November 22, 2011

Dr. Jayant Bondre, PhD
Vice-President – Engineering
Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: APPLICATION FOR CERTIFICATE OF COMPLIANCE NO. 9358 FOR THE MODEL NO. TN-LC PACKAGE – REQUEST FOR ADDITIONAL INFORMATION

Dear Dr. Bondre:

By letter dated August 17, 2011, you submitted your responses to the Request for Supplemental Information letter dated July 26, 2011, in connection with staff's acceptance review of your June 7, 2011, application for approval of the Model No. TN-LC as a Type B(U)F-96 package.

In connection with our detailed technical review, we need the information identified in the enclosure to this letter. We request that you provide this information by December 8, 2011. If you are unable to meet this deadline, you must notify us in writing no later than November 23, 2011, of your submittal date and the reasons for the delay. The staff will then assess the impact of the new submittal date and notify you of a revised schedule.

Please reference Docket No. 71-9358 and TAC No. L24543 in future correspondence related to this request. The staff is available to meet with you to discuss your proposed responses. If you have any questions regarding this matter, I may be contacted at (301) 492-3408.

Sincerely,

/RA/
Pierre Saverot, Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9358
TAC No. L24543

Enclosure: Request for Additional Information

November 22, 2011

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REQUEST FOR ADDITIONAL INFORMATION
FOR THE
MODEL NO. TN-LC PACKAGE

DOCKET NO. 71-9358

By application dated June 7, 2011, Transnuclear, Inc. (TN) submitted an application for approval of the Model No. TN-LC package. The NRC staff completed an acceptance review of this application on July 26, 2011. On August 17, 2011, TN submitted responses to staff's request for Supplemental Information.

This Request for Additional Information (RAI) identifies information needed by the staff in connection with its review of the Model No. TN-LC package application. The requested information is listed by chapter number and title in the applicant's Safety Analysis Report. The staff reviewed the application using the guidance in NUREG 1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

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

Chapter 1 General Information

1.1 Address the reason(s) for inconsistent statements about high burn up fuel with zirconium cladding that appear throughout the application, and provide a detailed description of the licensing basis for the Model No. TN-LC package.

As indicated in this RAI letter, several inconsistencies appear to indicate that this application does not establish a clear understanding of the licensing basis for high burn up fuel with zirconium cladding. For example, only criticality analyses consider bounding re-configured/damaged fuel geometries in accident conditions, thus taking no credit for the structural integrity of the fuel, without addressing reconfigured shielding and thermal sources. It is not clear why the statement that the fuel assemblies will "maintain their structural integrity during accident conditions of transportation" is then made. Similarly, the structural integrity of the payload fuel assemblies and elements during NCT side and end drops are evaluated but the material properties for high burn up fuel cladding are "unknown". The rationale for conjecturing "unknown" material properties for research reactor fuel data and using material properties after wet storage for dry transportation of commercial high burnup fuel needs to be addressed. In addition, a justification must be made on how the effects of vibration under NCT will not alter the geometric form of the high-burn up zirconium fuel cladding.

Thus, the application should describe the licensing basis of the package and justify every assumption made.

This information is required by the staff to determine compliance with 10 CFR 71.31(a), 71.33(b), 71.35(a), 71.47, and 71.51, 71.55(d)(2) and 71.71(c)(5).

Licensing Drawings

1.2 Amend all the engineering notes which permit deviations from the licensing drawings in the application. Include language requiring that the change meet the code of construction and with approval of the Certificate of Compliance (CoC) holder (TN). This includes alternate welding configurations. Examples of these engineering notes include notes 10 and 19 from drawing 65200-71-01.

Any changes to the design of the packaging should be in compliance with the code of construction and be approved by the CoC holder.

This information is required by staff to demonstrate compliance with 10 CFR 71.31(c) and 71.33(a)(5)(iii).

1.3 Justify the use of plate material and castings for the inner and outer shell, cited on Engineering note 24 on Licensing Drawing No. 65200-71-01. Limit the fabrication of the shell to only one material per TN-LC package.

The staff is concerned about the reduction of the strength and ductility that may occur when using alternative material processing techniques to produce safety related components, particularly of castings.

This information is required by staff to demonstrate compliance with 10 CFR 71.33(a)(5)(iii).

1.4 Provide all the minimum weld thickness on the licensing drawings.

All of the welds on the licensing drawings should indicate a minimum weld thickness that conforms to structural analysis in the safety analysis report and with the code of construction.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii).

1.5 Define equivalency for the glue used to fasten the wood together for the impact limiters.

The term "Equivalent" on Engineering note 7 of Licensing Drawing No. 65200-71-20 should be justified by listing the critical characteristics of the glue. The licensing drawings should specify that the CoC holder's approval is required for approval of equivalent glues.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii).

1.6 Clarify the consequences if the corners of the support spacer specified in Engineering note 20 on Licensing Drawing No. 65200-71-80 extended beyond the rail diameter.

Note 20 on Licensing Drawing No. 65200-71-80 makes an ambiguous statement that the "corners *should* not protrude past rail diameter."

This information is required by staff to determine compliance with 10 CFR 71.43(f).

1.7 Edit the Engineering Notes on the Licensing Drawings so that backing bars are removed prior to all weld examination techniques. Alternatively, justify (using the applicable code of construction) that retaining the backing bars after welding is an acceptable practice.

Backing bars are meant to be temporary construction aids, not permanent attachments to weldments.

This information is required by staff to determine compliance with 10 CFR 71.31(c).

1.8 Specify a welding code, (e.g., Section III, Subsection NF of the ASME Code or AWS) for the construction of the impact limiters.

No welding code is specified for impact limiters.

This information is required by staff to determine compliance with 10 CFR 71.31(c).

1.9 Specify a welding code, (e.g., Section III, Subsection NG of the ASME Code) for the TN-LC-1FA basket on the licensing drawings.

No construction code is specified for the basket.

This information is required by staff to determine compliance with 10 CFR 71.31(c).

1.10 Justify why the ASME SA-193 B8 bolts do not comply with Section III, Subsection NG of the ASME Code and do not have a Safety Category A. Clarify what the acronym "SHCS" means. See Licensing Drawing No. 65200-71-50, sheet 1 of 5.

The ASME SA-193 B8 bolts appear to be an integral part of the basket assembly, functioning to keep the contents in place and should therefore be constructed to similar level of quality. The term "SHCS" does not appear in the application.

This information is required by staff to determine compliance with 10 CFR 71.31(c) and 71.33(a)(5)(iii).

1.11 Identify on what sheet of Licensing Drawing No. 65200-71-40 the "Pipe" is shown.

Although listed in the Bill of Materials, the staff has not located Item #9, "Pipe."

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii).

1.12 Justify why the support spacer and other spacer components of the TN-LC-TRIGA basket do not comply with Section III, Subsection NG, of the ASME Code as Safety Category of A.

The support spacer and other spacers appear to be integral parts of the basket assembly and should therefore be constructed to a similar level of quality.

This information is required by staff to determine compliance with 10 CFR 71.31(c).

1.13 Describe, on the licensing drawings, the Poison Rod Assemblies (PRAs) used for criticality control for the PWR fuel assemblies.

Only components which are listed in the licensing drawings or adequately described in the CoC can be given credit for criticality control.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(ii).

1.14 Justify the code of construction and Safety Classification for the ASME SA-540 Gr. B23 Class1 impact limiter attachment bolts.

The impact limiter bolts are made of ferritic steel, which are susceptible to nil-ductility temperature embrittlement. Acceptance testing under a code of construction, e.g., Subsection NB, of the bolt material should be conducted to ensure that the bolts have adequate ductility at low temperatures. The SA-540 Gr.B23 bolts for containment will undergo low-temperature impact testing according to Subsection NB of the code. It is necessary that the impact limiter attachment bolts also undergo similar testing, since their ductility is necessary to maintain the function of the impact limiters during HAC.

This information is required by staff to determine compliance with 71.73(c)(1).

1.15 Remove the reference to MT on the code exception to NB-5221 and clarify the number of PT layers that will be used in lieu of a volumetric examination.

The inner and outer shells are non-magnetic, therefore MT is not an applicable inspection technique. The number of PT layers used in lieu of a volumetric examination should be clarified to ensure its adequacy.

This information is required by staff to determine compliance with 10 CFR 71.31(c) and 71.33(a)(5)(iii).

1.16 Justify the 1.0 joint equivalency factor for the tube longitudinal welds of the fuel basket.

Additional supporting documentation should be provided to the staff to make a safety evaluation, e.g., ASME Code cases, demonstrating that a joint equivalency factor 1.0 is a valid engineering assumption.

This information is required by staff to determine compliance with 10 CFR 71.31(c) and 71.33(a)(5)(iii).

1.17 Remove Engineering note 21 and note 22 from Licensing Drawing No. 65200-71-01, and all such similar notes on other licensing drawings. Clarify why the statement of note 21 “all welds shall be visually inspected to verify compliance with the requirements of NB-4424 or NF-4424” contradicts note 2 of drawing 65200-71-40 “All welds shall be visually examined in accordance with NG-5361 and NG-5362, as applicable.”

References to sub-tier sections of ASME Code are already incorporated into the code of construction and can be misinterpreted such that these specific sub-tier sections are used at the exclusion of other sub-tier sections of the code of construction.

The visual acceptance is also more stringent than NB-4424 or NF-4424. The applicant must specify which is the prevailing code.

This information is required by staff to determine compliance with 10 CFR 71.31(c) and 71.33(a)(5)(iii).

Chapter 2 Structural Evaluation

2.1 Provide a summary description of the impact limiter design criteria to clarify how the lock-up strain limits for the balsa wood and redwood are considered so that they can properly be modeled in the LS-DYNA cask analysis for the free drop tests and conditions. See page 2-7, Impact Limiters, of the application.

Section 2.1.2 of the application makes reference to Appendix 2.13.12 for the design of the impact limiters. The stress-strain curves of Figure 2.13.12-20 for the balsa display a lock-up strain of about 80%. The lock-up strain for the redwood shown in Figure 2.13.12-22 is about 60%. It is unclear how the lock-up strain limits are considered as design criteria and modeled in the LS-DYNA analysis of the package free drop tests and conditions.

This information is required by staff to determine compliance with 10 CFR 71.71(c)(1) and 71.73(c)(3)

2.2 Identify and tabulate individual weights and centers of gravity for all relevant package components, including applicable basket and payload configurations.

The application states that the center of gravity of the package is located on the axial centerline between 88.87 inches and 110.6 inches from the base of the package, depending on the basket in use. This characterization of the package weights configuration is insufficient for a safety evaluation. For instance, as reviewed in a latter RAI for the baseline cask rigid body decelerations calculated by the impact limiter finite element analysis, the information for the package's centers of gravity, including bounding locations, must be properly considered.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.3 Identify and re-assess the weakest part of the load path.

Section 2.5.1 of the application states that, for excessive load evaluation, "…the lowest safety margin occurs at the trunnion." However, based on the stress summary tabulated in page 2-16 and in Table 2.13.5-6, the lowest stress safety margin of 0.47 occurs at the outer cask shell, which is structurally an integral part of the package.

This information is required by staff to determine compliance with 10 CFR 71.45(a).

2.4 Clarify the inconsistent description presented in Section 2.6.7.1, “The load combination performed to evaluate these drop events are indicated in Table 2-8. In all cases, bolt preload effects and fabrication stress are included,” by noting that the fabrication stress is not included in Table 2-8, “Summary NTC Load Combinations.”

Table 2-8 does not appear to have considered fabrication stress, such as that of pouring lead into the annulus between the package’s inner and outer shells. See also the RAI below on considering fabrication stresses for cask component stress evaluation.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.5 Clarify and remove, as appropriate, the use of the terminology, “combined stress intensities,” for describing load combination stress intensity results in Section 2.6.7.1.

Use of the subject undefined terminology is confusing. Load combination stress intensities are often computed for the cask containment boundary evaluation by the ASME Code, Section III, Division 1, Subsection NB stress criteria. The terminology, “combined stress intensities,” however, has no meaning and should be modified to recognize proper use of ASME Code provisions.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.6 Provide a summary clarification for the statement in Section 2.6.11, “[t]he fabrication stresses remaining in the cask components at the time the cask will be used for transportation will be insignificant,” regarding the lead pouring between the inner and outer shells.

Position 1.5 in Regulatory Guide 7.8, “Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material,” states that fabrication stresses should be considered in determining the maximum resultant vessel stress. The applicable stress as related to lead pouring should be quantified even if it is considered insignificant in the context of fabrication stresses evaluation.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.7 Clarify the statement found in Section 2.7.1.4, “For cask body structural evaluation, the top and bottom ends of the cask are analyzed for 130 g acceleration… Therefore, the middle portion of the cask is bounded by the 75 g side drop evaluation.”

Provide a free-body diagram sketch to identify how the 130 g is applied as a side impact load for the cask analysis.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.8 Clarify the statement found in Section 2.11, “The structural adequacy of the fuel rods in the various baskets is not relied upon to provide containment of radioactive material under hypothetical conditions,” by noting that if analyzed fuel configurations are relied on for demonstrating criticality safety, it should also be recognized in this section of the

application.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.9 Replace the word “or” by “and” for the HAC (Level D) stress limits, in Table 2-3, “Cask Stress Limits,” for the P_m, P_l, and $(P_m + P_l) + P_b$ stresses by recognizing that both stress limit criteria are considered in the stress evaluation in accordance with ASME Code, Section III, Appendix F, provisions.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.10 Add a footnote to the tables, as appropriate, if the residual stress, including the lead pouring fabrication stress, is determined to be insignificant and needs not to be considered for meeting the Regulatory Guide 7.8 load combinations provisions.

Position 1.5, Regulatory Guide 7.8, provides that fabrication stresses should be considered in load combination evaluation of the package performance.

This information is required by staff to determine compliance with 10 CFR 71.17.

Appendix 2.13.1 TN-LC Cask Body Structural Analysis

2.11 Define which portion of the cask body shell is considered as the top and the bottom ends for stress analysis and reporting for the baseline 130 g side impact analysis in Section 2.13.1.6.1.4, “Slap Down Drop Evaluation.” Additionally, provide a confirmatory analysis using a more realistic loading assumption of linearly distributed side impact to demonstrate that the inner shell containment boundary stress is conservatively evaluated.

Tables 2.13.1-46 and 2.13.1-53 list stress results, based on plastic analysis, for the cask subject to the side impact of 75 g and 130 g, respectively. While maximum stress in the outer shell for the former load case is seen about 15% less than the latter, the maximum stresses in the inner shell are seen barely different at 49.9 ksi and 48.5 ksi for the two side impact conditions. Furthermore, the inner shell maximum stress of 49.9 ksi associated with the 75 g impact, shouldn’t be shown to be higher than that of 48.5 ksi of the case of 130 g side impact.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.12 Revise Tables 2.13.1-19 through 2.13.1-22 and Tables 2.13.1-37 through 2.13.1-40 to ensure all weight units are properly used.

The weight unit, kip, does not appear to be consistent with the listed values.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.13 Verify the stress allowables for the package top and bottom flanges and re-evaluate stress margins for the package components, as appropriate, in Tables 2.13.1-46 and 2.13.1-48.

There appears to be errors in entering the stress allowables into these tables.

Appendix 2.13.5 TN-LC Cask Lifting and Tie-Down Devices Structural Evaluations

2.14 Clarify the statement in Section 2.13.5.1.1, "10 CFR 71.45(a) requires that a minimum factor of safety of three and five are needed against material yields and ultimate strengths, respectively,…" by removing the reference to evaluate the factor of safety against material ultimate strength.

The regulation is incorrectly noted to also include an evaluation of stress factor of safety evaluation against material ultimate strength.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.15 Correct, as appropriate, the underscored typographical error on page 2.13.5-2, in particular, by recognizing that the cited trunnion drawing should read, "…drawing 65200-71-01, sheet 5."

This information is required by staff to determine compliance with 10 CFR 71.17.

2.16 Revise the sketch in Figure 2.13.5-1, "Trunnion Flange Stresses," by providing the section cut A-A with sufficient annotations to depict the details and locations considered for trunnion stress analyses presented in pages 2.13.5-2 through 2.13.5-8.

Section A-A, as identified, is not presented in Figure 2.13.5-1. Sufficient details should be provided to facilitate staff review of the trunnion analysis.

This information is required by staff to determine compliance with 10 CFR 71.17.

Appendix 2.13.7 TN-LC Lid Closure Evaluation Due to Delayed Impact

2.17 Clarify the statement, in page 2.13.7-6, regarding the bolt shear stress plot presented in Figure 2.13.7-16: "For the lid end drop there are no shear stresses; thus the tension plus shear stress limit is bounded by the tension limit," by noting that, if the "shear stress" at the bolt periphery is used as an indirect reference in the computational software for calculating the primary bending stress intensity, $P_l + P_b$, it should clearly be described and evaluated in the application.

As annotated in Figure 2.13.7-16, contrary to the "no shear stresses" assessment, the maximum shear stress is reported to be at 76.654 ksi. This inconsistency in reporting bolt shear stress or lack of it is confusing.

This information is required by staff to determine compliance with 10 CFR 71.17.

Appendix 2.13.8 TN-LC Basket Structural Evaluation

2.18 Provide the basis for the ASME Code bolt shear stress allowables of $0.8S_m$ and $0.6S_y$ for the evaluation of the 1FA basket in NCT and HAC conditions, respectively. See Section 2.13.8.5.5, "Frame/Sleeve Bolt Evaluation."

The application summarizes ASME Code, Section III, Subsection NG, stress allowables for stress evaluation of basket components except the bolts. It is unclear whether the ASME Code bolt stress acceptance criteria, including shear stress, have been properly considered for demonstrating the structural integrity of all 1FA basket components.

This information is required by staff to determine compliance with 10 CFR 71.17.

Appendix 2.13.11 TN-LC Fuel Assemblies and Fuel Elements under Impact Loads

2.19 Provide sketches depicting structural design attributes, including mechanical properties of the grid spacers as well as the clad and the U-AL fuel materials, for the NRU fuel assembly considered in the package side- and end-drop analyses.

The application notes the Young's modulus, E, of 9.1 x 10^6 psi and the yield stress, S_y, of 8,000 psi for evaluating the NRU/NRX fuel assembly. Table 2.13.11-3 lists a 0.03-inch thick clad and the U-AL fuel, which suggests that the fuel clad alone is counted for load resisting purpose. However, page 2.13.11-5 states, "For NRU/NRX, the fuel rod is modeled by beam elements since the fuel rod is solid." It is unclear whether the clad-fuel composite action is considered in the modeling for analyzing fuel rod response. The staff notes that, if a composite action assumption, as an artifact, is made for the fuel rod, to markedly increase the fuel rod load-resisting capability, such an assumption must be properly justified.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.20 Verify the statement in Section 2.13.11.1.2.2, "Side Drop Analysis." "The…NRU fuel cladding modeled by solid element…" by comparing it with the Section 2.13.11.1.3.3 statement, "the fuel cladding is modeled using beam elements."

Those two statements provide a different modeling description for the NRU fuel rod.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.21 Verify that correct fuel clad thickness reported in Tables 2.13.11-1 and 2.13.11-14 is considered for the PWR 16 x 16 fuel assembly in evaluating the fuel rod drop accidents.

Table 2.13.11-1 lists a clad thickness of 0.0198 inches for the PWR 16 x 16 fuel while Table 2.13.11-14 reports 0.0250 inches for the same fuel.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.22 Provide calculations to demonstrate that the end fitting lateral and rotational spring constants of 40,000 and 20,000 as Real Constants 9 and 10, respectively, are conservatively implemented for calculating peak bending stresses for the BWR fuel clads. See Figure 2.13.11-5, Finite Element Model Setup for BWR.

As depicted in Figure 2.13.11-6 for the model constraints assumed for the BWR fuel assembly and Figures 2.13.11-18 through 2.13.11-21 for the calculated bending stress diagrams, an inflection point is introduced to the "cantilevered" segment of the fuel rod

beyond the last hinge support. As such, the curvature reversal associated with the prescribed translational and rotational constraints in the respective UY and ROTZ degrees of freedom appear to have significantly brought down the calculated bending stresses of the rod. Since calculated maximum bending stresses may vary markedly, depending on the boundary conditions imposed, the constraints associated with the subject translational and rotational spring stiffness must be conservatively estimated.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.23 Verify that the maximum principal strain is used for the subject time-history response plot. See Figure 2.13.11-37, Maximum Principal Strain Time-History of NRU Rod.

It is unclear what the "maximum value data" labeled for the vertical plot axis stand for. The plot appears to suggest, however, that the complete fuel rod is always subject to compressive stress as the maximum principal strain is the algebraically largest, which reads "zero," in the plot. Given that a rod bowing of 0.01 inches is prescribed for the model, the rod is expected to also respond with positive maximum principal maximum strain during the end-drop event.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

Appendix 2.13.12 TN-LC Impact Limiter Analysis Using LS-DYNA

2.24 Provide appropriate "bolt strain" time-history response plots, for the package free drop with maximum calculated bolt stress of 148.6 ksi, to demonstrate that the high-strength bolts will not exceed the elongation limits upon the secondary impact of the tail impact limiter. The elongation capability of the SA-540 Gr. B23, Class 1, bolts should be shown bounding the calculated strains for demonstrating the bolt adequacy.

Page 2.13.12-12 notes the at-temperature bolt tensile strength, S_u, of 165 ksi at 300°F. The maximum calculated bolt stress of 148.6 ksi, however, exceeds the at-temperature bolt yield strength of 140.3 ksi. Given that elongation capability of the SA-540 Gr. B23, Class 1, bolts may be relatively limited and that the impact limiter components may have undergone a large deformation, a simple stress margin evaluation against the ASME Code, Section III, stress based acceptance criteria may not be sufficient for demonstrating the bolt structural adequacy.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.25 Re-evaluate the slapdown drop analyses, as appropriate, which were used for determining the baseline rigid body decelerations for cask structural evaluation in view of the bounding shift of the cask center of gravity location between 88.75 inches and 110.65 inches from the base of the package. See page 2.13.12-6 of the application.

Section 2.1.3 of the application notes the cask center of gravity location, on the axial centerline, at between 88.75" and 110.65" from the cask base. However, as noted in "Cask Model," the cask drop analysis considers a uniform weight distribution with the center of gravity located at the geometric center, 98.75," from either end. Effects of this

varied center of gravity location on slapdown drops can be significant and should be addressed accordingly.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.26 Clarify the statement on page 2.13.12-6, “The bulk modulus which defines the unloading curve is taken to be the same as the modulus of elasticity,” by providing a sketch depicting both the loading and unloading branches of the stress-strain curves for the cited wood properties presented in Tables 2.13.12-3 and -4.

Also explain, for instance, why the bulk modulus for balsa wood at 10,800 psi, which is interpreted as the modulus of elasticity, is necessarily reported to be smaller than the shear modulus of elasticity at 763,800 psi.

The modulus of elasticity, E, of a given material is known to be always greater than its shear modulus of elasticity, G. It is unclear how these two material constants are being considered in the finite element modeling of the wood in implementing the LS-DYNA Code.

This information is required by staff to determine compliance with 10 CFR 71.17.

2.27 Provide a sketch of finite element discretization details for the critical bolt-beam section between the impact limiter shell and the bolt boss as depicted in Figure 2.13.12-15. Provide also the corresponding stress analysis. On the basis of the stress analysis results, re-evaluate the structural performance of the attachment bolts, as appropriate.

The impact limiter attachment bolt may also fail in combined stress other than just average tensile stress as evaluated in Section 2.13.12.6.1. The attachment bolts have only been evaluated for tensile stress performance for the part not beyond the impact limiter casing. If they are also subject to other stress states, the applicant must re-evaluate the attachment bolts to ensure that the impact limiters will not become dislodged from the package body after the 30-ft drop test.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.28 Provide the impact limiter attachment bolts (LS-DYNA Beam Part IDs 18, 19, 31, and 32) time history results for axial force, axial stress, axial strains, and plastic strains.

Staff was unable to “post-process” the attachment bolt resultant stresses or strains of the submitted 30 Slapdown-10deg-firm\d3plot model that was contained in the Structural\2.13.12 folder.

It was noted that Table 2.13.12-8 tabulates the bolt forces results during the 10 degree slap down condition. However, staff was unable to duplicate the axial forces that are listed in this table during secondary impact of the impact limiter at time step (0.065s) with the highest deceleration value (130g) (reference Figure 2.13.12-32).

Table 2.13.12-8 lists the maximum axial force as 86434 lb at the location of impact surface. Staff noted (via LS-DYNA post processing) that the secondary impact limiter

attachment bolt at an angle 45° from the impact surface (element 237656) had an axial force resultant of 385000 lb, which was the largest axial force.

This information is required by staff to determine compliance with 10 CFR 71.73.

2.29 Justify the use of a beam element with a *MAT_ELASTIC_SPRING_DISCRETE_BEAM material definition to model the impact limiter attachment bolts.

The selection of this type of element is of concern because a “spring” element may not be able to provide a physical representation of bending or shear loading conditions, which may arise during the slap down drop orientation. Staff noted for this type of element, “Failure can [only] occur in either compression or tension based displacement values of CDF and TDF, respectively.” Reference Material Model 74: Elastic Spring for the Discrete Beam of the LS-DYNA theory Manual (page 19.107 of March 2006 version) and *MAT_074 and *SECTION_BEAM in the LS-DYNA Users Manual (version 971) for the capabilities of this element.

This information is required by staff to determine compliance with 10 CFR 71.73.

2.30 Explain how the potential bending and kinking of the attachment bolts next to the cask flange edge can adequately be modeled simply with “spring,” in lieu of beam, elements. See Figure 2.13.12-55, “30-ft Side Drop Soft Maximum Deformation.”

The subject plot on impact limiter deformation suggests that either the impact limiter casing has been torn open or the bolts have suffered fracture failure. The calculated impact limiter failure modes and corresponding design functions as sacrificial material and/or thermal protection of the package must be clearly evaluated in relevant Appendix 2.13.12 sections, including the Section 2.13.12.6.4 “Conclusions.”

This information is required by staff to determine compliance with 10 CFR 71.73(c)(1).

2.31 Justify that an upper limit of -31°C on the temperature of retraction of the elastomer seal is sufficient to maintain containment at -40°C.

Discussions with vendors of elastomer seals have led the staff to find that a temperature of retraction no greater than -35°C is necessary to maintain a static seal at -40°C. The maximum temperature of retraction for the elastomeric seal on a package such as the Model No. HI-STAR 60 package was -35°C.

The references cited by the applicant state that a temperature of 8°C lower than the lowest temperature of operation was acceptable for a static seal. The upper limit of -31°C would still not meet this limit. In addition, the TN-LC seal is a static seal that will be subjected to vibration; therefore the staff does not find a -31°C temperature of retraction acceptable.

This information is required by staff to demonstrate compliance with 10 CFR 71.43(f).

2.32 Justify that the absorbed dose will not significantly affect the elastomer seals used for containment, citing sources in literature. Alternatively, limit the transportation time to a maximum of 6 months.

In general, the staff finds that the radiation damage threshold for elastomer materials is 10^6 rads. However, the radiation threshold for fluoropolymers is significantly lower, approximately 1 x 10^4 rads. The O-ring containment seal will be subjected to the combined effects of heat and radiation damage.

This information is required by staff to demonstrate compliance with 10 CFR 71.43(d).

2.33 Provide an analysis that shear key bearing blocks fabricated from multiple pieces will have the same mechanical properties specified in the structural analysis. Also clarify if impact testing of the martensitic steel used to fabricate the shear key bearing blocks will be conducted to confirm the ductility of the temperature at low temperatures.

The staff is concerned about a potential reduction of mechanical properties that may occur in the shear key bearing block if it is fabricated from multiple pieces. The staff notes that the shear blocks are made of martensitic stainless steel, which is not known for good welding characteristics.

This information is required by staff to demonstrate compliance with 10 CFR 71.33(a)(5)(iii).

2.34 Explain how the following statement in Section 2.13.12.4.1 of the application is relevant to the properties of balsa wood under NCT.

Balsa properties at 150°F are also calculated by factoring the room temperature properties based on the temperature effect contained in Figure 15 of the JPL Technical Report (reproduced as Figure 2.13.12-19). A factor of 0.9 is calculated.

Depending on the packaging contents, the impact limiters are subject to temperatures above 150°F under NCT.

This information is required by staff to determine compliance with 71.73(c)(1).

2.35 Use bounding conservative estimates for the thermal expansion of zirconium and aluminum based alloys used for cladding materials which include the effects of irradiation growth to ensure proper fitting of the assemblies/rods in the TN-LC baskets.

The effects of irradiation growth do not appear in any of the calculations in Section 2.13.10.2. This is of particular concern for M5 fuel claddings and MOX clad fuel. The applicant should cite supporting documentation.

The irradiation growth of zirconium alloys should be based on conservative estimates from more recent documentation on the irradiation growth of zirconium alloys, e.g., NUREG/CR-7024.

This information is required by staff to determine compliance with 10 CFR 71.43(f).

2.36 Use the most conservative values for the thermal conductivities of the fuels loaded into the TN-LC-FA1 basket, or justify that there is a sufficient temperature margin for the loaded PWR assembly before thermal degradation of the cladding occurs.

The reference cited for the thermal conductivities of high burn up fuel do not appear to be the most conservative or comprehensive when compared to guidance prepared for the staff, e.g., FRAPCON-3 (NUREG/CR-6534), "FRAPCON-3 Code Updated with MOX Fuel Properties."

This information is required by staff to determine compliance with 71.73(c)(4).

2.37 Confirm the thermal properties of zirconium alloy using the current version of MATPRO.

The applicant is citing Version 11, Rev.2, of NUREG/CR-0497, “MATPRO” for the thermal properties of the zirconium, published in 1983. More recent versions of MATPRO are available.

This information is required by staff to determine compliance with 71.73(c)(4).

2.38 Justify that the wood used in the impact limiters will have energy-absorbing characteristics bounded by the stress/strain (crush) curves presented in the application and demonstrate how the mechanical proprieties of the woods listed in the application are used to generate the stress/strain (crush curves) used in the structural analysis.

Wood is highly variable material. The staff has reviewed the Wood Handbook (FPL-GTR-113), NASA Technical Report 32-944, NUREG/CR-0322, and the compression testing data presented in the SAR. It is not clear how all of the properties values listed in Tables 2.13.12-3 through 2.13.12-6 of the SAR were derived from these sources, how these values are necessarily conservative when compared to the values listed in the aforementioned documents, or (most importantly) how these values were used to generate the stress/strain (crush) curves in the SAR. The moisture contents and testing temperature of the wood samples used to generate the plots in Figures 2.13.12-16 and Figure 2.13.12-21 are also not stated.

Section 2.12.13.4 of “Balsa” states, “Similar testing [compared to the testing used to establish the proprieties of balsa wood parallel to the grain] were [sic] performed to determine the average crush stress perpendicular to the grain.” There is no description of this testing in the application.

The document, “Compression Tests of Redwood Samples,” is not readily available from the Forest Product Laboratory as cited and should be provided to the staff for verification.

This information is required by staff to determine compliance with 71.73(c)(2).

2.39 Provide a technical data sheet for RILSAN BMN-68.

The staff was unable to find an up-to-date technical data sheet for RILSAN BMN-68.

This information is required by staff to determine compliance with 71.73(c)(4).

2.40 Provide the properties for NITRONIC 60 (S21800) stainless steel which were used in the analyses in the SAR.

The properties of the NITRONIC 60 (S21800) used for the railing are not found in the SAR.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii).

2.41 Justify the statement, “The bulk modulus which defines the unloading curve is taken to be the same as the modulus of elasticity” in Section 2.13.12.4.1 of the SAR.

The bulk modulus and elastic modulus (Young's modulus) are fundamentally different material parameters.

This information is required to demonstrate compliance with 10 CFR and 71.33(a)(5)(iii).

2.42 Remove all references to the use of digital image analysis as a methodology to demonstrate the areal density of boron in the neutron absorber materials.

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

This information is required by staff to determine compliance with 10 CFR 71.31(c).

2.43 Provide the mechanical properties of 6063-O aluminum alloy in the application. Alternatively, state that the neutron shield holders were not incorporated into the FEA model used for the drop analysis.

The mechanical properties of the neutron shield holders on the TN-LC should be stated in the SAR if they influence the drop analysis.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii) and 71.73(c)(1).

2.44 Clarify if the fuel cladding for the various fuel elements is the only material to take structural credit during NCT and HAC. Explicitly clarify the type of fuel cladding materials used for the various fuel elements.

It is unclear if the proprieties of the composite fissile-aluminum material in the non-power reactors are credited for maintaining the integrity of the fuel elements during NCT and HAC. The SAR uses the properties of annealed 6061-O for the proprieties of the

cladding material, but, in some instances, e.g., TRIGA fuel, the weaker 1100-O alloy may have been used as cladding.

This information is required by the staff to determine compliance with 10 CFR 71.73(c)(1).

Chapter 3 Thermal Evaluation

3.1 Provide the basis for the reduced long-term peak cladding temperature limit for aluminum-clad fuel. Demonstrate that the long term limit is sufficient to assure long-term stability of this type of cladding.

The thermal design criteria for the Model No. TN-LC package (see p. 3-1, Section 2.1) appropriately references ISG-11, Rev. 3, for fuel cladding temperature limits for LWR fuel assemblies and 25-pin packages (i.e., maximum cladding temperature long-term limit of 752ºF (400ºC), short-term limit of 1058ºF (570ºC), plus limitations on thermal cycling in vacuum drying).

For NRUX/MTR/TRIGA fuel, the maximum cladding temperature limit is reduced to 400ºF (204ºC) for long-term conditions, and the short-term limit is defined as 1140ºF (613ºC) – the melting point of 1100ºF and 6063 aluminum alloys used as cladding in this type of fuel. No limitations are noted for thermal cycling. The basis for the limits for the research reactor fuels listed above has not been provided, and the staff requires a basis in order to make a finding on the adequacy of the temperature limits provided.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(v).

3.2 Justify the use of a linear extrapolation of burnup effects for fission gas inventory for package overpressure calculations and the determination of MNOP. Confirm that the B&W 15 x 15 fuel assembly remains the bounding fuel assembly for higher burnups.

Fission gas inventory predictions for cask overpressure calculations and to determine maximum normal operating pressure (MNOP) rely on linear extrapolation of fuel burnup effects from 55 GWd/MTU to 70 GWd/MTU, using data from Ref. 23[1]. Given the age of the reference cited, the staff believes that modern analysis tools, such as FRAPCON and modern isotopic depletion codes, could be used to effectively predict what the fission gas inventory could be without extrapolations from data for fuels with lower burnups. In addition, a linear extrapolation is believed to be potentially non-conservative in this application.

This information is required by staff to determine compliance with 10 CFR 71.33(b)(5).

3.3 Provide justification for the assumption that ‘cold’ dimensions are bounding for both NCT and HAC.

Dimensions of components and gaps in the ANSYS models provided for thermal

[1] Plannel, et al., Topical Report, Extended Fuel Burnup Demonstration Program. Transport Considerations for Transnuclear Casks, Project 1014, DOE/ET 34014-11, TN-E-4226, Transnuclear Inc., 1983.

analyses are based on 'cold' dimensions, as documented in the licensing mechanical drawings included in Chapter 1. This is presented in the applicaiton as a conservative assumption, as it does not account for reduction of gaps due to thermal expansion. However, given the dissimilar materials in the body of this package, and the wide range of geometries of the baskets it may contain, it is possible that thermal expansion could increase some gap widths in some geometries.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(v).

3.4 Justify the use of a linear charring rate of 55 mm/hr for exposed wood in Section 3.4.2, "Fire Test Conditions."

The linear charring rate is known to vary with density, moisture content, heat flux, hardwood-softwood classification, and species. However, the variation in these parameters is not clearly addressed in the SAR. The SAR should address the possibility of exposed balsa wood, in addition to redwood. Because balsa wood is classified as a hardwood with a very low density, the linear charring rate may exceed that proposed for redwood.

This information is required by staff to determine compliance with 10 CFR 71.73(c)(4).

3.5 Verify that the calculated specific heat values align with the referenced guidelines in Section 3.2.1, "Wood Properties."

The wood property specific heat values tabulated in Section 3.2.1 appear to be incorrect due to a unit discrepancy, Celsius versus Fahrenheit. Correct the specific heat values, corresponding density values, and any inconsistencies that result in the SAR as a result of these corrections. Alternatively, correct the "Temperature (ºC)" heading to read "Temperature (ºF)" and verify that these values were used appropriately in the analysis.

This information is required by staff to determine compliance with 10 CFR 71.43(g), 71.71 and 71.73.

Questions related to ANSYS analysis files

3.6 Provide an analysis that demonstrates the package's correct limiting configurations for both NCT and HAC.

Based on confirmatory analyses done by the staff, the limiting configurations for the Model No. TN-LC package under NCT and HAC have been incorrectly identified in the application. For the HAC condition, the staff believes that the identification of the fuel pin basket as the limiting configuration is incorrect. The limiting basket configuration under NCT is assumed to also be limiting for HAC, and therefore, the subsequent analyses completed for the limiting configuration in HAC are incorrect. In addition, the peak fuel cladding temperature for the TN-LC under HAC has been shown to occur with the package in the ISO container, due to the thermal inertia of the package in the fire transient, as demonstrated in NUREG/CR-6894, Rev. 1. This behavior is in contrast to the assumption in the application that the limiting configuration for the TN-LC package

under HAC is without the ISO container. While the temperatures presented in the application's analysis are generally conservative, the staff believes that the use of a "de-coupled" analysis approach for this package has led to the misidentification of the limiting configurations. This should be corrected in the current application, as well as for any future submittals.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(v).

3.7 Provide the text log and/or input file(s) used to generate the TN-LC-ISO-NCT-100F.db (and similar) files.

ANSYS database files do not allow the staff to complete even an audit review of the analysis approach used to generate the model geometry or meshing parameters. As stated in ISG 21 "Computational Modeling" the staff prefers text input files to database files, as it expedites the review of the analysis models in question.

This information is needed to ensure compliance with 10 CFR 71.33 (a)(5)(v).

3.8 Correct the error in specifying the impact limiter attachment block temperature which was found twice in analysis input file TN-LC-ISO-NCT-100F.inp, at approximately line 224. Update temperatures reported in the SAR as needed, and correct any other files that may have been impacted by this error.

While reviewing the ANSYS input files provided to the staff, it was discovered that in file TN-LC-ISO-NCT-100F.inp, at approximately line 224, the elements and nodes associated with material number 3011 were selected in order to determine the maximum temperature of that component, which is defined in file Mat_TN_LC.inp as the trunnion block. In the file LC-ISO-NCT-100F.inp, the maximum temperature is incorrectly assigned to the parameter for the impact limiter attachment block (ILAttach_), duplicating the previous assignment of this parameter to the maximum temperature for the elements and nodes associated with material 3010 (from line 222 of this file). This leads to an incorrect reporting of the maximum temperature for the impact limiter attachment block and no reporting for the maximum temperature of the trunnion block.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(v).

3.9 Correct the comment title of the file HC-ROOF.MAC to address Convection Coefficients for the TN-LC package.

The ANSYS macro file HC_ROOF.MAC, at line 6 states, "Total Convection Coefficients for HSM Roof." This should be corrected to refer to the TN-LC ISO container roof.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(v).

Chapter 5 Shielding Evaluation

5.1 Provide a detailed description of the benchmark experiments and a tabulated comparison of important parameters (such as isotopes used in the SAS2H source term calculation analysis) between these experiments to those used for the Model No. TN-LC

package shielding design.

Section 5.6.4.2.5.1 of the application states that "*Several references in published literature [4, 5, 6] were used to generate ratios for isotopes important to shielding in the form of the calculated value from SAS2H (C) over the measured value of the isotope (M). This is herein referred to as the C/M ratio. Activities of each isotope important to shielding were scaled by the C/M ratio. Radiological sources are adjusted based on the modified activities. Using response functions, the effect on the dose rates can be quantified.*" However, there is no description of the applicability of such referenced benchmark experiments to the Model No. TN-LC package.

This information is required by staff to determine compliance with 10 CFR 71.47.

5.2 Define the neutron response function and how it affects the neutron dose calculations.

Section 5.6.1.2.2 of the application states that the neutron source was comprised of both spontaneous fission and (α,n) reactions with the aluminum in the fuel matrix. Like the gamma calculation, a neutron response function was also generated for NCT. No details are provided on this neutron response function.

This information is required by staff to determine compliance with 10 CFR 71.47

5.3 Explain why the NRX fuel with heavy water source was the bounding source used in the dose rate calculations.

Section 5.6.2.2.2 states that the NRX fuel with heavy water source was the bounding source used in the dose rate calculations for the NRU and NRX fuels. However, there is no clear explanation on how the heavy water can give the bounding source for the NRX fuel.

This information is required by staff to determine compliance with 10 CFR 71.47

5.4 Clarify the uncertainty evaluation for the total dose rate.

Section 5.6.4.2.1 of the application states that adjusting radiological sources using the C/M ratios obtained from the references does not result in a dose rate increase more than 18%. However, the application reports dose rates as high as 9.12 mrem/hr at 2 m from the side of the transport vehicle. If this 18% in uncertainty is applied to such dose rates, regulatory limits could be exceeded.

This information is required by staff to determine compliance with 10 CFR 71.47.

5.5 Clarify why the 9 EPR MOX fuel yields a higher dose rate (9.12 mrem/hr) than the 25 EPR MOX fuel (9.06 mrem/hr), at 2 m from the vehicle.

Section 5.6.2.6.1 states that, at 2 m from the vehicle, the maximum dose rate of any fuel type is calculated for 25 EPR MOX rods to be 9.06 mrem/hr while it is 9.12 mrem/hr for the 9 EPR MOX rods.

This information is required by staff to determine compliance with 10 CFR 71.47.

Chapter 6 Criticality Evaluation

6.1 Explain the basis for assuming an intact fuel when damage to the cladding exceeds a pinhole leak or hairline crack.

Page 1.4.2-2 states that, for NRU/NRX fuel, "Damaged fuel assemblies with cladding damage in excess of pin hole leaks or hairline cracks are authorized. The extent of the damage is limited such that the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod." Page 1.4.3-4 states that, for MTR fuel, "MTR fuel elements with damaged cladding are authorized, provided the total surface area of the damage does not exceed 5% of the total surface area of the damaged element." It is unclear what the basis is for using 5% of the total surface area of the rod or fuel element as a determination of the "allowable" damage for a rod or fuel element without canning the rod(s)/fuel element(s) in a damaged fuel container when the damage exceeds a pinhole leak or hairline crack.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.2 Explain the basis for the statement on page 6.10.3-8 of the application that a configuration with the graphite reflectors of the TRIGA assemblies touching the end of the basket "can be considered to be a bounding geometry for a scenario under HAC in which the fuel is damaged."

It is unclear how not modeling the end fittings of the TRIGA fuel assemblies and allowing the graphite reflectors of the TRIGA assemblies to touch the end of the basket is a "bounding geometry for a scenario under HAC in which the fuel is damaged." The damaged fuel condition represents a potential loss of geometry control under transportation conditions, and fuel geometries other than that described for the TRIGA fuel have the potential to be much more reactive.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.3 Explain the basis for the statement on page 6.10.4-9: "For single shear and double shear analyses, it is shown that the BW 15 x 15 B11 fuel assembly results in the most reactive configuration. However, the WE 14 x 14 Std/LOPAR/ ZCA/ZCB fuel assembly remains the most reactive for damaged fuel cases, as shown in Table 6.10.4-12."

It is unclear what fuel the case ID's correspond to in the application. Also Table 6.10.4-13, "Single-Ended Shear Analysis – PWR Fuels," and Table 6.10.4-14, "Double-Ended Shear Analysis – PWR Fuels," (and elsewhere) refer to case ID's (e.g., P_F001) that have not been defined in the application (i.e., it is not clear what PWR fuel is being modeled). Finally, the case ID's in the application do not correspond to the case ID's of sample inputs and outputs that were provided on a disk to the staff.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.4 Revise the damaged fuel configuration for the TN-LC-1FA basket to consider loss of rods from the lattice after modeling pitch expansion.

The TN-LC-1FA basket criticality analysis considers only single- and double-ended rod shear and pitch expansion in the damaged fuel criticality analysis. The application should be revised to either demonstrate that these configurations are more reactive for the TN-LC-1FA. Alternatively, revise the evaluation to also consider loss of rods from the lattice.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.5 Revise the application to provide the most limiting fuel assembly parameters for the spent fuel assemblies to be transported in the TN-LC package (e.g., maximum fuel pellet outer diameter, minimum clad thickness, maximum pitch, etc.).

The application includes only nominal values for the spent fuel designs considered in the criticality safety analysis. The application should be revised to identify bounding fuel assembly parameters that will be listed in the Certificate of Compliance to identify the allowable contents of the package.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.6 Revise Sections 6.10.1.8.2, 6.10.2.8.2, 6.10.3.8.2, and 6.10.4.8.2 of the application to provide the results of the normality tests to ensure the applicability of the statistical approach for the selected benchmarking data set.

For USLSTATS analysis, the results of the normality tests, whether those from USLSTATS or other statistical packages, need to be provided to ensure applicability of statistical approach for the selected data set. Also, the applicant states on page 6.10.4-18 that USL functions are developed for five USL parameters, but then states that “all four USLSTATS6 runs meet the normality test.” Explain the apparent discrepancy.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.7 Revise the USL for the allowed contents of the TN-LC-MTR and TN-LC-TRIGA baskets to include a bias uncertainty derived from nonparametric methods.

The number of benchmark data is insufficient for establishing a normal distribution: only 17 benchmarks were directly applicable for the contents of the TN-LC-MTR basket (Note: Applicant stated 17, but text in Section 6.10.1.8.1, “Applicability of Benchmark Experiments,” only supports 16 as being directly applicable). The applicant only used 21 benchmarks for the contents of the TN-LC-TRIGA basket, for which only two were directly applicable. Less than 25 benchmarks would normally require the applicant to

use a nonparametric margin to calculate the USL. Revise the application accordingly, or show the number of experiments used to calculate the USL fits a normal distribution.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

6.8 Revise the application to clearly state whether TRIGA fuel is considered damaged or not for NCT and HAC.

The application states that the TRIGA fuel is modeled as damaged for both NCT and HAC. For example, page 6-2 states: "In the TN-LC-MTR, TN-LC-NRUX, and TN-LC-TRIGA basket analyses, fuel damage is conservatively modeled in both the NCT and HAC models for convenience." However, page 6.10.3-5 states: "Since the fuel assemblies are held in fairly small compartments, without much room to move during an accident, the baskets and fuel are assumed to be undamaged in all calculations performed for this analysis under both normal conditions of transport (NCT) and hypothetical accident conditions (HAC)." Page 6.10.3-7 states: "Structural analysis of the cask and basket in Chapter 2 has demonstrated that they maintain their structural integrity after an accident. Therefore, these calculations assume that both the basket and the fuel are completely undamaged and intact." Revise the application to clear up this discrepancy.

This information is required by staff to determine compliance with 10 CFR 71.55 and 71.59.

Chapter 7 Operating Procedures

7.1 Provide procedures for (i) loading a PWR fuel assembly with Poison Rod Assemblies (PRAs) prior to loading the fuel assembly into the TN-LC-1FA basket and (ii) ensuring the PRAs are loaded as prescribed by the requirements in Table 6.10.4-26 and Figures 6.10.4-12, 6.10.4-13, and 6.10.4-15 of the application.

Chapter 7 of the application provides operating procedures for all contents. However, staff noted that the procedures do not include loading procedures for the PRAs that are required to ensure subcriticality of a PWR fuel assembly for NCT and HAC. The procedures also do not specify a verification to determine the required number of PRAs are installed as stated in Table 6.10.4-26 or that they are installed in the proper locations as identified in Figures 6.10.4-12, 6.10.4-13, and 6.10.4-15 of the application. Because this is a new content and new configuration, procedures for loading and verifying the new content are needed.

This information is required by staff to determine compliance with 10 CFR 71.55, 71.59, and 71.87.

Chapter 8 Acceptance Tests and Maintenance Program

8.1 Provide acceptance testing or mixing methods for Resin F in the application and CoC by reference.

Acceptance tests and preparation procedures for preparation of the neutron absorbers VYAL B, but not Resin F, are included in the application and CoC by reference.

This information is required by staff to demonstrate compliance with 10 CFR 71.31(c).

8.2 Specify that accessible welds will be inspected prior to package loading use for signs of degradation and brought into compliance with the licensing drawings before shipment of a loaded package.

The maintenance program requires that threaded fasteners will be examined prior to the use of the TN-LC, and should include accessible welds as well.

This information is required by staff to determine compliance with 10 CFR 71.33(a)(5)(iii).

8.3 Describe the determination process for the acceptability of wood performance if the impact limiters are damaged and verify that the impact limiter will pass leak testing using the same acceptance criteria as the original leak tests.

The application states that if the impact limiter is damaged, that the internal wood will be examined for degradation. It is not clear how the wood will be examined to ensure that it has maintained its performance. In addition, it is not clear how the presence of free water (as determined by a humidity test) verifies that a repaired impact limiter adequately meets its original function.

This information is required by staff to determine compliance with 71.73(c)(2).

8.4 Establish acceptance tests in Chapter 8 of the application to verify the density, moisture content, and mechanical proprieties of the impact limiter wood.

The acceptance tests should include mechanical testing of the impact limiting wood to ensure that the wood will have the same impact characteristics stress/strain as the wood used in the impact analyses.

This information is required by staff to determine compliance with 71.73(c)(2).