March 5, 2001

Mr. Robert M. Grenier
President and Chief Operating Officer
Transnuclear West Inc.
39300 Civic Center Drive
Suite 280
Fremont, CA 94538

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING APPROVAL TO STORE SPENT NUCLEAR FUEL IN THE ADVANCED NUHOMS® STORAGE SYSTEM (TAC NO. L23203)

Dear Mr. Grenier:

By letter dated September 29, 2000, as amended on October 4, and November 1, 2000, Transnuclear West Inc. (TN West) submitted an application for a Certificate of Compliance to store spent nuclear fuel in the Advanced NUHOMS® Storage System. The staff has determined that additional information is required to assess compliance of the application 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 West 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 May 18, 2001. If you are unable to meet the May 2001 milestone, you must notify us in writing, at least 2 weeks prior to May 18 of your new response date and the reasons for the delay. The staff will then assess the impact of the new response date and issue a revised schedule. Please reference Docket No. 72-1029 and TAC No. L23203 in future correspondence related to this request.

If your response contains proprietary information please include a complete separate non-proprietary version of the response. Please direct any questions concerning this request to me at 301-415-8538.

Sincerely,

/RA/

Timothy J. Kobetz, Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1029

Enclosure: Request for Additional Information

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President and Chief Operating Officer
Transnuclear West Inc.
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Sincerely,

(Original Signed by:)

Timothy Kobetz, Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1029
Enclosure: Request for Additional Information
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TRANSNUCLEAR WEST INC.
DOCKET NO. 72-1029
TAC NO. L23203

REQUEST FOR ADDITIONAL INFORMATION

This document, titled Request for Additional Information (RAI), contains additional information requirements identified by the U.S. Nuclear Regulatory Commission (NRC) staff during its review of Transnuclear West Inc. (TN West) application to store spent nuclear fuel in the Advanced NUHOMS® Storage System.

Each individual RAI describes information needed by the staff for it to complete its review of the application and determine whether TN West has demonstrated compliance with the regulatory requirements. Where an individual RAI relates to TN West's apparent failure to meet one or more regulatory requirement or where an RAI specifically focuses on compliance issues associated with one or more specific regulatory requirement (e.g., specific design criteria or accident conditions), such requirements will be specified in the individual RAI.

Note that RAI items may refer to the Spent Fuel Project Office's (SFPO) Interim Staff Guidance (ISG). The ISG was developed as a result of management decisions on several key issues related to the review and approval of spent fuel storage systems and represents positions discussed in meetings with the Nuclear Energy Institute. The ISG will be incorporated into the next revision of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems (SRP)."

Chapter 1 General Information

1-1 Specify which transfer casks will be used to perform on-site transfer of the dry shielded canister (DSC).

Section 1.2.1.3.1 of the Safety Analysis Report (SAR) states that "the OS-197, MP-187, or any other NRC licensed transfer or transportation cask of sufficient size and payload capacity is acceptable for use with the Advanced NUHOMS® System subject to a site specific safety evaluation prior to the first usage." However, the SAR does not provide a methodology used to evaluate that the transfer or transportation cask has the appropriate shielding, heat transfer, structural integrity, and criticality characteristics to be used with the DSC-24PT. The application references previous evaluations of the OS-197 and MP-187 performed for the Standardized NUHOMS® System and the Rancho Seco site-specific application but does not demonstrate that those evaluations bound the DSC-24PT1

This information is required by the staff to assess compliance with 10 CFR 72.24(c)(3), to assess whether all structures, systems, and components important to safety will satisfy the design bases with an adequate margin of safety.

Chapter 2 Principle Design Criteria

2-1 Remove the statement in Sections 2.1.1 and 3.1.1.1, that refer to the storage of 'Greater than Class C' (GTCC) waste inside an advanced NUHOMS® Systems (AHSM).

Approval to store GTCC will not be addressed in a Certificate of Compliance, issued in accordance with 10 CFR Part 72, for the Advanced NUHOMS® Storage System. This changes is required by the staff to confirm compliance with 10 CFR 72.2.

2-2 Discuss design features of the Advanced NUHOMS® system to enhance decontamination and decommissioning.

The SAR did not contain sufficient detail with regard to decontamination and decommissioning of the Advanced NUHOMS® System. Include a discussion of the effect of the contamination limits stated in Section 12.5.2.4.b on potential off-site releases such as through the outlet vent at the top of the AHSM. The discussion should also include a statement on what affect the contamination level will have on the ability to decontaminate and decommission the concrete AHSM, the ISFSI pad, and the surrounding soils. Consideration of decontamination and decommissioning activities is required under 10 CFR 72.130 and 72.236(i).

Chapter 3 Structural

3-1 Justify the 392°F maximum concrete temperature for accident conditions given in Table 4.1-5. This temperature exceeds the allowable range of 0 - 350°F stated in the table.

Note 2 in Table 4.1-5 states testing will be performed to document that concrete compression strength will be greater than that assumed in structural analyses. The tests are to be on the exact concrete mix and are to acceptably demonstrate the level of strength reduction which needs to be applied, and to show that the increased temperatures do not cause deterioration of the concrete either with or without load. However, there is no discussion of what type of testing will be performed and why the testing is sufficient to confirm performance of the AHSM.   The test details are required for the staff to assess compliance with 10 CFR 72.146 (b).

3-2 Justify the use of the selected friction coefficients between the AHSM and the storage pad.

The friction coefficients determine the AHSM rocking/tipping and sliding responses. The AHSM sliding analyses are performed using a range of friction coefficients between the cask and pad surfaces. The range varies between 0.3 and 0.7. However, it is not clear why these values were selected nor if they are conservative. This information is required by the staff to assess compliance with 10 CFR 72.122(b)(2).

3-3 Provide the following clarifications for Table 11.2-2 of the SAR.

1. For Case 2, evaluate how a group of three AHSMs would respond to a design basis earthquake when only one is loaded with a 24PT1-DSC.

2. Explain why Cases 3 and 4 are more sensitive to the friction coefficient than Cases 1 and 2. For Cases 3 and 4 when the friction coefficient changes from 0.7 to 0.3 the X-displacement increases from 11.9 inches to 32.4 inches, however, similar friction coefficient changes for Cases 1 and 2 only result in an increase from 39.3 inches to 41.9 inches.

This information is required by the staff to assess compliance with 10 CFR 72.122(b)(2).

3-4 Discuss why the three sets of time histories each have a total duration of 40 seconds.

The discussion should include statements on what the affects a longer duration would have on the time histories or a statement stating why 40 seconds is sufficient for the time histories. This information is required by the staff to assess compliance with 10 CFR 72.122(b)(2).

3-5 Justify the use of a short term fuel cladding temperature limit of 1058°F for stainless steel cladding in lieu of 806°F as previously approved by the staff, or revise the allowable maximum temperature for the stainless steel cladding to 806°F.

Section 3.5.1.2.1 of the SAR references EPRI TR-106440 “Evaluation of Expected Behavior of LWR Stainless Steel Fuel in Long-Term Dry Storage” as the basis of the long term cladding temperature limit. However, this reference also states that dry storage at temperatures above 806°F should be discouraged because of the increased potential to sensitize the stainless steel.

The justification should address the effect of sensitization on the stainless steel cladding integrity and retrievability of the fuel assembly at temperatures above the short term allowable fuel cladding temperature limits for stainless steel cladding of 806°F.

Reference 3.33 of the SAR should also be provided along with a discussion on how the methods and data encompassed can be used to address the conclusions in Reference 3.30 of the SAR, specifically, that stress rupture by the development of a micron-sized pinhole leak is a possible failure mechanism for stainless steel cladding at temperatures above 806°F. Issues that should be addressed include:

1. The fuel cladding material’s microstructural properties and characterization after sensitization.

2. The material operating environment including maximum fuel rod internal pressure.

3. Fuel rod cladding susceptibility to the failure mechanism described in Reference 3.30 of the SAR during short term events.

4. Data and calculational methods, including applicable assumptions and codes, which address this failure mechanism.

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

3-6 Provide the San Onofre Nuclear Generating Station (SONGS) site-specific soil-structure interaction analysis.

The design basis response spectra of the Advanced NUHOMS® System design is based on the standard spectrum shape in NRC Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1, 1973, anchored at 1.5g zero period acceleration for the horizontal direction. The vertical design spectrum is set at 2/3 of the horizontal over the entire frequency range. The horizontal and vertical spectra are specified at the top of the basemat.

Nonlinear analyses were performed in Chapter 11 of the SAR to determine the maximum sliding and rocking/tipping response of the AHSM during a design basis seismic event. These analyses were based on acceleration time histories from the Taiwan earthquake in 1999, the Landers/Lucern earthquake in 1992, and the Tabas earthquake in 1978. However, the staff requires the analysis including the soils and structure interactions, the methodologies and computer model, and the specific acceleration histories that envelope the SONGS Unit 1 site parameters. This is required for the staff to assess compliance with 10 CFR 72.122(b).

3-7 Demonstrate that the LS-DYNA computer program has been benchmarked against actual test data using a quality assurance program in accordance with 10 CFR Part 72, Subpart G, or justify the use of this program for non-linear seismic stability analyses.

This is required for the staff to assess compliance with 10 CFR 72.122(b).

Chapter 4 Thermal

4-1 Specify which transfer casks will be used to perform on-site transfer of the dry shielded canister (DSC) and provide an analysis for each of the transfer cask designs to be used with the Advanced NUHOMS® System and provide design details or demonstrate why a previous analysis of the transfer casks bound its use with the 24PT1-DSC.

Section 1.2.1.3.1 of the SAR states that any NRC licensed transfer cask of sufficient size and payload capacity is acceptable for use. That statement has not been supported. The maximum heat load specified for the 24PT1-DSC is 14 kW. A previous calculation using a higher heat load for a different system may be bounding to determine the effects of that heat load on a transfer cask. However, the review is also focused on the response of the 24PT1-DSC components important to safety and the fuel cladding when subjected to the heat load and heat transfer characteristics associated with the system and particular transfer cask used. Additionally, Section 4 of the SAR provides an example which demonstrates that the MP 187 is not allowed for the content heat load in excess of 13.5 kW. This information is required by the staff to assess whether the system design can perform its function when subjected to the licensed content heat load under normal and accident conditions per 10 CFR 72.128(a).

4-2 Justify the assumption stated in Section 4.4.2.2 that the side and back surfaces of the AHSM should be modeled as adiabatic in order to simulate adjacent modules.

Applying the heat load generated in adjacent modules may be the more appropriate approach. The results of this calculation based on actual system arrangement will be used to determine compliance to 72.128(a).

4-3 When reformatting the SAR, ensure proper pagination.

Note that there are two pages identified as 4.1-3.

4-4 Clarify actual Pu enrichments in mixed oxide (MOX) fuel descriptions.

Fissile Pu enrichments listed on Page 5.2-2 (2.78, 3.05, 3.25 w/o) are different from those listed as footnote (1) of Table 2.1-2 (2.81, 3.1, 3.31 fissile Pu w/o), and Table 6.2-2 (2.84, 3.1, 3.31- fissile Pu w/o = 3.30, 3.65, 3.85 - total Pu w/o). Also, include the minimum fissile Pu enrichments in Table 12.2-1. Information must be presented accurately and in sufficient detail for the staff to assess compliance with 10 CFR 72.24(c)(3).

4-5 Include a discussion of MOX fuel He production effects on maximum nominal operating pressure and cladding failure.

Section 4.4.8 of the SAR refers to Table 4.4-9 and states that based on the information listed, the UO_2 assemblies are bounding for the analysis. However, the SAR does not include a discussion of the MOX rod void volume, a plot of gas generation over time, including He, and cladding strength which would provide a complete characterization of the condition of the contents stored. This information is required by the staff to assess whether the fuel cladding is protected against degradation that could lead to gross ruptures in accordance with 10 CFR 71.122(h)(1).

4-6 Clarify that a maximum heat load of 14 kW bounds the allowed contents of the 24PT1-DSC.

Section 1.2.1.1 of the SAR states that the 24PT1-DSC is designed for a maximum heat load of 14 kW. However, the staff found that the maximum heat load, based on allowable contents, is 16 kW. This information is required by the staff to assess compliance with 10 CFR 72.24(d)(1).

4-7 Justify the statement in Section 4.4.5 of the SAR that the conservative modeling approach precludes the necessity to perform thermal testing.

Section 4.4.5 of the SAR states that detailed, conservative evaluations were performed for heat transfer from the AHSM, OS 197 transfer cask and the 24PT1-DSC. However, as noted in RAI question 4-6 those evaluations could not be confirmed by the staff. In addition, the SAR did not provide design details for all transfer casks intended to be used with the 24PT1-DSC. This information is required by the staff to assess compliance with 10 CFR 72.128(a)(4).

4-8 Separately tabulate and provide all assumptions, including initial and boundary conditions, when describing component temperatures under normal, off-normal, and accident conditions.

Tables 4.1-3, 4, and 5, of the SAR assume different heat load assumptions that vary between 14 kW and 24 kW. As discussed in RAI question 4-6, it is not clear that assuming a 14 kW heat load is conservative. This information is required by the staff to assess compliance with 10 CFR 72.24(d)(1).

4-9 Explain why direct engulfment of the AHSM is unlikely and justify why any fire while the DSC is loaded in the AHSM is bounded by an engulfing fire around the transfer cask.

Section 11.2.4.1 of the SAR includes a discussion of the fire accident. A fire at the inlet of the AHSM may have a greater impact on the 24PT1-DSC than a fire engulfing the transfer cask and, therefore, should be addressed to determine reasonable assurance of adequate safety under normal and accident conditions. This information is required by the staff to assess compliance with 10 CFR 72.128(a).

Chapter 5 Shielding

5-1 Describe how ^{59}Co impurities in the fuel assembly structural material and cladding were modeled in determining bounding source terms. Section 5.2 "Source Specification" does not detail how this material was accounted for in the design basis bounding source term for the stainless steel fuel cladding, fuel hardware, and Inconel materials described in Tables 5.2-2 through 5.2-6.

Independent NRC staff calculations using ^{59}Co impurity levels described in EPRI TR-104329 "Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments" show that this impurity may be important and could lead to significant non - conservatism in both the shielding and thermal source terms.

Alternatively, provide references which characterize the ^{59}Co impurity levels of the materials used to generate the design basis source term and compare those to the impurity levels described in EPRI TR 104329. This information is required under 10 CFR 72.236(d) and (f).

5-2 Provide shielding analyses for all loading and unloading configurations that will exist during operations involving the Advanced NUHOMS® System. Include descriptions and shielding analyses of each transfer cask that will be qualified to transfer or transport the 24PT1-DSC during the four "Loading Stages" described in Section 5.3.1.2.

Alternatively, provide a table describing and comparing the characteristics of each transfer cask qualified to transfer or transport the 24PT1-DSC. Include a discussion which contrasts each cask characteristic and describes how the analysis in Section 5.3.1.3 (shown by Table 5.1-1, Figure 5.1-4, and Figures 5.4-4 through 5.4-7) is conservative or bounding. Provide reference for the OS197 transfer cask (as stated in Section 5.3) as the analyzed condition and compare any applicable characteristics such as material types, shielding thickness, streaming paths, neutron absorbing materials and other applicable characteristics described in NUREG 1536 for any requested alternative transfer casks.

Update the remaining sections of the SAR (including providing specific reference to allowable transfer cask designs) which detail the allowable transfer casks used in the Advanced NUHOMS® System (including Section 12.4.3.3) as appropriate. Figures which describe allowable transfer casks should reference specific casks (either submitted in response to this question or already documented at the NRC) and not general designs such as Figure 5.1-4.

Section 5 of the SAR does not describe all of the transfer casks that will be used in the operation of the Advanced NUHOMS® System. Further, the analysis does not explicitly state that the OS197 transfer cask is the basis for the transfer condition model or provide references to the applicable documents which contain OS197 specifications. The staff does not accept the use of transfer casks which are not analyzed and qualified for the Advanced NUHOMS® System. This information is required to show compliance with 72.236(d).

5-3 Clarify the particle transport model discussed in Section 5.4.1. Further describe the discrete ordinates method and limitations and how the DORT-PC computer code used addresses these limitations. Provide an in depth discussion and documentation (including benchmarking, verification, and validation) of reference 5.2 and how the analysis in Sections 5.3 and 5.4 addresses the limitations of the discrete ordinates method including:

1. The ability to solve mixed problem geometries such as mixed rectangular and cylindrical geometry systems encountered in the cylindrical DSC and the rectangular AHSM.

2. Problems with irregular boundaries and material distributions.

3. Production of spurious oscillations in the spatial distribution of the calculated flux density (also known as the ray effect) as an inherent consequence of the angular discretization. Discuss whether penetration through large non-scattering regions are encountered (where this effect may be particularly important).

4. How the code in reference 5.2 addresses multidimensional situations in which the flux density is anisotropic and in which the medium is many mean-free paths in size.

5. Numerical truncation errors introduced through the discretization of the spatial and angular variables and the required mesh fineness of the angular and spatial meshes which are required to obtain flux densities that are independent of the mesh size, particularly for large system models.

6. How modeling of streaming effects may be effected by the problems discussed in (a) - (e).

7. How the three models described in Sections 5.3 and 5.4 adequately address (a) - (f).

The staff notes that although reference 5.2 is distributed by ORNL/RSIC, no documentation is provided with respect to benchmarking, verification, and validation; and that similar code packages (e.g. CCC-650) may have more applicability as “industry standards.” This information is required by the staff to assess compliance with 72.24 and 72.236(d).

5-4 Submit reference 5.7, which is used to describe the “Normalized Conservative Burn-up Shape on WE 14 x 14 Fuel Assembly” shown in Table 5.4-1. This information is required by the staff to assess compliance with 72.236(d).

5-5 Provide concise figures which describe the DORT shielding model used. The figures should show clear descriptions of materials and thicknesses including boundaries and should contain dimensions and thickness that relate the model Figures (5.4-1 through 5.4-7) to the drawings shown in Section 1. The figures should also clarify what the letters and numbers (including bold or not bold fonts) shown in the model figures (5.4-1 through 5.4-7) physically mean. Further, the figures should show where dose rate measurements are taken and where and how the material “fluxdosium” is located. The figures should also address generic statements made in the SAR when addressing questions 5-2, 5-3, and 5-6 through 5-10 and should provide clear and concise comparison capabilities between the design basis drawings in Section 1 and the model figures 5.4-1 through 5.4-7.

The staff cannot determine the adequacy of the model based on Figures 5.4-1 through 5.4-7 alone. This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

5-6 Justify the statement in Section 5.3.1.1 that overestimating the cross-sectional area of the air vents always leads to conservative streaming estimates. Include discussion of how overestimation of these areas is balanced by the limitations of the discrete ordinates method described in question 5-3.

This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

5-7 Clarify the statement in Section 5.3.1.1 that “since the AHSM possesses limited azimuthal cylidrical symmetries, all approximations result in an overestimate of the dose rates on the AHSM surfaces.” Justify this statement in relation to question 5-3.

This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

5-8 Clarify the statement in Section 5.3.1.3 of the SAR that “The non-symmetric regions such as the 24 Neutron Shield Pannel support angles, the 4 trunnions, relief valves, clevises, eyebolts, etc are modeled such that the dose rate on the surfaces of the cask is overestimated." Describe how these devices are modeled and why resultant dose rates are overestimated.

This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

5-9 Clarify the statement in Section 5.3.2 that "...when the source is smeared into a cylinder, the source is moved closer to the surface of the source region. This results in less self-shielding of the source in the model as compared to the actual geometry, which results in an overestimate of the surface dose rates."

Provide a sensitivity analysis that compares the tradeoff between moving the source out closer to the surface of the DSC and the increased material density of the smeared source, which provides more self-shielding compared to helium. This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

5-10 Clarify the statement made in Section 5.3.2 that "For dose rate evaluations made on surfaces that are perpendicular to the spacer disks, credit is taken for the presence of the carbon steel spacer discs..." The staff assumes that the words parallel and perpendicular should be switched. In addition, clarify the statement "For the AHSM evaluation, no credit for the shielding properties of the spacer discs, fuel grid spacers and hold-down springs."

Provide a table which lists the various configurations of the Advanced NUHOMS® System (wet loading, dry transfer, AHSM evaluation) and the type of dose rate evaluation (axial or radial) and the amount of credit taken for the presence of the carbon steel spacer discs, fuel grid spacers, and hold down springs. Relate this table to the figures submitted in response to question 5-5.

This information is required by the staff to assess compliance with 72.24, 72.236(b) and 72.236(d).

Chapter 6 Criticality

6-1 Describe the calculational model used for the infinite array of casks which is found on page 6.3-2 of the SAR. Also justify in the SAR that this model is bounding for all actual storage, loading/unloading, and transfer configurations.

The model used for this scenario was not described in sufficient detail. It is not clear whether the casks are an infinite array of canisters in transfer casks or if the array consists of canisters in their typical storage configurations out of the transfer casks. This information is required for the staff to assess compliance with the nuclear criticality safety requirements specified in 10 CFR 72.124 and 72.236(c).

6-2 Provide a reference for the transfer cask used in the criticality safety analysis in Chapter 6 of the SAR.

The specific transfer cask that was used in the criticality safety evaluation in Chapter 6 is not identified. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-3 Justify that any other NRC licensed transfer cask is acceptable for use with the 24PT1-DSC without any consideration for criticality safety, as inferred in Section 1.2.1.3.1 of the SAR.

Section 1.2.1.3.1 of the SAR infers that any other NRC licensed transfer cask is acceptable for use with the 24PT1-DSC subject to a site specific evaluation which considers only the following cask characteristics; cavity dimensions, payload, heat load capacity, and shielding. The criticality safety analysis in Chapter 6 does not provide justification for use of other transfer casks, which may be constructed of different types and amounts of materials. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-4 Resolve the discrepancy between Tables 6.1-1 and 6.4-3.

Tables 6.1-1 and 6.4-3 report different k_{eff} values for what appears to be the same case for the most limiting WE 14x14 SC fuel scenario. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-5 Provide the fissile material compositions used in the criticality safety analysis in Tables 6.3-2 and 6.3-3.

The fissile material compositions used in the SCALE calculational models were not provided. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-6 Provide SCALE inputs for each of the following cases;

1. the most limiting normal cases for the intact UO2 fuel, intact MOX fuel, and damaged fuel (either UO2 or MOX), and
2. the most limiting accident cases for the intact UO2 fuel, intact MOX fuel, and damaged fuel (either UO2 or MOX).

The inputs used in the SCALE calculational models were not provided. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-7 Describe in greater detail in Section 6 how the Upper Subcritical Limit (USL) was determined.

The SAR does not discuss any bias and uncertainty associated with the USL determination, nor does it discuss any uncertainty due to modeling approximations. Note that only biases that increase keff should be applied. This is required for the staff to assess compliance with 10 CFR 72.124.

6-8 Describe which factors were considered when choosing the experiments used to validate the computer code used for the criticality safety analysis.

The bias and uncertainty should be established using critical experiments that are applicable to the package design. Several of the experiments, such as BW1810A through E, DSN 399-1 through -4 and W3269A through C contain materials such as gadolinium, hafnium, or cadmium which are not used in the criticality analysis computer models. Therefore, it is not clear why these experiments were used. This is required by the staff to determine the applicability of the critical experiments to the application and to assess compliance with 10 CFR 72.124.

6-9 Provide a table of correlation coefficients for the 6 parameters listed in table 6.5-2 and also for boron areal density.

The correlation coefficients are not given for the SCALE 4.4 validation. This is required for the staff to assess compliance with 10 CFR 72.124.

6-10 Discuss in Chapter 6 the failed fuel can screen mesh size and either justify that uneven draining is not credible or provide an analysis considering uneven flooding.

Uneven draining may be possible for screens with a mesh size of 350 or less because the water surface tension may be capable of supporting water. Uneven draining in the canister may be more reactive than a fully flooded cask. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

6-11 Clarify TS 12.2.1 d. which appears to disallow storage of intact MOX assemblies.

It is not clear whether the applicant intends to limit this application to storage of one damaged MOX assembly per cask only. The TS does not clearly address storage of any intact MOX assemblies. This is required by the staff to assess compliance with 10 CFR 72.24(g), 72.26, and 72.44(c).

6-12 Revise TS table 12.2-2 to include the number of guide tubes and/or instrument tubes.

The SAR does not address varying the number of guide tubes and/or instrument tubes. This is required by the staff to assess compliance with 10 CFR 72.24(g), 72.26, and 72.44(c).

6-13 Revise Section 12.4.0 to include the basket B-10 loading and the flux trap size.

The B-10 loading and flux trap size are design parameters important to criticality safety. This is required by the staff to assess compliance with 10 CFR 72.24(g), 72.26, and 72.44(c).

6-14 Justify that the postulated failure scenarios for damaged fuel is bounding.

The criticality safety analysis in Section 6.3 of the SAR does not provide justification that the scenarios considered for failed fuel bound other possible scenarios such as missing fuel pins. This is required for the staff to assess compliance with 10 CFR 72.124 and 72.236(c).

Chapter 7 Confinement

7-1 Justify the use of an ultrasonic test (UT) for the inner bottom cover plate to canister shell weld in lieu of a radiographic test (RT), as described in SAR section 7.1.3 and Figure 7.1-1. Provide evidence of the UT performance demonstration that was done to qualify the technique for this application, using full scale qualification samples of the same materials and geometry with imbedded cracks (not machined notches or drilled holes). Demonstrate that the sensitivity of the UT procedure meets or exceeds the sensitivity of an RT or the Code acceptance criteria for this application. Update other sections of the SAR as necessary (e.g. ASME code exceptions, and Chapter 9).

The application does not justify the use of the UT technology in an application where UT techniques are difficult and code requirements stipulate RT technology. The NRC notes that the use of ASME Section XI analytical methods and acceptance criteria has not been justified for this application. This information is needed to assure compliance with 10 CFR 72.236(j).

Chapter 10 Radiation Protection

10-1 Provide the dose contribution associated with the maximum contamination levels stated in Section 12.5.2.4 of the SAR.

This information is required for the staff to assess whether the maximum contamination levels bound, combined with the DSC source term, meet the off-site dose requirements of 10 CFR 72.104 and 10 CFR 72.106.

Chapter 11 Accident Analysis

11-1 Provide justification that inadvertently loading spent fuel assemblies, not allowed by Section 12 of the SAR, is not a credible occurrence as stated in Section 11.2.10.3 of the SAR.

NRC staff does not accept that inadvertent loading of spent fuel assemblies is not a credible event. The justification should address both the probability and consequences of loading spent fuel not allowed by Section 12 of the SAR.

This information is required by the staff to assess compliance with 72.94 and 72.236(l).

Chapter 12 Operating Controls and Limits

The following regulatory requirements are applicable in this chapter: 10 CFR 72.11, 72.24(g), 72.26, 72.44(c), 72.104, 72.106, 72.234(a), 72.236, and Subparts C, E, F, G, H, and I. It should be noted that other regulatory requirements may be applicable to this section.

12-1 Specify which transfer and/or transportation casks may be used to perform on-site transfer of the DSC-24PT and any required design criteria.

Technical Specification 12.4.3.3, does not specify transfer or transportation casks that may be used to perform on-site transfer of the DSC-24PT nor does it provide, or reference, a methodology or design criteria to be used to select casks appropriate to perform on-site transfer of the DSC-24PT.

12-2 Provide the analysis to support the 80°F AHSM limit stated in Section 12.5.2.5 of the SAR.

Section 9.1.6 of the SAR states that the heat removal capability is assured through monitoring of the AHSM concrete temperatures. Section 12.5.2.5 suggests corrective actions should be taken to avoid exceeding the concrete and cladding temperature limits if the temperature rises by more than 80°F. Describe how this temperature rise was chosen. Include an explanation of how the fuel cladding temperatures are related to this rise. This information is required by the staff to assess whether the fuel cladding is protected against degradation that could lead to gross ruptures per 10 CFR 71.122(h)(1).

12-3. Clarify the design criteria for Design Feature 12.4.2.2, Storage Pad.

Design Feature 12.4.2.2, refers to Chapter 2 which provides general information and also makes reference to other portions of the SAR. Design Feature 12.4.2.2, should contain sufficient information regarding the evaluation methods required for a licensee to ensure its site is bounded for the Advanced NUHOMS® System. This information is required by the staff to assess compliance with 10 CFR 72.102(c) and (e), 10 CFR 72.212(b)(2) and (3), and 10 CFR 72.122(b).

12-4. Revise Design Feature 12.4.3.1, Advanced Horizontal Storage Module to include a statement regarding the minimum number of AHSMs that must be attached and spacing on the pad that must be attached to each other to meet the design requirements for a design basis earthquake.

This information is required by the staff to assess compliance with 10 CFR 72.90.

12-5. Justify the statement in Section 12.5.2.3.b, of the SAR that the ISFSI will not create any radioactive materials or result in any credible liquid or gaseous effluent release.

The SAR did not provide sufficient information for not implementing effluent monitoring program for the Advanced NUHOMS® design. The justification should include an analyses and a discussion of all assumptions and methodology used, to support the statements made in Sections 12.5.2.3 and 12.5.2.4. The analyses should include the potential off-site dose release from the maximum number of Advanced NUHOMS® systems on site with a residual contamination limit of approximately 22,000 dpm/100 cm^2 β-α and 2,200 dpm/100 cm^2 α. The staff cannot confirm that the Advanced NUHOMS® system with the administrative controls program describing dose rates, surface contamination limits, and radioactive effluents will comply with the off-site dose limits of 10 CFR 72.104 for normal and anticipated occurrences.

In the past, the staff has accepted that surface contamination limits set at the values recommended by Regulatory Guide 1.86 will result in no effluent release from an ISFSI. Therefore, the staff has concluded that an effluent monitoring and reporting system in accordance with 72.44(d) (for site specific licensees) or 50.36a (for general licensees) is not required. This information is required for the staff to assess compliance with 10 CFR 50.36a in lieu of maintaining the contamination limits of Regulatory Guide 1.86, “Termination of Operating Licenses for Nuclear Reactors.”