Non-proprietary Request for Additional Information
NAC International
Docket No. 71-9356
Certificate of Compliance No. 9356
Model No. MAGNATRAN® Transportation Package

By application dated November 26, 2012, as supplemented on February 15 and March 29, 2013, December 1, 2014, and January 13 and 21, 2015, NAC International (NAC) submitted an application for approval of Certificate of Compliance No. 9356, for the Model No. MAGNATRAN® transportation package. This request for additional information identifies non-proprietary information needed by the U.S. Nuclear Regulatory Commission staff in connection with its review of the application. The requested information is listed by chapter number and title in the applicant's safety analysis report (SAR). The staff used the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," in its review of the application.

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

Chapter 1 – General Information Evaluation

1. Provide the following information:

    a. Revise Section 1.3.2, "Contents" to ensure that the description of the contents is consistent with the rest of the SAR. Item 12.i. on page 1.3-28 states "Spent fuel content shall be loaded in accordance with the loading tables in Chapter 5, Section 5.8.10 and Table 5.8-49 of this SAR." Section 5.8.10, does not appear to contain any loading tables.

    b. Specification regarding application of the specific activity limit for greater than class C (GTCC) waste contents. While a package may be loaded with material that may have the allowable specific activity when averaged over the entire contents; this averaging allows potentially significant variations in specific activity in the contents. This variation could result in much higher dose rates than analyzed. Thus, the specific activity limits should include limits on potential variability of GTCC activity.

    c. Specification of replacement rods for boiling-water reactor (BWR) fuel as solid, unirradiated rods, or, with appropriate analysis, the applicant could provide a specification similar to that for pressurized- water reactor (PWR) fuel. The PWR content limits include a restriction on the number of irradiated steel replacement rods in an assembly, their exposure and cooling time, and the number of assemblies in the package with these rods. The current application only appears to support use of solid, unirradiated replacement rods for BWR contents.

    d. Maximum exposure and minimum cooling time limits for Combustion Engineering (CE) neutron source assemblies. No specifications are currently provided, and it is not clear that control element assemblies (CEAs) bound neutron source

assemblies (NSAs) since NSAs have steel/Inconel components in the active fuel zone whereas CEAs do not.

e. Clarification of the cooling times for PWR assemblies with irradiated steel replacement rods. It appears that PWR assemblies with irradiated steel replacement rods must be cooled an additional year, but also have minimum cooling times not less than 20 years.

f. Minimum density and hydrogen content for the neutron shield (Drawing Nos. 71160-502-3P and 71160-502-3NP). These specifications, together with the B_4C content limit (already included on the drawing), are important for the shield's performance.

g. Confirmation that damaged Combustion Engineering (CE) 16x16 fuel (fuel type 16a) is not an allowable package content. The calculation package 71160-5508, Revision 0, indicates that damaged CE 16x16 fuel is not an allowed content; however, the application does not clearly indicate one way or the other.

h. Clarification of the additional cooling time requirements for fuel in damaged fuel transportable storage canisters (DF TSCs). It appears that the additional cooling time applies to all assemblies (damaged and undamaged) in the DF TSC when damaged fuel is present and that this cooling time is, for undamaged assemblies, in addition to any cooling time that must be added for loading of non-fuel hardware.

i. Clarification that CE burnable poison rod assemblies (BPRAs) and thimble plug devices (TPDs) are not allowed contents. The shielding analysis (application Section 5.8.5) does not include CE BPRAs or TPDs as contents. However, per Section 5.8.10.2, Table 5.8-49 includes additional cooling time requirements for CE 14x14 assemblies with BPRAs or TPDs when these assemblies are loaded with damaged fuel in the DF TSC.

j. Clarification whether damaged fuel with enrichments as low as 1.3 weight percent are allowed contents. It is not clear that the damaged fuel shielding analysis covers these low enrichments.

k. Clarification that the added cooling times applied to DF TSC contents when damaged fuel is loaded also apply when high burnup fuel is loaded. Since high burnup fuel is treated as damaged fuel regardless of its actual physical condition at loading, the contents specifications should state that contents requirements for the DF TSC when loaded with damaged fuel also apply when the TSC is loaded with high burnup fuel.

l. Tolerances on the technical drawings for items relied on in the shielding analysis. It is not clear that the drawings include the tolerances for items important to shielding (e.g., thickness of the package's lead shielding, steel inner and outer shells, neutron shield assembly components, lid, and base). It is also unclear that the dimensions in the shielding analysis are consistent with the package's tolerances.

Shielding analyses should be modified, as necessary, to support the proposed contents specifications and package tolerances.

This information is needed to confirm compliance with 10 CFR 71.33 and 71.35(a).

Chapter 2 – Structural Evaluation

1. Revise the impact limiter dimension description on page 1.3-7 of the application.

   On page 1.3-7 of the application, the impact limiter dimension description is "81 inches deep." The impact limiter dimension description should be consistent with that depicted in Drawing No. 71160-531, Rev. 1P, "66 inches deep."

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

2. Revise the following statement in Section 2.6.7.5 of the application, as appropriate, to recognize the use of a half-symmetry finite element analysis (FEA) model for evaluating the package subject to a side drop event.

   Section 2.6.7.5 of the application states: "[T]he side drop simulations were performed using a quarter-symmetry model with a quarter of the package assembly in the finite element model…"

   The staff notes that the boundary conditions associated with the inertia effects of a side drop cannot be properly modeled using only a quarter of the package in a FEA model.

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

3. Provide NAC Calculation No. 71160-2121, Rev. 0, "Gap Between the Lead and Steel Shells of the Transport Cask."

   In its December 1, 2014, response to NRC's request for additional information (RAI), NAC introduces Revision 3 to Calculation No. 71160-2018, which changes the previously calculated lead slump of 2.33 inches to 0.87 inches at the top of the lead annulus. Section 6.1.5.3 of Calculation No. 71160-2018, Rev. 3, refers to Calculation No. 71160-2121, Rev. 0, for a calculation to which this new calculated lead slump is attributed. The subject calculation is not available for determining the shielding performance of the package associated with the package hypothetical accident condition free drop tests.

   This information is needed to confirm compliance with 10 CFR 71.73(c)(1).

Materials Questions

Questions to be provided by June 19, 2015.

Chapter 3 – Thermal Evaluation

1. Provide energy balance calculations on the personnel barrier to determine its maximum temperature.

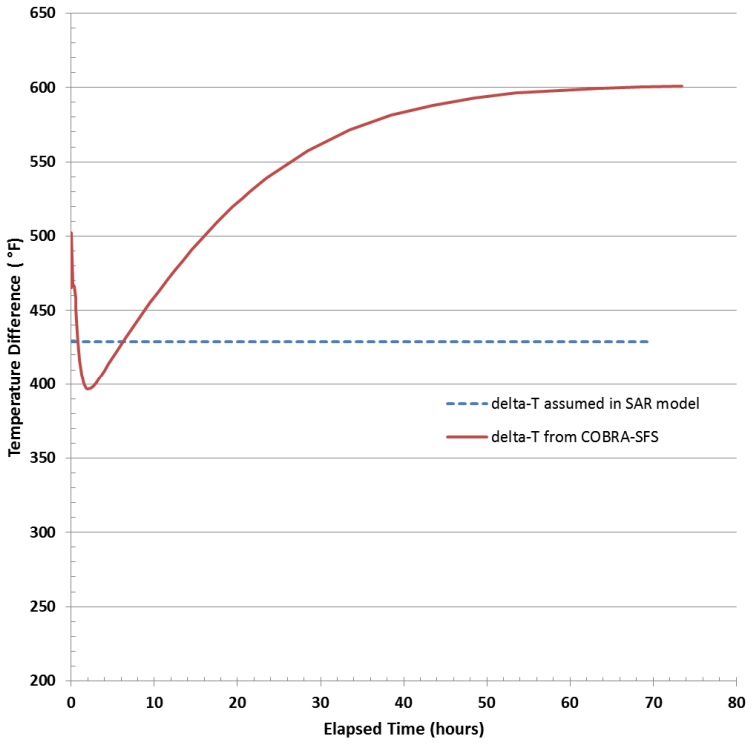
The NAC response to NRC's question 3-15 in letter dated January 13, 2015, stated that the personnel barrier temperature is an output from the FLUENT analysis. However, the reported temperature is a value at a certain location in the flowfield; the barrier was not modeled. Therefore, the reported temperature does not consider the influence of radiation heat transfer from the package's hot surface, etc. The energy balance calculation should be provided and assumptions used in the calculation justified, considering there is less than 20°F margin from the exclusive use, maximum temperature in the regulations.

This information is needed to confirm compliance with 10 CFR 71.43(g) and 10 CFR 71.71.

2. See Enclosure 1.

3. Modify the thermal analysis for hypothetical accident conditions to take into account the thermal inertia effects or justify that the thermal methodology for hypothetical accident conditions will generate accurate, or bounding, component temperatures, considering that the thermal inertia of the TSC and contents are not taken into account.

   The NAC response to question 3-21 and previous NRC's questions (letter dated February 28, 2013, see ADAMS Accession No. ML13059A711) indicated the staff's concern with the hypothetical accident conditions methodology and its results. Rather than model the thermal inertia of the TSC/contents, the non-physical methodology mixes separate analyses for normal conditions of transport and hypothetical accident conditions by relying on the addition of a "delta_T" parameter to the temperature of the package inner surface for normal conditions of transport. Contrary to the NAC response to NRC's question 3-7(f), there is potential for the current methodology to produce non-conservative results for content with certain masses. A confirmatory analysis, which included the thermal inertia effects, was used to understand the behavior of the "delta_T" parameter, with the result shown in the figure below. As can be seen in the figure, the NAC hypothetical accident conditions methodology results in a constant "delta_T" parameter throughout the hypothetical accident conditions transient, compared to the varying temperature parameter from the physical reality-based NRC confirmatory analysis. The results in the figure demonstrate the inadequacy of the hypothetical accident conditions methodology to model physical behavior and therefore, show there is uncertainty that the hypothetical accident conditions methodology produces accurate, or bounding, temperatures.



This information is needed to confirm compliance with 10 CFR 71.73.

4. Provide the pressure spreadsheet discussed in NAC's response to question 3-14 in letter dated January 13, 2015, and include/confirm the volume of tubes in the pressure calculation of Calculation Package 71160-3022, Rev. 1.

   a. The pressure calculation mentions a spreadsheet, but it was not included and therefore a review cannot be performed.

   b. Section 6.1.3.2 includes an equation for V_{tubes} that should be clarified/confirmed within the calculation package text and the spreadsheet; the equation does not appear to be correct.

   This information is needed to confirm compliance with 10 CFR 71.71 and 10 CFR 71.73.

5. See Enclosure 1.

6. Clarify the meaning of an "area-weighted" y+ value so that a review of the FLUENT fin model can be performed.

   NAC's response to NRC question 3-1 in letter dated January 13, 2015, states that the area-weighted y+ value of 0.55 is appropriate for the model. Confirm that all of the local grid/nodes have an appropriate y+ value, recognizing that y+ values should be determined locally at each grid node near the surface and not determined as an average over a surface.

This information is needed to confirm compliance with 10 CFR 71.35, 10 CFR 71.71, and 10 CFR 71.73.

7. Provide a more accurate package boundary condition heat transfer coefficient during the fire analysis that recognizes fins tend to augment heat transfer (including the contribution due to convection from the fire) or provide a bounding boundary condition.

   a. According to NAC's response to question 3-1 in letter dated January 13, 2015, the MAGNATRAN® fin effectiveness is 3.1. However, NAC's response to question 3-12 did not address the effect of fins, which augment heat transfer from the fire's forced convection, during the thermal test for hypothetical accident conditions. Rather, the hypothetical accident conditions thermal model relies on input from a reference that describes a thermal analysis for hypothetical accident conditions from a package that is unfinned. A more accurate heat transfer coefficient, or a bounding boundary condition (i.e., 800 °C temperature), should be modeled.

   b. Explain how the radiation thermal input from the fire is imposed on the model. It is understood that currently, a heat transfer coefficient boundary condition of 0.01833 Btu/hr in^2-°F is imposed during the fire.

   c. NAC's response to question 3-12 in letter dated January 13, 2015, indicates a 171 Btu/hr-in^2 average total peak flux. Define "average total peak" heat flux (average versus peak) and show how it is calculated and imposed on the model to achieve the current convection coefficient of 0.01833 Btu/hr- in^2-°F. Clarify whether it is the portion attributed only to convection effects.

   This information is needed to confirm compliance with 10 CFR 71.73.

8. Clarify the use of the half- and full-length models and correct the temperatures calculated for normal conditions of transport and hypothetical accident conditions, including Table 3.4-1, 3.5-1, and 3.5-2.

   a. NAC's response to question 3-2 in letter dated January 13, 2015, compares results from half-length and full-length thermal models. It appears that the full-length model has approximately 15% more insolation than the half-length model but has lower component temperatures. This does not seem reasonable considering that the O-ring temperature for the half-length model was at a lower temperature of 184°F). (Note: Provide the higher temperature referred to in RAI 3-21 response.) Likewise, although item 2 on page 3.4-6 mentions that the top portion is 30% longer than the bottom portion, "which provides more surface area for heat rejection to the ambient," it does not address that the area covered by impact limiters is modeled as adiabatic (page 3.4-3), indicating there is no heat rejection.

   b. Previously, a half-length model that did not include the lid and O-ring section was used to determine component temperatures. Section 3.4.1.1.1 in the SAR very briefly described a new full-length three-dimensional MAGNATRAN® ANSYS model. The full-length model resulted in an O-ring temperature during normal conditions of transport of 232°F, an increase from the half-length model value of

184°F. Except for the higher O-ring temperature, it appears that Table 3.4-1 lists temperatures from the half-length model. Explain the rationale for not relying on the full-length model results in the SAR that provided more accurate and bounding temperatures near the critical O-ring area and provide the component temperatures that use the full length model.

c. Explain whether the O-rings, which are part of the containment boundary, were explicitly modeled in the full length model; if not, explain how the O-ring temperatures were determined.

This information is needed to confirm compliance with 10 CFR 71.51, 10 CFR 71.71, and 10 CFR 71.73.

9. Provide a grid sensitivity analysis as part of the normal conditions of transport and hypothetical accident conditions thermal analyses in the SAR.

NAC's response to question 3-18 in letter dated January 13, 2015, indicates that a grid sensitivity analysis for the ANSYS model is unnecessary for an FEA model. Grid sensitivity analyses are often performed for FEA models and, in fact, Appendix F of Calculation Package 71160-3020, Rev. 3, provides a grid sensitivity study for the MAGNASTOR® ANSYS FEA model, which indicated a temperature dependency on grid resolution. Grid sensitivity should be analyzed, considering the relatively small temperature margin of peak clad temperature (PCT) to the allowable value for normal conditions of transport. NRC confirmatory analysis indicates that the PCT may be above 752°F.

This information is needed to confirm compliance with 10 CFR 71.71 and 10 CFR 71.73.

10. Confirm that there are no adverse consequences (e.g., containment, shielding, thermal effects of changing gap sizes, etc.) due to temperature gradients/thermal stresses that may occur during hypothetical accident conditions.

Section 2.7.4.3 and Section 3.5.5 in the SAR indicates that thermal stresses were not calculated for hypothetical accident conditions. The potential effects of temperature gradients (e.g., gap size changes due to different thermal expansion coefficients of inner shell steel and lead gamma shield, etc.) and thermal stresses on packaging performance must be considered.

This information is needed to confirm compliance with 10 CFR 71.73.

11. Clarify and correct how gaps were modeled in the thermal ANSYS calculations (normal conditions of transport, hypothetical accident conditions, etc.) so that a review of the analyses can be performed. Include, in the ANSYS models, a more accurate, or bounding, representation for the contact area between the basket and canister shell, contact area between canister shell and the package, and the thermal conductivity between those gaps.

Gaps within a thermal model have an impact on thermal results. NAC's response to question 3-10 in letter dated January 13, 2015, did not adequately address the following, which would have an effect of component temperatures, including PCT:

a. Provide results that confirm the four “contact locations,” and the area between them, shown in the Figure 3-10.1 (NAC response to question 3-10), pictorial actually represent physical contact between the separate components. It should be understood that an “arc of contact” does not necessarily represent a contact area.

b. NAC’s response to question 3-10, page 10 of 37 of Calculation package 71160-3014, Rev. 0, indicates that applying the helium thermal conductivity to gaps within the package would be “overly conservative.” However, SAR Section 3.2.2.3 states that the gas conductivity of the gap is used (which is not the 0.8 or 2 Btu/hr-in-°F imposed value). There is no basis for assuming a thermal conductivity of 0.8 Btu/hr-in-°F for gaps listed as 0.03 inch (or larger, some of which were listed as 0.28 inch)and therefore, these gaps should be modeled using the thermal conductivity of the surrounding/backfill gas.

c. Provide the calculations that show how the thermal conductivities were determined. For example, there was no calculation showing the basis for choosing a contact thermal conductivity of 0.8 Btu/hr-in-F between the stainless steel components, considering that stainless steel can have a thermal conductivity of 0.867 Btu/hr-in-F and there would be some form of contact resistance between different components.

d. Provide further details associated with the calculation that showed a 1°F change in temperature when thermal conductivity changed from 2 Btu/hr-in-F to 0.8 Btu/hr-in-F.

e. There are many instances in Calculation Package 71160-3014, Rev. 0 (page 14, 15, etc.) in which gaps are modeled by “effective properties” that are “adjusted” by “factors” (0.2044, etc.). There is no clear discussion or calculation that explains the rationale or procedure for modeling a gap with effective properties or applying an adjustment factor; the discussion in SAR Section 3.2.2.3 is very limited and there is nothing showing the relative contribution to conduction and radiation heat transfer in the presented results. Based on the limited description and presented results, gaps should be modeled with the surrounding/backfill gas thermal conductivity, especially because the radiation heat transfer component would be small.

f. NAC’s response to question 3-7(a.2) in letter dated January 13, 2015, indicates an assumed gap between the lead gamma shield and the steel body of the transportation package is conservative. This may be conservative for the cooldown phase of the fire test for hypothetical accident conditions, but it would not be conservative during normal conditions of transport because the higher temperature on the inner face and the small surface area would tend to over-estimate the heat transfer across the gap compared to putting the gap on the outer surface. This may have an effect on results; a sensitivity study should be performed considering the relatively small temperature margin of PCT during normal conditions of transport.

g. Include a calculation that describes the intent of Calculation Package 71160-3014, item 10 on page 15 of 37 (e.g., why 75% of helium thermal properties). Based on the limited description, it would appear that gaps should be modeled with the helium thermal conductivity.

h. NAC's response to question 3-10 in letter dated January 13, 2015, discusses the contact arc. However, it is the actual area of good contact (basket-to-shell and canister-to-shell) that affects heat transfer. It is difficult to understand how "contact" can be assumed for gaps that are 0.03 inch and larger. Provide information on the following contact areas that were used in the thermal analysis in the SAR:

   1. The linear contact length in the thermal model at each location (as illustrated in Figure 3-10.1 and Figure 3-10.2) in the basket cross-section where it is assumed that there is perfect contact between the basket and the canister shell.

   2. The axial length that the perfect contact between the basket and the canister shell is assumed to exist over in the thermal model.

   3. The total contact area assumed at each location in the thermal model where it is assumed that there is perfect contact between the basket and the canister shell (Note: this is a redundant check on the information in (1) and (2) above; just for completeness.)

   4. The linear contact length in the thermal model between the canister outer shell and the packaging inner liner for the package cross-section.

   5. The axial length in the thermal model that contact between the canister shell and the packaging inner liner is assumed to exist over in the thermal model.

   6. The value(s) of thermal conductivity used to represent contact conductance between elements assumed to be in contact, for the basket-to-shell contact region and the canister-to-shell package region.

i. The diagrams in Figure 3-10.1 and 3-10.2 are based on what appears to be an optimum orientation of the basket within the canister shell, relative to the assumed direction of the force of gravity. That is, this orientation appears to be one that would produce the maximum contact area. Is it expected that orientation of the canister within the package will be controlled to the degree that it can be ensured that this orientation will be maintained in all transport configurations? If not, a sensitivity study is needed to show the effect of canister orientation on contact area. The minimum contact area should be used in the thermal model, to ensure conservatism in the predicted canister component temperatures, including PCT.

This information is needed to confirm compliance with 10 CFR 71.71, and 10 CFR 71.73.

12. Provide Calculation Packages 71160-3001, Rev. 3, and 71160-3002, Rev. 1, so that a review of the methodology and assumptions for the effective conductivity modeling can be performed.

    NAC's response to question 3-3P in letter dated January 13, 2015, provided the effective thermal conductivity curves for the PWR and BWR fuel, which are different from expectations and require further review. In addition, specify whether the poison plates were homogenized with the fuel assemblies.

    This information is needed to confirm compliance with 10 CFR 71.35.

13. Justify the use of an average heat transfer coefficient correlation that is applied to the outer package surface during cooldown for both normal conditions of transport and hypothetical accident conditions thermal evaluations.

    NAC's response to question 3-1 in letter dated January 13, 2015, and SAR Section 3.2.3, indicates that an average heat transfer coefficient correlation is applied as an outer package boundary condition. An average heat transfer correlation generally uses the average temperature over the particular geometry/length-scale of the correlation rather than the local temperature. The local heat transfer correlation would tend to be used when local temperature conditions are applied; the local correlation can be much less (e.g., 25% to over 30%) than the average heat transfer correlation. This reduction in heat transfer coefficient can be important considering the relatively small margin of the PCT to the allowable value for normal conditions of transport.

    This information is needed to confirm compliance with 10 CFR 71.35, 10 CFR 71.71, and 10 CFR 71.73.

14. Clarify the areas associated with the 2D unfinned model and 3D finned model described in the NAC's response to question 3-1 in letter dated January 13, 2015, Appendix H Calculation Package 71160-3045 so that a review can be performed.

    a. The model outer areas are listed in the second table of Appendix H of Calculation Package 71160-3045. Clarify whether the 2D half-symmetry, unfinned geometry models only one-half of the neutron shell. It is not clear how the listed area was determined.

    b. The 3D finned model area (0.7132 m^2) in the table appears to be based on 360 degrees. However, the model is 180 degrees. It is not clear how the listed area was determined.

    This information is needed to confirm compliance with 10 CFR 71.35 and 71.71.

15. Provide the two experimental test reports (GA-A20770 and GA-A19897) discussed in NAC's response to question 3-9 and question 5-6P, in letter dated January 13, 2015.

    NAC's responses refer to experiments performed on the NS-F-4 neutron absorber at high temperatures, but it was not certain whether the test conditions corresponded to those for the thermal test for hypothetical accident conditions (800°C for 30 minutes),

and therefore, it is uncertain whether the results confirm the behavior of the neutron absorber during the 30 minute fire.

This information is needed to confirm compliance with 10 CFR 71.73.

16. Update the thermal models or demonstrate that the emissivity and absorptivity values used in the thermal models (SAR text and ANSYS input files) are realistic/bounding so as to provide accurate/conservative results.

According to NAC's response to question 3-18(d), the emissivity values listed on SAR pages 3.2-4 through 3.2-8 are also used as absorptivity values. Although this can be appropriate when dealing with spectral properties, it may not be valid when applying average emissivity and absorptivity values. For example, as noted in the response to question 3-18(d), a relatively low stainless steel emissivity would lessen the radiation heat transfer rate out of the package for normal conditions of transport. However, dust or surface coatings would tend to increase the absorptivity beyond a 0.36 value. A higher absorptivity would increase the insolation thermal input, which is a relatively large component, as noted in Table 3-2.1 and Table 3-2.2 (response to question 3-2).

This information is needed to confirm compliance with 10 CFR 71.71 and 71.73.

Chapter 4 – Containment Evaluation

No Questions

Chapter 5 – Shielding Evaluation

1. Demonstrate that the:

   a. radial dose rates at the personnel barrier surface are less than the 200 mrem/hr limit, and

   b. radial dose rates at the appropriate locations (i.e., package surface, personnel barrier surface, and 2 meters from the vehicle edge) meet their respective limits, accounting for the impacts of the alternating 2-inch and ¼-inch void gaps at the top of the neutron shield assemblies.

Maximum surface dose rates on the package radial surface above the neutron shield are significantly higher than 200 mrem/hr for both undamaged and damaged fuel contents. While these dose rates are lower than the limit for a package surface when the package is in an enclosure (i.e., personnel barrier), they cannot be relied on to show that the dose rates at the enclosure surface meet the dose rate limit of 200 mrem/hr.

Additionally, the mesh tally analysis in Section 5.8.12 of the application indicates that void areas in the neutron shield can result in significant radiation streaming. However, this analysis is limited to the thermal insulation and expansion foam areas near the neutron shield mid-plane area. Based on this analysis, the 2-inch tall void areas at the top of some of the neutron shield assemblies would also be areas of significant radiation streaming that, due to their size, could have a significant impact on package dose rates at the locations listed in b above. Since not all neutron shield assemblies have this size

of void in their top end, this effect is not adequately captured by surface detectors that average the dose rate around the package circumference. Maximum radial dose rates for damaged fuel and undamaged fuel contents occur near this area of the package and so may be influenced by this azimuthal variation in the top ends of the neutron shield assemblies. Thus, the shielding analysis should be modified to address this effect using detectors of reasonable size both axially and azimuthally (e.g., 3 inches).

For both items a and b, above, the analyses should appropriately capture other azimuthal effects (e.g., spent fuel contents with specific characteristics being limited to specific basket locations) and include both spent fuel and non-fuel hardware contributions for both damaged fuel and undamaged fuel contents.

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

2. Provide support for the statements regarding high burnup fuel and the analysis approach used for high burnup fuel described in Section 5.2 of the application, both in terms of radiation source and heat load.

   Section 5.2 of the application cites NUREG/CR-6701, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel," specifically Appendices A and B. The information in those appendices and the nuclides considered there are important to criticality safety. Also, Appendix A states, in part in Section A.8, that the "sensitivity studies described [in Appendix A] only assess the influence of the underlying data parameters on the concentrations of the actinides and fission products as they [affect] the k of a spent fuel system." The same nuclides are not necessarily important to the radiation source term and the heat load, and the appendices' applicability as a basis for the proposed approach is unclear. It is also unclear that the appendices provide the discussions and conclusions described by the applicant.

   It is also unclear that the applicant's approach results in any margin with respect to the shielding analysis and dose rates since the applicant changed its analysis method in response to the staff's previous request for information. Bounding dose rates are now calculated with high burnup fuel having the burnup, enrichment, and cooling time specifications in Table 5.8-16 that are at the reduced heat load limit for high burnup fuel.

   This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

3. Revise the analysis to address the condition of short PWR assembly types being loaded into the long TSC.

   The current analysis for PWR assemblies appears to be based on PWR assemblies being loaded into the short TSC, with the exception of the CE 16x16 assembly type. However, per Section 1.3.2 of the application all PWR assembly types can be loaded into the long TSC. The analysis does not appear to address this condition. A look at the input files provided with calculation package 71160-5508 shows that the PWR assemblies (excepting the CE 16x16 assemblies) were analyzed in the short TSC and not in the long TSC. Thus, it is not clear that the dose rates reported for PWR assemblies are bounding and meet the dose rate limits at the regulatory locations (package surface, personnel barrier surface and 2 meters from the vehicle edge). With

short PWR assemblies in the long TSC, the upper hardware zones of these assemblies can now extend above the neutron shield and the lead shield since neither of these shields extends the whole length of the package cavity. Also, the fuel region can be closer to the top ends of the neutron shield and lead shield and so be less shielded versus when in the short TSC. The revised analysis should also account for the uncertainties in the method. The staff's evaluation of the uncertainties arising from the applicant's analysis method (e.g., due to assumptions and simplifications) indicates that dose rates for the currently analyzed configuration may have little margin to the personnel barrier surface and 2-meter dose rate limits.

This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

4. Justify how the assembly response functions can be used for estimating dose rates from BPRAs and TPDs if credit is taken for the masses of these non-fuel hardware (NFH) types, or modify the analysis appropriately.

   The discussion of response functions for these NFH types remains unclear. The change to Section 5.8.5.1 of the application did not adequately address this issue. Differences in material properties in the models (NFH present vs. not present) result in different response functions. Thus, to take credit for the NFH mass, different response functions would need to be used. Otherwise, this implies that dose rates for assemblies without NFH are calculated with response functions that include the masses of the NFH, which would be incorrect.

   This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

5. Modify the shielding analysis chapter to correct errors and inconsistencies within the chapter, as well as inconsistencies between the shielding chapter and the shielding calculation package and its associated files.

   There are many inconsistencies and errors in the shielding chapter and between the shielding chapter and the calculation package. These items make it difficult to understand the application. They include differences in reported dose rates and bounding source term specifications, and incorrect model figures (e.g., the neutron shield assemblies figure has incorrect dimensions). These errors and inconsistencies are within the chapter (differences between text, figures, and tables) and between the chapter and the calculation package (text, figures, and tables). The GTCC sample input file should be updated to show the use of new surface detector divisions (for capturing axial lead slump, etc.). Also, revise the shielding analysis to change "doses" to "dose rates," as only dose rates are calculated.

   This information is needed to confirm compliance with 10 CFR 71.33 and 71.35(a).

Chapter 6 – Criticality Evaluation

1. See Enclosure 1.

2. See Enclosure 1.

3. Revise the application to ensure that absorbers in the SCALE/TRITON model are depleted by constant flux as opposed to constant power.

   In the sample TRITON depletion input files provide for fuel (Figure 6.10.1-8) and for reactor control cluster assemblies (RCCAs) (Figure 6.10.1-20), it appears that Wet Annular Burnable Absorbers (WABA) and RCCA materials are depleted using constant power. Section 6.3 of NUREG/CR-7041 states:

   > *"While depletion assuming a constant power is generally representative of a fuel element, this is not true for nonfuel elements. Targets, structural materials, and burnable poisons are generally "driven" by fluxes from neighboring fuel elements and do not contribute significantly to power production. Forcing a constant power in a burnable poison rod, for example, would result in a rapidly increasing flux due to depletion of the poison material, which is nonphysical. Allowing mixed-mode depletion within SCALE/TRITON better approximates the time-dependent flux behavior across an assembly."*

   The application should ensure that absorber materials, including WABAs, RCCAs, and any other burnable absorbers or control elements, are modeled appropriately in all depletion models.

   This information is needed to confirm compliance with 10 CFR 71.55 and 10 CFR 71.59.

4. Revise the application to clarify how the burnup uncertainty is incorporated into the determination of maximum initial enrichment for the PWR loading curves.

   Burnup credit loading curves are typically constructed so that minimum burnup is a function of known initial enrichment, and the package user compares reactor burnup record values to the required minimum burnup. The burnup value to be used for loading acceptance is the reactor record value reduced by the uncertainty in that value (typically two times the in-core measurement uncertainty). For the MAGNATRAN® system, the applicant proposes PWR loading curves where the allowable initial enrichment is determined based on a known burnup value. It is not clear how burnup uncertainty is taken into account in this case.

   This information is needed to confirm compliance with 10 CFR 71.55 and 10 CFR 71.59.

5. Revise the application to include administrative loading procedures to prevent misloads in the PWR and PWR-DF canisters for contents evaluated with burnup credit.

   A misload analysis should be coupled with additional administrative procedures to ensure that the transportation system will be loaded with fuel that is within the specifications of the approved contents. Procedures considered to protect against misloads in transportation packages that rely on burnup credit for criticality safety may include:

   - Verification of the location of high reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool both prior to and after loading,
   - Qualitative verification that the assembly to be loaded is burned (visual or gross measurement),

- Verification, under a 10 CFR Part 71 quality assurance program, of the canister or package inventory and loading records prior to shipment for previously loaded systems,
- Quantitative measurement of any fuel assemblies without visible identification numbers,
- Independent, third-party verification of the loading process, including the fuel selection process and generation of the fuel move instructions, and
- Minimum soluble boron concentration in pool water, to offset potential misloads during loading and unloading.

This information is needed to confirm compliance with 10 CFR 71.55 and 10 CFR 71.59.

6. Revise the application to ensure that all calculated k_{eff} values for the MAGNATRAN® package are below the calculated Upper Safety Limit (USL).

   Tables 6.10.3-1 and 6.10.3-2 appear to show $k_{eff} + 2\sigma$ values that are above the system USL of 0.9376 for some BWR fuel in the MAGNATRAN® system with the 0.027 g/cm^2 areal density neutron absorber plates. Revise the application to demonstrate that the maximum calculated $k_{eff} + 2\sigma$ values for all fuel and canister types are below the system USL.

   This information is needed to confirm compliance with 10 CFR 71.55 and 10 CFR 71.59.

7. Revise the application to clarify the PWR fuel depletion assumptions regarding Control Element Assembly (CEA) insertion during irradiation.

   Table 6.10.1-4 of the application lists several irradiation parameters assumed in the PWR depletion analysis, including assumptions regarding CEA insertion. For "CE Assembly Type CEA," the table gives the analyzed parameter as "Full Insertion," implying that CEAs were assumed to be fully inserted for the full burnup credited. However, this assumption is not discussed anywhere in the associated text of Section 6.10.1. The text of this section should be revised to clarify the CEA exposure assumption for CE fuel types.

   Additionally, for "B&W and WE CEA," Table 6.10.1-4 gives the analyzed parameter as "≤ 20 cm insertion and/or ≤ 10 GWd/MTU Full Insertion at Power." While the 20 cm insertion parameter is discussed in the text of Section 6.10.1, the assumption of 10 GWd/MTU full insertion during irradiation is not. The application should be revised to clarify the assumptions regarding B&W and WE CEA insertion, and provide supporting analyses demonstrating that the 10 GWd/MTU CEA insertion assumption is appropriate.

   This information is needed to confirm compliance with 10 CFR 71.55(b).

Chapter 7 – Operating Procedures Evaluation

1. Provide a verification in Chapter 7, "Package Operations," to ensure that accessible package surface temperatures will not exceed the limits specified in 10 CFR 71.43(g).

The thermal analyses indicate that surface temperatures are near the exclusive use shipment temperature limit; a test to determine surface temperature prior to shipment will help ensure that surface temperatures will remain below the allowable value.

This information is needed to confirm compliance with 10 CFR 71.43(g) and 71.87(k).

2. Clarify and confirm that the allowable loading and unloading time periods presented in Chapter 7 accurately reflect the current MAGNATRAN® design.

   Chapter 7 (including Sections 7.1.2, 7.1.4, 7.2.2, etc.) updated SAR pages include various loading and unloading time periods. For example, Section 7.1.2 lists a 44 hour time limit for loading into the MAGNATRAN® package. Page O17 of O18, Appendix O of Calculation Package 71160-3020, dated December 2010 lists a 44 hour time period, but it is based on a transfer cask with air transfer, according to Appendix L of Calculation Package 71160-3020. The applicability/relevance of certain Calculation Package 71160-3020 analyses to the MAGNATRAN® loading/unloading operations is uncertain. Confirm that the analyses presented in Calculation Package 71160-3020 were part of previously approved MAGNASTOR amendments, recognizing that the MAGNATRAN contents are not identical to the MAGNASTOR contents and to ensure that the MAGNASTOR thermal analyses are valid for the MAGNATRAN.

   This information is needed to confirm compliance with 10 CFR 71.35 and 71.71.

3. Modify the package operations to address the following:

   a. Section 7.1.5, steps 10 and 14 include options that are not supported as options in the technical drawings to be incorporated into the certificate of compliance.

   b. Section 7.1.1, Step 5 note incorrectly refers to 71.87(i) and 71.47 for an empty package.

   Section 7.1.5, step 10 includes a statement that the TSC closure ring is optional, and Drawing No. 71160-785, Rev. 2, includes a note stating the same thing. However, Drawing No. 71160-500, Rev. 4P, includes a note that states that GTCC TSCs without closure rings are not transportable.

   Additionally, Section 7.1.5, Step 14 includes a statement that the outer vent and drain port covers are optional as well as an option to use both the inner and outer cover plates and only weld the outer cover plates. While the notes in Drawing No. 71160-785, Rev. 2, support the outer cover plates being optional, the notes in Drawing No. 71160-500, Rev. 4P, do not. Also, the notes in Drawing No. 71160-785, Rev. 2, do not appear to support the option for including both inner and outer cover plates with only the outer cover plates being welded.

   Lastly, the appropriate reference for contamination and dose rate limits for an empty package is 49 CFR 173.428.

   This information is needed to confirm compliance with 10 CFR 71.87.

4. Clarify or remove the statement on page 7.1-1 which states that the procedures for the loading, closure, and preparation for transport of a TSC are outside the scope of the application.

   Page 7.1-1 states "*The procedures for the loading, closure and preparation for transport of a TSC, and for the removal of a loaded TSC from storage in a MAGNASTOR concrete cask (CC) at an ISFSI, are provided in the MAGNASTOR FSAR. Those procedures are outside the scope of these MAGNATRAN operating procedures.*" The procedures for loading a TSC and preparation for storage are outside the scope of the application, but procedures for loading a TSC and preparation for transport are within the scope of the application. Thus, the MAGNATRAN® operating procedures should ensure that the TSC is loaded, closed and prepared for transport in accordance with the MAGNATRAN® package design.

   This information is needed to confirm compliance with 10 CFR 71.87.

5. Revise page 7.1-5 to state that the maximum assembly average burnup for fuel assemblies to be transported off-site as undamaged is limited to ≤ 45,000 MWd/MTU.

   Page 7.1-5 appears to contradict itself by saying that spent fuel assemblies to be transported must have a maximum assembly average burnup less than 45,000 MWd/MTU. The next sentence states that a fuel assembly for transportation that has a maximum assembly average burnup greater than 45,000 MWd/MTU but less than 60,000 MWd/MTU shall be transported as damaged fuel.

   This information is needed to confirm compliance with 10 CFR 71.87.

Chapter 8 - Acceptance and Maintenance Tests Evaluation

1. Modify the shielding acceptance test in Section 8.1.6.3 (and the equivalent maintenance test in Section 8.2.3) to compare the measured dose rates to estimated (i.e., calculated) dose rates for the specific contents loaded in the package for this test, with these estimated dose rates being the acceptance criteria.

   The shielding effectiveness tests for both the acceptance tests and the maintenance program tests appear to use the regulatory dose rate limits for acceptance criteria. While meeting these limits demonstrates the particular shipment meets the regulations, it does not adequately demonstrate that the shielding performs as designed. The appropriate criteria for demonstrating the shielding performs as designed are estimated (i.e., calculated) dose rates that are representative of either a check source(s) that may be used for the test or for the loaded contents (e.g., assembly type, enrichment, burnup, cooling time), if the test is done with a loaded package.

   This information is needed to confirm compliance with 10 CFR 71.85 and to ensure the package maintenance activities are performed in a way that assures continued compliance with the requirements in 10 CFR Part 71, Subparts E and F, during its service life.