Request for Additional Information
Transnuclear, Inc.
Docket No. 71-9302
Model No. NUHOMS®-MP197HB Package

By letter dated March 2, 2012, Transnuclear, Inc. (TN) submitted an application for revision to Certificate of Compliance (CoC) No. 9302 for the Model No. NUHOMS®-MP197HB package, including high burnup fuel assemblies and the 32PT, 24PTH, 32PTH, 32PTH1, and 37PTH dry storage canisters (DSCs) as proposed contents of the package. On August 15, 2012, the U.S. Nuclear Regulatory Commission staff (the staff) received your responses to the request for supplemental information dated July 13, 2012, and your application was accepted for review on September 24, 2012.

This Request for Additional Information (RAI) identifies information needed by the staff in connection with its review of the Model No. NUHOMS®-MP197HB package application to confirm whether the applicant has demonstrated compliance with regulatory requirements. The requested information is listed by chapter number and title in the package application. NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," was used for this review.

Chapter 2 – Structural Evaluation

2-1 though 2-16: Refer to Proprietary Enclosure

2-17 Demonstrate that the assumptions used for the full scale dynamic analysis are appropriate for determining the G loads used in the package stress analysis.

The methodology used for the 1/3 scale benchmarking was replicated for the full scale package analysis including the use of a surrogate representation of the package body. While this modeling approach would be sufficient for comparison to a test specimen for the same configuration, it is not necessarily appropriate for generating G-loads for further quasi-static analysis. The full scale detailed package analysis should simulate the actual geometric configuration of the design to the extent practicable with finite element analysis techniques.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-18 Provide input and output files for representative hypothetical accident conditions (HAC) analyses.

Staff requests both the 30-foot end drop and 30-foot side drop input and output files of the detailed package model(s) - see RAI 2-17.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-19 Provide a direct comparison of the stress results obtained from ANSYS and LS DYNA for the 30-foot end drop and 30-foot side drop.

While the LS DYNA approach used has not been fully benchmarked, a direct comparison of the stresses generated in the quasi-static analysis to the stresses generated in the dynamic analysis provide reasonable assurance that the structural behavior is representative of that obtained with a full scale test.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-20 Provide a sensitivity analysis which demonstrates that the constant stress element formulation is appropriate for the analyses presented.

Despite the constant stress element being a default choice within the LS DYNA software platform, the staff has concerns about whether that element formulation is appropriate when considering the loading(s) that the package is subjected to during drop events.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-21 Demonstrate that Nelm's equation is sufficient to evaluate outer shell and neutron shield performance under HAC puncture loads.

Nelm's equation is an empirical approach developed exclusively for lead backed steel shells. Given that the design of this package deviates from that configuration with the presence of a neutron shield, the request for moderator exclusion, and the result of Nelm's equation showing that the overall shell thickness is inadequate, staff does not have reasonable assurance that a breach of the package will not occur or that the assumptions used in other analyses are conservative.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(3).

2-22 Demonstrate that the use of the simplified bolt modeling technique (spring and beam elements) is sufficient to accurately capture the bolt behavior and the interaction with surrounding structural components.

Staff does not have reasonable assurance that the simplified bolt modeling technique is appropriate nor sufficiently benchmarked.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-23 Justify the use of 120 Hertz filtering for the slapdown case when all other cases utilize 180 Hertz.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

2-24 Provide a sensitivity study demonstrating that automatic general contact is an appropriate modeling choice over automatic single surface contact.

Staff does not have reasonable assurance that the deformation patterns exhibited during the 30-foot drop analyses are representative of actual deformations. If the contact definitions are not defined correctly, energy dissipation and subsequent damage can be non-conservative.

This information is required by the staff to determine compliance with 10 CFR 71.73.

2-25 Provide a comparison of the time history of the package lid for the detailed package model (RAI 2-17) to the time history of the cask node in the single pin model.

Staff does not have reasonable assurance that using a filtered end drop peak G magnitude and corresponding constant force-deflection characteristics are representative of the G-loads that a fuel assembly or pin may be exposed to during an actual 30-foot drop.

This information is required by the staff to determine compliance with 10 CFR 71.73 (c)(1).

Chapter 3 - Thermal Evaluation

3-1 Clarify the maximum heat load for the 61BTHF and 24PTHF Dry Shielded Canisters (DSCs) allowed for transportation.

Page A.3-1 of the application includes a table with a maximum heat load per DSC allowed for transportation, but a value is not provided for the above mentioned DSCs.

This information is required by the staff to determine compliance with 10 CFR 71.71 and 71.73.

3-2 Clarify which specific canisters are required to have spacers to accommodate short fuel assembly fuels.

Page A.3-5 of the application states that there are two types of 37PTH canisters: a short canister type designated as the 37PTH-S and a medium canister type designated as the 37PTH-M. Use of spacers may be required to accommodate short fuel assemblies. The application should clearly state what DSCs require the use of spacers. The loading procedures should clearly indicate this as well.

This information is required by the staff to determine compliance with 10 CFR 71.71 and 71.73.

3-3 Provide the thermal conductivity values of the poison material to cover the expected temperature range during normal conditions of transport (NCT) and HAC.

Page A.3-28 of the application provides a table with the minimum conductivity of the poison material. However, the temperature range does not appear to cover the maximum temperatures during NCT and HAC.

This information is required by the staff to determine compliance with 10 CFR 71.71 and 71.73.

3-4 Provide an analysis to demonstrate that ignoring helium conductivity in the axial and radial effective thermal conductivity of the internal sleeve is conservative.

Page A.3-50 of the application states that, due to the low conductivity of helium in comparison to aluminum, helium conductivity can be conservatively ignored. The staff performed the calculation by considering the helium conductivity to verify this statement and was not able to draw the same conclusion.

This information is required by the staff to determine compliance with 10 CFR 71.71.

3-5 Perform a grid convergence index (GCI) calculation to obtain the discretization error by following the procedures described in American Society of Mechanical Engineers Verification and Validation 20-2009 (ASME V&V 20-2009), "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer."

Page A.3.73 of the application states that the maximum neutron shield temperature is 316°F (158°C) for NCT, which is below the long term limit of 320°F (160°C). However, this temperature is very close to the long term limit. Therefore, the discretization error should be obtained to obtain the uncertainty in the calculation through the GCI calculation to verify that the limit will not be exceeded. Also, a sensitivity study of the model input parameters should be performed to assure bounding calculated results are obtained.

This information is required by the staff to determine compliance with 10 CFR 71.71 and 71.73.

3-6 Clarify why the presence of water within the annulus maintains the maximum DSC shell temperature below 212°F during loading/unloading operations.

Page A.3-84 of the application states that the water in the annulus is replenished with fresh water to prevent boiling and maintain the water level if excessive evaporation occurs. Presence of water within the annulus maintains the maximum DSC shell temperature below the boiling temperature of water in open atmosphere. This assumption is valid at the water surface. However, at the bottom of the cask the water temperature will be higher due to the hydrostatic head.

This information is required by the staff to determine compliance with 10 CFR 71.71.

3-7 Explain why thermal initial conditions obtained from NCT for the thermal evaluation under HAC are acceptable for the case of a homogenized basket. Explain the process of transferring the temperature distribution from the detailed DSC analysis to the homogenized basket used during HAC so realistic or conservative temperatures are used for the DSC with a homogenized basket model.

Page A.3-86 of the application states that the initial temperatures for the MP197HB package transient model before the fire accident are determined using the same boundary conditions for NCT (100°F ambient with insolation). However, the process of

transferring the temperature distribution from a detailed DSC analysis to a model with homogenized basket is not clear.

This information is required by the staff to determine compliance with 10 CFR 71.71.

3-8 Perform a grid convergence index (GCI) calculation to obtain the discretization error by following the procedures described in American Society of Mechanical Engineers Verification and Validation 20-2009 (ASME V&V 20-2009), "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer."

Page A.3-95 of the application states that, during HAC, the maximum seal temperature for fluorocarbon seals is 394°F at the cask lid for a 32 kW heat load in a 69BTH DSC when the package is equipped with external fins. This temperature is very close to the long term limit. Therefore, the discretization error should be calculated to obtain the uncertainty in the calculation through the GCI calculation to verify that the limit will not be exceeded. Also, a sensitivity study of the model input parameters should be performed to ensure that bounding calculated results are obtained (see also RAI 3-5).

This information is required by the staff to determine compliance with 10 CFR 71.71 and 71.73.

Chapter 4 – Containment Evaluation

4-1 Clarify that the containment welds are a part of the containment boundary.

Given the containment welds shown in Figures A.4-1 and A.4-2 (Part 2) of the application, the applicant should modify the statement in Section A.4.1.1 as: "The containment boundary for NUHOMS-MP197HB cask consists of a cylindrical inner shell, a bottom plate with a RAM access closure plate with seal, a cask body flange, a top lid with seal, vent and drain port closure bolts and seals, <u>and all containment welds</u>."

This information is required by the staff to determine compliance with 10 CFR 71.33.

4-2 Clarify whether the lid bolt, the ram access closure bolt, and the bottom plate are part of the containment components in Section A.4.1.1 or notes of Figure A.4-1.

The applicant described the containment boundary in Section A.4.1.1 and listed it in the notes of Figure A.4-1 of the application. With the inconsistency between Section A.4.1.1 and Figure A.4-1, the applicant should clarify in Section A.4.1.1 and in the notes of Figure A.4-1 whether the lid bolt in Item 1 and the ram access closure bolt in Item 6 are the containment components and add the bottom plate as the containment component in the notes of Figure A.4-1.

This information is required by the staff to determine compliance with 10 CFR 71.33.

4-3 Include a statement in Part 3 of Document E-33299 to show that the DSC will meet all conditions of the appropriate storage CoC before returning to storage.

The applicant delineated in step 9 of Part 3, "Helium Leakage Test Procedure," of Document E-33299 (Enclosure 7 to TN E-32784) that, if the leakage rate is greater than 1×10^{-3} ref-cm^3/s, the DSC cannot be transported and shall be returned to the HSM. The applicant should include a statement that clearly acknowledges understanding that

all of the storage conditions must be met if the DSC is not verified for transport and is needed for continued storage.

This information is required by the staff to determine compliance with 10 CFR 71.35 and 71.39.

Chapter 5 – Shielding Evaluation

5-1 Describe the modeling approach for the shielding analysis of the MP-197HB package.

Page A.5-2 of the application states: "The maximum radiation dose rates for NCT are shown in Table A.5-1. Dose rates shown in Table A.5-1 are for intact fuel only. Reconfigured fuel is described in Section A.5.4.3.2." However, there is no detailed description of the approach in Section A.5.4.3.2 and the section seems only to covers the 69BTH package. The applicant needs to provide a detailed description of the modeling approach of the shielding analysis of the MP-197HB package.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-2 Provide shielding analyses for packages with reconfigured fuel at a horizontal position.

The application provides analyses for some high burnup fuel reconfiguration scenarios. However, fuel reconfiguration in packages at a horizontal position was not considered in these analyses.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-3 Demonstrate that the structural integrity of the impact limiter, subsequent to the 30-foot end drop and fully engulfing fire, is consistent with the assumptions made in the shielding evaluations with respect to dose rate calculation locations.

Page A.5-2 of the application states: "For purpose of HAC, the impact limiters are replaced with air." Page A.5-13 further states: "Dose rates are calculated using mesh tallies 1 m below the bottom impact limiter, 1 m above the top impact limiter, 1 m from the side of the cask, and on the side of the cask at the surface of the impact limiter. Note, the impact limiters are modeled as air since it is assumed that no wood is left under HAC. While the steel shells of the impact limiters are also conservatively modeled as air, they are expected to remain intact. Therefore, it is still appropriate to choose dose rate locations with respect to the impact limiter dimensions."

It is not clear if the appropriate structural analyses (static crushing of impact limiter shell(s), post fully engulfing fire) have been performed to support this conclusion. The applicant needs to demonstrate that the structural integrity of the impact limiter, subsequent to the 30-foot end drop and fully engulfing fire, is consistent with the assumptions made in the shielding evaluations with respect to dose rate calculation locations.

This information is required by the staff to determine compliance with 10 CFR 71.51.

5-4 Refer to proprietary enclosure.

5-5 Clarify the fuel burnup for the dose rates data shown in Table A.5.

Page A.5-2 of the application states: "The maximum radiation dose rates for NCT are shown in Table A.5-1. Dose rates shown in Table A.5-1 are for intact fuel only.

Reconfigured fuel is described in Section A.5.4.3.2." However, Section A.5.4.3.2 appears to cover only the 60BTH and 37PTH canisters. The applicant needs to clarify if the dose rate data shown in Table A.5-1 are for low burnup or high burnup, or both.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-6 Refer to proprietary enclosure.

5-7 Refer to proprietary enclosure.

5-8 Check and correct the corresponding energy of the 90,000 Ci Co-60 and recalculate the allowable GTCC.

Page A.5-5 of the application indicates that the corresponding energy of GTCC equivalent to 90,000 Ci of Co-60 is 225 GeV/s. The staff's calculation shows that the radiation energy for 90,000 Ci of Co-60 is 8.325×10^{12} GeV/s rather than 225 GeV/s. Because this value will be used by the users to determine the quantity of the allowable GTCC, the applicant needs to check and correct the 90,000 Ci Co-60 corresponding energy and if necessary, recalculate the allowable GTCC. In addition, explain how this error was not prevented by the quality assurance programs implemented during the MP-197HB design.

This information is required by the staff to determine compliance with 10 CFR 71.47, 71.51, and Subpart H of 10 CFR Part 71.

5-9 Clarify if the scenarios analyzed for high burnup fuel in a package under NCT are bounding reconfigurations and, if so, explain why.

Page A.5-11 of the application states: "Three NCT configurations are analyzed: (1) no damaged fuel assemblies, (2) up to 4 damaged assemblies, and (3) between 5 and 24 damaged assemblies. Damaged fuel assemblies are modeled with the active fuel length reduced to 75% of the nominal value." However, it is not clear if these scenarios represent the worst-case fuel reconfiguration scenarios with respect to shielding requirements.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-10 Provide detailed information on the determination and results of the additional cooling time required to meet the regulatory limits of 10 CFR 71.47 and 71.51 for packages with reconfigured fuel.

Page A.5-11 of the application states: "Three NCT configurations are analyzed. When 24 fuel assemblies are rubblized to reduce the radial dose rate, additional cooling time is applied to the damaged fuel assemblies. The dose rates are driven by the outer damaged assemblies, and the additional cooling time lowers the radial dose rates considerably." However, there is no specific information for the differences in terms of cooling time for various packages. The applicant needs to provide detailed information on the determination and results of the additional cooling time required to meet the regulatory limits of 10 CFR 71.47 and 71.51 for packages with reconfigured fuel.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-11 Provide a justification for the assumption that all compacted fuel is considered to be flat.

Page A.5-12 of the application states: "The axial profile for all compacted fuel is assumed to be flat for both gammas and neutrons consistent with the recommendations of NUREG/CR-6802." However, this assumption may not be valid particularly for packages at horizontal position.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-12 Provide justification for the RCA samples used in depletion code benchmarking for the source term calculations of high burnup fuel.

The application presented benchmark analyses for the source term calculation computer code TRITON. However, these samples may not be appropriate because a majority of the samples selected are in the low burnup range. As such the selected samples may not be able to represent the high burnup range of the fuel for which the packages are designed.

In addition, there appears to be errors in the data, particularly the last two rows, presented in Tables A.5-42 to A.5-44. The applicant needs to check and correct the data, if necessary, in Tables A.5-42 to A.5-44 and revise the source term calculations with new correction factors.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-13 Refer to proprietary enclosure.

5-14 Provide justification for using the technique in biasing the RCA data.

Page A.5-17 of the application indicates that a special technique obtained from reference 12 was used to adjust the uncertainty of measured data. However, it is not clear if the said data manipulation technique is appropriate for this use. If so, why was this technique not recommended in code benchmarking analyses for burnup credit? The applicant needs to provide a justification for using the special technique for biasing the RCA data in depletion code benchmark analyses.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-15 Provide the design basis neutron and gamma sources before adjustments

Tables A.5-36 and A.5-37 of the application provides gamma and neutron sources for the design basis PWR and BWR fuel assemblies. However, the data shown in these tables seem to be neutron and gamma sources after applying various adjustments. While these data are useful to the staff, they do not provide information on the impact of these various adjustments. The applicant needs to provide the design basis neutron and gamma sources before adjustments. Also, the applicant needs to provide the specific design parameters, such as power density, moderator density, cooling time, down time between cycles, in the models used to calculate the sources.

This information is required by the staff to determine compliance with 10 CFR 71.47 and 71.51.

5-16 Refer to proprietary enclosure.

5-17 Refer to proprietary enclosure.

5-18 Refer to proprietary enclosure.

5-19 Refer to proprietary enclosure.

5-20 Refer to proprietary enclosure.

Chapter 6 – Criticality Evaluation

6-1 Revise the fuel specifications for each DSC in Chapter A.1 and A.6 to state that replacement rods in reconstituted fuel assemblies should displace an equal or greater amount of water in the active fuel region than the original rods.

Replacing rods with smaller water displacement will cause the resulting fuel assembly to be more reactive than evaluated in the criticality analysis due to increased moderation by water. A restriction on replacement rod water displacement is stated in the criticality analysis of some, but not all, DSCs evaluated in the SAR. This restriction should be present in the fuel specification tables in each of the Appendix A.1 Chapters A.1.4.1 through A.1.4.9, and in appropriate tables in Appendix A.6.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-2 Refer to Proprietary Enclosure.

6-3 Revise the description of the allowable contents for the NUHOMS-24PT4, 24PTH, and 61BTH DSCs in Appendix A Sections A.1.4.1.3, A.1.4.3.3, and A.1.4.8.3, respectively, to describe the configuration of the rod storage basket, and to clarify whether or not the rod storage basket is required to be transported inside a failed fuel can.

The rod storage basket is briefly described in Sections A.1.4.1.3, A.1.4.3.3, and A.1.4.8.3 of Appendix A as allowable configurations for damaged fuel in the NUHOMS-24PT4, 24PTH, and 61BTH DSCs, and k_{eff} results are reported for two criticality analyses of the rod storage basket in Table A.6.5.3-11. However, the configuration of the rod storage basket is not specified, either in the allowable contents description or in the text describing the criticality analysis performed for this configuration. The description should be revised to state the materials of construction, range of dimensions, and fuel material limits (e.g., MTU) for the rod storage basket, as well as whether or not the tubes that make up the basket are closed or screened, and whether or not the rod storage basket should be transported inside a failed fuel can. Also, the criticality analyses in Sections A.6.5.1, A.6.5.3, and A.6.5.5 of the SAR should be revised to reflect the credible range of configurations that might exist in the rod storage basket.

The applicant shall identify the most reactive credible configuration consistent with the chemical and physical form of the contents.

This information is required by the staff to determine compliance with 10 CFR 71.55.

6-4 Revise the fuel specification for all fuel assemblies to include maximum active length.

The specifications for all of the various PWR and BWR fuel assemblies in Appendix A.1 of the SAR include maximum total fuel assembly length, but do not include the maximum active fuel length evaluated in each of the DSC criticality analyses.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-5 Revise the maximum planar or lattice average initial enrichment and minimum burnup combinations (as appropriate) for damaged fuel in each DSC Appendix section as necessary to clarify the use of these limits.

The maximum planar or lattice average initial enrichment limits, or combined maximum planar or lattice average initial enrichment and minimum burnup requirements for DSCs evaluated with burnup credit, are given in separate tables for intact and damaged fuel in Appendix A.1 for each DSC. However, it is not always clear if the limits in the damaged fuel tables are for all fuel in the DSC, or for the damaged fuel only. Note that Appendix A.1.4.3 for the NUHOMS-24PTH DSC is clear in Table A.1.4.3-9 that the limits are for "All Fuel Assemblies when Damaged Fuel Assemblies are Loaded."

This information is required by the staff to determine compliance with 10 CFR 71.33.

6-6 Refer to Proprietary Enclosure.

6-7 Refer to Proprietary Enclosure.

6-8 Revise the criticality analysis for BWR fuel in the 61BT, 61BTH, and 69BTH DSCs to describe the modeling of partial length rods.

The parameters for BWR fuel assemblies for shipment listed in Tables A.6.5.1-3, A.6.5.1-53, and A.6.5.2-2 for the NUHOMS-61BT, -61BTH, and -69BTH DSCs, respectively, indicate the presence of partial length rods in several assembly types. The criticality analyses for these DSCs do not discuss how partial length rods were modeled. If modeled explicitly, as opposed to assuming all rods are full length, then the numbers, lengths, and specific locations of partial length rods in the evaluated fuel assemblies should be listed as fuel parameters in the aforementioned tables.

The applicant shall identify the most reactive credible configuration consistent with the chemical and physical form of the contents.

This information is required by the staff to determine compliance with 10 CFR 71.55.

6-9 Refer to Proprietary Enclosure.

6-10 Refer to Proprietary Enclosure.

6-11 Clarify the statement in Item 15 of the list in Section A.6.5.1.4.1 of Appendix A regarding the assumption of volume of aluminum and stainless steel in the DSC rails.

This section states that "it is assumed that the volume of the aluminum and stainless steel in the R45 rails is not greater than 56,000 in^3 against the current value of 46,608 in^3." It is not immediately clear that assuming there is more volume of metal in the rails than as designed is conservative, particularly for stainless steel.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-12 Revise the application to demonstrate that the ABB fuel assemblies in the 61BTH DSC BWR criticality analysis have been appropriately considered in the analysis to determine most reactive fuel assembly design.

The text of Appendix A.6.5.1 states that the most reactive fuel assembly design of all those evaluated and listed in Table A.6.5.1-53 is the GE12/GE14 10x10. However, it is not clear that all of the ABB fuel assembly designs have been appropriately considered. Table A.6.5.1-56 shows the calculated k_{eff} for each fuel design, and shows that the GE12/GE14 10x10 is more reactive than the ABB-10-3 10x10 fuel assembly. Table A.6.5.1-52 lists four ABB fuel assembly types (SVEA-64, SVEA-92, SVEA-96, OPTIMA, OPTIMA 2, and SVEA-100). It is not clear which of these types is the ABB-10-3 10x10 considered in the analysis represents, or that this is the most reactive of the ABB fuel types. Additionally, the criticality analysis of the 69BTH in Appendix A.6.5.2 shows that the SVEA-96 OPTIMA2 assembly modeled in the package exceeds the reactivity of the GE12/GE14 10x10, and determines a reduced initial enrichment value for this assembly design. Either provide a similar analysis for the 61BTH, remove this assembly type as an authorized content, or otherwise clarify why this assembly type is acceptable as an authorized content.

The applicant shall evaluate the most reactive credible configuration consistent with the chemical and physical form of the material.

This information is required by the staff to determine compliance with 10 CFR 71.55(b) and (e).

6-13 Revise Sections A.6.5.1.6 and A.6.5.3.6 of Appendix A to include relevant details of the fresh fuel benchmarking analyses of the NUHOMS-61BTH and -24PT4, respectively.

Each of these sections refers to previous benchmarking analyses in support of criticality analyses for Part 72 storage applications. Although the details of the benchmarking analyses can be referred to in another application, relevant details, such as the number and type of critical experiments modeled, should be included in this application. Also, note that Part 72 storage criticality analyses for PWR fuel assemblies are typically performed with water containing soluble boron. The critical benchmark set used for such a system may not be appropriate for analyses of fuel in unborated water.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-14 Revise Section A.6.5.2.5 to clarify the use of mixed-oxide fuel (MOX) experiments for benchmarking fresh UO_2 fuel criticality evaluations.

Table A.6.5.2-17, “USL Determination for Criticality Analysis,” for the NUHOMS-69BTH, contains a determination of USL-1 based on plutonium enrichment, indicating that some of the critical experiments used in the benchmarking analysis were with MOX fuel. The benchmarking analysis section in the text associated with this table should include a discussion of how it was determined that these MOX fuel experiments were applicable to a low-enriched, fresh UO_2 system.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-15 Revise Section A.6.5.3.2 to clarify the amount of credit assumed for the boron loading in the borated stainless steel poison rodlet used for criticality control in certain configurations of the NUHOMS-24PT4 DSC.

This section states that 64% credit is taken for the ^{10}B content in the B_4C-filled stainless steel tube variation of the poison rodlet, but does not state what amount of credit is taken for the borated stainless steel variation. Note that NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," states that the packaging model for the criticality evaluation should generally consider no more than 75% of the specified minimum neutron poison concentrations. NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," provides applicable acceptance testing criteria for taking greater than 75% credit for minimum neutron absorber concentrations.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-16 For DSC configurations with failed fuel cans or damaged fuel top and bottom end caps, revise the criticality evaluation to consider the potential for differential water draining or filling rates to result in preferential flooding of the system.

The screened design of the failed fuel cans used for failed fuel configurations, or of the damaged fuel top and bottom end caps, have the potential for slower water draining and filling during such operations in the DSC. This could result in a preferential flooding scenario, where water is inside the failed fuel cans or damaged fuel locations and not in the intact fuel locations, or vice versa. Such a situation should be evaluated in the criticality analysis to ensure it is not more reactive than the fully flooded scenario.

The applicant shall evaluate the most reactive credible configuration consistent with the chemical and physical form of the material.

This information is required by the staff to determine compliance with 10 CFR 71.55(b) and (e).

6-17 Revise the criticality evaluation to clarify if fuel assembly guide tubes are modeled, and if so provide limiting dimensions in the fuel specifications in Appendix A.1.

It is not clear from the discussion in the criticality analyses for the various DSCs if the fuel assembly guide tubes are modeled or not. The most reactive condition would be without guide tubes, as this would allow more water in the fuel envelope. If the guide tubes are modeled, then the limiting dimensions with respect to criticality (i.e., tube thickness, inner and outer diameter) should be included in the fuel specifications in Appendix A.1.

The applicant shall evaluate the most reactive credible configuration consistent with the chemical and physical form of the material.

This information is required by the staff to determine compliance with 10 CFR 71.55(b) and (e).

6-18 Revise the criticality analysis of failed fuel in the NUHOMS-24PTH DSC in Appendix A.6.5.5 to consider potential movement of fuel particles above the neutron absorber plates in the top of the failed fuel can.

The evaluation of failed fuel in the NUHOMS-61BTH DSC in Appendix A.6.5.1 considers a 16-inch axial region of fuel outside of the basket. This model accounts for a space where loose rods from compartments with failed fuel may slide into and increase the reactivity of the system due to the lack of neutron absorbers in this region. A similar analysis should be performed for any DSC which contains failed fuel, if such a configuration is possible. Also, since the criticality analysis for this DSC includes credit for burnup, evaluation of low burnup regions of the failed fuel moving axially within the failed fuel can should be evaluated.

The applicant shall evaluate the most reactive credible configuration consistent with the chemical and physical form of the material.

This information is required by the staff to determine compliance with 10 CFR 71.55(b) and (e).

6-19 Revise the criticality analysis for each DSC to clarify the “Least Material Condition” used in the model, specifically with respect to total aluminum and poison plate thickness, and stainless steel strip thickness, where applicable.

Although the criticality analysis for each DSC has an evaluation of poison plate thickness effect on reactivity, there is no evaluation of variation of k_{eff} with total paired poison and aluminum plate thickness. Similarly, there is no evaluation of k_{eff} variation with stainless steel strap thickness, where it exists in a particular DSC design. Both of these materials exist between fuel assemblies in the DSC basket, and minimizing their dimension would have the effect of bringing fuel assemblies closer together, which has been demonstrated to increase system reactivity.

This information is required by the staff to determine compliance with 10 CFR 71.55.

6-20 Refer to Proprietary Enclosure.

6-21 Revise Figure A.6.5.7-1 to correct an apparent typographical error and to verify that there is at least one poison plate between each pair of assemblies in the NUHOMS-37PTH DSC.

This Figure shows a fuel compartment in the west half of the basket apparently missing a set of poison plates. Correct the figure and ensure that there is a poison plate between each set of fuel assemblies in the basket.

This information is required by the staff to determine compliance with 10 CFR 71.55 and 71.59.

6-22 Refer to Proprietary Enclosure.

6-23 Refer to Proprietary Enclosure.

6-24 Refer to Proprietary Enclosure.

6-25 Refer to Proprietary Enclosure.

6-26 Refer to Proprietary Enclosure.

6-27 Refer to Proprietary Enclosure.

6-28 Refer to Proprietary Enclosure.

6-29 Refer to Proprietary Enclosure.

6-30 Refer to Proprietary Enclosure.

6-31 Refer to Proprietary Enclosure.

6-32 Refer to Proprietary Enclosure.

6-33 Refer to Proprietary Enclosure.

6-34 Refer to Proprietary Enclosure.

6-35 Refer to Proprietary Enclosure.

Chapter 7 – Package Operations

7-1 Clarify step 5 of Section A.7.1.1 and step 8 of Section A.7.1.2.

The applicant stated in Section A.4.1.4 of the application that the seals used for the penetrations are either *fluorocarbon elastomer O-rings* or *metallic O-rings*. Step 5 of Section A.7.1.1 for cask loading preparation, and step 8 of Section A.7.1.2 for cask wet loading state that the O-ring seals shall be discarded after each use. The applicant should clarify that this requirement of discarding O-rings applies to both fluorocarbon elastomer and metallic O-rings, or only to metallic O-rings.

This information is required by the staff to determine compliance with 10 CFR 71.43(f) and 71.87.

7-2 Check or correct the typo in step 20 of Section A.7.4.1.

The applicant should check the typo of 7×10^{-3} ref-cm^3/s or correct it to 1×10^{-7} ref-cm^3/s in step 20 of Section A.7.4.1 *Assembly Verification Leakage Testing of the NUHOMS MP197HB Cask Containment Boundary.*

This information is required by the staff to determine compliance with 10 CFR 71.35 and 71.87.

7-3 Describe in Appendices A.7.7 of the application the special control to prevent the debris from entering the package/DSC annulus when unloading the DSC to the fuel pool.

The applicant states that it will “take precautions to prevent debris from entering the cask/DSC annulus” in each step 6 of Appendices A.7.7 for unloading a DSC (e.g., 24PT4, 32PT, 24PTH, 32PTH1, 37PTH, 61BT, 61BTH, or 69BTH) to the fuel pool. The applicant is required to specify special controls needed to prevent the debris from entering the cask/DSC annulus through the vent port or the siphon port when unloading the DSC to the fuel pool.

This information is required by the staff to determine compliance with 10 CFR 71.35(c).

7-4 Explain why “taking precautions to prevent debris from entering the cask/DSC annulus” is not mentioned/required in Appendix A.7.7.4.4 for unloading 32PTH DSC to a fuel pool.

The applicant should explain why “taking precautions to prevent the debris from entering the cask/DSC annulus” in step 6 is mentioned/required in Appendices for unloading 24PT4, 32PT, 24PTH, 32PTH1, 37PTH, 61BT, 61BTH, and 69BTH DSCs to a fuel pool, but not mentioned/required in Appendix A.7.7.4.4 for unloading 32PTH DSC to the fuel pool.

This information is required by the staff to determine compliance with 10 CFR 71.35(c).

7-5 Clarify why the two additional actions described in step 5 of Appendix A.7.7.4.4 for unloading the 32PTH DSC to the fuel pool are not required for unloading other types of DSCs to the fuel pool.

Compared to 24PT4, 32PT, 24PTH, 32PTH1, 37PTH, and 61BT DSCs, the additional actions of (a) the vent cavity gas may include steam, water, and radioactive material, and should be routed accordingly and (b) monitor the vent pressure and regulate the water fill rate to ensure that the pressure does not exceed 15 psig are required when unloading a 32PTH DSC to a fuel pool (step 5 of Appendix A.7.7.4.4). The applicant is required to explain why these two additional actions described above are only required for 32PTH DSC, but are not for other types of DSCs.

This information is required by the staff to determine compliance with 10 CFR 71.35(c).

7-6 Provide specific instruction and requirements for verification of the fuel assemblies to be transported.

On page A.7-2, step 13 of the Operating Procedures for the package instructs the users to verify that the fuel assemblies selected for loading meet the criteria of the applicable fuel specification as listed in Table A.7-2. However, it was not clear how the selected fuel assemblies should be verified. Correctly loading the package is critical in that any mistakes could potentially impact the safety of the package. The applicant needs to provide specific instructions for fuel qualification and verification procedures and requirements for the users to develop procedures that meet the quality assurance requirements of Subpart H of 10 CFR Part 71.

This information is required by the staff to determine compliance with 10 CFR 71.89 and 10 CFR Part 71, Subpart H.

7-7 Revise the Unloading Operating Procedures to instruct the users to proceed to the abnormal operating model when airborne radioactive particles are found in the cask/DSC annulus gas sample.

Page A.7-19 of the application states: "If airborne radioactive particles are found in the cask/DSC annulus gas sample, appropriate filters should be used to preclude the uncontrolled release of any potential airborne radioactive particles during unloading. This will protect both personnel and the operations area from potential contamination. Personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program." However, when airborne radioactive particles are found in the cask/DSC annulus gas sample, cladding might have already breached. If cladding failure has occurred, the situation is beyond the need for protecting the operating personnel and the operations area from potential contamination. The unloading process shall turn into an abnormal operating model, such as unloading the package in a hot cell, to prevent airborne radioactive particles from releasing to the outside environment. The applicant needs to reconsider revising the Unloading Operating Procedures to instruct the users to proceed to abnormal operating model when airborne radioactive particles are found in the cask/DSC annulus gas sample.

This information is required by the staff to determine compliance with 10 CFR 71.89.