approach, in section Appendix A.1.4.A of the application, that required to:

(i) Use the canister as a second barrier for moderator exclusion for evaluation of the system during HAC. Under HAC, the fuel will not be assumed to maintain its configuration during a side drop.

(ii) Verify that the fuel is intact, during NCT, prior to unloading the canister.

(iii) Make a shipment only to a facility that can handle damaged fuel.

A complete materials review was conducted for the seven DSCs when they were previously approved for storage and transportation of LBF. The expected materials properties of the fuel were reviewed and found acceptable when they were previously approved for storage. Therefore, the DSCs and fuels were only reviewed for degradation that might occur during storage that would affect issues impacting their behavior during transportation such as:

(i) Drying of the high burnup fuel in the 69BTH DSC during direct loading for transportation (see Section 2..2.1).

(ii) Degradation of the DSCs during storage and unloading from storage (see Section 2.2.2.1).

(iii) Degradation of the internal components of the canister important to safety- basket and neutron absorber (see Section 2.2.2.2).

(iv) Potential change in HBF condition during storage prior to transport, primarily the effect of hydride reorientation on the mechanical properties of the cladding (see Section 2.2.2.3).

(v) Definition of damaged fuel (see Section 2.2.2.3.1).

(vi) Condition of the fuel after transport (see Section 2.2.2.3.3).

Only the changes to the drawings made since the previous approval for low burnup fuel were checked. These changes, as indicated in Enclosure 4 of the RAI responses, were found acceptable by the staff.

2.2.1 Directly Loaded Canisters

The licensing approach when HBF is directly loaded from the pool is the same as for the transport of loaded canisters that had been in dry storage, except that the examination procedure, established in Figure A.7-3 of the application to determine if there are weld cracks, is not applicable since the canister is new. The drying procedure is specified in Appendices A.7.7.1 through A.7.7.10 of the application, and is the accepted procedure for drying in the Standard Review Plan, NUREG-1536.

2.2.2 Previously stored DSC

There are two types of situations for canisters that have been previously stored: 1) canisters that are in their initial storage period, and 2) canisters that are in a renewal period and are maintained under an approved aging management plan. The plans for assuring the suitability of canisters stored under the two different situations may be different and these differences are noted in Section 7.1.3 of the application.

2.2.2.1 Degradation of the DSCs during storage and unloading from storage

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2.2.2.2 Degradation of the internal components of the canister important to safety- basket and neutron absorber

The baskets in all the DSC are constructed of similar materials. The basket structure consists of welded ASME SA-240 Type 304 stainless steel tubes. Either non-structural neutron poison plates or ASME B 209 Alloy 100 aluminum spacer plates are on the inside surface of the tubes. The basket is open at each end except for the cells designed to accommodate damaged fuel that will have top and bottom caps. The cap design, materials specifications, and weld designations if applicable are given in the drawings for the individual DSCs. The basket is surrounded by stainless steel and aluminum type 6061 support rails to maintain orientation within the package.

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2.2.2.3 Potential change in content (HBF) condition during storage prior to transport

10 CFR 71.33(b)(3) requires that the application must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The description must include -- (b) With respect to the contents of the package –and (3) Chemical and physical form.

2.2.2.3.1. Damaged Fuel Definition

The definitions of damaged fuel in canisters previously loaded for storage are consistent with the definitions used in this amendment for transportation. The damaged fuel is canned and assumed to be in one of a number of modified configurations for criticality, thermal, and shielding calculations. Thus, the definition of damaged fuel is consistent with the definition, based on functionality, as specified in ISG-1, Rev. 2. The ability of the fuel in modified configurations to meet the criticality, shielding, and thermal requirements is evaluated in the thermal, criticality and shielding sections of the application. Since all damaged fuel is canned, and there are no mechanisms for degradation of the cans during storage, the physical and chemical condition of initial damaged fuel is in a known condition for transport after storage.

2.2.2.3.2 Uncanned Fuel

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2.2.2.3.3 Potential change in content condition during NCT

10 CFR Part 71 contains the following two requirements:

- 10 CFR 71.55(d)(2) - A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.71 ("Normal conditions of transport"); (2) The geometric form of the package contents would not be substantially altered; and
- 10 CFR 71.89 - Before delivery of a package to a carrier for transport, the licensee shall ensure that any special instructions needed to safely open the package have been sent to, or otherwise made available to, the consignee for the consignee's use in accordance with 10 CFR 20.1906(e).

Demonstrating compliance with these regulatory requirements for high burnup fuel is made difficult because of a limited availability of spent fuel cladding material properties which are necessary to perform analytical evaluations of fuel performance. The applicant is meeting these regulations by using a defense in depth approach:

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(a) Cladding mechanical properties.

The DSCs have been approved for storage of HBF. Most cladding, other than M5®, has both hydrogen contents above 200 ppm, and increased hoop stresses resulting from a larger fission gas production, and release at high burnup. Since the cladding will have to undergo a drying cycle, the presence of radial hydrides is expected. The applicant states in Section A.2.13.11.1 of the application that "the presence of radial hydrides has no effect on the longitudinal mechanical properties used to evaluate bending stress resulting from 1 foot side and end drop (both NCT). The staff agrees with this statement and evaluated the applicant's calculation of the mechanical properties of the cladding as if no radial hydrides were present. The modulus, yield of the Zircaloy cladding were calculated in Section A2.13.11.1 using the correlations developed by Geelhood and Beyer, a NRC recommended reference for Zircaloy properties of cladding without any hydride reorientation. The calculations of the mechanical properties were done at temperatures up to 400°C (750°F). The application calculated the appropriate mechanical properties as a function of temperature and tabulated them in Table A.2.13.11.1 of the application for use in the structural calculations for evaluating bending stress resulting from a 1 foot side and end drop.

(b) Cladding inspection program that is described in Chapter A.7.2.1 of the application.

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Thus, compliance with 10 CFR 71.89 is obtained.

2.2.2.4 Evaluation of expected thermal and irradiation behavior of Overpack/Cask

The expected temperatures of the Transport Cask (TC) components that would be affected by the higher heat load are summarized in the table below along with the limiting temperature for their satisfactory performance.

TC item		Maximum temperatures (°F)							Criterion limit
		24PT4	24PTH	32PTH	32PTH1	37PTH	61BTH	69BTH	
Lid seal	HAC		373					321	204°C (400°F)
Neutron shield	NCT	265	285	280	280	263	276	290	160°C (320°F)
Gamma shield	HAC		508					552	327.5ºC (621ºF)

HAC values for other DSCs are not reported (24PT4, 32PTH, 32PTH1, 37PTH, and 61BTH). The application states that, based on thermal analysis presented in Section A.3.3.1.1 and results shown in Table A.3–8 and Table A.3–9 for NCT, the maximum seal temperatures of MP197HB TC are bounded by: a) 69BTH DSC with 32 kW heat load in TC with external fins, b) 69BTH DSC with 26 kW heat load in TC without internal sleeve, and c) 24PTH DSC with 26 kW heat load in TC with an internal sleeve and without external fins.

The neutron shield material, VYAL B, is a proprietary vinyl ester resin mixed with alumina hydrate and zinc borates, which are added for their fire retardant properties. The resin is cast in 6063-T651 Al tubes which are protected from damage or loss by a coated SA-516-70 steel enclosure. The applicant provided short term testing data and modeling of the VYAL B neutron shield material to show it should be thermally stable at the maximum allowable temperature under NCT of 160°C (320°F). The staff agrees with this assessment. The maximum expected temperature of the neutron shield is 143° C (290°F), which is less than the allowable limit for normal use. The cask neutron shielding will be tested before each shipment (Section A.8.1.6.2 of the application) to ensure the dose limits are met.

Gamma shielding is provided by cast ASTM B29 copper lead shielding poured between the two shells of the containment vessel using a carefully controlled procedure. "To prevent melting of the gamma shield (lead) under NCT, an allowable maximum temperature of 327.5ºC (621ºF – melting point of lead) is considered for the gamma shield [Section A.3.1, Table 11]. The maximum expected gamma shield temperature is 288°C (552°F); lower than the melting point of the lead.

The lid is sealed to the cask body using dual fluorocarbon O-ring seals, as shown on Drawing MP197HB-71-1003. Each of the vent and drain ports is sealed with a single 3/4 inch SA-540, Grade B23, Class 1 bolt, having a gasket under the head of the bolt, that is tightened to the values shown in Drawing MP197HB-71-1006. The long term maximum temperature for the fluorocarbon seals (Viton O-rings) in the containment vessel for NCT and HAC is 204°C (400°F). This is the range specified by the Parker Handbook when the Viton is in contact with ordinary atmospheres. The seals are expected to reach temperatures no higher than 189°C (373°F), which is below the maximum operating temperature limit.

The O-rings are the only TC component expected to be affected by an increased radiation dose. According to Chapter A.5, Table A.5-1, of the application for NCT, the maximum dose rate at the surface of the package is approximately 52 mrem/hour gamma, and 111 mrem/hour neutrons. At the end of 1 year of continuous exposure, this would result in absorbed energy in the seals of about 8×10^4 rad, well below the threshold of polymer damage, generally about 10^6 rad. In addition, the seals will be changed after every shipment to minimize any radiation damage. The staff does not expect any radiation deterioration of the polymer seals in this application.

In all cases, the components are expected to remain below the limiting value. The staff does not expect any thermal or irradiation deterioration of the above components due to high burnup fuel content in this application.

2.2.2.5 Conclusions

The DSCs having been previously approved for storage, only those materials that affected the ability to meet requirements strictly related to transportation were evaluated. Since the MP-197HB package is very similar to the MP197 package previously approved, only those materials that changed, or the effects of all the materials to meet the transportation requirements, were evaluated in detail. The remainder of the packaging materials properties was only spot-checked.

(a) The staff agrees with the applicant that, if any galvanic, chemical, corrosive or radiological interactions take place, such interactions are limited by the design of the system or administrative controls and that the system meets the requirements of 10 CFR 71.43(d)(3).

(b) The staff expects that, with the specified inspections, and required actions, the welds and seals will be in a condition and have the materials properties to behave as expected over both the temperature and stress ranges expected for NCT and HAC, and not compromise the containment of radionuclides, as required in 10 CFR 71.43(f), 10 CFR 72.51(a)(1), and 10 CFR 71.51(a)(2).

(c) A procedure for determining the complete chemical and physical description of the contents of the DSCs was given, including a definition of damaged fuel meeting 10 CFR 71.33(b)(3), and 10 CFR 71.55(e)(1). The staff found that the fuel and cladding properties of the undamaged cladding were sufficiently accurate to analyze the behavior of the contents during NCT and HAC other than side drops, which are evaluated using a reconfiguration analysis and, as such, meet the requirements of 10 CFR 55(d)(2).

(d) The staff reviewed, in an SER for a previous amendment, the mechanical and thermal properties of the materials of construction and found them to be accurate and suitable for analysis of the behavior of the system over the ranges of temperature and stress applicable to this amendment to maintain containment and shielding, thus meeting the requirements of 10 CFR 71.33(a)(5), 10 CFR71.43(f), 10 CFR 71.47(a), 10 CFR 71.55(d)(4), and 10 CFR 71.55(e).

(e) The staff found that the absorbers were described in sufficient detail, with reasonable quality assurance and acceptance plans, to meet the requirements of 10 CFR 71.33(a)(5)(iii), 10 CFR 71.111, and 10 CFR 71.123.

(f) Applicable codes or defensible code alternates are listed. They were reviewed in an SER for a previous amendment and found to meet 10 CFR 71.31. The drawings are complete, containing the lists of the materials of construction, weld specifications, and acceptance codes meeting requirements of 10 CFR 71.107(a).

(g) As indicated by the tables A.3-8 to A.3-19 of the application, all the materials stayed below their temperature limits for both NCT and HAC, thus meeting the requirements of 10 CFR 71.33(a)(5), 10 CFR71.43(f), 10 CFR 71.47(a), 10 CFR 71.55(d)(4), and 10 CFR 71.55(e).

2.9 Evaluation Findings

The staff has reviewed the package structural design description and concludes that the contents of the application meet the requirements of 10 CFR 71.31. Staff reviewed the design criteria, acceptance criteria, loadings, and load combinations, and codes and standards and found them acceptable. The staff has reviewed the packaging structural evaluation and concludes that the application meets the requirements of 10 CFR 71.35.

The staff has reviewed the packaging structural performance for NCT and HAC. Calculation packages were reviewed to verify modeling methodologies, input values including material properties, and that the results from the dynamic and quasi-static evaluations were reasonable and conservative. Staff also verified values obtained from the analytical models by recreating reported data and associated graphical representations showing stress contours and deformed and un-deformed configurations of the package. The staff concludes that there will be no substantial reduction in the effectiveness of the packaging, and that the packaging has adequate structural integrity to satisfy the sub-criticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

3.0 THERMAL REVIEW

3.1 Review Objectives

The objective of the review of the Model No. MP197HB package is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC, and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.2 Description of the Thermal Design

3.2.1 Design Features

The Model No. NUHOMS®-MP197HB package, also called transport cask (TC) by the applicant, consists of multiple shells to transfer decay heat to the package outer surface. The package includes optional features such as aluminum internal sleeves to accommodate DSC diameters smaller than 69.75 inches. It also includes an aluminum shell with external circular fins. Aluminum boxes that contain the shielding material are designed to fit tightly against the steel shell surface, thus improving heat transfer across the neutron shield.

Heat dissipation from the package outer surface is via natural convection and radiation. White painted outer surface enhances thermal radiation. Steel-encased wood impact limiters provide protection to the lid and bottom regions during HAC. A personnel barrier, which consists of a stainless steel mesh attached, encases the package body between the impact limiters. It has an open area fraction of about 80%.

The TC can accommodate the following DSC types, as presented in the table below:

DSC Type	DSC OD (inches)	TC Sleeve	TC External Fins	Maximum DSC Heat Load for Transport (kW)
69BTH	69.75	No	No	26.0
			Yes (Optional)	29.2
			Yes (Optional)	32.0
61BTH Type 1	67.25	Yes	No	22.0
61BTH Type 2	67.25	Yes	No	24.0
61BT	67.25	Yes	No	18.3
37PTH	69.75	No	No	22.0
32PTH/32PTH Type 1	69.75	No	No	26.0
32PTH1 Type 1	69.75	No	No	26.0
32PTH1 Type 2	69.75	No	No	24.0
32PT	67.19	Yes	No	24.0
24PTH-S or –L (w/ Al inserts)	67.19	Yes	No	26.0
24PTH-S or –L (w/o Al inserts)	67.19	Yes	No	26.0
24PTH-S-LC	67.19	Yes	No	24.0
24PT4	67.19	Yes	No	24.0

The 69BTH DSC basket supports up to 69 BWR spent fuel assemblies. The basket structure consists of 9 and 6 compartment fuel cell subassemblies held in place by basket rails in combination with a hold-down ring. The compartment subassemblies are held together by welded stainless steel plates wrapped around the fuel compartments, which also retain the aluminum and/or neutron absorbing plates sandwiched between the fuel compartments. The aluminum and neutron absorbing plates provide the necessary criticality control and heat conduction paths from the fuel cells to the perimeter of subassemblies. The aluminum plates retained between the subassemblies provide a heat conduction path from the subassemblies to the perimeter of the basket. The space between the basket and DSC shell is bridged by solid aluminum transition rails. No convection heat transfer is considered within the basket. Heat transfer inside the fuel compartments is included through the use of the effective thermal conductivity model which includes both conduction and radiation heat transfer. Decay heat from the canister is conducted to the MP197HB TC inner shell via conduction and radiation.

The 37PTH DSC basket has 37 fuel compartments for PWR spent fuel assemblies. Each compartment accommodates aluminum and/or neutron absorbing plates that provide the necessary criticality control and heat conduction path from the fuel assembly to the basket grid. The space between the basket and the DSC shell is bridged by solid aluminum transition rails. No convection heat transfer is considered within the basket. Heat transfer inside the fuel compartments is included through the use of the effective thermal conductivity model which includes both conduction and radiation heat transfer. Decay heat from the canister is conducted to the MP197HB TC inner shell via conduction and radiation.

The design features of DSC types 61BT, 32PT, and 24PTH are described in Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-003, Rev. 11, Appendices K, M, and P, respectively. The design features of

24PT4 DSC are described in Updated Final Safety Analysis Report for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, ANUH-01.0150, Rev. 3, Appendix A. The design features of DSC types 61BTH and 32PTH1 are described in Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-003Appendices T and U, respectively. The design features of 32PTH and 32PTH Type 1 DSCs are described in Final Safety Analysis Report for NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel, Rev. 2.

3.2.2 Thermal Design Criteria

Several thermal design criteria are established by the applicant for the MP197HB package to ensure that the package meets all its functional and safety requirements. These criteria are listed below:

(a) Maximum fuel cladding temperature limits of 752°F (400°C) for NCT and 1058°F (570°C) for HAC are considered for the fuel assemblies with an inert cover gas.

(b) The fuel cladding temperature is limited to 400°C (752°F) for short term operations such as vacuum drying.

(c) Containment of radioactive material and gases is a major design requirement. Seal temperatures must be maintained within specified limits to satisfy the leak tight containment requirement. A maximum temperature limit of 400°F (204°C) is considered for the Fluorocarbon seals (Viton O-rings) in the containment boundary for NCT and HAC. The maximum operating temperatures for the metallic seals in the ram plate and test/drain port seals are 644°F (340°C) and 1100°F (593°C), respectively. A maximum seal temperature of 644°F (340°C) is considered for all metallic seals for thermal evaluation for NCT and HAC.

(d) To maintain the stability of the neutron shield resin, a maximum allowable temperature of 320°F (160°C) is considered for the neutron shield for NCT.

(e) A temperature limit of 445°F (229°C) is considered conservatively for polypropylene to prevent thermal degradation of resin in trunnion plugs.

(f) To prevent melting of the gamma shield (lead) under NCT, an allowable maximum temperature of 621°F (327.5°C – melting point of lead) is considered for the gamma shield.

(g) A temperature limit of 320°F (160°C) is considered for wood to prevent excessive reduction in structural properties at elevated temperatures.

(h) In accordance with 10 CFR 71.43(g) the maximum temperature of the accessible packaging surfaces in the shade is limited to 185°F (85°C).

(i) The NCT ambient temperature range is –20°F to 100°F (-29°C to 38°C) per 10 CFR 71.71(b).

3.2.3 Content's Decay Heat

The thermal analysis of the MP197HB TC loaded with existing DSC qualified for storage along with 69BTH and 37PTH DSCs is based on a range of a maximum total heat load of 18.3 to 32 kW.

Specific total decay heat per each canister type is provided in the table above. The MP197HB TC is designed to transport a payload of up to 56.0 tons of dry irradiated and/or contaminated non-fuel bearing solid material in secondary containers with a total decay heat load of 5 kW.

The permitted heat load zoning configurations (HLZC) for all DSCs are listed in Chapter A.1, Appendix A.1.4.1 through Appendix A.1.4.9, of the application. These design basis HLZCs are symmetrical and show maximum allowable heat load per FA and per DSC, which result in bounding maximum fuel cladding and DSC component temperatures. Possible asymmetry in HLZC (within specified FA and DSC limits) means reduction of heat load in a particular FA resulting in reduction of local and maximum temperatures of fuel cladding and DSC components. A peaking factor is considered along the active fuel length for calculation of the decay heat profile of the fuel assemblies as described in Section A.3.3.1.4 and Section A.3.3.1.6 of the application for 69BTH and 37PTH DSCs, respectively.

The staff reviewed the design features, design criteria, and content's decay heat of the MP197HB package. Based on the information provided in the application regarding these items, the staff determines that the application is consistent with the guidance provided in Section 3.5.1 (Description of the Thermal Design) of NUREG-1617. Therefore, the staff concludes that the description of the thermal design is acceptable.

3.2.4 Summary Tables of Temperatures

The summary tables of the package component temperatures, Tables A.3-11, A.3-12, A.3-13, and A.3-19 of the application, were reviewed. The components include spent fuel cladding, spent fuel basket, containment shell, neutron shield, packaging surface, impact limiters, primary closure lid, secondary closure lid, containment base plate, primary and secondary lid seals and aluminum basket shims. The temperatures are consistently presented throughout the application for both NCT and HAC. For HAC, the applicant presented the pre-fire, during-fire, and post-fire component temperatures. With the exception of the impact limiters and neutron shield, all components remain below their material property limits listed in Tables 3.2.10 to 3.2.12 of the application. The temperatures and design temperature limits for the package components were reviewed and found to be consistent throughout the application.

3.2.5 Summary Tables of Maximum Pressures

The summary tables of the containment pressure under NCT and HAC (Tables A.3-20, A.3-22, and A.3-23) were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation sections of the application. These tables reported the Maximum Normal Operating Pressure (MNOP) for normal conditions of transport and hypothetical accident conditions (fire). These pressures remain below the design pressures for NCT and HAC.

The staff reviewed the design description of the Model No. MP197HB package thermal design and finds it acceptable. The staff reviewed the temperature and pressure design limits and calculated temperatures and pressures for the package and found them to be acceptable and consistent in the application.

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

The package application provided material thermal properties such as thermal conductivity, density, specific heat, and emissivity for all modeled components of the package. The staff found these properties acceptable. The applicant specifies the natural convection heat transfer coefficient as a function of the product of Grashof and Prandtl numbers. This product is a function of length scale, surface-to-ambient temperature difference, and air properties. The thermal properties used for the analysis of the package were appropriate for the materials specified and for the conditions of the cask required by 10 CFR Part 71 during NCT and HAC.

The staff reviewed the thermal properties used for the package analyses and determined that they were appropriate for the materials specified and for the package conditions required by 10 CFR Part 71 during NCT and HAC.

3.3.2 Component Specifications

The application provided component thermal technical specifications for the MP197HB containment seals and poison plates used in the DSC basket.

The MP197HB TC seals are Fluorocarbon seals (Viton O-rings). The seals will have a minimum and maximum temperature rating of -40°F and 400°F, respectively. The metallic seals will have a minimum and maximum temperature rating of -40°F and 644°F, respectively.

The 69BTH DSC design allows the use of different neutron absorber materials based on the heat load zoning configuration (HLZC). Boral, Metal Matrix Composite (MMC), or Borated Aluminum can be used as poison materials for HLZC # 1, # 2 and # 3 in the 69BTH basket. For the 69BTH basket with HLZC # 4, only borated aluminum can be used as poison material. The HLZCs for 69BTH are described in Section A.3.3.1.4 of the application.

The 37PTH DSC design allows the use of different neutron absorber materials. Boral plates paired with Al1100 plates or single plates of metal matrix composite (MMC) or borated aluminum can be used as poison materials in the 37PTH basket with a 22 kW heat load.

3.4 Thermal Evaluation under NCT

3.4.1 Thermal Models

The applicant developed three finite element models to analyze the MP197HB TC loaded with a DSC. The applicant used these models to analyze the case where the physical configuration of the fuel assemblies is not altered.

A half-symmetric, three-dimensional (3-D) finite element model of the MP197HB TC is developed using the ANSYS code. ANSYS is a thermal, structural and fluid flow analysis package capable of solving steady state and transient thermal analysis problems in one, two, or three dimensions. Heat transfer via a combination of conduction, radiation, and convection can be modeled by ANSYS. The applicant's MP197HB TC model contains the cask shells, cask bottom plate, cask lid, impact limiters, DSC shell, and DSC end plates without the basket. The DSC dimensions correspond to nominal DSC dimensions listed in Table A.3–1 of the application for variations of the MP197HB TC model. The applicant considered the following gaps in the MP197HB TC thermal model:

a) 0.0625" axial gap between thermal shield and impact limiter shell
b) 0.0625" axial gap between thermal shield standoffs and the cask top or bottom end surfaces
c) 0.10" diametrical gap between cask lid and cask inner shell

d) 0.01” axial gap between cask lid and cask flange
e) 0.01” axial gap between ram closure plate and cask bottom plate
f) 0.01” radial gaps between neutron shield boxes and surrounding shells
g) 0.025” radial gap between gamma shield and cask outer shell
h) 0.01” radial gaps between the cask inner shell and aluminum sleeve
i) 0.01” radial gap between the finned aluminum shell and the cask shield shell
j) 0.0625” axial gaps between the DSC bottom shield plug and bottom cover plates
k) 0.0625” axial gaps between the DSC top inner cover and the adjacent top shield plug and top outer cover plate
l) 0.025” axial gaps between the lead shield plugs and encapsulating plates for 24PT4 DSC
m) 0.01” gaps between trunnion replacement plugs and the trunnion attachment blocks

A half-symmetric, three-dimensional finite element model of the 69BTH basket and DSC is developed using ANSYS. The model contains the DSC shell, the DSC cover plates, shield plugs, aluminum rails, basket plates, and homogenized fuel assemblies. The applicant considered the following gaps in the 69BTH DSC canister/basket model at thermal equilibrium:

a) 0.30” diametrical hot gap between the basket outer surface and the canister inner surface
b) 0.125” axial gap between the bottom of the basket and the DSC bottom inner cover plate
c) 0.01” gap between any two adjacent plates or components in the cross section of the basket
d) 0.125” gap in axial direction between the aluminum rail pieces
e) 0.01” gap between the sections of the paired aluminum and poison plates in axial direction.
f) 0.1” gap between the two small aluminum rails at the basket corners.
g) 0.1” gap between the two pieces of large aluminum rails at 0° -180° and 90° - 270° orientations
h) 0.0625” gap between DSC shield plugs and DSC cover plates for calculation of effective conductivities in axial direction

A three-dimensional finite element model of the 37PTH DSC/basket model is developed using ANSYS. The model contains the DSC shell, the DSC cover plates, shield plugs, aluminum rails, basket plates, and homogenized fuel assemblies. The following gaps are considered in the 37PTH basket/DSC model at thermal equilibrium:

a) 0.45” diametrical hot gap between the basket outer surface and the canister inner surface
b) 0.45” diametrical hot gap between the shield plugs and the canister shell inner surface
c) 0.01” gap between the basket rails and compartment plates
d) 0.0075” gap between any two adjacent plates or components within the cross section of fuel compartments
e) 0.125” gap in axial direction between the aluminum rail pieces
f) Two pieces of MMC plates with 0.0075” contact gap are assumed to model single MMC plate in the model
g) 0.01” gap between any two adjacent plates between shield plugs and canister cover plates
h) 0.1” axial gap between the canister inner bottom plate and bottom basket assembly

The applicant used these models to perform steady state evaluations for NCT conditions. Decay heat load is applied as a uniform heat flux over the inner surface of the DSC shell covering the basket length. Radiation and conduction between the DSC and the TC inner shell/internal sleeve is considered by calculating effective conductivities for helium gaps between the components listed above. Insolation is applied as a heat flux over the TC outer surfaces using average insolation values from 10 CFR Part 71.

The insolation values are averaged over 24 hours and multiplied by the surface absorptivity factor to calculate the solar heat flux. The cask external fins are not considered explicitly in the TC model. Instead, an effective heat transfer coefficient is applied over the outer surface of the un-finned aluminum shell to simulate the heat dissipation from this area. Convection and radiation heat transfer from the “un-finned” cask surfaces are combined together as total heat transfer coefficients. The total heat transfer coefficients are calculated using free convection correlations.

The DSC shell temperatures for NCT are retrieved from the MP197HB TC model and transferred to the DSC/basket models to evaluate the maximum fuel cladding and basket component temperatures. Decay heat load is applied as volumetric heat generation rate over the elements representing homogenized fuel assemblies. The base heat generation rate is multiplied by peaking factors along the axial fuel length to represent the axial decay heat profile. A correction factor is used to avoid degradation of decay heat load due to imperfections in application of peaking factors.

The staff reviewed the applicant’s description of the package thermal models. Based on the information provided in the application regarding the developed thermal models, the staff determines that the application is consistent with the guidance provided in Section 3.5.3 (General Considerations for Thermal Evaluations) of NUREG-1617. Therefore, the staff concludes that the description of the thermal models is acceptable.

3.4.2 Heat and Cold

The applicant performed steady state analysis using the MP197 TC thermal model without insolation to determine the accessible surface temperature of the impact limiters in the shade. A heat load of 32 kW and boundary conditions at 100°F and no insolation are considered in the cask model to bound the maximum accessible surface temperature under shade. The maximum accessible surface temperature of impact limiter and personnel barrier under these conditions are 121°F and 152°F, respectively. The maximum temperature of the cask outer surface is 302°F and belongs to a part of shield shell uncovered by the external fins in the model.

Tables A.3–8 and A.3–9 of the application present the maximum temperatures for the TC components and DSC Shells.

The applicant stated that the DSC types 61BTH, 61BT, 32PTH, 32PTH1, 32PT, 24PTH, and 24PT4 were evaluated previously for NCT under 10 CFR 72 requirements. The applicant compared the DSC shell temperature profiles of these DSCs in MP197HB TC model to the corresponding profiles from the safety analysis reports (SAR) submitted under 10 CFR Part 72. The maximum DSC shell temperatures for NCT under 10 CFR Part 71 requirements are compared to the corresponding data for transfer conditions under 10 CFR Part 72 requirements in Table A.3–24. This table shows that the fuel cladding and the basket component temperatures in 10 CFR Part 72 SARs represent the bounding values for these DSCs under transport conditions.

The maximum temperatures for the DSC contents for all DSCs to be transported in the MP197HB TC for NCT are listed in Table A.3–10 of the application. The NCT thermal analysis demonstrates that the MP197HB TC with up to 32 kW heat load meets all applicable requirements. The highest maximum temperatures are summarized in Table A.3–11 of the application.

The seal O-rings are not explicitly considered in the models. The maximum seal temperatures are retrieved from the models by selecting the nodes at the locations of the corresponding seal O-rings. The maximum seal temperature (382°F, 194°C) for NCT is below the long-term limit of 400°F (204°C) specified for continued seal function. The maximum neutron shield temperature is 290°F

(143°C) for NCT, which is below the long term limit of 320°F (160°C). No degradation of the neutron shielding is expected. The maximum temperature of gamma shield is 397°F (203°C) for NCT, which is below the melting point of lead (621°F, 327°C). The predicted maximum fuel cladding temperature (733°F, 389°C) is below the allowable fuel temperature limit of 752°F (400°C) for NCT.

Under the minimum ambient temperature of -40°F (-40°C), the resulting packaging component temperatures will approach -40°F if no credit is taken for the decay heat load. Since the package materials, including containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the MP197HB TC. The maximum component temperatures for ambient temperatures of -40°F and -20°F with maximum decay heat and no insulation are calculated for the 69BTH DSC and the 37PTH DSC to use for structural evaluations. These temperatures are listed in Table A.3–12 and Table A.3–13 of the application.

The average temperatures of helium gas in the TC cavity, and the average temperatures of fuel assemblies and helium within the 37PTH and 69BTH DSC cavities for NCT are listed in Table A.3–14 of the application. These temperatures are used to evaluate the maximum internal pressures within the TC and DSC cavities. Thermal stresses for the MP197HB TC loaded with DSC are discussed in Chapter A.2 of the application. The staff confirms that the applicant's calculated maximum temperatures are below the material temperature limits with sufficient margin and finds them acceptable.

3.4.3 Maximum Normal Operating Pressure

The maximum pressures in the cask cavity calculated for a loaded MP197HB TC are presented in Table A.3–20 of the application. The case of the 69BTH DSC in MP197HB TC with 32 kW heat load is bounding for the maximum TC cavity pressure for all DSCs. The maximum pressure in the cask cavity of the MP197HB TC for NCT is 12.7 psig. Due to low heat load of the secondary containers and an expected low amount of fission gas release, the internal pressure of the package loaded with the DSCs is bounding for the package loaded with the secondary containers.

The maximum internal pressure for the 69BTH DSC within the MP197HB TC for NCT is determined based on the maximum allowable heat load for each HLZC discussed in Section A.3.3.1.4 of the application and the maximum assembly average burnup of 62 GWD/MTU. The maximum pressures are summarized in Table A.3–22 of the application. As seen in this Table, the maximum internal pressures in the 69BTH DSC are within the design limits.

The maximum internal pressure for the 37PTH DSC within the MP197HB TC for NCT is determined based on the maximum allowable heat load of 22 kW discussed in Section A.3.3.1.6 of the application and the maximum assembly average burnup of 62 GWD/MTU. The maximum pressures for 37PTH DSC are summarized in Table A.3–22 of the application. As seen in this table, the maximum internal pressures in 37PTH DSC are within the design limits.

The maximum internal pressures for the 61BTH, 61BT, 32PTH, 32PTH1, 32PT, 24PTH, and 24PT4 DSCs for storage and transfer conditions under 10 CFR Part 72 requirements are determined in the respective storage SARs. The applicant states that, since the maximum temperatures during NCT are bounded by the storage configurations, the pressures obtained during storage configuration also bound the NCT conditions.

The applicant also states that, since 100% fuel rod rupture is assumed for both transfer accident conditions and transport HAC, and maximum DSC shell temperatures for transfer accident conditions bound those for transport HAC as noted in Section A.3.4 of the application (which results in lower average DSC helium temperature), the internal pressures calculated for storage licensed DSCs for transfer accident conditions bound the internal pressures for transport in MP197HB TC during HAC.

The MNOP are below the containment design pressure, as reported in the application and therefore is acceptable. The staff reviewed selected calculations and results of the Model NUHOMS®-MP197HB package for NCT conditions and found them acceptable.

3.5 Thermal Evaluation During Drying Operations

After completion of fuel loading, the TC and DSC are removed from the pool and the DSC is drained, dried, sealed, and backfilled with helium. These operations occur when the annulus between the TC and DSC remains filled with water. Water in the DSC cavity is forced out of the cavity (blowdown operation) before the start of vacuum drying. Helium is used as the medium to remove water and subsequent vacuum drying occurs with a helium environment in the DSC cavity. The vacuum drying operation does not reduce the pressure sufficiently to reduce the thermal conductivity of the helium in the canister cavity.

The maximum temperatures including the maximum fuel cladding temperature are bounded by those calculated for transport operation if the DSC shell temperature under NCT is higher than the DSC shell temperature of 212°F maintained during vacuum drying. As shown in Table A.3–8 and Table A.3–9 for all DSCs in the MP197HB TC, all DSC shell minimum temperatures are higher than 212°F. Therefore, no additional thermal evaluation is needed. The presence of helium during blowdown and vacuum drying operations eliminates the thermal cycling of fuel cladding during helium backfilling of the DSCs subsequent to vacuum drying. Therefore, the thermal cycling limit of 65°C (117°F) for short term operations set by ISG-11 is satisfied for vacuum drying operation in the MP197HB package.

Per the operating procedures during vacuum drying, the cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg or less. The maximum fuel cladding temperature during a reflooding event is significantly less than the vacuum drying condition owing to the presence of water/steam in the canister cavity. Based on the above rationale, the maximum cladding temperature during an unloading operation is bounded by the maximum fuel cladding temperature for a vacuum drying operation.

Based on the described model and the thermal evaluation results, the staff finds the package thermal evaluation during vacuum drying operations acceptable.

3.6 Thermal Evaluation Under Hypothetical Accident Conditions

The finite element model of the MP197HB TC developed by the applicant to perform NCT is modified in this evaluation to determine the maximum component temperatures for HAC. For the transient runs considering HAC, the basket and hold-down ring are homogenized.

3.6.1 Initial Conditions

The initial temperatures for the MP197HB TC transient model before the fire accident are determined using the same boundary conditions for NCT (100°F ambient with insolation) described in Section A.3.3.1.1 of the application, except that the decay heat load is applied as a uniform heat generation rate over the homogenized basket for the transient runs.

3.6.2 Fire Test Conditions

Based on the requirements in 10 CFR 71.73, a fire temperature of 1475°F, fire emissivity of 0.9 and a period of 30 minutes are considered for the fire conditions. A bounding forced convection coefficient of 4.5 Btu/hr-ft^2-°F is considered during the burning period based on data from the report entitled "Thermal Measurements in a Series of Long Pool Fires," SANDIA Report, SAND 85-0196, TTC-0659, 1987. A surface emissivity of 0.8 is considered for the packaging surfaces exposed to fire based on 10 CFR 71.73.

3.6.3 Maximum Temperatures and Pressure

The maximum component temperatures for transient runs are listed in Table A.3–16 of the application. The seals are not explicitly considered in the models. The maximum seal temperatures are retrieved from the models by selecting the nodes at the locations of the corresponding seals.

The maximum HAC temperatures for all DSC contents are listed in Table A.3–18 of the application. The maximum TC and DSC component temperatures for HAC are summarized in Table A.3–19 of the application. This table shows that the maximum temperatures of the MP197HB TC components, calculated for HAC, are lower than the allowable limits. The maximum seal temperature for the fluorocarbon seals is 394°F at drain port for 32 kW heat load in the 69BTH DSC when the TC is equipped with external fins.

The short-term temperature limit for the fluorocarbon seals is 482°F as shown in the Parker O-ring Handbook 5700, Y2000 Edition, 1999. The maximum seal temperature for the metallic seals is 434°F at drain port for 32 kW heat load in 69BTH DSC when the TC is equipped with external fins. This temperature is below the long term limit of 644°F specified for continued seal function.

The maximum temperature of the gamma shield (lead) is 574°F, which is below the lead melting point of 621°F. The maximum fuel cladding temperature is between 668°F and 693°F for 69BTH DSC with 26 kW to 32 kW heat loads. For 37PTH DSC, the bounding maximum fuel cladding temperature is 671°F with a 22 kW heat load. These temperatures are below the limit of 1058°F (570°C).

The maximum internal pressures inside the 37PTH and 69BTH DSCs are summarized in Table A.3–22. The maximum internal pressures are 102.64 psig and 95.54 psig for the 37PTH DSC and 69BTH DSC under HAC, respectively. These maximum internal DSC pressures are below the design limit of 140 psig specified in Section A.3.1 of the application for HAC for the 69BTH and 37PTH DSCs.

3.6.4 Maximum Thermal Stresses

Thermal stresses for the MP197HB TC loaded with DSCs are discussed in Chapter A.2 of the application.

3.6.5 Accident Conditions for Fissile Material Packages for Air Transport

The package is not designed for air transport.

The staff reviewed the applicant's analysis of the package during HAC. Based on the information provided in the application regarding HAC analysis, the staff determines that the application is consistent with the guidance provided in Section 3.5.6 (Thermal Evaluation under Hypothetical Accident Conditions) of NUREG-1617. Therefore, the staff concludes that the HAC analysis is acceptable.

3.7 Thermal Evaluation for NCT and HAC for Altered Physical Configuration of Fuel Assemblies

THIS SECTION IS PROPRIETARY

3.8 Thermal Tests

A thermal test of the Model No. MP197HB fabricated package is performed as described in Section A.8.1.8 of the application. This thermal test if performed to measure the effective thermal conductivity of the TC in the radial direction over an approximately 10-ft exposed length within the neutron shield region. These measured thermal conductivities will be used as thermal input for the ANSYS model described in Section A.3.3.1.1 of the application for the NCT thermal analysis.

The temperature distribution computed with the measured conductivity of the packaging will be compared against the corresponding values in Tables A.3-8, and A.3-10 to demonstrate that the thermal performance of the fabricated TC is equal to or exceeds the predicted performance reported in the application.

3.9 Confirmatory Analyses

The staff reviewed the thermal models developed by the applicant to perform the thermal evaluation of the package. The staff checked the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were applied. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the analysis model. The material properties presented in the application were reviewed to verify that they were appropriately referenced and used.

3.10 Findings

The staff reviewed the package description and evaluation and found reasonable assurance that they satisfy the thermal requirements of 10 CFR Part 71. The staff reviewed the material properties and component specifications used in the thermal evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71. The staff reviewed the methods used in the thermal evaluation and found reasonable assurance that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.

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

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

The staff reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-time limits during HAC, consistent with the tests specified in 10 CFR Part 71.

4.0 CONTAINMENT REVIEW

The objective of the review is to verify that the package containment design is adequately described and evaluated under NCT and HAC, as required per 10 CFR Part 71.

4.1 Description of the Containment System

The Model No. NUHOMS®-MP197HB package containment system consists of the following components: (1) a cylindrical inner shell, (2) a bottom plate with a RAM access closure plate with seal and bolts, (3) a cask body flange, (4) a top lid with seal and bolts, (5) the vent and drain ports with closure bolts and seals, and (6) all containment welds. The containment vessel prevents leakage of radioactive material from the package cavity and maintains an inert atmosphere in the package cavity.

The containment system components and their material of construction are listed in Table 4-1 below:

Table 4-1 MP197HB Containment System Components

Part	MATERIAL
Inner Shell	SA-203, Grade E
RAM Closure Plate	SA-203, Grade E or SA-240, Type 304
RAM Closure Seal	Fluorocarbon or metallic
Cask Body Flange	SA-350-LF3
Bottom End Closure	SA-350-LF3
Top Lid	SA-350-LF3 or SA-203, Grade E
Top Closure O-Ring Seal	Fluorocarbon or metallic
Drain and Vent Port Bolts	Brass or A193 B8
Drain and Vent Port O-Ring Seal	Fluorocarbon or metallic

The containment penetrations include the vent and drain ports, the RAM access closure plate and the lid. Each penetration is designed to maintain a leakage rate of 1×10^{-7} ref-cm^3/sec or less, which is defined as "leaktight" per ANSI N14.5.

The containment seals are located at the lid, the RAM access closure plate, the vent plug, and the drain plug. The containment seals are either fluorocarbon elastomer or metallic O-rings. The fluorocarbon O-rings (VM835-75) are used in a temperature range from -40°F to 400°F, while the metallic seal for the RAM closure plate is a Helicoflex® aluminum jacketed seal designed to maintain a maximum helium leak rate of 1.0×10^{-9} ref-cm^3/sec with an operating temperature range from cryogenic to 644°F (340°C). The metallic seal for the drain plug is a Parker O-ring made of alloy X-750 with an operating temperature range from -40°F to 1100°F (593°C). The inner seal in all cases is the primary containment seal. The outer, secondary, seals facilitate leak testing of the inner containment seal of the RAM access closure plate and the lid.

All containment boundary welds are full penetration bevel or groove welds to ensure structural and sealing integrity. The containment boundary welds are fully examined by radiographic or ultrasonic methods and are tested by liquid penetrant examination.

4.2 Containment under NCT

4.2.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed, that the maximum pressure (MNOP) of the MP197HB package under NCT is 12.7 psig and would not exceed the pressure of 30.0 psig for which the package has been evaluated in Appendix A.2.13.1, Section A.2.13.1.4 of the application.

4.2.2 Containment Criteria

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

The results of the applicant's structural and thermal analyses show that the containment system retains the capability to maintain 1×10^{-7} ref-cm^3/sec or less under the conditions specified in 10 CFR 71.71. Therefore, the staff concludes that the loss or dispersal of radioactive material from the package will be less than 10^{-6} A_2 per hour under NCT, as required in 10 CFR 71.51(a)(1).

4.3 Containment under HAC

4.3.1 Pressurization of Containment Vessel

Section A.2.7.4 of the application noted that the maximum pressure of the MP197HB canister under HAC is 14.4 psig which is bounded by the internal pressure of 120 psig used in the HAC structural analysis.

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 HAC.

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 package inner shell will not buckle under HAC.

Results of the structural and thermal analyses in Chapters A.2 and A.3 of the application demonstrated 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 leaktight criteria of American National Standards Institute 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 Special Requirements

Solid plutonium is not an authorized content for the secondary containers containing dry irradiated and/or contaminated non-fuel bearing solid materials.

4.5 Verification of DSC Acceptance for Transport

4.5.1 Helium Diffusion Verification

The applicant performed a helium diffusion analysis to verify the existence of helium in the DSC after storage. The applicant assumed in the calculation that (a) the complete helium leakage is plausible, (b) the helium leaks through the through-cracks in the DSC shell, and (c) the minimum detectable crack by the NDE is 0.1 mm deep and 1.0 mm long for a DSC with typical nominal shell thickness of 0.5 inches. The analysis shows it would require 9,544 years for the DSC cavity to be completely empty of helium and would require 63 years for a crack of approximately four times of the minimum detectable diameter for complete helium diffusion.

The staff reviewed the Part 2 "Helium Diffusion Verification" of Enclosure 7 to TN Report E-34471, and performed the confirmatory analyses to validate the applicant's calculations. After comparing the applicant's results of Table 1 (Leakage Rate and Elapsed Time) and Table 3 (Sensitivity Study at Varying Temperatures) with the staff's confirmatory analytical results of Tables 2 and 4, the staff concludes that any DSC within 56 years of storage still contains helium.

Table 1. Applicant's Calculations of Leakage Rate and Elapsed Time for a Hypothetical Crack

D	a	T	P_u	P_d	P_a	F_c	F_m	L_u	t	
cm	cm	K	atm	atm	atm	cm^3/atm-s	cm^3/atm-s	cm^3/s	s	y
3.00E-03	1.27	450	1.27	1	1.14	6.025E-03	7.562E-04	1.648E-03	1.989E+09	63
2.00E-03	1.27	450	1.27	1	1.14	1.190E-03	2.241E-04	3.437E-04	9.537E+09	302
1.00E-03	1.27	450	1.27	1	1.14	7.439E-05	2.801E-05	2.488E-05	1.317E+11	4177
8.00E-04	1.27	450	1.27	1	1.14	3.047E-05	1.434E-05	1.089E-05	3.010E+11	9544

Table 2. Staff's Confirmatory Calculations of Leakage Rate and Elapsed Time for a Hypothetical Crack

D	a	T	Pu	Pd	Pa	Fc	Fm	Lu	time	time
cm	cm	K	atm	atm	atm	cm^3/atm-s	cm^3/atm-s	cm^3/s	(seconds)	(years)
3.00E-3	1.27	450	1.27	1	1.135	6.034E-3	7.570E-4	1.639E-3	2.000E9	63
2.00E-3	1.27	450	1.27	1	1.135	1.192E-3	2.243E-4	3.417E-4	9.592E9	304
1.00E-3	1.27	450	1.27	1	1.135	7.449E-5	2.804E-5	2.474E-5	1.325E11	4201
8.00E-4	1.27	450	1.27	1	1.135	3.051E-5	1.435E-5	1.083E-5	3.026E11	9597

Table 3. Applicant's Sensitivity Analyses for a 0.003 cm Diameter Crack at Varying Temperatures

T_{DSC}	D	a	T	µ	P_u	P_d	P_a	F_c	F_m	L_u	t	
°F	cm	cm	K	cP	atm	atm	atm	cm^3/atm-s	cm^3/atm-s	cm^3/s	s	y
200	3.00E-03	1.27	367	2.283E-02	1.27	1	1.14	6.957E-03	6.826E-04	1.857E-03	1.765E+09	56
300	3.00E-03	1.27	422	2.517E-02	1.27	1	1.14	6.309E-03	7.325E-04	1.711E-03	1.915E+09	61
400	3.00E-03	1.27	478	2.747E-02	1.27	1	1.14	5.782E-03	7.792E-04	1.594E-03	2.055E+09	65
500	3.00E-03	1.27	533	2.854E-02	1.27	1	1.14	5.564E-03	8.233E-04	1.552E-03	2.112E+09	67

Table 4. Staff's Confirmatory Sensitivity Analyses for a 0.003 cm Diameter Crack at Varying Temperatures

T	D	a	T	µ	Pu	Pd	Pa	Fc	Fm	Lu	Time	Time
F	cm	cm	K	cP	Atm	Atm	Atm	cm^3/atm-s	cm^3/atm-s	cm^3/s	(seconds)	(years)
200	3E-3	1.27	367	2.283E-2	1.27	1	1.135	6.956E-3	6.836E-4	1.843E-3	1.778E9	56
300	3E-3	1.27	422	2.517E-2	1.27	1	1.135	6.309E-3	7.330E-4	1.699E-3	1.929E9	61
400	3E-3	1.27	478	2.747E-2	1.27	1	1.135	5.781E-3	7.801E-4	1.583E-3	2.070E9	66
500	3E-3	1.27	533	2.854E-2	1.27	1	1.135	5.565E-3	8.238E-4	1.541E-3	2.127E9	67

The staff reviewed the Helium Diffusion Verification described in TN Report E-33299 (Enclosure 7 to TN E-34471) and verified that the helium will be present in the DSC after its extended storage time.

4.5.2 Verification of Moderator Exclusion

The applicant stated in Part 3 of TN Report E-33299 that, if the leakage test verifies that the leakage rate is less than $1x10^{-3}$ ref-cm^3/s, water will not leak into the DSC. The staff performed the confirmatory analysis and concludes that the helium leakage rate of $1.0x10^{-3}$ ref-cm^3/s can be used as the acceptance criterion for water leakage detection. Therefore, any DSC exhibiting a leakage rate less than $1.0x10^{-3}$ ref-cm^3/s through a pre-shipment leak testing will preclude water from entering the canister.

The staff confirms that both helium diffusion and moderator exclusion verifications provide support that the DSCs transported in the package will be able to maintain their containment safety function.

4.6 Evaluation Findings

The staff has reviewed the containment evaluation of the package and concludes that the Model No. MP197HB package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71 and meets the containment criteria of ANSI N14.5.

The staff has reviewed the evaluation of the containment system under NCT and concludes that the Model No. MP197HB package meets the containment requirements specified in 10 CFR 71.71, 71.43(f), and 71.51(a)(1) for NCT. The staff has reviewed the evaluation of the containment system under HAC and concludes that the Model No. MP197HB package meets the containment requirements specified in 10 CFR 71.73 and 71.51(a)(2) for HAC.

5.0 SHIELDING REVIEW

The purpose of this evaluation is to verify that the Model No. NUHOMS®-MP197HB package shielding design provides adequate protection against direct radiation from its contents and the package design meets the dose rate limits set forth in 10 CFR Part 71.47 and 71.51.

5.1 Shielding Design Description

The package is designed to transport spent PWR and BWR fuel with burnups greater than 45 GWd/MTU (high burnup) and irradiated and/or contaminated non-fuel bearing solid materials in radioactive waste container. The Model No. MP-197 low burnup fuel (≤ 45 GWd/MTU) package designs were approved in the previous certificate of compliance.

The Model No. MP197HB package consists of a transport overpack and a Dry Shielded Canister (DSC). The DSC is a cylindrical spent fuel basket with several configurations for loading of 24, 32, or 37 high burnup PWR fuel assemblies or 61 or 69 high burnup BWR fuel assemblies. The maximum allowable assembly average burnup is 62 GWd/MTU for both PWR and BWR fuels.

Control components (CC), which include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources are authorized contents. Nonfuel hardware that are inserted in the fuel assembly after the fuel assembly is discharged from the core, such as guide tube or instrument tube tie rods or anchors, guide tube inserts, are also considered as allowable control components. Details on these various fuel basket configurations and allowable contents are discussed in Chapter 1 of the application, along with all corresponding appendices.

The shielding design of the packaging includes two cylindrical stainless steel shells, a lead shell, and a borated VYAL-B resin layer at the radial direction and a bottom plate and a top lid at the axial directions. In the radial direction, the lead layer is encased in two stainless shells. Borated resin is encased in slender tubes that are inserted into the space between the outmost stainless steel shell and the outer shell that holds the lead layer. The total thickness of stainless steel shells is 4 inches. The lead layer and the resin layer are 3 and 6 inches respectively. The top lid and the bottom plate are 4.75 inches and 6.5 inches respectively. The impact limiters are made of balsa wood and redwood encased in a 0.25 inch thick stainless steel shell and are 26.5 inch long and 125.53 inch in diameter.

The inner cavity of the packaging is the same for all fuel basket designs. An aluminum sleeve is used to hold the baskets, in the center of the package, when those are smaller than the overpack cavity. The lead layer and stainless steel shells provide gamma shielding and the borated resin provides chief neutron shield at the radial direction. The top and bottom assemblies provide shielding at the top and the bottom ends of the package respectively. The impact limiters provide additional shielding mainly at the axial directions by the additional distance.

Solid aluminum transition rails serve to center the fuel compartment clusters inside the DSC. Borated aluminum-poison plates for criticality safety control are attached to the walls of the tubes by stainless steel covers. The wall and top and bottom plates of the DSC for the spent fuel also provide significant shielding in the radial and axial directions of the package.

The package is designed for exclusive use. The package is assumed to be as wide as the open vehicle. A personnel barrier is mounted to the transport frame of the vehicle to prevent unauthorized access to the package body. Normal conditions of transport dose rates are computed for exclusive-use transport in an open vehicle. The package is designed to be transported by private carriers and the operators are required to wear a dosimeter and are subject to the requirements of 10 CFR 20.1502 on occupational dose which satisfies the requirements of 10 CFR 71.47(b)(4).

The allowable contents for the various packages are as follows:

5.1.1 NUHOMS®-24PT4 DSC

The 24PT4 basket is designed to accommodate up to 24 intact or a combination of the intact and damaged PWR fuel assemblies. The 24PT4 can hold up to 12 damaged Westinghouse-CENP 16x16 (CE 16x16) fuel assemblies in specially designed Failed Fuel Cans (FFCs) with the balance being loaded with intact fuel. The specifications and design characteristics of intact and/or damaged CE 16x16 fuel assemblies acceptable for transport in the 24PT4 DSC are shown in Table A.1.4.1-1, Table A.1.4.1-2, and Table A.1.4.1-3 of the SAR. The fuel to be transported in the 24PT4 DSC is limited to a maximum initial enrichment of 4.85 wt.% U-235. The maximum allowable assembly average burnup is given as a function of initial fuel enrichment but does not exceed 62,000 MWd/MTU. The minimum cooling time is 15 years.

Reconstituted assemblies are authorized contents provided that these fuel assemblies contain no more than eight replacement stainless steel rods or replacement Zircaloy clad uranium rods in places of damaged fuel rods. These rods must displace an amount of water equal to or greater than that displaced by the original fuel rods in the active fuel region of the fuel assembly.

The 24PT4 fuel basket is divided into inner and peripheral zones, as shown in Figure A.1.4.1-5 of the SAR. Table A.1.4.1-5 of the SAR provides cooling time requirements for the fuel assemblies in the inner zone. Additional cooling time, Δt, in years is required for fuel in the peripheral zone as defined in Figure A.5-12 is determined by the following equation:

$$\Delta t = 26.19 \times \ln\left(\frac{m}{440}\right) + B \qquad (5\text{-}1)$$

Where m is the uranium loading of the fuel assembly to be shipped in kilogram and the value of parameter B is given in Table A.1.4.1-5a of the application. Figure A.5-12 of the application provides the loading pattern for the 24PT4 DSC.

5.1.2 NUHOMS®-24PTH DSC

The 24PTH baskets are designed to accommodate 24 intact or a combination of intact and up to 12 damaged fuel assemblies. The intact fuel assemblies may also contain control components. The NUHOMS®-24PTH configurations are designed to transport intact (including reconstituted) and/or damaged PWR fuel as specified in Table A.1.4.3-2 and Table A.1.4.3-4 of the SAR. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235. The maximum allowable assembly average burnup is 62 GWd/MTU and the minimum cooling time requirements are given in Table A.1.4.3-2 of the SAR. Table A.1.4.3-5 of the SAR provides the PWR Fuel Qualification Table for NUHOMS®-24PTH DSC. Additional cooling time is required for fuel loaded at the peripheral locations and with burnup greater than 51 GWd/MTU. The required additional cooling time is determined by Equation 5-1. The value of parameter B in equation 5-1 for this basket configuration is given in Table A.1.4.3-5a of the application.

Intact fuel assemblies transported in the 24PTH DSC may include reconstituted assemblies containing replacement rods up to 10 stainless steel rods per assembly. There is no restriction on the number of lower enrichment UO_2 replacement rods per assembly. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. A 24PTH DSC containing less than 24 fuel assemblies may contain either empty slots or dummy fuel assemblies in the empty slots. This is not a concern for shielding because this configuration contains fewer radiation sources than does the irradiated fuel and is therefore bounded by their shielding analyses. Figure A.5-12 provides the loading pattern for 24PTH DSC.

5.1.3 NUHOMS®-32PTH DSC

The 32PTH baskets are designed to accommodate 32 intact or up to 16 damaged with the remainder for intact PWR fuel assemblies with or without control components. There are two designs for the NUHOMS®-32PTH DSC, namely 32PTH and 32PTH1 or 32PTH Type 1. The difference between the 32PTH and 32PTH Type 1 design is the length of the basket to accommodate PWR fuel assemblies with different lengths. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235. The maximum allowable assembly average burnup is 62 GWd/MTU and the minimum cooling time requirements are given in Table A.1.4.4-5 of the application. For fuel in the peripheral zone of the fuel basket, certain burnup and enrichment combinations require additional cooling time determined by Equation 5-1 with B values determined based on Table A.1.4.4-5a of the application.

5.1.4 NUHOMS®-37PTH DSC

The 37PTH baskets are designed to hold 37 all intact, or up to 4 damaged plus the remainder intact, PWR fuel assemblies with or without control components. There are two designs for the NUHOMS®-32PTH DSC, namely 37PTH-S and 37PTH-M for short and medium canisters respectively to accommodate PWR fuel assemblies with different lengths. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time requirements are given in Table A.1.4.6-2 of the SAR. Spacers, instrument tube tie rods, and anchors that were used to facilitate handle of fuel assemblies may also be loaded as part of the fuel assemblies.

Reconstituted assemblies may contain up to 10 replacement irradiated stainless steel rods or stainless steel clad rods per assembly. For reconstituted fuel assemblies with replacement rod made of lower enrichment UO_2 rods, Zircaloy (including other Zirconium based alloy) rods, Zr pellets, or unirradiated stainless steel rods, there is no limit on the allowable number of replacement rods per fuel assembly. For fuel assemblies reconstituted with stainless steel replacement rods, the maximum exposure of the stainless steel rods shall not exceed two-thirds of the burnup of the rest of fuel rods that were originally loaded in the assembly. For reconstituted fuel assemblies using UO_2 fuel rods as replacement rods, the enrichment of the replacement rods shall not exceed the initial enrichment of the fuel assembly as manufactured. The replacement rods with lower enrichment UO_2 are assumed to have the same irradiation history as the entire fuel assembly for radiation source calculation purpose. A 37PTH DSC may also contain fewer than 37 fuel assemblies and these cells may be replaced with dummy fuel assemblies. The fully loaded basket bounds the ones with fewer fuel assemblies in terms of meeting shielding requirements.

The fuel basket is divided into inner and peripheral zones, as shown in Figure A.1.4.6-2 of the application. For fuel in the peripheral zone, certain burnup and enrichment combinations require additional cooling time determined by using Equation 5-1 with the B values from Table A.1.4.6-5a of the application.

Some of the PWR fuel DSC designs use zoned loading of fuel with various burnup and cooling time (decay heat). The loading pattern presented in Figure A.1.4.4-1 of the SAR is a specific example and needs special attention from the users and should be noted in Chapter 7. These loading patterns could create a potential misload. To ensure correct loading, users follow the allowable loading patterns presented in Chapters 1 and 7 of the application, as well as those referenced in the CoC.

The NUHOMS DSC designs include several complex loading patterns. While the 32PTH has two zones for the center four fuel assembly cells, the 32PTH1 allows damaged fuel only in the center 16 locations with three zones.

A 24PTH DSC containing less than 24 fuel assemblies may contain either empty slots or dummy fuel assemblies in the empty slots. The 32PT DSC is not authorized to ship fuel with burnup greater than 45 GWd/MTU. The users must follow exactly the loading tables when loading the DSCs.

5.1.5 NUHOMS®-61BTH DSC

The 61BTH baskets are designed to accommodate 61 intact, or up to 16 damaged with up to four (4) Failed Fuel Cans (FFCs) loaded with damaged fuel with the remainder intact BWR fuel assemblies with or without fuel channels. The maximum burnup of the allowable BWR fuel assemblies for the 61BTH cask is 62 GWd/MTU. The minimum required cooling times are given in Tables A.1.4.8-6 and A.1.4.8-7 of the SAR.

The 61BTH DSC system includes three design configurations, 61BTH Type 1, 61BTH Type 2, and 61BTHF. These different types have same capacity but with three different canister lengths to accommodate fuel assemblies with different lengths. Table A.1.4.8-1 of the application provides the overall lengths and outer diameters for each 61BTH DSC configuration. Details of the 61BTH fuel baskets are shown in the drawings in Section A.1.4.10.9 of Appendix A.1.4.10 of the application.

The 61BTH DSC allows for reconstituted fuel assemblies that contain up to four replacement irradiated stainless steel rods per assembly or 61 non-zircaloy clad fuel with lower enrichment UO_2. The stainless steel rods must have less than two-thirds exposure as the remaining fuel rods in the same assembly. The reconstituted UO_2 rods must not exceed the burnup as those rods were originally loaded in fuel assembly.

The 61BTH DSC uses zoned loading. For the fuel assemblies in the peripheral zones, additional cooling time is required. The additional cooling time required is determined by Equation 5-1 with the B values for the equation as given in Table A.1.4.8-7a of the SAR. However, if the result of additional cooling time, determined by Equation 5.31 is less than 0, the additional cooling time should be zero.

5.1.6 NUHOMS®-69BTH DSC

The 69BTH basket is designed to accommodate 69 intact, or up to 24 damaged plus 45 intact BWR fuel assemblies with or without fuel channels. The maximum burnup of the allowable BWR fuel assemblies for the 69BTH cask is 62 GWd/MTU. The details of the 69BTH fuel basket are shown in the drawings in Section A.1.4.10.10 of Appendix A.1.4.10 of the SAR.

The characteristics of the allowable spent fuel contents of the 69BTH fuel basket are given in Table A.1.4.9-1 of the SAR. The minimal cooling time for each fuel is given in Tables A.1.4.9-54 and A.1.4.9-5 of the SAR. Because the 69BTH fuel basket uses zoned loading, additional cooling time may be necessary for some of the fuel assemblies to be loaded in the peripheral fuel cell locations. The additional cooling time required is determined by Equation 5.31 with the B values for the equation as given in Table A.1.4.9-5a of the SAR. However, if the result of additional cooling time, determined using Equation 5-1, is less than 0, the additional cooling time should be zero.

The 69BTH basket has a variety of complicated loading patterns. One loading pattern has a zone that has one location with the decay heat limit of 0.1 kW/FA (Figure A.1.4.9-2). Because of the complicated loading pattern design for the 69BTH basket, there is a greater potential for misloading the package. The users of the packaging must pay close attention to this issue. Table A.1.4.7-3 of the SAR provides specific requirements for various damaged and intact fuel loading configurations. The shippers shall follow closely the loading table for each configuration to avoid misloads.

5.1.7 Radioactive Waste Canister

The package uses Radioactive Waste Container (RWC) to transport dry irradiated and/or contaminated non-fuel-bearing solid materials. Each RWC system includes an outer cylindrical shell assembly. The RWC system consists of two design configurations, a Welded Top Shield Plug Design (RWC-W) and a Bolted Top Shield Plug Design (RWC-B). The drawings contained in Section A.1.4.10.11 of Appendix A.1.4.10 of the SAR provide detailed structural layout and the dimensions of the RWCs. The payload in the RWC is dry and in an air or inert gas environment. The total activity of the RWC payload is limited to 90,000 Ci of cobalt-60. Typical composition of the payload may consist of any individual or a combination of the following irradiated non-fuel hardware:

1. BWR Control Rod Blades
2. BWR Local Power Range Monitors (LPRMs)
3. BWR Fuel Channels
4. BWR Poison Curtains
5. PWR Burnable Poison Rod Assemblies (BPRAs)
6. PWR and BWR Reactor Vessel and Internals

Licensing drawings NUHRWC-71-1001 Rev 1, NUHRWC-71-1002 Rev 1, and Figure A.1.1 in the SAR provides a general sketch of the MP197HB packaging system. Engineering drawings MP197HB-71-1001 to 1014 provide details of the geometric dimensions and the structural materials of the packaging system. NUH24PT4-71-1001 to NUH24PT4-71-1003, NUH24PTH4-71-1001 to NUH24PTH4-71-1009, NUH61BT-71-1000 to NUH61BT-71-1002, NUH61BTH-71-1100, NUH61BTH-71-1102 to NUH61BTH-71-1106, NUH69BTH-71-1000, NUH69BTH-71-1002, NUH69BTH-71-1002, NUH69BTH-71-1011, and NUH69BTH-71-1012 provide details of fuel basket structures that are important to shielding for the six different DSC designs. NUHRWC-71-1003 Rev 0 provides detailed information on the RWC design and geometric dimensions.

5.2 Radiation Source Specification

The design basis fuel contents are the B&W 15x15 Mark B10 for PWR fuels and the GE-2 7x7 Type G2A for BWR fuels since these two fuel assemblies contain the largest amount of heavy metal load in their respective fuel types among the authorized fuel contents. The shielding performance of the package is analyzed based on a bounding source (both gamma and neutron).

The B&W 15x15 Mark B10 and the GE-2 7x7 are evaluated as the design basis (DB) PWR and BWR fuel assembly (FA) respectively in the shielding evaluation of the MP197HB transportation package based on the maximum heavy metal weight for their type with respect to the same burnup and cooling time. The final qualification for fuel loading is determined by the required cooling for the combination of burnup and initial enrichment of a fuel assembly. The GE 7x7 fuel assembly bounds all allowable fuel assembly designs as it contains the highest amount of steel, and Inconel among all other BWR fuel designs qualified for contents. The applicant further demonstrated that the 69BTH package containing sixty nine (69) GE-2 7x7 BWR fuel assemblies contains the highest amount of fuel, steel, and Inconel and therefore bounds all other PWR and BWR loading configurations of the MP-197HB packages in terms of shielding analyses.

The applicant calculated the radiation source terms of the spent fuel contents using the TRITON module of the SCALE computer code package. The 44GROUPNDF5 cross section library is used in the TRITON/T-DEPL depletion calculations. To address the concern if the SCALE package used for the source term calculations is appropriate for high burnup fuel, the applicant performed benchmark analyses for the TRITON module of the SCALE code for source term calculations using publicly available data as provided in, NUREG/CR-6968, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors," Oak Ridge National Laboratory, February, 2010, for fuel samples obtained from the Calvert Cliffs, TMI, and Takahama reactors and NUREG/CR-7012, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," Oak Ridge National Laboratory, January, 2011, for high burnup fuel obtained from the Vandellos reactor, and NUREG/CR-7013, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor," January 2011, for fuel samples obtained from Gosgen and GKN II reactors. The burnup of the high burnup samples ranges from 46 to 78.3 GWd/MTU. Although these samples do not have a full coverage of the actinides and fission products that are important to shielding, the staff considers these samples sufficient for the purposes of benchmarking source term calculations because these samples do contain the actinides and fission products that are the major contributors of gamma and neutron sources.

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5.2.1 Gamma Source

The gamma source terms comprise of three sources: spent fuel region, activated fuel structural materials and inserts, and the source term from (n, γ) reactions. The primary source of activity in the non-fuel regions of a fuel assembly comes from the activation of ^{59}Co to ^{60}Co because of the impurities in the steel structural material above and below the fuel. The activity of the ^{60}Co is calculated using ORIGEN-S with the assumption that the in-core fuel region flux was at full power. The calculated activation ^{60}Co source was modified using the scaling factors listed in Chapter A.5, Section A.5.2 of the application account for the fact that the end fittings and the plenum section of a fuel assembly were exposed to only a fraction the flux of that the assembly was exposed to.

The masses of the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.15, respectively in the fuel assembly model to account for the fact that the neutron flux that activates the metal is much lower and softer than that in the fuel region. This is consistent with the recommendation of ORNL/TM-11018, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," Oak Ridge National Laboratory, December 1989, and PNL-6906, "Spent Fuel Assembly Hardware Characterization and 10 CFR Part 61 Classification for Waste Disposal," PNL-6906, Volume 1, Pacific Northwest Laboratory, Richland, WA, June 1989. Table A.5-18 of the SAR provides the bounding gamma source for all authorized CCs.

The second source of photons comes from (n, gamma) reactions in the basket and package materials. The (n, gamma) photons are properly accounted for in the coupled neutron-photon MCNP5 calculations by using the PHYS:n, p option and tallies for both neutron and photons in the neutron particle transport simulation.

The gamma radiation spectrum is presented in an 18 energy group structure consistent with the SCALE 27 n-18 g energy group cross section library. The lower bound energy range in this library is 0.05 MeV. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

The staff performed a confirmatory calculation for the gamma source terms of the design basis fuel assembly and finds the applicant's source terms acceptable and conservative.

5.2.2 Neutron Source

The neutron sources are comprised of primarily the neutron radiation from spent fuel (both α-n reactions and spontaneous fission) with neutrons produced by sub-critical multiplication in the fuels.

The applicant calculated the neutron sources of the design basis fuel, B&W 15x15 Mark B10 for the 37PHT and the 69BHT for the 37 PWR assembly package and the 69 BWR assembly package respectively. The applicant calculated the average number of neutrons produced by the spent fuel and obtained the neutron source as a function of fuel assembly burnup.

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5.3 Shielding Model Specification

The applicant performed the shielding analysis of the MP197HB package using the MCNP5 computer code and the continuous energy ENDF/B-VI cross section libraries. For NCT, the MP197HB package model includes the neutron shield and impact limiters. For HAC, the neutron shielding material as well as the trunnion plugs at the package side was assumed lost. The shielding model further assumes a 0.375 gap in at the end of the gamma shield or 0.1 inches of gap along the axial length of the lead shell to simulate the lead slump for vertical drop accident and horizontal drop accidents respectively. The impact limiters were assumed lost.

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Using the "response functions method," the applicant developed fuel qualification tables. The fuel qualification Tables A.1.4.1-5, A.1.4.7-4, A.1.4.8-6, A.1.4.8-6, and A.1.4.9-4 in the SAR provide the minimal cooling time required for each burnup and enrichment combination. It is important to note that these fuel qualification tables are based on shielding analyses only. Fuel meets the requirements specified in these tables may not guarantee them to satisfy requirements imposed by thermal and criticality safety requirements. The final fuel qualification must meet simultaneously all of the requirements of shielding, thermal and criticality safety.

The applicant calculated the dose rates for the package under hypothetical accident conditions. The applicant assumed that all neutron shield and the impact limiters are lost. Under HAC, the maximum dose rate at 1 meter from the surface of the package is 868.37 mrem/hr and 866.54 mrem/hr for the 37 PWR and 69 BWR fuel assembly package respectively, thus meeting the requirements of 1,000 mrem/hr.

The material compositions used in the shielding models are typical material compositions and densities for stainless steel, lead, aluminum, boron, and UO_2. The neutron shield material VYAL-B is a non-standard material and the SAR provides a detailed description of the material composition in Table A.5-13.

5.4 Fuel reconfiguration consequence analyses

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5.5 Shielding Evaluation

The applicant used the TRITON model of SCALE 6.0 to determine the gamma and neutron sources of the high burnup fuel. The TRITON code is a two dimensional fuel assembly lattice analysis code, and the applicant performed benchmark analyses of the codes for source term calculations to determine bias and uncertainties associated with the computed isotopic composition of the high burnup fuel using TRITON. The calculated source terms are then adjusted to include the uncertainty potentially being introduced by the code.

The applicant uses the MCNP5 code [1], version 1.4 for the shielding analyses. The MP197HB package is modeled with full three-dimensional details, the active fuel region is modeled as a homogenized mass and the upper and bottom end fitting regions of the design assembly are modeled as mixture of steel and void. The packaging body, neutron shield, radial steel ribs, and impact limiters are modeled according to the engineering design drawings.

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5.6 Evaluation Findings and Conclusion

The staff reviewed the description of the package design features related to shielding and the source terms for the design basis fuel and finds them acceptable. Based on its review, the staff determined that the methods used are consistent with accepted industry practices and standards.

The staff reviewed the maximum dose rates for NCT and HAC and determined that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51 for an exclusive use package.

The dose rate requirement prescribed in 10 CFR 71.47(b)(4) does not apply because the package must be transported by private carriers and the operators, required to wear a dosimeter, are subject to the requirements of 10 CFR 20.1502 per 10 CFR 71.47((b)(3).

Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance, and that the package design and contents satisfy the shielding and dose rate limits in 10 CFR Part 71 with the following conditions placed in the CoC fuel qualification tables:

1. The maximum length of the natural or low enrichment uranium blankets shall not exceed 5% of the assembly length.
2. The maximum average burnup is 62 GWd/MTU and 70 GWD/MTU for authorized PWR and BWR fuel types respectively.
3. The maximum quantity of non-fuel bearing radioactive material loaded into a package shall not exceed 90,000 Ci of ^{60}Co.
4. The users of this packaging system shall determine the condition of the high burnup fuel in the package prior to shipment following the instructions provided in the Operating Procedures which are referenced by the certificate of compliance of this package.
5. The shipper shall terminate further operations, should fuel damages be detected under NCT.
6. The package must be transported by private carriers and the operators, required to wear a dosimeter, are subject to the requirements of 10 CFR 20.1502 on occupational dose.

References:

1. X-5 Monte Carlo Team 2003, *MCNP—A General Monte Carlo N-Particle Transport Code, Version 5*, LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, N.M.
2. M.C. Billone, T.A. Burtseva, and R.E. Einziger "Ductile to Brittle Transition Temperature for High-Burnup Cladding Alloys Exposed to Simulated Drying Storage Conditions", Journal of Nuclear Materials, Available online October 22, 2012.

6.0 CRITICALITY REVIEW

This section presents the findings of the criticality safety review for the Model No. NUHOMS®-MP197HB package. This amendment includes a criticality analysis that credits reduced reactivity due to fuel burnup for some contents. Some contents, including those evaluated using burnup credit, also include high burnup fuel.

The staff evaluated the package for its ability to meet the fissile material requirements of 10 CFR Part 71, including the general requirements for fissile material packages in 10 CFR 71.55, and the standards for arrays of fissile material packages in 10 CFR 71.59. The staff reviewed the criticality safety analysis of the package presented in the application, and also performed independent calculations to confirm the applicant's results.

The staff's review considered the criticality safety requirements of the radioactive material transportation regulations in 10 CFR Part 71, as well as the review guidance presented in NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*.[1] Additionally, for contents evaluated using burnup credit, the staff's review considered the recommendations presented in Interim Staff Guidance 8, Revision 3, *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks* (ISG-8, Rev. 3).[2]

6.1 Description of the Criticality Design

6.1.1 Packaging Design Features

The packaging design consists of a cylindrical, steel shell containment system, with a flat bottom and bolted closure lid at the top, and with nine different available transportable canisters with internal basket structures for maintaining the position of the spent fuel contents. Each canister type consists of a stainless steel cylinder with a double seal-welded closure, and with similar basket structural materials and geometry, as described in the application.

Criticality safety is maintained by the fixed geometry of the basket, as well as by the borated aluminum, aluminum/B_4C metal matrix composite, or Boral® neutron poison plates present between each spent fuel assembly location.

For high burnup fuel, where reduced cladding properties may result in fuel reconfiguration under HAC conditions structural loads, criticality safety also depends on moderator exclusion, as described in Interim Staff Guidance 19, *Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)* (ISG-19).[3]

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6.1.2 Codes and Standards

The applicable regulations considered in the review of the criticality safety portion of this application include the fissile material requirements in 10 CFR Part 71, specifically the general requirements for fissile material packages in 10 CFR 71.55, and the standards for arrays of fissile material packages in 10 CFR 71.59. The staff also used the review guidance contained in NUREG-1617, as well as in ISG-8 and ISG-19.

6.1.3 Summary Table of Criticality Evaluations

The applicant provided a summary of maximum calculated k_{eff}s in Section A.6.1.3 of the application, which is summarized in the following table for all DSCs. All results include the calculated k_{eff} plus two times the Monte Carlo uncertainty.

Results for DSCs evaluated with burnup credit (24PTH, 32PT, 32PTH, 32PTH1, and 37PTH) also include the bias due to actinide depletion from the isotopic depletion code (0.0175), and a bias due to the horizontal burnup gradient (0.0050). The Upper Safety Limit (USL) includes the criticality code bias and bias uncertainty.

DSC	**K_{eff}**	**USL**
24PT4	0.9393	0.9411
32PT	0.9375	0.9412
24PTH	0.9375	0.9412
32PTH	0.9375	0.9412
32PTH1	0.9375	0.9412
37PTH	0.9375	0.9412
61BT	0.9364	0.9414
61BTH	0.9400	0.9415
69BTH	0.9406	0.9415

Note that the 32PTH, 32 PTH1, and 32PTH Type 1 DSCs are identical from a criticality safety standpoint, and are evaluated together in the application. The k_{eff} value reported for the 32PTH DSC applies to both the 32PTH, 32 PTH1, and 32PTH Type 1 DSCs.

6.1.4 Criticality Safety Index

The applicant demonstrated that infinite arrays of packages are adequately subcritical under NCT and HAC. Therefore, the criticality safety index (CSI), determined in accordance with 10 CFR 71.59(b), is 0.0.

6.2 Spent Nuclear Fuel Contents

The package is designed to transport a maximum of 69 boiling water reactor (BWR) and 37 PWR fuel assemblies in DSCs. Specific DSC contents are summarized below:

- NUHOMS®-24PT4 – designed to transport up to 24 intact or reconstituted Combustion Engineering 16x16 PWR spent fuel assemblies, or up to 12 damaged assemblies in failed fuel cans, with the remaining fuel intact. Intact and damaged fuel assembly specifications are contained in Tables 1.4.1-1 and 1.4.1-2, respectively. Fuel assembly design characteristics important to criticality safety are given in Table 1.4.1-3. Fuel assembly maximum initial enrichment and neutron poison requirements are given in Table 1.4.1-4.

- NUHOMS®-32PT – designed to transport up to 32 intact or reconstituted PWR spent fuel assemblies, with or without control components. Fuel assembly specifications are given in Table 1.4.2-2, and individual fuel assembly type characteristics important for criticality safety are given in Table 1.4.2-3. Poison rod assembly requirements for loading are given in Table 1.4.2-5, with poison rod assembly characteristics as given in Figure 1.4.2-1. Initial enrichment, burnup, and cooling time requirements determined in the burnup credit analysis are given in the loading curve in Table 1.4.2-7.

- NUHOMS®-24PTH – designed to transport up to 24 intact or reconstituted PWR spent fuel assemblies, with or without control components, or up to 12 damaged fuel assemblies or 8 failed fuel assemblies in failed fuel cans, with the remaining fuel intact. Fuel assembly specifications are given in Table 1.4.3-2, and individual fuel assembly type characteristics important for criticality safety are given in Table 1.4.3-4. Initial enrichment, burnup, cooling time, and neutron absorber type requirements determined in the burnup credit analysis are given in the loading curves for intact and damaged fuel configurations in Tables 1.4.3-8 and 1.4.3-9, respectively.

- NUHOMS®-32PTH and 32PTH Type 1 – designed to transport up to 32 intact or reconstituted PWR spent fuel assemblies, with or without control components, or up to 16 damaged fuel assemblies with the remaining fuel intact. Fuel assembly specifications are given in Table 1.4.4-2, and individual fuel assembly type characteristics important for criticality safety are given in Table 1.4.4-4. Initial enrichment, burnup, cooling time, and neutron absorber type requirements determined in the burnup credit analysis are given in the loading curves for intact and damaged fuel configurations in Tables 1.4.4-7 and 1.4.4-8, respectively.

- NUHOMS®-32PTH1 – similar to the 32PTH and 32PTH Type 1 canisters, and designed to transport up to 32 intact or reconstituted PWR spent fuel assemblies, with or without control components, or up to 16 damaged fuel assemblies with the remaining fuel intact. Fuel assembly specifications are given in Table 1.4.5-2, and individual fuel assembly type characteristics important for criticality safety are given in Table 1.4.5-4. Initial enrichment, burnup, cooling time, and neutron absorber type requirements determined in the burnup credit analysis are given in the loading curves for intact and damaged fuel configurations in Tables 1.4.5-8 and 1.4.5-9, respectively.

- NUHOMS®-37PTH – designed to transport up to 37 intact or reconstituted PWR spent fuel assemblies, with or without control components, or up to 4 damaged fuel assemblies with the remaining fuel intact. Fuel assembly specifications are given in Table 1.4.6-2, and individual fuel assembly type characteristics important for criticality safety are given in Table 1.4.6-4. Initial enrichment, burnup, and cooling time requirements determined in the burnup credit analysis are given in the loading curve for intact and damaged fuel configurations in Table 1.4.6-6.

- NUHOMS®-61BTH – similar to the previously approved 61BT canister, and designed to transport up to 61 intact or reconstituted BWR spent fuel assemblies, or up to 16 damaged fuel assemblies, or up to 4 failed fuel assemblies in failed fuel cans, with the remaining fuel intact. Intact and damaged fuel assembly specifications are contained in Table 1.4.8-2. Fuel assembly design characteristics important to criticality safety are given in Table 1.4.8-3. Fuel assembly maximum initial enrichment and neutron poison requirements for intact and damaged or failed fuel are given in Tables 1.4.8-4 and 1.4.8-5, respectively.

- NUHOMS®-69BTH – designed to transport up to 69 intact or reconstituted BWR spent fuel assemblies, or up to 24 damaged fuel assemblies with the remaining fuel intact. Intact and damaged fuel assembly specifications are contained in Table 1.4.9-1. Fuel assembly design characteristics important to criticality safety are given in Table 1.4.8-2. Fuel assembly maximum initial enrichment and neutron poison requirements for intact and damaged fuel are given in Table 1.4.9-3.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

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6.3.2 Material Properties

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6.3.3 Computer Codes and Cross Section Libraries

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6.3.4 Demonstration of Maximum Reactivity

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6.4 Single Package Evaluation

6.4.1 Configuration

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6.4.2 Results

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6.5 Evaluation of Package Arrays

6.5.1 Configuration

The applicant reflected the most reactive single package models in the criticality model on all sides to produce infinite arrays of packages, conservatively ignoring the impact limiter, neutron shield, and neutron shield stainless steel skin. The applicant varied the water density between packages to find the most reactive condition.

6.5.2 Results

Infinite array k_{eff}s are not significantly different than those for the water-reflected single package models. This indicates that the package is neutronically isolated, as expected due to the large amount of structural and shielding material between fissile material in adjacent packages. Infinite array k_{eff} results for each of the DSCs in the NUHOMS®-MP197HB package are contained in the application's Appendices A.6.5.1 through A.6.5.7. Since an infinite array of packages is demonstrated to be subcritical under both NCT and HAC, the resulting CSI is 0.0.

6.6 Fresh Fuel Benchmark Evaluations

The applicant performed a fresh fuel benchmarking analysis of the CSAS25 module of SCALE 4.4, with KENO V.a and the 44-group ENDF/B-V cross section library, used in the criticality analysis for the package with the 61BT, 61BTH, 69BTH, and 24PT4 DSCs. The details of this analysis are provided in Section A.6.5.2.5 of the application.

6.6.1 Experiments and Applicability

The applicant performed a fresh fuel benchmarking analysis using 125 fresh uranium oxide experiments, chosen to have characteristics similar to the TN NUHOMS® MP-197 package. The experiment parameters evaluated for range of applicability and trends included:

- ^{235}U enrichment,
- Fuel rod pitch,
- Water/fuel volume ratio,
- Assembly separation, and
- Average energy group causing fission (AEG).

The applicant evaluated each critical experiment using the same code, cross section library, computer platform, and modeling techniques as was used for the criticality evaluation.

6.6.2 Bias Determination

The applicant determined the USL for the package using USL Method 1 in accordance with NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water- Reactor fuel in Transportation and Storage Packages*.[4]

The applicant determined a separate USL for each of the four DSCs evaluated using the fresh fuel assumption, summarized in Appendices A.6.5.1, A.6.5.2, and A.6.5.3 for the 61BT and 61BTH, 69BTH, and 24PT4 DSCs, respectively. The trending parameter values from the limiting criticality analysis were compared to the USL equations for those parameters to determine a set of USLs based on each parameter. The most limiting (i.e., lowest) USL from this set was chosen as the USL for that particular DSC. The resulting USLs, and the trending parameter that they are based on, are summarized in the following table:

DSC	Trending Parameter	Bounding USL
24PT4	Fuel rod pitch	0.9411
61BT	Water/fuel ratio	0.9414
61BTH	Assembly separation	0.9415
69BTH	Assembly separation	0.9415

The staff reviewed the fresh fuel benchmarking analysis performed by the applicant, and determined that the USL was determined in accordance with relevant NRC guidance. The critical experiments chosen are appropriate for the system being evaluated, and the resulting USLs are bounding and acceptable.

6.7 Burnup Credit

6.7.1 Limits for the Licensing Basis

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6.7.2 Licensing-Basis Model Assumptions

6.7.2.1 Isotopic Depletion

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6.7.2.2 Criticality

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6.7.3 Code Validation – Isotopic Depletion

The applicant performed depletion calculations with SAS2H using the 44-group ENDF/B-V cross section library. The NITAWL-III and BONAMI modules were used by the applicant to determine the resonance self-shielding corrections based on the data available in the cross-section libraries. The applicant performed a validation based on a comparison of SAS2H predictions to publicly available isotopic assay data described in Section A.6.5.8.2 of the SAR.

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6.7.4 Code Validation – K_{eff} Determination

The applicant performed a burnup credit criticality benchmarking analysis of the CSAS25 module of SCALE 5, with KENO V.a and the 44-group ENDF/B-V cross section library, used in the criticality analysis for the package with the 32PT, 32PTH, 32PTH1, 37PTH, and 24PTH DSCs. The details of this analysis are provided in Sections A.6.5.9 and A.6.5.10 of the SAR.

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Additionally, a similar benchmarking evaluation was previously approved by staff for the TN-40 spent fuel transportation package.

6.7.5 Loading Curve and Burnup Verification

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6.7.5.1 Misload Analyses

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6.7.5.2 Administrative Procedures

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These additional procedures are comparable to those recommended in ISG-8, Rev. 3, and are acceptable for reducing the likelihood and severity of misload events.

6.8 References

1. U.S. Nuclear Regulatory Commission, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG-1617, March, 2000.
2. U.S. Nuclear Regulatory Commission, *Division of Spent Fuel Storage and Transportation Interim Staff Guidance – 8, Revision 3 – Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks*, U.S. NRC, September, 2012.
3. U.S. Nuclear Regulatory Commission, *Division of Spent Fuel Storage and Transportation Interim Staff Guidance – 19 – Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)*, U.S. NRC, May, 2003.
4. U.S. Nuclear Regulatory Commission, *Criticality Benchmark Guide for Light-Water- Reactor Fuel in Transportation and Storage Packages*, NUREG/CR-6361, ORNL/TM-13211, March 1997.
5. U.S. Nuclear Regulatory Commission, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, NUREG/CR-6801, ORNL/TM-2001/273, March, 2003.

6.9 Findings

The applicant has demonstrated that the package, when loaded with fuel assemblies meeting the characteristics of the contents described in Appendices A.1.4.1 through A.1.4.9 of the application, will be adequately subcritical under all conditions.

Therefore, the applicant has shown and the staff agrees that the TN NUHOMS® MP-197HB package meets the fissile material requirements of 10 CFR 71.55 for single packages, and 10 CFR 71.59 for arrays of packages, with a CSI of 0.0.

7.0 PACKAGE OPERATIONS

Chapter 7.0 of the application provides a summary description of package operations, including package loading and unloading operations, to ensure that the package is operated in a safe and reliable manner under NCT and HAC. The preparation of an empty package for shipment is also described.

7.1 Package Loading

The use of the packaging to transport fuel, dry irradiated and/or contaminated nonfuel bearing solid materials offsite involves (1) the preparation of the package for use, (2) verification that the fuel assemblies or waste to be loaded meet the criteria set forth in the appendices of the application for each type of fuel or waste, and (3) the installation of a DSC or RWC into the package.

In preparing the package for use, the user shall verify that the O-ring seals (metallic or elastomer) used at the penetrations have been discarded after each use. For the inner top cover plate welding operations for the wet loading of 24PT4, 32PT, 32PTH, 32PTH1, 37PTH, 61BT, 61BTH, and 69BTH DSCs, the user needs to verify that the measured hydrogen concentration does not exceed a limit of 2.4%. If the limit is exceeded, the user needs to stop all welding operations and purge the DSC cavity with 2-3 psig helium to reduce the hydrogen concentration safely below the 2.4% limit.

The user needs also to verify that the package has been fitted with an internal aluminum sleeve if it transports any of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or an RWC. The intact, damaged and failed fuel assemblies to be transported in a specific DSC model must be evaluated (by plant records or other means) to verify that they meet the criteria of the applicable fuel specification as listed in Table A.7-2 of the application.

For the transportation of fuel within the NUHOMS®-32PT, 24PTH, 32PTH, 32PTH1, or 37PTH DSCs, which use burnup credit, additional administrative controls to prevent misloading are also implemented. Regarding content's loading, the user needs to determine the maximum decay heat to limit the hydrogen generation and verify that the contents do not exceed this decay heat for contents which is loaded wet or contains the water. The user is also required to ensure the contents, the secondary container, and the packaging are chemically compatible and will not react to produce the flammable gases. The lid O-ring (elastomer) of the lid shall be discarded after each use of cask wet loading. The new O-ring seals at lid, drain port, vent port, and test port shall be installed during preparation of the MP197HB package for down-ending.

DSCs in dry storage under their approved license renewal period are subject to the provisions of an Aging Management Program (AMP) under the requirements of 10 CFR Part 72 and in accordance with the guidance of NUREG-1927. The AMP is designed to ensure that no aging effects result in the loss of the intended safety function of the DSCs over the storage period, including those required during transport, e.g., preventing water from leaking into the DSC during transportation. Prior to transport of a DSC in the MP197HB package, a specific evaluation is performed to ensure that their integrity is maintained. Transportation of a DSC that fails a leak test is not permitted.

If the packaging contains high burnup fuel assemblies, both a radiation survey (both neutron and gamma) and a thermal survey of the package are performed to evaluate the axial radiation and thermal source distributions prior to transportation. The user should verify that the exterior surface of the package does not exceed the temperature limits of 185^{o}F in an exclusive-use shipment, in compliance with 10 CFR 71.43(g) and 49 CFR 173.442.

7.2 Package Unloading

For the inner top cover plate cutting operations for the wet loading of 24PT4, 32PT, 32PTH 32PTH1, 37PTH, 61BT, 61BTH and 69BTH DSCs, the users need to verify that the measured hydrogen concentration does not exceed a limit of 2.4%. If the limit is exceeded, the users need to stop all welding operations and purge the DSC cavity with 2-3 psig helium to reduce the hydrogen concentration safely below the 2.4% limit. In the event of a confirmation of a fuel breach, during unloading operations described in Chapter A.7, Table A-7-5, of the application, further transports shall be suspended until the cause of the breach is determined. If the breach occurred during transportation, the underlying cause must be remedied prior to the resumption of transports.

7.3 Preparation of Empty Package for Transport

Previously used and empty packages shall be prepared for transport per the requirements of 49 CFR 173.427.

7.4 Evaluation Findings

The staff reviewed the Operating Procedures in Chapters A7 and 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation.

On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures, as outlined in the application, provide adequate measures and reasonable assurance for safe operation of the package in accordance with 10 CFR Part 71.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application identifies the inspections, acceptance tests and maintenance programs to be conducted on the Model No. MP-197HB package and verifies their compliance with the requirements of 10 CFR Part 71.

8.1 Acceptance Tests

The containment welds of both the packaging and of the DSCs are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB. Welds of the non-containment structure are inspected per the NDE acceptance criteria of ASME B&PV Code, Subsection NF. The DSC fuel baskets are designed, fabricated, and inspected in accordance with the ASME B&PV Code Subsection NG. Fusion weld tests as required are shown on drawings, and alternatives to the code are described in Chapter A.2, Section A.2.1.4, and Appendix A.2.13.13 of the application.

Trunnions, either a non-single failure proof or a single failure proof depending on site and transfer operation requirements, are fabricated and tested in accordance with ANSI N14.6. A load test of 3.0 times the design lift load (for single failure proof trunnions) or 1.5 times the design lift load (for non-single failure proof trunnions) is applied to the trunnions for a period of ten minutes, to ensure that the trunnions can perform satisfactorily.

A pressure test is performed, in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 or NB-6300, on the packaging assembly at a pressure between 40.0 and 45.0 psig, i.e., above 1.5 times the MNOP of 12.7 psig.

Prior to lead pour and final machining of the inner shell, the cylindrical portion of the containment boundary, including the bottom end closure, is leak-tested per ANSI N14.5 or ISO-12807, using temporary closures and seals for the ram access cover plate and lid. Leakage testing is performed during the fabrication process in conjunction with the non-destructive examination of the inner shell welds in accordance with Section III, Subsection NB. An MT or PT examination of every weld layer in the shell-to-top-forging closure weld and an MT or PT examination of all final machined weld surfaces of the inner shell is performed per the Code.

The fabrication verification leakage tests include the following: (i) vent port closure bolt seal integrity, (ii) drain port closure bolt seal integrity, (iii) lid seal integrity, and (iv) RAM access closure plate seal integrity. The acceptance criterion requires each component to be individually leak-tight, i.e., less than 1×10^{-7} ref cm^3/s. The DSC is leakage tested to be leak-tight in accordance with ANSI N14.5. The welds of the DSC inner top cover plate and vent/drain cover plates are also leakage tested to $\leq 1.0 \times 10^{-7}$ ref-cm^3/s.

The integrity of the poured lead shielding is confirmed via gamma scanning prior to installation of the neutron shield. The radial neutron shield is protected from damage or loss by the aluminum and steel enclosure. The neutron shield material, VYAL B, is a proprietary vinyl ester resin mixed with alumina hydrate and zinc borate, which are added for their fire retardant properties. The minimum resin density in acceptance testing is 1.75 g/cm^3. The neutron absorber used for criticality control in the DSC baskets may consist of either (i) Boron-aluminum alloy (borated aluminum), (ii)

Boron carbide/Aluminum metal matrix composite (MMC), or (iii) Boral®. To assure performance of the neutron absorber's design function, only the presence of B^{10} and the uniformity of its distribution need to be verified, with testing requirements specific to each material. Qualification testing is specified in Chapter 8 of the application and is also subject to the process controls specified in Chapter 8. Neutron absorbers shall be 100% visually inspected in accordance with the certificate holder's QA procedures. Neutron Absorber thermal conductivity acceptance testing shall conform to ASTM E1225, ASTM E1461, or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material.

However, to provide additional assurance that the thermal performance of the fabricated cask is equal to or exceeds the theoretical performance reported in the application, a thermal test is performed after fabrication of the Model No. MP197HB package.

8.2 Maintenance

A maintenance inspection and test program schedule is provided in Table 8.2.1 of the application.

Within 14 months prior to any lift of the Model No. MP197HB package, the front trunnions are subject to either a test load equal to 300% of the maximum service load for single failure proof trunnions or a test load equal to 150% of the maximum service load for non-single failure proof trunnions. Dimensional testing, visual inspection and nondestructive examination of accessible critical areas of the trunnions, including the bearing surfaces, are performed in accordance with paragraph 6.3.1 of ANSI N14.6.

The lid, RAM access closure plate, drain and vent ports, which are all containment boundary components, shall be subject to periodic maintenance, and pre-shipment leakage testing in accordance with ANSI N14.5 or ISO-12807. At a minimum, the package lid bolts are replaced at least every 250 shipments (round-trip) to ensure that adequate fatigue strength is maintained. Impact limiters are leakage tested once every five years to ensure that water has not entered the impact limiters. If the leakage test indicates that the impact limiters have a leak, a humidity test will be performed to verify that there is no free water in the impact limiters.

There are no periodic tests or inspections required for the shielding of the package, but radiation surveys are performed on the package exterior to ensure that the limits specified in 10 CFR 71.47 are met prior to each shipment. The material composition of the VYAL-B neutron shielding resin employed in the shielding calculations are based on minimum guaranteed values that are determined as a result of extensive tests under various (including extreme) environmental conditions. The 10CFR Part 71 dose rate compliance measurements serve to indicate the shielding effectiveness of the package. Therefore, periodic tests for the neutron shielding resin are not necessary.

8.3 Evaluation Findings

The containment boundary components of lid, vent port, drain port, and RAM access closure plate shall be subject to maintenance and periodic leakage testing with each component $\leq 1 \times 10^{-7}$ ref-cm^3/s and pre-shipment leakage testing with each component $\leq 1 \times 10^{-3}$ ref-cm^3/s.

Based on the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71. Further, the certificate of compliance is conditioned to specify that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

CONDITIONS

The following changes were made to the certificate of compliance:

Condition No. 5(a)(2) was rewritten for clarity.

Condition No. 5(a)(3) was rewritten for clarity. Also, the nine NUHOMS® DSCs (24PT4, 24PTH, 32PT, 32PTH, 32PTH1, 37PTH, 61BT, 61BTH, and 69BTH were introduced. A clarification was added for the fins to be considered as an optional feature for heat loads less than or equal to 26 kW while they are required to accommodate the NUHOMS®-69BTH DSC with heat loads greater than 26 kW. A clarification was also added for spacers to be installed in the MP197HB overpack or DSC cavity, if necessary, to limit the axial gaps between the components, as specified in Chapter A.7, Table A.7-1 of the application.

Condition No. 5(a)(4) was modified to add one drawing to the MP-197 list of drawings

Condition No. 5(a)(5) was modified to add 64 drawings to the MP197HB list of drawings.

Condition No. 5(b)(1), Table 1, was modified to add units to the pellet diameter.

Condition No. 5(c) was modified to specify the type and form of contents as either (a) Fuel assemblies stored inside any of the nine DSCs, as described in Chapter A.7, Section A.7.1 of the application, or (b) dry irradiated and/or contaminated nonfuel bearing solid materials in an RWC as described in Chapter A.7, Section A.7.1, of the application. The maximum quantity of material per package is specified in Chapter A.7, Section A.7.1, of the application. The maximum peaking factor of the fuel assembly average burnup in all fuel contents shall not exceed 1.212 and 1.152 for BWR and PWR fuel, respectively, for burnups greater than 45 GWd/MTU.

Condition No. 8 was modified to clarify the additional operating requirements of the NUHOMS®-MP197 package including (i) verification of the basket type, (ii) verification that the fuel assemblies meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits, and (iii) replacement of the package lid bolts after 85, or fewer, roundtrip shipments to ensure that the allowable fatigue damage factor will not be exceeded during NCT.

Condition No. 9 was added to specify the additional operating requirements of the NUHOMS®-MP197HB package, including: (i) development of alternate procedures to address site specific conditions and requirements; (ii) verification of the integrity of the DSC that has been used in storage, prior to transportation. Condition No. 9 also requires that the effectiveness of the inspection and verification techniques shall be demonstrated on mockups or working systems, prior to transportation, and that the package lid bolts shall be replaced after 250, or fewer, round trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during NCT.

Condition No. 12 authorizes the use of the previous certificate for approximately one year.

The Reference Section was modified to include Transnuclear, Inc., safety analysis report for the NUHOMS®-MP197 Transport Packaging, dated January 10, 2014, as supplemented on March 12 and April 22, 2014.

CONCLUSION

Based on the statements and representations contained in the application, and the conditions listed above, the staff concludes that the Model No. NUHOMS®-MP197HB package has been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9302, Revision No. 7,
on April 23, 2014.