CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATION NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9302	11	71-9302	USA/9302/B(U)F-96	1 OF	19

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, “Packaging and Transportation of Radioactive Material.”

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION.

c. ISSUED TO (Name and Address)

TN Americas LLC
7160 Riverwood Drive, Suite 200
Columbia, MD 21046

d. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NUHOMS®-MP197 Transportation Packaging Safety Analysis Report, Revision No. 21, dated November 2022.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos: NUHOMS®-MP197, NUHOMS®-MP197HB

(2) Description: NUHOMS®-MP197

The NUHOMS®-MP197 package consists of an outer packaging, used for the transport of the NUHOMS®-61BT dry shielded canister (DSC). Weights and dimensions noted below are approximate values.

Packaging

The NUHOMS®-MP197 packaging is fabricated primarily of stainless steel. Non-stainless-steel items include the lead shielding between the containment boundary inner shell and the structural shell, the O-ring seals, the neutron shield, and carbon steel closure bolts. The body of the packaging consists of a 1.25 inch thick, 68 inch inside diameter, stainless steel inner (containment) shell and a 2.5 inch thick, 82 inch outside diameter stainless steel structural shell, without impact limiters, which sandwich the 3.25-inch-thick cast lead shielding. The packaging is 208 inches long and has an outer diameter of 91.5 inches. The weight of the packaging body is 148,840 pounds including about 10,000 pounds of neutron shield and 60,000 pounds of cast lead.

5.(a)(2) Description, NUHOMS®-MP197 (continued)

The containment system of the NUHOMS®-MP197 packaging consists of the inner shell, a 6.50-inch-thick bottom plate, a 2.5-inch-thick radioactive material (RAM) access closure with a 24 inch diameter, a top closure flange, a 4.5 inch thick top closure lid with closure bolts, drain port closures and bolts, and double O-ring seals for each penetration. The packaging cavity is pressurized to above atmospheric pressure with an inert gas, helium. Radial shielding is provided by 4 inches of stainless steel, 3.25 inches of lead, and 4.5 inches of neutron shielding. Four removable trunnions are provided for handling and lifting of the package.

Dry Shielded Canister (DSC)

The DSC allows the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless-steel shell, with an outside diameter of 67 inches and an external length of 200 inches, and of a basket assembly designed to accommodate 61 intact boiling-water reactor (BWR) fuel assemblies, with or without fuel channels.

The basket structure consists of a welded assembly of stainless-steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless-steel boxes and support rails. The poison plates, constructed from borated aluminum, provide criticality control and a heat conduction path from the fuel assemblies to the canister wall. No credit is given to the DSC as a containment boundary.

Impact Limiters

The two impact limiters, consisting of a laminate of balsa wood and redwood encased in stainless steel shells, are attached to the top (front) and bottom (rear) of the packaging by 12 bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two hoist rings for handling. The hoist rings are threaded into the impact limiter shell. During transportation, the impact limiter hoist rings are removed. An aluminum thermal shield is added to the bottom impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts is approximately 28,000 lbs. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the body of the package during transport.

5.(a)(3) Description, NUHOMS®-MP197HB

The NUHOMS®-MP197HB package consists of an outer packaging, which is used for the off-site transport of any one of the nine NUHOMS® DSCs (24PT4, 24PTH, 32PT, 32PTH, 32PTH1, 37PTH, 61BT, 61BTH, and 69BTH). It is also used to transport a Radioactive Waste Container (RWC) with dry irradiated and/or contaminated non-fuel bearing solid materials. Weights and dimensions are approximate values, unless otherwise noted.

Packaging

The MP197HB packaging is a modified version of the MP197 packaging described in 5(a)(2).

The packaging is fabricated primarily of nickel-alloy steel (NAS). Other materials include the cast lead shielding between the containment boundary inner shell and the structural shell, the O-ring seals, the resin neutron shield, and the carbon steel closure bolts. Socket headed cap screws (bolts) are used to secure the lid to the package body and the RAM access closure plate to the bottom of the package. The body of the packaging consists of a NAS inner shell, 1.25 inch thick with a 70.5 inch inside diameter, and a NAS outer shell, 2.75 inch thick with a 84.5 inch outside diameter, which sandwich the 3 inch thick cast lead shielding material.

The packaging is 271.25 inch long with a diameter of 126 inches, when both impact limiters are installed. The packaging diameter, including the radial neutron shield, is 97.75 inches without the fins or 104.25 inches with the fins. The fins are an optional feature for heat loads less than or equal to 26 kW. The packaging cavity is 199.25 inches long and 70.5 inches in diameter without the internal sleeve (discussed below) or 68 inches in diameter with the sleeve.

The MP197HB uses an internal sleeve for smaller diameter DSCs and secondary containers. The inner sleeve is designed with slots to accommodate the existing rails inside the packaging and to provide rails inside the sleeve on which the smaller diameter DSCs or secondary containers slide during horizontal loading or unloading of the package.

The gross weight of the loaded package is 152 tons including a maximum payload of 56 tons. Four removable trunnions, attached to the package body, are provided for lifting and handling operations, including rotation of the packaging between the horizontal and vertical orientations.

The package containment boundary consists of the inner shell, a 6.5-inch-thick bottom plate with a 28.88 inch diameter, 2.5 inch thick RAM access closure plate with seal and bolts, a package body flange, 4.5 inch thick lid with seal and bolts, vent and drain ports with closures bolts and seals, and all containment welds.

5.(a)(3) Description, NUHOMS®-MP197HB (continued)

For contents loaded in a dry shielded canister (DSC), an inert atmosphere (helium) is maintained in the package cavity. Helium assists in heat removal and provides a non-reactive environment to protect the fuel assemblies against fuel cladding degradation. Radial shielding is provided by approximately 4 inches of steel, 3 inches of lead and 6.25 inches of neutron shielding assembly.

To accommodate the NUHOMS®-69BTH DSC with heat loads greater than 26 kW, removable external fins are provided for the packaging.

Dry Shielded Canister (DSC)

The function of the DSC, which is placed within the transport package, is identical to that described for the MP197 cask in 5(a)(2) above. The DSC consists of a stainless-steel shell and a basket assembly. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is taken for the DSC as a containment boundary.

There are nine DSC designs and a radioactive waste canister authorized for transport in the NUHOMS®-MP197HB packaging. The packaging cavity is designed to accommodate the larger 69.8-inch diameter DSCs (32PTH, 32PTH1, 37PTH, and 69BTH DSC). To accommodate the smaller 67.3-inch diameter DSCs (24PT4, 24PTH, 32PT, 61BT, and 61BTH DSC) or secondary container (RWC), an inner sleeve is provided. To accommodate the varying lengths of the DSCs and secondary containers, stainless steel or aluminum spacers are provided to limit axial movement of the payload. Spacers are installed in the MP197HB overpack or DSC cavity, if necessary, to limit the axial gaps between the components, as specified in Chapter A.7 of the application. The maximum weight of the payload (DSC including the fuel) is limited to 56 tons.

The DSC basket poison plates are constructed from Boral®, borated aluminum or aluminum/B_4C metal matrix composite (MMC) and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Radioactive Waste Container (RWC)

The RWC confines its contents (payload of dry irradiated and/or contaminated non-fuel bearing solid materials) to provide gamma shielding to limit external radiation levels, but the containment boundary is provided by the MP197HB inner shell, lid, and seals.

The RWC assembly, together with cask cavity spacers, shall provide an equivalent of either 1.70 inches or 0.50 inches of minimum steel shielding, depending upon the RWC design, in the radial direction. A minimum of 5.75 inches equivalent steel shielding shall be provided at the bottom of the canister and a minimum of 7.00 inches equivalent steel shielding at the top of the canister, which for one variant of the RWC are combinations of steel and lead. The maximum weight of the payload (RWC, including waste) is limited to 56 tons.

5.(a)(3) Description, NUHOMS®-MP197HB (continued)

Impact Limiters

Impact limiters consisting of balsa wood and redwood encased in stainless steel shells are attached at the front and rear end of the package during shipment by twelve (12) attachment bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving internal pressure. Each impact limiter has three hoist rings for handling, and two support angles for supporting the impact limiter in a vertical position during storage. The hoist rings are threaded into the impact limiter shell, while the support angles are welded to the shell. Prior to transport, the impact limiter hoist rings are removed and replaced with bolts. An aluminum thermal shield is added to each impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts is 25,000 lbs. A personnel barrier is mounted to the transportation frame to prevent access to the body of the package.

5.(a)(4) Drawings, NUHOMS®-MP197

The package shall be constructed and assembled in accordance with the following TN Americas LLC, Drawing numbers:

1093-71-1, Revision 0,
NUHOMS®-197 Packaging
Transport Configuration

1093-71-2, Revision 1,
NUHOMS®-197 Packaging
General Arrangement

1093-71-3, Revision 1,
NUHOMS®-MP197 Packaging
Parts List

1093-71-4, Revision 1,
NUHOMS®-MP197 Packaging
Cask Body Assembly

1093-71-5, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-6, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-7, Revision 0,
NUHOMS®-MP197 Packaging
Lid Assembly & Details

1093-71-8, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Assembly

1093-71-9, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Details

1093-71-10, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Assembly

1093-71-11, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Details

1093-71-12, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Details

1093-71-13, Revision 1, NUHOMS®-61BT Transportable Canister for BWR Fuel General Assembly

5.(a)(4) Drawings, NUHOMS®-MP197 (continued)

1093-71-14, Revision 1, NUHOMS®-61BT Transportable Canister for BWR Fuel General Assembly

1093-71-15, Revision 2, NUHOMS®-61BT Transportable Canister for BWR Fuel Shell Assembly

1093-71-16, Revision 0, NUHOMS®-61BT Transportable Canister for BWR Fuel Shell Assembly

1093-71-17, Revision 2, NUHOMS®-61BT Transportable Canister for BWR Fuel Canister Details

1093-71-18, Revision 1, NUHOMS®-61BT Transportable Canister for BWR Fuel Canister Details

1093-71-20, Revision 0, NUHOMS®-MP197 Packaging Regulatory Plate

1093-71-21, Revision 0, NUHOMS®-MP197 Packaging on Transport Skids

5.(a)(5) Drawings, NUHOMS®-MP197HB

The NUHOMS®-MP197HB package shall be constructed and assembled in accordance with the following TN Americas LLC drawings:

MP197HB-71-1001 Rev 5	NUHOMS®-MP197HB Packaging Transport Configuration (2 sheets)
MP197HB-71-1002 Rev 10	NUHOMS®-MP197HB Packaging Parts List (2 sheets)
MP197HB-71-1003 Rev 4	NUHOMS®-MP197HB Packaging General Arrangement (1 sheet)
MP197HB-71-1004 Rev 7	NUHOMS®-MP197HB Packaging Cask Body Assembly (1 sheet)
MP197HB-71-1005 Rev 10	NUHOMS®-MP197HB Packaging Cask Body Details (3 sheets)
MP197HB-71-1006 Rev 6	NUHOMS®-MP197HB Packaging Lid Assembly and Detail (1 sheet)
MP197HB-71-1008 Rev 5	NUHOMS®-MP197HB Packaging Impact Limiter Assembly (1 sheet)

MP197HB-71-1009 Rev 5 NUHOMS®-MP197HB Packaging Impact Limiter Details (1 sheet)

MP197HB-71-1011 Rev 1	NUHOMS®-MP197HB Packaging Transport Configuration Outer Sleeve with Fins Option (1 sheet)
MP197HB-71-1014 Rev 3	NUHOMS®-MP197HB Packaging Internal Sleeve Design (1 sheet)
NUH24PT4-71-1001 Rev 0	NUHOMS® 24PT4 Transportable Canister for PWR Fuel Basket Assembly (5 sheets)
NUH24PT4-71-1002 Rev 0	NUHOMS® 24PT4 Transportable Canister for PWR Fuel Main Assembly (8 sheets)
NUH24PT4-71-1003 Rev 0	NUHOMS® 24PT4 Transportable Canister for PWR Fuel Failed Fuel Can (4 sheets)
NUH32PT-71-1000 Rev 0	NUHOMS® 32PT Transportable Canister for PWR Fuel Summary Dimensions (1 sheet)
NUH32PT-71-1001 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel Main Assembly (5 sheets)
NUH32PT-71-1002 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel Shell Assembly (3 sheets)
NUH32PT-71-1003 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel “A” Basket Assembly (16 Poison/16 Compartment Plates) (8 sheets)
NUH32PT-71-1004 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel Aluminum Transition Rail – R90 (2 sheets)
NUH32PT-71-1005 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel Aluminum Transition Rail –R45 (1 sheet)
NUH32PT-71-1006 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel “A/B/C/D” Basket Assembly (20 Poison/12 Compartment Plates) (6 sheets)
NUH32PT-71-1007 Rev 1	NUHOMS® 32PT Transportable Canister for PWR Fuel “A/B/C/D” Basket Assembly (24 Poison/8 Compartment Plates) (8 sheets)
NUH24PTH-71-1000 Rev 1	NUHOMS® 24PTH Transportable Canister for PWR Fuel Main Assembly (5 sheets)
NUH24PTH-71-1001 Rev 1	NUHOMS® 24PTH Transportable Canister for PWR Fuel Basket Shell Assembly (4 sheets)
NUH24PTH-71-1002 Rev 1	NUHOMS® 24PTH Transportable Canister for PWR Fuel Shell Assembly (4 sheets)
NUH24PTH-71-1003 Rev 2	NUHOMS® 24PTH Transportable Canister for PWR Fuel Basket Assembly (8 sheets)

NUH24PTH-71-1004 Rev 1	NUHOMS® 24PTH Transportable Canister for PWR Fuel Transition Rails (4 sheets)
NUH24PTH-71-1008 Rev 1	NUHOMS® 24PTHF Transportable Canister for PWR Fuel Failed Fuel Can (2 sheets)
NUH24PTH-71-1009 Rev 1	NUHOMS® 24PTHF Transportable Canister for PWR Fuel Basket Assembly (8 sheets)
NUH32PTH-71-1001 Rev 2	NUHOMS® 32PTH Transportable Canister for PWR Fuel Parts List (1 sheet)
NUH32PTH-71-1002 Rev 1	NUHOMS® 32PTH Transportable Canister for PWR Fuel Main Assembly (1 sheet)
NUH32PTH-71-1003 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Siphon Pipe Details (1 sheet)
NUH32PTH-71-1004 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Inner Top Cover Details (2 sheets)
NUH32PTH-71-1005 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Outer Top Cover Details (1 sheet)
NUH32PTH-71-1006 Rev 0	NUHOMS® 32PTH Transport able Canister for PWR Fuel Shell Assembly (1 sheet)
NUH32PTH-71-1007 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Shell Bottom Details (1 sheet)
NUH32PTH-71-1008 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Grapple Ring Details (1 sheet)
NUH32PTH-71-1009 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Assembly (1 sheet)
NUH32PTH-71-1010 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Assembly Details (1 sheet)
NUH32PTH-71-1011 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Assembly details (1 sheet)
NUH32PTH-71-1012 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Assembly – Details (1 sheet)
NUH32PTH-71-1013 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Rail A180 (1 sheet)
NUH32PTH-71-1014 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Basket Rail A90 (1 sheet)

NUH32PTH-71-1015 Rev 0	NUHOMS® 32PTH Transportable Canister for PWR Fuel Damaged Fuel End Caps (1 sheet)
NUH32PTH Type 1-71-1000 Rev 1	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Main Assembly (4 sheets)
NUH32PTH Type 1-71-1001 Rev 2	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Basket Shell Assembly (4 sheets)
NUH32PTH Type 1-71-1002 Rev 1	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Shell Assembly (4 sheets)
NUH32PTH Type 1-71-1003 Rev 2	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Basket Assembly (7 sheets)
NUH32PTH Type 1-71-1004 Rev 2	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Transition Rails (4 sheets)
NUH32PTH Type 1-71-1010 Rev 1	NUHOMS® 32PTH Type 1 Transportable Canister for PWR Fuel Alternate Top Closure (6 sheets)
NUH32PTH1-71-1000 Rev 1	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Main Assembly (4 sheets)
NUH32PTH1-71-1001 Rev 1	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Basket Shell Assembly (5 sheets)
NUH32PTH1-71-1002 Rev 1	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Shell Assembly (4 sheets)
NUH32PTH1-71-1003 Rev 2	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Basket Assembly (8 sheets)
NUH32PTH1-71-1004 Rev 1	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Transition Rails (7 sheets)
NUH32PTH1-71-1010 Rev 1	NUHOMS® 32PTH1 Transportable Canister for PWR Fuel Alternate Top Closure (6 sheets)
NUH37PTH-71-1001 Rev 2	NUHOMS® 37PTH Transportable Canister for PWR Fuel Main Assembly (4 sheets)
NUH37PTH-71-1002 Rev 3	NUHOMS® 37PTH Transportable Canister for PWR Fuel Basket Shell Assembly (5 sheets)
NUH37PTH-71-1003 Rev 3	NUHOMS® 37PTH Transportable Canister for PWR Fuel Shell Assembly (4 sheets)
NUH37PTH-71-1004 Rev 3	NUHOMS® 37PTH Transportable Canister for PWR Fuel Alternate Top Closure (6 sheets)

NUH37PTH-71-1011 Rev 2	NUHOMS® 37PTH Transportable Canister for PWR Fuel Basket Assembly (7 sheets)
NUH37PTH-71-1012 Rev 1	NUHOMS® 37PTH Transportable Canister for PWR Fuel Transition Rails (7 sheets)
NUH37PTH-71-1015 Rev 0	NUHOMS® 37PTH Transportable Canister for PWR Fuel Damaged Fuel End Caps (1 sheet)
NUH61BT-71-1000 Rev 1	NUHOMS® 61BT Transportable Canister for BWR Fuel Parts List (1 sheet)
NUH61BT-71-1001 Rev 1	NUHOMS® 61BT Transportable Canister for BWR Fuel Basket Assembly (1 sheet)
NUH61BT-71-1002 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Basket Details (1 sheet)
NUH61BT-71-1003 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1004 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1005 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1006 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1007 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1008 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1009 Rev 0	NUHOMS® 61BT Transportable Canister for BWR Fuel Basket Details (1 sheet)
NUH61BT-71-1010 Rev 1	NUHOMS® 61BT Transportable Canister for BWR Fuel Additional Basket Details – Damaged Fuel (4 sheets)
NUH61BTH-71-1000 Rev 1	NUHOMS® 61BTH Type 1 Transportable Canister for BWR Fuel Main Assembly (5 sheets)
NUH61BTH-71-1100 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Main Assembly (7 sheets)
NUH61BTH-71-1101 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Shell Assembly (2 sheets)
NUH61BTH-71-1102 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Basket Assembly (8 sheets)

NUH61BTH-71-1103 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Transition Rails (2 sheets)
NUH61BTH-71-1104 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Damaged Fuel End Caps (1 sheet)
NUH61BTH-71-1105 Rev 1	NUHOMS® 61BTHF Type 2 Transportable Canister for BWR Fuel Failed Fuel Can (2 sheets)
NUH61BTH-71-1106 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister for BWR Fuel Top Grid Assembly Alternate 3 (2 sheets)
NUH69BTH-71-1001 Rev 3	NUHOMS® 69BTH Transportable Canister for BWR Fuel Main Assembly (4 sheets)
NUH69BTH-71-1002 Rev 3	NUHOMS® 69BTH Transportable Canister for BWR Fuel Basket – Shell Assembly (4 sheets)
NUH69BTH-71-1003 Rev 3	NUHOMS® 69BTH Transportable Canister for BWR Fuel Shell Assembly (4 sheets)
NUH69BTH-71-1004 Rev 6	NUHOMS® 69BTH Transportable Canister for BWR Fuel Alternate Top Closure (7 sheets)
NUH69BTH-71-1011 Rev 3	NUHOMS® 69BTH Transportable Canister for BWR Fuel Basket Assembly (5 sheets)
NUH69BTH-71-1012 Rev 4	NUHOMS® 69BTH Transportable Canister for BWR Fuel Transition Rail Assembly And Details (6 sheets)
NUH69BTH-71-1013 Rev 4	NUHOMS® 69BTH Transportable Canister for BWR Fuel Holddown Ring Assembly (2 sheets)
NUH69BTH-71-1014 Rev 2	NUHOMS® 69BTH Transportable Canister for BWR Fuel Damaged Fuel Modification (1 sheet)
NUH69BTH-71-1015 Rev 2	NUHOMS® 69BTH Transportable Canister for BWR Fuel Damaged Fuel End Caps (1 sheet)

RWC-BA-71-1001 Rev. 0	RWC-BA Main Assembly (7 sheets)
RWC-BA-71-1002 Rev. 0	RWC-BA Outer shell Assembly (4 sheets)
RWC-24PT1-71-1001 Rev. 0	RWC-24PT1 Main Assembly (5 sheets)
RWC-24PT4-71-1001 Rev. 0	RWC-21PT4 Main Assembly (8 sheets)
RWC-DD-71-1001 Rev. 0	RWC-DD Main Assembly (7 sheets)
RWC-WA-71-1001 Rev. 0	RWC-WA Main Assembly (6 sheets)

5.(b) Contents of Packaging, NUHOMS®-MP197

(1) Type and Form of Material

(a) Intact irradiated BWR fuel assemblies with or without fuel channels, with uranium oxide pellets and zircaloy cladding. Channel thickness is limited to 0.065 to 0.120 inches. Prior to irradiation, the fuel assemblies must meet the dimensions and specifications of Table 1. Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the NUHOMS®-61BT DSC.

(b) The maximum burn-up and minimum cooling times for the individual assemblies shall meet the requirements of Table 2.

In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of condition 5(b)(1)(c). The maximum total allowable cask heat load is 15.86 kW.

(c) The maximum assembly decay heat of an individual assembly is 260 watts

(d) BWR fuel assembly poison material shall meet the design requirements of Table 3.

TABLE 1[1]

Assembly Type	7x7 49/0	8x8 63/1	8x8 62/2	8x8 60/4	8x8 60/1	9x9 74/2	10x10 92/2
Maximum Initial Enrichment (wt% ^{235}U)	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
Number of Fuel Rods per Assembly	49	63	62	60	60	66-full 8-partial	78-full 14-partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Minimum Cladding Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Pellet Diameter	0.487	0.416	0.410	0.410	0.411	0.376	0.345
Maximum Active Fuel Length (in)	144	146	150	150	150	146-full 90-partial	150-full 93-partial

[1])Maximum Co-59 content in the Top End Fitting region is 4.5 g per assembly
Maximum Co-59 content in the Plenum region is 0.9 g per assembly
Maximum Co-59 content in the In-Core region (including the whole fuel channel) is 4.5 g per assembly
Maximum Co-59 content in the Bottom region is 4.1 g per assembly

TABLE 2

Intact BWR Fuel Assembly Characteristics	
Physical Parameters:	
Fuel Design	7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel
Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR fuel"
Channels	Fuel may be stored with or without fuel channels
Maximum assembly weight	705 lbs
Radiological Parameters:	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	6-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.0 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.65 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

Intact BWR Fuel Assembly Characteristics	
Radiological Parameters:	
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.38 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	15-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.4 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

TABLE 3
Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment

NUHOMS®-61BT DSC Basket Type	Maximum Fuel Assembly Lattice Average Enrichment (wt.% ^{235}U)	Minimum Boron-10 Areal Density for Boral® (g/cm^2)	Minimum Boron-10 Areal Density for Borated Aluminum, Metamic®, and Boralyn® (g/cm^2)	Areal Density Used in the Criticality Evaluation [75% Credit for Boral®] (g/cm^2)
	Intact Fuel Assemblies			
A	3.7	0.025	0.021	0.019
B	4.1	0.038	0.032	0.029
C	4.4	0.048	0.040	0.036

5.(b) Contents of Packaging, NUHOMS®-MP197 (continued)

(2) Maximum quantity of material per package

(a) The quantity of material authorized for transport is 61 intact standard BWR fuel assemblies with or without fuel channels. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For material described in 5(b)(1) the approximate maximum payload is 43,505 lbs.

5.(c) Contents of Packaging, NUHOMS®-MP197HB

(1) Type and Form of Material

(a) Fuel assemblies stored inside any of the nine DSCs, as described in Chapter A.7, Section A.7.1 of the application.

(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.

(2) Maximum quantity of material per package: as specified in Chapter A.7, Section A.7.1 of the application.

(3) 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.

5.(d) Criticality Safety Index: "0"

6. For the 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 that displace an equal or greater amount of water.

7. In addition to the requirements of Subpart G of 10 CFR Part 71, the NUHOMS®-MP197 and NUHOMS®-MP197HB packages shall:

   (a) Be prepared for shipment and operated in accordance with the Operating Procedures in Chapters 7.0 and A.7 of the application, respectively, as supplemented; and

   (b) Meet the Acceptance Tests and Maintenance Program of Chapters 8.0 and A.8 of the application, respectively.

8. Additional operating requirements of the NUHOMS®-MP197 package include:

   (a) Verification of the basket type A, B, or C, by inspection of the last digit of the serial number on the grapple ring at the bottom of the DSC.

   (b) Verification that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Tables 2 and 3. The enrichment limit must correspond to the basket type determined in 8(a) above.

   (c) 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 normal conditions of transport.

9. Additional operating requirements of the NUHOMS®-MP197HB package include:

   (a) Transportation of DSCs is limited to facilities that have the capability to handle uncanned damaged fuel assemblies.

   (b) Detailed site-specific procedures shall be developed to address site specific conditions and requirements that may require the use of different equipment and ordering of steps to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.

   (c) Prior to transportation of DSCs, the condition of the DSC must be evaluated to verify that (i) the containment function of the DSC is maintained and (ii) the degradation of neutron absorbers and basket materials has not occurred to the extent they would no longer comply with applicable materials and dimensions, as specified in condition 5(a)(5). The verification of the containment function shall follow the instructions outlined in Chapter A.7, Section A.7.1.3.1, Step 5 "Evaluation" of the application. The effectiveness of the inspection and verification techniques, outlined in Chapter A.7, Section A.7.1.3.1, Step 5, shall be demonstrated on mockups or working systems, prior to transportation.

(d) The aging management plan and evaluation for each DSC, or set of DSCs, shall be submitted to the NRC prior to shipment.

(e) Replacement of the package lid bolts after 250, or fewer, round-trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during normal conditions of transport.

10. Transport by air is not authorized.

11. The NUHOMS®-MP197 and NUHOMS®-MP197HB packages authorized by this certificate are hereby approved for use under the general license provisions of 10 CFR 71.17.

12. Revision No. 10 of this certificate may be used until August 31, 2023.

13. Expiration Date: January 31, 2028.

REFERENCES

NUHOMS®-MP197 Transportation Packaging Safety Analysis Report, Revision No. 21, dated November 2022.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Signed by Diaz-Sanabria, Yoira
on 01/30/23

Yoira K. Diaz-Sanabria, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: January 30, 2023

SAFETY EVALUATION REPORT
Docket No. 71-9302
Model No. NUHOMS® MP-197HB Package
Certificate of Compliance No. 9302
Revision No. 11

SUMMARY

By letter dated November 30, 2020, ORANO USA (TN Americas LLC) submitted an application for a revision of the Certificate of Compliance (CoC) No. 9302 for the Model No. NUHOMS® MP-197HB transportation package. TN Americas LLC requested to add optional specifications to the packaging and radioactive waste canister(s) (RWC) designs to provide assurance that an RWC currently in storage or an RWC that will be loaded and placed in storage can be transported in the MP-197HB package.

The U.S. Nuclear Regulatory Commission (NRC) staff issued a first request for additional information by letter dated April 7, 2021 (Agencywide Documents Access and Management System [ADAMS] Accession No: ML21090A290) and received responses by letter dated May 28, 2021 (ML2148A067). Staff issued a second request for additional information by letter dated August 6, 2021, (ML21210A202) and received responses by letter dated May 31, 2022 (ML22151A065 and ML22151A066).

TN Americas LLC also requested, during the course of this review by the NRC staff, several changes that were not directly related either to the original amendment request or to the staff's two requests for additional information but came as a result of due diligence being performed by the applicant on the RWC design drawings or to account for non-conforming conditions resulting from the fabrication of the first package. On January 16, 2023, the applicant provided all approved changes and modifications in Revision No. 21 of its consolidated application.

By letter dated May 31, 2022, the applicant requested, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.38, the renewal of the certificate, which had an August 31, 2022, expiration date, at the conclusion of the review of this amendment request.

The staff used the guidance in NUREG-2216, the standard review plan for spent fuel and radioactive material packages, to conduct this review.

EVALUATION

1.0 GENERAL INFORMATION

The package consists of a cask body that has a steel-lead-steel radial wall with a neutron shield enclosed in a steel shell attached to the radial wall's outer steel shell. The cask has a steel base with a ram access port and a steel lid. Impact limiters, made of wood encased in steel shells, are attached to each axial end of the cask. Other components provide some spacing between the impact limiters and the cask lid and base.

The proposed changes to the packaging are confined primarily to the ram access closure plate, and how it is sealed, by limiting the material specifications to one material, instead of several material options currently authorized for the closure plate in the package design.

As part of this amendment request, the applicant revised existing licensing drawings and added new ones for the RWC canister, replacing the previously approved three variants with five variants:

RWC-BA-71-1001 Rev. 0, RWC-BA Main Assembly
RWC-BA-71-1002 Rev. 0, RWC-BA Outer Shell Assembly
RWC-24PT1- 71-1001 Rev. 0, RWC-24PT1 Main Assembly
RWC-24PT4-71-1001 Rev. 0, RWC-24PT4 Main Assembly
RWC-DD-71-1001 Rev. 0, RWC-DD Main Assembly
RWC-WA-71-1001 Rev. 0, RWC-WA Main Assembly

Two of the variants are dry shielded canisters (DSCs) that are modified for use as RWCs (the 24PT1 and 24PT4). The DSC-based RWC variants have a minimum RWC shell thickness that is significantly thinner than the other three RWC variants (0.50 inches versus 1.70 inch).

The RWC drawings specify a minimum axial shielding thickness for each RWC variant. For the 24PT4 variant, those specifications also include minimum specifications for the lead thicknesses in its top and bottom shield plugs. The drawings indicate also that for all the other RWCs the RWCs' axial top and base are made entirely of steel.

As it was also the case for the DSCs, spacers are provided for the RWCs to minimize the axial space within the package's cask cavity and the potential axial movement of RWCs in the package. The necessary spacer size(s) vary by RWC. Spacers may be at either or both ends of the RWC.

The staff reviewed the drawings and verified that they adequately describe the package and reflect the applicant's proposed changes for the package.

2.0 STRUCTURAL AND MATERIALS EVALUATION

The applicant requested welding flexibility and proposed the use of American Society of Mechanical Engineers (ASME) Section IX with ASME Section V inspection requirements, as an alternate to American Welding Society (AWS) D1.1 with AWS D1.6 inspection requirements as identified in the CoC. The staff found that the applicant's response to a request for additional information presented an appropriate basis for the substitution when needed. The code substitution does not change the weld orientation, nor the weld throat size as clarified in the drawing note and the weld performance in both cases is identical. Only the weld preparation is different, with the weld quality inspected to the respective code inspection procedures. The provided comparison between ASME NF-5360 and AWS show that the two codes compare favorably in meeting the fabrication and quality requirements of the weld, thus ensuring adequate weld performance under the conditions experienced by the RWC, during transportation and storage.

The staff concludes that the information provide adequate basis to meet the requirements of 10 CFR 71.31(a) and 71.31(c) in the using either of the welding design codes in the fabrication of the RWC. The changes to the drawing notes allowing this during fabrication are consistent with the basis provided for code substitution.

For the weld design, it was unclear how the applicant had selected a weld allowable value of 21.4 ksi to determine the weld stress ratio for the ASME code welding process. The applicant's response provided additional information on the derivation of the weld allowable stress using the material properties for the appropriate environmental conditions. The applicant stated that the RWC design is based on the criteria specified in the Code and Appendix F. The use of this code was based on the recommendation in NUREG-3854, Fabrication Criteria for Transportation Packages. The stress criteria are in accordance with the ASME BPVC Section III, Subsection NF referenced in Safety Analysis Report (SAR) Appendix A.2.13.4, Table A.2.13.4-1 for partial groove/fillet welds with the allowable stress value for SA-240 Type 304 stainless steel in Table U and Table Y-1of ASME Code, Section II, Part D. The staff concludes that the applicant's basis for the selection of the allowable weld stress of 21.4 ksi is consistent with the design code and acceptable for use with the design of the RWC weld.

Based on the above findings, the staff concluded that the use of the ASME code welding processes as identified above is acceptable for use in fabrication. The staff concludes that the package with the proposed changes will perform its intended functions and maintain its structural integrity to meet the requirements of 10 CFR 71. The staff finds that the materials and materials specifications for the package comply with the requirement in 10 CFR 71.33(a)(5).

Based on the review of the statements and representations in the application, the staff concludes that the package design has been adequately described and evaluated and that the package meets the requirements of 10 CFR 71.

3.0 THERMAL EVALUATION

The objective of the thermal review is to verify that the MP-197HB package design and performance, through the submitted set of change pages, satisfy the thermal requirements of 10 CFR 71.

On page 12 of the revised pages, the applicant proposed a change, which is thermal in nature, but also pertains to the loading procedures. The change being proposed is that the backfill pressure range from 2.5 ± 1.0 psig to 0 to 3.5 psig. The lower limit is reduced but will keep the upper limit.

The applicant provided a justification in Section A7.7/10-2 of the application: the thermal properties of backfilled gases (helium or air) used for the design basis thermal evaluation of the MP-197HB with the RWC are based on properties at atmospheric pressure (0 psig) as listed in SAR Section A.3.2.1, Item 15 (helium) and Item 16 (air). The backfilled pressure range between 0 to 3.5 psig does not reduce any thermal conductivity of the backfilled gases (helium or air) in the RWC. Therefore, the proposed change in the backfill pressure range has no impact on the design basis thermal evaluation.

Staff reviewed the thermal properties listed in this amendment request. After reviewing these values and the appropriate regulations in 10 CFR 71 regarding thermal properties, staff found that the description meets the requirements of 10 CFR 71.33(a)(5).

Staff reviewed the thermal properties for the loading procedures listed in this amendment request and found that their description meets the requirements of 10 CFR 71.31(c) and 10 CFR 71.87.

Staff has reviewed the material properties used in the thermal evaluation and concludes that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR 71.

4.0 CONTAINMENT EVALUATION

The objective of the containment review is to verify that the MP-197HB package design and performance satisfy the containment requirements of 10 CFR 71 and will not exceed the allowable radionuclide release rates under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The applicant provided the following requested changes applicable to the containment review:

1. Contents specification – Contents for the Radioactive Waste Canister (RWC) were modified to allow the presence of organic material and residual water with operating conditions added to require that the effect of gas generation from radiolysis is evaluated prior to shipping, including that flammable gases may be generated.

2. Engineering Drawings – Drawing NUHRWC-71-1001 was replaced by six new drawings for the five RWC variants.

3. Operating Instructions, and Acceptance Tests and Maintenance Instructions – Relevant changes to the containment evaluation include consideration of residual water in the RWC after drying and leak test criteria.

4.1 Contents Specification

4.1.1 Addition of organic material and gas generation characterization requirements

Applicant Justification: The RWC Contents description is modified to be consistent with NRC guidance on organics to say that gas generation from organics or biological growth need to be considered and evaluated for each shipment. The package user will be responsible for characterizing the contents and evaluating the gas generation from thermal decomposition, radiolysis, or biologic growth.

As identified by the applicant, the package user is required to characterize the package contents including organics or biological growth as well as evaluating any gas generation due to thermal decomposition, radiolysis, or biologic growth. The NRC noted that the applicant provided a reference procedure developed by the U.S. Department of Energy (DOE) (NUREG/CR 6673 “Generation of Hydrogen in TRU Waste Transportation Packages”) as a means for calculating the effects of gas generation due to radiolysis, but not for biological material or thermal decomposition as they identified in their justification. That section of the SAR lacked guidance or a sample calculation on how to use the DOE reference document to adequately evaluate these conditions. The NRC determined that it was not sufficient to simply require the package user to develop this procedure or perform gas generation calculations without some minimal guidance including a sample calculation.

As part of a request for additional information (RAI) the NRC requested that the applicant provide a procedure to characterize and quantify any organic material or biological growth within the RWC such that any gas generation due to those contents could be evaluated as a contributor to flammable gas concentration or increases in pressure (Mean Normal Operating Pressure).

In response to the RAI question, the applicant provided a sample calculation using the DOE method referenced above to determine the maximum amount of allowable residual water that could remain in the RWC and still meet the self-imposed 4 percent lower flammability limit and the NRC finds the inclusion of this calculation acceptable as a sample calculation as well as an acceptance criterion for the maximum amount of residual water in an RWC.

Further discussion on biologic growth is continued in this section while discussion on thermal decomposition is continued in section 4.3.2 of the Safety Evaluation Report (SER).

Biologic Material/Growth

The applicant provided the following unsupported statement as a response to the RAI and as supplemental text in the SAR change pages regarding biological material/growth:

> *"There is no history showing that commercial LLRW [Low Level Radioactive Waste emp. added] is generating combustible gas from either biological sources (methane) or rusting of waste container internals."*

The NRC inferred that this statement was related to updated language in the SAR referencing organic growth. As identified above, the applicant removed original text discussing organic materials and biologic growth and subsequently replaced it with updated text discussing organic growth from spent fuel pool water.

The statement regarding no history showing commercial LLRW generating combustible gas is at odds with what the NRC noted in [1], that "*Restricting the contents of the transportation package to solid inorganic materials and prohibiting explosives, pyrophorics, and corrosives (pH less than 2 or greater than 12.5) will preclude the potential for gas generation from biological activity".* The NRC in this instance clearly identified biological activity as a source of gas generation under specific conditions. Further, the statement presented above by the applicant appears to be a line taken verbatim from the NRC response in a *Federal Register* Notice, Volume 80, No.133 (2015), in which a comment from the public and response from the NRC regarding the requirement of a registered user to perform calculations on hydrogen gas generation was published. The comment presented to the NRC objected to the 'requirement' for calculating gas generation from radiolysis because such an activity is not risk informed, and that the NRC should instead focus on observable history of gas generation due to "*…biological sources (methane) or rusting of waste container internals (hydrogen)…*".

The NRC response was such that it only rejected the commenters assertion with respect to generation of methane from biologic material or combustible gases due to rusting of container internals. The NRC did not make any generic finding with respect to gas generation by biologic material, which is again consistent with the NRC position presented in [1]. Since the applicant provides no upper limit on the amount of specific additional materials listed as possible contents and provided no margin of safety on the calculated maximum allowable residual water in the RWC, the additional allowed contents of *"small pieces of cloth or nylon rigging (e.g., choker, rope….), rags, adhesive tape, small rubber pieces from loading operations outside the reactor vessel, or organic growth (algae or microbiological)"* may result in the lower flammability limit exceeding 4% if the allowable residual water was at a maximum.

Since the expected residual water in the RWC is likely to be a small fraction of the allowable residual water to meet the lower flammability limit and it is further unlikely that there will be a significant contribution to gas generation from residual biologic material as currently specified, there is reasonable assurance that the 4 percent flammability limit will not be exceeded from the combined gas generation from radiolysis and biologic material decomposition.

4.2 Engineering Drawings

4.2.1 Metallic seal option: tolerances, seal specification, seal selection evaluation

Applicant Justification of seal material: The use of a metallic seal is an improvement to the existing SAR drawing. Compared to the use of rubber material, the metallic seal is able to last significantly longer in service when properly installed. The use of the metallic seals used for the Nine Mile Point RWC is analyzed in Calculation 35208-0200. The project-specific metallic seals have a pressure rating of 14 psi and temperature rating of 125 °F, which is in alignment for the design conditions of the RWC per the design criteria document 35208-DCD-001.

The NRC requested as an RAI that the applicant submit additional information that indicates required tolerances, seal specifications, and seal selection criteria such that the metallic seal option being requested in this amendment can be evaluated. The applicant elected to provide the referenced calculation 35208-0200 from an evaluation at Nine Mile Point. The NRC reviewed this submittal and finds that it provides the necessary information, including tolerances, seal specifications, and seal selection evaluation to justify the use of a metallic seal option.

4.3 Operating Instructions, and Acceptance Tests and Maintenance Instructions

4.3.1 Content Specification

Applicant Justification: The contents were revised to allow the presence of some organic materials. Allowable contents will not include liquid wastes, sludge or resins. Waste containing organic material is acceptable provided that gas generation from water and organic materials does not lead to potentially flammable of explosive conditions, including the formation of corrosive constituents from radiolysis, biodegradation or chemical reaction. The package user is responsible for characterizing the contents and evaluating the gas generation from thermal decomposition, radiolysis, or biologic growth.

As noted in section 4.1.1 of the SER, the applicant elected to remove references to thermal decomposition and gas generation due to organic material. Gas generation due to thermal decomposition is considered in the following section.

4.3.2 Gas Generation

Applicant Justification: Residual water may remain in RWC not dried to 10 mbar. The amount of residual water may result in exceeding the MNOP [Maximum Normal Operating Pressure, emp. added] or design pressure for the RWC or cask during fire HAC. Both vapor pressure of the water and gas generation from radiolysis need to be considered. In addition, radiolysis could result in exceeding the lower flammability limit for hydrogen.

As part of the response to the RAI referenced in section 4.1.1 of this SER, the applicant provided a calculation for the MNOP considering gas generation due to radiolysis, gas pressure increases due to increased internal temperature, and vapor pressure. The NRC noted for this calculation that the pressure increase due to thermal decomposition presented in [1] was not considered in the calculation for MNOP and no justification was provided by the applicant for this omission.

NRC guidance referenced in [1] states that:
"…to determine the MNOP and to ensure that the MNOP calculation has considered all possible sources of gases, such as the following:

- *hydrogen or other gases resulting from the radiolysis of water*
- *gases initially present in the package*
- *saturated vapor, including water vapor from the contents or the packaging*
- *hydrogen or other gases from the thermal decomposition of materials"*

Further, [1] also states "*Although the gases released from the thermal decomposition or the thermal degradation of materials (e.g., O-ring) are not expected to generate significant hydrogen or other hydrogen gases, the analysis of the pressure in the containment vessel should consider them."*

When reviewing the calculations provided, the portion of MNOP due to radiolysis and vapor pressure due to residual water is approximately 60 percent of the pressure value used for stress calculations for Normal Conditions of Transport. This indicates that there is significant margin remaining in the system design to withstand small to moderate increases to MNOP due to thermal decomposition. Since [1] stipulates that the pressure component of thermal decomposition is expected to be small, there is low risk in not considering the MNOP pressure component due to thermal decomposition. This conclusion must be reconsidered if the proportion of pressure due to radiolysis and vapor pressure increase such that significant margin no longer remains with respect to the NCT design pressure.

4.3.3 Editorial change to ANSI N14.5 Requirements

Change "Each component individually" to "The sum of all components". ANSI N14.5 requires the summation of all individual leakage rates when the criterion is not "leak-tight". This change is consistent with ANSI N14.5 guidance, therefore the NRC finds this change acceptable.

4.4 Findings

The NRC staff reviewed documentation provided by the applicant to verify that statements presented by the applicant are accurate and within acceptable engineering practices. Based on the review of the statements, representations, and calculations in the application, the staff concludes that the containment design and function has been adequately described and evaluated and that the package continues to have adequate containment integrity to meet the requirements of 10 CFR 71.

4.5 References

[1] J. Chang, P. Lien, M. Waters, "Evaluation of Hydrogen Generation and Maximum Normal Operating Pressure for Waste Transportation Packages," Presented at Waste Management Conference, Phoenix, AZ, 2011

5.0 SHIELDING EVALUATION

The purpose of the shielding review is to confirm that the package (the packaging together with its contents) meets the external radiation requirements in 10 CFR 71. The MP-197HB package is designed to transport non-fuel bearing solid irradiated and contaminated materials in radioactive waste canisters (RWCs) as well as commercial spent nuclear fuel assemblies in dry shielded canisters (DSCs). The certificate holder (also referred to as the applicant) has applied to revise the certificate and design of the package to:

1) incorporate various changes in the packaging components, particularly for the package's cask body and the RWCs,

2) specify distinct contents limits for packages containing irradiated and contaminated materials in the RWC variants with the thinner minimum shell thickness of 0.5 inches, and

3) specify the inclusion of organic material in the RWCs' authorized contents.

The staff used the guidance in NUREG-2216, the standard review plan for spent fuel and radioactive material packages, to conduct this review.

The staff, in its review, identified multiple locations in the application that refer to functions and descriptions that pertain to topics that are not related to transport. The staff did not review nor did the staff approve those items as they are not pertinent to approving the package for transportation under 10 CFR 71 requirements. This review is related only to the 10 CFR 71 transportation package certificate. The staff only evaluated those topics that are pertinent to the transportation package certificate.

5.1 Shielding Design Description

5.1.1 Shielding Design Features

5.1.1.1 Package – Cask Body

The package consists of a cask body that has a steel-lead-steel radial wall with a neutron shield enclosed in a steel shell attached to the radial wall's outer steel shell. The cask has a steel base with a ram access port and a steel lid. Impact limiters, made of wood encased in steel shells, are attached to each axial end of the cask. Other components provide some spacing between the impact limiters and the cask lid and base. For the commercial spent fuel contents, the DSCs also provide some radial and axial shielding. The RWCs also provide some radial and axial shielding for the non-fuel bearing solid irradiated and contaminated materials which are loaded in them.

The proposed changes to the packaging include changes to the cask body. These changes are confined primarily to the ram access closure plate and how it is sealed. The staff reviewed the changes and identified that they do not affect the package shielding through the package axial base. No differences in access closure plate materials specifications are introduced. The changes only limit the material specifications that can be used for one option, or type, of the closure plate to one material of the material options that are currently authorized for the closure plate in the package design.

The applicant also added a minimum thickness specification to the cask lid thickness in the drawings, which is a specification that is relevant for shielding and package axial radiation levels. Therefore, other than the cask lid minimum thickness, the staff finds that the changes to the cask body do not affect the package's shielding function.

The staff recognizes that there is a separate specification for minimum thickness of the radial lead shielding for the Unit 01 MP-197HB packaging. This change from the standard design-specified thickness and the impacts on shielding performance and content limits were evaluated in a previous revision of the package certificate (see the staff's safety evaluation report for Revision No. 9, ML19112A168). Based on that previous staff review, the RWCs' contents were limited, in the certificate, to the specifications for the RWC in the Unit 01 package regardless of whether the RWCs are in a MP-197HB packaging with the standard design-specified lead thickness or in the Unit 01 packaging with the thinner lead. The applicant revised the application in this current revision request to only analyze the RWCs in the Unit 01 packaging and to remove any other analysis and contents specifications for the RWCs. In keeping with this approach, the applicant analyzed, and the staff reviewed the impact of the changes to the RWCs and the RWCs' contents in this current request for the RWCs being loaded in the Unit 01 packaging.

The limits and conditions derived from this evaluation will apply to all MP197HB packages containing loaded RWCs. If changes in the future result in separate limits for the Unit 01 packaging versus other packagings fabricated in accordance with this CoC, then the impacts of such changes on what was evaluated and approved for this request will need to be considered. However, the limits derived from evaluations for the Unit 01 package can apply to RWCs in all packagings that have a lead thickness of at least the same thickness as in the Unit 01 packaging, which is 2.77 inches.

5.1.1.2 DSCs

The applicant did not propose any changes to the DSCs or the currently authorized spent fuel contents. The applicant did, in response to staff questions, modify the shielding acceptance tests in Section 8.1.6.1 of the application to provide acceptance tests and criteria for the lead shielding layer in the top and bottom shield plugs of DSCs and RWCs that use lead. Thus, the staff did not review the package's shielding performance for the spent fuel contents except to confirm the impact of the acceptance test and its criteria on package radiation levels.

Lead is used in one RWC and in two DSCs (both are for pressurized water reactor spent fuel). The acceptance test acceptance criteria indicate the amount of void space in the cavity containing the lead which is available for the lead to deform, or slump, under hypothetical accident conditions as well as define the minimum shielding effectiveness of the lead shielding under normal conditions of transport.

The staff's evaluation of the acceptance criteria and their impacts on the package shielding analysis and performance is described in Section 5.3 of this SER.

Though not a design change, per se, for the DSCs, the applicant did modify the application to remove Table A.7-1 of the package operations, which provided dimensional specifications for spacers used with each DSC to minimize the space in the cask cavity and potential DSC axial movement. Instead, the applicant only specifies a maximum axial gap size (0.5 inches) within the package's cask cavity for DSCs at the appropriate operations steps (e.g., see Sections A.7.1.2 and A.7.1.3.2 of the application).

Spacer lengths will be adjusted accordingly with DSC length to meet this package operations requirement. These spacers may be at either or both ends of the DSC. Other necessary spacer and gap size requirements for DSCs (i.e., axial gaps for fuel assemblies and DSC baskets), are also provided with the appropriate package operations steps.

5.1.1.3 RWCs

The applicant did propose several changes to the RWCs that the staff identified as having potential impacts on the shielding design. These changes include modifications of the RWC descriptions and drawings to account for existing and new variants of the RWC. The applicant is replacing the previously approved three variants with five variants, which are described in the revised package drawings for the RWCs and Appendix A.1.4.9A.

Two of the variants are DSCs that are modified for use as RWCs (the 24PT1 and 24PT4). The DSC-based RWC variants have a minimum RWC shell thickness that is significantly thinner than the other three RWC variants, with the thinnest being a minimum of at least 0.50 inches. The other three RWCs have a specified minimum thickness of 1.70 inch minimum, which is slightly less than the minimum specified in previous CoC revisions.

The RWC drawings, which are incorporated by reference in the CoC, also specify minimum axial shielding thicknesses for each RWC variant. For the 24PT4 variant, those specifications also include minimum specifications for the lead thicknesses in its top and bottom shield plugs. The drawing for the 24PT4 RWC explicitly states the minimum lead thickness in the bottom shield plug and in the top shield plug. Other than for the 24PT4 RWC which uses lead in its top and bottom shield plugs, the RWC drawings specify that the RWCs' axial top and base are made entirely of steel. The drawings also specify maximum radial and axial gaps between the lead and steel components of the 24PT4 RWC's shield plugs.

The staff notes that the drawings indicate a gap between the lead and the top and bottom steel components of each shield plug. The staff expects that this allows for an axial gap to exist between either the lead and the steel component above the lead or between the lead and the steel component below the lead versus allowing for a gap of the specified size to exist between the lead and both steel components above and below the lead (i.e., either a gap above the lead or below the lead but not both above and below the lead). However, even if it allowed for gaps to exist at both locations, the applicant's analysis for the RWCs (as described and evaluated in this SER chapter) remain bounding for both normal conditions of transport and hypothetical accident conditions.

The staff also notes that, per Note 2 of the drawing for the 24PT4 RWC, the dimensions shown for the lead in the shield plugs are the dimensions of the lead and not the cavity in which the lead is placed. Thus, the shown dimensions are the amount of lead to be evaluated in the shielding analysis.

The drawings allow for crediting of multiple plates or shells toward compliance with the specified minimum radial shell, base, and top thicknesses for the RWCs. To be credited as part of the minimum RWC shielding that is specified in the RWC drawing, these plates or shells must meet the material specifications for the axial and radial RWC components as described in the respective RWC drawing. Furthermore, they must ensure that the full RWC cavity length has the specified minimum radial thickness with the specified steel. This will ensure that the contents cannot have a configuration with less shielding than the applicant evaluated in the shielding analysis, which corresponds to the minimum in the RWC drawings. In any case the minimum shielding specified in the RWC drawings must be met in terms of both the materials and the minimum dimensions.

In its review, the staff recognizes that these plates or shells may be components of liners, baskets, or sleeves that are used with the RWCs, as allowed by the drawings. The staff finds their use as described above to be acceptable as it ensures the minimum shielding specified in the drawings for the RWC is met and will be consistent with the shielding configuration in the applicant's analysis. The staff recognizes that liners typically include openings in the bottom to allow draining into the cavity of the RWC. The staff anticipates the same to be true of the baskets and sleeves that may be used in the RWCs.

Thus, the staff had a concern about these items' ability to prevent material of small enough size from moving into less shielded configurations in the RWC cavity. The applicant addressed this concern by analyzing package radiation levels due to loose contamination accumulating in a less shielded configuration in the RWC, as described later in this SER.

The drawing for the 24PT4 RWC provides a material specification for the lead used in that RWC variant. The staff finds that the use of lead as a shielding material in the RWC axial base and top results in a need to evaluate the impact on package radiation levels of the gaps between the lead and the steel components under normal conditions of transport and slump and creep under hypothetical accident conditions. Section 5.3 of this SER discusses the treatment of this RWC's lead shielding, including lead slump.

Just like for the DSCs, spacers are provided for the RWCs to minimize the axial space within the package's cask cavity and the potential axial movement of RWCs in the package. The necessary spacer size(s) to achieve this space requirement (a maximum of 0.5 inches) will vary by RWC. This requirement is now specified explicitly in the appropriate package operations steps now that the applicant has removed Table A.7-1 of the package operations. Spacers may be at either or both ends of the RWC.

The staff notes that the application indicates that the shielding analysis assumes that the axial position of the RWC is such that its base is about 2.7 inches above the base of the package's cavity. As described in the application, it is possible that the RWC may rest on, or touch, the base of the package cavity. Thus, it is not clear that the specification regarding spacers is sufficient to ensure the actual package configuration is consistent with the shielding analysis.

However, the staff does not expect that shifting the RWC down to the base of the package cavity will result in noticeable increases in package radial radiation levels. The staff also expects that, while there may be small increases in package radiation levels at the package base, these increased radiation levels will be relatively small and well within the margins to the regulatory limits. Thus, the staff finds the specification of the spacers in terms of maximum axial gap size to be acceptable.

From the information provided initially in the application, it appeared that at least one RWC variant may have the option for filtered venting. The applicant however stated that this vented configuration is for an RWC in a dry storage configuration and that the penetrations in the RWC top cover plate that allow venting during storage will be sealed either with port covers over or plugs inserted into those penetrations prior to the RWC being loaded into the package. Thus, for transportation, no RWC will be vented. The staff finds that this is acceptable because it ensures there is no potential for radioactive materials, including loose contamination, to get into a configuration with less shielding than is analyzed in the applicant's shielding analysis. The staff also confirmed that the RWC drawings and package operations descriptions include sufficient information to ensure RWC's are not vented in the transport configuration.

5.1.1.4 Package Drawings

As part of the review, the staff reviewed the package drawings to identify the changes the applicant made to the drawings to ensure identification of all changes potentially impacting package shielding performance. The staff also reviewed the drawings to ensure the drawings adequately describe and reflect the applicant's proposed changes for the package. Based on its review, the staff finds that the package drawings, with the proposed changes, adequately describe the packaging features that impact shielding performance. This includes having the needed dimensions and material specifications for the RWCs.

While the package drawings do not identify tolerances or minima on all dimensions, the drawings do specify the materials standards for the RWC items and for the cask body items that are standard materials. These standards include requirements for dimensional tolerances. With these standards specified in the drawings, the as-fabricated packaging components will need to meet the tolerances in those standards. The RWC drawings also specify information to identify maximum gap sizes, or reasonably maximum gap sizes, between components relevant to shielding, which can be important for radiation streaming. The drawings also provide enough information about the vent and drain ports, which can also be important for radiation streaming.

For components like the packaging's radial lead shielding and the neutron shielding, minimum dimensional limits and material specifications that impact the package's shielding function are specified in the acceptance tests. Fabricated packages must meet these minimum specifications since they are incorporated by reference into the package certificate.

5.1.1.5 Summary

Based on its review, as described above, the staff finds that the application adequately describes the proposed changes to the packaging design that impact the shielding performance of the package.

5.1.2 Codes and Standards

Regarding the design of the packaging, as has been the case with previous revisions to the package certificate, the applicant did not use any codes or standards. For the evaluation of the package's shielding performance, the applicant used the same computer codes and industry standard for conversion of gamma and neutron fluence rates to radiation levels that the applicant used in previous revisions. The shielding evaluation section of this section of the SER includes a description of these codes and this standard.

5.1.3 Summary Tables of Maximum External Radiation Levels

As described above, the authorized MP-197HB package contents include commercial spent nuclear fuel assemblies in DSCs and solid, non-fuel bearing irradiated and contaminated materials in RWCs. The cask body changes do not affect the package's shielding performance and radiation levels for the spent fuel contents; thus, there are no impacts to the maximum package radiation levels for the spent fuel contents due to the cask body changes. The application tables summarizing these radiation levels remain unchanged from before.

As described in Section 5.3 of this SER, there are implications from the new acceptance test in Section 8.1.6.1 of the application and its acceptance criteria for the radiation levels for the package containing spent fuel in the DSCs. Thus, the applicant performed an analysis to evaluate those implications and the impacts on radiation levels. Based on its analysis, the applicant identified that the impacts on radiation levels are relatively small and so did not modify the tables summarizing these radiation levels. The staff's evaluation of that analysis is described in Section 5.3 of this SER.

For the RWCs, changes to the minimum shell thickness for the thick shell RWCs to be 1.70 inches and the introduction of RWCs with a minimum shell thickness of no less than 0.5 inches affect the shielding capacity of the package and hence the package radiation levels and the allowable contents for a package loaded with an RWC. The applicant also modified the shielding analysis method for the RWCs (e.g., approach to tallies, accounting for additional packaging tolerances and uncertainties), which affects the analyzed package radiation levels.

The applicant determined that other changes, such as allowing organic materials to be present in the RWC contents, do not affect package radiation levels or the RWCs' contents. Given those changes that do affect package radiation levels, the maximum radiation levels for the package for the RWCs have changed.

Based on the staff review and the applicant's proposed changes, as described in Section 5.1.1 and Section 5.3 of this SER, the staff finds that the package radiation levels for the spent fuel contents will not change, except as evaluated in regard to the implications of the new acceptance test and its criteria that are described in Section 8.1.6.1 of the application. Also, while the applicant specified a tolerance for the cask lid, the applicant did not change the analyses with the DSC to account for this tolerance. Given the small impact of this tolerance on the RWC analyses and the portion contributed by gamma radiation to the radiation levels for the DSCs, the staff expects the impact on the DSC analyses would be minor or negligible. Also, based on its review, as described in the following sections of this SER, the staff finds that the maximum radiation levels for the package containing RWCs have changed. The evaluations use only the Unit 01 packaging for determining radiation levels for a package with an RWC; thus, the application now only includes radiation levels for an RWC in the Unit 01 packaging.

Tables A.5-64 and A.5-65 provide the maximum radial radiation levels for normal conditions of transport and hypothetical accident conditions for the package containing an RWC. Table A.5-64 is for the RWC variants that have a minimum radial shell thickness of 1.7 inches, and Table A.5-65 is for the RWC variants that have a minimum radial shell thickness of 0.5 inches. The values in the tables show that the radiation levels are below the applicable regulatory limits for an exclusive use shipment. They also show that the limit at 2 meters from the package's radial side and vehicle side (with the vehicle width assumed to be equal to the diameter of the package's impact limiters) is the most restrictive limit.

The applicant has set the maximum radial 2-meter radiation level to not exceed 9.00 mrem/hr. This provides a margin of 1 mrem/hr to compensate for things such as loose contaminants or debris (also referred to as Chalk River unidentified deposits or CRUD) collecting in the RWC cavity, potentially in a less shielded area of the RWC, as described in the application. The staff's review of the package radiation levels for the RWCs is described in the following sections of this SER.

5.2 Radioactive Materials and Source Terms

The applicant has not proposed any changes to the authorized spent fuel contents. Therefore, the staff only reviewed the RWC contents and the changes to those contents. Section A.1.4.9A.2 of the application includes a description of the RWC contents. The contents are dry, solid, non-fuel bearing irradiated or contaminated materials. The application describes the contents as typically being activated metals or metal oxides that may also have surface contamination.

For the RWCs with a minimum radial shell thickness of 1.70 inches, the maximum allowed activity is 60,800 Ci of Co-60 or equivalent. Section A.5.5.6 of the application shows the maximum analyzed amount to be 60,829 Ci of Co-60, but the difference in source strength is minimal. Since package radiation levels also depend upon the concentration of the radiation source (e.g., higher source concentrations for the same total activity can lead to higher package radiation levels), the application includes a maximum specific activity, based on the shielding analysis, of 6.46 Ci Co-60 or equivalent per kilogram of contents.

The specific activity, per Table A.7-2b is determined by only crediting the mass of contents materials that have a shielding effectiveness at least equal to that of carbon steel. For the RWCs with a minimum radial shell thickness of 0.5 inches, the maximum allowed activity is 8,400 Ci of Co-60 or equivalent with a maximum specific activity of 3.78 Ci of Co-60 or equivalent per kilogram of contents. The same restriction applies for determining the specific activity (i.e., only the mass of contents materials that have a shielding effectiveness at least equal to that of carbon steel can be credited).

The maximum activity and specific activity limits are such that the package's radiation levels do not exceed 9.00 mrem/hr at 2 meters from the radial side, accounting for package tolerance and uncertainties. Recognizing that the RWC contents may include other sources in addition to Co-60, Tables A.7-2d and A.7-2e provide activity limits for contents with gamma radiation at seventeen specific gamma energies for their respective RWCs. The activity limits for each gamma energy result in radiation levels at 2 meters from the radial side that are equal to the radiation levels for the respective Co-60 source.

The applicant developed these activity limits using a response function, calculated with MCNP, for each energy for the respective RWCs. Tables A.5-61 and A.5-62 show the response functions along with the activity limits. These activities and the respective Co-60 activity and specific activity are used to determine the specific activity limits for these gamma energies. A sum of fractions approach is to be used when gamma sources of multiple energies are present in the contents. The staff did perform some simple calculations to confirm the different energies' sources resulted in the same package radiation levels at 2 meters. Those calculations are described below.

The staff considered the applicant's proposed method for determining an RWC's specific activity and the credited contents materials. The staff did a comparison of mass attenuation coefficients over the range of gamma energies for which the applicant proposed contents limits. This comparison showed that some materials would have greater shielding effectiveness than carbon steel at some energies but not at others. Some of these materials are materials that the staff would not find to be appropriate to credit in determining the contents' specific activity.

Consideration of the materials' density (so the comparison was of linear attenuation coefficients) showed only materials that the staff would find acceptable to credit as have shielding effectiveness similar to or better than carbon steel. Thus, the staff expects that the package user appropriately considers the contents' properties affecting shielding effectiveness to determine which materials to credit in the specific activity calculation.

Table A.7-2b of the application also contains requirements related to volume size over which the specific activity is to be determined. The staff finds these requirements are important to ensure against significant variations in the specific activity with the contents of an RWC, which could then lead to high source concentrations that result in unacceptable package radiation levels.

The applicant has proposed to allow organic materials to also be present in the RWCs' contents. Table A.7-2b of the application includes the descriptions of materials to which this organic material is limited. The allowed organic materials do not include liquid wastes, sludge, or resins. Instead, it is limited to small pieces of non-metallic items from loading operations or organic growth from water in the pool where RWC loading takes place. Given that this material, other than the organic growth, originates from non-metallic items used in RWC loading operations, the materials won't be activated themselves, or will be negligibly activated or contaminated. Also, given the practices to ensure pool water cleanliness and limit organic growth as well as to limit the introduction of foreign materials into the pool during loading operations, the staff expects that the quantities, or masses, of these organic materials from organic growth in the pool should be quite limited. Further, the materials, as described will not contribute to the radiation source term of the contents.

The applicant's shielding analysis neglects any impact on shielding that the presence of these materials may have. The staff finds this assumption appropriate since the amount of material may vary and the amount of shielding the material might provide is likely to be quite limited. Given the description of allowed materials and their neglect in the shielding analysis, the staff finds the inclusion of organic materials in the contents as limited in Table A.7-2b of the application to be acceptable in terms of shielding. As described above, the package certificate includes limits on the contents' specific activity in addition to total activity. Based on the foregoing description of the organic materials, the staff finds it acceptable that determination of the specific activity of the loaded RWC contents will not include the organic materials.

Given the material and source specification of the RWCs' contents, the applicant only considers gamma radiation. Related to potential neutron radiation sources, the applicant specifies that any fissile materials in the RWCs will be limited such that they do not exceed the 10 CFR 71.15 limits for exempting material from classification as fissile material. One intent of this limit is to ensure the neutron source from these materials is negligible.

Based on staff experience and conservative evaluations, the neutron source from these materials at the quantity limits in 10 CFR 71.15 could contribute a few percent to package radiation levels. The staff included this consideration in its review of the package radiation levels described below. The staff notes that the limits in 10 CFR 71.15 do not address non-fissile neutron-emitting nuclides. However, based on the descriptions of the waste contents, including the organic materials that may be present, the staff does not expect that non-fissile neutron-emitting nuclides will be present in any amount that will result in a non-negligible contribution to package radiation levels.

The staff also reviewed the source description in the shielding chapter of the application. The maximum source is described. Based on its review, the staff identified that the shielding chapter's content description is adequately consistent with the description in other parts of the application, focusing on those aspects that are important to the package's shielding performance and that the applicant credited in the shielding analysis.

5.3 Shielding Model and Specifications

5.3.1 General

The applicant used shielding models that are similar to the models for the RWCs in previous analyses but with some important differences. The applicant introduced a new model for an RWC with a 0.5-inch minimum radial shell thickness. This model addresses the new RWC variants that have minimum shell thicknesses that are significantly less than the currently approved RWCs. The applicant also reduced the minimum shell thickness in the model for the currently approved RWCs to 1.70 inches. The RWC lid and base thicknesses are the same between the two models and are the smallest of the minimum thicknesses specified for the RWCs in the package drawings. While one of the new RWC variants uses lead in its lid and base, the applicant models the RWCs as just being carbon steel.

The RWC is positioned within the package cavity such that the RWC's base is about 2.7 inches above the base of the package cavity. The staff understands that this positioning may be to account for the spacers that help to fix the RWC's axial position. However, the staff notes that the application allows for the spacers to be either above or below the RWC or a combination of the two. Thus, the RWC may be positioned to rest on the base of the package cavity. The staff's evaluation of the impact of this difference in positions is described in Section 5.1.1 above.

The model uses the specifications of the Unit 01 packaging (with the radial lead shielding at a thickness of 2.77 inches). The applicant modified the model to use the minimum thicknesses of the inner and outer steel shells that surround the radial lead shielding layer. The axial thickness of the wood in the impact limiters is slightly less than the minimum specifications in the package drawings. For the RWCs with the 1.7-inch-thick shell, the applicant modeled the cask lid and base at the minimum thickness allowed by the package tolerances. The remaining packaging components, including the cask lid and base for the 0.5-inch-thick shell RWCs, are modeled at nominal dimensions (thicknesses).

For axial package radiation levels, the staff did some simple calculations to evaluate the impacts of packaging tolerances applicable to each RWC model. Based on those calculations, the staff finds that the margins to the limits are sufficient to compensate for the impacts of packaging components' tolerances for those components that were modeled at nominal dimensions. Since the applicant's model uses nominal dimensions for some radial packaging components, the staff considered the impact of those components' tolerances as part of the staff's evaluation of uncertainties for the package radial radiation levels, as described below. Based on that uncertainty evaluation, the staff finds that modeling of those package components at nominal values is acceptable.

The applicant modeled the 0.5-inch-thick RWC's (also referred to in this SER section as the thinner RWC) contents as a cylinder of 66.18 inches diameter and 100 centimeters height. The applicant's model for the 1.70-inch-thick RWCs' contents is a cylinder of 66 inches diameter and 168 inches height. Since the thinner RWC's contents are more axially compacted and the contents can be positioned at any axial location within the RWC cavity, the applicant performed calculations for the contents being in the bottom of the RWC, at the cask mid-height, and in the top of the RWC. The staff finds that this approach to modeling the contents is acceptable because it ensures that the maximum package radiation levels will be identified for all possible locations of the contents within the RWC.

The applicant modeled the contents in both RWC models as 1g/cc carbon steel with the radioactive source distributed uniformly throughout the content volume. This density, together with the contents' volume and the maximum total activity limit for each RWC model results in the maximum specific activity specified for the respective RWCs in the Table A.7-2b contents specifications.

The models do not include organic material in the contents. The description of this material is discussed in Section 5.2 above. Based on the description of the allowed organic materials and the inclusion of this description for the RWC contents specifications in Table A.7-2b of the application, the staff finds that neglecting the organic materials in the models is acceptable. This is because doing so excludes the shielding effect of the organic materials, which, though small, is conservative and because these materials contribute negligibly to the contents' radiation source.

5.3.2 Loose Radioactive Particulates

As described above, the package design includes the possibility that the minimum RWC shell thickness can be the total thickness of the RWC shell and a liner, sleeve, or basket that can be used inside the RWC. Previously, the application included descriptions of some of the liners that could be used. Based on those descriptions, the staff found that it was unclear that the liners cover the full cavity length of the RWC or sufficient length of the cavity to preclude irradiated or contaminated items or contamination from moving into less shielded areas in the RWC cavity that are outside the liner.

Further, the descriptions indicated that the liners have holes to allow drainage of water; however, it was not clear how the design of those holes prevents content materials from moving out of the liner. While the application indicated that one liner may have a lid or shield plug, it did not indicate that was true for the other RWC liners. Therefore, the staff had concerns about the possibility of radioactive contents shifting into configurations that are less shielded than analyzed in the shielding chapter.

The applicant addressed this concern by including an analysis of the impact of loose contamination, or CRUD, collecting at one point. The applicant calculated the radiation levels at 2 meters from the side of the package that result from this collection of material. The applicant assumed an amount of concentrated Co-60 activity based on information in Section A.4.6.1.2.1 of the application for containment analyses.

The applicant positioned the material in a location of minimal shielding and assumed an RWC shell thickness of 0.50 inches. The applicant calculated a radiation level of 0.3 mrem/hr at 2 meters from the package. The applicant's analysis indicates that there are sufficient margins to the radiation level limits to compensate for this added 0.3 mrem/hr.

The staff reviewed this analysis and its supporting information. The staff considered the information in the containment chapter of the application from which the applicant derived the source terms for its analysis. Considering the ranges of factors used in determining the source strength and the total mass of contents allowed by the certificate, the staff's evaluation indicated a potential increase of 60 percent over the applicant's source strength. The staff also considered the minimum thicknesses that had been specified in a note in the previously proposed RWC drawing revision, which applied only to the thicker shell RWCs, for potential impact to further bound the evaluation, even though these minima were restricted to be localized thicknesses.

A simplified evaluation performed by the staff showed the impacts of these minimum thicknesses to be an increase in radiation levels of less than 30 percent above those calculated with a 0.5-inch minimum RWC shell thickness. The staff considered these impacts in its evaluation of uncertainties described below.

The staff recognizes that the applicant has now modified the descriptions of the RWCs to remove detailed descriptions of any liners, sleeves, or baskets that may be used to attain the minimum shell thickness. The applicant has also removed any descriptions that allow for localized minimum thicknesses that are thinner than the overall minimum thickness of the RWCs' shells. Additionally, the proposed RWC drawings now indicate that the shells for any of the RWC variants may be comprised of multiple plates.

The staff still had some concerns about the potential for the radioactive contents to get into a less shielded configuration if the liners, sleeves, or baskets don't extend the full length of the RWC cavity. However, a review of the modified RWC drawings shows that the minimum thickness must be attained for the full length of the cavity; thus, the contents should never get into a less shielded configuration. Therefore, the staff finds its concern in this regard has been addressed and the applicant's analysis for the CRUD to be acceptable.

5.3.3 Axial Positioning of RWCs and RWCs' contents

As described in Section 5.1.1 above, the length of the RWCs may vary and therefore so does the length of the spacers for the RWCs. The staff considered that these different lengths could have implications for the modeled axial position of the RWC in the package's cask body cavity. The package operations also include options for placement of the cavity spacers such that the spacers may only be in the cavity base, only in the cavity top area, or in both areas. This means the RWC can be sitting on the cavity base, touching the cask lid, or some distance from both the base and lid.

In the applicant's model, the base of the RWC is 2.71 inches above the base of the cask body's cavity. Depending on the RWC length, this means that the RWC will be near the cask lid or several inches below the cask lid. In addition, for the 1.7-inch-thick shell RWCs, the model of RWC contents means that the modeled contents are several inches below the RWC lid (amount varies by RWC).

The staff checked the axial package radiation levels in the application and did simple evaluations of the potential impacts on axial radiation levels when the RWC is moved from its position in the model to touch the cask cavity base or cask lid. For the 1.7-inch-thick shell RWCs, the staff also evaluated the shift of the contents to touch the RWC lid. The staff found the impacts to radiation levels at the base of the package to be minimal and that the margins to the regulatory limits are more than sufficient to compensate for those impacts. The staff's evaluation indicated that the impacts at the axial top end of the package are not insignificant; however, the margins to the regulatory limits are sufficient to compensate for those impacts.

The staff also evaluated for potential impacts of the RWCs' and the RWC contents' axial position on the radial radiation levels of the package. This evaluation included consideration of the possible position of the RWC and its contents in the model that the cavity spacer placement options allow (e.g., see Section A.7.1.3.2 of the application), the lengths of the different RWCs, the axial extent of the package's radial shielding versus the cavity, and, for the 0.5-inch-thick shell RWCs, the three axial positions of the contents within that RWC's cavity.

Based on this evaluation, the staff considered that the applicant's shielding analysis adequately covers the potential impacts of axial positioning of the RWCs and the RWCs' contents on radial package radiation levels.

5.3.4 Dimensional and Material Variations and Penetrations

The staff evaluated the potential impacts of RWC cavity length and diameter variations allowed by the RWC drawing. Since the package's axial shielding is radially uniform, the staff determined that there should be no impact on axial package radiation levels. This conclusion applies even when accounting for gaps in the axial shielding. The staff's determination is based on the applicant's shielding analysis showing that the maximum allowed gap sizes do not impact package axial radiation levels.

In its evaluation regarding radial package radiation levels and axial variation of radial shielding, the staff determined that the outcome of its evaluation of the longer RWCs described above also applies to the evaluation of the allowed dimensional variations. The staff does note, though, that there is a possibility of different RWC diameters introducing some small uncertainty in the radial radiation levels calculated for the package (~1 percent). The staff considered this uncertainty in its evaluation of uncertainties, which includes whether the package radiation levels will be below the regulatory limits when these uncertainties are considered.

The staff notes that the new specifications for the RWC variants include different types of steel, particularly stainless steel 316, for components of some RWC variants. This material specification differs from the previously approved RWC specifications. The shielding model uses carbon steel for the RWCs' steel components. The model uses carbon steel for all the package's steel components that are included in the model, some of which are in actuality stainless steel. Differences in composition and density can have an effect on package radiation levels. Based on a review of those properties, simple evaluations, and staff experience, the staff determined the addition of the new material specification is acceptable from a shielding perspective and the shielding model bounds the effects of the materials that can be used to fabricate the RWCs.

As can be seen in the RWCs' drawings, the RWCs have penetrations through the top cover plates and shield plugs for performing draining and drying operations. The cover plates that cap these penetrations do not extend through the full thickness of the top cover plates and shield plugs. Therefore, these penetrations represent potential areas of radiation streaming. The potential for radiation streaming is particularly pertinent to the thicker shell RWCs for which the configuration of the penetrations is a simple, direct penetration through the top cover plates and top shield plugs.

The applicant's analysis model includes a penetration that has the size of the largest penetration identified in the drawings for these thicker shell RWCs. The model also conservatively ignores the cover plate over the penetration. For the thinner shell RWCs, the configurations of the vent and drain ports are such that there is no direct line of sight, and their size is relatively small. So, the model for these RWCs does not include penetrations. The staff reviewed the models and, based on the descriptions of the penetrations in the RWCs' drawings and experience with shielding analyses related to vent and drain ports, finds the applicant's models to be acceptable.

5.3.5 Lead Shield Plugs, Acceptance Test Criteria, Slump for RWC and DSCs

One RWC variant is derived from a DSC that includes lead in both the top and bottom shield plugs. This lead is precast and fitted to the cavities of the shield plugs in which it is placed. Even in this fabrication method, there will be gaps, both radial and axial gaps, between the lead and the steel components of the shield plugs. The drawings specify the maximum allowed gaps between the lead and the steel in the shield plugs. Also, as noted in Section 8.1.6.1, the gamma shield test for package acceptance tests, the lead may also include flaws up to 0.13 inches thick. This flaw specification is part of a new acceptance test and acceptance criteria for precast lead components that the applicant added to Section 8.1.6.1. The staff considers that these flaws are voids in the lead.

The shielding models, including the model that applies to the RWC variant, use the smallest total minimum thickness of the cover plates and shield plugs for both the top and base of the RWCs. Also, they are modeled as being entirely carbon steel. The models include radial gaps of the maximum dimension specified in the RWC drawings for gaps between the lead and steel in the shield plugs.

Additionally, for the hypothetical accident conditions analysis, the applicant reduced the amount of steel in the top and base of the RWC model that applies to this RWC variant. The remaining thickness of steel is slightly less than the actual amount of steel in the top and base of this RWC variant. Thus, the model is equivalent to having all the lead removed from the RWC's top and bottom shield plugs, which is conservative for treating lead slump under hypothetical accident conditions. Since the total thicknesses of the base and top of the RWC model are less than for the actual RWC and uses carbon steel, which has a significantly lesser shielding effectiveness than lead, the staff finds the RWC model for normal conditions of transport to be acceptable for this RWC variant. Also, since the model for the hypothetical accident conditions uses steel thicknesses in the top and bottom which equates to removal of all lead, the staff finds that model also acceptable for this RWC variant.

The same acceptance test and criteria for the lead in the shield plugs for this RWC variant also apply to the two DSCs that have lead in their shield plugs and are included in the application for this package. These DSCs are the 24PT4 and 24PTH. Thus, the lead in these DSCs' shield plugs will have the same allowed gaps and flaw size specifications, which are important for package radiation levels for a package as prepared for transport and under normal conditions of transport. The lead in these shield plugs is also subject to lead slump under hypothetical accident conditions to the extent that these allowed gaps and flaw size allow.

Given these considerations, the staff reviewed the shielding analyses for the DSCs to ensure the allowed specifications and acceptance criteria for the precast lead shielding did not impact that shielding analysis and package radiation levels for the DSCs. The applicant's shielding analysis uses the 37PTH DSC to bound the package analyses and radiation levels for the 24PT4 and 24PTH. The 37PTH DSC is an all-steel DSC.

The staff used some simple, conservative methods to compare the axial shielding performance of these three DSCs versus each other. The staff's comparison considered both the DSCs' axial shielding and the expected bounding contents for each DSC in terms of radiation source terms. The staff did this comparison for normal conditions of transport, hypothetical accident conditions with lead slump due to an axial drop, and hypothetical accident conditions with lead slump due to a horizontal drop.

The staff used the acceptance criteria and information in the DSCs' drawings to estimate the amount of lead slump that could occur in both hypothetical accident conditions' drop configurations. For the axial drop, the lead slumps to fill the cavity radially, but its axial thickness is reduced. For the horizontal drop, the lead slumps such that its thickness is the size of the DSC shield plug lead cavity, but it leaves portion of the cross-sectional area of the cavity, which is not insignificant, without any lead to shield the DSC contents.

Thus, particularly for the horizontal drop configuration, it was not clear to the staff that the axial radiation levels for the 37PTH DSC would bound those of the 24PT4 DSC or 24PTH DSC. Nor was it clear that the radiation levels for these latter two DSCs would remain below the regulatory limits.

For both the normal conditions of transport and hypothetical accident conditions of transport with slump due to an axial drop, the staff's evaluation indicated that the 37PTH DSC will be bounding for the 24PT4 DSC. The staff's evaluation did not clearly indicate that the 37PTH DSC will be bounding for the 24PTH. However, the staff's evaluation did indicate that there is more than sufficient margin to the regulatory limits applicable to these conditions to compensate for any increases in axial radiation levels for the 24PTH DSC versus the 37PTH DSC.

As noted in Section 5.1.3 above, the applicant performed some additional analyses for DSC radiation levels. These analyses are described in Section A.5.3.1.2.2 of the application and address lead slump for the horizontal drop hypothetical accident conditions. The applicant's calculations address both the differences in shielding and consideration that differences in shielding configuration can lead to different source terms resulting in bounding radiation levels. The applicant's analysis showed increases of less than 10 percent in total radiation levels versus those analyzed with the 37PTH DSC. The applicant also performed bounding calculations that completely removed the lead from the shield plugs. The radiation levels for these latter cases greatly increased but still remained below the regulatory limits.

In its evaluations of the horizontal drop hypothetical accident conditions, the staff determined the extent of the shield plug cavity that would be void of lead due to slump and considered the staff's expected bounding contents for the DSCs. The staff's evaluation focused on the 24PTH DSC versus the 37PTH DSC since the staff's evaluation indicated that the 24PTH DSC would be bounding for the 24PT4 DSC. The staff's evaluations resulted in a void area in the shield plug cavity of size that was very similar to that determined by the applicant. The staff's simple, conservative evaluations indicated that while axial radiation levels of the 24PTH DSC with lead slump could potentially be significantly higher than those reported in Table A.5-2 of the application for the 37PTH DSC, there would still be very large margins to the regulatory limits.

Additionally, the staff finds the applicant's bounding cases (no lead in the shield plugs) provide further confidence that there will be sufficient margin to the regulatory limits for the 24PTH and 24PT4 DSCs with lead slump due to a horizontal drop.

Thus, the staff finds that, while the 37PTH DSC's axial radiation levels aren't always bounding for the 24PT4 and 24PTH DSCs with lead in their shield plugs that meets the drawing specifications and acceptance criteria in Section 8.1.6.1 of the application for all conditions, the axial radiation levels for these latter two DSCs will not exceed the regulatory limits, with margin.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant used the same methods and codes (MCNP5) for the analyses for the new RWC design and contents as for the currently authorized packaging design and contents, with a few exceptions. However, these changes do not affect the types of input and output data; they remain unchanged. The applicant continued to use the staff-accepted conversion factors for converting fluence rates to radiation levels (i.e., the conversion factors in ANSI/ANS 6.1.1-1977).

The applicant's approach is to specify a Co-60 source activity and calculate package radiation levels for that source. Since other radionuclides may also be in the RWCs' contents, the applicant used MCNP to determine a response function for seventeen specific gamma energies. The response function specifies radiation level in mrem/hr per source particle at each gamma energy. The applicant only developed a response function for the radiation levels at two meters from the side of the package.

The applicant's calculations with the Co-60 source indicate that this location is the most restrictive for the package in terms of regulatory radiation level limits. Using the response function at each gamma energy, the applicant developed a maximum source strength for each gamma energy that results in the same package radiation level as the Co-60 source. Tables A.5-61 and A.5-62 of the application show the response functions and maximum source strengths for the RWCs with 1.70-inch-thick shell and for the RWCs with the 0.5-inch-thick shell, respectively.

For contents with gamma energies at different energies than the specified energies, the limit for the next highest energy is used to determine the activity limit. For example, the limit for a 1.60 MeV gamma source would be the limit for 1.75 MeV gammas. The resulting package radiation levels for the Co-60 contents are in Tables A.5-64 and A.5-65 of the application.

For determining the response functions and source strengths for this thinner shell RWC, the applicant did separate MCNP calculations for each gamma energy. Also, the applicant used a mesh tally for both the Co-60 analyses and the response functions instead of using a surface tally for the response functions, as had been done previously. Use of the mesh tally removed a source of uncertainty arising from the difference in results between a mesh tally and a surface tally for the same source.

The applicant also set a target maximum radiation level at the 2-meter location of 9.00 mrem/hr. For the Co-60 source and the source for each gamma energy, the applicant ensured that the calculated radiation level plus one standard deviation does not exceed 9.00 mrem/hr. So, for calculating the maximum source strength at each gamma energy, the response function is the calculated (mrem/hr)/(gamma/sec) plus one standard deviation. The applicant set the target to 9.00 mrem/hr to ensure margin to the regulatory limit to compensate for remaining uncertainties in the analysis and the contribution of the CRUD.

The application describes the mesh sizes of the mesh tally for the DSC analyses. These mesh sizes are fairly large, including on the package surface. The applicant used the same mesh sizes for the RWC analyses. For the mesh at 2 meters from the package surface, the staff considers this to be a reasonable size. For the package surface, where differences like gaps could be expected to impact radiation levels, this size seems to be a bit large. The staff would expect a mesh node size that is more consistent with the size of common radiation detectors (a few inches).

However, the staff notes that there is significant margin to the limits for the surface radiation levels. The staff also noticed that differences in radiation levels between mesh and surface tallies in previous analyses did not exceed about 10 percent. The staff also expects that the impacts would be relatively small for maximum gap sizes of relevance to shielding even for mesh nodes that are more like common detectors in size. Therefore, given these considerations, the staff finds that the mesh size for package surface radiation levels is also acceptable.

5.4.2 External Radiation Levels

Tables A.5-64 and A.5-65 provide the package radiation levels for the RWCs with 1.70-inch-thick shells and RWCs with 0.5-inch-thick shells, respectively. The tables show maximum package radiation levels for both axial and radial locations at the package and vehicle surfaces and at 2 meters from the package surfaces for normal conditions of transport. The tables also show the radiation levels for locations of occupied spaces assumed by the applicant. They also show the radiation levels at 1 meter from the package surfaces for hypothetical accident conditions.

As part of this review, the staff performed a variety of confirmatory evaluations. The purposes of these evaluations included confirming the reasonableness of the radiation levels reported for the two groups of RWCs (the thinner shell RWCs and the thicker shell RWCs), including relative to each other, and the similarity of the variation in radiation levels depending on package surface and distance from that surface. These evaluations were done in ways that accounted for differences in source strength and specific activity, or source strength per Unit mass, between the two groups of RWCs, as well as considered differences between their shielding models.

The staff also performed calculations to confirm the proposed source strengths for different gamma energies result in the same radial 2-meter radiation levels as the Co-60 source and that these radiation levels will not exceed the 9.0 mrem/hr target set by the applicant. Additionally, the staff evaluated the impact of uncertainties and non-conservatisms that the staff identified in the applicant’s analysis and method to confirm whether radiation levels will not exceed the regulatory limits.

The staff performed these evaluations using methods of varying complexity. These methods included relative comparisons of radiation levels by determining transmission fractions using half value thicknesses of materials for a variety of gamma energies. Another method involved adjusting the radiation levels for one group of RWCs by the ratios of source strength and specific activity between the two groups of RWCs to get an estimate of the radiation levels for the other, or second, group of RWCs.

Additionally, the staff used MicroShield for some evaluations and MCNP 6.2 for others. The staff used each method with a recognition of their respective limitations.

The staff used progressively complex methods where that complexity was necessary to provide sufficient confirmation of the applicant's analysis. MCNP 6.2 provides the most complex calculation method of those the staff used with the fewest limitations. In this case, the limitations of the method derive from the complexity of the staff's input including the geometry, variance reduction, and definition of tallies for calculating radiation levels.

The staff used MCNP for the comparison of 2-meter radial radiation levels for the Co-60 source and several gamma energies to each other and versus the applicant's target radiation level. For these calculations, the staff developed a model that sufficiently accurately represented the package's axial and radial shielding configurations, implemented variance reduction, and a sufficiently complex tally structure.

In the early part of the review, the staff identified some concerns based on its evaluations. These included differences in radiation levels for various gamma energies versus the radiation levels for the Co-60 source. Some of these sources resulted in noticeably higher radiation levels. The staff also identified that, with the uncertainties and non-conservatisms in the method used in the initial application for the certificate revision, it was unclear that the regulatory limits would not be exceeded.

The staff's evaluations indicated the limits would be exceeded. Since the applicant's analysis method was the same as for the already approved RWCs and their contents, the staff's concerns extended to those RWCs and their contents too.

Thus, to address those concerns as well as additional changes to the RWCs' specifications, the applicant revised its analysis method and the radiation level results reported in the application. The applicant changed the target 2-meter radial radiation level to be 9.0 mrem/hr to leave a 1.0 mrem/hr margin to address the impacts of CRUD and other remaining uncertainties. The applicant modified the models to include the tolerances on the inner and outer steel shells of the package's cask body.

The applicant's approach now is that the calculated radiation level plus one standard deviation must not exceed the 9.0 mrem/hr target radiation level. Additionally, for the response function, the same tally type and structure is used to calculate radiation levels as for the Co-60 source and a separate calculation is done for each gamma energy.

The staff applied its same evaluation approach and methods to the revised analysis and results. The staff's evaluations of selected gamma energy sources confirmed that the 2-meter radial radiation levels for the different gamma energies will result in similar radiation levels as the Co-60 source and that these levels will not exceed the applicant's 9.0 mrem/hr target.

The staff also confirmed that the radiation level results for the two groups of RWCs are reasonable and consistent with each other considering the differences in the source strength and specific activity allowed for each RWC group and differences in the RWC model (e.g., penetrations in the lid to capture the effects of vent and drain ports, cask lid and base at nominal dimensions versus minimum dimensions).

The staff did identify that there are some remaining uncertainties or non-conservatisms in the applicant's analysis. For example, the neutron shield and shield jacket are at nominal dimensions. The cask base and cask lid are also at nominal thickness for the analyses of the 0.5-inch-thick shell RWCs. The positioning of the RWCs within the package's cask cavity also has some impacts on radiation levels.

Also, contents that do not exceed the fissile exempt classification may still have a small neutron contribution to package radiation levels. Additionally, while fissile nuclides are limited so that the materials can be exempted from fissile material classification per 10 CFR 71.15, this does not have a bearing on the amounts of any non-fissile neutron-emitting nuclides being present. However, as described in Section 5.2 above, based on the contents descriptions and limiting specifications, the staff finds that if any such nuclides are present, they will only be present in quantities that contribute negligibly to package radiation levels.

The staff evaluated the impact of these uncertainties and non-conservatisms together with the contribution from CRUD described earlier. For the neutron contribution, the staff used a conservative approach. Based on these evaluations, the staff determined that the margin to the regulatory limits for the radial side of the package is sufficient to compensate for these effects and ensure the package radiation levels will not exceed the limits.

The staff also reached the same conclusion for the axial ends of the package. Thus, based upon the applicant's analyses and the staff's evaluations, the staff has reasonable assurance that the package radiation levels will not exceed the regulatory limits for a package containing the thick shell RWCs and their contents.

The staff also had performed some simple analyses for investigating the impacts of lead slump on axial package radiation levels for a package containing either a 24PT4 DSC or a 24PTH DSC. The applicant's analysis of lead slump for these DSCs and the staff's evaluation are described in Section 5.3.5 above.

Thus, based on that analysis and evaluation, the staff has reasonable assurance that the package radiation levels for a package containing these DSCs will not exceed the regulatory limits.

5.5 Evaluation Findings

Based on a review of the information and representations provided in the application and the staff's confirmatory calculations, the staff has reasonable assurance that the MP-197HB package with the proposed changes to the packaging and contents satisfies the shielding requirements and limits in 10 CFR 71.

6.0 CRITICALITY EVALUATION

The purpose of the criticality review is to confirm that the package together with its contents meet the requirements in 10 CFR 71 for criticality safety. The MP-197HB package is designed to transport non-fuel bearing solid irradiated and contaminated materials in radioactive waste canisters (RWCs) as well as commercial spent nuclear fuel assemblies in dry shielded canisters (DSCs). The certificate holder (also referred to as the applicant) has applied to revise the certificate and design of the package. These changes include changes to the package's cask body and changes to the RWCs and their contents. The applicant also made changes to information in the criticality safety analysis and the package operations related to criticality safety. The staff used the guidance in NUREG-2216, the standard review plan for spent fuel and radioactive material packages, to conduct this review.

The applicant has limited the contents of the RWCs to materials that do not exceed the limits for classification as fissile exempt material in 10 CFR 71.15. Therefore, the changes to the RWCs and their contents do not involve criticality safety concerns and a criticality review was not needed. The staff considered that the proposed changes to the package's cask body could affect the package's criticality safety function and analyses.

Thus, the staff reviewed the criticality safety design description and analyses in the application to understand how the design is defined and analyzed. Based on that review and the description of the proposed cask body changes, the staff found that the proposed changes do not impact the package's criticality safety function and are acceptable.

The applicant did provide a new page A.6.14-28l, describing the change as a revision to provide the correct table. The staff, in its review, identified that this change provides information the current criticality analyses are based on and so does not impact the criticality analyses.

The applicant made a change to the package operations, on page A.7-5 of the application, to reference the correct table for specifying the depletion parameters, or reactor operating conditions, that must be complied with in loading spent fuel contents in a DSC that relies on burnup credit. The staff confirmed that the correct table is now referenced in the package operations.

The applicant also revised the referenced table, Table A.6.5.8-7, to include a note that is important for clarifying how they are to be used in the package operations to confirm that the spent fuel contents loaded in DSCs using burnup credit comply with the parameter values in this table. The note specifies that the values of the fuel temperature, moderator temperature, soluble boron concentration, and specific power are bounding upper values. This means that spent fuel contents in DSCs using burnup credit must have irradiation histories for which the values of these parameters do not exceed the values in Table A.6.5.8-7 to be considered to comply with parameter values in that table and acceptable package contents.

The staff reviewed the revised description and confirmed that specifying the table's values for these parameters as bounding upper values is consistent with the applicant's burnup credit analyses and supporting information in the appendix regarding burnup credit in NUREG-2216's criticality chapter (Chapter 6) with the exception of specific power.

The applicant's analysis indicates that, for the type of burnup credit the applicant uses, lower specific power results in increased reactivity and so the table value for specific power should be a bounding lower value. This outcome is consistent with the information in the aforementioned NUREG-2216 Chapter 6 appendix.

However, the staff also recognizes that for the same type of burnup credit, the relationship between specific power and reactivity is not all that clear cut. The staff also recognizes that the impact of specific power on reactivity is relatively small.

Based on these considerations, together with conservatisms built into the approach to burnup credit, the staff finds specifying the specific power value in Table A.6.5.8-7 as a bounding upper value is acceptable in this case. Thus, the staff finds that, with the new note on Table A.6.5.8-7, the package operations are adequate to ensure spent fuel contents in DSCs using burnup credit are acceptable package contents.

Based on a review of the information and representations provided in the application and the staff's confirmatory evaluations, the staff has reasonable assurance that the MP197HB package, as modified in the application, satisfies the criticality safety requirements in 10 CFR 71.

7.0 OPERATING PROCEDURES

7.1 Clarifications to the Operating Procedures

The applicant made some changes to Chapter 7 of the application to clarify, in particular, that the steps for dry loading are for previously loaded DSCs or RWCs that are stored on an independent spent fuel storage installation (ISFSI) under the appropriate license for that ISFSI site. For RWCs, this would be either a 10 CFR 72 site-specific license or a 10 CFR 50 or 52 license. The wet loading procedure, as referred to in this amendment request, refers to loading fuel from a spent fuel pool into a DSC or loading irradiated waste into a RWC using the MP-197HB package.

Chapter A.7 has been revised to remove Table A.7-1 and move the information regarding axial fuel spacers and basket spacers for DSCs and cask cavity axial spacers for both DSCs and RWCs to the appropriate package operations steps where these spacers are used. This includes adding notes about the fuel spacers and basket spacers for the DSCs to Table A.7-3. The descriptions regarding the cask cavity spacers removes specific dimensions and retains the criterion that they be sized so as to maintain an axial gap in the cask cavity of not more than 0.5 inches. To meet this criterion, the cask cavity spacers' dimensions will be based on as-fabricated RWC or DSC canister and MP-197HB inside cavity dimensions.

The applicant explained that recent operating experience with loading RWC-DD in the MP-197HB has shown that, in addition to specified tolerances, the actual overall length may be affected by the flatness of the canister bottom and lid.

Thus, the applicant concluded that cask cavity spacers lengths need to be based on as-built dimensions for the canisters, and the spacers will be custom-fabricated based on as-built dimensions and flatness of the canister bottom or lid. Therefore, specifying spacer dimensions in the operating instructions was not necessary. The staff finds this to be acceptable because fabricating spacers that ensure the specified maximum gap size criterion is met will ensure the configurations of the package contents will be consistent with the package analyses.

7.2 Package Operations for RWCs and RWCs' Contents

The staff reviewed the proposed changes to the package operations to incorporate the modifications to the RWCs and their contents. The applicant also modified the package operations for the RWCs to remove unnecessary details and describe only the essential elements of the operations.

The staff confirmed that the descriptions related to operations such as draining and drying of the RWCs, placement of top cover plates and top shield plugs, and sealing or plugging the vent and drain ports are consistent with the RWCs' descriptions in the drawings. The staff confirmed the descriptions ensure configurations of contents materials will be maintained in configurations consistent with the package analyses, including the shielding analysis.

The staff confirmed that the package operations descriptions are appropriate for and account for the different RWC variants. This includes identifying differences in operations sequences that are necessary to accommodate use of such RWC variations, providing sufficient descriptions related to configuration and use of axial spacers in the package that are adequately consistent with the package analyses, and providing sufficient specifications of the allowable waste contents. The staff finds the modified package operations are consistent with the shielding design description and analyses.

The staff also reviewed the RWCs' allowed contents descriptions in the package operations chapter. The contents descriptions are specified in Tables A.7-2b, A.7-2d, and A.7-2e. The tables distinguish maximum activity and maximum specific activity limits based on the RWCs' shell thickness. They also specify that the specific activity for sources of gamma energies other than Co-60 must not exceed a specific activity that is equivalent to the specific activity limit for the Co-60 source applicable to the RWC in which those contents are loaded. Since Chapter 7 is incorporated by reference into the CoC and includes these activity and specific activity limits, the CoC no longer needs to include those limits directly in a separate CoC condition. The contents limits are all now based on the Unit 01 package with 2.77-inch-thick radial lead shielding. Separate content limits for packages with thicker radial lead shielding have been removed from the CoC and Chapter 7 of the application. Thus, all packages must meet the content limits derived from the Unit 01 package.

The contents description includes a clear description of the types of organic materials that may be part of the RWCs' contents. Based on the evaluation of these materials, as described in Chapters 4 and 5 of this SER, the staff finds these materials to acceptable as part of the contents and that the contents description ensures the allowed contents are consistent with that evaluation. Further, the contents description includes key requirements for determining the specific activity of the contents and ensuring the maximum allowed specific activity is not exceeded.

As described in Table A.7-2b of the application, only contents materials with a shielding effectiveness equivalent to or greater than carbon steel can be credited toward the contents mass used in determining the loaded contents' specific activity. Chapter 5 of this SER describes the staff's considerations for the application of this requirement and its acceptability. For example, organic materials would not be counted toward the mass used in the specific activity calculation.

7.3 Package Radiation Levels, Supplemental Shielding, Occupied Space Requirements

The staff also identified that the applicant initially proposed a change to the package operations related to the final radiation surveys and compliance with the radiation level limits in 49 CFR

173.441 and 10 CFR 71.47. In particular, the change was in relation to the limit in 10 CFR 71.47(b)(4) and 49 CFR 173.441(b)(4) that limits the radiation level in any normally occupied space to not more than 2 mrem/hr. The initially proposed changes described the option to add supplemental shielding to the vehicle as long as it is not attached to the package (without prior NRC approval). They also described an option for the carrier to implement radiation dosimetry requirements to use the exception to the 2 mrem/hr limit in the regulation.

The staff determined that these initially proposed changes are not consistent with the regulations and, without modifications, could lead to acceptance of package conditions that are not consistent with the shielding evaluations demonstrating compliance with the regulatory limits for package radiation levels. For example, the exception that is allowed in 10 CFR 71.47(b)(4) applies only to a private carrier and the dosimetry devices must conform to 10 CFR 20.1502 requirements.

The U.S. Department of Transportation's "Radioactive Material Regulations Review" (December 2008, and available at https://www.phmsa.dot.gov/training/hazmat/ramreview2008), Section XI. A., "Carrier Requirements," states that a private carrier is a company that provides transportation of its own cargo, generally owns the radioactive material to be transported, and for which transport activities are incidental to their regular business activities. Such carriers are always licensed by the NRC or an agreement state to possess and transport the radioactive material.

For the exception in 10 CFR 71.47(b)(4), the carrier must be an NRC licensee given the reference to requirements in 10 CFR 20. For all other carriers, the 2 mrem/hr limit applies. The MP-197HB package operations do not restrict shipments to be by private carrier; thus, use of the exception is not appropriate.

Also, the shielding analysis does include distances for a package containing a loaded RWC at which package radiation levels would not exceed the 2 mrem/hr limit. These contents are design basis contents. Thus, supplemental shielding should only be necessary when the conveyance is not long enough to allow such distances between the end of the package and the normally occupied spaces for a package that is loaded and operated correctly.

Therefore, in the event supplemental shielding were determined to be necessary to not exceed regulatory radiation level limits, the package operations would have needed to include steps to first confirm the package has been loaded and prepared in compliance with the conditions of the package certificate. This would ensure the use of supplemental shielding was not compensating for a problem with the package. The staff also determined that the description of the use of supplemental shielding would have needed to be clear enough so that it addressed the radiation level limit for normally occupied spaces (e.g., the vehicle cab) as the shielding analyses have shown that the package loaded with design basis contents will not exceed the other exclusive use radiation level limits.

Based on the staff's determinations, the applicant modified the proposed changes to the package operations. The applicant's modification restores the operations descriptions regarding the final radiation survey to once again be as they are for the currently approved operations descriptions, with one minor difference. The current operations descriptions refer to the regulatory radiation level limits in both 49 CFR 173 and 10 CFR 71; the modified descriptions only refer to the limits in 10 CFR 71. The limits in 10 CFR 71 and 49 CFR 173 are the same, with some differences in the conditions for exemption from the limit for the normally occupied space for exclusive use shipments. Compliance with 10 CFR 71 limits will ensure compliance with 49 CFR 173 limits.

The staff finds the modified operations descriptions regarding package radiation levels and regulatory compliance to be consistent with the regulations and their implementation and therefore acceptable. The staff also notes that while the applicant did add some statements regarding transport being by exclusive use and private carrier in the shielding chapter, the package operations chapter does not include such requirements. Also, the shielding chapter is not incorporated into the package certificate. Thus, these statements do not impose requirements upon the package user.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

Section 8.1.6.1 of the application, which specifies the gamma shield tests for the package acceptance tests, was revised to distinguish between acceptance criteria for poured and rolled lead shielding material. The applicant provided a new acceptance test and acceptance criteria for precast lead shielding. This new acceptance test and its criteria are for the RWC and DSCs that use this lead in their top and bottom shield plugs, which is precast, rolled and cut to fit the shield plugs' cavities for this lead.

The staff evaluated the acceptability of the test and acceptance criteria in terms of ensuring the specifications on the package drawings were met as well as how the shielding analyses support these acceptance criteria. The staff finds the test and criteria ensure the specifications for the lead in the drawings will be met. The criteria for gaps and minimum thickness are consistent with the specifications in the drawings.

The criterion for flaw size is not in the drawings and so is an additional criterion that is only specified in the acceptance test. This is acceptable as the acceptance test is part of the CoC requirements as long as the shielding analyses support the acceptance criteria.

The staff's review related to the acceptance tests and the shielding analyses is described in Section 5.3 of this SER. Based on that review, the staff finds the acceptance criteria to be acceptable and supported by the shielding analysis.

Therefore, the staff finds the proposed acceptance test, including the acceptance criteria, for the precast lead to be acceptable and sufficient to ensure the package meets the specifications in the package drawings and will fulfill its shielding function as designed and evaluated in the application.

CONDITIONS

The following changes have been made to the certificate:

Item 3.c was edited to remove the comma between the words TN Americas and LLC, as the official legal name of the applicant is TN Americas LLC.

Item 3.d was revised to identify the final revised application as Revision No. 21, dated November 2022.

Condition No. 5(a)(2) clarified that the shielding description was for radial shielding.

Condition No. 5(a)(3) was revised to (i) clarify the radial shielding description, (ii) refer to Chapter A.7 since the information about spacer lengths is included now into the appropriate procedures steps, (iii) indicate that, for one of the RWC variants, the RWC-24PT4, thicknesses are a combination of steel and lead for the top and bottom since the shield plugs' designs for this RWC include lead, with less steel than the specified minimum steel equivalent thickness.

Condition No. 5(a)(5) was updated to include updated drawing revisions for the packaging cask body, packaging general arrangement, lid assembly details, impact limiters, and packaging transport configuration as well as new licensing drawings for the radioactive waste canister variants.

Condition No. 5(c)(2) was revised to clarify that Section A.7.1 of the application covers the maximum contents for this package, for both the RWCs and the DSCs' contents.

Condition No. 10 was revised to delete the "exclusive use" requirement. The package meets the regulations and does not exceed the limits in 71.47. An analysis for the bounding/design basis contents, with a TI greater than 10 while not exceeding the limits in 71.47(b) for exclusive use, does not translate into a regulatory requirement that the package can only be used as exclusive use when a particular shipment is shown by measurement at the time of shipment that it can be done as non-exclusive use.

Condition No. 12 was updated to authorize the use of Revision No. 10 of the CoC until August 31, 2023.

Condition No. 13 was updated to renew the certificate for an additional 5 years.

The References Section has been updated to refer to revision 21 of the application.

CONCLUSION

Based on the statements and representations contained in the application and the conditions listed above, the staff concludes that the design has been adequately described and evaluated, and the Model No. MP-197HB package meets the requirements of 10 CFR 71.

Issued with CoC No. 9302, Revision No. 11.