UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 17, 2019

Mr. W. Scott Edwards
Director of Transportation
TN Americas LLC
7135 Minstrel Way
Columbia, MD 21045

SUBJECT: REVISION NO. 9 OF CERTIFICATE OF COMPLIANCE NO. 9302 FOR THE MODEL NO. NUHOMS® – MP197HB PACKAGE

Dear Mr. Edwards:

As requested by your application dated February 28, 2018, as supplemented June 27, August 15 and December 14, 2018, January 16 and April 16, 2019, enclosed is Certificate of Compliance No. 9302, Revision No. 9, for the Model No. NUHOMS® – MP197HB package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of Title 49 of the *Code of Federal Regulations* 173.471.

If you have any questions regarding this certificate, please contact me or Pierre Saverot of my staff at (301) 415-7505.

Sincerely,

/RA/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9302
EPID - L-2018-LLA-0000

Enclosures:
1. Certificate of Compliance No. 9302, Rev. No. 9
2. Safety Evaluation Report

cc w/encls. 1&2: R. Boyle, DOT
J. Shuler, DOE,
c/o L. F. Gelder

SUBJECT: REVISION NO. 9 OF CERTIFICATE OF COMPLIANCE NO. 9302 FOR THE MODEL NO. NUHOMS® – MP197HB PACKAGE, DOCUMENT DATE: April 17, 2019

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or
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SAFETY EVALUATION REPORT
Docket No. 71-9302
Model No. NUHOMS® MP-197HB Package
Certificate of Compliance No. 9302
Revision No. 9

SUMMARY

By application dated February 28, 2018, as supplemented June 27, August 15 and December 14, 2018, January 16 and April 16, 2019, TN Americas LLC (TN, or the applicant) submitted an amendment request to revise the certificate of compliance (CoC) for the Model No. NUHOMS® MP-197HB package.

The applicant requested to add the Dismantling and Decommissioning Radioactive Waste Container (RWC-DD), as a variant of the previously approved RWC design configuration, for the transport of dry irradiated and/or contaminated solid waste in the MP-197HB package. The amendment request also included modifications of the internal sleeve, fabrication and operational enhancements, and leak testing criteria in compliance with ANSI N. 14.5-2014.

TN Americas LLC also requested, during the course of this review by the NRC staff, several changes which were not directly related either to the original application 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.

For this amendment request, staff reviewed the application using the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material" and associated Interim Staff Guidance.

EVALUATION

1.0 GENERAL INFORMATION

As part of this amendment request, the applicant proposed the addition of the Dismantling and Decommissioning Radioactive Waste Container (RWC-DD), as a variant of previously approved RWC design configurations for use with the NUHOMS®-MP197 transportation package.

The RWC-DD canister is designed to transport reactor-related dry irradiated or contaminated non-fuel bearing solid materials or a combination of both, and to facilitate on-site operation with ancillary equipment. The RWC-DD has a bolted top closure design that allows for the re-use of the container. The inner sleeve is designed with slots to accommodate the rails inside the package, while also providing rails inside the sleeve on which the RWC slides during horizontal loading or unloading of the package. The sleeve has been modified with an option to use stainless steel and aluminum materials for transport of the RWC.

The content specification, as described in the application, is the same as for the previously-approved RWC-W and RWC-B canister designs, i.e., radioactive material contents typically in the form of neutron activated metals, or metal oxides, in solid form. Surface contamination may also be present on the irradiated components. A typical composition of the payload may consist of any individual or combination of the following irradiated hardware: BWR Control Rod Blades, BWR Local Power Range Monitors (LPRMs), BWR Fuel Channels, BWR Poison Curtains, PWR Burnable Poison Rod Assemblies (BPRAs), and PWR and BWR Reactor Vessel and Internals.

Although Co-60 is the most significant contributor to the external dose rate, the staff noted that other radionuclides with significant gammas (such as Fe-59, Co-58, and Mn-54) can also have some contribution to external dose rates and that characterizing activated steel only by Co-60 would neglect these other nuclides. The staff noted that Title 10 of the *Code of Federal Regulations* (10 CFR) 71.33(b) requires than an application identify radioactive constituents and its maximum radioactivity. As discussed in NRC RIS 2013-04 "Content Specification and Shielding Evaluation for Type B Transportation Packages", multiples of A2 do not satisfy this requirement. In response to staff's request, the applicant revised the application and developed the response function at 2 meters from the side of the package in normal conditions for transport for gamma activity limits at 18 energy groups between 0.6-10 MeV. The maximum quantity of radioactive material in an RWC was 8,182 A_2 in the previous application, which is equivalent to 90,000 Ci of Co-60. A reduced lead thickness, between the inner and outer shells, in locations associated with the longitudinal weld seams on the outer shell, was observed from a gamma scan during the manufacturing of the first package and the applicant assessed the impact of this reduced lead thickness condition with respect to the limiting dose rates for spent fuel and RWC contents loaded into the package. The maximum quantity of material for all RWC contents is now limited to a maximum quantity of 70,000 Ci Co-60 or equivalent.

The RWC-DD has the same dimensions as those of the approved RWC-W and RWC-B, except its length. The RWC-DD has a 1.75" shell thickness, a 5.75" bottom plug, a 7.0" top plug similar to the RWC-W and RWC-B, but the cavity length of the RWC-DD is 184.75" (the application mentions a maximum length of 197") compared to 167.3" for the RWC-W and-B. As a result of staff's requests to justify the recategorization of several items and the removal of quality assurance requirements, the applicant revised the drawing NUHRWC-71-1001 to include the grapple ring assembly and its quality category classification as Quality Category C, included details of the vent and drain ports with a quality classification as Quality Category B for the threaded plug and cover plate, and also incorporated weld details required for the fabrication of the RWCs. In addition, details T1 and T 2 on the drawing MP-197HB – 71-1005 and details D and G on drawing MP-197HB-71-1006 were also revised to identify the nominal O-ring groove dimensions for seal contact surfaces along with the corresponding tolerances in accordance with Interim Staff Guidance (ISG)-20.

Table A.1.2 of the application was revised to add a row for the RWC to provide the required sleeves, spacers and a recommendation regarding fins to be used on the cask. Table A.2.13.14-1 was revised to add a note specifying the cold gaps for the RWC designs while Table A.7-1 was revised to correct the spacer height for the RWC-DD to 2.25 inches.

2.0 STRUCTURAL EVALUATION

2.1 Structural Evaluation

The MP-197HB package introduces a new RWC-DD canister, longer and heavier (112,000 lbs) with smaller spacers than the previously approved RWC containers, RWC-W and RWC-B, which weigh less than 110,000 lbs. However, the weight of the RWC-DD, when contained in the MP197HB package (303,600 lbs), is still bounded by previously approved analyses. In addition, the center of gravity of the MP-197HB, when loaded with the RWC-DD, is at 102 $\pm$ 4 inches, which is also bounded by previous analyses. Thus, staff agrees that the MP-197HB package containing a RWC-DD canister will continue to function as previously approved as the demands on the package are still bounded by previously approved analyses.

Staff did not examine the RWC-DD explicitly for normal conditions of transport (NCT) or hypothetical accident conditions (HAC) since the applicant states that the RWCs, in general, are not part of the containment boundary. Although the RWC canisters do not provide a containment function, they do provide a shielding function, which is ensured even if the lid of an RWC detaches from the rest of canister as in the case of a complete lid bolt failure. The lid geometry, spacers, and dunnage ensure that only a 0.5" space between the lid of the RWC and the overpack is present which also limits any significant streaming path.

In addition to the RWC-DD canister, multiple changes to the licensing drawings were also noted. Most of these changes removed minor details and do not impact the structural performance of the package while allowing for more flexibility in the fabrication of the package by increasing some of the tolerances on components such as the impact limiters. The applicant did make several small changes to the drawings, which required that some calculations be revisited. The applicant expanded existing analyses of the RWCs and added an expansion gap detail for the Vyal B neutron absorber material along with supporting calculations. The applicant updated the double shoulder trunnion bolt calculations as a result of correcting a small typographical error on the drawings. The safety margins reported by the licensee continued to be positive after such updates. The dimensions to the transportation skid, which is not part of the package, have been changed to reference values.

Based on the review of the statements and representations in the application, the staff concludes that the MP-197HB package meets the structural requirements of 10 CFR Part 71.

2.2 Materials Evaluation

The staff reviewed the changes in the new revisions of the RWC design-basis drawings. The RWC-DD canister's nominal diameter is the same as the approved RWC configurations but longer in length. The internal structural components of the RWC-DD are fabricated from the same alloys of stainless steel or carbon steel as those of the previously-approved RWC-W and RWC-B designs. An inner sleeve is used to accommodate the RWC within the MP197HB transport cask. The inner sleeve design for all RWCs, per drawing MP197HB-71-1014, Revision 2, retains the previously-approved design but was revised to include the option for fabrication with both stainless steel and aluminum (instead of only aluminum, as previously approved). The material property values for RWC subcomponents, as listed in Section A.2 of the application remained unchanged from prior approvals. Therefore, the staff considers these values as acceptable. The staff has addressed concerns for chemical, galvanic, or other reactions later in this SER.

The applicant revised drawing NUHRWC-71-1001 as Revision 5, to show a general arrangement for the new RWC-DD, and the previously-approved RWC-W and RWC-B. Further, the previously-approved NUHRWC-71-1002, Revision 1, and NUHRWC-71-1003, Revision 0, drawings were both removed, because the applicant stated those drawings are now superseded by NUHRWC-71-1001, Revision 5.

Both NUHRWC-71-1001, Revision 1, and NUHRWC-71-1003, Revision 0, included multiple subcomponents previously identified as Quality Category A. In NUHRWC-71-1001, Revision 5, these subcomponents were reclassified to Quality Category B. The applicant justified the reclassification by clarifying that these RWC subcomponents do not serve a primary containment function and are only considered for maintaining an analyzed configuration of the RWC contents per the shielding evaluation described in Section A.5.3.1.3 of the application.

The applicant provided a structural evaluation (Section A.2.13.7.4.6 of application) to demonstrate that these RWC subcomponents (reclassified as Quality Category B) and their welds satisfy the stress limits required by ASME B&PV Code, Section III, Subsection NF. Per this evaluation, the applicant concluded that the structural performance of the RWC assumed for the shielding evaluation is valid. The applicant therefore concluded that the RWC is not subject to reconfiguration and the classification of the RWC subcomponents as Quality Category B is appropriate. The staff has reviewed the adequacy of this structural evaluation elsewhere in this SER and, per that review, considers that the reclassification of RWC subcomponents is adequately justified and consistent with the classification in NUREG/CR-6407.

The applicant also revised the allowable closure details for the various RWC designs in drawing NUHRWC-71-1001, Revision 5. The staff confirmed that the drawing identifies the weld requirements for the closure, which provide reasonable assurance that the contents debris inside the RWC will remain as analyzed in the shielding evaluation of the package. The staff reviewed the revisions and considers that the revisions are acceptable and consistent with the classification in NUREG/CR-6407.

Section A.1.4.9A of the application states that all RWC welding procedures, welders, and welding are performed in accordance with the requirements of AWS D1.1 and AWS D1.6. The staff verified that Drawing NUHRWC-71-1001, Revision 5, identifies that all visual (VT) and liquid penetrant (PT) examinations are performed in accordance with these structural welding codes.

Drawing NUHRWC-71-1001, Revision 5, states that alternate welds of equivalent strength may be used for RWC fabrication with the approval of the CoC holder. The weld sizes defined in the drawing are minimum. The staff considers that substituting alternative welds, of equivalent strength, is reasonable for demonstrating structural performance of the RWC per the shielding evaluation described in Section A.5.3.1.3 of the application, upon considering that the RWC does not serve a primary containment function during transport.

The applicant revised the application and the Bill of Materials in Drawing MP197HB-71-1002, Revision 7, to add flexibility for use of alloy-steel SA-540 Grade B24 Class 1 in addition to the previously-approved Grade B23 Class 1, for the allowed materials for lid bolts, impact limiter attachment bolts, ram access closure plate bolts, and trunnion bolts for the package. Tables A2.13.2-3, A.2.13.2-3, and A.2.13.5-1 of the application define equivalent mechanical properties for both grades of alloy steel. The staff verified that the yield strength and tensile strength for both grades of alloy steel are equivalent per the ASME BPVC Section II Part D. Section A.2.13.2 of the application also states that bolt materials fabricated with Grade B24 Class 1 will have the same minimum tensile strength as the previously-approved Grade B23, Class 1. The staff verified this requirement is defined in Drawing MP197HB-71- 1002, Revision 7.

The applicant made additional revisions to the design-basis drawings for the package in response to staff's questions. The staff followed the guidance per Interim Staff Guidance (ISG)-20 to determine if these changes removed either safety-related information or other features that may be important for other reasons (e.g., ease of handling radioactive material within a facility, product protection, or cosmetic reasons). Per ISG-20, the staff confirmed that the drawings identified containment-related seal surfaces and O-ring groove details, including surface finish, groove dimensions within strict tolerances, O-ring size, type, and material. The applicant confirmed that the groove inner and outer diameters for the metallic seals are only relevant for fabrication and assembly; as they are sized to fit the seal but their size has no bearing on the physical characteristics of the seal such as compression. The applicant further clarified that the metallic seals are not relevant to the safety analysis of the package because, as long as the groove has the depth recommended by the manufacturer for the specified seal to ensure the seal is properly crushed, the seal will perform its function. Therefore, these dimensions are left as reference dimensions on detail T1 of drawing MP197HB-71-1005. The staff considers the justification to be acceptable.

The materials used in the fabrication of the RWC-DD have been previously evaluated for their radiation resistance in the NUHOMS®-MP197 package. The package operations require that all elastomeric seals are replaced prior to 12 months of use. The fusible plug, which is meant to melt during a fire event to prevent pressure build-up within the impact limiter shell, was revised from steel to a nylon-type material. This material does not serve a structural function (Quality Category C, no QA recordkeeping requirements) and therefore the staff does not consider that its radiation resistance could reasonably impact its intended function. Further, the staff expects that routine determinations, as required per 10 CFR 71.87(b), will provide defense-in-depth assurance that the fusible plug remains as intended. The staff finds that the package complies with the requirement in 10 CFR 71.43(d).

The applicant also revised the allowable moisture content for the wood used in the impact limiter. The compressive strength values for Redwood and Balsa wood impact limiter materials are applicable to the same specified values for the density, but with moisture content revised to range from 6% to 12%, and the temperature range of −20°F to 165°F (−29°C to 74°C). The applicant stated that the range of both Redwood and Balsa wood compressive strengths were specified over a temperature range of −20 to 165°F (−29 to 74°C) and moisture contents ranging from 6% to 12%. The applicant stated that the range of moisture contents was typical for kiln dried wood. The applicant provided references to support the range of compressive strengths of 5000 to 6500 psi (34.47 to 44.82 MPa) for Redwood and 1560 to 2010 psi (10.75 to 13.86 MPa) for Balsa wood.

The staff reviewed the information on moisture content and compressive strength of wood products. A moisture content of 12 percent is typical of most wood for most of the U.S. In dryer climates, the moisture content can be less (typically 6 to 9 percent), and in the Pacific Northwest the moisture content can be up to 16 percent. The staff concluded that the range of moisture content cited by the applicant is typical of most of the U.S. The staff reviewed the available information on the effect of moisture content and temperature on mechanical properties of wood. Compressive strength parallel-to-the-grain increases 35% when the moisture content is decreased from 12% to 6%. For Redwood, testing conducted at Sandia National Laboratories showed that compressive strength was a function of moisture content and grain orientation. Values reported by Cramer et al. (1996) generally agree with the range of compressive strengths cited by the applicant for Redwood over the range of moisture contents from 6 to 12%. The staff determined that the applicants specified values of compressive strength for Redwood ranging from 6500 psi at −20°F to 5000 psi at 165°F are appropriate.

For Balsa wood, the available information showed that Balsa wood with a density of 10 lb/ft^3 had a mean compressive strength generally above 1500 psi for moisture contents ranging from 0 to 10 percent. At a higher moisture content of 15%, the mean compressive strength decreased to approximately 1300 psi. For Balsa wood with a moisture content of 0%, the mean compressive strength ranges from 2000 psi at −20 °F to 1500 psi at 165 °F. The staff determined that the values of compressive strength for Balsa wood cited by the applicant are appropriate and because they are in agreement with reported values over the range of moisture content values and temperatures.

The applicant noted that effects of wood properties and initial cask drop angle were considered in numerous computer analysis runs to determine the forces and decelerations used in the cask body structural analysis. Balsa wood and Redwood densities have been selected to limit the maximum cask body inertia loads due to the one-foot NCT drop and the thirty-foot HAC accident drop. The staff determined that the compressive strength values for Redwood and Balsa wood materials are applicable to the same specified values for the density when the wood moisture contents are revised to range from 6% to 12%. Because the proposed changes resulted in no impact limiter design changes, including the Redwood and Balsa wood compressive strengths, the staff concludes that the forces and decelerations used previously for package approval continue to apply. This demonstrates that the revised wood moisture contents will not reduce the effectiveness of the package in meeting 10 CFR Part 71 requirements.

The staff concludes that the package with the proposed changes will perform its intended functions and maintain structural integrity to meet the requirements of 10 CFR Part 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 Part 71.

3.0 THERMAL EVALUATION

The minimum required thermal conductivities for stainless steel and anodized aluminum, as listed in Section A.3.2.1, remain unchanged from prior approvals. The staff confirmed that references for the technical specifications of package components were identified, as they pertain to the revisions in this amendment. Consistent with the guidance in NUREG-1609, the staff confirmed that the minimum allowable service temperature of all components is less than or equal to - 40°C (- 40°F).

The staff finds that the material properties and component specifications provided in the application are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71. Because the RWC heat load is less than 5 kW, no RWC configurations, i.e., RWC-W, RWC-B and RWC-DD, has any impact on the thermal evaluation of the package which is approved for 26 kW for the spent fuel contents within the DSCs for the MP-197HB package without external cooling fins.

4.0 CONTAINMENT EVALUATION

The containment boundary for the NUHOMS-MP197HB package consists of a cylindrical inner shell, a bottom plate with a ram access closure plate with seal and bolts, a cask body flange, a top lid with seal and bolts, vent and drain ports with closure bolts and seals, and all containment welds. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity, when needed. Radioactive waste canisters (RWCs) are not a containment feature and provide no secondary containment boundary.

The NUHOMS-MP197HB used for shipping RWC contents does not require leak tight criteria for containment. A reference air leak rate less than leak tight criteria is allowed for shipment of the

RWC contents. Leak rate criteria for RWC contents are established in Appendix A.4.6. The allowable leak rates are calculated in accordance with ANSI N14.5-2014 “American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment”.

The applicant clarified a number of issues regarding package containment:

> The ram access cover plate comes in two options, per note 3 of drawing MP197HB-71-1002 and details T1 and T2 of drawing MP197HB-71-1005): (i) a metal seal required to transport spent fuel DSCs, and (ii) an elastomer seal option (with an inner containment seal and an outer seal to perform the leak-tightness test of the inner seal, used to transport RWCs.
>
> Because the groove required for a metallic seal is different from the grooves required for two O-ring elastomer seals, only one of the two options may be used on any given fabricated cover plate; therefore, two cover plates are fabricated: the ram access cover plate with metallic seals is used only with spent fuel DSCs and the ram access cover plate with elastomer seals is used for shipping RWCs. The heat load of RWCs is very low, and therefore there is no need for a metallic seal to address high temperatures like there is for high burnup spent fuel DSCs.

The inner elastomer O-ring is always the containment seal, with the outer seal being used for leak-tightness testing. Note 7 of drawing MP197HB-71-1001 lists the components that belong to the containment boundary. For the lid, the inner O-ring (item 24) is the containment boundary seal (the outer seal is used for testing).

The staff has reviewed the description and evaluation of the containment system and concludes that: (1) the application identifies established codes and standards for the containment system; (2) the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package; (3) the package is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction.

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

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

In summary, the staff has reviewed the Containment Evaluation section of the application and concludes that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71, and that the package meets the containment criteria of ANSI N14.5-2014.

5.0 SHIELDING EVALUATION

The staff has reviewed the proposed changes pertaining to this amendment request, including (1) the addition of the RWC-DD with the effect of the reduced lead thickness on the RWCs, and

(2) the effect of the reduction in lead thickness on the spent nuclear fuel (SNF) contents. During the fabrication of the MP197HB Unit 01, the applicant found that the lead thickness did not meet the minimum 2.85" acceptance criteria in localized areas and, instead, conformed to a minimum thickness of 2.77". The RWC spacer height does not affect the shielding performance of the package, as the material is not credited within the shielding evaluation, and therefore is not discussed in this SER.

In reviewing the changes to this application, the staff used the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," March 1999. This evaluation documents the staff's review that verifies that the shielding design of the transportation package MP197HB, including the above changes, provides adequate protection against direct radiation from its contents and that the package design meets the dose rate limits set forth in 10 CFR Part 71.47 and 71.51 under NCT and HAC.

5.1 Addition of RWC-DD and Reduced Lead Thickness for MP197HB Unit 01 on RWC

The applicant provided the detailed structural layout and the dimensions of the RWCs in the drawings contained in Section A.1.4.10.11 of Appendix A.1.4.10 of the application. The RWC-DD has a 1.75" shell thickness, a 5.75" bottom plug, a 7.0" top plug. The staff verified that these are documented as minimum thicknesses within the drawings. The dimensions are similar to RWC-W and RWC-B, but the cavity length of the RWC-DD is 184.75" compared to 167.3" for the RWC-W/B. The source can be closer to the bottom or top of the MP197HB transfer cask cavity since the RWC-DD is 17.45" longer than RWC-W/B.

The allowable contents for the RWC canisters are described in Table A.7-2b of the application and includes various types of dry irradiated and contaminated non-fuel bearing reactor hardware limited to 56.0 tons in an air or inert gas environment. The payload will vary from shipment to shipment. A typical composition of the payload consists of the following components either individually or in combinations: BWR Control Rod Blades, BWR Local Power Range Monitors (LPRMs), BWR Fuel Channels, BWR Poison Curtains, PWR Burnable Poison Rod Assemblies (BPRAs), and PWR and BWR Reactor Vessel and Internals. The radioactive material is typically in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. Although the Co-60 is the most significant contributor to the external dose rate, there are also contributions from other nuclides. To address other non-Co-60 nuclides that have different gamma energies, the applicant determined equivalency by specifying activity limits as a function of gamma energy such that higher gamma energies have a lower activity limit and lower gamma activities have a higher activity limit.

The tables of allowable activity include 18 energy group between 0.6-10 MeV and are specified in Table A.7-2c of the application for a 90,000 Ci Co-60 payload, and Table A.7-2d of the application for a 70,000 Ci Co-60 payload (reduced lead configuration for MP197HB Unit 01). The tables include a sum of fractions procedure to determine the allowable contents for a source with gammas from multiple energy bins. These tables are applicable for all RWC variants.

The staff found this to be an acceptable way of defining allowable contents and meets the requirements in 10 CFR 71.33(b) based on its discussion in NRC RIS 2013-04 "Content Specification and Shielding Evaluation for Type B Transportation Packages," April 2013. Table A.7-2b of the application retains the limit of a maximum of 8,182 A2. This is for containment limits as well as for providing a limit for non-gamma emitting nuclides (such as alpha and beta).

Neutron radiation was not evaluated for the RWC and based on the allowable contents is only expected to be present in trace amounts.

The applicant discussed the modeling of the RWC in Section A.5.2.1.5 of the application. Radioactive waste occupies only a portion of the inner volume of the canister. The applicant modeled the waste as carbon steel with a reduced density of 1.0 g/cm^3 distributed within a cylindrical volume with a 66.0" diameter and 168" height and the rest of the inner volume of the canister is modeled as air. These waste dimensions would fill the entire RWC-W and RWC-B cavities and is equivalent to 9.4 metric tons. Since the waste material acts as self-shielding, the CoC contains a condition that limits the specific activity of discrete components to a maximum of 7.45 Ci Co-60 or equivalent per kilogram where equivalent activity limits as a function of gamma energy for isotopes other than Co-60 are shown in Table A.7-2d for the 70,000 Ci limit. The basis for this value is the 70,000 Ci Co-60 or equivalent divided by the 9,400 kg of self-shielding material. If a user were to ship less than the design basis activity limit, then the required amount of self-shielding material would be reduced accordingly. The staff found that this ensures that the amount of self-shielding material required within each shipment is consistent with the analysis within the application that demonstrates that the package meets regulatory dose rate limits.

The applicant models the source as fully distributed within the self-shielding medium. This assumption is not necessarily conservative. It is possible for contents of this type to have non-uniform activation profiles or hot spots and to be non-uniformly distributed with areas of higher activity. It is also possible for contents to shift and settle upon transport exposing higher activity areas of the payload and change the dose rate profile measured during pre-shipment measurement. Therefore, the CoC contains a condition that states: *"The volume of discrete components shall be divided into ten or more nearly equal volumes no greater than 0.1 m^3. The specific activity of each volume must be assessed (through measurements, calculations, or process knowledge) and the specific activity of individual volumes shall not exceed 7.45 Ci Co-60 or equivalent per kg."*

As perfectly uniformly distributed contents are unrealistic, this would allow contents with a realistic level of non-uniformity to be shipped that are still represented by the design basis analysis for calculating dose rates. Dividing discrete components into 10 nearly equal volumes and requiring that these volumes do not exceed the 7.45 Ci/kg limit, ensures that the specific activity limit is applied for reasonably small areas of components so that components with relatively higher areas of specific activity could be shipped as long as the specific activity is below the amount demonstrated to meet dose rate limits.

The applicant models the source material as carbon steel. The allowable components described consist of materials other than carbon steel that may have better or worse shielding properties. The applicant did not demonstrate that carbon steel was the most conservative self-shielding material; however, the staff still found it acceptable based on its experience that the self-shielding material (for a given density) for the payload described doesn't result in a significant difference in dose rates, and that the radiation source would likely be from the activated steel components. Other materials like zirconium or aluminum, even at an equivalent density, may provide less shielding than carbon steel. Based on the example contents described in Table A.7-2b of the application, the staff finds that there would be a mix of materials, likely dominated by steel and that potentially reduced shielding from other materials would not significantly affect the dose rates and found that modeling the source's self-shielding as carbon steel is reasonable.

The applicant modified the MCNP model for 68BTH Dry Shielding Canister (DSC) to calculate the NCT and HAC dose rates for the RWC. The staff found the modeling of the MP197HB outer packaging acceptable under NCT and HAC in its review of Revision 8 of the CoC (Letter from J.McKirgan (NRC) to J. Bondre (TN), "Revision No. 8 of Certificate of Compliance No. 9302 for the Model No. NUHOMS-MP197 Package," May 23, 2017). The applicant replaced the 68BTHDSC with the carbon steel RWC. The applicant modeled the RWC as a cylinder with a 70.50" diameter and 189.19" height. The applicant assumed that the thickness of the cylindrical shell on side of the canister is 1.75" and the thickness of the shield plugs on the bottom and top of the canister are 5.75" and 7.00" respectively. The staff found this acceptable as these are consistent with the minimum dimensions within the RWC drawings. The applicant modeled the cylinder as centered on the cask axis and located it 2.71" from the cask bottom plug.

The applicant models the lead shielding within the package at a nominal thickness of 3". This is non-conservative as the acceptance criteria for this package is 2.85". For the reduced lead configuration (MP197HB Unit 01), the applicant reduced the lead thickness to 2.77", which is equivalent to the acceptance criteria. Since modeling the lead shield at 3" is non-conservative with respect to the 2.85" acceptance criteria, there is a CoC condition for all RWCs that the contents be limited to 70,000 Ci Co-60 or equivalent supported by the 2.77" lead thickness analysis per the requirements of Table A.7-2d of the application.

The limiting dose rate, as calculated by the applicant for the RWC, is at 2 meters from the package side for 70,000 Ci Co-60 for the reduced lead configuration (MP197HB Unit 01) and is 9.41 mrem/hr. As shown in Tables 8-1 through 8-6 of the calculation file "Shielding Performance of MP197HB Transportation Cask Containing a Canister with Irradiated/Contaminated Waste," CALC-MP197HB-0504, as well as in Table A.5-34 of the application, the dose rate at 2 meters is the limiting dose rate as compared to the surface dose rate or the HAC dose rate. This meets the limit in 10 CFR 71.47(b)(3) that requires that, for an exclusive use package, the dose rate does not exceed 10 mrem/hr at any point 2 meters from the lateral surface of the vehicle.

The maximum dose rate at the surface under NCT is 72.2 mrem/hr at the bottom of the package, from Table A.5-34 of the application. This meets the requirement in 10 CFR 71.47(b)(1) that requires that, for an exclusive use package under NCT, the dose rate on the surface not exceed 200 merm/hr (unless certain conditions are met then the dose rate limit is higher).

The applicant evaluated the maximum dose rate under HAC, which is 103 mrem/hr at 1 meter from the package surface as stated in Table A.5-34 of the application. This meets the requirement in 10 CFR 71.51(a)(2) which requires that under HAC the external radiation dose not exceed 1000 mrem/hr at 1 meter from the package surface.

Although the applicant calculated the surface dose rate and the HAC 1-meter dose rates using the nominal lead shield thickness and the higher Co-60 Ci content, these calculations show that there is significant margin in these dose rates as compared to the 2-meter dose rate. Based on the margin to the dose rate limit, the staff found that these evaluations provide reasonable assurance that the reduced lead shield configuration would also meet surface and HAC dose rate limits.

For gammas other than those from Co-60, the applicant calculated the activity (gammas/sec) for each energy group that results in a 2 meter dose rate of 8.71 mrem/hr at 2 meters for the reduced lead configuration (MP197HB Unit 01). The staff found this acceptable as this dose rates meets the regulatory limit in 10 CFR 71.47(b)(3). The staff found 2 meters as the limiting dose rate location from the Co-60 results discussed above applicable for these other energies because the source geometry is the same.

The staff found that all maximum calculated dose rates meet the regulations in 10 CFR Part 71.

5.2 Reduced Lead Thickness for MP197HB Unit 01 on SNF Contents

The applicant submitted a justification that the reduced lead (MP197HB Unit 01) configuration when containing the allowable SNF contents would not exceed regulatory dose rate limits in 10 CFR Part 71 in Section A.5.5.5.1 of the application. The applicant evaluated the dose rate by reducing the lead thickness 3mm from the nominal thickness of 3" for a 30" wide section for the limiting NCT and HAC SNF configuration, which is the 69BTH DSC. This equals to a lead thickness of 2.88". The results, as displayed in Table A.5-63 of the application, show for this configuration that the dose rates remain below the regulatory limits in 10 CFR Part 71.

The limiting SNF configuration is for higher burnup fuel where the dose rate is dominated by the contributions from neutron emissions. The reduction in lead thickness would be more significant for sources that are dominated by gamma emissions as the lead acts as a gamma shield. The source terms where gammas dominate are lower burnup assemblies. This is illustrated by the applicant in Figures A.5-21 and A.5-22 of the application. This is also illustrated in NUREG/CR-6700, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," January 2001. Although the applicant's figures show a much greater neutron contribution to dose rate, the contribution to dose rate from neutrons versus gammas is highly dependent on the shielding and the staff finds this to be expected. The NUREG only serves to further illustrate the principle that, with higher fuel assembly burnups, the dose rate becomes more and more dominated by neutron contributions.

Based on the dose rate results in Table A.5-30 of the application, lower burnup assemblies (burnup at 45 GWd/MTU and below) have a higher margin to regulatory dose rate limits. From Figure A.5-22 of the application, the gamma contribution to the dose rate is about 50%. The staff performed a MICROSHIELD calculation using a spent fuel gamma source spectrum and found that a reduction in lead shielding by 4 mm would result in about a 25% increase in dose rates. The margin to the regulatory dose rate limits in Table A.5-30 for fuel assemblies with gamma dominated sources (fuel assemblies burned to 45 GWd/MTU and below) are large enough that if the contribution to the dose rate from gammas increases by 25%, the package will still meet regulatory limits. Therefore, the staff found that a 4 mm reduction in lead to about 2.85" would not affect the package's ability to meet regulatory dose rates.

The applicant did not provide calculations demonstrating that SNF meets regulatory dose rates for the reduced lead acceptance criteria of 2.77." However, the staff found this reduced lead thickness acceptable for an acceptance criteria for the MP197HB Unit 01. The staff based its finding on the gamma scan for the particular package in which this applies in Figure 2-1 in Enclosure 3 from the letter from W.S. Edwards (TN) to the NRC, "Application for Revision 9 to Certificate of Compliance No. 9302, Response to Second Request for Additional Information, Docket No. 71-9302 and EPID-L-2018-LLA-0000," December 14, 2018. This figure shows that the reduction in lead is localized. When determining the 2-meter dose rate, localized effects will be averaged out. There is a significant margin to the limits for the surface dose rates shown in Tables A.5-1 and A.5-1a of the application that the reduction in lead is not significant enough to cause them to exceed regulatory limits. It is the staff's judgment that these localized effects for this particular package will not cause the dose rates to exceed regulatory limits in 10 CFR 71.47 when loading the allowable SNF contents.

5.3 Conclusions and Conditions

Based on the analysis submitted by the applicant and the findings and conditions within this SER, the staff found the addition of the Dismantling and Decommissioning Radioactive Waste Container (RWC-DD), and the addition of the acceptance criteria for localized areas of the lead shield of the MP197HB Unit 01 package conforming to a minimum thickness of 2.77" does not affect the MP197HB package's ability to meet regulatory dose rates in 10 CFR Part 71. The CoC will contain the following condition 5(c)(2) that applies to all the MP197HB RWCs:

Maximum quantity of material per package: as specified in Chapter A.7, Section A.7.1 of the Application with the exception that all RWC contents are limited to a maximum quantity of 70,000 Ci Co-60 or equivalent. Specific activity of discrete components is limited to a maximum of 7.45 Ci Co-60 or equivalent per kilogram. Equivalent activity limits as a function of gamma energy for isotopes other than Co-60 are shown in Table A.7-2d for the 70,000 Ci limit. The volume of discrete components shall be divided into ten or more nearly equal volumes no greater than 0.1 m^3. The specific activity of each volume must be assessed (through measurements, calculations, or process knowledge) and the specific activity of individual volumes shall not exceed 7.45 Ci Co-60 or equivalent per kg.

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

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

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

6.0 CRITICALITY EVALUATION

There is no fissile material payload content in the RWC; thus, no criticality evaluation is required.

7.0 OPERATING PROCEDURES

The RWC-DD is intended to be a reusable transport container and, as such, not intended for extended storage. The RWC may be unloaded with the MP-197HB being in a vertical position.

The applicant revised loading operations to clarify that the transport cask cavity is backfilled with an inert gas for transporting only dry shielded canisters. The loading procedures for all RWCs with dry irradiated and/or contaminated non-fuel bearing solid materials, including the RWC-DD, were revised to allow for cask backfilling with air, nitrogen or inert gas environment following drying operations. The staff recognizes that the inert environment is not necessary for the transport of RWCs, as these are not credited for containment. The drying procedures for wet-loaded RWCs are reasonable for ensuring that chemical or galvanic interaction corrosion will be negligible or limited for short transport periods. Further, the staff expects that routine determinations, as required per 10 CFR 71.87(b), will provide defense-in-depth assurance that corrosion remains insignificant. Therefore, the staff agrees with the assessment that hydrogen gas generation is not credible for the transport of RWC-DDs, as was found for the previously-approved RWC-W and RWC-B designs.

Changes related to operating procedures include: (i) specific operational steps, (ii) the use of a transfer skid or of a specialized handling frame for down-ending operations, (iii) a new procedure for vertical unloading operations for the RWC, noting that horizontal unloading had been previously covered in the previous amendment request, and the transfer into a staging area for unloading of the RWC, (iv) the sampling of the cask cavity only for a DSC since the RWC does not contain fission product gases that could be released to the cask cavity, (v) specific procedures for the loading, drying and sealing operations of the RWC-DD, (vi) the use of a ram access plate design that uses elastomeric seals for the RWC shipments instead of the metallic seals required for spent fuel shipments in DSCs, (vi) the notification that elastomeric seals are inspected for each shipment and replaced within 12 months of a shipment. Changes made to the sleeve design have no impact on operation.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

The staff confirmed that the applicant did not revise the acceptance tests performed on the bolts, as discussed in Section A.8.1.2 of the application. This discussion states that the structural materials are (1) to be chemically and physically tested to confirm that the required properties are met, and (2) base materials and welds are examined in accordance with the ASME BPVC requirements. Therefore, since both SA-540 Grade B23 Class 1 and Grade B24 Class 1 have equivalent mechanical properties, and the same acceptance test requirements remain in place for both alloy steel grades, the staff considers the revision to be acceptable.

The staff has reviewed the change to the acceptance criteria for the gamma shielding within Chapter 8, Section A.8.1.6.1. This section states that the calibration test block lead thickness shall be 2.85”, with the exception of the MP197HB Unit 01, which shall be 2.77”. The staff found the acceptance criteria for 2.85” and 2.77” lead thickness to be acceptable as they are consistent with the reduction of lead accounted for within the shielding evaluation for the reduced lead configuration.

For the 2.77” acceptance criteria, the staff further found this acceptable for the MP197HB Unit 01 based on the gamma scan for this particular cask, which shows that the reduction in lead is only applicable for localized areas.

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 corresponding application as Revision No. 19, dated April 2019.

Condition No. 5(a)(3) was revised to include materials other than aluminum pertaining to the internal sleeve, and specify that an inert atmosphere is maintained in the package cavity for contents loaded in a Dry Shielded Canister (DSC).

Condition No. 5(a)(4) was updated to include new drawings for the packaging cask body, packaging general arrangement, lid assembly details, impact limiters, and packaging transport configuration as well as the new radioactive waste canister (RWC-DD). The drawing MP197HB-71-1007 pertaining to the packaging regulatory plate was removed from the licensing drawings.

Condition No. 5.(c)(2) was revised as follows: Maximum quantity of material per package: as specified in Chapter A.7, Section A.7.1 of the Application with the exception that all RWC contents are limited to a maximum quantity of 70,000 Ci Co-60 or equivalent. Specific activity of discrete components is limited to a maximum of 7.45 Ci Co-60 or equivalent per kilogram. Equivalent activity limits as a function of gamma energy for isotopes other than Co-60 are shown in Table A.7-2d for the 70,000 Ci limit. The volume of discrete components shall be divided into ten or more nearly equal volumes no greater than 0.1 m^3. The specific activity of each volume must be assessed (through measurements, calculations, or process knowledge) and the specific activity of individual volumes shall not exceed 7.45 Ci Co-60 or equivalent per kg.

Condition No. 9 was edited to specify that it was related only to the transportation of DSCs.

Condition No. 12 was updated to authorize use of Revision No. 8 of the certificate of compliance until April 30, 2020.

The expiration date of the certificate was not changed. The References Section has been updated to refer to Revision 19 of the application dated April 2019.

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 Part 71.

Issued with Certificate of Compliance No. 9302, Revision No. 9
on April 18, 2019.