December 13, 2004

Mr. Michael Mason
Chief Engineer
Transnuclear, Inc.
Four Skyline Drive
Hawthorne, NY 10532

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE TRANSNUCLEAR NUHOMS® HD HORIZONTAL MODULAR STORAGE SYSTEM (TAC NO. L23738)

Dear Mr. Mason:

By letter dated May 5, 2004, as supplemented July 6 and October 28, 2004, Transnuclear, Inc., (TN) submitted an application for NUHOMS® HD Certificate of Compliance (CoC) No. 1030. This application proposes a new horizontal modular storage system, designated the NUHOMS® HD. The staff has determined that additional information is required to assess compliance with 10 CFR Part 72. Enclosed is the staff's request for additional information (RAI) for the continued review of your request.

To the extent practicable, we request that TN respond to this RAI by providing a response to each item in the RAI. We would be willing to meet with you to discuss and clarify the enclosed RAI. Your response to the enclosed RAI is expected by February 18, 2005. If you are unable to meet the February 2005 milestone, you must notify us in writing, at least 2 weeks prior to February 18 of your new response date and the reasons for the delay. The staff will assess the impact of the new response date and issue a revised schedule.

Please reference Docket No. 72-1030 and TAC No. L23738 in future correspondence related to this request. If you have questions concerning this request, please contact me at 301-415-3781.

Sincerely,

/RA/

Mary Jane Ross-Lee, Senior Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-1030
TAC No. L23738

Enclosure: Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION
TRANSNUCLEAR, INC.
DOCKET NO. 72-1030

By application dated, May 5, 2004, as supplemented July 6 and October 28, 2004, Transnuclear, Inc. requested approval of the NUHOMS® HD Horizontal Modular Storage System. This request for additional information (RAI) identifies additional information needed by the U. S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application. The requested information is listed by chapter number and title used in the applicant's safety analysis report (SAR). NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems (SRP)," was used by the staff in its review of the application.

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

Chapter 1 General Information

1-1 Revise the drawings of the transfer cask to show:

  b. Either section A-A and associated material dimensions, whose slice is shown on Drawing No. 10494-72-19, or specify the location of section A-A on another drawing,
  c. The thickness of the resin in the bottom of the transfer cask,
  d. The thickness of the radial neutron shield and the stainless steel enclosing the water.

  Section A-A is needed in order to be able to determine the thickness of lead shield. Additionally, the dimensions of the resin material in the bottom of the transfer cask as well as the neutron shield thickness and steel casing are not specified in the drawings.

  This information is needed to satisfy the requirements of 10 CFR 72.11.

1-2 Clarify Drawing No. 10494-72-19, which appears to show lead shielding in the base of the transfer cask instead of the borated resin material. The application does not indicate that lead is contained in the transfer cask anywhere else except as radial shielding.

  This information is needed to satisfy the requirements of 10 CFR 72.11.

1-3 Revise the application to provide a material specification for the Vyal B resin material in the DSC.

  This information is needed to satisfy the requirements of 10 CFR 72.11.

1-4 Revised Table 1-1 to indicate the name(s) and the technical specifications to indicate the name of the Metal Matrix Composites to be used for criticality control.

  This information is needed to satisfy the requirements of 10 CFR 72.11.

1-5 In Drawing Nos. 10494-72-1 and 10494-72-15, justify the use of Non Code,(i.e., commercial grade) material for the many components of the DSC and TC.

The specification for the plates, bolts, and pipes is shown in the PART LIST but the material is Non Code.

This information is required by the staff to assess compliance with 10 CFR 72.11 and 10 CFR 72.236(b).

1-6 Editorial: Drawing No. 10494-72-10, it appears that Section U - U should have been Section Y-Y.

Drawing No. 10494-72-19, Section AA - AA is not shown on the drawing.

Section 1.1 - should be OS187H TC, not OS817H.

Chapter 2 Principal Design Criteria

2-1 Justify the assumption for calculating the maximum internal pressure in the NUHOMS 32PTH DSC, that 1% of the fuel rods are damaged for normal conditions and up to 10% of the fuel rods are damaged for off-normal conditions.

Conservatively, the basis for calculating the maximum internal pressure of the DSC should be based on 32 intact assemblies.

This information is required by the staff to assess compliance with 10 CFR 72.236(l).

2-2 Revise Section 2.1.1 to provide a definition(s) of damage fuel that is consistent with the guidance in Interim Staff Guidance -1, Revision 1 (ISG-1, rev.1).

In accordance with 10 CFR 72.236(c), the spent fuel must be maintained subcritical under credible conditions. Further, 10 CFR 72.236(m) seeks to ensure safe fuel storage and handling and to minimize post-operational safety problems with respect to retrievability of the fuel from the storage system.

2-3 Revise the table in Chapter 2, entitled, “Minimum ^{10}B areal density” to include the volume percent of boron carbide in all of the neutron absorbers.

The high loadings of boron carbide in the aluminum may cause the absorbers to become brittle, similar to a ceramic, and thus, affect the durability of the Metal Matrix Composites (MMCs) neutron absorber.

This information is required by the staff to assess compliance with 10 CFR 72.236(c).

2-4 Revise Section 2.1.2 (and Chapter 4, Thermal) to state that the number of thermal cycles that the cladding experiences is less than 10.

Thermal cycling (repeated heatup/cooldown cycles) can enhance the amount of hydrogen that eventually re-precipitates in the form of radial hydrides. The extent of the formation of radial hydrides is dependent on many factors including the maximum temperature, change in temperature, number of thermal cycles, applied stress, hydrogen concentration, and solubility of hydrogen in the material. As stated in ISG-11, Rev. 3, the formation of radial hydrides in spent fuel cladding can be minimized by restricting the change in cladding temperatures to less than 65°C and minimizing the number of cycles to less than 10.

2-5 Editorial: Verify the statement on page 2-9, Section 2.2.9, “Ambient Variations (including solar insulation).”

It appears that the bullet should read “Ambient Variations (including solar insolation).”

This information is needed to satisfy the requirements of 10 CFR 72.11.

Chapter 3 Structural Evaluation

3-1 The 32PTH DSC stability criteria (i.e., allowable buckling loads) stated in Section 3.1.2.1.2 are not acceptable. Revise the application to calculate the critical loads for buckling of the DSC shell and the basket structure. The buckling evaluation should consider elastic, plastic, and local buckling, if applicable. Reasonable safety factors for the allowable buckling loads should be provided to take into account material and geometrical imperfections.

The application stated that: “The acceptance criteria (allowable buckling loads) are taken from ASME Code, Section III, Appendix F, paragraphs F-1341.3, Collapse Load.” Buckling is a stability consideration. Collapse load evaluation for a given combination of loads on a given structure is to ensure the strains or deflections of the structure are acceptable for load carrying purposes. Thus, collapse load evaluation is a strength consideration and cannot be used to substitute a buckling evaluation.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-2 Provide the basis and justification for applicability of the temperature dependent material properties shown in Table 3-6 and Table 3-7.

The maximum temperature of the HSM-H concrete surface exceeds the temperature limits in the ACI 349, A.4.2. The temperature dependent material properties of the HSM-H concrete are provided in Tables 3-6 and 3-7. Provide relevant information or the pages of the references to show applicability to the HSM-H concrete.

This information is required by the staff to assess compliance with 10 CFR 72.11 and 10 CFR 72.236(b).

3-3 Justify in Appendix 3.9.1, Table 3.9.1-7, why the allowable stress intensities for the end closure welds were based on the material tensile stress, when the end closure welds are partial penetration welds and the allowable stresses should be based on the material shearing stresses. In addition, on page 3.9.1-55, Lifting Block Weld Stresses, allowable stresses appear to be based on the material yield stresses when it should be revised to be based on the shearing stresses.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-4 The collapse load approach to evaluate buckling loads of 32PTH DSC fuel basket plates and support rails is not acceptable (See Chapter 3, RAI 1). Given the fact that the fuel basket plate supports are fusion welded to the stainless steel fuel compartments, it is not clear why a small three-dimensional ANSYS finite element model is needed to calculate the buckling load. (Section 3.7.1.1.2) In addition, the drawing shows the basket plates are aluminum (B209) not SA-240 Type 304 stainless steel. However, the application stated that a buckling analysis of the full size basket was conducted for a 45-degree azimuth basket orientation drop (Page 3.9.1-29). The inelastic finite element analysis converged up to a load of 86.4g. The design g-load is 75g and thus the safety factor is approximately 86.4g/75g=1.15. Please justify that this small safety factor is adequate to prevent buckling of the basket components during a 75g side drop of the transfer cask. Confirm that the buckling analysis was preformed based on the correct materials and material properties.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-5 Provide justification for a concurrent 30 psig internal pressure in the evaluation of the structural adequacy of the OS187H Transfer Cask inner shell with respect to buckling.

Section 3.7.3.4, Transfer Cask Inner Containment Buckling Analysis, stated that: the loads considered for buckling analysis includes an internal pressure of 30 psig and a 75g top and bottom drop load in both hot (115 F) and cold (-20 F) ambient environments. The Tech Spec only specifies a helium backfill pressure of 2.0±1.0 psig. Resolve the discrepancy and evaluate the effects of internal pressure on buckling loads for the shell.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-6 The OS187H transfer cask impact analysis presented in Appendix 3.9.7 is based on the Reference: “Structural Design of Concrete Storage Pads for Spent Fuel Casks,” EPRI Np-7551. NRC has not endorsed the EPRI report and the analysis results (e.g., g-loads and damages to the cask) may be unacceptable. The transfer cask dynamic impact analysis should be revised and performed using an approach such as the nonlinear finite element code DYNA3D on a cask-pad-soil finite element model as described in NUREG/CR-6608. Note that the g-loads probably will be different as the result of the revised evaluation and the cask closure bolt analysis will have to be redone.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-7 Perform a top end corner drop dynamic impact structural analysis of the transfer cask to show the adequacies of the top inner shell weld to the top forging.

The inner shell is attached to the top flange (forging) by fillet welds (Drawing No. 10494-72-19). It is unclear whether the welds have adequate strength to prevent separation of the inner shell from the top flange for a top end corner drop of the transfer cask.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-8 Perform structural analysis to demonstrate the canister rails (3.0 inches wide and 0.12 inches thick) are adequately strong to support the DSC canister without significant deflections.

The canister rails are supported by the cask inner shell which is in turn supported by chemical lead shield. Appendix 3.9.2, page 3.9.2-19, stated that: " Pressures applied in the radial direction in the 3-dimensional finite element model are based on cosine distributions. These pressure distributions simulate the internal cask contents applying pressure to the inner cask wall." This pressure distribution may not be conservative for the canister rails because the applied loads are rather concentrated and the lead backed inner shell may not have sufficient stiffness to prevent significant deflection of the rails along the length of the cask.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-9 Section 3.7.3.4, revise the buckling analysis of the transfer cask inner shell by including the lateral pressure of the lead on the inner shell in an end drop impact condition.

The lateral pressure of the lead on the inner shell is significant at the impacted end and thus must be considered for the shell buckling analysis.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-10 Justify the handling loads used for the structural analysis of the transfer cask trunnions (e.g., DW+0.5g axial+0.5g transverse+0.5g vertical, Appendix 3.9.5, page 3.9.5.1).

The handling loads for the transfer cask trunnions should include the loads during TRANSFER OPERATIONS. Provide the references for the loads assumed in analysis.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-11 Appendix 3.9.8, page 3.9.8-1, stated that: "Damaged fuel assemblies may be only stored in the peripheral compartments of the NUHOMS 32PTH DSC." However, Chapter 2, page 2-1, stated that: "Damaged fuel assemblies shall be placed into the sixteen inner most basket fuel compartments, as shown in Figure 2-2, which . . ." Resolve the apparent contradiction. Also, the temperatures effects should be considered on the allowable stresses of the cladding.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-12 Justify that the maximum g-load acting on the damaged fuel rod subjected to 1 foot end drop is 30g.

The basis for the 30g end drop g-load is not clear. If this is the design basis loading considered for the NUHOMS HD System, please clearly state it in the SAR.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-13 Justify the fuel rod moment of inertia (MI) used for the side drop fuel rod structural integrity evaluation. The fuel rod moment of inertia has been assumed to be equal to the net tube MI plus net fuel MI.

The application stated that: "Where it is conservatively assumed that the net tube MI is equal to one half of the total tube MI, and the net fuel MI is equal to one half of the total fuel MI." The basis to include fuel MI in the stress calculation is not conservative and subject to challenge. Please provide the justification for the assumption.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-14 Describe the controls during LOADING OPERATIONS and TRANSFER OPERATIONS, such that the fuel rods will either accelerate from 0 initial velocity to a maximum velocity of 5 MPH, or decelerate from a maximum velocity of 5 MPH to 0 final velocity.

Under normal conditions, the structural integrity of the fuel assemblies is calculated based on the kinetic energy of the mass and the velocity. It is thus important to ensure the maximum velocity of the fuel assemblies does not exceed the 5 MPH limits as assumed.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-15 Demonstrate that the Vyal B resin used in the transfer cask is non-reactive with cask internal components and will remain adherent when exposed to the various environments during loading operations. The manufacturer's data/specification should be submitted to support your argument.

This information is required by the staff to assess compliance with 10 CFR 72.236(g).

3-16 Provide a brittle fracture analysis to evaluate the fracture toughness of the transfer cask bolts.

The statement concerning brittle fracture is not a concern because all the components comprising the transfer cask are all stainless steel, is not a true statement. As stated in 3.1.1.3 of the SAR, non-stainless steel members include the carbon steel closure bolts. The SAR does not have any supporting analysis or evaluation to demonstrate that these bolts will not fracture while in-service.

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

3-17 Provide correct irradiated material properties; in particular, the ultimate strength, yield strength, and oxide thickness used for the fuel rod integrity during drop scenario at 750°F. Additionally, show that they are bounding for the irradiated fuel type chosen for the analysis. Further, clarify if an extrapolation was done to obtain the mechanical properties at 750°F, and if so, provide the calculations to obtain the mechanical properties.

The staff has reviewed a data-base of properties, EPRI reports, and technical papers that do not agree with the properties used for the applicant's inputs for the structural analysis on rod integrity in Section 3.5.3. Furthermore, the estimate of the cladding corrosion and corresponding wall thinning computed should be approximately a wall thinning of ~120/1.76 = ~77 microns, using a conservative value for the pilling-bedworth ratio. Based on data, the 77 microns could be low for high burnup Zircaloy-4 fuel cladding that has absorbed up to 600 ppm (total hydrogen content) of hydrogen during reactor operation. The properties in Section 3.5.3 appear to be acceptable for low burnup fuel, but not for high burnup fuel cladding.

In accordance with 10 CFR 72.236(c), the configuration of the spent fuel geometry should be maintained to assure subcriticality under all credible normal and design basis events of storage. Further, 10 CFR 72.236(m) seeks to ensure safe fuel storage and handling and to minimize post-operational safety problems with respect to retrievability of the fuel from the storage system.

3-18 Verify the statement in Tables 3.9.2-2 to 3.9.2-7 which reads "Allplied Loads."

It appears that the heading should read "Applied Loads."

This information is needed to satisfy the requirements of 10 CFR 72.11.

Chapter 4 Thermal Evaluation

4-1 Provide additional justification for coupling the nodes at the location of the "two middle rails to represent the contact area at those two locations," as stated on page 4-11, second paragraph, last sentence.

It seems that the coupling of the nodes would indicate that the two bodies are welded together, however, it is stated earlier in the first sentence of the same paragraph that the "DSC shell rests on the four rails in the transfer cask during transfer operations." Taking credit for the perfect contact will aid in heat transfer from the DSC shell to the rails, which will then act like cooling fins. If the perfect contact assumption cannot be justified, an equivalent resistance or gap can be placed between the DSC shell and the four rails.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

4-2 Justify the assumption "no gap is considered between the paired aluminum and poison plates," as stated on page 4-17, second paragraph, first bullet, second dash, last sentence.

The lack of this gap is evident in Figure 4-13, Detail B. In the previous analysis on page 4-6 of the equivalent conductivity, the thermal resistance of the contact gap is also neglected. Neglecting this gap effectively models a perfect contact between the aluminum and poison plate. The two pieces of metal are touching each other, which does not guarantee a perfect contact. If the perfect contact assumption cannot be justified, please modify your models to account for this gap.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

4-3 Explain the issues with stabilizing the Ansys run, which is described on page 4-69, second to last paragraph.

An instability issue could be a sign for a non-converged or incorrect solution.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

4-4 Provide additional justification for the use of the Nu_l correlation for laminar flow, which is found on page 4-48 and page 4-63.

As per reference [5] on page 6-60, it is stated that "Both cylinders are assumed to be horizontal, and their axial length is assumed to be very much greater than their mean gap size L." The mean gap size was calculated to be 5.0625 inch. The axial length was found on Drawing No. 10494-72-21, Detail W to be 11.00 inch. Therefore, the axial length is approximately only twice the length of the mean gap size. This calculation is in conflict with the assumptions provided for the Nusselt number correlation. Provide either experimental data or a computational fluid dynamics (CFD) model to justify the Nusselt numbers calculated in Table 4-14, 4-15, 4-16 and 4-20. With this analysis, use the appropriate internal heat flux and external convection and radiation boundary conditions.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

4-5 Explain the discrepancy between the values in Table 4-18, third table, in the column "Calculated k_eff" versus the values in the column "k_eff in Model." These two values are not similar. The values in the model appear to be non-conservative at some times, while conservative at other times, and address the discrepancy between the calculated Nusselt number in Table 4-18, and the calculated Nusselt number in Table 4-14. The effect of a improperly calculated Nusselt number will affect the results.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

4-6 Clarify the statement on page 4-33, section 4.5.1.4, first paragraph, third sentence, that "A margin of about 20 °F is considered for conservatism in determining the time limit."

The maximum fuel cladding temperatures are summarized in Table 4-8". In Table 4-8, Procedure B, the fuel assembly temperature is within 8 °F of the allowable temperature limit.

This information is needed to satisfy the requirements of 10 CFR 72.11.

4-7 Provide additional information regarding the welds between the 7/8" SQ. STOCK STAINLESS STEEL and the 3/8" STOCK STAINLESS STEEL found in Drawing 10494-72-12, Item 27, Basket Rail A90 and Drawing 10494-72-11, Item 19, Basket Rail A180.

In particular, these welds should be full penetration welds which extend through the entire axial length. This is necessary in order to assume a perfect contact in the basket rail.

This information is needed to satisfy the requirements of 10 CFR 72.11.

4-8 Justify that your fire transient thermal analysis is conservative. Justify neglecting the ends of the cask or entire cask body in the fire transient analysis.

The conservatism could be explained by stating that water in the annular region will conduct the heat from the fire better than air, and that will create higher temperatures, which is conservative. On page 4-23, it is stated that "the liquid neutron shield (water) will be released at high temperatures (~417 ^{o}F) when its saturation pressure exceeds the set point of the pressure relief value (40 psig)." At 40 psig (54.7 psia), the saturation pressure of water is 286.9 ^{o}F. However, radiation heat transfer will have a significant effect after the water has escaped from the neutron shield region. The heat transfer due to radiation in the annular region will affect the overall temperature during the transient analysis.

Typically, the end volumes have a greater amount of surface area (per volume) exposed to the external heat source. This effect will increase the heat transfer into the cask ends. The ends of the cask are composed primarily of metal, unlike the center annular regions which have numerous air gaps and a liquid neutron shield. The difference in materials will allow heat to transfer into the cask easier in the ends of the cask, than in the center region of the cask.

This justification is needed to satisfy the provisions of 10 CFR 72.11 and 10 CFR 72.236(f).

The following questions are editorial in nature (completeness and accuracy of all information provided to the NRC is required per 10 CFR 72.11.).

4-9 Fill in the box for the maximum allowable temperature for the liquid neutron shielding at 40 psig in Table 4-1.

NRC staff believes the temperature is 286.9 ^{o}F.

4-10 Verify the statement on page 4-2, first paragraph after two bullets, second sentence, which reads "The loading requiremenst described in Section 4.3.1.3".

It appears that the sentence should read "The loading requirements described in Section 4.3.1.3".

4-11 Verify that the 2nd and 3rd column titles in the Lead material property table on page 4-5 should be reversed.

4-12 Verify the statement on page 4-7, Section 11, second paragraph which contains "(0.084 ibm/in^3)."

It appears that the sentence should contain "(0.084 lbm/in^3)".

4-13 Verify the statement on page 4-37, Section 4.6.2, which contains "temperature before backfillig."

It appears that the sentence should contain "temperature before backfilling."

4-14 Verify the statement on page 4-45, last paragraph, which contains "simulate heat transfer by radiation and convection."

It appears that the sentence should read "simulate heat transfer by radiation and conduction."

4-15 Verify the statement on page 4-47, second paragraph, first sentence, which contains "The calculated transverse effective conductivties for vacuum."

It appears that the sentence should contain "The calculated transverse effective conductivities for vacuum."

4-16 Verify the statement on page 4-48, the fourth equation reads, which contains "N_{COND}."

It appears that it should read "Nu_{COND}."

4-17 Verify the equation on page 4-48, which contains "(ln D_o/D_i)."

It appears the equation should contain "ln (D_o/D_i)." The equation in reference [5], page 6-60 appears to have a typographical error, which was copied into the SAR.

4-18 Add subscripts to g_1 and g_2 on page 4-49 Equation 4.9-3,

4-19 Verify on page 4-63, second paragraph that Reference [12] refers to Reference [5].

4-20 Verify that the units of h_c should be changed from (BTU/hr-ft^2-°F) to (BTU/hr-in^2-°F) in table 4-19.

4-21 Verify that Detail A and Detail B are reversed in Figure 4-14.

4-22 Remove the dash from the caption in Figure 4-23.

This dash could be mistaken as a negative sign, instead of a dash mark.

Chapter 5 Shielding Evaluation

5-1 Revise Table 5-7 to provide assembly materials and masses for the Framatome Cogema MK BW 17x17 fuel assembly and ensure that the values provided for the Westinghouse fuel assemblies are appropriate for all the Westinghouse fuel assemblies to be stored. Note that the bounding source term comes from the Framatome Cogema MK BW 17x17 fuel assembly, however information on the fuel assembly materials was not provided.

This information is needed to satisfy the requirements of 10 CFR 72.104.

5-2 Clarify whether the specific power, provided on page 5-3 for the burnup cycles, is 25 MW/MTU or 25 MW/assembly.

This information is needed to satisfy the requirements of 10 CFR 72.104.

5-3 Revise the application to explain why the top dose rates for transfer casks during welding and transfer-storage decrease as water is drained from the transfer cask. During decontamination the transfer cask is essentially completely full of water and during welding water is drained from the canister, but not the region between the transfer cask and canister. For canister transfer to storage, the transfer cask and canister system is completely dry.

This information is needed to satisfy the requirements of 10 CFR 72.104.

Chapter 6 Criticality Evaluation

6-1 Provide surveillance requirements for verifying the appropriate soluble boron concentration in the pool.

The applicant takes credit for the soluble boron in the spent fuel pool; however, no surveillance is identified to ensure the appropriate concentration. Given the small margin to the USL, the soluble boron concentration needs to be verified prior to loading the casks.

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

6-2 Provide acceptance testing requirements to verify the basket type placed into the cask.

Table 6-1 identifies eight basket types with increasing areal densities of neutron absorber loading with corresponding required minimum soluble boron concentrations. Neither Chapter 8 or Chapter 9 of the SAR provide requirements for verifying that the appropriate basket type has been placed into the cask. Borated aluminum basket types B and C have the same fixed neutron absorber thicknesses but different neutron absorber loadings. Therefore, it is not apparent that the appropriate basket can be verified solely through a visual inspection. Given the small margin to the USL, the appropriate basket type needs to be verified prior to loading the casks.

This information is required by the staff to assess compliance with 10 CFR 72.236(c).

6-3 Provide information on how the analytical studies of the fuel compartment size and the fuel compartment thickness were performed. Identify manufacturing tolerances on structures affected by the analytical studies, including the fuel compartment box thickness, the egg crate structure thickness, and the egg crate structure fuel compartment size.

On page 6-15, the applicant indicates an analysis was performed where the fuel compartment size was varied from 8.650 inches to 8.750 inches. The minimum fuel compartment size was found to be most reactive. However, details are not provided on how the remaining basket structures are modeled. The applicant then states that an analysis was performed on the fuel compartment box thickness. The applicant indicates that the most reactive configuration is the nominal thickness.

The NRC staff performed an independent analysis of the effect of the compartment box thickness and found that increasing the compartment box thickness from the nominal 0.1875 inches to 0.2275 inches increased the reactivity several times the statistical uncertainty. The NRC model displaced water inside the fuel compartment box with the change in the compartment box thickness, while the rest of the cask model remaining unchanged. The NRC notes that the optimum compartment box thickness produces a fuel compartment box size comparable to that found by the applicant.

This information is required by the staff to assess compliance with 10 CFR 72.124(a).

Chapter 7 Confinement Evaluation

7-1 The staff suggests that the applicant change the wording in SAR Sections 7.2.1, 7.3.1, and 7.4.1 in the Safety Analysis Report from “...required per ISG-5" to “... in accordance with ISG-5.”

Interim Staff Guidance is not a requirement in the Federal Regulations and should not be referred to as such.

7-2 SAR Sections 7.1.2, 7.2, 7.3.2 should be changed to reflect that the confinement boundary has no credible leakage, not that it is “leak-tight.”

The term “leak-tight” is defined by ANSI N14.5 - 1997, and implies that leak testing has been performed on the confinement boundary as a whole.

7-3 Clarify how the end caps fit onto the basket fuel compartments to confine damaged fuel assemblies. Additionally, clarify whether the damaged fuel would be retrievable by normal means (grapple and crane).

It is not clear how the end caps are attached to the fuel basket. Additionally, it is not clear whether the damaged fuel assemblies will be retrievable by normal means with the use of end caps as opposed to a damaged fuel can. Current staff guidance contained in Interim Staff Guidance - 1, "Damaged Fuel," states that damaged fuel should be placed into a damaged fuel can that is retrievable by normal means.

Retrievability is addressed in 10 CFR 72.236(h), 72.236(m), and in Interim Staff Guidance - 2, 'Fuel Retrievability.'"

Chapter 9 Acceptance Tests and Maintenance Program

9-1 Indicate the correct particle size for the boron carbide in all of the neutron absorbers to be used in the matrix of the absorber

The applicant has stated in Section 9.5.3.1 of the SAR that the boron carbide particles for the metal matrix composites considered have an average size in the range 10-40 microns. This Section also states that no more than 10% of the particles are over 60 microns. This is not true for Boral according to the fabricator. Boral has an average particle of approximately 85 microns.

This information is needed to satisfy the requirements of 10 CFR 72.11.

9-2 Indicate, in tabular form, the boron credit, minimum effective areal density, and the boron carbide volume fraction for each neutron absorber (i.e., Boral, Metamic, etc.) and provide qualification test data that shows that the neutron absorbers will maintain their durability in the range of 15-35 volume percent boron carbide.

Staff has reviewed and approved data at the 15 volume percent level for Metamic, but not at higher volume percent to ensure durability over the service life. Staff has no test data on Boral related to this degradation that will ensure the durability of this material over the service life.

This information is needed to satisfy the requirements of 10 CFR 72.11.

9-3 Remove the justification for not conducting thermal and corrosion testing for qualifying a neutron absorber.

Staff has reviewed the literature and some proprietary data, in detail, and does agree with the applicant that accelerated radiation testing need not be done on newer absorbers. However, staff does not agree with the applicant that thermal and corrosion testing should not be conducted. A review of the literature shows that the few tests do not consider synergistic effects of pool chemistry, temperature, galvanic coupling, etc. There is not enough published data to unequivocally state that thermal and corrosion testing should not be done. This argument is confirmed by the applicant's one referenced technical paper footnoted to support the argument.

This information is needed to satisfy the requirements of 10 CFR 72.11.

Chapter 10 Radiation Protection

10-1 Either revise the application to determine the occupational dose rates for the transfer cask with the neutron shield tank empty or ensure that the transfer cask can not be moved with the neutron shield tank empty. It appears from Section 8.1.1.4 (1) that the transfer cask will be downended and transferred, with the neutron shield tank empty.

This information is needed to satisfy the requirements of 10 CFR 72.104.

Chapter 11 Accident Analysis

11-1 Revise the accident dose calculations for the cask drop, in section 11.3.1, to show that the off-site dose is less than 5 rem. Note that the evaluation in the application shows the maximum dose rate at the surface, but the acceptance criteria is an off-site dose of 5 rem.

This information is needed to satisfy the requirements of 10 CFR 72.106(b).

11-2 Revise section 11.3.3.5 to show that scabbing of the HSM-H concrete due to a missile impact will not cause the off-site dose to exceed 5 rem.

This information is needed to satisfy the requirements of 10 CFR 72.106(b).

Chapter 12 Operating Controls and Limits

12-1 Revise the technical specifications to include a provision for measuring dose rates at various locations around either the loaded transfer cask or the HSM-H. Dose rate measurements after loading, ensure the cask is loaded properly and that the facility will meet the 25 mrem off-site dose requirements of 10 CFR 72.104(a).

This information is needed to satisfy the requirements of 10 CFR 72.104.

12-2 Clarify whether the reference to Table 2-3 in TS 12.2.1 (c) is correct or whether the reference should be Table 12-3.

This information is needed to satisfy the requirements of 10 CFR 72.11.

12-3 Revise the application to ensure consistency between each of the TSs and between all of the TS and the fuel storage limits specified in Section 2. Note that Table 12-1 specifies the maximum burnup as 60 GWD/MTU but Table 12-4 shows maximum burnups of > 60 and 60.1 GWD/MTU. Additionally, Table 12-1 states that the minimum cooling time is five (5) years, while Table 2-3 shows the minimum cooling time of seven (7) years. Also there is a discrepancy between the maximum gamma sources in tables 2-3 and 12-4. Note that if a burnup of greater than 60 GWD/MTU is desired, the shielding evaluation should be revised to reflect the higher burnup.

This information is needed to satisfy the requirements of 10 CFR 72.104.

12-4 Justify why a helium leakage test is not performed on the TC to ensure no leakage of helium.

Tech Spec SR 12.3.1.3.1, Surveillance only requires one-time verification of the OS187H cavity/annulus helium backfill pressure (e.g., 2.0± 1 psig) after the installation of the TC lid. It is not clear why this surveillance requirement is adequate to ensure no leakage of helium during TRANSFER OPERATION. Since the thermal analysis assumes a helium gap between the DSC and the TC, it is important to ensure no helium leakage during TRANSFER OPERATION.

This information is required by the staff to assess compliance with 10 CFR 72.236(f).

12-5 Supplement the SAR to include tests to prove the concrete capability for elevated temperatures (or reduction of strength) for the HSM-H concrete structure.

Temperature requirements of ACI 349, A.4.2 state that the concrete surface temperatures shall not exceed 350 °F, for accidents or any short term period (200 °F for normal operations). Because the HSM-H concrete temperature is higher than ACI 349 Code allowables, tests should be performed for the cement type and aggregates selected to confirm concrete capability (e.g., compressive strength and shielding capability).

This information is required by the staff to assess compliance with 10 CFR 72.236(b).

12-6 Revise Tech Spec. 12.5.3.1 Transfer Cask Lifting Heights to include applicable LOADING OPERATIONS.

LOADING OPERATIONS include all licensed activities on a 32PTH DSC while it is being loaded with INTACT or DAMAGED FUEL ASSEMBLIES, and in a TRANSFER CASK while it is being loaded with a 32PTH DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. Based on this definition of LOADING OPERATIONS, it seems necessary to address cask lifting heights for LOADING OPERATIONS as well.

This information is required by the staff to assess compliance with 10 CFR 72.11 and 10CFR 236(b).

12-7 Editorial: Table 12-2, Assembly length should be note (3).