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, 2013, 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. Clarify that the personnel barrier extends the entire length between the package impact limiters.

   Figure 1.2-2 appears to show that the personnel barrier does not prevent access to the entire package surface between impact limiters. It appears that the upper flange area is exposed and accessible. In order to credit the personnel barrier, it must prevent access to all portions of the package body between the impact limiters.

   This information is needed to determine compliance with Title 10, *Code of Federal Regulations* (10 CFR) 71.35(a).

2. Revise Section 1.3.2 to provide the following with regard to the proposed package contents:
   a. Clarify whether the maximum blanket length of 6 inches applies to low enriched and annular blankets as well as un-enriched blankets.
   b. Clarify that only un-irradiated non-fuel hardware (NFH) is allowed to be loaded with pressurized-water reactor (PWR) fuel into damaged fuel cans (DFCs).
   c. Clarify the specifications of the proposed greater than class C (GTCC) waste contents in item 7 on page 1.3-25 to indicate these contents are limited to activated and surface contaminated steel and plasma cutting debris.
   d. Justify the use of the ^{60}Co curie limits in Tables 1.3-17 and 18 as an alternative to the irradiation exposure (i.e., burnup) and cooling time limits in the same tables for defining acceptable NFH contents (see 11.i and 12.l).
   e. Clarify that un-enriched rods that are used as replacement rods must be un-irradiated (see 11.k, 12.n, and 13), that any other type of replacement rods must be un-irradiated solid non-fuel rods (see 11.i, 12.l, and 13.n), and that steel rods used to displace guide tube 'dashpot' water must be un-irradiated (see 11.i and 12.l).
   f. Change references to Figure 1.3-6 to Figure 1.3-4 for item 12, since item 12 deals with the basket depicted in Figure 1.3-4.

g. Revise item 12 to specify the appropriate decay heat and assembly length limits. It appears that the high burnup fuel decay heat limit is 590.5 W and that the axial length inside of the transportable storage canister (TSC) containing damaged fuel will not accommodate 178.3-inch long assemblies.
h. Clarify which TSCs use burnup credit and ensure the contents specifications for those TSCs include the appropriate loading pattern requirements to use burnup credit (burnup credit loading patterns are specified for the TSC with a damaged fuel basket assembly). Also, ensure that the loading patterns specified in the items 12.j and 13.k for partially-loaded TSCs (not to address burnup credit) are appropriate. It appears that the loading patterns should be symmetric about the TSC's axis to balance the load, but some of these patterns are not (see 12.j and 13.k).
i. Clarify the appropriate tables for 13.b to include Table 1.3-22 and not Table 1.3-20.
j. Clarify the maximum decay heat per storage location; the value in 13.c does not match the value in Table 1.3-19 or Table 5.8.41.
k. Clarify the maximum assembly weight (including channels and spacers); the value in 13.f differs from the value in Table 1.3-19.
l. Clarify whether the loading pattern for partially loaded boiling-water reactor (BWR) baskets should be the same for items 13.i and 13.k.
m. Clarify whether the maximum decay heat per fuel location includes the contribution from NFH.
n. Provide minimum enrichments in Tables 1.3-6 and 1.3-19 for the PWR and BWR contents that are supported by the analyses; based on the shielding analysis, it appears that the minimum allowable average enrichment should be 2.1 weight-percent.
o. Clarify that no interpolation is allowed for the cool time and burnup (i.e., maximum exposure) limits in 11.l and 12.o.

These clarifications are sought to ensure the proposed contents are supported by the analyses in the application. If the proposed contents descriptions differ from the requested clarifications, the applicant should ensure the analyses support its proposals. Appropriate changes to figures and tables should also be made.

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

3. Revise the content qualification in 12.d (see pg. 1.3-28 of the application) to specify that fuel with burnup greater than 45,000 MWd/MTU (i.e., high burnup fuel) must be treated as damaged fuel in determining the fuel qualifications for loading with respect to criticality (loading curve), shielding, and thermal; in other words, that high burnup fuel is always considered as damaged fuel regardless of its actual physical condition at the time of loading.

   The application indicates that high burnup fuel must be loaded and handled in a damaged fuel can. The contents specifications, or qualification criteria, are different depending upon whether or not damaged fuel is loaded in the TSC in some respects. For example, the limits on enrichment, decay/cooling time, and burnup credit loading curves are different, as indicated in 12.b and 12.c on pages 1.3-27 and 28 of the application. Since the cladding performance of high burnup fuel during transportation is not well understood, the proposed solution is to treat high burnup fuel as damaged in the MAGNATRAN spent fuel transportation package and load all high burnup fuel into DFCs. Therefore, the fuel qualification should clearly specify that high burnup fuel should always be treated as damaged fuel to avoid confusion and misloading of the package.

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

4. Modify the contents descriptions, including the maximum mass of uranium, for the assembly types in Tables 1.3-7 and 1.3-20 to be consistent with those evaluated in the application.

   The maximum mass of uranium per assembly is an important parameter for shielding analyses and the values given in Tables 1.3-7 and 1.3-20 exceed those used in the shielding analyses for the hybrid fuel assembly classes that relate to the assembly types listed in these tables. Thus, the shielding analyses do not support the proposed contents. The applicant should also describe the correlation between the assembly types in these tables and the hybrid assembly classes used in the shielding analysis. In correlating the assembly types, the applicant should ensure that all specifications, in addition to maximum uranium mass, are consistent. For example, in its review of the assembly type specifications, the staff found that several specifications described in the PWR and BWR source calculation packages are inconsistent with those provided in the application (e.g., Tables 1.3-7 and 1.3-20).

   This information is needed to determine compliance with 10 CFR 71.33(b), 10 CFR 71.47 and 10 CFR 71.51.

5. Revise the proposed drawings to address the following:
   a. Drawing No. 71160-500 should include a GTCC waste TSC.
   b. Drawing Nos. 71160-684 and 71160-685 should include a shield collar for both the vent and drain ports on the TSC lid.
   c. Provide the material composition and vendor specifications of all four O-rings listed on drawing 71160-505, sheet 1 of 4. These are only specified as metallic and ethylene propylene diene monomer (EPDM) rubber O-rings in the drawings. Per Appendix 4.5.2 the maximum rated temperature has a wide variation depending on the materials used for the metallic O-rings. Drawing No. 71160-505 (sheet 1 of 4) lists the metallic O-ring and EPDM O-ring as commercial, but there are many different construction and materials used in the metallic O-rings that are "commercial" and there are many different compositions of EPDM with different service specifications. More details should be provided on these components to assure they work properly within the specified temperature range.

   The changes are needed to help understand the proposed package design and to ensure the analyses are adequate to demonstrate that the design meets the regulatory requirements.

   For item b, Drawing Nos. 71160-684 and 71160-685 show a shield collar for only one of the TSC lid ports whereas the drawings (e.g., Drawing No. 71160-584), show a shield collar for both lid ports. It is not clear why some TSC lids require shield collars for each port and others do not. Also note that Drawing No. 71160-785 may also need to be updated in this regard since it uses the same lid design from Drawing No. 71160-584.

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

6. Confirm that the short spent fuel basket cannot be used in the long TSC, or modify the analyses to address this configuration.

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

7. Clarify the package components' safety classification.

   The package components' safety classification should be provided to ensure proper quality assurance during design, fabrication, and inspection.

   This information is needed to determine compliance with 10 CFR 71.37.

Chapter 2 – Structural Evaluation

1. Revise the lead paragraph in Section 2.6.7.5 to recognize that the parametric studies results in Section 2.12.2.3.14 on effects of shallow-angle package drops are determined to be bounded by the side drop.

   The description of evaluating the package for the most damaging drop test orientation is incomplete and appears misleading as to the test conditions being examined to determine compliance with 10 CFR 71.73(c)(1).

2. Revise the second paragraph on page 2.6.7.5-5 and resubmit the balsa wood static stress-strain curves used in the LS-DYNA finite element analysis of the impact limiter to substantiate the statement, "[S]tress-strain curves are developed in the parallel and perpendicular-to-the-grain directions for hot (200^{o}F) and cold (-40^{o}F) conditions."

   A number of data reduction steps are briefly discussed in the application for the generation of the stress-strain curves suitable for defining the wood finite element material properties model. For clarity, a typical set of these curves should be presented in the application to facilitate staff review of how the materials properties are implemented in the LS-DYNA modeling of the impact limiter performance under the free drop conditions.

   This information is needed to determine compliance with 10 CFR 71.71(c)(7) and 10 CFR 71.73(c)(1).

3. Explain how the volumetric strain/stress values of Table 2.6.7-31 are converted to the uniaxial stress/strain values similar to those of Table 2.6.7-32 or vice versa for generating the stress-strain curves such as those described in page 2.6.7.5-5 for the LS-DYNA modeling of the impact limiter wood properties.

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

4. Revise Table 2.6.7-33, "Balsa Wood Stress-Strain Properties (Static)," to:
   a. Include the balsa wood stress-strain properties data similar to those of Table 2.6.7-32 for the redwood tested at an applicable strain rate. It's unclear why the strain-rate

dependent stress-strain properties data for the balsa wood is not reported in the application.

b. Verify that the compressive stresses for the hot (perpendicular to grain) condition are correctly reported in the table.

c. Explain why the stresses corresponding to the strain up to 0.4 are increasingly higher than that of 173 psi at a strain of 0.1.

The stresses listed for the perpendicular-to-grain direction are more than one order of magnitude larger than those typically observed. Also, at a strain between 0.1 and 0.4, the balsa wood crush stresses have generally been observed to increase only slightly, which are in contrast markedly to the listed data.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (7) and 10 CFR 71.73(c)(1).

5. Revise Section 2.7.13 of the application to provide a fuel basket stability evaluation for the PWR damaged fuel basket.

   For completeness, a geometric stability evaluation of the PWR damaged fuel basket is needed.

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

6. Revise Section 2.12.2.3 to recognize, as appropriate, that there were two series of scale-model impact limiter tests performed for the Model No. NAC-STC transportation package (Docket No. 71-9235). The revision should include:

   a. ascertaining the use of balsa wood LS-DYNA material models for individual scale-model tests used in benchmarking modeling approaches for specific drop orientations and,

   b. modifying and annotating the sketches in Figures 2.12.2-6 through 2.12.2-8 to depict the applicable impact limiter designs/orientations tested.

   Figures 2.12.2-6, -7, and -8 appear to have depicted two different impact limiter designs, the NAC-STC and NAC-STC-CY impact limiters, for which scale-model drop test data were used to benchmark the modeling approaches to calculate package rigid body response for the side- and end/corner-drop events, respectively. However, starting with “Introduction” in the section, there is a lack of clarity in describing which parts of the two tests were considered in the benchmarking associated with the balsa wood material performance being represented by LS-DYNA material model Mat_Honeycomb or Mat_026. Specifically, in Figure 2.12.2-6, sketches should be properly annotated to depict that only the side- and slapdown-drops of the NAC-STC impact limiter are considered for model benchmarking, which would be used for analyzing the MAGNATRAN package side-drop accident. Conversely, if the NAC-STC-CY impact limiter scale model was only tested for the end and corner drops, sketches similar to those of Figure 2.12.2-6 on drop orientations should also be presented to facilitate staff review.

   The information is needed to facilitate staff review of the application for meeting the requirements of 10 CFR 71.71(c)(7) and 10 CFR 71.73(c)(1).

7. Correct the following typographical or editorial errors, as appropriate:
   a. Revise page 2.6.7.5-6 to change Figure 2.6.7-7 to read 2.6.7-6,
   b. Revise pages 2.6.7.5-9 and 2.7.1.4-1 to change Table 2.6.7-39 to read 2.6.7-37, and
   c. Page 2.12.2-21 to change Table 2.12.2-1 to read 2.12.2-2.

8. Clarify that the lid and cover plate O-ring/groove designs can withstand the imposed internal and external pressures.

   Pages 2.7.7-1 and 3.5-4 mention high external and internal pressures may exist on the package but do not clarify that the effectiveness of the O-ring/groove designs is equivalent to the inward and outward pressures.

   This information is needed to determine compliance with 10 CFR 71.33, 71.61.

Materials Evaluation

9. Correct the coefficient of thermal expansion for the NS-4-FR neutron shielding material given in Table 2.2.1-13.

   It is necessary to know what impact any thermal expansion has on the neutron shield material (in terms of density) versus what is credited in the shielding analysis. Is this going to result in a significant density reduction? If so, how much? The thermal expansion coefficient in the Table 2.2.1-13 at 302°F is 58.9 x 10^{-6} in/in/°F. The thermal expansion coefficient in the reference is given in "in/in/deg." The x-axis is in °C. Therefore in all probability "in/in/deg" means in/in/°C not in/in/°F. In the reference at 302°F =150°C the thermal expansion coefficient for restrained conditions is 65 x10^{-6} in/in/°C. Both the values and the units in the table appear to be wrong.

   This information is needed to determine compliance with 10 CFR 71.51(a)(2).

10. Provide the fabrication controls on the density of the precast foam pour to ensure that it meets the design specifications used in the thermal analysis.

    Note 10 of Drawing No. 71160-502 claims the foam can be precast.

    This information is needed to determine compliance with 10 CFR 71.33, 10 CFR 71.71, and 10 CFR 71.73.

11. Provide the applicable standards and codes, such as those presented in the MAGNASTOR SAR (Docket 72-1031), for the neutron shielding fabrication, examinations of the welds, and qualifications of the personnel.

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

12. Evaluate the potential radiation damage to the polymer seals depicted in Drawing No. 71160-505. Provide calculations of the expected gamma dose over a year at the location of the seals and compare that to the minimum radiation dose before degradation occurs.

Polymers are known to deteriorate in radiation fields after a dose, which is dependent on the material used. The deterioration may result in a loss of seal tightness and/or introduction of corrosive gases into the system.

This information is needed to determine compliance with 10 CFR 71.43(d), and 10 CFR 71.51(a).

13. Identify where in the packaging that Viton will be used, and the dose that the Viton will receive.

    SAR Section 2.2.2.8 indicates that Viton polymers will be used in the package. With sufficient radiation Viton can generate fluorine that is corrosive to a number of metals. The reviewer was not able to locate where Viton is used in the system and the radiation dose it is exposed to.

    This information is needed to determine compliance with 10 CFR 71.43(d), and 10 CFR 71.51(a).

14. Provide the minimum effective areal density for borated metal matrix composite, borated aluminum, and Boral on Drawing Nos. 71160-571, 71160-572 and 71160-671. Further, indicate the neutron absorber material to be used on the drawings.

    Use of the wording metallic composite excludes borated aluminum, which is an alloy, and Boral, which is a metallic laminate.

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

15. Specify the moisture content of the redwood and balsa wood in the licensing drawings.

    The moisture content of the wood controls the mechanical properties of the wood.

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

16. Clarify whether structural credit is given to the epoxy joining the wood in the impact limiters during and after the tests for hypothetical accident conditions.

    Critical characteristics of the epoxy should be specified if the epoxy is given credit for maintaining the configuration of the wood in the impact limiter.

    This information is needed to determine compliance with 10 CFR 71.51(a).

Chapter 3 – Thermal Evaluation

1. Provide the following additional details on the package finned computational fluid dynamics model, so that an evaluation of the numerical model can be performed.
    a. The calculations demonstrating thermal fin efficiency and fin array effectiveness greater than one,
    b. Literature that validates the computational fluid dynamics approach used to determine an accurate finned heat transfer coefficient and justification that the transitional k-ω turbulence model is the appropriate choice,

c. Clarification that the mesh is modeled such that the "y+" values are less than one at the wall, and
d. Confirm the validity of the FLUENT model used to determine the finned surface heat transfer coefficient by:
    i. Performing sensitivity studies on the mesh resolution, in particular the transition from the fine-mesh region near the fins to the larger mesh region representing ambient air,
    ii. Performing sensitivity studies to determine that the height of the mesh extends sufficiently far from the package to preclude non-physical 'edge' effects at the boundary of the mesh,
    iii. Developing an axial representation of the model mesh that correctly reflects the actual periodic geometry of the finned outer shell, such that the model includes the 1-inch axial gaps between the 6-inch axial segments of each fin, and
    iv. Demonstrating the appropriateness of the turbulence model selected by such means as showing that calculated values of dimensionless parameters are within the turbulence model's ranges of applicability.

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

2. Provide details on the insolation thermal boundary conditions applied to the numerical half-model and a full-scale package.
    a. Provide a table that includes the areas of the three-dimensional fins, areas of the non-finned package surface, and the corresponding solar insolation values applied to those areas; the insolation values also should be justified. A bounding approach should be modeled if a detailed analysis (surface interactions, view factor effects, etc.) is not performed.
    b. From this information, provide a calculation showing the insolation thermal input to the numerical half-symmetry model and a full-scale package.

    The insolation thermal boundary conditions, which are an important part of evaluating the numerical model, were not clearly presented.

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

3. Discuss how the thermal models bound the different shipping configurations shown on Drawing No. 71160-500.

    It is not possible to evaluate whether the numerical thermal model bounds the numerous package variations.

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

4. Justify that appropriate volumetric heat generation values are included in the thermal analyses.

    Figure 3.4-3 and Figure 3.4-7 provide the design basis PWR and BWR fuel assembly axial power distributions. It would appear that the area under the curve should be equal to one, and if the area under the curve is less than one, then a correction factor should be applied to the volumetric heat generation to the active fuel region of the model.

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

5. Revise the table found on page 3.3-1 to provide the operating temperatures and design temperature limits for the components in Tables 3.4-1 and 3.5-1, in order to confirm that components meet their allowable temperature limits. In addition provide:
   a. The temperatures associated with decomposition, auto-ignition temperature, etc., should be considered when determining temperature limits. Components include EPDM O-rings, adhesive, silicone foam (page 3.4-5), thermal insulator (page 3.4-5), wood impact limiters, etc.
   b. The allowable cold and hot temperatures for the EPDM O-rings and their calculated temperatures for the heat test for normal conditions of transport and fire test for hypothetical accident conditions.
   c. References for the minimal acceptable operating temperature for normal conditions of transport "Cold" at -40°F for the components in the table in Section 3.3.2. Note that the safe operating range of the metallic O-rings is obtained from the technical information presented in Section 4.5.2. This section is Helicoflex literature which only specifies the upper operating temperature. The lower operating temperature is not provided.

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

6. Justify the use of the half-axial model for the normal conditions of transport thermal analyses.

   A thermal model should include critical components so that their temperatures can be determined. Page 3.4-2 of the SAR indicates that the thermal analyses for normal conditions of transport are based on the lower half of the package. However, the metallic and EPDM O-ring are at the upper half of the package. Justify how the model bounds, or can accurately predict, O-ring temperatures.

   This information is needed to determine compliance with 10 CFR 71.71.

7. Provide a detailed accounting for the effects of gaps that exist in the TSC and MAGNATRAN thermal models so that an evaluation of a well-represented model can be performed.
   a. Table 3.2-13 lists gaps that were incorporated in the MAGNATRAN model. The applicant should consider the following additional gaps between:
      - TSC and canister spacer,
      - gamma shield and outer shell,
      - outer shell and neutron shield,
      - outer shell and cooling fin,
      - neutron shield and fin-A,
      - mounting ring and fins,
      - fin-A and neutron shield and fin-B,
      - neutron shield and fin-C,
      - fin-A and fin-B, and
      - neutron shield and mounting ring.

b. It is unclear whether the gaps that make up the neutron shield (NS-4-FR, thermal insulator, stainless steel shell) are considered in the model. If so, provide the gap sizes utilized in the model.
c. Only two axial gaps were listed in Table 3.2-13. Are these the only two axial gaps in the TSC and MAGNATRAN assemblies?
d. Explain how gaps were incorporated in the model. What was the thermal conductivity or contact resistance applied to these gaps?
e. The thermal resistances due to gaps have a cumulative effect on heat transfer. Provide a detailed accounting for the effects of these gaps in the thermal model and analysis.
f. The gaps and contact resistances should reflect normal conditions of transport and hypothetical accident conditions to conservatively estimate the heat input to the package. For example, large gaps are often assumed during normal conditions of transport and post-fire hypothetical accident conditions while non-existent gaps are often assumed during the 30-minute fire for hypothetical accident conditions.
g. Table 3.2-13 indicates that the gap between the canister and package is a uniform 0.125 inches. However, the gap can be larger than this value based on the analyses' assumption of a non-uniform gap (i.e., having contact between the canister and package). Account for this discrepancy in the model.

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

8. Confirm that constant thermal properties listed in Section 3.2 and Calculation Package No. 71160-2101 R06, such as NS-4-FR, are bounding for the normal conditions of transport and hypothetical accident condition thermal analyses.

   Bounding conditions should be used in the analyses if the calculation inputs are not functions of temperature.

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

9. Provide additional details on the effects of the hypothetical accident conditions tests on the package so that an evaluation can be performed. Incorporate the appropriate effects discussed below in the hypothetical accident conditions fire model.
   a. Revise Section 3.5 to either provide the following information about the effects of the puncture, and fire tests on the package or provide a cross reference to the details in the structural evaluation.
      i. Temperatures of the neutron shield, gamma shield, O-rings, etc., due to local damage from the drop/puncture tests to the fins and neutron shield.
      ii. Discuss whether the neutron shield material can degrade (combust, char, etc.) at high temperatures and provide the thermal effects on the package due to locally high temperatures of a charring/combusting neutron shield.
      iii. It is stated in the SAR that NS-4-FR in the neutron shield is assumed destroyed and "the result is a lower conductivity." Provide the "lower conductivity" value, provide the basis for this lower conductivity, and explain if it is bounding.
      iv. If in response to item 9.a.ii above, the neutron shield combusts or chars, show that it will not be entirely consumed during the fire so that the fire can impinge directly on the outer shell.

b. Discuss the effects of the hypothetical accident conditions tests on the impact limiter and the resulting thermal effects, including:
   i. Pages 2.6.7.5-8 and 2.6.7.5-9 indicate that the impact limiter can be deformed during the hypothetical accident conditions drops. Would this result in tearing of the impact limiter's thin stainless steel covering?
   ii. See item 5.b in the proprietary request for additional information.
   iii. See item 5.c in the proprietary request for additional information.
   iv. Explain the extent of combustion of the impact limiter's wood components.
   v. Justify the assumed adiabatic surface at the impact limiter locations during hypothetical accident conditions, considering the above, item 5 in the proprietary request for additional information, and the potential for heat input from the fire.

c. Describe the effects of the hypothetical accident conditions tests on the condition of the fins and the resulting effect on heat transfer.
   i. The description of the package after the drop tests on page 3.5-3 of the SAR does not describe the damage to the fins. For example, the damage from the drop and puncture tests was not described satisfactorily. What fraction of the area is damaged? Does the damage provide a pathway for the high temperature fire to penetrate closer to critical components, such as lead and O-rings?
   ii. See item 6.b in the proprietary request for additional information.

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

10. Provide detailed justification for the choice of contact area between the basket and canister shell, contact area between canister shell and the package inner shell, and the thermal conductivity between the gaps. The model should be updated to reflect appropriate conditions so that accurate temperature distributions can be reported.

    a. Radial contacts are, essentially, point contacts and therefore it is difficult to envision large contact angles between two components. Provide detailed calculations showing the PWR and BWR contact areas between the basket and canister shell and the contact areas between canister shell and the package inner shell.
    b. The choice of a 2 Btu/hr-in-°F thermal conductivity between the two components does not appear reasonable, considering that the thermal conductivity of stainless steel components is approximately 0.867 Btu/hr-in-°F. A more appropriate assumption would apply the thermal conductivity of the helium backfill gas or a bounding contact resistance. Incorporate an accurate thermal conductivity between gaps, including the basket/canister shell gap and canister shell/package inner shell gap.

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

11. Provide an energy balance equation on the package surface during the hypothetical accident conditions fire condition.

    It is difficult to understand the thermal inputs to the package due to the hypothetical accident conditions fire from page 3.5-2 and 3.5-3. Energy balance equations on the package during the hypothetical accident conditions fire and post-fire should be provided, including the flame emissivity, package absorptivity, etc., so that an evaluation can be performed.

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

12. Include an appropriate convection heat transfer coefficient when modeling the hypothetical accident conditions fire simulation.

    Appropriate heat transfer coefficients must be applied to numerical models in order to obtain accurate or bounding results. The thermal analysis, however, included a heat transfer coefficient based on a published paper that assumed fire convection coefficients of 5 and 10 W/m^2; no justification for those low values was provided. Include appropriate heat transfer coefficients in the model that take into account increased convection (i.e., Gregory, et al., "Thermal Measurements in a Series of Large Pool Fires", Sandia Report SAND85-0196 TTC – 0659), augmentation by the fins, etc., during the hypothetical accident conditions fire simulation.

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

13. Justify that homogenizing the neutron shield in the thermal model for normal conditions of transport accurately determines the neutron shield temperature and ensure that the neutron shield temperature is below its allowable temperature of 300°F.

    Package components must be below allowable temperatures to ensure proper operation. The model's neutron shield is homogenized; homogenization results in averaging which numerically lowers actual peak temperatures. Provide an explanation that shows the reported homogenized neutron shield temperature (264°F) accurately portrays the highest temperature experienced by the neutron shield.

    This information is needed to determine compliance with 10 CFR 71.33 and 10 CFR 71.71.

14. Provide calculations for the pressures in the PWR and BWR transport package during normal conditions of transport and hypothetical accident conditions.

    Knowledge of pressures within the transport package is necessary in order to evaluate its integrity. Provide calculations that support the statements in Sections 3.4.4.1, 3.4.4.2, and 3.5.4.

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

15. Provide additional information on the personnel barrier so that an evaluation can be performed. The personnel barrier is used to ensure package surface temperatures remain below the regulation requirement. Page 3.4-14 provides insufficient detail of the personnel barrier: "… made from aluminum mesh with a large ratio of open area."
    a. Provide details of the mesh in the drawing, including maximum wire size, minimum percent opening, etc.
    b. Calculation Package No. 71160-3045 states that the barrier is at least 6.7 inches above the fins; include this requirement in the drawings or operating conditions.
    c. Provide calculations and a complete energy balance that support the page 3.4-15 statement that the maximum temperature of the personnel barrier is 162°F.
    d. Page 3.4-14 states that the personnel barrier is not explicitly modeled in the computational fluid dynamics analysis. Provide an indication in the vector plots where the personnel barrier would be located (as described in Calculation Package No. 71160-3045) in order to confirm rising hot air does not impinge on the barrier.

This information is needed to determine compliance with 10 CFR 71.43 and 10 CFR 71.71.

16. Clarify the relation between radiation elements and the “water condition.”

    It is not clear what is meant by the statement (page 3.4-8): “… radiation elements are not used for the water condition.”

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

17. Clarify whether copper and/or aluminum fins are included in the PWR and BWR three-dimensional models.

    Page 3.4-10 lists only copper cooling fins as being modeled. Calculation Package No. 71160-3014 indicates that aluminum fins were modeled. Clarify this discrepancy, the choice of fin emissivity, and whether the resistances (contact, etc.) of the copper and aluminum fins are included in the PWR and BWR models.

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

18. Provide analyses that show the effects of grid sensitivity and modeling uncertainties on thermal results so that an evaluation can be performed.

    The maximum fuel rod cladding temperature is listed as 726°F (Table 3.4-1 of the SAR). This value is near the acceptable allowable of 752°F. As part of the thermal review, sensitivity and uncertainty analyses should be provided. Grid sensitivity and modeling uncertainties include those associated with:

    a. boundary conditions,
    b. uncertainty of applied heat transfer coefficients (heat transfer coefficient correlations often have uncertainties of 25%),
    c. effective thermal conductivity methodology (sensitivity of a range of effective thermal conductivities, etc.),
    d. ranges of internal and external emissivities/absorbtivities (emissivity may not equal absorptivity) and corresponding absorbed solar insolence,
    e. modeling simplifications (number of ray tracings used in radiation heat transfer modeling, etc.), and
    f. consider using the descriptions of grid sensitivity and grid convergence index analyses found in NUREG-2152, “Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications” and the American Society of Mechanical Engineers Verification and Validation 20-2009 (ASME V&V 20-2009), “Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer.”

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

19. Revise the application to include the bolt torque analysis for the lid and port cover plate.

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

20. Explain the choice of ANSYS elements used in the thermal analysis.

    Page 3.4-7 indicates that PLANE55 conduction and MATRIX50 radiation elements are used, whereas page 3.4-8 indicates that LINK31 radiation elements are used. Provide the reason for the different modeling approaches.

    This information is needed to determine compliance with 10 CFR 71.71.

21. Justify the lid seal temperatures presented in the SAR.

    The staff does not agree with the applicant's thermal methodology presented in the SAR and NAC's response (see ADAMS Accession No. ML13093A233) to the NRC's thermal request for additional information (see ADAMS Accession No. ML13059A711). Recognizing the thermal interaction that occurs between package mass, content mass, and content decay heat, the staff had a three-dimensional, COBRA-SFS thermal model generated as part of a confirmatory analysis. COBRA-SFS explicitly modeled (i.e., coupled) the transportation package (including contents) to determine steady-state and transient temperatures of the package during normal conditions of transport and hypothetical accident conditions, respectively. The analysis has shown differences with the SAR, including temperature profiles and temperatures. For example, confirmatory results indicate temperatures of critical package components, such as the lid seal, are higher than the values presented in the SAR.

    This information is needed to determine compliance with 10 CFR 71.51.

Chapter 4 – Containment Evaluation

1. Provide clarification on the term "weldment" used in the SAR and how the weldment is being tested.

   On page 8.1-6 of the SAR, the term "weldment" is used; however, staff needs clarification on what's included within this term here in this SAR. In addition, the applicant mentions the containment boundary weldment being hydrostatic pressure-tested. Staff needs clarification on what is actually being tested in the containment boundary weldment.

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

2. Verify whether the welds and base metal are not potential paths for escape of radioactive material from the transport package containment during transport operation.

   On page 4.1-1 of the SAR, it was mentioned that the potential paths for escape of radioactive material from the transport package containment during transport operation included through the O-ring seal on the lid or the past inner O-ring seal on the lid coverplate. Provide verification on whether leakages through the welds and through the base metal are not other possible ways of escape of radioactive material.

   This information is needed to determine compliance with 10 CFR 71.43(f) and 10 CFR 71.51.

3. Provide the basis for the O-ring compression and O-ring groove dimensions in the drawings, such as from manufacturer data sheets. There are a total of six O-rings used in the MAGNATRAN package.

   a. Provide the manufacturer and product designation for the six O-rings on the engineering drawings.
   b. The drawings should indicate both the dimensions and tolerances of the groove dimensions and O-rings, including surface roughness, to ensure compression of the O-rings.

   This information is needed to determine compliance with 10 CFR 71.33, 10 CFR 71.43, and 10 CFR 71.51.

Chapter 5 – Shielding Evaluation

1. Describe how the shielding analysis addresses the condition that NFH may be NFH components and not entire NFH devices, modifying the analysis as needed.

   The definition of NFH in Chapter 1 of the application includes components of the devices listed in the definition, such as individual rods. This means one of two things. First, the location of the source due to the irradiated NFH component may move axially in the assembly within which it is placed and thus could affect dose rates at different locations of the package. Second, components of multiple NFH assemblies may be loaded into the same assembly (e.g., the rods from multiple thimble plug devices). It is not clear that the shielding analysis addresses these conditions. The impacts to axial (bottom and top) and radial dose rates should be addressed.

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

2. Provide requirements for additional cooling time for assemblies loaded with NFH that are based on appropriate shielding analyses for all NFH types proposed to be transported in the package.

   The proposed contents include several types of NFH. The application describes some shielding analysis regarding the NFH, including additional cooling time required for assemblies containing NFH (see Tables 1.3-16 and 5.8-40). However, the proposed additional assembly cooling times are not supported by the analysis descriptions in Sections 5.8.5 and 5.8.6 of the application. The tables include cooling times that are shorter than appear to be supported by these two sections. The tables also include cooling times that are longer than what appears to have been analyzed in these two sections, which indicates the analysis may be insufficient. Furthermore, it is not clear that the necessary additional cooling times address all the proposed NFH contents. For example, Table 5.8-40 doesn't include neutron source assemblies (NSAs) or hafnium flux reduction assemblies (HFRAs). Also, the definition of NFH includes in-core instrument thimbles, but no analysis is provided for these items. Finally, the applicant should ensure that acronyms and labels for the different types of NFH are used consistently throughout the application.

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

3. Justify the cobalt impurity level assumed in the analysis for the steel and Inconel components of the assembly hardware and NFH.

   The applicant assumed a 0.8 g/kg impurity level. However, this is not supported by the applicant's own source term calculation packages. It is also not supported by PNL-6906, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal" by A. Luksic (June 1989), which shows Inconel impurity levels as high as 1.5 g/kg and steel impurity levels as high as 2.1 g/kg for the assembly types in the proposed contents. Staff notes that while a cobalt reduction campaign was begun in the late 1980s, the cooling time and exposure levels proposed for the assembly and NFH contents indicate that at least some of the contents would have been manufactured prior to this campaign and therefore would have higher levels of cobalt. Thus, a higher impurity level should be used in the analyses.

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

4. Modify the analysis for NFH to use bounding hardware masses in the different assembly axial zones where NFH may be irradiated, including using appropriately bounding insertion configurations for reactor control components (RCCs).

   Staff's review of DOE/RW-0184, "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation" (Dec. 1987), indicates that the characteristics of the NFH associated with the different assembly types in the application exceed the characteristics used in the applicant's analysis. This includes the masses of the NFH in the different assembly axial zones (i.e., upper nozzle, upper plenum, and active fuel zones). In addition some of the NFH types have hardware that extends into the active fuel, which the applicant has not analyzed. These NFH types include thimble plug devices and NSAs (even those without poison rods) and RCCs (also referred to as control element assemblies by the applicant). Given the characteristics of NSAs, which include hardware masses that exceed those of burnup poison rod assemblies (BPRAs), the current analysis appears to be inadequate to address NSAs. Also, any analysis of NSAs should address both the neutron and the gamma contributions and look at the impacts on axial as well as radial dose rates. It is not clear that the analysis covers NSAs for Combustion Engineering 14x14 and 16x16 assembly types. It is also not clear what kind of insertion configurations were assumed for the RCC analysis or the basis for using those configurations. Additionally, the applicant should clarify if and how the analysis for RCCs addresses axial power shaping rods. Finally, the applicant should provide information about HFRAs to a similar level of detail as for the other NFH types (e.g., see Tables 5.8-26 and 5.8-27) and ensure the analysis is bounding for these HFRA characteristics. Based on the analysis, specific limitations with regard to the NFH types may be necessary in the certificate.

   This information is needed to determine compliance with 10 CFR 71.33(b), 10 CFR 71.47 and 10 CFR 71.51.

5. Clarify whether and if so, how fuel spacers and DFC spacers are credited in the shielding analysis.

   It is not clear whether the shielding analysis relies on spacers to maintain the spent fuel contents' axial position fixed. From some of the figures in Chapter 5, it appears that there is space between the base of the assemblies and the base of the TSC cavity. Yet, it is not clear if the position of the fuel is altered for the dose rate calculations at different locations around the package (e.g., the base vs. the top). The applicant should explain what kind of credit is taken (e.g., whether it is position of the assemblies only or the spacer material is also credited) and justify the credit that was taken. The applicant should also clarify whether the spacers are always beneath or above the assemblies and DFCs.

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

6. Clarify the determination and treatment of bounding sources in the analysis for assembly burnup, cooling time, and enrichment combinations for which dose rates, not heat load, are limiting.

   The current analysis indicates that the dose rates are presented based on bounding heat loads (see Section 5.1.2). The staff's experience is that heat loads are bounding for some burnup, cooling time, and enrichment combinations, but dose rates are bounding for other combinations. It is not clear that the application recognizes this or how it addresses these conditions, such as for application of uncertainties and other analysis assumptions.

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

7. Modify the hypothetical accident conditions analysis to consider radial and axial lead slumps that are consistent with or bound the lead slump determined in the structural evaluation, with the entire axial slump occurring at the same end of the lead shielding area.

   Based on the structural analysis in the application (see Section 2.7.1.5), the maximum radial lead slump is 0.9 inches (2.286 cm) and the maximum axial lead slump is 2.33 inches (5.92 cm). The descriptions of the hypothetical accident conditions shielding model indicate smaller slumps were analyzed for shielding. Additionally, the descriptions in the application indicate that the amount of axial lead slump was divided between the ends of the package's lead shielding region (i.e., some slump is at the base and some is at the top of the lead cavity). The analysis should model all axial void due to slump at the end of the lead cavity that maximizes the impact of this slump. The analysis should be modified to address the impacts of this lead slump, since a different source term may result in dose rates at the slump region that are bounding (as opposed to the currently bounding source term).

   This information is needed to determine compliance with 10 CFR 71.51.

8. Correct or confirm the following:
   a. Confirm the spectra for the correct bounding source for the spent fuel contents are provided in Tables 5.2-6 and 5.2-7; other tables indicate that the selected PWR source is not the bounding source;
   b. Confirm the calculations for reconstituted assemblies use direct MCNP calculations or use the response function approach, explaining how if the response function approach was used;
   c. Correct the discussions of dose(s) to be dose rate(s);
   d. Confirm the entries of limiting cooling times in Table 5.6-1 and similar tables in Section 5.8 are correct; some do not appear to fit the pattern described in the beginning of Section 5.6.1.2;

e. Confirm the RCC dose rate analysis uses response functions that were uniquely developed for RCCs (vs. the response functions developed for assemblies and their hardware), providing justification if otherwise;
f. Confirm that response functions are calculated for each assembly type, and each assembly type has multiple response function sets (a normal conditions of transport and a hypothetical accident conditions response function set for each axial zone of the assembly), providing justification if otherwise;
g. Correct the minimum cooling time for WE 14x14 assemblies with burnup between 40 and 45 GWd/MTU and an enrichment between 2.7 and 2.9 weight percent in Tables 5.8-15 and 5.8-38; one is not correct;
h. Confirm or correct the minimum cooling time for all BWR/4-6 fuel assembly types for several burnup and enrichment combinations in Tables 5.8-23 and 5.8-41; several cooling times are the same between the two tables, which appears to be incorrect since the decay heat source differs between the two tables;
i. Confirm the statement that addition of NFH such as BPRAs does not affect the maximum package dose rates (e.g., see Section 5.8.5.2); this appears to be incorrect, especially since the analysis does not account for the effects of increased assembly cooling for assemblies with NFH;
j. Confirm (or provide) the cooling time used in the analysis of the HFRA source term in Section 5.8.5; the analysis doesn't provide this information;
k. Confirm the dose rates for Westinghouse (WE) 17x17 control element assemblies bound those for WE 15x15 control element assemblies; it is not clear this is true given the amount of material in the latter's plenum region and the higher source per unit mass of Ag-In-Cd (vs. steel and inconel) in a number of gamma energy groups;
l. Correct Figures 5.8-9 and 5.8-21; both show PWR and BWR basket models that are not consistent with the text description of the basket model with respect to absorber plates; and
m. Either correct Tables 5.8-38 and 5.8-39 to remove the proposed cooling times for WE 15x15 assemblies with burnup and enrichment combinations of 30 to 35 GWd/MTU and 2.1 to 2.3 weight-percent and 45 to 50 GWd/MTU and 2.7 to 2.9 weight-percent or revise the shielding analysis to evaluate these items. The analysis in Section 5.8.3 does not support these cooling times.

This information is needed to determine compliance with 10 CFR 71.33(b), 10 CFR 71.47 and 10 CFR 71.51.

9. Confirm whether:
   a. there is an axial lead gap due to fabrication tolerances and any slump due to normal conditions of transport test conditions, and
   b. the shielding analysis addresses these effects.

   It is not clear from the application whether or not these two conditions exist and how they are addressed in the normal conditions of transport shielding calculations. These effects can impact the maximum dose rates of the package and should be addressed in the shielding analysis.

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

10. Provide in the application the necessary fuel parameters to develop shielding models for the different source regions of the fuel assemblies.

Sections 5.8.3.1 and 5.8.4.1 refer to tables in Section 5.8.1 as the sources of information for developing the shielding models of the different assembly types' source regions. However the tables in Section 5.8.1 do not contain sufficient information to support the model specifications for the assemblies' fuel region. The tables are missing information that includes the fuel rod parameters (e.g., fuel pellet diameter, clad thickness, pitch) and guide tube and instrument tube parameters (e.g., outer diameter and tube thickness).

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

11. Justify how the fuel assembly response functions can be used for estimating dose rates from BPRAs and thimble plug devices if credit is taken for the masses of these NFH types, or modify the analysis to appropriately treat these NFH types.

    Alterations of material properties in the model result in different response functions. Thus, it is not clear that the assembly response functions can be used to estimate NFH dose rates when the added mass of the NFH is credited.

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

12. Analyze the GTCC waste contents for the material and source densities that bound the dose rates.

    The package proposes a maximum contents mass of 55,000 pounds and a maximum specific and total activity limit for ^{60}Co. The current shielding analysis is based on the contents mass that equates to the maximum ^{60}Co activity limit at the maximum specific activity limit and homogenizes the mass and source throughout the GTCC waste liner's cavity. Since the allowable mass of material is much greater than is analyzed for shielding, the source may be more concentrated volumetrically than in the analysis. A more concentrated source may result in higher dose rates. The application should identify the configuration of the GTCC waste contents that results in bounding dose rates and demonstrate those dose rates comply with regulatory limits. The application should also clarify that the specific activity limits apply to any loose crud that may be included in the waste. Also ensure that the operating procedures for loading GTCC waste in Chapter 7 are consistent with this bounding analysis.

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

13. Provide an evaluation of the conservatisms and the uncertainties in the analysis.

    The bounding dose rates for the proposed package design and its contents are analyzed to be at the regulatory limits. The analysis uses various assumptions that introduce uncertainties into the analysis. The applicant should address these uncertainties as part of the analysis to demonstrate compliance with the regulatory limits. This can be done by identifying the conservatisms along with the uncertainties and showing, in at least a semi-quantitative manner that the conservatisms adequately compensate for the uncertainties. This approach may be an acceptable way to address some of the issues in the other shielding questions (e.g., items 2 and 3, in enclosure 1).

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

Chapter 6 – Criticality Evaluation

1. Provide additional information on the neutron poison plates.

   Page 6.1.1-1 of the safety SAR states: "The minimum as-manufactured loading of the neutron absorber sheets depends on the effectiveness of the absorber and the minimum effective absorber areal density. Effectiveness of the absorber is influenced by the uniformity and quantity of the ^{10}B nuclide within the absorber base material. Depending on the absorber type, 75% or 90% ^{10}B effectiveness is credited. Any material meeting the ^{10}B areal density and physical dimension requirements will produce similar reactivity results. See Table 6.1.1-1 for effective versus "credit" adjusted absorber areal densities." This definition does not uniquely qualify the poison plates. A material meeting the ^{10}B areal density and physical dimension requirements does not guarantee that the neutron poison material will produce similar reactivity results. In addition, the poison plates must meet certain chemical and mechanical properties to ensure the assumptions used in the criticality safety analyses remain valid under all conditions as specified in 10 CFR 71.71 and 10 CFR 71.73. The applicant needs to provide additional information to:
   a. uniquely identify the materials used for neutron poison plates in Table 6.1.1-1 by providing names and specific composition of the materials that will be used in the neutron poison plates. See item 17, in the Materials Evaluation, above,
   b. state which poison plate can take how much credit (75% or 90%) and state the analytical conditions that are assumed when taking additional boron credit, and
   c. demonstrate that any material used as poison plates meets the assumptions used in the criticality safety analyses under all conditions specified in 10 CFR 71.71 and 10 CFR 71.73.

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

2. Clarify whether analyses have been performed for all allowable fuel assembly designs or a bounding fuel assembly design has been selected for all of the package designs.

   The title of Table 6.1.2-1 indicates this table provides bounding PWR fuel assembly loading criteria. The data shown in this table include a list of fuel assembly designs. For the BWR fuel packages, the applicant provided three tables for the qualified fuel assembly designs. From the data presented in this table, it seems that safety analyses had been performed for each of the fuel assembly types. The applicant needs to clarify if analyses have been performed for all allowable fuel assembly designs or a bounding fuel assembly design has been selected for all of the package designs.

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

3. Provide references for the data in Table 6.2.1-2 on BWR fuel assembly characteristics, and Table 6.2.1-1 on PWR assembly characteristics.

   Some of the numbers in a spot check do not agree with the reviewers independent sources, especially on the minimum fuel mass. References for the fuel masses in SAR Tables 1.3-7, 5.2-1, and 5.2-2 should also be provided.

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

4. Justify how the planar average enrichment bounds actual enrichment for BWR fuel contents.

   Page 6.1.2-1 of the SAR states: "Maximum enrichment is defined as peak rod enrichment for PWR assemblies and the maximum peak planar-average enrichment for BWR assemblies. The maximum initial peak planar-average enrichment is the maximum planar-average enrichment at any height along the axis of the fuel assembly." However, planar-average enrichment alone may not be an appropriate parameter for limiting the fuel assembly enrichment because many combinations of rod enrichments can give the same planar average enrichment. Page 6.10.3-4 of the application states: "Fuel assembly types studied are the GE 8x8 60 and 62 fuel rod, and GE 9x9 74 fuel rod assembly types. Each of the fuel assemblies is evaluated at a planar-average homogeneous enrichment and the actual documented enrichment pattern. Results of the analysis, listed in Table 6.10.3-6, show that for all cases, the heterogeneous enrichment produces a lower k_{eff} than the homogeneous planar-average (in this case assembly-average) enrichment case." The enrichment patterns presented in Table 6.10.3-6 show that some of the fuel rods have enrichment exceeding 5 weight-percent, which is the enrichment limit of commercial reactor fuel per the requirement of 10 CFR 50.68(b)(7). Provide the following information:
   a. Justify that planar average enrichment alone is bounding for any and all enrichment patterns for BWR fuel that can be loaded in the MAGNATRAN. Consider adding a maximum enrichment to accompany the average limit.
   b. Address the concern that the analyzed BWR fuel enrichment in Table 6.10.3-6 exceeds the regulatory limit of 10 CFR 50.68(b)(7), i.e., if there is any spent fuel design that has been approved by the U.S. Nuclear Regulatory Commission for use in commercial reactors.
   c. Justify that the 8x8 and 9x9 studies are bounding for all BWR fuel types – especially the 10x10 assemblies – allowed for transport in the MAGNATRAN.

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

5. Explain how the maximum enrichment was determined for each allowable fuel assembly design.

   The SAR for the MAGNATRAN spent fuel transportation package design provides maximum enrichment limit for each fuel assembly design of the allowable fuel contents. It is not clear how the maximum allowable enrichment was determined. The applicant needs to explain how the maximum enrichment was determined for each allowable fuel assembly design, including detailed description and calculations as necessary.

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

6. Provide a list of all allowable NFH inserts for the PWR fuel assemblies in the MAGNATRAN package.

   On page 6.2.2-1 of the SAR, the applicant states: "As inserts are modeled as neutronically transparent zirconium alloy, no differentiation between insert type is made in this chapter. Inserts allowed include, but are not limited to, reactor control rod assemblies (RCCAs), BPRAs, thimble plugs (TPs), NSAs, and HFRAs. BPRAs may include PYREX and wet annular burnable absorber types of absorbers, but are not limited to these particular vendor specific definitions." Since there may be NFH, such as Plutonium-Beryllium neutron sources, that may not be bounded by zirconium, the applicant needs to revise the NFH qualification statements to provide a list of all allowable NFH inserts with justification for their impact to criticality safety of the package.

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

7. Provide a definition for the term "off-normal conditions" and justification for the conclusion that there will be no permanent deformation of the fuel basket that will impact criticality safety under "off-normal conditions."

   Page 6.3.1-3 of the SAR states: "Fuel assembly and basket will retain their structure and will not show any significant permanent deformation during normal conditions or off-normal or accident events." However, the term "off-normal conditions" is not defined in the SAR or in the regulations of 10 CFR Part 71. The applicant needs to provide a definition for the term "off-normal conditions" and justification for the conclusion that there will be no permanent deformation of the fuel basket that will impact criticality safety under "off-normal conditions"

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

8. Clarify if the fuel optimization analysis intends to optimize fuel loading in a package. If so, provide definitions for the objective function(s), constraint functions, and the approach or algorithm used in the fuel loading optimization analysis and justify that the optimal fuel loading bounds all loading criticality safety.

   Page 6.6.2-1 of the SAR states that the fuel loading is optimized in the criticality models using the several conditions. It is not clear what the optimization analyses intend to optimize. If the objective of the fuel loading optimization is to maximize fissile material loading, provide definitions for the objective function(s), constraint functions, and the approach or algorithm used in this fuel loading optimization. The applicant also needs to demonstrate that the package with optimal fuel loading is conservative in terms of criticality safety. In addition, the applicant needs to explain why and how the removal of BWR channels and the presence of NFH are relevant to fuel loading optimization.

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

9. Demonstrate that the selected critical experiments are appropriate for the configurations of 33 PWR and 82 BWR fuel assembly baskets.

   Section 6.8 of the SAR presents discussion on and results of benchmarking of the criticality safety analysis code MCNP for the MAGNATRAN package. However, it was not clear if these selected critical experiments are appropriate for the packages of 33 PWR fuel assembly package or 82 BWR fuel assembly packages because it is not clear if the selected critical experiments include configurations that have assembly-size water-filled holes. The applicant needs to demonstrate that the selected critical experiments are appropriate for the configurations of 33 PWR and 82 BWR fuel assembly baskets.

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

10. Provide information on the convergence of the MCNP model for the 33 PWR and 82 BWR fuel assembly packages

    Based on the paper published by Forrest Brown, "On the Use of Shannon Entropy of the Fission Distribution for Assessing Convergence of Monte Carlo Criticality Calculations," PHYSOR-2006, ANS Topical Meeting on Reactor Physics, Organized and hosted by the Canadian Nuclear Society, Vancouver, BC, Canada, 2006 September 10-14, the MCNP code will need more cycles to converge on the source distribution in comparison with the k_{eff} convergence for systems containing large water holes. Since the 33 PWR and 82 BWR fuel assembly packages contain assembly-size water-filled holes in the center of the basket, the model might have terminated prematurely if the k_{eff} was used as the sole convergence criterion. The applicant needs to provide information on the convergence of the MCNP model for the 33 PWR and 82 BWR fuel assembly packages.

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

11. Clarify if the poison loadings presented in Table 6.10.1-9 are optional.

    Page 6.10.1-11 of the SAR states: "Using the limiting model description from Section 6.10.1.2, the maximum enrichments are summarized in Table 6.10.1-9 for the optional PWR absorber loadings." The data in the table however do not seem to indicate that the absorber loadings are "optional." The applicant needs to clarify whether these poison loadings are optional or not and make corrections if necessary.

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

Chapter 7 – Operating Procedures Evaluation

1. Modify the package operations descriptions to ensure that the essential elements of operations involving the TSC are described in the "Package Operations" chapter.

   The purpose of the "Package Operations" chapter is to describe the essential elements needed to prepare the package for shipment in order to ensure its performance under normal and accident conditions will be as described in the package evaluation. Options in the operations sequence or the operations to be performed should be described in the chapter. It is not clear that some of the descriptions meet this criterion. These descriptions include:
   a. The statement at the beginning of Section 7.1.2 that indicates site-specific procedures may differ from those described in the section. If different hardware is to be allowed for the operations in that section, the operations descriptions in the section should be written to allow for that.
   b. The statement at the end of the note following Section 7.2.1, step 5, that says "The site-specific procedures can address other unloading facility arrangements."

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

2. Modify the package operations descriptions to provide the following information:
   a. A torque value appears to be missing from Section 7.1.4, step 11,
   b. Section 7.1.5 appears to be missing a description for visual inspection of the GTCC waste basket liner and placement of the liner into the GTCC TSC,

c. Section 7.1.5 is missing a description of the installation, welding, weld exam of the closure ring to the TSC shell and lid, and recording of the results,
d. Section 7.1.5, steps 8, 10, and 12 appear to be missing acceptance criteria for their respective operations (and discussion of recording of results), and
e. Section 7.2.2, step 9 should also address the package cavity spacer when it is present.

Based on the other operations descriptions it appears that the requested information is important to the operation of the package and should be included in the operations descriptions.

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

3. Clarify why hydrostatic or helium leak tests during the loading of the TSC with GTCC waste contents is not used.

   Page 7.1-11 of the SAR, provides the steps to be followed in order to load the TSC with GTCC waste contents. However, it appears that the procedure lacks a step to perform a hydrostatic or helium leak test. Provide clarification on why this step is not used in this portion of the SAR.

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

4. Modify the package operations to ensure references to appropriate regulatory requirements are given for the operations being performed.

   The package operations descriptions include several references to regulatory requirements. A number of these are not appropriate for the particular operations for which they are used. Also, some descriptions are missing specific references for the operations. These include the following:
   a. Section 7.2.1, step 2 should also refer to the radiation level and contamination level requirements of Part 71 per 10 CFR 20.1906 since there are requirements for the package surface, which isn't accessible until the personnel barrier is removed in step 2,
   b. Section 7.2.2, step 24 should include a specific reference to 49 CFR 173.428(d),
   c. Section 7.3, step 8 should refer to 49 CFR 173.428, 10 CFR 71.87(i), and 10 CFR 71.47 apply to loaded packages and not to empty packages, and
   d. Section 7.3, step 10 should refer to 49 CFR 173.428; the current reference appears to be incorrect.

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

5. Modify the description on page 7.1-5 regarding assembly average burnup to indicate the limit on burnup applies to the maximum assembly average burnup and not the nominal assembly average burnup.

   The shielding analysis is based on the maximum assembly average burnup not exceeding 45,000 MWd/MTU unless the assembly is loaded into a DFC, in which case the maximum assembly average burnup must not exceed 60,000 MWd/MTU.

This information is needed to determine compliance with 10 CFR 71.47, 10 CFR 71.51, and 10 CFR 71.87.

6. Clarify the intended helium backfill pressure within the MAGNATRAN package.

   Page 7.1-7 of the SAR indicates that the MAGNATRAN package is to be backfilled with helium to 14.7 (+1, -0) psia. Recognizing that the pressure within the package will fluctuate with temperature, the helium pressure within the package could become less than atmospheric when temperatures fall below the initial backfill temperature, especially for low decay heat packages. Confirm whether a higher helium backfill pressure than 14.7 psia was intended to ensure that the pressure in the package is always higher than atmospheric pressure.

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

7. Clarify the loading and unloading time restrictions with different TSC heat loads.
   a. Provide procedures for loading and unloading in the event that the time limits of 41, 44, and 6 hour limits are exceeded, for movement from the concrete cask pedestal to placement on the transport vehicle, movement to the transport vehicle for TSCs loaded immediately prior to transport and lifting the cask lid to closing the transfer cask shield doors, and unloading, respectively.
   b. For the time limited items listed above in item 7.a, explain whether TSC's with heat loads less than 23 kW for PWR fuel and 22kW for BWR fuel have similar time restrictions.
   c. Provide the boundary conditions, calculations, and their description that support the time restrictions for loading and unloading.

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

8. Ensure the descriptions of the package components in the operating procedures, acceptance tests, and maintenance program are consistent with the package design description, including the certificate drawings.

   All sections of the SAR that will be referenced in the certificate of compliance should use terminology that is consistent with the terminology used to describe the package design. For example, Section 7.1.2, step 26, refers to a canister spacer. The SAR discusses both a canister spacer and a cask cavity spacer, which appear to be the same item. Additionally, the package cavity spacer is included in the proposed certificate drawings.

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

8. Describe in the operating procedures that the package exterior will be decontaminated of chloride-containing salts and other corrosive agents.

   Current operating procedures state "Perform contamination and radiation surveys and record results. Verify results comply with the applicable requirements of 10 CFR 71.87(i) and 10 CFR 71.47." These regulations only deal with radiation levels and do not address decontaminations methods or levels to assure removal of corrosive agents. The removal of the salt is not a radiation issue but rather it is needed to avoid setting up conditions for stress corrosion cracking of the canister and should be added to the procedures.

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

9. Provide in the "Package Operations" Chapter the closure lid welding procedure to ensure no hydrogen explosion could occur.

   Section 2.2.2.3 of the SAR indicates a small amount of hydrogen will be produced due to the interaction of the aluminum in the basket with the water during loading. The amount generated must be assessed to determine if an explosive situation will develop when the closure lid is welded on the TSC.

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

Chapter 8 - Acceptance and Maintenance Tests Evaluation

1. Provide clarification on the helium fabrication leakage rate test mentioned in Section 8.1.4.1 of the SAR.

   On page 8.1-6 of the SAR, the text states that a "helium fabrication leakage rate test of the containment boundary weldment and lid will be performed in accordance with the requirements of Section V, Article 10, of the ASME Boiler and Pressure Vessel Code." Provide clarification on what makes this test different than the American National Standards Institute (ANSI) N14.5-1997, "Radioactive Materials - Leakage Tests on Packages for Shipment," test and whether the base metal is being leak tested.

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

2. Provide the following for the gamma shielding test in Section 8.1.6.1:
   a. clarification that the test is performed prior to installation of the package fins (copper, steel, aluminum) as well as the neutron shield assemblies,
   b. justification that the scanning speed is sufficiently slow so that the choice of detector for the scan will ensure the scan will detect problems that may exist in the fabricated shield; the appropriate speed is dependent on the detector's characteristics (e.g., response time and dead time),
   c. clarification that retesting of the shield after repair will include the areas surrounding the repaired area that also may be affected by the repair or corrective action,
   d. an acceptance criterion that is based on the shielding analysis and package design as specified in the proposed certificate drawings, and
   e. explanation and justification of the specified count time being at least 1 minute.

   For item d, the applicant proposes that dose rate measurements from a steel and lead shielding mockup be used as an acceptance criterion for the fabricated packaging's shielding. This is an acceptable approach; however, the mockup should use steel and lead shielding at the minimum thicknesses specified in the drawings and supported by the shielding analyses in order to correctly ensure the as-fabricated packaging is within the design specifications in the certificate drawings. The currently proposed mockup would allow for acceptance of a packaging that is not within those specifications since the acceptance criterion is based on dose rates from a mockup with minimum dimensions that are less than those specified in the proposed certificate drawings. For item e, it is not clear what is meant. It appears that an integrated count is being made over the 4.5 feet that the detector is to travel per minute, which does not seem to be an appropriate method for verifying the lead shielding's effectiveness and can only be done with certain detector types.

This information is needed to determine compliance with 10 CFR 71.85.

3. Justify the appropriateness of using a dry steam methodology in the thermal acceptance test. There are a number of issues that need to be resolved in order to confirm that the acceptance test would provide valid results/conclusions:
    a. The energy balance provided in Section 8.7.1 of the SAR only includes the effect of the condensate. Provide an explanation describing why other energy components were not included, including exit steam quality, exit steam temperature, uniformity (if any) of the exit steam's saturation, etc. In addition, provide a thorough energy balance.
    b. Justify the appropriateness of using a single surface temperature, which is influenced by local ambient conditions, to confirm steady state operation, which is necessary to achieve meaningful results.
    c. Describe the criteria for increasing the number of thermocouples in the test.
    d. Calculations/analysis with assumed test parameters (using an analytical tool such as a spreadsheet or other mathematical software) is presumed to have been performed to confirm the feasibility of the acceptance test procedure and assess the sensitivity of test results to perturbations in the test conditions. Provide the results of this study. Note: The test description is very brief and does not sufficiently describe the test. It is critical to have accurate temperature and flow measurement. Weighing the collected water might be appropriate, but there would be an inevitable time-lag in the data. This is an additional source of uncertainty/sensitivity, and needs to be considered in the calculation spreadsheet analysis.

    This information is needed to determine compliance with 10 CFR 71.43 and 10 CFR 71.87.

4. Modify Section 8.1.6.2 to include a test of the as-fabricated neutron shield for each package.

    The currently proposed section describes chemical analysis to ensure composition and density and states that installation will be done using qualified procedures. However, there is no acceptance test to show the neutron shielding of the as-fabricated package performs as designed. Such a test is necessary to ensure each package meets the requirements described in 10 CFR 71.85(a) and (c). A test comprised of measurements with a check source and an appropriately sized grid over the entire neutron shield surface that are compared to calculated dose rates using minimum dimensions and composition specifications is one satisfactory method for acceptance testing. An alternative comparison would be to use a mockup having the minimum composition specifications and minimum dimensions. The proposed test method should be properly justified as satisfying the acceptance test requirement. Further guidance regarding shielding acceptance tests is contained in NUREG/CR-3854, "Fabrication Criteria for Shipping Containers."

    This information is needed to determine compliance with 10 CFR 71.85.

5. Describe in the Maintenance Chapter of the SAR the testing done to ensure that the steel covering for the neutron shielding units and impact limiters remains hermetically sealed.

The arguments that the impact limiters and neutron absorber materials do not corrode or rot are that they are in controlled atmospheres provided by a permanently sealed stainless steel sheathing (SAR Sections 2.2.2.1, and 2.2.2.2.1), however, clarification is needed to further describe what kind of leak tests are being used to test the shell welds.

This information is needed to determine compliance with 10 CFR 71.43(d), and 10 CFR 71.47(a).

6. Provide the following:
   a. A description of any maintenance needed for the rotation trunnions in Section 8.2.1, similar to that provided for the lifting trunnions,
   b. Clarify whether the tests and leak rates in Section 8.2.2.4 apply to all parts of the containment boundary after repair, and
   c. An acceptance criterion for the repairs described in the 5th sentence of the 1st paragraph of Section 8.2.5 that includes the repaired area conforming to the certificate drawings.

   It seems that some kind of maintenance inspection is needed for the rotation trunnions similar to what is proposed for the lifting trunnions. The maintenance tests descriptions imply that other items of the containment boundary, in addition to the metallic seals, may need maintenance or repair (e.g., weld repairs). So, it appears that some kind of acceptance criterion and testing is needed, along with the test sensitivity for these items too. It is not clear from the current application that the tests and acceptance criterion in Section 8.2.2.4 apply to all parts of the containment boundary. For item c, the application should be clear that the acceptance criteria for any repair includes conformance with the certificate drawings.

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

7 Place the language "in accordance with the methodologies and requirements of ANSI N14.5-1997 using approved written procedures" in the "Pre-shipment Leak Testing" section of the SAR.

   On page 8.2-2 of the SAR, it is written that "a pre-shipment leakage rate test shall be performed prior to each loaded transport to confirm that the MAGNATRAN transport package containment system is properly assembled for shipment." If this language is correct, place "in accordance with the methodologies and requirements of ANSI N14.5-1997" at the end of this sentence.

   This information is needed to determine compliance with 10 CFR 71.43(f) and 10 CFR 71.51.

8. Clarify the type of test used and the qualification requirements for the individual who approves the periodic leakage rate testing.

   On page 8.2-2 of the SAR, it is mentioned that "the periodic leakage rate test will be performed with approved written test procedures." However, clarification is needed to determine what type of test is being used and the qualification requirements for the individual who approves the periodic leakage rate tests.

This information is needed to determine compliance with 10 CFR 71.37.

9. Justify that the proposed method for verifying the shielding effectiveness (described in Section 8.2.3 of the application) is sufficient for that purpose.

   The current application proposes to rely on pre-shipment dose rate measurements as a maintenance test of the package's shielding effectiveness. Pre-shipment measurements provide verification of shielding effectiveness only if they are compared to dose rates that are calculated for the package containing the loaded contents. It is not clear from the currently proposed Section 8.2.3 that this comparison will be done. Thus, the pre-shipment measurements are not adequate for verifying continued shielding effectiveness. A maintenance program that compares pre-shipment measurements to dose rates calculated for the shipment's contents at a number of package surface locations that is sufficient to provide assurance of continued effectiveness of the shielding and at some appropriate frequency (i.e., it need not be for every shipment) would be an acceptable approach.

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

10. Revise the application (either maintenance program or operating procedures) to provide a test or evaluation to verify that the neutron absorbing material used for to ensure criticality safety is evaluated prior to transport.

    This information is needed to determine compliance with 10 CFR 71.87(b)(3), 10 CFR 71.55, and 10 CFR 71.59.