May 23, 2017

Dr. Jayant Bondre
Chief Technical Officer
TN Americas, LLC.
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: REVISION NO. 8 OF CERTIFICATE OF COMPLIANCE NO. 9302 FOR THE MODEL NO. NUHOMS® – MP197 PACKAGE

Dear Dr. Bondre:

As requested by your application dated August 16, 2016, as supplemented February 1, February 28 and May 4, 2017, enclosed is Certificate of Compliance No. 9302, Revision No. 8, for the Model No. NUHOMS® – MP197 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 49 CFR 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
CAC No. L25139

Enclosures:
1. Certificate of Compliance No. 9302, Rev. No. 8
2. Safety Evaluation Report

cc w/encls. 1&2: R. Boyle, Department of Transportation
L. Gelder, DOE-SRNL

REVISION NO. 8 OF CERTIFICATE OF COMPLIANCE NO. 9302 FOR THE MODEL NO. NUHOMS® – MP197 PACKAGE, DOCUMENT DATE: MAY 23, 2017

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SAFETY EVALUATION REPORT
Docket No. 71-9302
Model No. NUHOMS® –MP917 Package
Certificate of Compliance No. 9302
Revision No. 8

SUMMARY

By application dated August 16, 2016, supplemented February 1 and 28, 2017, TN Americas LLC (TN, or the applicant) submitted an amendment request to revise the certificate of compliance (CoC) for the Model No. NUHOMS® -MP197 package. The applicant requested an increase of the maximum allowable assembly average fuel burnup from 62 GWd/MTU to 70 GWd/MTU for the NUHOMS® 69 BTH Dry Shielded Canister (DSC).

By letter dated May 4, 2017, the applicant also requested timely renewal of the certificate. The certificate was renewed for a 5 year period.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). The analyses performed by the applicant demonstrate that the package provides adequate structural, thermal, containment, shielding protection, and criticality control under NCT and HAC conditions.

For this amendment request, NRC staff reviewed Chapters 2, 3, 5 and 7 of the application using the guidance in "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617 and associated Interim Staff Guidance (ISG). Based on the statements and representations in the application, and the conditions listed in the certificate of compliance, the staff concludes that the package meets the requirements of 10 CFR Part 71.

2.0 STRUCTURAL EVALUATION

The objective of the structural review is to verify that the structural performance of the package continues to meet the requirements of 10 CFR Part 71.

2.1 Description of Structural Design

The structural design of the 69BTH DSC for a 70 GWd/MTU fuel is identical to that of the previously approved revision. The applicant has proposed the addition of an optional permanent backing ring under the root of the inner top cover plate (ITCP) to the DSC shell weld. Because there is no change to the weld size or geometry, the staff concludes that there is no impact on the strength of the weld and, therefore, no impact on the previously approved DSC. As a result, the staff determines that only the increase in fuel burnup for the 69BTH DSC is within the scope of this evaluation.

2.1.1 Discussion

In Appendix A.2.13.11 of the application, the applicant stated that, due to uncertainties in the fuel cladding material properties after storage, one cannot assume intact fuel for NCT and HAC in the thermal, shielding and criticality evaluations. As such, the structural integrity of the fuel is not credited in the analyses. The applicant does however evaluate the fuel cladding for NCT, specifically side and end drops, in Appendix A.2.13.11 of the application, as a means to provide reasonable assurance that the fuel cladding will remain intact during NCT.

The proposed increase in fuel burnup will affect the temperature of the 69 BTH DSC and NUHOMS®-MP197, as well as the pressure within the individual boiling water reactor (BWR) fuel rods. Because the increase in the heat load is bounded by the previously approved value, the staff concludes that the increase in temperature, as a result of the increase in fuel burnup, is also bounded by the previously approved revision. Therefore the increase in the heat load has minimal impact on the integrity of the fuel rod.

In Appendix A.2.13.11 of the application, the applicant revised the side and end drop structural analyses of the BWR fuel assemblies to account for the higher pressure resulting from the increase of the maximum authorized assembly average burnup from 62 to 70 GWd/MTU for the 69BTH DSC. Based on fission gas inventory calculations, the applicant determined that the fuel rod internal pressure would increase by a factor of 1.14 as a result of the increase in average burnup. The applicant then increased this pressure by over three times to envelope the internal pressure of the BWR fuel rod.

The applicant used the same methodology, but with increased internal pressure, for the side drop and end drop analyses, to determine the maximum stress and strain in the fuel cladding.

For the side drop analysis, the applicant compared the calculated maximum bending plus axial stress with the yield strength of the cladding. The applicant calculated a safety factor (yield stress/combined calculated stress) of 2.15 for the side drop evaluation.

For the end drop analysis, the applicant compared the calculated maximum principal strain with the yield strain of the cladding. The applicant calculated a safety factor (yield strain/maximum principal strain) of 2.33 for the end drop evaluation.

These safety factors indicate that the combined calculated stress and strain within the cladding material due to the drop analyses, with increased internal pressure, is less than half of their respective yield values.

The staff concludes that the applicant’s method to calculate these safety factors is reasonable. Because of these conservative safety factors and because the analyzed pressure load is over three times the calculated pressure, the staff concludes that the increase in average burnup will have minimal effect on the structural performance of the fuel cladding for NCT, and is therefore acceptable.

2.2 Findings

Based on a review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated and that the package has adequate structural integrity to continue to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

3.1 Review Objectives

The objective of the review is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC, and that the package design satisfies the thermal requirements of 10 CFR Part 71. This review also determines whether the package fulfills the acceptance criteria listed in Section 3 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," as well as associated Interim Staff Guidance (ISG) documents.

3.2 Description of the Thermal Design

3.2.1 Design Features

Except for the new heat load zoning configuration (HLZC), identified as HLZC No. 8 for the 69BTH dry shielded canister (DSC), the packaging design features documented in the application have been reviewed and accepted previously.

3.2.2 Thermal Design Criteria

Several thermal design criteria are established by the applicant to ensure that the package meets all its functional and safety requirements. These criteria are listed in the application and have been reviewed and accepted previously.

3.2.3 Content's Decay Heat

The applicant revised the thermal analysis in Chapter 3 to include HLZC No. 8 for the 69BTH DSC only. This new HLZC provides an alternate zoning option for the 69BTH DSC to accommodate BWR fuel assemblies with decay heats as high as 950 watts (W) and with a maximum burnup of 70 GWd/MTU. Heat source evaluation for this burnup is provided in Section 5.2. However, the maximum heat load for HLZC No. 8 is 30.2 kW and remains below the current approved maximum heat load of 32 kW.

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

3.2.4 Summary Tables of Temperatures

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

3.2.5 Summary Tables of Maximum Pressures

Summary tables of the containment pressure under NCT and HAC conditions were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation SAR sections. These tables reported the Maximum Normal Operating Pressure (MNOP) for both NCT and HAC (fire). These pressures remain below the design pressures for NCT and HAC.

The staff reviewed the design description of MP197HB package thermal design and finds it acceptable. The staff reviewed the temperature and pressure design limits and calculated temperatures and pressures for the package and found them to be acceptable and consistent in the SAR.

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

The package application provided material thermal properties such as thermal conductivity, density, specific heat, and emissivity for all modeled components of the cask. The staff reviewed these properties and finds them acceptable because they cover the temperature range encountered during transport for normal and accident conditions. The thermal properties used for the analysis of the package were appropriate for the materials specified and for the conditions of the cask required by 10 CFR Part 71 during normal and accident conditions.

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

3.3.2 Component Specifications

The application provided component thermal technical specifications for the MP197HB containment seals and poison plates used in the DSC basket. The specifications are described in the MP197HB SAR and have been reviewed and accepted previously.

3.4 Thermal Evaluation under Normal Conditions of Transport

3.4.1 Thermal Models

To evaluate the thermal performance of the package, the applicant developed three-dimensional (3-D) ANSYS finite element models. These models have been reviewed and accepted previously and, since there were no fundamental changes in the design, the previous technical evaluation by the staff remains valid. The applicant used these models to perform a steady state evaluation of NCT conditions for HLZC No.8.

The staff reviewed the applicant's description of the package thermal models. Based on the information provided in the application regarding the developed thermal models, the staff determines that the application is consistent with the guidance provided in Section 3.5.3 (General Considerations for Thermal Evaluations) of NUREG-1617. Therefore, the staff concludes that the description of the thermal models is acceptable because the description is consistent with NUREG-1617 for meeting 10 CFR Part 71.

3.4.2 Heat and Cold

The applicant performed a steady state analysis, using the thermal model without insolation, to determine the accessible surface temperature of the impact limiters in the shade. A heat load of 32 kW, with boundary conditions at 100°F and no insolation, is considered in the cask model to bound the maximum accessible surface temperature under shade as compared to 30.2 kW. No changes were made to the thermal analysis for 32 kW and, therefore, the previously calculated maximum accessible surface temperature remains valid and bounding.

Calculation results are provided in the application for total heat loads of 32 and 30.2 kW. The SAR results show that both the maximum and average temperatures of the package components for HLZC No. 8 are bounded by those for HLZC No. 4. Therefore, the package components are within the temperature limits for their respective materials, and perform their intended safety function within the operating range for the 69BTH DSC with HLZC # 8 for NCT hot conditions.

Under the minimum ambient temperature of -40°F (-40°C), the resulting packaging component temperatures will approach -40°F if no credit is taken for the decay heat load. Since the package materials, including the containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the package. The maximum component temperatures for ambient temperatures of -40°F and -20°F with maximum decay heat and no insulation are calculated for the 69BTH DSC to use for structural evaluations. These temperatures are used to evaluate the maximum internal pressures within the package and the DSC cavities. Thermal stresses for the package loaded with a 69BTH DSC are discussed in Chapter A.2 of the SAR. The staff confirms that the applicant's calculated maximum temperatures are below the material temperature limits specified in the SAR with sufficient margin and finds them acceptable.

3.4.3 Maximum Normal Operating Pressure

The maximum pressures in the cask cavity calculated for a loaded package are presented in the SAR. The case of the 69BTH DSC in the package, with a 32 kW heat load, is bounding for the maximum cavity pressure for all DSCs. These calculations were reviewed and approved previously and therefore remain valid. The applicant's calculated MNOP is below the containment design pressure, as reported in the application and therefore is acceptable. The staff reviewed selected calculations and results of the package for NCT conditions and found them acceptable.

3.5 Thermal Evaluation during Drying Operations

Thermal evaluations drying operations are documented in the SAR. The thermal evaluation and analysis results have been previously reviewed and accepted.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

The applicant stated that the previous 69BTH DSC fire analysis documented in the SAR for 32 kW and HLZC No. 4 remains bounding for predicted maximum temperatures, as compared to 30.2 kW and new HLZC No. 8. The staff finds the applicant's justification acceptable because all thermal external loads remain unchanged and the total heat load is lower for the new heat load configuration.

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

Therefore, the staff concludes that the HAC analysis is acceptable because the analysis and results are consistent with NUREG-1617 for meeting 10 CFR Part 71.

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

The 69BTH DSC in the package, with a 32 kW heat load, is considered to present the bounding case for reconfigured fuel assemblies. The analysis approach and assumptions were reviewed and approved previously and therefore remain valid. The maximum temperatures of the neutron and gamma shields and the maximum temperature of the wood in impact limiters remain below the allowable limits for NCT and the maximum gamma shield temperature remains below the limits for HAC. The results show that all the confinement and shielding design functions of the package are assured for NCT and HAC, and the design criteria specified in the SAR are satisfied.

The staff reviewed the applicant's analyses during NCT and HAC for altered physical configuration of fuel assemblies. Based on the information provided in the application regarding these analyses, the staff determines that the application is consistent with the guidance provided in Section 3.5.5 (Thermal Evaluation under Normal Conditions of Transport) and Section 3.5.6 (Thermal Evaluation under Hypothetical Accident Conditions) of NUREG-1617. Therefore, the staff concludes that the HAC analysis is acceptable because the analysis and results are consistent with NUREG-1617 for meeting 10 CFR Part 71.

3.8 Thermal Tests

The previous thermal test of the MP197HB fabricated packaging remains unchanged. This test is described Section A.8.1.8 of the SAR.

3.9 Confirmatory Analyses

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

3.10 Findings

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

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

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

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

5.0 SHIELDING EVALUATION

The purpose of this evaluation is to verify that the shielding design of the package provides adequate protection against direct radiation from the requested 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 Shielding Design Description

The Model No. NUHOMS®-MP197HB package is currently authorized to transport spent BWR fuel with a burnup up to 62 GWd/MTU (high burnup). This application requests an amendment to the CoC to increase the allowable burnup limit from 62 GWd/MTU to 70 GWd/MTU for BWR fuel that is loaded in the 69BTH DSC.

The applicant provided a description of the package design. The package consists of a transport overpack and a DSC. The DSC is a cylindrical spent fuel basket with several configurations for various fuel types, including the 69BTH DSC for shipping 69 high burnup BWR fuel assemblies.

The overpack is made of a lead shell and two concentric stainless steel shells holding the lead layer. A borated resin layer is encased in slender aluminum tubes and attached to the outer shell that holds the lead layer. The lead layer and stainless steel shells of the overpack provide gamma shielding and the borated resin provides primary neutron shield at the radial direction. The top and bottom lids provide shielding at the top and bottom ends of the package. The impact limiters provide some additional reduction of radiation mainly at the axial directions by the additional distance.

The inner cavity of the overpack is the same for all fuel basket designs. An aluminum sleeve is used to hold the fuel baskets to the center of the overpack cavity when the fuel baskets are smaller than the cavity of the overpack. Solid aluminum transition rails are built on to the inner wall of the overpack to help center the fuel compartment clusters inside the DSC. Borated aluminum-poison plates for criticality safety control are attached to the walls of the fuel cell tubes by stainless steel sheath. The wall and top and bottom plates of the DSC for the spent fuel also provide significant shielding in the radial and axial directions of the package.

The applicant stated that the package is designed for exclusive use. A personnel barrier is mounted to the transport frame of the vehicle to prevent unauthorized access to the package body. The operators are required to wear a dosimeter and subject to the requirements of 10 CFR 20.1502 on occupational dose which satisfies the requirements of 10 CFR 71.47(b)(4).

The applicant does not propose any change in the shielding design of the package. The package is assumed to be as wide as the open vehicle. Normal conditions of transport dose rates are computed for exclusive-use transport in an open vehicle. The details of the 69BTH DSC fuel basket are shown in the drawings in Section A.1.4.10.10 of Appendix A.1.4.10 of the SAR.

The applicant provided the characteristics of the allowable spent BWR fuel contents of the 69BTH DSC fuel basket in Table A.1.4.9-1 of the SAR. The minimal cooling time for each fuel type is given in Tables A.1.4.9-54 and A.1.4.9-5 of the SAR. The applicant stated that, because the 69BTH DSC fuel basket uses zoned loading, additional cooling times may be necessary for some of the fuel assemblies to be loaded in the peripheral fuel cell locations. The required additional cooling time is determined by Equation A.1.4.9-1 and the B values for the equation are given in Table A.1.4.9-5a, Equation A.5.31 and Table A.5-52 of the SAR. However, the applicant stated if the result of additional cooling time, determined using Equation 5.31, is less than 0, the additional cooling time should be zero.

In addition, this amendment request introduces a new five zone loading pattern that requires both loading low burnup fuel/low decay heat (in zone 1-4) fuel and use of dummy fuel assemblies made of aluminum, which is also called Heat Load Zoning Configuration No. 8 for the 69BTH DSC Basket. Figure A.1.4.9-5a presents a layout of the required new loading pattern.

5.2 Radiation Source and Decay Heat Specification

The applicant calculated the radiation source terms of the spent fuel contents at an assembly average burnup of 70 GWd/MTU using the TRITON/ORIGEN-ARP sequence of the SCALE 6.0 computer code package. The applicant benchmarked the code using the Radiochemical Assay (RCA) samples as published in NUREG/CR-7013, “Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation- Vandellos II Reactor.” The staff notes that although the applicant has benchmarked the code with the best available data, this data, however, is insufficient to support source term calculation for BWR fuel with burnup to 70 GWd/MTU for two reasons: (1) the high burnup fuel RCA data is mainly from PWR fuel rather than BWR fuel and (2) there are only very few data points for burnup at or exceeding 70 GWd/MTU. The applicant considered these facts and applied an extra penalty, of about 10%, to the calculated dose rate limits, which was converted into additional cooling time, in order to compensate the code’s limitation in calculating source terms for BWR fuel at burnup beyond the available RCA data.

The staff evaluated the applicant’s approach for alleviating the code’s limitation on source term calculation and determined that the extra penalty applied to the dose rate calculations is sufficient to offset the code uncertainty in source term calculations at 70 GWd/MTU. This determination is based on two facts. First, the RCA data includes some data points at or beyond 70 GWd/MTU burnup. The code’s ability to provide reliable source term calculation is assured for major isotopes in the spent fuel at around 70 GWd/MTU. Second, the extra penalty is sufficient to account for the uncertainty in the calculated source terms at the burnup not much beyond the burnup range for which the code is fully benchmarked.

The applicant also calculated the decay heat with bounding parameters for the spent BWR fuel at 70 GWd/MTU using the SCALE 6.0 computer code. Similarly, the applicant applied an additional 10% safety margin to the calculated decay heat. The staff determined that this additional safety margin for decay heat calculation is adequate to account for the uncertainty of the code in the burnup range between 62 GWd/MTU and 70 GWd/MTU in lieu of code benchmarking with sufficient RCA data for decay heat and therefore is acceptable. This decay heat is used in the thermal evaluation of the package.

The applicant applied burnup profiles to the source terms calculated based on average fuel burnup. Based on the burnup profile, the applicant further developed neutron and gamma sources as a function of fuel burnup along the axial direction of the fuel assembly. Detailed discussions on the neutron source distribution in the package are found in Section 5.2.2 of the previous SER. The staff verified the burnup profile used by the applicant. The staff compared the burnup profile with the known burnup profiles of typical normally discharged BWR fuel and finds it appropriate and acceptable.

The applicant performed structural performance analyses for the BWR high burnup fuel assemblies with up to 70 GWd/MTU under NCT and HAC. The applicant stated the results of the analyses show that the fuel with a burnup up to 70 GWd/MTU will remain intact under these conditions.

However, since the available data is limited to ensure the material properties of the cladding, and the performance of the fuel, the applicant performed shielding analyses assuming the BWR high burnup fuel assembly could lose its geometric shape. The applicant concluded that, with increased cooling time, the results of the analyses show that the package, with a burnup up to 70 GWd/MTU, continues to meet the regulatory requirements of 10 CFR 71.47 and 71.51 even if the fuel were to reconfigure. This licensing approach is the same as used in the safety analyses in the previous application. The staff finds this approach acceptable because it has considered the worst case scenarios of the fuel assembly performance and conservatively calculated the package dose rates assuming that the fuel, therefore the source, would reconfigure.

5.2.1 Gamma Source

The gamma source terms comprise of three parts in the content: spent fuel, activated fuel structural materials and inserts, and the source term from (n, γ) reactions. The applicant uses the same gamma source calculation method as used in the SAR for the previously approved MP-197HB package design. The staff verified the gamma source calculated by the applicant using the Origen/Arp software in the SCALE 6.1 software package and confirmed the applicant's calculated gamma source. On this basis, the staff finds that the applicant has correctly calculated the gamma source and the results are acceptable.

5.2.2 Neutron Source

The neutron sources are comprised of primarily the neutron radiation from spent fuel (both α-n reactions and spontaneous fission) with neutrons produced by the sub-critical multiplication in the fuels. The applicant uses the same neutron source calculation method as used in the SAR for the previously approved MP-197HB package design. The staff verified the neutron source calculated by the applicant using the Origen/Arp software in the SCALE 6.1 software package and confirmed the applicant's calculated neutron source. On this basis, the staff finds that the applicant has correctly calculated the neutron source and the results are acceptable.

5.3 Shielding Model Specification

The applicant performed a shielding analysis for the package using the MCNP5 computer code and the ENDF/B-VI cross section libraries. Under NCT, the package model includes the neutron shield and impact limiters. Under HAC, the applicant modeled the package assuming that the package completely lost the neutron shield and the impact limiters.

The applicant calculated the dose rates on the surface and two meters from surface of the package loaded with the 69BTH DSC. The applicant modeled the overpack, the DSC, and its contents, explicitly as built. The applicant included tolerances of the fuel basket and overpack in the shielding models. The applicant modeled the fuel assemblies in the package as 18 homogenized discrete segments along the axial direction. Gamma and neutron sources are adjusted with the burnup in each zone to account for the source term as a function of fuel burnup.

The applicant modeled the package under NCT as a carbon steel cylindrical shell with a cavity 70.50 inches in diameter and 199.25 inches in height with other dimensions, as shown in licensing drawings. The package cavity and the fuel basket are modeled as built. Voids and the fuel assembly in the fuel compartment are homogenized with fuels and cladding material. The staff determined that this is an acceptable modeling approach for representing the fuel regions in the shielding calculations because modeling the details of the fuel assembly is not necessary for shielding calculations.

The package is secured in a horizontal position on a skid attached to a railcar, or any other trailer that has a deck or floor, during transportation. The applicant modeled the package assuming the dimension of the railcar is the same as the package outer dimension; no credit was taken for the edge of the vehicle. The staff finds this assumption is conservative because the package outer dimension is smaller than the outer edge of the transportation vehicle.

The impact limiters are modeled as wood encased in a 0.25 inch thick steel shell. There is a 3.5 inch gap between the impact limiter pocket inner surface and the transport package lid. The interior steel gussets are conservatively neglected in the shielding model. Wood thickness between the end of the impact limiters and the package ends is modeled 23.75 inches in MCNP models. The outer diameter of the impact limiters is 126 inches.

The applicant calculated the dose rates at different points from the design basis packages with the model as described above. Based on its calculations, the applicant determined the dose rates for the package with the design basis content. The maximum radiation dose rates for intact fuel during NCT are provided in Table A.5-1 of the SAR. Dose rates are computed to bound all authorized burnup and enrichment combinations defined in the fuel qualification tables in Chapter A.1 of the SAR.

The applicant used the Cm-244 spontaneous fission spectrum as the neutron source energy distribution in the shielding calculation using the MCNP5 code. The applicant used the “nonu” and “PHYS: n, p” cards in the neutron transport calculations of the MCNP model to determine the dose rates of neutron and secondary gammas. A separate gamma transport calculation was made for determining the dose rates from the fuel and hardware gamma radiation. The applicant used the ANSI/ANS-6.1.1-1977 flux-to-dose factors to convert the mesh tally fluxes to dose rate. The staff finds this modeling approach is a common practice and the flux-to-dose factors used are consistent with the guidance provide in the NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel.” Therefore, the staff finds the modeling approach acceptable.

Based on the results of the applicant’s shielding analyses under NCT, the calculated maximum dose rate is 46.10 mrem/hr on the surface of and 8.16 mrem/hr at 2 meters from the surface of the package with 69 BWR fuel assemblies at 70 GWd/MTU. With consideration of fuel reconfiguration, the calculated maximum dose rate is 120.44 mrem/hr on the surface of and 8.61 mrem/hr at 2 meters from the surface of the package with 69 BWR fuel assemblies at 70 GWd/MTU.

The applicant calculated the dose rates for the package under HAC. The applicant assumed that all neutron shield and the impact limiters are lost. The calculated maximum dose rate at 1 meter from the surface of the package is 836.86 mrem/hr for the 69 BWR fuel assembly package at 70 GWd/MTU with consideration of fuel reconfiguration. The applicant determined the calculated results show that the package shielding design meets the regulatory requirement, i.e., not to exceed 1,000 mrem/hr at 1 meter from the package under HAC. Based on the calculated dose rate, the staff determined that package design for BWR fuel contents with the new burnup limit meet the regulatory requirements of 10 CFR 71.47 and 71.51.

The material compositions used in the shielding models are typical material compositions and densities for stainless steel, lead, aluminum, boron, and UO_2. The neutron shield material VYAL-B is a non-standard material and the SAR provides a detailed description of the material composition in Table A.5-13 of the SAR. The staff finds that the material properties used in the models for shielding safety design of the package are consistent with the package design and the material specifications in common shielding analysis practice. On the basis of the above findings, the staff determined that the material properties used by the applicant in the shielding models acceptable.

5.4 Fuel Reconfiguration Consequence Analyses

The applicant performed a structural evaluation in Appendix A.2.13.11, based on available cladding material property data, to demonstrate that the fuel remains intact under NCT and HAC. Since the material property data for BWR high burnup fuel is limited at the present time, the applicant performed shielding analyses assuming reconfigured fuel assemblies for the package under NCT and HAC conditions to demonstrate that the package still meets the regulatory requirements of 10 CFR 71.47 and 71.51 for radiation shielding, even if the fuel were to reconfigure.

The applicant developed an MCNP model of the package containing the 69BTH DSC under NCT, assuming reconfigured fuel with the bounding source characteristics of 5.0 wt.% and 70 GWd/MTU. Twenty five axial nodes are used in the MCNP model with burnup dependent spent fuel neutron and gamma source and material composition. The isotopic compositions (28 isotopes of interest) of each axial node are calculated using ORIGEN-ARP with a bounding BWR fuel assembly at 5.0 wt.% enrichment and an average burnup of 70 GWd/MTU.

The applicant presents the calculated dose rates with reconfigured fuel in Table A.5-1b of the SAR. For a package containing the 69BTH DSC with fuel at 70 GWd/MTU burnup, the results show that the calculated maximum dose rates are about 124.22 mrem/hr on the surface of the package and 8.56 mrem/hr two meters from the surface of the package under normal conditions of transport. In this case, the applicant modeled the package "as built", i.e., the models included the neutron poison plates.

The applicant performed additional analyses on the consequences of fuel reconfigurations with various compaction factors. The additional evaluations include fuel compaction cases up to the maximum theoretical fuel compaction assuming broken fuels rodlets in a closely packed square of material inside the basket compartment walls. The applicant stated that these evaluations demonstrate that the shielding performance of the packaging is maintained during transportation under NCT and the radiation levels remain below the limits even if the fuel assemblies were to reconfigure. With these calculated dose rates, the applicant demonstrated that the shielding evaluations provide assurance that all applicable regulatory requirements as specified in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2) for an exclusive-use transportation in an open vehicle are satisfied even under the assumption of the non-mechanistic loss of cladding integrity.

Radiological and thermal surveys, as described in Section A.7.2.1, will be performed prior to unloading operation. These surveys will indicate if axial reconfiguration/relocation of fuel had occurred during NCT. In the unlikely event that fuel reconfiguration/relocation has occurred, tests and procedures described in Table A.7-5 allow for unloading operations using a dry cell and provide instructions for adjusting the boron concentration, using filters, special tools, etc., if wet unloading operation is used.

In addition, the user must stop further shipments and notify the NRC if, during the unloading operations, as described in Chapter A.7, Table A.7-5, an unexpected cladding damage is detected that has occurred during NCT. The shielding evaluations for HAC, assuming reconfigured high burnup BWR fuel assemblies, are presented in Section A.5.3.1.2 (Configurations 2 and 3). Similar to the HAC evaluations with the intact fuel assemblies (Configuration 1), no credit is taken for the presence of the neutron shielding material when the high burnup fuel assemblies are assumed to be reconfigured. The assumptions considered for the reconfigured fuel assemblies under HAC are the same as those described for NCT except for the axial source profile. The applicant also evaluated dose rates for the package with a source term for a hypothetically reconfigured fuel (Configuration 3). Additional sensitivity evaluations on axial compressed fuel configurations are performed in Section A.5.3.1.2 for the 69BTH DSC. The staff finds that, through these additional sensitivity analyses, the applicant demonstrated that the scenarios chosen represent the maximum possible dose rates and are therefore acceptable.

5.5 Shielding Evaluation

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

The applicant used MCNP5, version 1.4 for the shielding analyses. The MP-197HB package is modeled with full three-dimensional details, the active fuel region is modeled as a homogenized mass and the upper and bottom end fitting regions of the design assembly are modeled as mixture of steel and void. The overpack body, neutron shield, radial steel ribs, and impact limiters are explicitly modeled in accordance with the design as shown in the design drawings.

In the applicant's MCNP5 model, surface flux type tallies are performed at the package surface, at 1 meter from the surface and 2 meters from the surface of the package. The MCNP5 mesh tally was used to determine the neutron and gamma fluxes. The applicant used the ANSI/ANS-6.1.1-1977 flux-to-dose factors to convert the mesh tally fluxes to dose rate. This is consistent with the acceptance criterion of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." The maximum dose rates are determined at key locations along the sides of the package, i.e., radial mid-plane, top, bottom, and along the centerline of the impact limiters.

The staff reviewed the shielding design of the package. Based on the dose rates together with the analyses, the assumptions and approximation used in the analyses as presented in the application, the staff determined the package design acceptable and there is a reasonable assurance that the dose rates of the package meet the regulatory requirements of 10 CFR 71.47 and 71.51 under both NCT and HAC.

Based on the conclusions from the material and structural evaluations, staff determined that the high burnup fuel cladding will retain its integrity under NCT and HAC. With this assessment, the staff believes that the proposed high burnup BWR fuel can be transported as intact fuel. To address concerns on the limitation of experimental data on the mechanical performance of high burnup fuel, the staff determined the additional safety margin included by the applicant demonstrates that the package will meet the regulatory requirements of 10 CFR Part 71 even if the fuel were to reconfigure. The staff determined the fuel reconfiguration consequence analyses provides additional assurance of public safety in the worst case scenario.

The staff reviewed the fuel reconfiguration consequence analyses, including the consequence analysis logic, the conclusion on fuel reconfiguration impact on package radiation safety, the derivation of formulas for calculation of the additional time needed, and the method for deriving the coefficients for the equations. Based on its review, the staff determined that the methods developed are adequate for the reasons discussed above and the package shielding design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

Since the package will be transported in a horizontal position, the horizontal fuel reconfiguration represents the NCT and a position for unloading operations during which the package will be lifted first to allow for removal of the impact limiter and the closure lids.

It is important to note that the staff considered fuel reconfiguration analyses as an assessment on the consequence of fuel reconfiguration rather than the design basis of high burnup fuel transportation package. Fuel with known damage(s) must be treated as damaged fuel. In addition, the shipper of the package must be able to follow the operating procedures, prescribed in Chapter 7 of the application, to determine the conditions of the fuel.

5.6 Evaluation Findings and Conclusion

The staff reviewed the description of the package design features related to shielding and the source terms for the design basis fuel. 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 with the following conditions placed in the CoC fuel qualification tables:

1. The maximum length of the natural or low enrichment uranium blankets shall not exceed 5% of the assembly length.
2. The maximum average burnup is 70 GWd/MTU for authorized BWR fuel types in the 69BTH DSC.
3. For 69BTH Type F BWR fuel package at 70 GWd/MTU burnup, the five zone loading pattern as presented in Figure A.1.4.9-5a must be used.
4. The minimal B-10 areal density in the fuel basket neutron poison plates should be 18.9 mg/cm^2.
5. The package must be transported by private carriers and the operators, required to wear a dosimeter, are subject to the requirements of 10 CFR 20.1502 on occupational dose.

CHAPTER 7 OPERATING PROCEDURES

The operating procedures for the 69BTH DSC, in Chapter A.7 and Appendix A.7.7.9 (specifically Table A.7-2a), direct the user to Table A.1.4.9-1 to determine applicable fuel specifications for loading the 69BTH DSC.

More specifically, for HLZC 8, Table A.1.4.9-1 refers the user to Figure A.1.4.9-5a which indicates to the user which of the 69BTH DSC cells must be loaded with fuel and which cells must be loaded with aluminum dummy assemblies, while Table A.1.4.9-5b indicates to the user the acceptable combinations of burnup, enrichment and cooling times acceptable for loading in each zone of HLZC 8.

Therefore, the operating procedures currently indicated in the SAR, through the use of Table A.7-2a, alert the user to all the requirements related to the use of HLZC 8. The applicant provided also some clarity for loading the 69BTH Type F DSC.

CONDITIONS

The following changes are included in Revision No. 8 to Certificate of Compliance No. 9302:

Item 3(c) has been modified to reflect the new name, TN Americas, LLC, of the Certificate Holder.

Item 3(d) has been modified to reference revision No. 18 of the application dated April 2017.

Condition No. 5(a)(5) was modified to include Revision No. 5 of drawing MP197HB-71-1005.

Condition No. 12 authorizes use of the previous revision for a period of approximately one year, until May 31, 2018.

Condition No. 13 was modified to reflect the timely renewal request of the certificate. The new expiration date of the certificate is August 31, 2022.

The References section was modified to include the consolidated application, Revision No. 18, dated April 2017.

CONCLUSION

Based on the statements and representations contained in the application, and the conditions listed above, the staff concluded that the Model No. NUHOMS® – MP197 package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9302, Revision No. 8
On May 23, 2017.