**Non-Proprietary Request for Additional Information**
**Docket No. 71-9382**
**Model No. Model Nos. TN Eagle-STC SC and TN Eagle-STC LC**
**Revision No. 0**

By letter dated December 30, 2020 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML20365A018), as supplemented on April 29, 2021 (ADAMS Accession No. ML21119A307), TN Americas LLC submitted an application for a new certificate of compliance for the Model Nos. TN Eagle-STC SC (standard canister) and TN Eagle-STC LC (large canister) spent fuel packages. Unless specified, TN Eagle-STC refers to both model numbers.

This request for additional information identifies 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 in the applicant's safety analysis report (SAR). The NRC staff used NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material: Final Report," in its review of the application. Each question describes information needed by the staff for it to complete its review of the application and to determine whether the applicant has demonstrated compliance with regulatory requirements.

## 1.0 General Information

1-1 Clarify the material composition of the cask internal sleeves. Provide an operational procedure step(s) for installation and removal of the internal sleeve.

Section 1.2, "General Packaging Arrangement:" states that, "*a sleeve is used to limit the radial gap of DSCs [dry shielded canisters] in TN Eagle SC Casks*. Section 3.1.1, "Other Non-EOS DSC Types" states in part that, "*Other non-EOS DSCs including 24PT4, 32PT, 32PTH1, FO/FC/FF, and 24PT1 DSCs are allowed to be transported in the TN Eagle SC with internal sleeves*."

Section 3.6.5.1.2, "Thermal Model," state in part that, "*Internal sleeve and rail extensions are assumed* **[[**proprietary information removed**]]** *with material properties listed in Table 8-5 of [8]*". If a different material of the internal sleeve is considered, the material properties will be verified to ensure the values used in the thermal evaluations remain bounding. If they do not remain bounding, the thermal model in this appendix will be re-evaluated to ensure all criteria in Section 3.2.3 are satisfied." However, Section 1.6.3.1, "24PT4 DSC Description," states that, "*Under normal transport conditions, the canister rests on four canister rails attached to the inside surface of the aluminum inner sleeve of the transport cask*."

NUREG 2216, Section 8.1.1.1, "Preparation for Loading," states, "*verify that the application describes the procedures for package loading preparations sequentially in the order of performance, and ensure that the procedure descriptions, at a minimum…* The contents are authorized in the certificate of compliance, including the use of a secondary container or containment, shoring, or dunnage, as applicable." The NRC staff were unable to identify any procedural step for installation or verification of the sleeve.

This information is needed to determine compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 71.33 and 10 CFR 71.35(c).

## 2.0 Structural Evaluation

2-1 In Appendix 2.11.1, “TN EAGLE Cask NCT [normal conditions of transport] Evaluation,” the applicant stated that the weight of the impact limiters was not included for the lifting condition. Since the impact limiters are part of the packaging, clarify why this weight was not included.

If the TN Eagle-STC package is lifted without impact limiters either at an NRC-licensed facility or during transport, provide either a revised cask drop analysis without the impact limiters installed, or a commitment to utilize a single failure proof handling system and describe associated single failure proof design features. These additional analyses, or design features, ensure that if the package is lifted either during preparation, shipment, or receipt, that the package is maintained within its design configuration. Note that lifts at an NRC facility licensed under 10 CFR Part 50 or 10 CFR Part 72 would also be evaluated under the provisions of 10 CFR 50.59 or 10 CFR 72.48, accordingly.

Section 8.1.3 indicates that impact limiters are not installed during cask lifting operations. Specifically, “Step 3. Using an appropriately sized lift beam and rigging, engage the outer ends of the cask body with slings in a basket configuration. Step 4. Lift the cask out of the transfer skid and place it in the transport frame on the conveyance. Step 5. Install the transport frame tie-down straps. Step 6. Install the impact limiters on the cask...”

Additionally, a cask lift height restriction should be established to ensure the package without its impact limiters is not lifted to a height greater than that for which a drop evaluation has been performed, or alternatively has been deemed incredible by utilizing a single failure proof handling system. This operational restriction ensures that the package is handled within the limitations of its design.

The NRC staff was unable to determine whether a failure of a lifting system would impair the ability of the package to meet the requirements of 10 CFR Part 71 in the event of a drop event when the impact limiters are not installed.

This information is necessary to determine compliance with 10 CFR 71.47, 10 CFR 71.51(a) and 10 CFR 71.55(b).

2-2 In Appendix 2.11.1, when discussing accelerations during transport (Appendix 2.11.1.4.9), the applicant discusses various reactions that were hand calculated and input into the systems since some of these components were not explicitly modeled. However, the applicant did not provide these hand calculations, therefore the staff is unable to ascertain how the reactions were calculated and consequently modeled. Provide calculations that demonstrate the reaction forces for the components and identify their input locations in the models as discussed in Appendix Section 2.11.1.4.9.

This information is necessary to determine compliance with the requirements in 10CFR 71.45(a).

2-3 See Enclosure 2.

2-4 See Enclosure 2.

2-5 In appendix 2.11.5.1.5, "Shielding Rings" the applicant provides a qualitative discussion on the effects of puncture on the shielding rings and states that local damage may occur after this accident condition and refers to an analysis of a package with a similar mass as the TN Eagle-STC. Provide the following clarification:

a. Provide the reference for the package with a similar mass and justify why the comparison is appropriate.
b. Evaluate how shielding requirements continue to be met after this accident condition given that local damage to the shielding rings may occur.

This information is necessary to determine compliance with the requirements in 10 CFR 71.73.

**3.0 Thermal Evaluation**

No questions.

**4.0 Containment Evolution**

No questions.

**5.0 Shielding Evaluation**

5-1 Provide additional information related to Note 1 of Table 1.6.7-1 of the application that makes allowance for cooling times shorter than those specified for the 24PT1 DSC.

Note 1 to Table 1.6.7-1 for the loading of Westinghouse (WE) 14×14 fuel class for the 24PT1 DSC states:

*"The 39 year cooling requirement may be shortened if it can be demonstrated that a payload meeting all other requirements specified above will result in dose rates that are bounded by 3.76 weight % U-235, with 45,000 NWd/MTU burnup with 39 years cooling. For this analysis, a method of classification of fuel assemblies similar to that specified for the [Babcock and Wilcox] B&W 15x15 fuel may be applied. Alternatively, the acceptability for transport of WE 14x14 stainless steel clad (SC) fuel assemblies with less than 39 years of cooling may be demonstrated by measured doses on the Package surface prior to shipment. This method must apply a 25% margin to account for potential errors in the value measured. The measured dose for normal operation conditions with the added margin shall be compared to the 10 CFR 71 limits and is acceptable if within these limits (including margin)."*

a) Provide clarifying information on what is meant by the following statement and add this information to the operating procedures as necessary: *"a method of classification of fuel assemblies similar to that specified for the B&W 15x15 fuel may be applied."* Based on the 24PT1 DSC contents being for mixed oxide and stainless steel clad fuel, the method would need to address these differences in source term from the B&W 15×15 as the safety analysis report (SAR) methods using the B&W 15×15 are based on zirconium based clad for $UO_2$ light-water reactor fuel.

b) Remove the statement: *"Alternatively, the acceptability for transport of WE 14x14 stainless steel clad (SC) fuel assemblies with less than 39 years of cooling may be demonstrated by measured doses on the Package surface prior to shipment."* 10 CFR 71.35(a) states that an application for approval must demonstrate that the design of the package satisfies the requirements of Subpart E. Staff does not accept pre-shipment measurement as an evaluation demonstrating that the package meets the dose rate requirements in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2) (contained in Subpart E). Staff cannot determine that a package meets these dose rate requirements at the time of application based on pre-shipment measurements that happen in the future. The regulations in 10 CFR 71.31(b) states that an application for a package design must include sufficient information to demonstrate that the proposed design satisfies the package standards in effect at the time the application is filed. (See NRC RIS 13-04, "Content Specification and Shielding Evaluation for Type B Transportation Packages," https://www.nrc.gov/docs/ML1303/ML13036A135.pdf).

This information is needed to determine compliance with the requirements in 10 CFR 71.31(b), 10 CFR 71.33(b)(1), 10 CFR 71.35(a), 10 CFR 71.47(b) and 10 CFR 71.51(a)(2).

5-2 Clarify where the fuel qualification table (FQT) is located for the FO/FC/FF DSCs that differentiate between Type I and Type II assemblies or provide this FQT. Discuss how control components have been accounted for in the decay heat limits.

Section 1.6.6.6 of the application states: *"Fuel assemblies are separated into two types, Type I and Type II. Type I assemblies are those assemblies which meet the 0.764 kW assembly heat load limit, but not the cask average (0.563 kW per assembly) heat load limit. Type II assemblies are those assemblies which meet the 0.563 kW per assembly cask average decay heat limit. As shown in Figure 1.6.6-1, Type I assemblies shall only be loaded into the inner four fuel cells of the FO DSC. Type II assemblies may be loaded into any fuel cell in the FO DSC. Cooling times necessary to reach these heat loads are tabulated versus burnup in Fuel Qualification Tables."*

The FQT associated with the FC/FO/FF DSC is Table 8.7.3-1 of the application. This table does not have a distinction between Type I and Type II assemblies. The staff estimated the decay heat for higher allowed burnup (38,000 MWd/MTU and above) allowed by Table 8.7.3-1 of the application using the guidance in Regulatory Guide (RG) 3.54, Rev. 2, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," and it exceeds 0.563 kW decay heat without consideration for control components.

This information is needed to determine compliance with the requirements in 10 CFR 71.35(b)(7).

5-3 Provide the following additional information related to control components (CCs):

a) Provide the loading requirements for control components in heat load zone configuration 2 (HLZC2) for the NUHOMS® EOS-37PTH (EOS-37PTH) DSC. Table 1.6.1-17 of the application only has 3 zones and the note only references Figure 1.6.1-1 of the application and does not mention HLZC2 or Figure 1.6.1-2, however Figure 1.6.1-2 indicates that control components are allowed in HLZC2.

b) Revise Table 1.6.1-17 of the application to include a description of Note 2. Table 1.6.1-17 of the application includes a reference to a Note 2 on "Zone 1" but there is no Note 2 present on the table.

c) Update the operating procedures to include instructions for determining allowable control components for the EOS-37PTH DSC where nuclides other than Cobalt 60 (Co-60) have a significant contribution to dose rate. Authorized control components include control rods, neutron source assemblies and neutron sources which contain radionuclides other than Co-60, however limits associated with control components are in terms of limiting Co-60. The applicant states in Section 5.2 of the application: "*Control components with hafnium or silver-indium-cadmium (Ag-In-Cd or AIC) as absorber materials may have different gamma spectrum after irradiation due to activation of hafnium and silver*." There is a discussion in Section 5.4.1.5 of the application about performing Co-60 equivalence using response functions. Include the appropriate steps defining this procedure within the operating procedures for the TN-Eagle-STC.

d) Justify that including control components with significant source at the bottom of the TN-EAGLE meets regulatory dose rate limits. The CC source assumed in Tables 5-32 and 5.6.1-9 of the application do not contain a concentrated source term at the bottom of the package. Some control components such as control rods and axial power shaping rods (APSRs) can have a significant source that increases dose at the bottom of the cask.

e) The radiation source term for CCs in the 32PT and 32PTH1 DSCs are limited in Tables 1.6.4-4 and 1.6.5-3 of the application. These tables are in terms of gamma/sec/assembly and decay heat. Provide a maximum energy for the gamma source in these Tables and discuss how these source specifications are bounded by the CC source assumed in the shielding evaluation in Table 5.6.1-9 of the application.

f) Provide additional information limiting the radiation source term for the CC for the FC/FO/FF DSC in Section 1.6.6 of the application. Section 1.6.6.7 of the application has maximum decay heat for the allowable fuel assemblies and a minimum cooling time, but this is not enough information to determine that the radiation contribution from this source term will meet regulatory dose rate limits with the analysis provided.

This information is needed to determine compliance with the requirements in 10 CFR 71.33(b)(3), 10 CFR 71.47(b) and 10 CFR 71.51(a)(2).

5-4 Clarify if the intent for FQTs in Appendix 8.7 of the application is to qualify fuel to meet decay heat limits. If so, justify that fuel assemblies described by the FQTs would all have decay heat within the limits of their appropriate HLZC.

Section 5.4.1.4 and Note D for Tables 8.7.1-2 through Table 8.7.1-30 and Note C for Tables 8.7.2-2 through Table 8.7.2-10 state conditions for loading fuel with shorter cooling times then those required in the standard FQTs. These sections state: *"Condition 1: the total decay heat of this FA is verified by means other than the standard FQTs to ensure that the total decay heat of this FA is less than the decay heat limit of* *the zone in where this FA is loaded."* This statement implies that fuel assemblies meeting the FQTs which were developed to limit dose rate also meets decay heat requirements. The staff notes that decay heat and dose rate are not necessarily correlated (Cumberland, Riley, Radulescu, Georgeta, and Banerjee, Kaushik. "The Relationship Between Dose Rate and Decay Heat for Spent Nuclear Fuel Casks". United States.) and typically cask loading procedures require an independent determination that fuel assemblies meet decay heat requirements because FQTs developed with the goal of meeting dose rate requirements may not correlate.

The staff sampled from allowable burnup/enrichment/cooling times (BECTs) from Tables 8.7.1-2, 8.7.1-3 and 8.7.2-2 of the application and estimated the decay heat using RG 3.54, Rev. 2 and a significant number of pressurized-water reactor (PWR) BECT combinations had a decay heat value that exceeded the allowable decay heat from the application in Figure 1.6.1-1 while nearly all of the boiling-water reactor (BWR) BECT combinations had a decay heat value that exceeded the allowable decay heat in Figure 1.6.2-1 of the application. Some burnup/enrichment combinations cannot be calculated using RG 3.54 Rev. 2 and the maximum burnup for BWR fuel exceed the applicability range of RG 3.54 Rev. 2. Also, EOS-37PTH loading plans 1 and 2 are designed to have different dose rate contributions for each zone, but not different decay heat.

If the intent for FQTs in Appendix 8.7, is to qualify fuel assemblies to meet decay heat limits, justify that a fuel assembly with each allowable BECT meets the decay heat limit. Describe the evaluation method along with other depletion assumptions used. Provide procedures for how to determine decay heat for fuel assemblies that contain control components or stainless steel replacement rods. Although the outer zone has additional cooling time to compensate, the inner zones do not, and a control component or stainless steel replacement rod could increase decay heat for these assemblies.

For exceptions to these FQTs, where: *"the total decay heat of this FA is verified by means other than the standard FQTs,"* discuss the other means used to determine decay heat and add this to the operating procedures as necessary.

This information is needed to determine compliance with the requirements in 10 CFR 71.35(b)(7).

5-5 Explain the difference between "maximum planar average burnup" and "FA average burnup" and how this accounts for the uncertainty associated with the depletion of fuel assemblies with large axial blankets. Also, for assemblies with axial blankets less than 5% active fuel length, justify that no additional uncertainty is needed.

The applicant states in Section 5.2: *"Fuel may contain axial blankets, which are regions at the end of the FA that contain natural or depleted uranium. To account for the uncertainty associated with the depletion of fuel assemblies with large (>5% active fuel length) natural uranium blankets, the fuel qualification is based on the maximum planar average burnup instead of the FA average burnup."*

Blanketed fuel is characterized by a more peaked axial burnup profile compared with non-blanketed fuel which can result in an underprediction in dose rates at the axial peak.

This information is needed to determine compliance with the requirements in 10 CFR 71.47(b)(2).

5-6 Provide additional information justifying the amount of cobalt assumed for the fuel hardware.

a. Provide additional information justifying that **[[**proprietary information removed**]]** is an appropriate assumption to account for the Co-59 impurity within fuel structural materials. The applicant states that **[[**proprietary information removed**]]**. Although the staff understands this concept in general to be true, the staff does not have information on **[[**proprietary information removed**]]**. Also, **[[**proprietary information removed**]]** than the values listed for Inconel materials that cited in Table 5.1 of ORNL/TM-11008 which are around 5,000–6,500 ppm. Provide data justifying the **[[**proprietary information removed**]]** Co-59 impurity for all fuel structural materials for fuel assemblies that will be transported in the TN-Eagle-STC or NRC will include a condition within the certificate of compliance that limits Co-59 impurities for fuel assembly structural material.

b. Provide additional information explaining how the amount of cobalt in Tables 5-6 and 5-7 of the application were derived given the information in Tables 5-3 and 5-4 of the application. Tables 5-3 and 5-4 of the application give the mass of the materials (stainless steel, Inconel, zircalloy) that make up each fuel hardware zone (plenum, top nozzle, etc.). Section 5.2.2 and Table 5-5 of the application state that the cobalt in Inconel and stainless steel is assumed to be **[[**proprietary information removed**]]** while Table 5-5 of the application also states that the cobalt in zircalloy is **[[**proprietary information removed**]]**. However, when applying these amounts to the masses in Tables 5-3 and 5-4, this does not equal the cobalt amount listed in Tables 5-6 and 5-7 of the application which show the amount of cobalt input into the ORIGEN-ARP for the depletion/activation calculation. The amount of cobalt shown in Tables 5-6 and 5-7 of the application are, for most regions, significantly lower.

Cobalt 59 is activated into Co-60 which emits high energy gammas and is known to be a significant contributor to dose. The staff needs to ensure that the assumed amount of cobalt is appropriate to determine if the TN Eagle-STC meets regulatory dose rate limits.

This information is needed to determine compliance with requirements in 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-7 Provide additional content limits for neutron source assemblies (NSAs) and justify that the impact of the NSAs on the shielding performance of the TN Eagle-STC can be neglected.

NSAs can have a significant neutron source that could increase dose rate. Section 5.2.8 of the application states: **[["**proprietary information removed***."]]***

Provide the following additional information related to NSAs:

a. Provide Reference 19, "TN Document, DI-83016-006, "Neutron Sources," Rev. 0." Section 5.2.8 of the application states, *"***[["**proprietary information removed***."]]***"

b. Section 5.2.4 of the application states: **[[*"***proprietary information removed***"]]*** Update Appendix 1.6.1 of the application to include this restriction.

c. Update Section 1.6 of the application to include a limit to the neutron source for allowable NSAs.

d. Provide additional justification that **[[**proprietary information removed**.]]** Outer assemblies may provide some shielding from inner assemblies, however outer fuel assemblies tend to provide more significant gamma than neutron shielding. Additional neutron source from NSAs also produces secondary gammas.

e. For the TN-EAGLE-SC DSCs, Section 5.6.1.2.8 of the application states: **[[**proprietary information removed**.]]** However, there is no limit to the number of NSAs within the TN-Eagle-STC SC DSCs or a limit to the neutron source. Include a limit within Section 1.6 of the application for NSAs for the DSCs that contain them (32PT, 32PTH1, 24PT1 and FO/FC/FF) and justify that the **[[**proprietary information removed**.]]**

This information is needed to determine compliance with the requirements in 10 CFR 71.33(b)(3), 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-8 Provide additional information on how the relative error from the Monte Carlo method was incorporated into the response functions used to generate the FQTs.

Since MCNP is a Monte Carlo code, the calculated values have a relative error associated with them. This relative error needs to be considered in the dose rate evaluation.

This information is needed to determine compliance with the requirements in 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-9 Update the drawings and/or acceptance test procedures to include the minimum density and composition of the VYAL-B resin.

The drawings do not specify the composition or minimum density of the VYAL-B resin used for the neutron shield. Section 9.1.7.1 of the application specifies the acceptance tests for the neutron shielding material and it only specifies a minimum density and acceptance criteria for composition with respect to boron and hydrogen. Table 5-39 of the application shows that **[[**proprietary information removed**.]]** Alternatively provide sensitivity studies showing that neglecting the other VYAL-B elements does not have an impact on the package's ability to meet regulatory dose rates limits.

This information is needed to determine compliance with the requirements in 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-10 Clarify Table 1.6.1-1 of the application to clearly state what is allowed in HLZC2.

Table 1.6.1-1 of the application under *"Heat Load Zone Configuration (HLZC) and Fuel Qualification,"* states: "*Per Figure 1.6.1-1 for HLZC #1 or HLZC #2."* Clarify why this does not include Figure 1.6.1-2 for HLZC2. Also, it does not appear that damaged/failed fuel is allowed in HLZC2 of the EOS-37PTH, as there are no designated locations in Figure 1.6.1-2 of the application. Verify that this is the case.

This information is needed to determine compliance with the requirements in 10 CFR 71.33(b)(3), 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-11 Update the FQT in Section 8.7.1-1 of the application to be consistent with operating procedures in Section 8.4.1 of the application.

Section 8.4.1 of the application, Step 9, states: *"For EOS 37PTH DSCs, the cooling time of the fuel at time of shipment will be at least five years."* This is inconsistent with the loading tables in Section 8.7.1 of the application where Tables 8.7.1-2 through 8.7.1-29 for the EOS-37PTH DSC contain allowable cooling times less than 5 years. The applicant should update these tables to include the restriction that allowable cooling time must be at least 5 years so that loading requirements are consistent.

This information is needed to determine compliance with the requirements in 10 CFR 71.33(b)(3), 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-12 Update Note E for Tables 8.7.1-2 through 30 of the application to include the correct table reference.

Note E for Tables 8.7.1-2 through 30 of the application explains requirements for control components. It states: *"The maximum Co-60 equivalent activity for the CCs stored in the EOS-37PTH DSC is specified in Table 8.7.1-2."* Table 8.7.1-2 does not include this information. Provide this information or update the table reference.

This information is needed to determine compliance with the requirements in10 CFR 71.33(b)(3), 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

5-13 Update Appendix 1.6.1 of the application to be consistent with Note C to the FQTs in Appendix 8.7.1 of the application.

Appendix 1.6.1 of the application states that there are up to 37 assemblies allowed that include reconstituted fuel which includes irradiated stainless steel rods. There is no limit on the number of rods that are allowed, however there is a limit in Note C to the FQTs in Appendix 8.7.1 of the application. This is inconsistent with Appendix 1.6.2 of the application that limits the number of reconstituted steel rods per assembly to five, which is also limited by Note B to the FQTs in Appendix 8.7.2 of the application.

This information is needed to determine compliance with the requirements in 10 CFR 71.33(b)(3), 10 CFR 71.47(b)(2) and 10 CFR 71.51(a)(2).

## 6.0 Criticality Safety

6-1 Revise the application to clarify the loading requirements for damaged and failed Combustion Engineering (CE) 14×14 class fuel in the 32PT DSC with the 24 poison plate basket configuration.

Table 1.6.4-8 of the application provides the acceptable average initial enrichment, burnup, and cooling time combinations for damaged CE 14×14 class fuel loaded in the 32PT DSC with the 24 poison plate basket configuration. It is not clear, however, if these limits are applied to the damaged fuel only, or for all fuel in the DSC when damaged fuel is loaded. Similarly, Table 1.6.4-9 of the application provides the acceptable average initial enrichment, burnup, and cooling time combinations for failed CE 14×14 class fuel loaded in the 32PT DSC with the 24 poison plate basket configuration. It is not clear if these limits are applied to the failed fuel only, or for all fuel in the DSC when failed fuel is loaded. Revise the application to clarify the damaged, failed, and intact fuel loading requirements.

This information is needed to determine compliance with 10 CFR 71.33.

6-2 Revise the application to demonstrate that the TN Eagle-STC transportation package is within the range of applicability of the benchmarking analysis for both fresh fuel and burnup credit criticality analyses.

Table 6.8.1-7 of the application provides the critical benchmark experiment parameters for those experiments the applicant chose to benchmark the fresh fuel calculations with BWR fuel in the EOS-89BTH canister. Similarly, Tables 6.8.2-17 through 6.8.2-20 of the application provide the critical benchmark experiment parameters for those experiments the applicant chose to benchmark the fresh and burned fuel calculations with PWR fuel in the EOS-37PTH canister. The experiment parameters listed in these tables ($^{235}$U enrichment, fuel rod pitch, assembly separation, moderator-to-fuel volume ratio, hydrogen to fissile atom ratio (H/X), average fission energy group (AEG), and energy of the average lethargy of fission (EALF)) define the range of applicability of the benchmarking analysis. Provide a comparison of similar EOS-89BTH and EOS-37PTH canister system parameters to demonstrate that the TN Eagle system is within the range of applicability of the benchmarking analysis.

This information is required for the staff to determine compliance with the requirements in 10 CFR 71.55(b).

6-3 Revise the application to ensure that critical experiment modeling in the benchmarking analysis uses the same cross section data as was used in the modeling of the TN Eagle-STC package.

Section 6.8.2.4.1 of the application states that for all calculations, the TN Eagle-STC is modeled using either the CSAS5 or STARBUCS sequences of the SCALE 6.1.3 code, with ENDF/B-VII cross section data. However, Section 6.8.2.5.1 of the application states: "**[[**proprietary information removed**.]]**" Revise the criticality evaluation to ensure that the same methodology (code version and cross section library) was used to model the selected critical benchmark experiments as was used to perform the criticality evaluation of the TN Eagle system. Note that ANSI/ANS 8.24-2017, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," states that "The calculational methods and analysis techniques used to analyze the set of benchmarks should be the same as those used to analyze the system or process to which the validation is applied."

This information is required to determine compliance with regulatory requirements in 10 CFR 71.55(b).

6-4 Revise the criticality analysis for failed CE 14×14 class PWR fuel in the 32PT DSC with the 24 poison plate basket configuration to consider rod removal.

Section 6.8.5.5.1 of the application states that **[[**proprietary information removed**.]]**. The damaged and failed fuel configurations modeled in the EOS-37PTH DSC in Section 6.8.2 of the application considered various failed fuel assembly configurations, including the removal of multiple fuel rods from the lattice. Some of these configurations showed that the system was more reactive than without rod removal. Failed fuel assemblies may include fuel debris or partial fuel assemblies, which may be more reactive than a full lattice of rods and should be considered in the failed fuel evaluation to bound potential failed fuel configurations.

This information is required to determine compliance with the requirements in 10 CFR 71.55(b).

**Materials Evaluation**

7-1 Provide additional information to address the unloading of a DSC with high burnup fuel as described in SAR Section 8.2.2 and Section 8.4.2 as follows:

a. Clarify whether the TN Eagle-STC transportation package would be immersed in borated water as described in SAR Section 8.4.2 Step 4.

b. If the TN Eagle-STC transportation package would be immersed in borated water as described in SAR Section 8.4.2 Step 4, provide an evaluation of the potential corrosion reactions and the means to mitigate the potential for the generation of flammable gas such as hydrogen during the unloading operation while the TN Eagle-STC transportation package is immersed.

c. If the TN Eagle-STC transportation package would be immersed in borated water as described in SAR Section 8.4.2 Step 4, provide a description of the transfer operation that would need to occur prior to the immersion and reflooding operation described in SAR Section 8.4.2 Step 4.

SAR Sections 8.2.2 and 8.4.2 states that unloading of the DSC may take place in a hot cell or in borated water if the DSC contains high burnup fuel assemblies and the cask/DSC annulus gas sampling indicates the presence of airborne radioactive particles, which would be an indication that the DSC confinement barrier has been compromised. It is not clear if the DSC would still be inside the TN Eagle-STC transportation package for the unloading operations in borated water as described in SAR Section 8.4.2 Step 4, or if the DSC would be transferred to a transfer cask prior to immersion in borated water.

This information is needed to determine compliance with the requirements in 10 CFR 71.43(d).

7-2 Provide additional information regarding the provisions to ensure that neutron streaming will not occur as a result of resin shrinkage under conditions of extreme cold. Provide additional information regarding the acceptance tests conducted to confirm the neutron shielding effectiveness and verification of lack of voiding or defects.

This information is needed to determine compliance with the requirements in 10 CFR 71.43(f), 10 CFR 71.85(a), 10 CFR 71.87(b), and 10 CFR 71.93(b).

7-3 Provide additional information regarding the periodic tests or evaluations to ensure that the neutron shielding effectiveness will not degrade on account of the combined aging effects of heat and radiation field.

This information is needed to determine compliance with the requirements in 71.43(f), 10 CFR 71.85(a), 10 CFR 71.87(b), and 10 CFR 71.93(b).

7-4 Provide additional information on the zinc-aluminum thermal spray described in Section 7.8 of the SAR as follows:

a. Provide additional information to support the statement, "*This coating is more than adequate for the dry loading environment of the TN Eagle interior*," in Section 7.8.4 of the SAR, including the results of coating qualification tests, temperature compatibility, resistance to radiation damage, and resistance to abrasion that may be encountered in loading and unloading operations. Examples of successful prior use on transfer casks or transportation packages may also be included.

b. Explain why the thermal spray coating does not appear on the drawings, nor identified in the maintenance program in Section 9.2 of the SAR (coating inspection is limited to exterior surfaces of the cask component). Table 7-1 refers to the thermal spray as an anticorrosion coating, but it is unclear if this is credited as a coating that is important to safety, which would therefore need to be inspected. Please clarify and modify the SAR accordingly.

c. Provide an explanation of the International Standards Organization Standard No. 12944-5, "Paints and varnishes — Corrosion protection of steel structures by protective paint systems — Part 5: Protective paint systems," standard that is referenced a single time in the application Section 7.8.3. This standard is included as a parenthetical without any context or discussion, and it does not appear in the list of references at the end of Chapter 7.

This information is needed to verify compliance with the requirements of 10 CFR 71.31(c) and 10 CFR 71.43(d).

7-5 Provide an explanation for the differences in the tensile properties shown in Table 7-8 and 7-11. The table captions indicate that these tables are for the same material, but the tensile strength (Su) and yield strength (Sy) values are not consistent.

This information is needed to determine compliance with the requirements of 10 CFR 71.31(c).

7-6 Provide additional information regarding the contents of Table 7-8A.

a. Provide justification for references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code materials properties in Table 7-8A when the material used is not an ASME B&PV Code material. These references are misleading and inaccurate and could infer that an ASME B&PV Code material is being used for this application.

b. Provide justification for the use of testing data to demonstrate stress values. Limited testing results could bias the results and be influenced by conditions of the testing program, resulting in a nonconservative material value. Provide information on the programmatic controls to ensure that each heat of material is bounded by the values provided in the limited testing program cited for the material properties. Demonstrate that the sample size used for the limited testing program provides statistical confidence that the minimum values assumed are bounding. Alternatively, provide a copy of EN 10269 and identify the applicable stress values from the standard incorporated into SAR Table 7-8A including the temperature dependent values for yield strength, tensile strength, elastic modulus, and design stress intensity values.

This information is needed to determine compliance with the requirements of 10 CFR 71.31(c).

7-7 Provide additional information regarding the contents of Table 7-11

a. Clarify the source of information for the modulus of elasticity as a function of temperature for ASME SA-540/540M B24/B23 Class 2 material. Table 7-11 identifies the source of the information as p.835 of ASME B&PV Code, Section II, Part D (Metric) Table TM-1, for Group B materials. However, ASME B&PV Code, Table TM-1 Group B materials does not identify the 2Ni–3/4Cr–1/3Mo material. Further, available data on the room temperature elastic modulus for AISI 4340 steels are higher than the value shown in ASME B&PV Code, Section II, Part D (Metric) Table TM-1, for Group B materials.

b. Clarify the source of information for the calculation of the thermal conductivity and heat capacity as a function of temperature. SAR Table 7-11 states that these values were calculated using the information in ASME B&PV Code, Section II, Part D (Metric) Table TCD, p. 819, for Group D materials. However, ASME B&PV Code, Section II, Part D (Metric) page 819 contains thermal expansion coefficient data for Nickel alloys.

This information is needed to determine compliance with the requirements of 10 CFR 71.31(c).

7-8 See Enclosure 2.

7-9 See Enclosure 2.

7-10 Provide additional information to justify the maximum long term temperature limit of **[[**proprietary information removed**]]** provided in the SAR for the VYAL-B resin and how it will ensure acceptable material performance over the life of the package. Further, the SAR states that the VYAL-B will exceed this maximum temperature limit during NCT. Provide additional information to demonstrate how the assumption of damage for this resin outside of the maximum temperature limit is verified as bounding.

This information is needed to determine compliance with the requirements of 71.43(f).

7-11 Provide additional information regarding the silicone-acrylic paint system, as follows:

a. The qualification process did not assess the effects of radiation on the coating system. Any coating used for a transportation package must have been tested to demonstrate the coating's performance under all conditions of loading and transportation, including the regulatory test conditions. The conditions evaluated should include exposure to radiation, unloading, and transfer operations. Provide information supporting how the paint is qualified for the radiation environment. The staff notes that Section 7.10 of the SAR mentions that the silicone-acrylic paint could be affected by gamma irradiation but does not later discuss the maximum total gamma energy deposit rate as it does for the resin and polytetrafluoroethylene (PTFE) materials.

b. The thermal aging was conducted at a maximum temperature of **[[**proprietary information removed**]]** but Table 3-8 of the SAR provides a maximum cask body temperature of 367 °Fahrenheit for load case 1. Provide information that addresses this apparent exceedance of the qualification bounds.

The document NTE-20-031053-000 Rev. 1, "International system coating qualification," provides a qualification process for the silicone-acrylic paint used on the exposed carbon and low alloy steel on the outside surface of the cask body. This process assessed resistance to solvents, borated water, and thermal aging; however, the testing was not conducted over a range of temperatures that bound the maximum temperature under NCT and does not include data on radiation resistance.

This information is needed to verify compliance with the requirements of 10 CFR 71.43(d).

## 8.0 Operating Procedures Evaluation

8-1 Provide procedures for controlling the radiation level limits on unloading operations and procedures for addressing situations when surface contamination and radiation surveys are too high.

Step 6 in Section 8.2.1 of the operating procedures states: *"Verify that the cask surface removable contamination levels meet the requirements of 49 CFR 173.443."* Step 7 of the same section states: *"Perform a radiation survey of the cask to verify compliance with 10 CFR 71.47."* However, there are no procedures associated with controlling radiation level limits and addressing situations when contamination and radiation levels exceed regulatory limits. RG 7.7, "Administrative Guide for Verifying Compliance with Packaging Requirements for Shipping and Receiving of Radioactive Material" Section 4.0, "Receiving and Opening a Package," contains an approach that the staff considers acceptable for meeting the requirements associated with receipt of radioactive material in 10 CFR Part 71 and 10 CFR Part 20. In addition, NUREG/CR-4775, "Guide for Preparing Operating Procedures for Shipping Packages," may contain useful information.

This information is needed for a user to be able to comply with 10 CFR 71.89 and 10 CFR 20.1906.

8-2 Provide justification that a review of DSC loading reports provides adequate information to verify that the DSC was not damaged during the insertion or extraction process. Alternatively, provide an inspection process to verify that the DSC is in unimpaired physical condition. Provide qualitative acceptance criteria to verify the performance of the DSC shell to the original design requirements for safety functions and that package is unimpaired physical condition.

Section 8.4.1, “DSC Evaluation for Transport,” Step 6, states that, “for DSCs being stored under the initial 10 CFR 72 licensed period, the loading reports were reviewed to ensure the DSC was not damaged during the insertion or extraction process and that, if necessary, appropriate evaluations were performed to verify the performance of the DSC shell to the original design requirements for safety functions.”

NUREG 2216, Section 8.1.1.1, “Preparation for Loading,” states in part, to “verify that the application describes the procedures for package loading preparations sequentially in the order of performance, and ensure that the procedure descriptions, at a minimum, assure… the package is in unimpaired physical condition.”

10 CFR 71.87, “Routine determinations” states, in part, that, “before each shipment of licensed material, the licensee shall ensure that the package with its contents satisfies the applicable requirements of this part and of the license. The licensee shall determine that… (b) The package is in unimpaired physical condition except for superficial defects such as marks or dents”

The NRC staff was unable to determine whether the loading procedures provided an adequate level of inspection and acceptance criteria to make the determination that the package is in unimpaired physical condition except for superficial defects such as marks or dents.

This information is needed for a user to be able to comply with 10 CFR 71.31(c), 71.35(c) and 10 CFR 71.87(b).

8-3 Revise the operating procedures to include steps for lifting a package without impact limiters during its transit from the point of origin to destination.

In addition to either a drop analysis or commitment for a single failure proof lifting devise (see item 2-1, above), lifting the package without the package’s impact limiters is a safety significant operation during transport.

This information is needed to determine compliance with 10 CFR 71.35(c).

## 9.0 Acceptance Tests and Maintenance Program

9-1 Include in the SAR a description of an appropriate periodic thermal test (to be performed usually every 5 years) or provide adequate technical justification why a periodic thermal test is not needed.

SAR Section 9.2.6 states that there are no periodic thermal tests required. However, adequate technical justification was not provided. The staff needs this information to verify that the heat-transfer capability of the packaging during its time in service are maintained.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and 10 CFR 71.73(c)(4)

9-2 Provide additional information on the neutron shielding material tests and maintenance program.

*a.* Section 9.1.7 of the application states: *"Acceptance testing of the neutron shield material shall be performed to verify the following criteria are met: a) Resin density is 1.75 g/cm³ or greater, b) Amount of hydrogen (H) in the resin is between 4.6 and 5.4 weight %, c) Amount of boron (B) in the resin is between 0.82 and 1 weight %."* Provide additional details on how the testing will ensure the neutron shield is functional for the package as-fabricated. Material testing before the neutron shield is installed would not address any defects resulting from the processes associated with fabrication and installation.

*b.* Section 9.2.5 states: *"There are no periodic shielding tests required."* Provide procedures including acceptance criteria for periodic tests verifying the neutron shield efficacy. Section 7.6.1 of the application states: **[[**proprietary information removed**.]]**

This information is needed so that staff can determine if appropriate procedures are in place to ensure compliance with regulations in 10 CFR 71.85(a), 10 CFR 71.85(c), 10 CFR 71.87(b), and 10 CFR 71.93(b).

9-3 Revise Section, 9.1.3, "Structural and Pressure Tests," to include pressure tests for each fabricated cask assembly, rather than first cask assembly (any model) fabricated to the design provided in the applicable drawings for package approval. Revise Section 9.1.3 to include ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6100, and NB-6400 with NB-6200.

Section, 9.1.3, "Structural and Pressure Tests," states that, "*A pressure test shall be performed on the first cask assembly (any model) fabricated to the design provided in the applicable drawings for package approval. Any change to the design that could impact the structural performance will require a new pressure test. The test pressure is between 20.0 and 25.0 psig and held for a minimum of 10 minutes. All visible joints and surfaces shall be visually examined for possible leakage after application of the pressure. The test shall be performed in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 [1].*"

Section 4.2, "Containment under Normal Condition of Transport," indicates that, "*The maximum normal operating pressure is calculated in Chapter 3 to be 83.4 kPa (12.1 psig) and the analyses in Chapter 2 demonstrate that the TN Eagle Cask effectively maintains containment leak-tight integrity with a cavity pressure of 83.4 kPa (12.1 psig).*"

Since the normal operating pressure has been determined to be 12.1 psig, a pressure test of the containment system is required in accordance with 10 CFR 71.85(c). The regulations in 10 CFR 71.85(c) requires this test be performed for "*any packaging*" not the "*first cask assembly (any model) fabricated to the design provided in the applicable drawings for package approval*." While a testing of the first cask assembly fabricated may show adequacy of the initial design, continued testing is necessary to demonstrated continued adequacy of fabrication of that design.

The hydrostatic pressure test shall be performed in accordance with all applicable portions of ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6000. Inclusion of the paragraphs NB-6100, and NB-6400 with NB-6200 ensure that adequacy of the test with appropriate general requirements and gauges.

This information is needed to comply with 10 CFR 71.85(c).

Editorial Comment:

Editorial comment: the first 2 paragraphs of Section 1.6.6.7 of the application are a repeat of Section 1.6.6.6 of the application.

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