May 8, 2008

Robert Grubb, Senior Vice President - Engineering
Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. TN-40 PACKAGE

Dear Mr. Grubb:

By application dated June 29, 2007, Transnuclear submitted an application to the U.S. Nuclear Regulatory Commission for approval of the TN-40 cask design as a transportation package under Docket No. 71-9313. In my letter to you dated August 24, 2007, I acknowledged receipt of your application and provided a proposed schedule for our review.

In connection with the staff's review, we need the information identified in the enclosure to this letter. We request that you provide this information by July 28, 2008. Inform us at your earliest convenience, but no later than July 14, 2008, if you are not able to provide the information by that date. To assist us in re-scheduling your review, you should include a new proposed submittal date and the reasons for the delay.

Please reference Docket No. 71-9313 and TAC No. L24106 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contact at 301-492-3338.

Sincerely,

/RA/
Meraj Rahimi, Senior Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9313
TAC No. L24106

Enclosure: Request for Additional Information

cc w/encl: R. Boyle, Department of Transportation
J. M. Shuler, Department of Energy

May 8, 2008

Robert Grubb, Senior Vice President - Engineering
Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. TN-40 PACKAGE

Dear Mr. Grubb:

By application dated June 29, 2007, Transnuclear submitted an application to the U.S. Nuclear Regulatory Commission for approval of the TN-40 cask design as a transportation package under Docket No. 71-9313. In my letter to you dated August 24, 2007, I acknowledged receipt of your application and provided a proposed schedule for our review.

In connection with the staff's review, we need the information identified in the enclosure to this letter. We request that you provide this information by July 28, 2008. Inform us at your earliest convenience, but no later than July 14, 2008, if you are not able to provide the information by that date. To assist us in re-scheduling your review, you should include a new proposed submittal date and the reasons for the delay.

Please reference Docket No. 71-9313 and TAC No. L24106 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contact at 301-492-3338.

Sincerely,

/RA/
Meraj Rahimi, Senior Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9313
TAC No. L24106

Enclosure: Request for Additional Information
cc w/encl: R. Boyle, Department of Transportation
J. M. Shuler, Department of Energy

DISTRIBUTION:
PUBLIC Docket File 71-9305 G:\SFPO\Rahimi\71-9313\REV0RAI.doc

OFC:	SFST	E	SFST	E	SFST	E	SFST	E	SFST	E	SFST	E
NAME:	MRahimi		DTang		REinziger		JIreland		MPanicker		ASotomayor	
DATE:	3/9/08		3/9/08		3/ 9/08		4/15/08		3/14/08		3/8/08	
OFC:	SFST	E	SFST	N	SFST	N	SFST	N	SFST	E	SFST	N
NAME:	ZLi		MDeBose		CRegan		MWaters		LCampbell		RNelson	
DATE:	4/21/08		3/25/08		5/8/08		5/8/08		4/ 25 /08		3/9/08	

C = COVER E = COVER & ENCLOSURE N = NO COPY OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

Docket No. 71-9313
Model No. TN-40
Certificate of Compliance No. 9313

This document, titled Request for Additional Information (RAI) contains a compilation of additional information requirements identified to date by the U.S. Nuclear Regulatory Commission (NRC) staff, during its review of Transnuclear's application for approval of the TN-40 transportation package under 10 CFR Part 71.

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

1.0 GENERAL INFORMATION

1-1 Provide the Criticality Safety Index (CSI) and Transport Index (TI) values for these packages.

On page 1-1, the Safety Analysis Report (SAR) states: “The Transport Index for nuclear criticality control for the TN-40 cask is determined to be zero (0) in accordance with 10 CFR 71.59 [1]. See Chapter 6 for details of this determination.” This statement is incorrect. It confused the Transport Index with Criticality Safety Index.

This information is needed pursuant to the requirements of 10 CFR 71.59.

1-2 Define “intact assemblies” to specify what kind of assembly hardware defects can be allowed and still consider the assembly as “intact.”

The only definition of “Intact Fuel” is found on page 1-1. This definition is adequate for defining intact rods but is not considered complete. Under this definition, assemblies with damaged hardware could be stored and shipped in this cask.

This information is needed to determine compliance with 10 CFR 71.55 (d)(1) and (2).

1-3 Clarify the statement on page 1-5, “[A]n aluminum spacer is placed on the cask lid prior to mounting the top impact limiter. The purpose of the aluminum spacer is to provide a smooth contact surface between the lid and the top impact limiter.” Also, revise the components list of Figure 1-1 of the application if the spacer is not used.

Drawings 10421-71-2 and -7 depict general arrangement and design of the 1.09-inch thick aluminum space, respectively. However, as called out in Drawing 10421-71-40, the Item 54 (aluminum spacer) is noted as “NOT USED” in the Parts List of Drawing 10421-71-41. It’s unclear whether the aluminum spacer is used for attaching the top impact limiter to the overpack. Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

1-4 Revise the statement on page 1-5, “[T]he upper trunnions… are designed to meet the requirements of NUREG-0612 for non-redundant lifting fixture. This is accomplished by evaluating the trunnions to the stress design factors required by ANSI N14.6. See

Section 2.11 for testing alternatives to ANSI N14.6," to require also the trunnion load test be three (3) times the design lift load.

Section 2.11, "ASME Code and NUREG-0612 Alternatives," of the application proposes a trunnion load test of 1.5 times the design lift load by invoking the May 11, 1995, NRC safety evaluation for accepting this test load for Part 72 storage. The staff notes, however, that the safety evaluation was performed on the deployed TN-40 storage system as part of the Prairie Island Nuclear Generating Plant site-specific Part 72 storage facility. Since a general license is requested, full compliance with industry consensus codes and standards is expected of the package.

This information must be provided in the application in accordance with the requirements of 10 CFR 71.7(a) and 71.35(a).

1-5 State in the SAR and CoC that in all cases where the cask has been loaded for storage that records have been maintained that indicate there has been no leakage of air into the canister.

Some TN-40 casks may have been loaded at a time before transportation was contemplated. The physical condition of the fuel was known at the time of loading for storage, and stored under atmospheric and temperature conditions indicated in ISG-11 (Cladding Considerations for the Transportation and Storage of Spent Fuel) that would assure no additional cladding breaches. If the records indicate air leakage into the container, the lack of cladding breaches should be confirmed using the methodology given in ISG-22 (Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere…)

This information is needed to determine compliance with 10 CFR 71.33(b)(3).

1-6 Correct the errors in the package content table in page 1-7 for the uranium load per assembly.

On page 1-7, the applicant defines in the inline table the contents that are permitted to load in the TN-40 packages. In the table, the SAR defines that the maximum MTU/Assembly as 360, 370, or 410. There is an error for the uranium loading/assembly unit in the table.

This information is needed pursuant to the requirements of 10 CFR 71.33(a)(6).

1-7 Provide copies of references for:

- Maximum MTU/Assembly (+2/-4% from staff reference) - SAR p. 1-7
- Rod pitch for all assemblies (staff calculate 0.564" not 0.556"), and pellet OD for Exxon Toprod (staff-0.344", SAR 0.3505") - Table 6-2
- Guide tube IDs and ODs (guide tube ID staff has 0.471", SAR 0.507" for example) – Table 6-2
- Cladding thickness for WE Std (staff has 0.025", SAR 0.0243") – Table 6-2

Ensure that values presented in the SAR are substantiated. If not, then corrections to values in Tables 2.10.7-1 and 5-3 may be necessary.

Assembly and rod specifications in the tables were reviewed by the staff. While in most cases there was agreement, in some cases the staff identified discrepancies with the staff's reference values (multiple sources).

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

1-8.1 Include the burnup credit loading criteria, in terms of minimum required burnup and maximum initial enrichment, in the Fuel Qualification Table in Table 1-2 and Page 1-7.

The maximum burnup and initial enrichment presented in Table 1-2 and Page 1-7 are based on shielding and heat load constraints. With the burnup credit as part of the design for criticality safety control, there is an additional constraint in terms of minimum required bunrnup as a function of initial enrichment. This additional constraint needs to be superimposed on the shielding and heat load design parameters for fuel qualification.

This information is needed pursuant to the requirements of 10 CFR Part 71.55 (b).

2.0 STRUCTURAL

Unless otherwise indicated, the following RAIs are needed to determine compliance with 10 CFR 71.35(a).

2-1 Clarify the statement on page 2-2, "[T]esting of the impact limiters is planned to confirm the analytical results from the Transnuclear in-house computer program (ADOC). The test specification is included in Appendix 2.10.9."

The statement is outdated. It should properly be revised to reflect the impact limiter testing described in Appendix 2.10.9 of the application.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

Section 2.1.2 Design Criteria

2-2 Justify the use of ASME Code, Section III, Subsection NB, as opposed to Subsection NG, as the basket weld stress acceptance criteria (page 2-4, 2-5, and 2-37).

NUREG-1617, "Standard Review plan for Transportation Packages for Spent Nuclear Fuel," provides staff guidance on reviewing welding criteria per Subsection NG, in recognizing that a variety of weld categories and associated examination procedures are acceptable for basket fabrication. The application notes that special inspections and tests were developed for welding, by fusion, the 13 gage stainless steel plates to the 1-3/8-inch diameter steel plugs to form fuel compartments. Drawing 10421-71-9 further notes that a 100% visual inspection is performed on these welds. The staff notes that Subsection NG, Table NG-3352-1, lists qualify factors for which weld stress intensity limits may be de-rated commensurate with the examination procedures. The application is unclear as to whether the weld quality factors are appropriately considered for evaluating the fuel basket.

2-3 Provide a discussion or calculation which shows that the various aluminum alloy canister components will meet their life-time design requirements when operating at temperatures where the material is subjected to creep-induced deformation. Provide or cite references for the long-term creep properties of any aluminum alloy canister component(s) which exceed the stress or temperature limits of the ASME Code, Section II, Part D. Show that these properties are adequate for meeting the component's design-life performance requirements during the specified operating condition(s).

This information is required for compliance with 10 CFR 71.33.

2-4 Revise, as appropriate, the statement on page 2-10, "[T]he two front trunnions…are evaluated to the requirements of ANSI N14.6 with alternatives listed in Section 2.11," to recognize also NUREG-0612 for non-redundant lifting.

Page 1-5 of the application identifies NUREG-0612 as the lead requirement to which the upper trunnions are designed as non-redundant lifting fixture. Both NUREG-0612 and its subordinate ANSI N14.6 should be cited for the upper trunnions.

This information should be provided with the application in accordance the 10 CFR 71.7(a) and 71.35(a) requirements.

Section 2.5.1 Trunnion Analysis

2-5 Provide calculations for the reported margins of safety of 0.06 and 0.28, for the front trunnions, against the yield and ultimate strengths, respectively. Furthermore, revise the application to add an excessive load evaluation for the trunnions.

It's unclear how the margins of safety were calculated in page 2-11. An evaluation must be performed, per 10 CFR 71.45(a), to demonstrate that failure of any lifting device that is part of the packaging would not impair its ability to meet other requirements of Subpart C of 10 CFR Part 71.

Section 2.6.1 Heat

2-6 Provide an evaluation of differential thermal expansion effects on the structural performance for various important-to-safety components of the cask.

Section 2.5.6.1 of the Standard Review Plan, NUREG-1609, provides that any differential thermal expansions and possible geometric interferences are considered in evaluating the cask for meeting the 10 CFR 71.71(c)(1) heat test requirements.

Section 2.7 Hypothetical Accident Conditions

2-7 Provide an upfront summary, under "Overview," to recognize (1) the use of the baseline decelerations as predicted in Section 2.10.8, by ADOC program, for analyzing the package structural performance, (2) the differences between the baseline decelerations and the tested decelerations of Section 2.10.9 from the 1/3-scale impact limiter testing, and (3) the calculated margins, as appropriate, for demonstrating sufficient margin when the baseline decelerations are less severe than the tested but are considered in structural analysis.

Table 2.10.9-1 compares rigid body decelerations measured by drop tests with those calculated by ADOC. The predicted decelerations are shown less than the tested ones for the end, side, and C.G.-over-corner drop tests. Section 2.10.9.7 presents evaluations to demonstrate that all analytical results based on the ADOC program predictions remain applicable. However, these evaluations are buried deep in the application as secondary reference and not readily recognizable. They should be discussed upfront to facilitate safety evaluations.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-8 Provide sketches to delineate the standard or selected representative stress report locations for the cask body flange and inner shell (pages 2-12 and 2-22 on "Reporting Method for Cask Body Stresses).

It's unclear where the stated stress report locations are. The report locations for the containment boundary should include those that are most susceptible to stress failure to facilitate safety review.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

Section 2.7.1 30-Foot Free Drop

2-9 Justify the selection of the 20^{o} drop angle for evaluating package free drop accidents for which maximum damage is expected.

10 CFR 71.73(c)(1) requires free-drop tests be conducted in a position for which maximum damage is expected.

This information should be provided with the application in accordance with the 10 CFR 71.73(c)(1) requirements.

Section 2.7.4.3 Immersion – Deep Water Immersion Test (Water Pressure of 290 psig)

2-10 Discuss how the "shrink-fit" fabrication stress is considered for the large displacement buckling analysis of the cask inner shell, and how the material and geometric nonlinearities are considered for the buckling evaluation, including meeting the minimum factor of safety of 1.34 of the ASME Code Case N-284 method commonly considered for metal containment shells.

As reported in Section 2.10.1, there is a shrink-fit interface pressure of 529.1 psi between the inner shell and the gamma shield shell. This pressure together with material and geometrical nonlinearities inherent to thin shell constructions should be considered for the buckling analysis of the inner shell.

This information should be provided to meet the requirements of 10 CFR 71.61.

Section 2.7.6 Summary of HAC Cask Body Structural Analysis

2-11 Clarify, as appropriate, the statement on page 2-30, "[F]rom the analysis results presented in Tables 2-18 and 2-19,…the containment functions of the cask and the support functions of the basket will be maintained."

The stress results in Tables 2-18 and 2-19 are for cask body components only. The cask body side-drop resistance limit is reported to be 85g. As described in Section 2.10.5 of the application, however, the basket is evaluated only up to a side impact of 75 g. Reference to the basket support functions in the context of the statement can be misleading and should be removed for clarity.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.(c)(1) requirements.

2-12 Justify the use of 0.8 S_m as the shear stress limit for the basket design (Table 2-5, "Basket Stress Limit).

The 0.8 S_m stress limit applies to shear at the periphery of a solid circular section in torsion per the ASME Code, NB-3227.2(b) or NG-3327.2(b) requirement. It's unclear how this stress category is considered for the basket components, which are not subject to torsion.

Its basis should be provided in the application in accordance with the requirements of 10 CFR 71.71(c)(1).

2-13 Provide sketches in the application to illustrate the locations and corresponding cross sections that have been determined to have the highest calculated stresses for individual components and load combinations for the normal conditions of transport and hypothetical accident conditions (Tables 2-14 and 2-19).

Graphic aid to the stress summaries is needed for facilitating the safety evaluation. This information should be provided to meet the intent of the Regulatory Guide 7.9, Sections 2.6.1.4 and 2.7.8, guidance on stress comparison in evaluating structural performance of the package.

2-14 Provide sketches to illustrate modeling details for the interface between the body flange and gamma shield comprised of a shim plate and a 1/2-inch deep grove weld (Table -19, Cases A11 and A12).

Considering lack of tensile stress load path accorded only by weld, provide an evaluation to demonstrate that the inner shell containment boundary next to the gamma shield is not to breach under the slap-down prying action as depicted in Figure 2.10.8-16. The shim plate as depicted in Detail E, Drawing 10421-71-3, is capable of carrying compressive force only. To evaluate effects of the normal force, R, and prying force, μR, as depicted in Figure 2.10.8, on the stress performance of this transition zone, the finite element model must consider the discontinuity associated with the shim placement.

Appendix 2.10.4 Fracture Toughness Evaluation of the TN-40 Cask

2-15 Provide temperature-dependent fracture property data for the filler metal and the heat affected zone (HAZ) in the temperature range of HAC to support the claim that the weld cracks in the base metal of carbon steel (SA-266, Class 4) are indeed stable during HAC in transport. Provide data to show that not only this statement is true but the HAZ is also tougher than the base metal.

The application provides fracture toughness data of the base metal (SA-266, Class 4) and uses it to show that potential weld cracks in the 10 critical locations remain stable during transportation. It is known that mechanical properties of filler material as well as HAZ, in general, can be dramatically different from that of the base metal. And, it is clear that data of the former two, not the latter, should be used as the cracks are located within the welds, not in the base metal. The application simply stated that the filler metal is tougher than the base but gives no data to substantiate the claim (Sec. 2.10.4.6) that the toughness of the weld material is higher than the base material.

This information is needed to determine compliance with 10 CFR Part 71.73(c)(1,2,3) and 10 CFR 71.33

Appendix 2.10.5 Structural Analysis of the TN-40 Basket

2-16 Justify that nodal coupling is conservatively implemented such that no composite action is developed to result in unrealistically high flexural rigidity for basket wall panels.

The applicant has stated on page 2.10.5-4 that “[T]he nodes between the steel tubes including the intermediate aluminum plates are coupled together in the out-of-plane direction so that they will bend in unison…to simulate through the thickness support provided by Boral plates.” The fuel tubes are structurally connected by fusion weld to steel plugs. It’s unclear whether the out-of-plane coupling also prevents nodes from relative displacements in the local x and y directions for assuring that the interface is modeled correctly.

2-17 Provide sketches to explain how the nodal coupling is implemented, in recognizing that the aluminum and Boral plate assemblies are essentially hung loose on the steel plugs to accommodate differential thermal expansion effects.

The applicant has stated on page 2.10.5-4 that “[T]he aluminum plates are coupled together at their intersection.” It’s unclear how the nodal coupling is implemented in the finite element model to reflect this physical condition.

2-18 Correct editorial errors on page 2.10.5-4 on material grade description of the basket support rails.

The basket rails are made of grade 6061-T6, or T651, aluminum, in lieu of SA-240, Grade 304 stainless steel.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-19 Describe how initial imperfections of the fuel tubes are implemented in the ANSYS nonlinear buckling analysis (page 2.10.5-9 on the fuel basket buckling analysis).

The through-the-thickness lateral support associated with intermediate aluminum plates may not be counted on in preventing the13-gage stainless steel wall panel from buckling away from the aluminum plates. It’s plausible that the compartment walls need not bend in unison with the aluminum plates for the 0^{o} and 90^{o} drop orientations of which maximum damage in terms of fuel compartment buckling is expected.

2-20 Include a footnote in the “Justification” column of Section 2.11 of the application similar to the one in Table 2.10.5-1 as part of the justification for using the code alternatives for establishing aluminum material properties at elevated temperatures and correct typographical errors in listing SA-240, Grade 304 stainless steel as the basket rail material.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-21 Include in the “Justification” column of Section 2.11 of the application, the footnote in Table 2.10.5-2, which compares allowable stress, S, as a code alternative for stress intensity limit, S_m, used in the design by analysis method for the basket structure.

The staff notes that the basket stress results for normal conditions of transport are evaluated for meeting only the stress intensity evaluation criteria per the ASME Subsection NG. It's unclear how the allowable stress criteria are implemented.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-22 Revise, as appropriate, the Table 2.10.5-6 basket fusion weld shear stress evaluation for NCT drops, per the quality factors commensurate with the implemented weld inspection procedures.

As commented previously, the ASME Code, Section III, Subsection NG, stress criteria, which recognize weld quality factors, should be considered for the basket fusion weld evaluation.

2-23 Justify that factors of safety in Table 2.10.5-11 are adequate for the 30^o, 45^o, and 60^o drop orientations at 1.18, 1.23, and 1.24, respectively. Revise the evaluation, as appropriate, by implementing also initial fuel tube imperfections in the analysis.

NUREG/CR-6322, "Buckling Analysis of Spent Fuel Basket," provides acceptable Level D factors of safety ranging from 1.42 to 2.12, which are embedded in the ASME Code for the stainless steel basket. The reported factors of safety do not satisfy the staff review guidelines.

Section 2.10.7 Structural Evaluation of the Fuel Rod Cladding Under Accident Impact

2-24 Clarify on page 2.10.7-1 the statement, "[T]he high burn up fuel assemblies experience higher temperatures and have lower cladding material properties."

From structural analysis perspective, the only lower cladding material property that should be recognized is the maximum elongation. The staff notes that, contrary to the stated, other key parameters, such as Young's modulus and yield strength, are higher than those with regular burn up.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-25 Justify the assumption that fuel rods in contact with the end fitting are to result in the most damage to the fuel clad during the end drop accidents.

This is referring to the "Contact Element." Secondary impact effects on fuel rod performance must be evaluated for the potential gap between fuel rods and the end fitting.

2-26 Reevaluate the maximum total strain, using the material properties associated with the fuel of regular burn up of no more than 45,000 GWD/MTU (refer to page 2.10.7-6, "Analysis and Results," on the maximum total strain of approximately 2.8%, which is based on higher than actual Young's modulus and yield strength).

The calculated maximum total strain of 2.8% appears to have been underestimated. At a lower yield strength associated with the fuel of regular burnup, the clad is likely to undergo a relatively larger total strain than the high-burn-up fuel in absorbing the same amount of kinetic energy associated with the end drop accident.

2-27 Revise, as appropriate, the summary g-loads, listed on page 2.10.7-11 for static analysis, based on the ADOC calculated maximum deceleration of 78 g reported in Table 2.10.9-1 of the application.

Table 2.10.9-1 reports the maximum cask decelerations of 61 g and 78 g for the measured and calculated slap down drops, respectively, which are much higher than the listed side-drop cask deceleration of 42 g. The staff notes that a bounding basket static g-load of 75 g is used after considering the dynamic amplification effect. When the same dynamic amplification of 1.25 is applied to the measured and ADOC calculated results, the respective loads of 76.3 g and 97.5 g are greater than the baseline deceleration of 75 g.

Section 2.10.8 Structural Evaluation of the Impact Limiters

2-28 Verify that the side-drop deceleration of 51 g is bounding for the basket structural analysis (refer to Table 2.10.8-10, “Loading used in Basket Structural Analysis, Appendix 2.10.5 versus Maximum g-load Predicted by ADOC Program”).

The staff notes, on page 2.10.7-11, that a side-drop deceleration of 42 g is used in the basket structural analysis. As commented previously, Table 2.10.9-1 reports a calculated slapdown deceleration of 78 g, which should be considered for side-drop analyses.

Complete and accurate information must be provided in the application in accordance with the 10 CFR 71.7(a) requirements.

2-29 Discuss how the Item 36 redwood, shown in Drawing 10421-71-42 and Figure 2.10.8-1, is modeled in by ADOC for the Region II material for impact limiter performance evaluation (refer to Figure 2.10.8-8, “Geometry of Impact Limiter Parameters,” and Figure 2.10.8-4, “ADOC Computer Model for TN-40 Transport Package”).

Figure 2.10.8-8 appears to suggest that the redwood directly backed by the cask end plate or closure lid can assume only a singe set of material properties for Region II redwood. It’s unclear how the redwoods with mixed grain orientations in a single region are considered for impact limiter analysis by ADOC.

Section 2.10.9 TN-40 Package Impact Limiter Testing

2-30 Justify why the ADOC calculated transverse deceleration of 78 g for the 20^{o} slap down drop in Table 2.10.9-1 is not used as a baseline deceleration for determining bounding decelerations for basket and fuel rod cladding analyses. Revise structural evaluations, as appropriate, for the applicable bounding cask decelerations considering the impact limiter testing, as supplemented by the ADOC predictions.

It's unclear why the drop tests results by ADOC underestimate those measured for all drop orientations except for the slap down drop. The higher deceleration should be considered for subsequent basket and fuel clad analyses when the measured deceleration is shown less than the calculated.

2-31 Revise the application (Section 2.10.9.3) to add the basis for not conducting a quasi-static impact limiter load testing to derive force-deflection curves for the ADOC evaluation of drop tests.

Section 2.5.2 of the standard review plan, NUREG-1617, notes that the adequacy of the method used for establishing impact limiter force-deflection characteristics should be verified by testing.

2-32 Incorporate in the application (page 2.10.9-18), by reference, appropriate documents evaluating the TN-68 impact limiter drop tested at low temperatures, which are considered also for the TN-40 package.

The last bullet states that the effect of low temperature on the impact limiter wood is not available due to lost test data. It notes also that, based on drop testing the TN-68 packaging, use of the impact limiters chilled to – 20 ^{o}F will increase the cask rigid body deceleration roughly by 15% to 20%, compared to those tested at ambient temperatures. A complete list of documents on the subject test must be captured as reference for staff review.

2-33 Revise Table 2.10.9-1 to recognize the potentially increased g-loads as bounding decelerations for package structural evaluation by analysis. Revise the table, as appropriate, by recognizing the low-temperature (i.e., – 20 ^{o}F) increased wood crush strengths for ADOC calculation of cask decelerations as bounding decelerations for package structural evaluation.

The listed decelerations in the table appear to be incomplete and potentially misleading. As a summary of decelerations for comparing the tested to the calculated decelerations, it should also list, for the applicable ambient temperatures, the baseline decelerations considered for developing bounding g-loads for package structural evaluation. This information must be provided in the application in accordance with the requirements of 10 CFR 71.35(a).

3.0 THERMAL

Section 3.2 Summary of Thermal Properties of Materials

3-1 Provide justifications and correct values for the thermal conductivities in Table 5, p. 3-5 of the SAR.

The thermal conductivity stated for SA350 Grade LF3 steel is that given in the reference for 0.5Cr-0.5 Ni-0.2 Mo alloy not a 3.5 Ni alloy. Coefficients of thermal expansion given on page 2.10.2-23 of SAR for SA-350 L43 appear to be those of 2.25 Cr-1Mo and not the 3.5 Ni SA-350. Note that the thermal expansion coefficient for this material is correct on page 2.10.1-3

This information is needed to determine compliance with 10 CFR 71.43(f) & (g).

3-2 Revise the application to clarify the discrepancy between the material specification used in the ANSYS material properties file for the impact limiter shell and the material specification used on Drawing 10421-71-41 for the impact limiter outer shell.

There is a discrepancy between the material used for the impact limiter shell in the ANSYS material properties file and the material specification for the impact limiter outer shell on Drawing 10421-71-41.

This information is necessary to determine compliance with 10 CFR 71.71 and 71.73

3-3 Revise the application to expand the thermal conductivity and specific heat property limits for aluminum 6061 on page 3-4 and the material used for the impact limiter shell.

The predicted temperatures exceed the material properties limits for thermal conductivity and specific heat in the SAR, specifically for aluminum 6061 and the material used for the impact limiter shell.

This information is necessary to determine compliance with 10 CFR 71.71 and 71.73.

3-4 Revise the application to clarify the apparent discrepancy between the thermal conductivity and specific heat values for SA-203, Gr. E SA-350, LF3 on page 3-5 and SA-516 Gr. 70 carbon steel on page 3-6 in the SAR and the values used in the ANSYS material properties file.

Thermal conductivity and specific heat values for SA 203, Gr. E SA-350, LF3, and SA-516 Gr. 70 carbon steel in the SAR are different from those used in the ANSYS file matTN40tr.mac.

This information is necessary to show compliance with 10 CFR 71.71 and 71.73.

3-5 Provide the Prandtl number used for air as a function of temperature.

The Prandtl number is necessary for natural convection calculations used in section 3.4.1.4 of the application.

This information is necessary to show compliance with 10 CFR 71.71 and 71.73.

3-6 Confirm that the thermal environment under normal conditions of transport does not degrade the materials (wood, adhesives, etc.), cause changes in material properties, or adversely affect the ability of the impact limiters to perform intended safety functions.

The package must be designed, constructed, and prepared for transport so that there will be no significant decrease in packaging effectiveness under the tests specified in 10 CFR 71.71 (normal conditions of transport.)

This information is necessary to show compliance with 10 CFR 71.71 and 71.73.

Section 3.4.1 Thermal Models

3-7 Clearly describe the flow regimes (i.e., laminar or turbulent) that are assumed for the NCT condition thermal analyses. Specify where and how these boundary conditions are applied in the analysis model.

The application should specify and justify that the appropriate flow regimes are being correctly applied to the analysis model.

This information is necessary to show compliance with 10 CFR 71.71.

Section 3.4.1.1 Basket Model and Section 3.4.1.2 Impact Limiter Model

3-8 Discuss the sensitivity of thermal analysis results to change in gaps in the finite element model. If a sensitivity analysis has been performed, discuss the results of that analysis.

The application should verify that the chosen gaps in the finite element model are producing the most conservative thermal results with respect to the dimensions and tolerances specified in the design drawings.

This information is necessary to show compliance with 10 CFR 71.35.

Section 3.4.1.4 Heat Dissipation

3-9 Revise the application to clarify the apparent discrepancy between the convection equation used for vertical flat plates in the SAR on page 3-12 and the one used in the given reference.

The equation used in the SAR is: $Nu_t = C_t^V Ra^{1/3}$, while the equation in the "Handbook of Heat Transfer 3rd Ed." is: $Nu_t = C_t^V Ra^{1/3} / \left(1 + 1.4 \cdot 10^9 \, \mathrm{Pr} / Ra\right)$ Eq 4.33c

This information is necessary to show compliance with 10 CFR 71.35.

Section 3.4.2 Maximum Temperatures and Tables 3-1, 3-2, and 3-3

3-10 Revise the application to describe where in the analysis model the maximum lid o-ring seal temperature is extracted from. These values are shown in Tables 3-1, 3-2, and 3-3.

The application should justify how the lid o-ring seal temperature is determined considering the seal is not modeled in ANSYS.

This information is necessary to show compliance with 10 CFR 71.35.

Section 3.4.3 Maximum Accessible Surface Temperature in the Shade

3-11 Demonstrate analytically that the surface of the personnel barrier remains below 185°F for all NCT conditions.

In order to protect personnel from temperatures that exceed the allowed accessible surface temperature of 185°F as described in 10 CFR 71.43(g), the personnel barrier must remain below 185°F for all NCT conditions.

This information is necessary to show compliance with 10 CFR 71.43(g).

Section 3.5.3 Crushed Impact Limiter Models

3-12 Provide a comparison of the impact limiter crush depths (derived from the structural evaluation) used in the thermal analysis and those witnessed in the scale model testing of the package with impact limiters. Provide a discussion of how the results of the structural analysis were validated by the results of the testing program.

Additional discussion is necessary regarding comparison of the impact limiter crush depths from the structural evaluation and those found in scale model testing as well as further discussion as to how the scale model testing validated the structural evaluation results.

This information is necessary to show compliance with 10 CFR 71.35.

3-13 Provide a reference for the assertion made in Section 3.5.3 (page 3-16 of the SAR) that the impact limiters "remain firmly attached to the cask," following the 30 foot drop.

Additional confirmation is necessary to support the assertion that the impact limiters remain firmly attached to the cask.

This information is necessary to show compliance with 10 CFR 71.35.

3-14 Provide the temperature of the wood impact limiter material during the scale 30 foot drop test. Describe how wood properties at the predicted NCT temperatures were accounted for in the drop testing results and subsequent analyses.

Wood properties can vary depending on temperature and humidity, which directly affect moisture content. The behavior of the wood predicted by the thermal analysis for NCT must be considered for the HAC evaluation of the 30 ft. drop. The SAR does not clearly discuss how the NCT temperatures of the wood were accounted for in either the physical HAC testing or the analysis of the HAC conditions.

This information is necessary to show compliance with 10 CFR 71.35.

Section 3.5.4 Summary of Results

3-15 Revise the application to provide a table of post-fire steady-state temperatures.

This information is necessary to determine compliance with 10 CFR 71.35.

Section 3.7.1 Effective Thermal Properties for Fuel Elements

3-16 Provide copy of the primary data document used in the Prairie Island SAR to substantiate the thermal properties of the UO_2 on page 3.7.1-1 of SAR.

The thermal conductivity values do not agree with the thermal conductivity stated in MATPRO. Some values are high some are low.

This information is needed to determine compliance with 10 CFR 71.43(f) & (g).

3-17 Provide a copy of the reference for hemispherical emissivity of 0.46 that is given for 304 stainless steel, on page 3.7.1-3 of the SAR.

The staff's reference (VTT Research Notes 2299, Steel emissivity at high temperatures, T Paloposki and L Liedquist) gives a value 0.35 to 0.3 in the temperature range of 200-400 C.

This information is needed to determine compliance with 10 CFR 71.43(f) & (g).

Section 3.7.1.4.2 Finite Element Model

3-18 Clarify the modes of heat transfer used in the two-dimensional quarter symmetry model of the fuel assembly.

In Section 3.7.1.4.2 it is stated that the "two-dimensional quarter symmetry model of the fuel assembly simulates heat transfer by radiation and convection" yet in the following paragraph it is stated that "no convection is considered within the fuel assembly model."

This information is necessary to show compliance with 10 CFR 71.35.

Section 3.7.1.4.3 Axial Effective Conductivity

3-19 Revise the application to clarify how the fuel axial thermal conductivity values were calculated in the SAR.

In Section 3.7.1.4.3 Axial Effective Conductivity, the equation $k_{axial} = \frac{cladding\ area}{4a^2} \cdot cladding\ conductivity$ is used to calculated fuel axial thermal conductivity. Using a = 4.025 inches, cladding area = 5.05 in^2, and conductivity values for Zircaloy-4 from Section 3.7.1.3, the staff calculated fuel k_{axial} values that were lower (by approximately 9%) than the values shown in the SAR.

This information is necessary to show compliance with 10 CFR 71.33(b).

Editorial: The following editorial comments are noted for information and should be corrected.

However, these items do not need to be addressed in an RAI response.

(a) Clarify that "dept" should be "depth" page 3-16, 5th paragraph.

(b) Minor in magnitude, but clarify why there is a change in the 4th digit after the decimal of Qreact at T_0 = 1000 °F in the table on page 3.7.1-5.

(c) In the table on page 3.7.1-5, clarify why 0.671 kW is used for the decay heat instead of 0.675 kW as stated earlier in the appendix.

(d) On page 23 and page 24 of Enclosure 3 to TN E-25513, the allowable fuel temperature limit is 716°F, clarify that should be 752°F like on Table 3-1 of the SAR.

(e) (Proprietary due to equation used) on page 28 of Enclosure 2 to TN E-25513 equation Aj is written as: $A_j = \sum_{1}^{n} \frac{(P_{i+1}+P_i)}{2(l_{i+1}-l_i)}$, but clarify that the equation should be: $A_j = \sum_{1}^{n} \frac{(P_{i+1}+P_i)\cdot(l_{i+1}-l_i)}{2}$.

(f) (Proprietary due to values and file name mentioned) clarify the discrepancy between the value used in the ANSYS file and in Enclosure 2 to TN E-25513, page 29, Table 10-1.

ANSYS file heatgen.mac Line # 43: pfact(9)=0.881.

In Enclosure 2 to TN E-25513 page 29 Table 10-1: $P_{avg}(9)$ = 0.883.

4.0 CONTAINMENT

4-1 Clarify which material (Nimonic 60 or Nimonic 90) is used for the spring detailed in Section 4.1.3. If Nimonic 60 material is used, provide technical and material specifications for Nimonic 60 material and its performance under normal and accident conditions. The technical and material specifications must include composition, ranges of temperature, pressure, and other materials properties. Also provide detailed description of “equivalent material” to Nimonic 90 or Nimonic 60 with respect to specific materials, and their properties.

Section 4.1.3, paragraph 2, of SAR states that *the spring is Nimonic 90 or equivalent material.* Section 4.1.3, paragraph 5, of the SAR indicates that the inner spring of the high performance Helicoflex metallic seal is made of *Nimonic 60 or equivalent material which ensures the seal will not be affected by relaxation and thus the seal can be maintained at the specified temperatures for extended periods.”* Please clarify which material (Nimonic 60 or Nimonic 90) is used for the spring. The application/SAR does not include any information on Nimonic 60 material or any references for it.

This information is required for safety evaluation in accordance with 10 CFR 71.33(a)(5) and 71.43.

4-2 Clarify and explain in detail how the package complies with 10 CFR 71.43(e) in retaining any leakage of radioactive contents of TN-40 packaging.

10 CFR 71.43(e) requires that “*A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operation and, except for a pressure relief device, must be provided with an enclosure to retain any leakage.”* The application does not contain any information to satisfy this criterion.

This information is needed to determine compliance with 10 CFR 71.43(e).

4-3 Provide a detailed calculation package for Basket Volume (1.05E+05 in^3) and Fuel Assembly Volume (1.64E+05 in^3) on page 4-4 of the SAR.

This item is used in the calculation of source activity released from fuel in transport and subsequently in the determination of permissible leakage rates for normal conditions of transport and for hypothetical accident conditions. A detailed calculation is not included in the SAR.

This calculation package is required to verify the applicant’s evaluation of package design under normal conditions of transport per as required by 10 CFR 71.71, and under hypothetical accident conditions per 10 CFR 71.73.

4-4 Provide the rationale for selecting Westinghouse 14 x 14 class PWR fuel assembly as having the bounding fuel parameters for containment analysis. Describe how the various fuel combinations in Table 1-2 were evaluated to determine the design basis assembly for containment

It is stated in Section 4.2.1.1 for the calculation of source activity from the fuel that SAS2H evaluations were performed to determine the design basis fuel for shielding. It is further stated that these SAS2H analyses were also evaluated to determine the bounding fuel parameters for the containment analysis. The bounding SAS2H evaluation was performed for the Westinghouse 14 x 14 standard fuel assembly with 39000 MWD/MTU burnup, enrichment 3.3 wt% U-235 and a cooling time of 15 years. Additional information is required to justify selection of standard Westinghouse 14 x 14 assembly at 39 GWd/MTU as the fuel having bounding parameters for containment analysis.

This information is required to verify the applicant's evaluation of package design under normal conditions of transport per as required by 10 CFR 71.71, and under hypothetical accident conditions per 10 CFR 71.73.

4.5 Explain in detail, how all the required leakage tests as listed in Chapter 7 of ANSI N14.5-1997 are performed. The leakage tests to be performed are listed in Chapter 7 "Test requirements" of ANSI N14.5 that includes Design leakage test, Fabrication leakage test, Maintenance leakage rate test, Periodic leakage rate test, and Preshipment leakage rate test.

Your response must include details such as, frequency of tests, affected components, required leak test sensitivity, specific results attained or expected from tests, main and alternate acceptable methods of testing, and acceptance criterion for each test.

This response is required by staff to determine whether all leak tests required by ANSI N14.5 are performed in compliance with 10 CFR 71.37(b), 10 CFR 71.43(f), and 10 CFR 71.87(c).

4-6 Explain the procedure and the instrument with which the user should perform the leak test described in Section 7.1.3.7 of Operating Procedures.

Section 7.1.3.7 of the Operating Procedures list to perform leak test the inner lid, inner vent, and drain port cover seals. In Section 8.1.3, "Containment Boundary Leak Tests" and in Section 8.2.2, "Leak Tests," it is stated that helium spectrometer or alternate methods are acceptable. However, for Section 7.1.3.7, no specific methods or details of testing are listed.

This information is required to show compliance by the licensee with 10 CFR 71.37(b), 10 CFR 71.43(f), and 10 CFR 71.87(c), and required by ANSI N14.5-1997.

4-7 Clarify the reason for not including Kr-85 in Table 4-1 for the hypothetical accidents.

This information is needed to demonstrate compliance with 10 CFR 71.51

5.0 SHIELDING

5-1 Provide the commercial name of the neutron absorber material and data or analyses to show that the neutron shield material will retain adequate properties for the application after being subjected to service conditions.

The section before 1.2.1.2 Gamma and Radial Neutron Shielding in the SAR states the neutron shield material is a borated polyester resin compound that surrounds the gamma shield shell and it is subject to thermal and radiation fields during service which have the potential for degrading properties of the material including its thermal conductivity.

This information is needed to determine compliance with 10 CFR 71.47(a).

5-2 Indicate in the SAR that any casks that have been loaded and used for storage will be inspected for cracks in the carbon steel shell that provides gamma shielding, containment, and structural stability.

This information is needed to determine compliance with 10 CFR 71.47(a), 71.73(c)(1,2,3), and 71.33.

5-3 Correlate the qualification and acceptance testing of the neutron shield with expected performance. Indicate how an adequate percentage of H and B are ensured in acceptance tests; describe the significance of the density measurement, and the sensitivity of measurements to the percentage of critical components of the material. Provide by reference to the SAR a Technical Specification requirement for controlling the manufacture and testing of all neutron poisons used in any future TN-40 spent fuel canisters.

This information is needed to determine compliance with 10 CFR 71.47.

5-4 Justify the use of the average peaking factors (1.049) for gammas instead of the maximum value of 1.16.

On page 5-3, the applicant states: “the peaking factors (for gammas) range from 0.700 at the bottom and top, to a maximum of 1.16 just below the middle.” However, on page 5-4 of the SAR the applicant states: “The source strength must be scaled approximately by the axial burnup normalization constant. For the axial profile the cumulative density of the burnup profile is 1.049.”

This information is needed pursuant to the requirements of 10 CFR 71.47.

5-5 Justify the use of the average peaking factors (1.367) for neutrons instead of the maximum value of 1.811.

This information is needed pursuant to the requirements of 10 CFR 71.47.

5-6 Provide more details about the response function that was utilized with SAS2H to estimate the 2 meter side dose rate as a function of different burnup, enrichment, and cooling time combinations.

The Safety Analysis Report should include the calculation description such as, summation process, or formula that explains how the method is used to sum doses; the axial and azimuthal locations that were considered, the method for statistically combining standard deviations, and how the burnup profile and adjustment for axial neutron and gamma source terms are utilized in the method. In addition, the two-step method should be validated against direct MCNP calculations for representative burnup, cooling time, and enrichment.

This information is needed pursuant to the requirements of 10 CFR 71.47.

5-7 Provide the definition for the term “cumulative density of the burnup profile.”

On page 5-4 of the SAR, the applicant states: “cumulative density of the burnup profile is 1.049.” From Table-5-12, it appears that the average axial source terms peaking factor for gammas is 1.049. It is not clear to the staff if these two terms are the same.

This information is needed pursuant to the requirements of 10 CFR Part 71.47.

5-7 Provide justifications on why 42 GWD/MTU and 24.4 years cooling time produce the bounding source term for the entire proposed spent fuel payload.

On Page 5-1, Section 5.1, the applicant states that the bounding design basis fuel for dose rate has an initial enrichment of 2.35 wt% of U-235 and a total maximum bundle-average burnup of 42,000 MWD/MTU with 24.4 years decay time. However, it is not clear to the staff what the basis for this statement is.

This information is needed pursuant the requirement of 10 CFR 71.47.

5-8 Explain why the maximum dose rate in Table-5-2 is different from that of Table 5-10.

The staff reviewed the maximum dose rate listed in Table 5-2 and Table 5-10 and found there is an inconsistency between these two tables. The applicant is requested to check these numbers and make corrections to the SAR as necessary.

This information is needed pursuant the requirement of 10 CFR 71.47.

6.0 CRITICALITY

Unless otherwise stated, the following information is needed pursuant to the requirements of 10 CFR 71.55, 71.59, and 71.73.

Section 6.1 Discussion and Results

6-1 Provide the technical basis for the conclusion that the bias due to the criticality safety evaluation code with additional benchmark data to account for the composition of the burned fuel is NOT required and is based on the reactor critical benchmark results.

On page 6-2, the applicant states: "The bias due to the criticality code with additional benchmark data to account for the composition of the burned fuel is not required and is based on the reactor critical benchmark results summarized in Reference [3]." The conclusion from Reference [3] is that reactor restarts can be used as benchmarks. The applicant needs to develop its own bias and the uncertainties associated with the cross sections used in SCALE 4.4 for TN-40 by modeling the specific reactor cores using the same cross sections.

6-2 Justify the conclusion that the results described in Ref. 3 are adequate for use as the basis for code bias in the spent fuel criticality safety evaluation.

On page 6-2, the applicant states that the bias due to the criticality code with additional benchmark data to account for the composition of the burned fuel based on the reactor critical benchmark results summarized in Reference [3]. The staff finds that the calculations in Ref. 3 may not qualify as a basis for determining the bias of the computer code for this particular application because the fuel burnup in Ref. 3 is only to 19 GWd/MTU, whereas the burnup in this application is 31 GWd/MTU. In addition, the cross section library used in Ref. 3 is different from the cross section library used in the evaluations of this application.

6-3 Provide specific definition for the terms "Maximum Average Initial Enrichment."

On page 6-41, Table 6-1, uses a term "Maximum Average Initial Enrichment." However, it is not clear what is the meaning of the term. If a special variation in assembly initial enrichment exists, a description of the variation should be provided and its effect on cask reactivity should be evaluated. Normally, the highest enrichment is used as the enrichment for the fuel assembly for conservatism.

6-4 Provide specific parameters and criteria that define the term "equivalent reload assemblies."

On page 6-42, the note to Table 6-2 states that equivalent reload assemblies from the manufacturers are also acceptable. However, no specific criterion is given that can qualify other spent fuel assemblies and NFAH as equivalent reload assemblies.

Section 6.3.1.1 Model Specification

6-5 Change the title of Table 6-3 to Required Fuel Assembly Parameters for SAS2H Models and add a note with respect to BPRAs.

On page 6-42, Table 6-3 provides some of the fuel assembly super cell parameters used in the SAS2H models. The title of Table 6-3 reads as “Required Fuel Assembly and Reactor Parameters for SAS2H Models.” There is, however, no parameter in the table that is reactor operating data. In addition, a table note should indicate that the SAS2H Region 1 contains assembly guide tube filled with either moderator or BPRAs and that the BPRA model radii are provided in Table 6-6. This would improve the description of the SAS2H models used in the safety analysis.

6-6 Provide justification for the values used for the limiting reactor operating parameters including specific power and fuel temperature. Indicate whether the soluble boron concentration used is the maximum average boron concentration. Also, indicate whether the limiting moderator density and temperature used in depletion calculations correspond to the average core outlet temperature for the reactor designs considered in the safety evaluation.

Maximum moderator temperature and the corresponding minimum density are required in depletion calculations since these parameters occur near the top of the fuel as demonstrated in published studies (e.g., DOE/RW-0472 Rev. 2 and NUREG/CR-6800). In a burnup-credit cask, the assembly top regions dominate the reactivity (see NUREG/CR-6801, Figure A.18). Furthermore, there are no discussions on specific power and fuel temperature.

6-7 Provide justification for the applicability of the recommended bounding axial profiles (that are not based on any data from Westinghouse and Exxon/ANF 14x14assemblies) for the W 14x14 assembly analyses.

Recommended bounding axial burnup profiles analyzed in NUREG/CR-6801 were used in the analyses. These recommend profiles were selected from a database of axial profiles that did not include any Westinghouse and Exxon/ANF 14x14 assemblies.

6-8 Justify 570 °K is a bounding value for the reactor core coolant temperature used in the depletion calculations for all fuel assemblies that are qualified to be loaded into TN-40.

On page 6-6 of the SAR, the applicant states: “It is conservative to utilize the highest [moderator] temperature and density in the depletion calculations since these parameters result in a slightly harder spectrum thereby producing more Pu-239 by way of capture in U-238 and hence increasing the residual U-235 at discharge.” Replace the words “and density” with the phrase “which corresponds to the lowest density” in order for the statement to be accurate. Secondly, the applicant needs to demonstrate that none of the fuel assemblies already loaded or to be loaded into TN-40 have been irradiated in reactor cores with a moderator temperature higher than 570 °K .

6-9 Provide technical basis for the assertion that 600 ppm soluble boron is adequate for the fuel assembly depletion analyses.

On page 6-6 of the SAR, the applicant states: “For this calculation, an average soluble boron concentration of 600 ppm was utilized for all the zones during depletion. This modeling is considered adequate and is equivalent to modeling a boron letdown curve where the soluble boron concentration is higher at the beginning of the cycle and lower (from 0 ppm to 100 ppm) at the end of cycle.”

The staff finds that using average soluble boron of 600 ppm may not be conservative because the analyses in Section 6.4.2.B states that the Prairie Island plant uses 675 ppm average boron concentration. First, SAS2H creates boron letdown curve using a linear equation that sets the Beginning of Cycle (BOC) to 1.9xAverage and 0.1xAverage at EOC (End of Cycle). Thus, the BOC boron concentration for average boron concentration of 675 ppm should be 1282 ppm rather than 1140 ppm as used in the analyses. In addition, higher soluble boron concentration at the BOC may prolong the cycle life slightly. Assembly depleted with higher BOC soluble boron concentration may result in higher residual reactivity for the same burnup.

6-10 Provide information on the maximum B_2O_3 used in the spent fuel assemblies to be transported by the TN-40 packages.

On page 6-6 of the SAR, the applicant states: “The BPRA material is Pyrex based with a B_2O_3 content of 12.615% by weight.” It is, however, not clear what was the maximum B_2O_3 load in the actual fuel assemblies while they were depleted in the reactor cores.

6-11 Justify that the assumptions with respect to the number of burnable poison rod assemblies, duration of their insertion into fuel assemblies during irradiation, and the number of such assemblies in the casks bounds the assemblies already loaded or to be loaded into TN-40.

The ISG-8, Rev. 2, indicates that the modeling assumption of assemblies being exposed to the maximum loading of burnable poison rods for the maximum burnup is an appropriate analysis assumption that encompasses the impact of exposure to burnable absorbers and to fully or partially inserted control rods. This SAR assumes 16 BPRA are present for two thirds, instead of full, depletion. It is not clear how those spent fuel assemblies which have already been loaded or future assemblies are to be loaded satisfy these assumptions.

6-12 Provide a description of the conclusions of Ref. 8 concerning the Westinghouse IFBA rods and the strategies recommended in Ref. 8, Section 5, to be used to address the small positive reactivity effect due to IFBA rods.

Reference 8 concluded that for assemblies with IFBA rods, such as the Westinghouse IFBA rods, there is a small positive reactivity effect that depends on the axial length of the IFBA coating. An analysis demonstrating that the effect of the IFBA rods is bounded by the effect of other modeling assumption (e.g., BPR exposure) is recommended in Ref. 8. Also, see comment 6-7.

6-13 Confirm that none of the fuel assemblies to be shipped in the TN-40 packages had major control rod insertions (i.e., > 20 cm) for a significant period of time during their depletion in the cores so that no analyses are warranted for the residual reactivity concern.

On page 6-7 of the SAR, the applicant states: “A parametric study of the effect of control rod insertions for PWR burnup credit was performed in Reference [7]. The report concludes that the effect of modeling a fully inserted control rod in all the guide tubes of the fuel assembly during the entire depletion results in a fuel assembly with a higher reactivity at discharge than one without control rods inserted. However, the PWR operations in the United States (U.S.) are generally performed for base load and do not have control rods inserted in the fuel assembly in the majority of the time. The report also concludes that the effect is insignificant for control rods inserted in the top segment of the fuel assembly. This control rod movement is more consistent with the operation of the U.S. PWRs and therefore, control rod movements are not considered in the SAS2H model." The staff agrees with the conclusions of the report [Ref. 7]. The staff, however, needs a confirmation that the spent fuel assemblies to be shipped by TN-40 packaging system do meet the conditions under which the conclusions were drawn. If it is true that the control rods were only inserted in the top segment of the fuel assemblies will be imposed in the CoC because the impact of control rod insertions to the residual reactivity of spent fuel assembly is positive and significant.

6-14 Describe the assumed orientations of assemblies with horizontal profiles in the casks and the calculated bias.

The orientations of spent fuel assemblies with tilted horizontal profiles in a casks affects the effective multiplication factor. The assumption with respect to the orientations of the fuel and associated bias needs to be provided.

Section 6.3.1.2 CSAS25 Models

6-15 Provide information on the evaluation that demonstrates that modeling the gap between the poison plates as aluminum is conservative over modeling that as internal moderator.

On page 6-9 of the SAR, the applicant states: "The poison plates are modeled with a thickness of 0.075 inches and a width of 7.50 inches. The gap between the poison plates between adjacent compartments is modeled conservatively as aluminum. An evaluation is performed to demonstrate that the modeling of aluminum between the poison plates is conservative over utilizing internal moderator." There is, however, no information on the evaluation that demonstrates that the modeling of aluminum between the poison plates is conservative in comparison with using internal moderator.

6-16 Clarify the definition of the term "design basis model."

On page 6-9 of the SAR, the applicant states: "Parametric calculations are done to make minor modifications to the previous model in order to determine the effect of tolerances on important basket design parameters. Fuel assembly positioning and burned fuel modeling are also included as part of these modification so that the resulting model (third model) is the design basis criticality model for criticality calculations." It is not clear from these statements if this model represents the most reactive cask loading.

6-17 Provide information on where in the application the criticality safety evaluation is furnished for the Hypothetical Accident Conditions as specified in 10 CFR 71.73.

This application seeks approval to transport spent nuclear fuel using the TN-40 packaging system. For this purpose, the design must meet all the requirements prescribed in 10 CFR Part 71. Regulatory Guide 7.9 provides guidance on the information that should be presented in the SAR.

This information is needed pursuant to the requirements of 10 CFR 71.73.

Section 6.3.2 Package Regional Densities

6-18 Provide a reference to the benchmark analysis with results statistically similar to those from Ref. 6 or indicate the report section describing this analysis.

The justification provided in Section 6.3.2, last sentence of the 3rd paragraph, is based on a benchmark analysis for which a reference is not provided.

6-19 Provide technical basis for the assertion that correction factors for Ru-101, Rh-103, and Ag-109 are identical to that for Ru-106 because these isotopes are expected to build up in a similar fashion as those whose factors are being utilized.

On page 6-10, the SAR states: "The four isotopes that do not have explicit benchmarking results for isotopic compositions but are utilized in this calculation are Mo-95, Ru-1 01, Rh-103, and Ag-109. The factor utilized for Mo-95 is 0.87 and is the same as that used for Tc-99. The factors utilized for Ru-101, Rh-103 and Ag-109 are identical to that for Ru-1 06 which is 1.00. This is done since these isotopes are expected to build up in a similar fashion as those whose factors are being utilized. These isotopes are part of the 28 isotopes that are utilized in the criticality calculations." The staff finds this conclusion and hence the values of the correction factors may not be valid without further justifications. The correction factors for isotopes need to be developed based on chemical assay data. Such data have been published in the past.

6-20 Provide information on which of the 28 isotopes in Table 6-10 were identified as exhibiting significant trending and what further corrections were made. Furthermore, justify why trending with respect to other parameters such as cooling time, a neutron spectral index, etc… are not needed.

On page 6-10a, the SAR states: "For isotopes that exhibited trending, the most conservative value was chosen as being applicable for the entire range effectively ensuring that the resulting factor is conservative. New "correction" factors based on best estimate evaluation of trending and assay comparison are obtained and are also shown in Table 6-10." The staff, however, was unable to find the "New correction factors" from Table 6-10. In addition, given the one dimensional code used for the depletion calculations, it is important to use other parameters such as a neutron spectral index to examine the adequacy of the computer code.

6-21 Provide information on resultant effective multiplication factor for the 25 isotope burnup credit cask and a discussion on the significance of this model.

On page 6-10a, the SAR states: "Criticality calculations are performed that take credit only for Rh-103 isotope among the isotopes that do not have benchmarking results (25 isotopes) to determine the bounding k_{eff}." The staff, however, was unable to find any further discussion on the significance of this model in comparison with the 28 isotope burnup credit cask. It is not clear if the analysis is based on 25 or 28 isotopes.

Section 6.4.1.4 Bases and Assumptions

6-22 Provide test results that can demonstrate that the fuel rods/assemblies will always bend inward toward the center when a cask is involved in a 30-feet drop as prescribed in 10 CFR 71.73.

On page 6-13, the SAR states: "The analytical results reported in Appendix 2.10.7 also demonstrate that the fuel assemblies do not undergo deformation significant enough to affect criticality evaluations. Though these analyses indicate that fuel rods undergo permanent plastic deformation following accidents, the resulting rod configuration is less reactive since it results in the fuel rods coming closer to each other." The staff needs further evidence to support this conclusion because there is no mechanism in the cask or assemblies to prevent the fuel rod to bend outward.

6-23 Justify using pure water instead of a mixture of water and steel for top and bottom nozzles is conservative.

Assumption 4 indicates that only the active fuel length of each assembly is explicitly modeled with water (albedo) boundary conditions on the ends. That may not be conservative.

6-24 Justify using pure water instead of actual cask bottom shell and top lid is conservative.

Assumption 4 indicates that only the active fuel length of each assembly is explicitly modeled with water (albedo) boundary conditions on the ends. That may not be conservative.

6-25 Clarify and justify the fuel density value used in calculations.

Assumption 8 states that [the model] used 95.5% theoretical density for fuel although this assumption conservatively increases the total fuel content in the model. It is not clear what the actual fuel density is and what is the basis for the conclusion that this assumption conservatively increases the total fuel content. In addition, the staff found that the SAS2H input file provided in Section 6.7.1 of the SAR uses 96% theoretical density. However, the input file provided in Section 6.7.2 uses UO_2 theoretical density of 0.975

Section 6.4.2 Fuel Loading Optimization

6-25 Provide information on exactly what neutron poison material will be used in the cask.

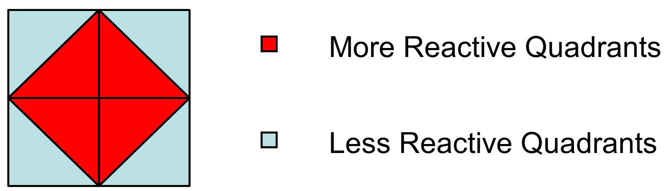
On page 6-16, the SAR states: "The next set of calculations evaluates the effect of the poison material modeling on the reactivity of the cask. The poison material is modeled with boral, instead of borated aluminum, with an absorber loading of 7.5 mg B-10/cm^2." The staff finds these statements confusing. It is not clear which neutron poison material will be used in the casks. If different neutron poison materials are to be used, neutronic as well material evaluations must be performed to qualify these materials because different materials that perform similarly may behave very differently in terms of chemical and mechanical properties.

6-26 Explain the k_{eff} result in Table 6-16 that was obtained for the following parameters: 3.85 wt% U-235, 31 GWd/MTU, 15 years cooling, 46 isotopes.

The k_{eff} value from a calculation that used 46 isotopes (0.9349) is much higher than that obtained from a calculation that used 28 isotopes (0.9324).

6-27 Justify using only four assemblies exposed to the specific horizontal burnup gradient in horizontal bias determination is appropriate.

The assembly configuration used in horizontal burnup gradient evaluation may not be conservative since only four assembly exposed to horizontal burnup gradient were used. Therefore, the analysis may be inadequate to serve as the basis for horizontal bias determination. In addition, the staff finds that the loading pattern depicted in Figure 6-8 may not be the most reactive configuration. As a matter of fact, the power density distribution and hence the burnup gradient always tilts across the diagonal rather than the x-y plane of a fuel assembly. The most reactive configuration may be resulted from the following loading pattern for the four center fuel assemblies.



6-28 Provide a description of the effect of utilizing a simplified axial burnup profile that is consistent with the k_{eff} results.

The effects are described as being statistically insignificant. However, the results in Table 6-16 show that the effect of utilizing a simplified axial burnup profile is statistically significant (and bounding) for the case of 3.85 wt% U-235 and 31 GWd/MTU (0.9346 versus 0.9318) and statistically insignificant for the other two cases.

6-29 Discuss why an extra complication is necessary to model soluble concentration as a constant in the SAS2H model if the system creates internally a boron letdown curve that is closer to the real fuel depletion process.

On page 6-19, the SAR discusses the impact of using constant soluble boron versus a letdown curve. The staff's understanding is that SAS2H will create a boron letdown curve using the input soluble boron concentration. This is much closer to the reality of the PWR operation. The staff does neither understand nor agree with the idea of introducing the extra complexity in using the SAS2H computer code. If the suggested constant soluble boron was used in the fuel assembly depletion analyses in the SAR, the entire criticality and possibly shielding evaluations need to be redone.

6-30 Develop a more refined burnup credit loading curve.

On page 6-19, the SAR states that a third order polynomial is fitted to determine the minimum required burnup as a function of initial enrichment based on the five burnup/enrichment combinations shown in Table 6-16. A step function needs to be used especially from the first point to the second. There is a minimum bias and uncertainties associated with burnup credit for which should be accounted in a step function fashion instead of continuous function with the enrichment starting at 1.65 wt%. In addition, enrichment increment needs to be justified. It appears that a loading curve based on a finer enrichment increment, especially from 1.65 wt% to 2.85 wt%, to be more appropriate.

6-31 Clarify if the BPRAs are loaded together with the spent fuel assemblies into the casks.

If the burnup credit loading is based on spent fuel assemblies with BPRAs inserted and needed for transportation, BPRAs become part of the criticality safety control system. Therefore, a safety assessment of BPRAs, especially under accident conditions, needs to be performed. Information regarding the structural integrity of BPRAs and assurance for their position under all transportation conditions including accident conditions are needed. If that is not the case, the "Discussion on Additional Reactivity Margin" with respect to BPRAs should examine the assumption on duration of BPRAs insertion into fuel assemblies during irradiation in a reactor core.

6-32 Provide justification that the simplification of using the characteristics of the spent fuel top ends is valid because the reactivity of a burnt fuel assembly is dominated by the reactivity of the most under-burned zone or the "top axial" zone.

The staff agrees that this is in general true but it may not be valid for low enrichment/high burnup fuel assemblies. Provide evaluation for fuel assemblies with 1.65 wt% enrichment and 31 GWd/MTU burnup.

6-33 Provide separate discussions for the different factors discussed in Item iv on pages 6-20, 6-21, and 6-21a and explain the physics that supports the conclusion that even a most conservative representation of the loaded cask configuration would result in a reduction of the k_{eff} by approximately 0.02.

On pages 6-20 through 6-21a, the applicant discussed factors that may provide additional criticality safety margins to the casks. Some of these conclusions are valid and others are not. For example, configurations A, B, and C are not fully described and is not clear what they are trying to demonstrate. Another example, is that it's not clear if the "simplified model" (i.e., DB-2D) is used for constructing the loading curve. If that is the case, a more robust justification is needed. Furthermore, as indicated in RAI 6-28, the sensitivity results on 8-zone vs. 18-zone axial burnup profile models is not "statistically insignificant." Therefore, going to a 1-zone model maybe even more significant.

6.5 Critical Benchmark Experiments

6-34 Justify the applicability of the selected benchmark critical experiments to the validation of a criticality calculation method based on burnup credit.

The 142 benchmark critical experiments selected as a component of validation methodology are not entirely representative of casks loaded with the spent fuel assemblies. Only 11 of the 142 experiments contain mixed oxide. The remainders are all uranium dioxide fuels. Further, the safety analysis indicated that the bias due to the criticality code with the additional benchmark data to account for the compositions of the burned fuel is not required based on the reactor core follow records summarized in Ref. 3. However a description of that basis is not provided. Therefore, the bias evaluated in this safety report may be inadequate and less conservative than a bias resulting from validations based on representative spent fuel compositions. The bias, as evaluated in this safety analysis, is more appropriate for fresh fuel assemblies than spent fuel assemblies.

6-35 Provide the statistical confidence level used to determine the USL functions presented in Table 6-21.

The statistic employed in the USL-1 method (Ref. 4) is a function of statistical confidence level.

Appendix 6A Assigned Burnup Loading Value

6A-1 Provide a discussion on criticality risk with misloaded assembly under 10 CFR Part 71 requirements.

The discussion presented in Section 6A.1 is based on administrative check to prevent misload and boron in the spent fuel pool to prevent criticality if fuel assemblies are misloaded in casks. With respect to administrative check, experience with spent fuel pool loadings has shown that misloads have happened and will continue to happen. One needs just to examine the NRC Licensee Event Report (LER). With respect to presence of boron in spent fuel pool, 10 CFR 71.55(b) requirement is not based on borated water and is not limited just to loading. Criticality safety requirements in 10 CFR 71.55 and 71.59 encompass loading, transport, and unloading. Therefore, the criticality safety design for transportation casks must be based on "if water were to leak into the containment system …"

One approach could be looking at the effect of a misload and determining the available safety margin. Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible.

6A-2 Provide the verification method with respect to irradiation history parameter value limit including cooling time associated with spent fuel assemblies loaded or to be loaded into TN-40.

The applicant needs to propose how the design assumptions with respect to moderator temperature, soluble boron letdown, and cooling time will be verified. The applicant identified two measures to prevent loading the wrong assemblies into the cask. However, it is not clear if any of the irradiation history parameter values are included in the consideration. The verifications should be included in Chapter 7 as part of the package operation.

6A-3 Provide justification on why the accumulative uncertainties of the fuel burnup records from cycle to cycle were not considered.

The applicant proposes use of core follow records as the basis of the fuel assembly burnup values. It discussed the possible uncertainties involved in these data. However, the accumulated uncertainties of these data were not considered.

7.0 PACKAGE OPERATION

Unless otherwise stated, the following information is needed pursuant to the requirements of 10 CFR 71.87.

7-1 Specify the spent fuel parameter values to be verified and how they will be verified in 7.1.2.4 prior to loading them into the cask.

The values for spent fuel irradiation parameters (e.g., in-core boron concentration, moderator density, etc…) including cooling time and burnup need to be verified prior to cask loading. The method of verification needs to be included in 7.1.2.4.

7-2 Include qualification procedures of the storage cask content for transport.

The same type of verification outlined in RAI 7-1 for the contents should be included for loaded casks recognizing this would be post-loaded verification.

7-3 Provide an analysis to show the cladding temperature limits are not exceeded during the vacuum drying operation or helium backfill operation for the TN-40. The analysis should specify any time limits and additional controls in the operation to ensure cladding integrity.

The application should confirm that the fuel cladding temperature limits will not be exceeded during the vacuum drying operation or helium backfill operation.

This information is required by the staff to assess compliance with 10 CFR 71.33.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1.2 Structural and Pressure Tests

8-1 Revise the load test to apply 3.0 times the design lift load to the lifting trunnions.

As commented previously, for a general license, full compliance with the ANSI N14.6 standard to test the two trunnions to 3 times the design lift load is expected of the non-redundant lift design.

This information is necessary to show compliance with 10 CFR 71.35(a).

Section 8.2.4 Shielding

8-2 Explain how the condition of the solid neutron shielding material will be verified prior to each shipment.

The condition of the solid neutron shielding material needs to be verified prior to transport to ensure there is no loss or damage.

This information is necessary to show compliance with 10 CFR 71.71 and 71.73.

8-3.1 Clarify how the "Acceptance Tests and Maintenance Program" are performed on the casks that have been fabricated and already placed in long-term storage. Clarify which procedures are performed prior to shipment but after storage.

This information is necessary to show compliance with 10 CFR 71.35(a).

ROUTING AND TRANSMITTAL SLIP

Date: 4/3/08

NAME	INITIALS	DATE
M. Rahimi		
D. Tang		
R. Einziger		
J. Ireland		
M. Panicker		
A. Sotomayor		
Z. Li		
M. DeBose		
C. Regan		
M. Waters		
L. Campbell		
R. Nelson		

***E

DO/NMSS TICKET NO(s).:
DUE TO DIVISION:
DUE TO NMSS:
DUE TO EDO:

****

MEMORANDUM/ TO: Packaging Technology

FROM: M. Rahimi

SUBJECT: First round RAIs on TN-40 Transportation Package

****

REMARKS:

****

ADAMS: ML

****

Reminders (√ if included) or (□ if not included/not applicable):

ADAMS Form 665 must be attached to all packages if the document(s) are placed in ADAMS and are being provided to the NMSS Front Office for concurrence and/or signature.

NMSS Front Office Group must be given "owner" rights to the document(s) in ADAMS that are being provided to the NMSS Front Office for concurrence and/or signature.

Form 6 must be attached to the package if it is a Commission paper.

Classified/Safeguards Document(s) - Verify that the envelope in which the classified or safeguards document(s) is to be mailed in is addressed to the proper classified address.

****

ORIGINATOR: M. Rahimi PHONE: 492-3338
SECRETARY: PHONE: 492-3300