Request for Additional Information
Nuclear Management Company, LLC
Docket No. 72-10
Related to Special Nuclear Material License No. 2506 Amendment Request
to Modify the TN-40 Cask Design at the Prairie Island
Independent Spent Fuel Storage Installation (ISFSI)

By application dated March 28, 2008, supplemented August 29, 2008, Nuclear Management Company, LLC (NMC) requested approval of an amendment to Special Nuclear Material License No. 2506 and the license Technical Specifications (TS) for the Prairie Island ISFSI in accordance with 10 CFR 72 (ML081400652). This amendment proposes to modify the TN-40 cask for storage of high enrichment and high burnup fuel and to reformat the TS.

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 amendment. The requested information is listed by chapter number and title in the applicant's safety analysis report. Where applicable, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," and NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," were used by the staff in the review of the amendment application.

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

ATT 1. REVISED TS

1.0 USE AND APPLICATION

1.1 DEFINITIONS

RAI ATT 1.1

Perform a transient impact structural integrity evaluation, similar to that of Section A4.2.3.8 of the SAR, of the fuel rod cladding for the 18-inch cask handling end-drop accidents, considering the "undamaged fuel assembly" characterized with: (1) uniform rod bowing and (2) missing, displaced, or damaged structural components that can still be handled with normal means.

The applicant defines undamaged fuel assemblies as those with uniform rod bowing and that can be handled by normal means, even if there exist missing, displaced, or damaged structural components. However, since fuel rod buckling performance has not been analyzed for the undamaged configurations described above, a structural evaluation must be included in the SAR to substantiate the subject definition.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

MATERIALS

Materials cut across several sections of the SAR. Therefore, the materials RAIs are presented here as a single collection. *The italicized RAIs were also asked for the TN40 transport license application* (ML0813006640 and/or ML0900300032).

RAI-M1

Specify the type of Never-seez to be used for lubrication of the trunnions. Provide justification for the compatibility with borated water and stainless steel. Specify the applicable temperature range of use.

Never-seez comes in a number of different varieties with different preferred applications and recommended environments for use.

This information is needed to determine compliance with 10 CFR 72.122(b)(2), and 72.122(1).

RAI-M2

Analyze the potential of a pyrophoric event during the loading, transporting, or unloading of the uranium replacement rods.

SAR Section A3.1.1 indicates "uranium" as a suitable replacement for fuel rods in reconstituted assemblies. The use of uranium requires the analysis of potential interactions and pyrophoric events.

This information is needed to determine compliance with 10 CFR 72.120(d), 72.166, and 72.122(h)(1).

RAI-M3

Provide copies of the references, or the NRC Agency Document and Management System (ADAMS) accession numbers if relevant, that substantiate the guide tube and instrument wall thickness (Table A7.2-1). Correct guide and instrument tube diameters (Table A3.3-19) to reflect the correct wall thicknesses, if necessary.

Assembly and rod specifications in the tables were reviewed by the staff. While in most cases there was agreement, in some cases the staff identified discrepancies with the staff's reference values (multiple sources). For example, the reviewers' sources indicate a substantially thicker tube wall (0.034 in).

This information is needed to determine compliance with 10 CFR 72.124(a) and 72.11.

RAI-M4

Correct or give references for the existing maximum MTU/assembly for the Westinghouse Electric Company (WEC) standard 14 x 14 fuel assembly in Table A7.2-1 and other tables.

This information is needed to determine compliance with 10 CFR 72.124(a) and 72.11.

RAI-M5

State the assumptions with respect to time out of reactor, uniformity of layer thickness, etc. used to determine the quantity of CRUD available to spall.

While the spallation fraction for the CRUD is stated in SAR Section A7A.8.5.1, no values and assumptions are given for CRUD quantities.

This information is needed to determine compliance with 10 CFR 72.126(d).

RAI-M6

Specify the radiation dose over 20 years at the location of the drain port valve and evaluate its affect on the Viton o-ring.

At significant dose, deterioration of the Viton o-ring may release fluorine into the cask resulting in loss of containment of the Zircaloy cladding. Such degradation would also affect the effectiveness of the seal.

This information is needed to determine compliance with 10 CFR 72.126(d).

RAI-M7

Specify in the Technical Specifications the % credit for the boron-10 for both the Boral and the B-Al alloy.

This information is needed to determine compliance with 10 CFR 72.124(a).

RAI-M8

Include an acceptance plan for the neutron poison plates in the SAR and include it by reference into the proposed Technical Specifications. Correlate the acceptance testing of the neutron absorber with expected performance. Indicate how the acceptance tests indicate an adequate percentage of H and B in the absorber material. Describe the significance of the density measurement, and the sensitivity of measurements to the percentage of critical components (H & B).

This information is needed to determine compliance with 10 CFR 72.124(a).

RAI-M9

Provide a reflood analysis.

This information is needed to determine compliance with 10 CFR 72.122(h)(1).

RAI-M10

Provide an acceptance plan for the neutron shield material. Provide data or analyses to show that the neutron shield material (both resin and polypropylene) will retain adequate properties for the application during the storage period. Include the testing procedure, and data that were collected to determine the maximum temperature that the resin can withstand without degradation. This plan should be included by reference to the SAR in the proposed CoC.

The neutron shield material is a borated polyester resin compound that surrounds the gamma shield shell. It is subject to thermal and radiation fields during service, which have the potential for degrading properties of the material including its thermal conductivity.

This information is needed to determine compliance with 10 CFR 72.126(6).

RAI-M11

Provide temperature-dependent fracture property data for the filler metal and the heat affected zone (HAZ) in the temperature range of Hypothetical Accident Condition (HAC) to support the claim that the weld cracks in the base metal of carbon steel (SA-266, Class 2) are stable (SAR Sec A4A.9).

This response should provide justification that any testing, using a limited combination of potential base metals, filler materials, and weld techniques, bounds the worst case fracture toughness expected from all potential combinations of these three parameters. Explain how the TransNuclear (TN) fabricators choose the combinations of weld processes, electrodes and base material to demonstrate the toughness of the weld and HAZ. Defend why any data provided are representative of all other possible combinations which can be used, or are these data the best case scenario?

Various weld techniques, parameters and/or procedural steps can be used to maintain or improve base metal, HAZ, and weld metal mechanical properties. For example, control heat input, bead placement, weld bead type, etc., are such parameters. For any test that results in abnormally high fracture toughness, the response should state the weld parameters utilized in the weld procedure.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

RAI-M12

Justify, as stated in the SAR, that the filler metal is as tough as the base metal (Sec A4A.9.5). Specify the code requirements that the weld filler materials satisfy.

The application provides fracture toughness data of the base metal (SA-266, Class 2) and presumably uses it to show that potential weld cracks in the 10 critical locations remain stable during storage since no fracture toughness for the welds is provided. It is known that mechanical properties of filler material as well as HAZ, in general, can be dramatically different from that of the base metal. It is clear that data of the weld material should be used as the cracks are located within the welds, not in the base metal.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

RAI-M13

Specify the weld inspection requirements for the fuel basket, and include these requirements in the proposed Technical Specifications.

The staff position is the basket must be inspected per the requirements of American Society of Mechanical Engineers (ASME) Code, Subsection NF, due to the prevalent use of fillet welds, not full penetration welds, as would generally be the case for Subsection NB construction.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

RAI-M14

Specify the acceptance standards or codes for the structural and containment welds. Include these standards or codes in the proposed Technical Specifications.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

RAI-M15

Specify the codes used for welders and weld procedures qualifications. These codes should be placed in the Technical Specifications.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

RAI-M16

a) Provide a discussion or calculation that shows that the various aluminum alloy canister components will meet their life-time design requirements when operating at temperatures where the material is subjected to creep-induced deformation.

b) Provide or cite references for the long-term creep properties of any aluminum alloy canister component(s) which exceed the stress or temperature limits of the ASME Code, Section II, Part D. Show that these properties are adequate for meeting the component's design-life performance requirements during the specified operating condition(s).

In Section A4B.1.5.6 of the SAR the applicant states: "The long term storage load compressive stresses in the limiting aluminum components were compared to allowable stress values that have been reduced to limit the effects due to materials creep."

This information is required for compliance with 10 CFR 72.122(b)(2).

RAI-M17

Justify the use of the hemispherical emissivity of 0.46 for 304 stainless steel, in SAR Section A3.3.2.2.3.6.2.3.

Staff's reference gives a value 0.35 to 0.3 in the temperature range of 200-400ºC.

This information is needed to determine compliance with 10 CFR 72.128(4).

RAI-M18

Provide references for the thermal characteristics of the neutron shield resins given on page 6 of SAR, Table A3.3-8.

This information is needed to determine compliance with 10 CFR 72.126(6).

RAI-M19

Provide thermal conductivities for fuel with a burnup of 60 GWd/MTU.

Values of thermal conductivity in SAR section A.3.3.2.2.3.6.2.2.1 are for unirradiated UO2.

This information is needed to determine compliance with 10 CFR 72.128(4) and 72.122(c).

A1. INTRODUCTION AND GENERAL DESCRIPTION OF STORAGE SYSTEM

RAI A1.1

Refer to the Proprietary Enclosure.

RAI A1.2

Refer to the Proprietary Enclosure.

A3. PRINCIPAL TN-40HT CASK DESIGN CRITERIA

RAI A3.1

NRC staff was unable to locate some general information on the proposed contents. Table A3.1-1 lists some general parameters for each of the approved fuel assembly classes however physical specifications of the assembly are missing, notably maximum assembly weight. Table A3.2-1 lists the presumed weight for all 40 assemblies, but isn't clear if this should be considered an upper bound for any particular class of fuel assembly.

Please present this information in chapter A3 or if located elsewhere in the SAR, indicate within chapter A3 where it may be found.

This information is necessary to verify compliance with 10 CFR 72.11.

RAI A3.2

Table A3.1-1 lists maximum MTU/assembly for the Westinghouse standard assembly as 410 MTU. It is believed the applicant intends this number to be 0.410 MTU and all confirmatory analyses have used this assumption.

Please correct the error in table A3.1-1.

This information is necessary to verify compliance with 10 CFR 72.124.

RAI A3.3

Provide the time-to-boil calculation for the liquid in the cask during wet fuel transfer operations.

In the Standard Review Plan for Dry Cask Storage Facilities (NUREG 1567) Section 6.5.1.2 states that the applicant should provide a time-to-boil calculation for the loaded cask during transfer operations. This calculation is important to determine if any conditions could exist that might impact the performance of the fuel cladding. The staff did not find the calculation for the time-to-boil in the application. If the time-to-boil calculation was mentioned within the application, provide the appropriate location of this information.

This information is needed to confirm compliance with 10 CFR 72.24(c)(3).

RAI A3.4

Provide a description of the reconstituted assemblies authorized to be stored in the TN-40HT cask. Include in the description, the enrichment, dimensions, and material of the stainless steel, inert, and uranium replacement rods. In addition, describe how reconstituted assemblies, having uranium rods, were addressed in the shielding evaluation.

Section A3.1.1 states, in part, that reconstituted assemblies (uranium, inert, or stainless steel rods replacing fuel rods) may also be stored in the cask. However, no information (i.e., dimensions, enrichment, etc.) was identified with regards to the uranium rods which may be used. Furthermore, use of the uranium rods was not analyzed in the shielding evaluation.

This information is needed to determine compliance with 10 CFR 72.24.

A4. STORAGE SYSTEM

RAI A4.1

<u>Section A4.2.3.3.3, Basket.</u>

Revise the underscored description in the statement, “[T]he required minimum tested capacity of the weld connection shall be based on a margin of safety (test to design) of 1.43 (see Appendix F, <u>Section F-132 (c)</u> of Reference 1), corrected for temperature difference between testing and basket operating conditions and the maximum weld load at any weld location in the basket."

Section F-132 (c) and related margin of safety requirement cannot be found in Appendix F of the ASME code.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

RAI A4.2

<u>Table A4.2-2, Containment Vessel Stress Limits.</u>

Revise the table to include also the stress allowable criteria for lid closure bolts as part of confinement boundary of the cask system.

Tables A4.2-2, -3, and -4 presents stress limits for the containment vessel, non-containment structures, and basket, respectively. To meet the 10 CFR 72.122(a) quality standard requirements, the lid closure bolt stress limits, which must be ASME Subsection NB compatible, should also be described in the SAR and tabulated accordingly to facilitate staff safety evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

RAI A4.3

<u>Table A4.2-10, Linearized Stress Evaluation for Normal Condition Load Combinations.</u>

With respect to load combination Case N5, use nodal stress intensities at Nodes 938 and 1218 and any intervening nodes, as appropriate, in an explicit calculation to verify that the stress linearization post-processing is properly implemented for calculating the primary membrane, P_m, and primary membrane-plus-bending, $P_l + P_b$, stress intensities.

The listed P_m and $P_l + P_b$ stress intensities of 1.98 ksi and 5.67 ksi, respectively, are much smaller than the referenced peak nodal stress intensity of 14.52 ksi. This raises a general concern on whether the ANSYS stress linearization post-processing is properly implemented for the cask body stress evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

RAI A4.4

Refer to the Proprietary Enclosure.

RAI A4.5

Refer to the Proprietary Enclosure.

RAI A4.6

Refer to the Proprietary Enclosure.

A4B. STRUCTURAL ANALYSIS OF THE TN-40HT BASKET

RAI A4B.1

Table A4B.1-1, Summary of Individual Loads for Storage Conditions – Basket.

Revise the table, as appropriate, to include also the load case associated with the 18-inch cask handling end-drop accident.

For clarity and completeness, the cask end-drop accident condition, as a licensing basis, should be included in the table to facilitate staff safety evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

RAI A4B.2

Section A4B.1.5.2.1, Finite Element Model Description.

Revise Figures A4B.1-2, A4B.1-3 and add additional sketches to provide sufficiently legible details to depict element types, discretization schemes, and interface as well as boundary conditions, as appropriate, for the structural analysis of the basket subject to lateral loads.

The SAR text and figures are short of necessary details for the adequacy of the basket finite element model.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

RAI A4B.3

Figure A4B.1-4.

Considering the connectivity between the 1.75-inch wide spacer bar and the fuel compartment walls, provide sketches to illustrate the interface conditions for which the load paths at the nodes other than the fusion weld locations must be properly accounted in the basket structural analysis.

Section A4B.1.5.2.1 of the SAR states: “[t]he strengths of aluminum plates and poison plates in the basket are neglected by excluding them from the finite element model.” Properly annotated modeling details are needed to facilitate staff review of the model assumptions made on interface conditions.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

A5. STORAGE SYSTEM OPERATIONS

RAI A5.1

Specify the sensitivity of the cask helium leakage rate test. Also, clarify that monitoring system boundaries are tested to a leakage rate equal to the confinement boundary.

The Technical Specifications should include a minimum test sensitivity of 5×10^{-6} atm-cm^3/sec for the cask helium leakage rate, consistent with ANSI N14.5-1997. ISG-5, “Confinement Evaluation,” states that monitoring system boundaries should be tested to a leakage rate equal to the confinement boundary. The staff could not find where this was described in the application. This information should be provided in the storage system operations or the Technical Specifications.

This information is required to determine compliance with 10 CFR 72.122(h)(4) and 128(a)(1).

A7. RADIATION PROTECTION

RAI A7.1
Provide information regarding burnup, enrichment, cooling time combinations for other candidate fuel assemblies authorized for storage in the TN-40HT cask.

Section A7.2.1 states that the 14x14 Westinghouse standard is the design basis fuel for shielding purposes because it has the highest initial metal loading and therefore results in the highest radioactive source terms for a given irradiation history. This includes a burnup, bundle average enrichment, and cooling time of 60 GWd/MTU, 3.4 wt% U-235, and 18-year cooling time, respectively. It is also noted that CRUD is maximized at the minimum cooling time of 12 years.

Section A3.1.1 states, in part, that fuel with various combinations of burnup, enrichment, and cooling time can be stored in the TN-40HT cask as long as the combination results in decay heat, surface dose rates, and radioactive sources for confinement that are bounded by the design basis fuel.

The SAS2H evaluation yielding the bounding source terms used for shielding and confinement were taken for what was identified as the design basis fuel. Additional information is required to justify the licensee’s selection of the initial enrichment wt.% U-235, burnup, and cooling time combination as having the bounding parameters for the shielding and confinement analyses.

This information is needed to determine compliance with 10 CFR 72.24 and 10 CFR 72.104(a).

RAI A7.2

Provide justification for the use of the particular burnup and cooling time values used in the calculation of the inserts. In addition, include information as to whether any downtimes existed between cycles during the overall burnup.

Section A7.2.1 describes the methodology for inclusion of the Fuel Insert Thimble Plug Device (TPD) and the Burnable Poison Rod Assembly (BPRA). The results of the SAS2H/ORIGEN calculations for the TPD and BPRA were included in the results for the design basis fuel gamma source. Staff has some degree of confidence that the TPD burnup was based on the total number of cycles. However, the basis for other relative assumptions (e.g., burnup of the BPRA for 30 GWd/MTU) used in the analysis concerning the TPDs and BPRAs are not discussed.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

RAI A7.3

Provide justification supporting your use of a lower boron concentration. Has a comparative analysis been performed on the change in boron concentration between 900 ppm and 600 ppm?

Section A7.2.1 states, in part, that typical cycle average boron concentration is on the order of 900 ppm. It also states that for modeling purposes in the current analysis, 600 ppm was chosen to be the average boron concentration for the first irradiation cycle, with the second having 95% of this value. The SAR makes reference in the paragraph that there is essentially no effect on dose rates and cooling times based on certain studies which were not discussed in adequate detail or referenced in the SAR. It is also stated in the discussion that “studies were performed showing that the use of a lower boron concentration leads to a tiny underproduction of decay heat, neutron and gamma source strength in the energy groups that contribute the most to casks dose rates.” The “studies” discussed in the paragraph provided no direction to sources supporting the use of a lower boron concentration.

This information is needed to determine compliance with 10 CFR 72.24.

RAI A7.4

Provide your technical justification for the use of 566°F as the moderator temperature.

The SAR states that moderator temperatures can vary between 500 – 600°F. The SAR states that a higher average moderator temperature results in increased epithermal absorption in U-238, which results in an increase in the actinide inventory in the fuel for a given total fuel burnup. The SAR states that a moderator density corresponding to a temperature of 566°F was used in the SAS2H calculation.

This information is needed to determine compliance with 10 CFR 72.24.

RAI A7.5

Identify the localized regions of elevated dose rates due to streaming. Please provide dose rates for vent and drain ports and what methods will be used to ensure doses are maintained ALARA.

In Section A7.4 of the SAR, it states that localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the vent and drain ports.

Section A1.3.2 states, in part, that penetrations exist for leak detection and venting. There are also vent and drain covers in the steel lid. Staff finding is that no dose rate estimates were identified for those regions where radiation streaming could occur, and no discussion was included detailing what the estimated radiological impacts were as a result.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

RAI A7.6

Provide confirmation as to whether fuel assemblies authorized for storage in the TN-40HT cask include natural uranium blankets.

Section A7.2.1 of the SAR provides information used in determining the neutron and gamma source terms. It states, in part, that the fuel assemblies acceptable for storage in the TN-40HT cask are listed in Table A3.1-1. Table A3.1-1 provides some detail about the authorized assemblies but gives no indication that natural uranium blankets were used with these assemblies.

From information found in a separate SAR, natural uranium blankets were used for fuel authorized for the TN-40 transportation package.

Confirmation is needed for fuel assemblies authorized to be stored in the TN-40HT cask.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

A7A. TN-40HT CASK DOSE ANALYSIS

RAI A7A.1

Identify the dimensions, conservatisms, and assumptions used in the TN-40HT cask model and the justification for all assumptions used in the shielding evaluation. Include all relevant dimensions, conservatisms, and assumptions used to generate the SAS2H and MCNP models, along with the justification for any differences between the TN-40HT cask design and the models used in the shielding evaluation.

Section A7A.4.1 states, in part, that the MCNP model used for normal and off-normal conditions is essentially based on the design details from the TN-40HT cask drawings, shown in Section A1.5, except for some conservative representations. The SAR must describe the computational models, data, and assumptions used in evaluating shielding effectiveness. More detail needs to be included (i.e., the distinct dimensions used in the models) in order for staff to confirm the adequacy of the shielding evaluation.

This information is needed to determine compliance with 10 CFR 72.24.

RAI A7A.2

Identify the differences between the gamma and neutron models used for the normal and off-normal conditions of the shielding evaluations and the justification for any assumptions and conservatisms used in the models.

Section A7A.4.1 states that two models were developed for determining the normal and off-normal dose rates. The gamma model containing a detailed segmentation of the thicker cask steel body is utilized to calculate the primary gamma dose rates. The neutron model is utilized to calculate the neutron and secondary gamma dose rates.

Staff finding is that although two different models were developed, no justification was included for the differences in the two models (e.g., why a thicker cask steel was used). In addition, the SAR states that the thickness of the gamma shield was reduced but the neutron shield thickness was increased as a consideration of the overall weight. More detail regarding the differences in dimensions need to be provided as part of this analysis.

This information is needed to determine compliance with 10 CFR 72.24.

RAI A7A.3

Provide relevant calculations and assumptions regarding the exponential function and decay constant used in specifying the total neutron and gamma source term strengths as described in Section A7A.7.1.

The discussion in Section A7A.7.1 addresses how the source term strengths can be approximated with an exponential function as a function of decay time. However, it is not clearly defined how the exponential function is used to approximate the source strength. No information was identified supporting the relationship between the source term strength and decay time. In addition, it is not clearly defined whether or not the relationship assumes that all nuclides decay at the same rate.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

RAI A7A.4

Table A7.2-6

Clarify if the light elements Co-60 and Ni-63 should be included in the radioactive inventory for the 14x14 design basis fuel assembly.

Section A7A.8.5.1 of the SAR states that Table A7.2-6 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in the design basis fuel, plus Iodine-129. It appears that the light elements Co-60 and Ni-63 contribute more than 0.1% of the activity contained in a design basis fuel (based on SAS2H results, 0.39% and 0.21% respectively), but they were not included in Table A7.2-6.

This information is required to determine compliance with 10 CFR 72.24(l)(1).

RAI A7A.5

Justify the use of a 45 day exposure period for off-normal conditions in Section A7A.8.5.2.

NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," states that for off-normal conditions, the bounding exposure duration should be the same as those for normal conditions which assumes that an individual is present at the controlled area boundary for one full year (8760 hours). Alternative exposure duration may be considered by the staff if the applicant provides justification.

This information is required to determine compliance with 10 CFR 72.104(a).

A8. ACCIDENT ANALYSIS

RAI A8.1

Section A8.2.8.2.1, Dynamic Impact Loads.

Considering the approach similar to that for the NUHOMS-HD storage system (Docket 72-1030), perform a transient dynamic impact dynamic analysis of the cask for the 18-inch handling end-drop accident to define applicable loading conditions for cask component evaluations.

A comprehensive review of the EPRI NP-7551 target hardness method and its benchmarking for TN-40HT application may involve long lead-time without certitude for closure. The staff will review other justifiable methods, including the NUHOMS-HD approach, for determining loading conditions for cask components.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

EDITORIAL COMMENTS

ED-1:

Clarify the apparent misspelling of the word properties found within the following sections of the SAR.

Within Sections A3.3.2.2.3.2, A3.3.2.2.3.3, and A3.3.3.2.2.3.4 of the SAR, properties was spelled “propoerties”.

ED-2: Tables A7A.8-3 through A7A.8-6

Clarify if in Table A7A.8-3 through A7A.8-6 of the SAR, Np237 should be Np239. Also clarify if in Table A7A.8-6 of the SAR, Cm243 should be Cm244.

ED-3: Enclosure 3 Attachment 2 “PI ISFSI Technical Specifications Bases”

In “PI ISFSI Technical Specifications Bases” ANSI 14.5 references are from 1977 and should be updated to be from 1997.