November 19, 2010

Mr. Donis Shaw
Licensing Manager
Transnuclear, Inc.
7135 Minstrel Way, Ste. 300
Columbia, MD 21045

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. NUHOMS-MP-197 PACKAGING, DOCKET NO. 71-9302

Dear Mr. Shaw:

By letter dated April 14, 2009, you submitted an application for revision to Certificate of Compliance (CoC) No. 9302 for the Model No. NUHOMS-MP-197 packaging (MP-197). The application proposes to add a modified version of the MP-197, designated the MP-197HB, and includes various NUHOMS dry shielded canisters as authorized payloads in the MP-197HB transport package. This application also proposes to update the design in order to be assigned a package identification number of B(U)F-96. Our established schedule provides a CoC issuance date of April 6, 2011.

The enclosed Request for Additional Information (RAI) identifies information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in order to complete its review of the application. Each individual RAI describes information needed by the staff for it to complete its review of the application to determine whether the applicant has demonstrated compliance with the regulatory requirements. NRC is requesting responses to these RAIs by February 28, 2011. Please inform us at your earliest convenience if you are not able to provide this information on the requested, and no later than February 14, 2011. To assist us in re-scheduling your review, you should also include a new proposed submittal date and the reasons for the delay. This letter confirms our phone call on November 9, 2010.

If you have any questions regarding this matter, I may be contacted at (301) 492-3175 or you may contact Chris Staab of my staff at (301) 492-3321.

Sincerely,

/RA/

Robert Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9302
TAC No. L24336
Enclosure: Request for Additional Information

cc: E. Redmond, Nuclear Energy Institute

Mr. Donis Shaw
Licensing Manager
Transnuclear, Inc.
7135 Minstrel Way, Ste. 300
Columbia, MD 21045

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Please reference Docket No. 71-9302 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contacted at (301) 492-3175 or you may contact Chris Staab of my staff at (301) 492-3321.

Sincerely,

/RA/
Robert Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
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Docket No. 71-9302
TAC No. L24336
Enclosure: Request for Additional Information
cc: E. Redmond, Nuclear Energy Institute
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Transnuclear, Inc.

Docket No. 71-9302

Request for Additional Information

Model No. NUHOMS-MP-197

By letter dated April 14, 2009, you submitted an application for revision to Certificate of Compliance (CoC) No. 9302 for the Model No. NUHOMS-MP-197 packaging (MP-197). Staff reviewed the application and transmitted initial RAIs on December 2, 2009. Staff reviewed TN's responses submitted on August 4, 2010. This second Request for Additional Information (RAI) identifies information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application. Each individual RAI describes information needed by the staff for it to complete its review of the application to determine whether the applicant has demonstrated compliance with the regulatory requirements.

2.0 Structural and Materials

2-1 Provide information on how the spacers between fuel and Dry Shielded Canisters (DSC) will be installed for DSCs loaded under storage licenses and/or other transport CoCs.

Table A.2.13.14-2 specifies the various spacer heights ranges to limit fuel-to-DSC gaps which affect Hypothetical Accident Conditions (HAC) drop considerations. However, it is not clear to the staff how these spacers will be implemented for DSCs loaded previously regarding considerations for transportation in the MP-197HB.

This information is needed to verify compliance with 10 CFR 71.73(c)(1).

2-2 Clarify the approach taken regarding fuel cladding integrity under HAC conditions, given the status of the TN-40 application cited in response to RAI 2-30 and 2-38.

The TN-40 review has not been finalized. Therefore, open questions regarding the methodology at this time are still open for the MP-197HB as well.

This information is needed to verify compliance with 10 CFR 71.55(e)(1).

2-3 Update the relevant results tables in the Safety Analysis Report (SAR).

Updated results from the revised structural analyses, applicable to various tables throughout the application, (e.g., crush depths in Table A.2.13.12-8), are not provided in the updated application.

This information is needed to verify compliance with 10 CFR 71.71(c)(7) and 71.73(c)(1).

2-4 Clarify the magnitude of g-loads for the HAC side drop basket analysis for the 32PTH1 basket.

Table A.2.13.8.1 lists the g-load for the 32PTH1 as 55 g, while for the 32PTH it is 75 g. However, Tables A.2.13.8-24 through A.2.13.8-26 list very similar results for maximum

stresses for the two similar baskets. It is unlikely that such a different loading would produce such similar effects.

This information is needed to verify compliance with 10 CFR 71.73(c)(1).

2-5 Remove the use of foreign code material as an alternative to American Society for Testing and Materials (ASTM), or American Society of Mechanical Engineers (ASME) code material for cask systems to be used in the United States. Specify exact alternative code material or properties of the code materials that have to meet specifications of the US code material such as yields, elongation, ultimate strength, etc.

A number of the drawings (notes and/or materials lists) include a statement that foreign specifications can be used.

This information is needed to satisfy the requirements of 10 CFR 71.111.

2-6 Reduce the maximum allowable assembly burnup for the 69BHT canister to 62.5 GWd/MTU peak rod average in Table A1.4.9-4, or provide justifications for the use of the temperatures in Interim Staff Guidance – 11 (ISG-11), Revision 3, "Cladding Considerations for Transportation and Storage of Spent Fuel."

Staff asked this same RAI in the 1st round (2-19). The applicant's justification was that there would be longer cooling times and the fast neutrons were the same, thus there was no thermal issue. The maximum assembly burnup allowable, if the temperature limits delineated in ISG-11, Revision 3, are invoked, is 62.5 GWd/MTU. This is based on creep and hydriding considerations. The burnup will affect the cladding oxide thickness, the cladding hydrogen content, and the fission gas release. All these will affect the cladding stress, and potentially the ductility, and creep. There is insufficient data on these parameters for the various fuels and claddings to substantiate these effects for reactor operations, hence the knowledge of these parameters for use in cladding stress calculations and creep calculations is insufficient for transportation applications. Table A.1.2.9-4 lists burnups as high as 70 GWd/MTU.

This information is needed to meet the requirements of 10 CFR 71.55(d)(1) and (2).

2-7 Show that the mechanical properties calculated using the Geelhood and Beyer correlations in SAR Section A2.13.11.1, apply to cladding with radial hydrides. Compare the ductility of cladding with radial hydrides with the stress imposed during a side drop, i.e., hoop plus crush stress.

Most of the DSCs have either been approved for storage of high burnup fuel or are asking for approval to transport high burnup fuel. The presence of radial hydrides is expected, since 1) the cladding will have had to undergo a drying cycle, and 2) most cladding, other than M5, has both hydrogen contents >200 wppm, with hoop stresses increased by the larger increase in fission gases at high burnup.

The applicant drew on the results of two papers, one by Chu and another by Aomi, to attempt to make the case that the formation of radial hydrides did little to decrease the

yield and ultimate strength of the cladding in the axial direction and only decreased them by 20% in the hoop direction. The staff accepts the conclusion for the axial direction , but rejects the conclusions for the hoop direction. Chu's work was conducted at applicable temperatures, stresses and hydrogen contents but only used charged unirradiated samples. The morphology of the initially charged samples (Chu, Fig. 4a) is completely different from the hydride morphology that occurs in irradiated cladding, and includes the artifacts of the circumferential hydrides in the body of the cladding, which if not dissolved, form a barrier to the propagation of radial hydrides and lowering of the yield stress. As a result, the results and conclusions from Chu's work cannot be generalized to irradiated cladding. Aomi uses burnups up to 59 GWd/MTU but conducts hydride orientation only up to a temperature of 300ºC and stresses only up to 70 MPa. Both conditions are too low to place a significant amount of hydrides into the radial direction although they form a preponderance of the hydrides in the body of the cladding. In addition, the cooling rate of 30 C/hr is too fast to allow for significant growth of the hydrides. Shorter hydrides have a lower chance to affect the yield strength. As a result, the reduced yield strengths calculated from these two papers, while only slightly lower than that calculated by Geelhood and Beyer, are rejected by the staff.

The proposed reduced yield stress was compared to the hoop stress in the cladding. The radial hydrides are important during the side drop where there are both hoop and crush stresses. Both must be accounted for when determining the stability of the cladding.

This same argument has been made by the applicant in the past and was rejected by the staff.

This information is needed to meet the requirements of 10 CFR 71.55(b)(1).

2-8 Amend the SAR referring to the neutron absorbing materials used for criticality control to be consistent with recent amendments (e.g., CoC No. 1030, Amendment 1).

a) Modify SAR Section A.8.1.7.1. The direct chill of an Al-Ti-B melt could precipitate other phases, e.g., AlB_{12}. The current language in the Technical Specifications (TS) would not permit such phases to act as the principle neutron absorbing material and may create a non-compliance issue with the TS, as the TS relate to transportation.

b) Specify In SAR Section A.8.1.7.2 that the boron carbide particles shall have an average size of 40 microns or less. Particle sizes may be determined by mesh size. The current language, which is that boron carbide particles "typically have an average size…," is vague and open to interpretation.

c) Specify In SAR Section A.8.1.7.3 that at least 80% by weight of the boron carbide particles should be less than 200 microns in size. This requirement is to control the final particle size of the boron carbide in the Boral plates after rolling, which affects neutron streaming.

d) Add a discussion of the thermal conductivity qualification testing of the neutron absorbers to the SAR. The thermal conductivity of the neutron absorbers influences the temperatures distribution of all the components of the entire package, and hence the

component performance. No qualification testing protocol was included in this application, consistent with previous applications.

e) Describe in SAR Section A.8.1.7.7.3.1, the delamination testing protocol. The delamination testing mentioned in the SAR is not described for the staff to evaluate. Delamination of clad neutron absorbers during drying may lead to unanalyzed geometries or deformations making retrievability of the fuel problematic.

This information is needed to determine compliance with 10 CFR 71.55(b).

2-9 Revise the SAR to add a plan to ensure, for any DSC that has spent an extended time in storage, that the contents and DSC itself meet all the requirements in the CoC. This plan should include inspections to obtain data or analysis to support: 1) the mechanical and thermal properties of the components of the DSCs related to safety, and 2) the contents, have not degraded during the storage period. Provide evidence that removal of the DSC from the storage overpack will not damage the DSC, and impact safety.

a) Describe how loading records will ensure that DSC damage is not done during the extraction process. Clarify what "appropriate evaluations" as indicated in the initial response, will be conducted.

b) Provide evidence that high burnup fuel in storage for 20 or more years will remain in its analyzed condition. To the staff's knowledge, the analysis on the condition of high burnup fuel after storage has not been experimentally confirmed.

c) Provide a discussion of how Time Limited Aging Analysis (TLAA) will be performed, and describe the aging management program that will be incorporated into the storage license to assure the condition of the basket, neutron absorbers, and fuel contents. Take into accounts comments in RAI M.3.

d) Describe how the applicant will ensure that if there is a storage license renewal beyond 20 years, the aging management plan will include periodic in-service inspections of the accessible canister surfaces as indicated at the end of the 1st round RAI response. Indicate the type of in-service inspections that will be required and what kind of damage will they look for. This section of the application is unique, as it will link a storage renewal application with a future transportation application for the first time.

All the mechanical and thermal properties of the materials of construction of the DSC used in this Part 71 analysis are for pristine materials. Many of the DSCs were constructed and loaded many years ago and are in storage. The materials properties used for the evaluation of the safety systems and contents of the DSCs that have already been in storage service must be representative of the conditions at the time of transport, not at the time of the loading of the DSC. A number of the issues were satisfactorily addressed in the initial RAI response. The issues cited above were not sufficiently resolved.

This information is needed to meet any thermal, shielding, criticality, or structural requirements of 10 CFR Part 71 where the materials properties are integral to the response of the system.

2-10 Describe the methodology used to determine the size of the gap between the fuel assembly and the DSC.

The gaps between the fuel assemblies and the DSC are given in Tables A.2.13.14-1 and 2. The size of the gap will depend on the dimensions of the DSC and the expected growth of the assembly, which depends on the burnup during irradiation. Without the expected growth of the assembly being stated, there is no way for the staff to determine if the stated gaps are the most limiting for determining the deformation of the assemblies during a hypothetical accident.

This information is necessary to meet the requirements of 10 CFR 71.73(c)(1).

5.0 Shielding

5-1 Demonstrate that the neutron shield will retain a uniform layer for the purpose of effective neutron shielding the under hypothetical accident conditions.

In its response to RAI 5-4, the applicant states: "Tests have shown that the neutron shielding material retains more than 60% of its principal contents (hydrogen, boron) following design basis fire accident…" In its response to RAI 5-6, the applicant further states: "The results of the fire tests performed during the testing of the resin indicate that a small layer gets charred under direct fire exposure and this layer is likely to be much smaller when there is no direct exposure to fire." The applicant further states: "The charring is localized …"

The staff's concerns, however, were not only with how much of the content would be retained or whether the tubes will detach or not. Another chief concern is if the neutron shielding material will be able to retain a uniform neutron shielding layer as assumed in the shielding model. A localized charring and further cracking of the resin may create neutron streaming paths that result in "localized" hot spots that void the effectiveness of the neutron shielding layer. The applicant is requested to demonstrate that the neutron shield will retain a uniform layer for the purpose of effective neutron shielding.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.51.

5-2 Provide shielding analyses that explicitly model the walls of the aluminum tubes that contain the neutron shield resin.

In its response to first round RAI 5-1, the applicant explained in detail the structural design of the resin tubes. From these explanations, it appears that the walls between two adjacent tubes will form an approximately 0.24 inch aluminum wall. Because aluminum has a much lower neutron absorption cross section than boron and a much lower slowing down power than the hydrogen and oxygen in the resin, these aluminum layers between the adjacent resin tubes may result in neutron streaming paths. It was not clear how these aluminum tube walls were treated in shielding analyses of the MP-197 packages. The applicant is requested to provide information on how the aluminum

tube walls were treated in the shielding models with adequate justification.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-3 Demonstrate via calculations that the MP197HB cask loaded with the 69BTH DSC containing design basis BWR fuel assemblies bounds all contents.

The staff requested, in the original RAI 5-3, which was transmitted to the applicant on December 2, 2009, that the applicant demonstrate that a MP197HB cask loaded with the 69BTH DSC containing design basis BWR fuel assemblies bounds all contents. In its response to the RAI, the applicant explained the bases on which it determined that the 69BTH DSC containing design basis BWR fuel assemblies bounds all contents with respect to the shielding safety analysis. However, the response used mainly qualitative discussion rather than calculations, and the discussions focused solely on a fuel load of a single fuel assembly. From the perspective of shielding safety analysis, the source terms (both neutron and gamma) are the determining factors for bounding design basis fuel. As such, the combination of total fuel quantity, burnup, cooling time, and initial enrichment are the major parameters that affect the source terms. The applicant is requested to demonstrate that the selected design basis content bounds all of the nine different cask designs, including those that contain damaged fuel and/or non-fuel hardware.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-4 Demonstrate via calculations that the source terms calculated using SAS2H are conservative.

In its response to the staff's RAI 5-8, transmitted to the applicant on December 2, 2009, and subsequent RSI 5-1, the applicant revised the SAR to provide qualitative sensitivity analyses to justify that the source terms calculated using SAS2H are conservative. However, the applicant did not provide any quantitative analyses to demonstrate that the calculated source terms are conservative and the calculated dose rates will not exceed the regulatory limits, e.g., 10 mrem/hr at 2 meters from the surface of the cask or enclosure. In fact, Table A.5-1a of revision 8 of the SAR shows that the maximum dose rate at two meters from the assumed railcar is 9.4 mrem/hr. The margin for safety is only 6%. Review of publications on SA2H validation against experimental data indicates that SAS2H underpredicts the concentrations of many isotopes that are important to shielding by over 10%. Given this fact, the applicant is requested to demonstrate with code benchmark results that the accumulated errors in the dose rates, including errors and uncertainties in source terms and dose rate calculations using the MCNP code will not exceed 6%. An analysis of the uncertainties associated with Monte Carlo method for shielding calculations will be helpful to the staff for determining if the cask design meets 10 CFR 71.47 and 71.51 requirements, given the fact that MCNP is a Monte Carlo method particle transport calculation code.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-5 Provide data to demonstrate that the uncertainty of SAS2H calculated gamma and neutron sources is 10% and 5%, respectively.

In its response to the staff's, RSI 5-1, transmitted to the applicant on June 17, 2010, the applicant states that the uncertainty of the SAS2H calculated gamma and neutron sources is 10% and 5%, respectively. In addition, the applicant states: "A more direct validation of source term calculations is to measure the dose rate from these sources. Numerous measurements from various spent fuel storage and transportation systems have indicated that the measured dose rates are typically lower than the calculated dose rates." The staff's review of the isotopes estimated by SA2H that are important to shielding indicates that many of them are underestimated by a large percentage. For example, the isotopic concentration for Cm-244, the major neutron emitter and high energy gamma emitter (average energy = 7.25 MeV and 9 MeV groups), is underestimated by as much as 21% and 12.5% for the TMI and REBUS samples, respectively. For the design basis fuel, the major gamma emitters, Cs-134 (average energy = 0.9 MeV group) were underestimated by 23%, 34.6% for two TMI samples, 20.8% for the Calvert Cliffs sample, 19.7% for the Takahama-3 sample, and 14.4% for the ARIANE sample, respectively. The applicant also referenced NUREG/CR-6701 and ORNL/TM-13315 for validation of SAS2H at burnups up to 75 GWd/MTU. The staff reviewed these publications and found they are not applicable to the TN MP197 package because (1) the only sample with burnup of 73 GWd/MTU used in NUREG/CR-6701 was taken from a reconstituted fuel assembly and (2) the maximum burnup for all benchmark data in ORNL/TM-13315 is 34 GWd/MTU. The staff also consulted the appropriate authors at Oak Ridge National Laboratory; the answer is stated that these reports were never intended to be used as basis to support that the validated range of SAS2H is up to 75 GWd/MTU. In addition, NUREG/CR-6701 and a presentation on CASMO validation presented during a recent Electric Power Research Institute Workshop on Light Water Reactor Physics Methods shows that the uncertainties of the calculated isotopic concentrations increase as the fuel assembly burnup increases. However, in the revised SAR and responses to the RAIs, no quantitative data were provided on the relationship between these uncertainties and fuel burnup. For these reasons, the applicant is requested to demonstrate that the gamma and neutron sources calculated for the MP-197 package shielding analysis are within 10% and 5%, respectively, for isotopes that are important to shielding. Additionally, the applicant should provide documentation to support the validation range of the SAS2H code up to 75 GWd/MTU.

The applicant is recommended to provide supporting data that the staff may not be aware of and make justification for the accuracy of the SAS2H source term calculation for high burnup fuels.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-6 Provide justification for why the neutron sources, as a function of the axial burnup profile, were not proportional to the fourth power of burnup.

Table A.5-15 of the revised SAR provides BWR axial peaking factors for neutron and gamma sources. A simple calculation of the peaking factor for neutron source along the axial direction seems to indicate that the neutron source is not proportional to the fourth power of local burnup. In comparison, Table A.5-16 shows that the neutron source for PWR fuel is proportional to the fourth power of burnup. On page A.5-7a of the revised SAR, the applicant states: "The ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup is

1.326 and 1.152 for BWR and PWR assemblies, respectively." However, it is not clear how these numbers are derived and what the bases are for these conclusions. Furthermore, the ratio for BWR fuel was used in Table A.5.-15, but not used for the PWR source determination in the same table. It is not clear why the same rule was not applied to both BWR and PWR fuel. The applicant is requested to provide justification for why the source term provided in this table does not follow the physical law.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-7 Demonstrate that dose rates under normal conditions of transport will not change significantly for fuel with natural uranium blankets, even though the source terms in the enriched fuel regions increases significantly for such fuel.

In its RAI 5-19, transmitted to the applicant on December 2, 2009, the staff requested the applicant explain how the natural uranium blankets were treated in the source term calculation and cask dose rate calculations. In the applicant's response, the applicant concluded that the natural uranium blankets caused the source terms to increase significantly, but the impact to dose rates of the cask is insignificant. This conclusion, however, appears to be in conflict with the basis of the "response function" method used for fuel qualification. The fundamental assumption of the "response function" method is that there exists a linear relationship between the outside dose rates and source terms in the cask. The applicant is requested to demonstrate its conclusion with detailed calculations.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-8 Pertinent to the response function method, explain why the response function calculated with a cask under normal conditions of transport can be used to determine the design basis HAC source terms.

In its response to the staff's RAI 5-18, transmitted to the applicant on December 2, 2009, and subsequent RSI 5-3, and the follow-up conference call on the RSIs, the applicant stated that the response function was not used for shielding analysis for the cask under hypothetical accident conditions. However, on page A.5-26 of Revision 8 of the SAR, the applicant states: "In summary, the response functions are calculated using the NCT shielding configuration models for the fuel qualification (to determine acceptable BECT combinations for various payloads). The neutron dose rates calculated using the response functions employed to determine the design basis HAC source terms to ensure that the fuel qualification methodology also ensures that the dose rates are within acceptable limits under HAC." There appears to be an inconsistency between these responses to the same question. The applicant is requested to explain exactly how the response functions calculated under normal conditions of transport were used in the safety analyses for the cask under hypothetical accident conditions and provide justification for such uses. Review of original RAI 5-18 and RSI-3 may be helpful to address the staff's concerns.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-9 Pertinent to the response function method, provide:

a) Detailed information on the definition of the response function, including the mathematical formulation and analytical derivation of the equations,

b) Technical basis for this method, i.e., how and why this approach produces reliable and accurate results, and

c) Validation and verification of the method, or

d) Publications and references that have demonstrated the validity of the methodology.

In response to the staff's RAI 5-18, the applicant revised the SAR to add some discussions on the response function method. However, the response to RAI 5-18 and the revised SAR do not address the above questions. Specifically, the response to the RAI did not provide any discussion on why this method is valid and reliable for casks loaded with various contents. The response did not provide any discussion on why the complicated particle transport problem can be simplified as a simple arithmetic problem so that the fuel qualification can be determined using a simple spreadsheet data table. The fundamental question is to demonstrate that the particle transport problem related to shielding calculations of the cask is independent of the material composition in the cask fuel region. The applicant is requested to (1) provide detailed information on the definition of the response function, including the mathematical formulation and analytical derivation of the equations, (2) technical basis for this method, i.e., how and why this approach works, and (3) validation and verification of the method or, (4) publications and references that have demonstrated the validity of the methodology.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47.

5-10 Explain exactly how the response function was used for determination of the design basis HAC source terms and provide justification for why such an approach is acceptable.

On page A.5-26 of the revised SAR, the applicant states: "The neutron dose rates calculated using the response functions are employed to determine the design basis HAC source terms to ensure that the fuel qualification methodology also ensures that the dose rates are within acceptable limits under HAC." In its response to RSI-3 of the acceptance review of the response to the RAI, the applicant states: "From the NCT dose rate results, the source with largest neutron dose rate fraction was selected as the bounding HAC source. This occurs for a burnup of 70 GWd/MTU and enrichment of 4.3% U-235, where the neutron fraction of the total NCT dose rate is 85%." However, it is not clear how the source giving highest neutron fraction of NCT dose rate will necessarily be that which will produce the highest neutron dose rate under HAC. In particular, if the neutron shielding material is highly effective, the neutron dose rate will be small for all practical purposes. Under HAC, both gamma shield and neutron shield will lose some shielding capabilities. There is no clear nexus between these two scenarios. The applicant is requested to clarify exactly how the response function was used for determination of the design basis HAC source terms and provide justification for

why such an approach is acceptable. A detailed description for the two types of base models, as mentioned in the last paragraph of page A.5-12, for both NCT and HAC, and the input/output files will be helpful for the staff to make the determination that the package meets the regulatory requirements.

In addition, the applicant is requested to provide code validation results that demonstrate that SAS2H is validated to 70 GWd/MTU burnup.

Also, the applicant should demonstrate the need for packages for spent fuel with burnups up to 70 GWd/MTU, given the fact that the Office of Nuclear Reactor Regulations limits maximum assembly burnup to 62.5 GWd/MTU.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.51.

5-11 Demonstrate that SAS2H can model fuel assemblies with low enriched or natural uranium blankets.

On page A.5-33 of the revised SAR, the applicant states that the use of axial blankets could be explicitly accounted for in the source term calculation by running SAS2H for each of the axial regions of the core. However, it was not clear how the power density and depletion environment were modeled in the source term calculations given the behavior of the neutron flux in the interface zones of the normal enrichment zone and low enrichment zone. In addition, the neutron flux in the low enrichment blanketed core is severely skewed. A model with local burnup may not be able to capture the flux distortion along the axial direction of the fuel assembly, and there is no axial burnup peaking factor that covers the fuel assemblies with natural uranium or lightly enriched uranium blankets. It was also not clear how the 3D effects were treated in a supercell model code such as SAS2H. The applicant is requested to demonstrate that SAS2H can model fuel assemblies with low enriched or natural uranium blankets.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47.

5-12 Provide justification for the conclusion that the extra conservatisms discussed in Section A.5.5.7 of the revised SAR will bound the uncertainties associated with the SAS2H source term calculations.

Section A.5.5.7 of the SAR discusses the uncertainties associated with the source terms calculated by the SAS2H code. However, some of the conservatisms discussed may not actually be present. For example, the fact that cobalt-60 contents for newer fuel designs are lower than in older designs may not be considered as conservatism, because the CoC, once approved, will not prohibit users from loading a cask with high-cobalt-60 fuel assemblies. Unless the CoC explicitly prohibits high cobalt-60 fuel from the authorized contents, this conservatism is not available as additional conservatism to dose rates. The same is true for other parameters listed on page A.5-34b of the revised SAR. The applicant is requested to provide justification for the conclusion that the listed conservatisms would compensate for the uncertainties of the SAS2H source term calculations.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

5-13 Correct the statement on page A.5-4a regarding the dose rate limit to the package accessible surface.

On page A.5-4a of the revised SAR, the applicant states: "External dose rate at any point on the outer accessible surface of the package under normal conditions: 1000 mrem/hr (max)." This is inaccurate because 10 CFR 71.47 requires the package surface dose rate to be 1000 mrem/hr only when the package is enclosed in a transport vehicle. The applicant is requested to revise this statement to make it consistent with 10 CFR 71.47.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47.

5-14 Verify the reference to Table 5.2-7 on page A.5-7 and make corrections as necessary.

Page A.5-7 of the revised SAR references Table A.5-7. The staff was unable to find this table. The applicant is requested to provide information on the location of this table.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47 and 71.51.

6.0 Criticality

Unless otherwise stated, the following information is required in order to ensure that the Model No. TN NUHOMS-MP-197HB package will meet the criticality safety requirements of 10 CFR 71.55 and 71.59, and that fissile material is packaged as if unknown properties have credible values that will cause the maximum neutron multiplication, per 10 CFR 71.83.

6-1 Revise Section A.6.2.4.A of the application, *Axial Burnup Distribution*, to clarify the number and burnup ranges of the axial profiles used in the criticality analysis.

The last paragraph in Section A.6.2.4.A states that the fourth profile is used in the criticality analysis for burnups greater than 42 GWd/MTU and greater than 4.0 weight percent initial enrichment. The fourth profile given in Table A.6-3 is for greater than 38 GWd/MTU, and does not state an initial enrichment limit. The application should be revised to clarify the burnup and enrichment limits for the fourth profile used in the criticality analysis.

6-2 Revise Section A.6.2.7.4, *Reactivity Effect of Specific Power During Depletion*, to either add a bias correction for burnups lower than 35 GWd/MTU, or justify that one is not necessary.

Section A.6.2.7.4 states that a bias of 0.0035 is appropriate for burnups lower than 35 GWd/MTU, as shown in Table A.6-36. This section then states that bias is not necessary since the SAS2H calculations are benchmarked to the DARWIN code with the same specific power. This conclusion is not supported by the text of this paragraph.

Revise this section to either use a more appropriate bounding specific power during depletion at low burnups, or to apply the 0.0035 bias determined for the burnup range.

6-3 Revise Section A.6.3.2.1, *SAS2H Fuel Depletion Benchmark Evaluation*, to justify using best-estimate correction factors without adjusting for uncertainty. Additionally, revise this section to justify crediting absorber isotopes with small numbers of radiochemical assay samples for depletion code validation.

Table A.6-10 of the application contains both the mean correction factor and standard deviation, but the standard deviation does not appear to be included to adjust the final isotopic correction factors for uncertainty. The isotopic correction factors should be adjusted to include the uncertainty, or this section should be revised to justify not doing so. Additionally, some credited isotopes have small numbers (i.e., less than 30) of radiochemical assay measurements used for depletion code validation. These small data sets should be subjected to a normality test to determine if the resulting isotopic correction factor values should be adjusted based on uncertainties determined using small-sample statistics.

6-4 Revise Section A.6.3.2.1, *SAS2H Fuel Depletion Benchmark Evaluation*, to include a trending analysis of the SAS2H calculated correction factors independent of the DARWIN scaling factor adjustment.

Table A.6-10 of the application presents single values of the TN SAS2H average correction factor, with no dependence on trending parameters important to the depletion analysis (e.g., burnup or enrichment). Section A.6.3.2.1 should be revised to include a trending analysis which demonstrates that the average correction factors do not have significant trends, or they should be adjusted to account for trends.

6-5 Revise Section A.6.3.2.1, *SAS2H Fuel Depletion Benchmark Evaluation*, to clarify how the isotopic scaling factors from DARWIN benchmarks were calculated, and how these scaling factors were applied to the SAS2H isotopic correction factor results.

The second paragraph on page A.6-17 of this section states that “A set of correction factors, based on the average ratios (DARWIN/SAS2H) for all three assembly types, was determined and these factors are shown in Table A.6-11.” It is not clear, however, exactly how these correction factors were determined. Additionally, it is not clear how the isotopic scaling factors determined from the DARWIN calculations were used to adjust the SAS2H correction factors. This section should be revised to clarify these two issues.

Unless otherwise stated, the following information is required in order for the staff to ensure that the Model No. TN NUHOMS® MP-197 package will meet the criticality safety requirements of 10 CFR 71.55 and 10 CFR 71.59 when loaded with the contents described in the application.

6-6 Revise Section A.6.3.2.2, *CSAS25 Criticality Benchmark Evaluation*, to justify that the critical benchmarks sufficiently cover the burnup range modeled in the criticality calculations.

For applicability of the critical benchmark experiments, this section claims that the highest “credited” burnup corresponds to the lowest burned axial segment of the highest

average burnup assembly, or 26.2 GWd/MTU at the top end of a 50 GWd/MTU assembly average burnup. In fact, the analysis is "crediting" the highest burned sections of the fuel as well, which may be significantly above the 50 GWd/MTU assembly average. This section should be revised to justify that the critical benchmarks sufficiently cover the burnup range modeled in the criticality calculations.

6-7 Revise Section A.6.3.4.1.2, *Single Fresh Assembly Misload Evaluation*, to justify the basket type and fuel assembly placement chosen for this evaluation.

This section states that the NUHOMS-32PT DSC basket is used for the evaluation due to low number of poison plates, but does not demonstrate that this is the most reactive case. Provide an analysis of a similar misload in another canister (e.g., 37PTH) to demonstrate that the 32PT is the most sensitive to a misload. This section also states that the misload evaluation is performed with the misloaded assembly in periphery of basket where there is no poison plate. This section should be revised to justify that this assumption is conservative compared to placing assembly in the center of basket. Additionally, this section does not state what the enrichment of the fresh assembly used in the evaluation is. Revise the application to provide this information. Note that although the basket may be administratively limited in the enrichment of the assembly that can be loaded, there may be higher enrichment fuel available in the pool for misloading. The misload analysis should consider the maximum reactivity assembly that may be misloaded.

6-8 Revise Section A.6.3.4.1.3, *Multiple Underburned Assemblies Misload Evaluation*, to state why Yankee Rowe, San Onofre 1, Haddam Neck, Maine Yankee, Zion 1 & 2, Indian Point 1, Rancho Seco, and Trojan were excluded from the misload evaluation.

It is not clear that these assembly types are specifically excluded as allowable contents in the NUHOMS-MP-197 package.

6-9 Revise Section A.6.3.4.1.3, *Multiple Underburned Assemblies Misload Evaluation*, to justify the basket type chosen for the misload evaluation.

This section states that the NUHOMS-32PTH1 DSC Type C basket is used for the misload evaluation as it is the most common in use. This analysis should consider the basket that is likely to be most sensitive to such a misload. Note that for the single fresh assembly evaluation, the NUHOMS-32PT was considered to be the most sensitive due to the low number of poison plates. Note also that although the 32PTH1 DSC may be the most common PWR canister in use today, this may not be the case in the future as utilities transition to higher density cask systems.

6-10 Revise the application to include a reduction in the reactor record burnup value by the uncertainty in that value.

It is not clear that uncertainties in reactor record burnup values are included when evaluating whether or not candidate fuel assemblies meet the appropriate loading curve burnup value. Interim Staff Guidance-8 (ISG-8), Revision 2, *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*, recommends that the assembly burnup value to be used for loading acceptance (termed

the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing that value by its associated uncertainty.

6-11 For canister loading curves crediting 40 years of cooling, revise the application to include a transportation cooling time limit of 160 years, beyond which the package would need to be reevaluated for criticality safety prior to transportation.

ISG-8 contains a maximum cooling time credit of 40 years, due to the fact that reactivity will decrease after this point until about 100 years, and then rise again to roughly the same reactivity at about 200 years. At the time ISG-8 was published, this transportation window of between 40 to 200 years was considered sufficient to allow transportation of most spent fuel. The NRC has recently undertaken an effort to identify issues related to potential long-term storage of dual-purpose spent fuel storage casks and transportation packages, considering a potential storage time frame of 300 years or more. Given this recent focus on long-term storage, staff considers it prudent to condition spent fuel transportation package certificates of compliance to require a re-evaluation of burnup credit if fuel is in storage for a period of time in which the reactivity may be higher than when originally evaluated.

Appendix A.6.5.1 - NUHOMS-61BT and NUHOMS-61BTH DSC Criticality Evaluation

6-12 Revise the damaged fuel configuration for the NUHOMS-61BT DSC to consider rod pitch expansion and loss of rods from the lattice.

Rod pitch expansion and loss of rods from the fuel lattice was considered for every other DSC for which damaged fuel is an allowable content, and was shown to be the most reactive condition of the damaged fuel. The NUHOMS-61BT criticality analysis appears to consider only single- and double-ended rod shear in the damaged fuel criticality analysis. The application should be revised to either demonstrate that these two configurations are more reactive for the NUHOMS-61BT DSC, or revise the evaluation to consider rod pitch expansion and loss of rods from the lattice, consistent with the other DSC damaged fuel evaluations.

Appendix A.6.5.4 - NUHOMS-32PTH/32PTH1 DSC Criticality Evaluation

6-13 Revise the application to clarify the damaged fuel configuration for the WE 14x14 class fuel assemblies in the NUHOMS-32PTH/32PTH1 DSC.

Section A.6.5.4.4.2.C discusses the determination of the most reactive WE 17x17 class damaged fuel assembly, but does not include a similar evaluation of the WE 14x14 class fuel assembly. Without such an analysis, it is unclear how the WE 14x14 assembly class acceptable average initial enrichment and burnup combinations for damaged fuel, given in Table A.6.5.4-15, were determined. The application should be revised to show how damaged WE 14x14 class fuel assemblies were modeled in the criticality analysis. Note that this information may also be necessary to clarify the modeling of damaged WE 14x14 class fuel assemblies in the NUHOMS-24PTH and -37PTH DSCs.

Appendix A.6.5.5 - NUHOMS-24PTH DSC Criticality Evaluation

6-14 Revise this section of the application to clarify the number of damaged fuel assemblies allowed to be shipped in the 24PTH DSC. Additionally, clarify whether or not CE 16x16 fuel assemblies are intended as allowable contents.

Section A.6.5.5.2 of the application, *Package Fuel Loading*, states that "a maximum of 8 damaged and remaining intact (for a total of 24) PWR fuel assemblies can also be transported within the NUHOMS-24PTH DSC. Section A.6.5.5.3 of the application, *Criticality Results*, states that the DSC can transport "up to 12 damaged (up to 8 failed)" fuel assemblies. The application should be revised to be consistent in all sections that discuss damaged/failed fuel. Additionally, Table A.1.4.3-2 of the application does not list CE 16x16 fuel assemblies as allowable contents, while they are listed in the loading curves described in Tables A.6.5.5-9 and A.6.5.5-10 of the application. The application should be revised to clarify whether or not CE 16x16 fuel assemblies are intended as allowable contents.

This information is needed to ensure that the contents of the Model No. TN NUHOMS MP-197 package are adequately described, per the requirements of 10 CFR 71.33.

6-15 Revise the application to clarify the loading requirements for damaged fuel in the NUHOMS-24PTH DSC.

Table A.6.5.5-10 provides the acceptable average initial enrichment and burnup combinations for damaged fuel loaded in the NUHOMS-24PTH DSC. It is not clear, however, if these limits are applied to the damaged fuel only, or for all fuel in the DSC when damaged fuel is loaded. The application should be revised to clarify the damaged and intact fuel loading requirements.

This information is needed to ensure that the contents of the Model No. TN NUHOMS MP-197 package are adequately described, per the requirements of 10 CFR 71.33.

7.0 Package Operations

7-1 Provide technical justification that Chapter 7 and Chapter 8 are not required to be referenced in their entirety in the CoC.

The applicant marked only part of the Operating Procedures and Acceptance Tests and Maintenance Programs as conditions of the CoC. The staff reviewed the proposed justifications and they are not sufficient. In particular, given the potential aging of the materials used in the cask because of prolonged storage before shipment, the entire Operating Procedures and Acceptance Tests and Maintenance Programs are required as conditions in the CoC. The example the applicant provided as rationale for not including the entirety of Chapter 7 and 8 is not relevant and not the common practice (see NUREG-1617, Section 8.2.4; NUREG-1609, Section 7.5; and ISG-20).

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47, 71.85, and 71.87.

7-2 Modify Chapter 7, A.7.1.5 to add dose rate measurements and calculation of Transport Index as part of the procedures.

Section A.7.1.5 of the SAR defines the operating procedures for the NUHOMS MP-197 packages. However, it appears that the dose rate measurements and calculation of the Transport Index were not included in the operating procedures. The applicant is requested to revise Chapter 7 of the SAR to add dose rate measurements and calculation of Transport Index as part of the procedures.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.47, 71.83, and 71.87.

8.0 Acceptance Tests and Maintenance Program

8-1 Provide evaluations of the neutron shielding material that include radiation effects as well as thermal effects in combination.

RAI 8-2, provided previously, requested test results that demonstrate the neutron shield will not degrade over time. Radiation from the package contents will also affect the neutron shielding material. The response did not include results of evaluations/tests for radiation effects combined with thermal effects. This information is important in understanding the neutron shield performance over time. Additionally, the application should address a time frame of greater than 20 years, since the canisters loaded with spent fuel could be in storage for a longer period of time. Otherwise, a neutron shield maintenance test of adequate periodicity should be included in Chapter A.8 of the application due to the possible use of a package for greater than 20 years.

This information is needed for the staff to determine if the package design meets the regulatory requirements of 10 CFR 71.83 and 71.87.