

February 14, 2012

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

SUBJECT: FIRST REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF  
AMENDMENT NO. 13 TO THE STANDARDIZED NUHOMS® SYSTEM  
(TAC NO. L24519)

Dear Mr. Shaw:

By letter dated February 9, 2011, Transnuclear, Inc. (TN) submitted an amendment request to the U.S. Nuclear Regulatory Commission (NRC) for Certificate of Compliance (CoC) No. 1004. This amendment proposes to include the following changes to the Standardized NUHOMS® storage system:

- Adds the 69BTH and 37PTH dry shielded canisters (DSC).
- Evaluates the 24PHB DSC to accommodate control components other than burnable poison rod assemblies (BPRAs), and to allow for storage of damaged fuel assemblies, and other minor enhancements.
- Evaluates the 32PT DSC for incorporation of high burn-up fuel assemblies with and without control components (CCs), and other minor enhancements.
- Evaluates the 61BTH and 24PTH DSCs for storage of failed fuel.
- Evaluates the high-seismic horizontal storage module (HSM) Model HSM-HS for storage of the following DSCs: 61BT, 32PT, 24PTH, 61BTH, 69BTH, and 37PTH.
- Extends the use of metal matrix composites (MMC) as a neutron absorber material in the 61BTH Type 1 and Type 2 DSCs for higher heat loads.
- Evaluates DSCs for the addition of BLEU and MOX fuel assemblies as authorized contents.
- Evaluates HSM-H and HSM-HS inlet vent shielding design modifications to achieve dose reductions.
- Evaluates the OS200 transfer cask (TC) to allow transfer of the 61T, 32PT, 24PTH, and 61BTH DSCs.
- Adds language in the updated final safety analysis report (UFSAR) to allow the use of Type III cement as an alternate equivalent to the Type II cement used in HSM construction.
- Changes Technical Specifications (TS) neutron absorber testing and acceptance requirements in order to remain consistent with similar requirements in other ongoing licensing actions, plus certain new changes in this area.
- Makes certain additional changes for consistency within the TS and the UFSAR.

The staff has determined that further information is needed to complete its technical review. The request for additional information (RAI) is in the enclosures. Your response should be provided by March 19, 2012. If you are unable to meet this deadline, please notify us in writing, at least one week in advance, of your new submittal date and the reasons for the delay. The

staff will then assess the impact of the new submittal date and notify you of a revised schedule. We have developed an updated schedule, as follows. This schedule assumes a second round RAI. If no second RAI is necessary, the schedule may be accelerated.

<b>ACTION</b>	<b>SCHEDULED DATE</b>
Request for Additional Information #1	02/14/12
Response to RAI #1	03/19/12
Complete Safety Evaluation Report, Draft CoC, and Draft TS	08/31/12

Please reference Docket No. 72-1004 and TAC No. L24519 in future correspondence related to this licensing action. If you have any questions, please contact me at (301) 492-3371.

Sincerely,

**/RA/**

B. Jennifer Davis, Senior Project Manager  
Licensing Branch  
Division of Spent Fuel Storage and Transportation  
Office of Nuclear Material Safety  
and Safeguards

Docket No. 72-1004  
TAC No. L24519

Enclosures:

- (1) Request for Additional Information (non-proprietary)
- (2) Request for Additional Information (proprietary)

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\*See Previous Concurrence

TRANSNUCLEAR INC.

DOCKET NO. 72-1004

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO AMENDMENT NO. 13 TO THE

STANDARDIZED NUHOMS® SYSTEM

By letter dated February 9, 2011, Transnuclear Inc. (TN) submitted an amendment request to the U.S. Nuclear Regulatory Commission (NRC) for Certificate of Compliance (CoC) No. 1004. This amendment proposes to include the changes listed in the cover letter, to the Standardized NUHOMS® storage system.

This request for additional information (RAI) identifies additional information needed by the NRC staff in connection with its review of the amendment. The requested information is listed by section number and/or page number in TN's Safety Analysis Report (SAR) and associated documentation. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," was used by the staff in its review of the amendment application.

Each individual RAI section describes information needed by the staff to complete its review of the application and the SAR and to determine whether the applicant has demonstrated compliance with the regulatory requirements.

### **3.0 Structural Evaluation**

**RAI 3-1:** Provide justification for classifying the Independent Spent Fuel Storage Installation (ISFSI) basemat as a component that is "Not Important to Safety" in Table 3.4-1 of the application (Ref. 1), "NUHOMS Major Components and Safety Classifications".

NRC NUREG-6407 (Ref. 2) identifies the ISFSI basemat as a structure Important to Safety (ITS), Category C. As NUHOMS® Amendment No. 13 is for a CoC, which will be used by general licensees, it is important to identify the basemat as ITS, Category C per the guidelines. In accordance with the definition of "structures, systems and components important to safety" in 10 CFR 72.3, this structure should be included as one which is ITS.

This information is required for compliance with 10 CFR 72.3.

**RAI 3-2:** Provide design details of the flat stainless steel heat shields (for roof and side) to be used when the 37PTH dry shielded canister (DSC) is stored in the horizontal storage module (HSM), HSM-H or HSM-HS.

The SAR, Appendix Z, Section Z.3.1.1.1, states that the flat stainless steel option is used when the 37PTH DSC is stored in the HSM-H or HSM-HS. Staff needs to verify the adequacy of the structural design details, connection details, and functional requirements, if any, for the heat shields discussed in the application.

This information is required for compliance with 10 CFR 72.236(b).

**RAI 3-3:** Provide an explanation for the discrepancy in the calculated weights reported in Table Z.3.2-1, “Summary of the NUHOMS®-37PTH System Component Nominal Weights.”

In Table Z.3.2-1, the “HSM-H / HSM-HS Single Module Weight Max. (Loaded)” shown seems to be incorrect. This weight should be a total of the Loaded DSC weight (wet) plus an empty HSM-H / HSM-HS single module weight. Discuss the implications of this in computations of margin of safety during the on-site transfer operations.

This information is required for compliance with 10 CFR 72.236(l).

**RAI 3-4:** Provide justification for using the allowable stress values at three different temperature limits of 500° F, 550° F and at 800° F as shown in the footnote for the table on page Y.3-84 titled, “Summary of Basket Assembly Stress Analysis due to Operation/Storage Loads.”

For the 69BTH DSC, the stress ratio for basket assembly due to operation/storage loads reported in the table in Section Y3.6.1.3.4 D, “Summary of Basket Assembly Stress Analysis due to Operation / Storage Loads,” uses allowable stress values at several different temperatures and lists the ASME Service Level A through D results for basket assembly. Staff needs to understand the logic supporting these different temperatures and the corresponding allowable stress values.

This information is required for compliance with 10 CFR 72.236.

**RAI 3-5:** Justify the use of the “Enveloped Load for Analysis” of 3.464g Vertical for Operation/Storage loads for baskets in HSM-HS modules for the 69BTH DSC and the 37PTH DSC.

A footnote shown for the table titled “Basket Loads in HSM-H/HSM-HS (Operation Storage loads) on page Y.3-78, in Section Y.3.6.1.3.2 indicates that the vertical load was increased to 3.464g for service level D “High Seismic” for the 69BTH DSC (and for the corresponding table on page Z.3-74 for the 37PTH DSC). This also appears in note number 4 for the table that the applicant provided as part of the response to Request for Supplemental Information (RSI) 16 (Ref. 3). Explain how this number was derived for convergence purposes.

This information is required for compliance with 10 CFR 72.236.

**RAI 3-6:** See Enclosure 2.

**RAI 3-7:** Provide or justify not providing the following LS-DYNA \*DATABASE\_OPTION output components for the ASCII files related to the LS-DYNA Finite Element Analysis (FEA).

The response to RSI-19 (Ref. 3) provided many of the \*DATABASE\_OPTION output components that were requested. However several of the files, listed below, did not contain the information needed to plot time history (e.g., Figure Z.3.7-4 (a) through Figure Z.3.7-7 (a) in the application (Ref. 1)).

**37PTH-14X14PWR [pressurized water reactor] - Fuel-Corner-Drop**

elout  
nodout

**37PTH-Basket-0-Deg-Buckling-Analysis**

bndout  
elout  
nodout

**37PTH-Basket-Accident-180-Deg-Side-Drop**

bndout  
elout  
nodout

**69BTH-7X7BWR [boiling water reactor] - Fuel-Corner-Drop**

elout  
nodout

Note that when opening the files using a .txt viewer (e.g. WordPad), the printout states only  
"NUH37PTH 14X14 80" CORNER DROP  
ls-dyna ls971d.7600.1224 R2 date 08/13/2007"

This information is needed to demonstrate compliance with 10 CFR 72.236(l).

**RAI 3-8:** Provide an ANSYS analysis for the 37PTH DSC basket assembly during the 80-inch accidental side drop.

The Amendment No. 13 SAR section Y.3.7.4.3.1 discusses the 69BTH DSC basket assembly side drop (accident) analysis by using ANSYS quasi-static analysis. Amendment No. 13, SAR section Z.3.7.4.3.1 evaluates the 37PTH DSC basket assembly side drop (accident) by a LS-DYNA transient dynamic finite element analysis.

The 37PTH DSC basket assembly analysis methodology using LS-DYNA to determine stress values (no direct stress intensities output) is inconsistent with the current ANSYS analysis technique for the 80-inch, 75g side drop. Staff notes that the SAR section Z.3.7.4.3.1 provides a justification for using LS-DYNA for basket analysis, "LS-DYNA is used for the accidental drop load cases because of its robust contact algorithms which are able to model contact between the different components in the model." However, the LS-DYNA capability for basket assembly analysis must be properly benchmarked for intended use. Specifically, the applicant should demonstrate that LS-DYNA analysis results can be properly post-processed for section-cut internal stress quantities (i.e. the ability to extract validated stress intensities consistent with ASME code criteria) relevant for a comprehensive structural integrity evaluation of the 37PTH fuel basket assembly.

This information is needed to demonstrate compliance with 10 CFR 72.236.

## 4.0 Materials Evaluation

**RAI 4-1:** Staff recommends excluding MOX from the list of potential contents. Alternatively, provide additional documentation that justifies what cladding materials are compatible with MOX fuel. If providing documentation, cite specific reactor experience where MOX has been used in reactors. State the maximum burn-ups achieved in those reactors with MOX fuels and the cladding materials that were used.

The use of MOX fuel in commercial American reactors is extremely limited, and characterization of MOX fuel for storage is therefore also limited.

This information is required to demonstrate compliance with 10 CFR 72.236(a). See also RAIs 4-8, 4-11, 5-5, 5-6 and 5-7.

**RAI 4-2:** Remove all references to the use of digital image analysis as a methodology to demonstrate the areal density of boron in the neutron absorber materials. (For example, see Section Y.9.1.7.7, "Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission.")

No documentation has been provided to the staff that this method has been used to demonstrate the area density of B<sup>10</sup> in neutron absorber materials for spent nuclear fuel applications.

This information is required to demonstrate compliance with 10 CFR 72.124(b).

**RAI 4-3:** Clarify engineering notes throughout the licensing drawings (e.g., Note 2 on Sheet 1 of drawing NUH-32PT-1005-SAR) allowing "alternate details or methods of attachments are acceptable with TN approval" to permit only mechanical joining.

The use of brazes or dissimilar welds combined with the heat loads within the canister may lead to failure of the aluminum/stainless steel join.

This information is required to demonstrate compliance with 10 CFR 72.236.

**RAI 4-4:** Clarify the term equivalency in the statement, "[w]eld metal is E/ER209 or TN Approved Equivalent," on the licensing drawings (e.g., Note 4 on Sheet 1 of Drawing NUH-32PT-1004 -SAR). Alternatively, remove the note on the licensing drawings and specify that the code of construction requires the use of appropriate weld metals.

Equivalency of materials should be verified by staff to make a Safety Evaluation. Welds that are qualified under Subsection NG under Section III of the ASME Code are required to use appropriate filler metals.

This information is required to demonstrate compliance with 10 CFR 72.236.

**RAI 4-5:** Provide a widely recognized industrial standard, e.g., ASME, specifying material properties for potential alternative materials.

The statement "[a]lternate material specifications (plate, bars or forging) to those specified may be used provided mechanical properties are equal to or greater than the material specified, and chemical composition is the same" (e.g., Note 1 on Sheet 1 of Drawing NU61BTH-72-1105)

should be qualified such that a widely recognized industrial standard dictates the material properties of the alternative materials.

This information is required to demonstrate compliance with 10 CFR 72.236.

**RAI 4-6:** Justify the use of Section III, Subsection NF of the ASME Code for construction of the basket hold down ring (e.g., Sheet 1 of Drawing NU69BTH-72-1013).

The hold down ring functions as a component of the fuel basket in the 61BTH DSC, which should be constructed to Section III, Subsection NG of the ASME Code following the guidance in NUREG-1567.

This information is required to demonstrate compliance with 10 CFR 72.144(b).

**RAI 4-7:** Staff recommends removing all references to all fuels with burn-ups above 62 GWd/MtU from the application.

There is insufficient data on fuel with burn-ups above 62 GW/MtU to make a safety determination for this application.

This information is required to demonstrate compliance with 10 CFR 72.236(a). See also RAIs 5-1, 5-8 and 6-1.

**RAI 4-8:** Clarify what operational procedures are in place to determine if fuel assembly growth has exceeded the dimensions required for placement and retrieveability in the fuel basket.

Damaged fuel assemblies, fuel assemblies using advanced cladding materials and/or claddings loaded with MOX fuel may be prone to excessive elongation during reactor operation.

This information is required to demonstrate compliance with 10 CFR 72.236(m). See also RAI 4-1.

**RAI 4-9:** Require TN or CoC-holder approval for all instances where "alternative" or "equivalent" weld joint configurations, materials, etc. are noted on the licensing drawings (e.g., Note 4 on Sheet 1 on Drawing NU69BTH-72-1015).

In order to ensure that the fabricated components of the NUHOMS system meet the requirements of the design presented to the staff for review, any significant alterations by the fabricator should be approved by the CoC-holder in such a way that changes are consistent with the licensing basis.

This information is required to demonstrate compliance with 10 CFR 72.236(b).

**RAI 4-10:** Provide well-established references for the thermal conductivity for 1100 and 6061 aluminum alloys at elevated temperatures. Provide these sources to the staff and update the thermal analyses in the SAR accordingly. Extrapolated or assumed values are not acceptable for intended application of aluminum alloys in the packaging.

The thermal properties of 1100 and 6061 aluminum alloys at elevated temperatures are not listed in the ASME Code.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

**RAI 4-11:** Incorporate the presence of higher fission product gases in MOX clad fuels into the maximum DSC internal pressure calculations. Assume the maximum potential quantity of fission product gases in the rods over the 20-year licensing period.

The maximum DSC internal pressure calculations do not appear to account for the presence of MOX clad fuel in the DSC.

This information is required to demonstrate compliance with 10 CFR 72.236(d). See also RAI 4-1.

**RAI 4-12:** Provide corrosion testing results of metal matrix composites produced by molten aluminum infiltration.

This information should be provided to confirm the adequacy of the material's corrosion resistance.

This information is required to demonstrate compliance with 10 CFR 72.236(g) and (h).

**RAI 4-13:** Use bounding conservative estimates for the expansion of zirconium-based alloys.

The thermal expansion coefficient of zirconium alloys cited in the SAR is not based upon bounding thermal expansion estimates, as shown in Figure 4.11 of the current version of NUREG/CR-6150 (Ref. 4).

The irradiation growth of zirconium alloys should be based on conservative estimates from more recent documentation on the irradiation growth of zirconium alloys, e.g. NUREG/CR-7024. The staff also notes that the reference to NUREG/CR-6150 cited in the SAR appears to be the original version (Ref. 5).

This information is required to demonstrate compliance with 10 CFR 72.236(m).

**RAI 4-14:** Provide a copy of the Interagency Agreement between the Tennessee Valley Authority (TVA) and the U.S. Department of Energy (DOE) which specifies the allowable concentrations of elements in Blended Low-Enriched Uranium (BLEU) fuel. Clarify the elemental chlorine content of the fuel, as mentioned in TVA, "Additional Use of Blended Low Enriched Uranium (BLEU) in Reactors at TVA's Browns Ferry and Sequoyah Nuclear Plants", May 2011. Also state if BLEU fuels with pinhole leaks or hairline cracks will be permitted for dry cask storage.

The presence of certain elements in the BLEU fuel has the potential for inducing stress corrosion cracking in the cladding or the confinement boundary.

This information is required to demonstrate compliance with 10 CFR 72.236(d).

**RAI 4-15:** Explain the basis for the assumed 10% reduction in strength for the mechanical properties of concrete above 350°F as stated in Sections Y.4.4.8 and Z.4.4.4 of the SAR.

The SAR provides no justification for the assumed 10% reduction of strength for concrete at temperatures above 350°F.

This information is required to demonstrate compliance with 10 CFR 72.236(l).

**RAI 4-16:** Provide a copy of letter No. FAB10-449, "Transmittal of M5<sup>®</sup> Data for Application to Oconee Spent Fuel," dated May 19, 2010, Areva NP Inc.; to the staff, to validate the properties of irradiated M5<sup>®</sup> cladding material.

The source of the mechanical properties for irradiated M5<sup>®</sup> cladding should be provided to the staff to make a safety evaluation. Additional information supporting the reported properties of M5<sup>®</sup> cladding after operation in the reactor may aid the staff in the review of the application.

This information is required to demonstrate compliance with 10 CFR 72.236(d).

**RAI 4-17:** Perform structural analyses of the spent nuclear fuel using conservative estimates which incorporate engineering uncertainties for the mechanical properties for high burn-up fuel.

The applicant cited PNNL-17700, "Stress/Strain Correlation for Zircaloy," as a reference for the mechanical properties of the spent nuclear fuel. Section 5, "Application of Material Property Models to Spent Fuel Cladding Analyses," of this document expressly states, however:

*Due to the lack of mechanical data for RXA BWR fuel cladding there is a large uncertainty in their mechanical properties and due to the lack of data it is difficult to quantitatively estimate uncertainties. In addition, hydrides in RXA cladding appear to have a more random orientation than SRA cladding such that some hydrides are orientated in the radial direction that may result in lower ductilities (uniform elongation) than SRA material at an equivalent excess hydrogen level. Therefore, an engineering judgment is made that the uncertainties in the RXA data are a factor of 2 larger than those for SRA mechanical properties with fast fluences and excess hydrogen levels above  $3 \times 10^{25} \text{ n/m}^2$  and 100 ppm, respectively.*

Engineering uncertainties were not incorporated in the structural analyses for either PWR or BWR fuel assemblies.

This information is required to demonstrate compliance with 10 CFR 72.236(d).

**RAI 4-18:** Justify that the thermal properties of the concrete listed in the SAR are conservative.

Conservative values for the thermal properties of concrete should be used for the thermal analysis in the SAR. Alternatively, the applicant should demonstrate through acceptance testing that the concrete's thermal properties are bounded by the analysis in the SAR.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

**RAI 4-19 (QUALITY):** Correct the typographical errors for SA-336 steel on the licensing drawings.

ASME 336 Type 304 austenitic stainless does not appear in Section IID of the ASME code.

This information is required to demonstrate compliance with 10 CFR 72.236(b).

**RAI 4- 20:** Specify the material of construction for the stainless steel heat shield in the licensing drawings.

It is not clear if the heat shield is made of ferritic or austenitic stainless steel in the licensing drawings.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

**RAI 4-21:** Explain how the assumption of 10% reduction in concrete material properties is conservative for the structural analysis of HSM-H concrete components when the concrete temperature is 83°F above the limit in a blocked vent accident.

The applicant evaluated the blocked vent accident condition for the 69BTH DSC in Appendix Y, Section Y.4.4.8 and calculated a maximum concrete temperature of 433°F at the end of 40 hours, which is above the 350°F limit given in Section 4 of ACI-349 for accident conditions. To account for the effect of higher concrete temperature on the concrete compressive strengths, the applicant performed a structural analysis of HSM-H concrete components in Section Y.3, by assuming 10% reduction in concrete material properties. The applicant should explain how the assumption of 10% reduction in concrete material properties is conservative for structure evaluation when the concrete temperature is 83°F above the limit under the blocked vent accident.

This information is required to assess compliance with 10 CFR 72.236(a).

## 5.0 Thermal Evaluation

**RAI 5-1:** Verify/ensure that (1) the referenced axial heat flux profile (in Appendix T, Figure T.4-37) is bounding for all BWR spent fuel assemblies that can be stored in the 69BTH DSC and (2) the axial heat flux profile for BWR fuel will exhibit the same flattening with burn-up that is noted for PWR fuel (in DOE/RW-0472, Rev. 2 (Ref. 6)).

The applicant stated in Appendix Y, Section Y.4.6.4 that the axial heat flux profile of Figure T.4-37 is applicable to the BWR spent fuel with the maximum burn-up of 45 GWd/MTU. However, Figure T.4-37 indicates that this axial heat flux profile was derived based on in-core data from “an operating BWR facility” and the measurements were taken for 25 axial locations along the length of a fuel assembly, at 5 fuel assembly locations in the core.

Therefore, the applicant should clarify whether the axial heat flux profile (Fig. T.4-37) is appropriate to thermal evaluation for all 69BTH DSCs with burn-up greater than 45 GWd/MTU because (1) the facility, the power level of the assemblies, and the fuel type for the 69BTH DSCs are not identical to those in Appendix T, and (2) DOE/RW-0472 specifically indicates that the investigations are needed to determine the appropriate axial heat flux profile per burn-up credit for BWR assemblies to be loaded into high density casks.

1. Since the axial heat flux profile of Figure T.4-37 is based on a limited and incompletely defined data set, why is it assumed that this heat flux profile is bounding for all BWR spent fuel assemblies that can be stored in the 69BTH DSC, including the fuel assembly with burn-up greater than 45 GWd/MTU or up to 70GWd/MTU.
2. Since the cited reference (DOE/RW-0472) specifically indicates that studies are needed to determine the burn-up effects for BWR fuel, why is it assumed that the axial heat flux profile for BWR fuel will exhibit the same flattening with burn-up that is noted for PWR fuel (in DOE/RW-0472, Rev. 2).

This information is required to assess compliance with 10 CFR 72.236(a), (d), and (f). See also RAI 4-7.

**RAI 5-2:** Clarify/evaluate whether the fuel assembly FANP 9x9 with only 72 fuel rods bounds the fuel assembly FANP 9x9-2 with 79 fuel rods for thermal evaluation of 69BTH DSC.

The applicant stated in Appendix Y, Section Y.4.9.1 that the fuel assembly FANP 9x9 is bounded by the fuel assembly FANP 9x9-2 in thermal evaluation because of the higher number of active fuel rods loaded in FANP 9x9 design in comparison to FANP 9x9-2 design. However, this section also stated that the number of fuel rods for the FANP 9x9 design varies between 72 and 81, while the FANP 9x9-2 design has 79 fuel rods. The applicant should clarify whether the FANP 9x9 design with only 72 fuel rods still bounds the FANP 9x9-2 design with 79 fuel rods for thermal evaluation of the 69BTH DSC.

This information is required to assess compliance with 10 CFR 72.236(a) and (f).

**RAI 5-3 (QUALITY):** (1) Clarify or correct the number of fuel rods, the fuel rod diameter, and the fuel rod thickness for fuel assemblies FANP 9x9 and LaCrosse 10x10 and (2) clarify or correct the water tubes with the water hole in Table "Fuel Assembly Effective Density and Specific Heat" (page Y.4-57).

The table, "Fuel Assembly Effective Density and Specific Heat" (page Y.4-57) appears to contain a number of errors, with FANP 9x9 data interchanged with LaCrosse 10x10 data:

- a) The number of fuel rods is listed as 96 for the FANP 9x9; this assembly design has only 81 pin positions in the array.
- b) The number of fuel rods is listed as 81 for LaCrosse 10x10; this fuel assembly has a 10x10 array with 100 pin positions and can contain 96 or 100 fuel rods.
- c) The fuel rod diameter and clad thickness appear to be switched between FANP 9x9 and LaCrosse 10x10.
- d) The SVEA-92 has a fractional number of water tubes; in fact, this design has no water tubes, but rather an open rectangular central water hole.

This information is required to assess compliance with 10 CFR 72.236(a).

**RAI 5-4:** Explain (a) how a 30% reduction of the maximum decay heat of LaCrosse fuel assembly is determined for the thermal evaluation of the 69BTH DSC and (b) how the decay heat is correlated to enrichment, burn-up, and cooling time for the LaCrosse fuel assemblies.

The applicant proposed to load the LaCrosse fuel assembly in the 69BTH DSC, and evaluated the effective properties with the 2-D model for LaCrosse fuel assembly in Appendix Y, Section Y.4.9.2. The applicant noted in Appendix Y, Figure Y.2-1 that the payload in the 69BTH DSC will be adjusted to maintain the total DSC heat load within the specified limit and reduce the maximum decay heat of the LaCrosse fuel assembly to 70% of the listed value for all six heat load zoning configurations (HLZCs). The applicant should detail how the 30% reduction is determined. This information is needed to ensure the thermal evaluation is conservative for LaCrosse fuel assemblies.

This information is required to assess compliance with 10 CFR 72.236(a), (d), and (f).

**RAI 5-5:** Provide the decay heat generation of MOX fuel assemblies and the loading procedure to ensure the decay heat limit is met in each zone of HLZCs 2, 3, 4, and 5 of the 69BTH DSC and HLZC 3 of the 37PTH DSC, even though the MOX has higher decay heat generation and burn-up.

The applicant proposed to replace the UO<sub>2</sub> fuel assembly with the MOX fuel assembly in the 69BTH and 37PTH DSCs. The applicant stated that the characteristics of the MOX fuel are almost identical to the UO<sub>2</sub> fuel they are replacing, except for thermal conductivity. The applicant summarized the thermal conductivity of irradiated UO<sub>2</sub> fuel and MOX fuel for the 69BTH DSC in Appendix Y, Section Y.4.9.3 and the 37PTH DSC in Appendix Z, Section Z.4.9.1. With the reduced thermal conductivity and the increased decay heat for MOX fuel, the thermal evaluation of MOX fuel could be less conservative. The applicant should provide:

1. The decay heat generation of MOX fuel and,
2. The procedure to ensure that the loading of MOX fuel with higher decay heat and burn-up still meets the specified limit in each zone of HLZCs 2, 3, 4, and 5 of the 69BTH DSC and HLZC 3 of the 37PTH DSC.

This information is required to assess compliance with 10 CFR 72.236(a), (d), and (f). See also RAI 4-1.

**RAI 5-6:**

1. Describe the characteristics of LaCrosse, FANP and SVEA fuel assemblies and provide the corresponding heat load generation and burn-up for each, and
2. Mark the loading zones of LaCrosse, FANP, and SVEA fuel assemblies, respectively, in all six HLZCs of the 69BTH DSC and
3. Provide the loading procedures of LaCrosse, FANP, SVEA to ensure the decay heat limits are met in each HLZC of the 69BTH DSC.

The applicant noted in Appendix Y, Section Y.4.9 that the FANP, LaCrosse, and SVEA fuel assemblies can be loaded in the 69BTH DSC. To accelerate the staff's review, the applicant should:

- (a) Provide or repeat the descriptions of LaCrosse, FANP, and SVEA and their heat load generations and burn-ups in the current Appendix Y,
- (b) Mark the loading zones of LaCrosse, FANP, and SVEA fuels in all six heat load zoning configurations (HLZCs) of 69BTH DSC as done for the aluminum dummy and MOX fuel assemblies (Figures Y.2-1 thru Y.2-6), and,
- (c) Provide the loading procedures of LaCrosse, FANP, and SVEA fuels to ensure the decay heat limits are met in each HLZC of the 69BTH DSC.

This information is required to assess compliance with 10 CFR 72.236(a), (d), and (f). See also RAI 4-1.

**RAI 5-7:** Demonstrate/verify the assumptions and/or the equations used to derive the transverse effective conductivity ( $k_{\text{MOX-irrd}}$ ) for MOX fuel pellets and provide the data for  $k_{\text{MOX-irrd}}$  from ORNL's report (Ref. Y.4.15) in the current Appendix Y.

The applicant described in Appendix Y, Section Y.4.9.4 that the transverse effective conductivity of the MOX fuel is calculated using the model for the bounding FANP 9x9-2 fuel assembly with the assumption that the reduction rate of the MOX fuel pellet conductivity due to irradiation is the same as that for UO<sub>2</sub> fuel pellet. The applicant stated that the irradiated conductivity ( $k_{\text{MOX-irrd}}$ ) for the MOX fuel pellet calculated for FANP 9x9-2 fuel assembly is lower than that from ORNL's report (SAR Appendix Y, Ref. 4.15).

To verify assumption and equation used to derive  $k_{\text{MOX-irrd}}$ , the applicant should demonstrate that (1) the calculation of  $k_{\text{MOX-irrd}}$  takes into account the burn-up dependence as described in PNNL's report of FRAPCON-3 Code for MOX Fuel Properties (FRAPCON-3 is a U.S. Nuclear Regulatory Commission-sponsored code) and (2) the reduction rate of UO<sub>2</sub> fuel pellet is equivalent to the reduction rate of MOX fuel pellet.

The applicant should also provide the data of  $k_{\text{MOX-irrd}}$  from ORNL's report (Ref. Y.4.15) in the current Appendix Y, Section Y.4.9.4.

This information is required to assess compliance with 10 CFR 72.236(a), (d), and (f). See also RAI 4-1.

**RAI 5-8:** Provide more information (procedures and derivations) to validate the maximum internal pressure calculation of 69BTH DSC.

The applicant analyzed the maximum internal pressures for 69BTH DSC under normal, off-normal and accident conditions based on the maximum allowable heat load of 35 kW, and on the maximum burn-up of 70 GWd/MTU (if applicant continues to include). The applicant accounts for the canister free volume, the quantities of canister backfill gas, fuel rod fill gas, and fission products, and the average canister cavity gas temperature in the analysis.

To help the staff justify the pressure evaluation, the applicant is requested to explain how the data below are derived and provide the data derivations in the Amendment No. 13 SAR Thermal chapter:

- a) the bounding (maximum) plenum volume of 2.136 in<sup>3</sup> (SVEA FAs) among all fuel assemblies,
- b) the free helium volume in DSC (263687 in<sup>3</sup>), and the volume of helium in fuel assemblies along active fuel (244059 in<sup>3</sup>) as listed in Table Y.4-12, "Volume and Average Helium Temperatures in 69BTH DSC for Storage/Transfer Conditions",
- c) the bounding (maximum) initial rod pressure of 160 psia, and
- d) usage of only 98% of the DSC free cavity volume.

This information is required to assess compliance with 10 CFR 72.236(a). See also RAI 4-7.

**RAI 5-9:** Clarify/verify whether or not the load case T9 is bounded by the load case T6 for thermal analysis of 69BTH DSC. If not bounded, perform the thermal analysis for load case T9 and provide the results in Appendix Y.

The applicant justified in Appendix Y that the load case T9 (35 kW, HLZC 5, transient, vertical operations, 140°F ambient, without insolation) is bounded by the load case T6 (35 kW, HLZC 5, transient with forced convection, normal transfer, 100°F ambient, with insolation). Given the

conditions that the load case T9 has vertical operation, and an ambient temperature of 140°F, which is much higher than the ambient temperature of 100°F in the load case T6, the justification that the load case T9 is bounded by the load case T6 in Appendix Y may not be adequate.

The applicant should provide the thermal analysis for the load case T9 and provide the results in Appendix Y for staff's review to ensure the thermal criteria are met, in accordance with ISG-11 (Ref. 7).

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-10:** Clarify/verify whether or not the load case T10 is bounded by the load case T6 for thermal analysis of 69BTH DSC. If not bounded, perform the thermal analysis for the load case T10 and provide the results in Appendix Y.

The applicant justified in Appendix Y that the load case T10 (35 kW, HLZC 5, transient with forced convection, off-normal transfer, 117°F ambient, without insolation) is bounded by the load case T6 (35 kW, HLZC 5, transient with forced convection, normal transfer, 100°F ambient, with insolation). Given the conditions that the load case T10 has a higher ambient temperature (117°F) and a higher cladding temperature limit of 570°C for off-normal-transfer, which are higher than the ambient temperature of 100°F and the cladding temperature limit of 400°C in the load case T6 for normal transfer, the justification that the load case T10 is bounded by the load case T6 in Appendix Y may not be adequate.

The applicant should provide the thermal analysis for the load case T10 and provide the results in Appendix Y for staff's review to ensure the thermal criteria are met, in accordance with ISG-11.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-11:** (a) List/summarize (in current SAR) the minor deviations in dimensions of the fuel assemblies listed in Appendices Z, M and P, and (b) explain how it is determined that the deviations have no effect on the bounding effective PWR fuel properties for thermal evaluation.

The applicant described in Appendix Z, Section Z.4.9 that the effective fuel properties (density, thermal conductivity, and specific heat) for PWR fuel assemblies to be stored in 37PTH DSC are back-referenced to Appendix M, Section M.4.8 and Appendix P, Section P.4.8; the applicant also noted in Appendix Z that there are minor deviations between the dimensions of the fuel assemblies listed in Appendix Z.2 and those studied in Appendices M and P, but these differences have no effect on the bounding effective PWR fuel properties. To ensure the bounding effectiveness in thermal evaluations, the applicant is requested to (a) list/summarize these minor deviations in dimensions of the fuel assemblies and summarize the deviations in the current Appendix Z of NUHOMS Amendment No. 13, and (b) explain how it is determined that these deviations have no effect on the bounding effective PWR fuel properties for thermal evaluation.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-12:** Demonstrate how the effective densities are calculated for the 37PTH basket and the 32PTH1 basket, respectively in Appendix, Section Z.4.4.2 and justify the results, relative to the 32PTH1 basket.

The applicant compared the effective properties between 37PTH basket and 32PTH1 basket in the Table of page Z.4-9 of Appendix Z, Section Z.4.4.2. It does not seem reasonable that the effective density of the 37PTH basket is listed higher than that of the 32PTH1 basket in the Table, because the 37PTH basket contains openings for 5 more fuel assemblies than does the 32PTH1 basket. The applicant is requested to demonstrate how the effective densities are calculated for the 37PTH and the 32PTH1 baskets, respectively, and justify the results, relative to the 32PTH1 basket.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-13:** Provide more information (e.g., derivations) to validate the maximum internal pressure calculation of 37PTH DSC.

The applicant evaluated the maximum internal pressures for 37PTH DSC under normal, off-normal and accident conditions based on the maximum heat load of 30 kW and the maximum burn-up of 62 GWd/MTU. The applicant accounts for the canister free volume, the quantities of canister backfill gas, fuel rod fill gas, and fission products, and the average canister cavity gas temperature in the calculations.

To help the staff justify the pressure evaluation, the applicant is requested to explain how the data below are derived and provide the data derivations in the Amendment No. 13 SAR Thermal chapter:

- a) the bounding (maximum) plenum volume of 1.326 in<sup>3</sup> among all fuel assemblies,
- b) the free helium volume in DSC (273058 in<sup>3</sup>) and the volume of helium in FAs along active fuel (231507 in<sup>3</sup>) listed in Table Z.4-6,
- c) the bounding (maximum) initial rod pressure, and
- d) usage of only 98% of the DSC free cavity volume.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-14:** Provide a schematic figure to show the cooling air flow path (from inlet to outlet) within the NUHOMS<sup>®</sup> HSM system.

The applicant should provide a schematic figure or drawing showing the cooling air flow path (from inlet to outlet) within the HSM, either for the 69BTH DSC or the 37PTH DSC design. The figure can be generated either by drawing or from a computer model (similar to Figure Y.4-1). The staff needs this information to understand the thermal design and the corresponding heat exchange.

This information is required to assess compliance with 10 CFR 72.236(a).

**RAI 5-15:** Explain how the transfer time limits of the 69BTH DSC and the 37PTH DSC are interpolated/determined from the 32PTH1 DSC thermal analyses under transfer conditions.

The applicant defined the transfer time limits of the 69BTH DSC with a 13-hour time limit for HLZCs 1 to 5 (decay heat  $\leq$  35 kW), and no time limit for HLZC 6 (decay heat  $\leq$  24 kW), as shown in Appendix Y, Section Y.4.5.2; as well as the transfer time limits of the 37PTH DSC with no time limit for HLZCs 1 and 2 (decay heat  $\leq$  24 kW), and a 14-hour time limit for HLZC 3 (decay heat  $\leq$  30 kW) in Appendix Z, Section Z.4.5.4, based on the bounding analyses of the 32PTH1 DSC with decay heat loads of 31.2 kW and 24 kW, respectively.

Given the different HLZCs, basket configurations, and “different” transfer time limits of 32PTH1 DSC (e.g., no limit for HLZC 3, 13 hours for HLZC 1, 14 hours for HLZC 2 intact fuel, and 10 hours for HLZC 2 damaged fuel in the 32PTH1 DSC), the applicant is requested to provide more information to explain how the transfer time limits of the 69BTH DSC and the 37PTH DSC are interpolated and determined from the 32PTH1 DSC thermal analyses under the varying transfer conditions.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-16:** See Enclosure 2.

**RAI 5-17:** Provide the calculations/derivations for the “maximum allowable air temperature rises” through the HSMs loaded with 69BTH DSCs and 37PTH DSCs.

The applicant specified that the “maximum allowable air temperature rise” through the HSM is a function of the DSC heat load and the HSM model, and summarized the “maximum allowable air temperature rises” of 100°F for the 69BTH DSC and 90°F for the 37PTH DSC, respectively, in Technical Specification (TS) LCO 3.1.4. The applicant should describe in the SAR how these temperature rises are derived and provide the corresponding parameters/conditions used for the derivations. The staff needs this information to ensure the evaluations of the “allowed maximum air temperature rises” in TS LCO 3.1.4 are acceptable for the 69BTH and the 37PTH DSCs.

This information is required to assess compliance with 10 CFR 72.236(a), (d) and (f).

**RAI 5-18:** Clarify or revise inconsistency of the hydrogen/flammability limits required by Appendices K, M, N, T, Y, Z, and Technical Specification 5.2.6 for all types of DSCs, including the 69BTH and 37PTH.

Appendices Y (69BTH) and Z (37PTH) state that “[m]onitoring of hydrogen concentration, before and during welding operations, will be performed to ensure that the hydrogen concentration does not exceed 2.4.” Appendices K (p. K.8-19, 61BT), M (p. M.8-17, 32PT), N (p. N.8-4, 24PHB), and T (p. T.8-18, 61BTH), however, state “[p]rovide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations to ensure that a safety limit of 2.4% is not exceeded *and in compliance with Technical Specification 5.2.6.* [Emphasis added.]”

However, Technical Specification 5.2.6 points out “For the 24P, 52B, 24PHB, 61BT, 32PT, 24PTH, 61BTH, 32PTH1, *69BTH and 37PTH* DSCs, while welding the inner top cover plate during loading operations, and while cutting the outer or inner top cover plates during unloading operations, hydrogen monitoring of the space under the shield plug in the DSC cavity is required, to ensure that the combustible mixture concentration remains below the flammability limit of 4.0%.”

It will be more conservative if the hydrogen/flammability limit is revised from 4.0% to 2.4% in Technical Specification 5.2.6 for controlling/monitoring the hydrogen concentration for all types of DSCs. The applicant should either clarify or revise the inconsistent hydrogen/flammability limit between all Appendices and Technical Specification.

This information is required to assess compliance with 10 CFR 72.236(a).

## 6.0 Shielding Evaluation

**RAI 6-1:** Provide validation of the SAS2H code for calculating the gamma and neutron source terms of fuel with burn-ups up to 70 GWd/MTU.

During the acceptance review, the staff requested that the applicant provide justification for using the SAS2H code for source term calculations for fuel with up to 70 GWd/MTU burn-up. In its response to the RSI (Ref. 3), the applicant provided a basis for using the SAS2H code for burn-ups greater than 62 GWd/MTU, which consisted of an evaluation of the conservatisms in the HLZC, assumed specific power, occupational exposure estimate, and ISFSI site dose calculation.

The staff reviewed these responses and did not find that the response established the basis for using the SAS2H code to calculate shielding source terms for fuel with up to 70 GWd/MTU burn-up. Specifically, the decay heat does not appear to have a linear relationship with the source term. Therefore, a 20% decay heat margin does not necessarily translate into a 20% safety margin for the shielding source terms. Regarding the conservatism provided by using higher specific power versus actual specific power, the magnitude of the dependency of the source term on specific power is not quantified, and it is not clear that this dependency is large enough to counter uncertainties in the calculated source term. In order to be able to take credit for the conservatism, a quantitative analysis is needed. Regarding the conservatism of using the 61 fuel assembly cask for occupational dose and site boundary dose calculations, it is not clear why there is a 20% dose reduction, given that the 69BTH DSC has 8 more fuel assemblies than the 61BTH DSC.

The applicant should provide:

1. An analysis to show a quantitative relationship between source terms and decay heat for the burn-up range between 62 GWd/MTU and 70 GWd/MTU to demonstrate that the decay heat can envelop source terms for the entire range of qualified fuels with respect to enrichments, burn-up, and cooling time;
2. A quantitative analysis to establish the relationship between source terms and specific power, and a quantification of the conservatism; and
3. A demonstration that the dose rate for the 61BTH DSC is 20% more than that of the 69BTH DSC. A comparison of the design basis source terms of the 69BTH and 61BTH DSCs would be acceptable for responding to this request.

This information is needed to ensure that the storage system design meets the requirements of 10 CFR 72.236(a). See also RAI 4-7.

**RAI 6-2:** See Enclosure 2.

## 7.0 Criticality Evaluation

**RAI 7-1:** Revise Section Y.6.3.1 of the SAR and associated CSAS25 input files to ensure that the correct input file unit numbers are provided and used by the code.

In Section Y.6.3.1, the third paragraph discusses the axial layout model as a sequence of units 58, 59, and 79 as defined in array 21. The CSAS25 input deck for the design basis intact fuel assembly case, Y.6.6.2, does not use units 59, 79, or array 21. If this section is discussing a different intact fuel assembly model, provide the correct input deck(s). These units are also specified in Section Y.6.4.2.B. Since these units are not part of the provided input deck, staff needs to ensure that the correct model was evaluated by the applicant.

This information is necessary to determine compliance with 10 CFR 72.236(a) and (c).

**RAI 7-2:** Revise Section Y.6.3.1 of the SAR and associated CSAS25 input files to ensure that the correct input file enrichments are provided and used by the code.

In Section Y.6.3.1, under the damaged fuel assemblies model assumptions, the lattice average enrichment is stated to be 4.1 wt% U-235, and the damaged row of fuel is modeled with a peak enrichment of 4.7 wt% U-235. However, the included models Y.6.6.3 and Y.6.6.4 have different enrichments listed (i.e., 4.8, 4.4 and 3.7 wt% U-235). If these assumptions used different models than those included in Appendix Y, please provide the correct inputs. Otherwise, explain the discrepancy. Since these parameters were not used in the provided input decks, staff needs to ensure that the correct models were evaluated by the applicant.

This information is necessary to determine compliance with 10 CFR 72.236(a) and (c).

**RAI 7-3:** Revise Section Z.6.4.2.B of the SAR and associated CSAS25 input files to ensure that the optimum moderator densities were evaluated over the full potential range of densities.

Section Z.6.4.2.B states that in order to determine the optimum internal moderator density the internal moderator densities were varied between 60% and 100%. Since there is no discussion in this section on why lower densities were not considered, provide an explanation or justification as to why these lower densities are not evaluated for the 37PTH DSC. Since this parameter was not evaluated for internal moderator densities less than 60%, it is unclear if the optimum moderator density is as stated in the evaluation.

This information is necessary to determine compliance with 10 CFR 72.236(a) and (c).

**RAI 7-4:** Revise the calculation provided in Section Z.6.6.2.4 to ensure that it is the correct input file for this amendment application.

In Section Z.6.6.2.4, "Most Reactive Damaged-WE 17x17", the input file lists this run as "NUHOMS 37PTH CoC 1004 Amendment 12." Clarify if this input file listing is correct. This application is for Amendment No. 13. This may be a typo, however, it is possible that the incorrect input file was submitted with the application.

This information is necessary to determine compliance with 10 CFR 72.236(a) and (c).

## 8.0 Confinement Evaluation

**RAI 8-1:** Clarify the components providing the redundant sealing of the confinement system and the sealing procedures employed in the test port located in the outer top cover plate.

Sections Y.7.1.1 and Z.7.1.1 describe the procedures for installing the outer top cover plate and optional usage of the test port located in the outer top cover plate. These sections state that the outer top cover plate provides redundant sealing and that a test port located in it could be used to leak test the inner top cover plate. After completing this leak testing, a test port plug is threaded into the outer top cover plate and seal welded in place. However, the staff does not have reasonable assurance on whether the test port plug is part of the redundant sealing, since it is not specified as part of the redundant sealing. A clarification of the components providing redundant sealing and the respective sealing procedures is needed to accept the provision of redundant sealing.

This information is necessary to satisfy the regulatory requirements of 10 CFR 72.236(e).

**Observation TS-1:** Under Technical Specification (TS) 5.2.4.c, the applicant provided a reference for the leak testing procedures performed to the confinement boundary after loading the DSC. For the NUHOMS<sup>®</sup>- 69BTH and NUHOMS<sup>®</sup>- 37PTH configurations, the confinement boundary is tested through two procedures in order to meet the leak tight criteria per ANSI N.14.5. The first procedure is performed to test the inner bottom cover plate, the canister shell and associated welds during fabrication and the second procedure tests the remaining components of the confinement boundary. This second procedure is referred to in T.S 5.2.4.c. However, the first procedure performed to leak test the confinement boundary is not referred to in the TS or the CoC although it is part of the acceptance tests required for approval of this application.

**Observation 8-1:** The dates of the references to the ASME Code and the ANSI standard provided under sections Y.7.4 and Z.7.4 are inconsistent. Staff is not aware of any specific differences highlighted through the confinement evaluation that would depend on the usage of one specific version of these codes and standards. However, staff notes that TS 4.2.2 refers to the ASME Code 2004 Edition, with Addenda through 2006, as the applicable code for the NUHOMS<sup>®</sup>- 69BTH and NUHOMS<sup>®</sup>- 37PTH configurations.

**Observation 8-2:** The confinement evaluation chapter and the drawings provided refer to the vent and siphon ports as vent and siphon ports, but the chapters on acceptance tests and operating systems refer to those as vent and drain ports. Consistency in the reference to the components of the confinement boundary is necessary to ensure that the appropriate components are being tested according to their specific safety bases.

### References:

1. Transnuclear, Inc., "Application for Amendment 13 to Standardized NUHOMS<sup>®</sup> Certificate of Compliance No. 1004 for Spent Fuel Storage Casks, Revision 0," E-29954, February 9, 2011 (ML110460541).
2. U.S. Nuclear Regulatory Commission, NUREG-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," 1996.

3. Transnuclear, Inc., "Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 13 to Standardized NUHOMS® System, Response to Request for Supplemental Information (Docket No. 72-1004; TAC No. L24519)," E-31217, July 22, 2011 (ML11217A03).
4. U.S. Nuclear Regulatory Commission, NUREG-1650, "MATPRO -A Library of Materials Properties for Light-Water-Reactor Accident Analysis," Vol. 4, Rev. 2, 2001 (ML010330363).
5. U.S. Nuclear Regulatory Commission, NUREG-1650, "MATPRO -A Library of Materials Properties for Light-Water-Reactor Accident Analysis," Vol. 4, 1998 (ML070820423).
6. U.S. Department of Energy, DOE/RW-0472, Rev. 2, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages, September 30, 1998 (ML070780668, non-public).
7. U.S. Nuclear Regulatory Commission, Interim Staff Guidance #11 (ISG-11) "Cladding Considerations for the Transportation and Storage of Spent Fuel," Rev. 3, 2003.