Non-Proprietary Request for Supplemental Information
Model No. No. CASTOR® geo69
Docket No. 71-9383

This request for supplemental information (RSI) identifies information needed by the staff in connection with its acceptance review of an application for a certificate of compliance for the Model No. CASTOR® geo69 spent fuel transportation package, dated January 14, 2021 (Agencywide Documents Access and Management System Accession No. ML21033A353). The U.S. Nuclear Regularly Commission (NRC) staff reviewed the application using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material."

The requested information is listed by chapter number and title in the applicant's safety analysis report (SAR).

Request for Supplemental Information

2.0 Structural Evaluation

2-1 Provide information to demonstrate that the structural design of the package meets the moderator exclusion criterion or revise the criticality safety analyses to evaluate a flooded package under normal conditions of transport (NCT) and hypothetical accident conditions (HAC) as well as for the criticality safety index calculations.

The applicant assumed that the internals of the package were dry in some of the criticality safety analyses. In the application, the applicant states: "Due to a very low reactivity of dry fuel, the cask reactivity under normal and hypothetical accident conditions is bounded by the reactivity of the fully flooded cask." Although the packaging design includes two closure seals, the structural design has not demonstrated that the package design is qualified for moderator exclusion pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 71.55(d) and 71.55(e) because the structural analysis did not provide specific acceptance criteria linked to moderator exclusion for both normal conditions of transport and hypothetical accident conditions.

This information is required to determine compliance with 10 CFR) 71.55(d) and 71.55(e).

2-2 Provide the following documents to confirm that requirements related to lifting, NCT, and HAC have been satisfied:

The following references are needed to confirm bolt loads, material properties, impact limiter performance, and loads to be applied for NCT and HAC drop tests.

1014-TR-00029, Rev. 0
Structural Evaluation
Impact Limiters, Transport Package CASTOR® geo69

DIN EN 1993-1-5, 2019/10
Eurocode 3: Design of steel structures
Part 1-5: Plated structural elements

DIN EN 1999-1-1, 2014/03
Eurocode 9: Design of aluminum structures

VDI-Richtlinie 2230, Part 1
Systematic calculation of highly stressed bolted joints,
Joints with one cylindrical bolt.
Version 11/2015

VDI 2230:2014-12 Part 2
Systematic calculation of highly stressed bolted joints
Multi bolted joints
Safety Standards of the
Nuclear Safety Standards Commission (KTA)
KTA 3201.2

Components of the Reactor Coolant Pressure Boundary
of Light Water Reactors
Part 2: Design and Analysis
2017-11

1014-TR-00009
Material Data CASTOR® geo69

This information is required to determine compliance with 10 CFR 71.71, 10 CFR 71.73, and 10 CFR 71.45.

2-3 Provide justification/benchmarking used to support and validate drop simulations and stress analysis in ANSYS and LS-DYNA for NCT and HAC.

The applicant provided drop test results for NCT and HAC that were based on simulation without any supporting physical testing or any reference to actual drop testing to verify such results. Without physical testing to confirm simulations (LS-DYNA and ANSYS) which are meant to replace actual physical testing, staff cannot confirm any of the results or assumptions made in the safety analysis report (SAR). Data related to physical testing should be provided so that staff can confirm that the simulations produce reasonable results. The physical data that staff needs to confirm drop simulations should include specimen description, drop orientation, target information, prototype details (if any), specimen construction, package instrumentation, atmospheric test conditions, testing methodology, validation, etc. Note that drop test data used to benchmark the CASTOR® geo69 analysis does not have to be for this specific package but could be for a package whose behavior and/or physical properties (i.e. shape, dimensions, weight, impact limiters characteristics, etc.) are similar to this package. Attachment 2A of NUREG 2216, specifically Section 2A.4 “Computer Model Validation”, provides additional validation guidance.

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

2-4 Provide information to demonstrate that the borated metal matrix composite (MMC) material used in the basket will not deform as a result of the tests for NCT and HAC.

The package design uses borated MMC plates to make the fuel basket cells in the canister. Therefore, the MMC plates serve as both the structural components and neutron poison plates. The applicant performed criticality safety analyses for the package with the assumption that the borated MMC plates will retain their geometric shape and dimensions. The shielding analyses also credit the borated MMC plates. Because the borated MMC is an aluminum based alloy, it is not clear if the MMC plates will deform under NCT and HAC.

This information is necessary for the staff to determine compliance with 10 CFR 71.47, and 10 CFR 71.51, 10 CFR 71.55(d), 10 CFR 71.55(e) 10 CFR 71.59, 10 CFR 71.71, and 10 CFR 71.73.

2-5 Provide the following documents in English referenced in the CASTOR® geo69 SAR for the proprietary materials properties.

The CASTOR® geo69 package uses several proprietary materials that are not described in the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code, including the neutron shielding material, fuel basket structure (i.e., fuel cell partitioning plates), impact limiter foam, and containment seals. The staff needs the following documents referenced in the SAR Section 2.2, "Materials," to verify that the materials properties support the safety analyses:

[4] Material data by manufacturer Ticona GmbH, GUR {proprietary information removed}. Domininghaus, "Die Kunststoffe und ihre Eigenschaften", Springer, 1998
[6] {proprietary information removed}
[7] Material data by manufacturer General Plastics, USA
[8] HELICOFLEX Federelastische Metalldichtungen, Garlock Sealing Technologies Catalogue July 2017 {proprietary information removed}

In Document No. 1014-TR -00028, Structural Evaluation Basket and Shielding Elements Transport Package CASTOR® geo69 provide the following reference:

[11] {proprietary information removed}

This information is needed to determine compliance with 10 CFR 71.31(c) and 10 CFR 71.33(a)(5).

2-6 Provide details of the fuel contents, cask drying process, and the structural performance of the fuel cladding during NCT and HAC (if needed to demonstrate that the analyzed fuel configuration of the fuel will be maintained).

The staff needs the following information to evaluate the allowable fuel contents and potential content reactions:

- The specific types of allowable cladding alloys (e.g., "Zircaloy-2"),
- The definition of "undamaged" fuel assemblies,
- The cask drying criteria (e.g., vacuum pressure, hold time) that ensure a dry, inert environment, and
- The maximum allowable fuel cladding temperatures during short-term operations (e.g., cask drying).

In addition, if the geometric form of the fuel is assumed to be unaltered during NCT and HAC, the staff needs the following information on the structural performance of the fuel assemblies:

- The cladding mechanical properties used in the structural analysis,
- A description of how the structural evaluation accounts for a reduced cladding thickness due to oxidation during reactor service,
- An analysis of the fatigue performance of the fuel rods,
- For high burnup fuel in dry storage for more than 20 years prior to transport, a demonstration that long term aging degradation issues will not prevent the fuel from fulfilling its safety functions (See NUREG-2224, "Dry Storage and Transportation of High Burnup Spent Nuclear Fuel – Final Report," Figures 4-1 through 4-3 for some examples of approaches considered to be acceptable),

Alternatively, if the fuel is assumed to reconfigure during NCT or HAC, revise the criticality, shielding, and thermal analyses with consideration of fuel reconfiguration, beyond what is already addressed in the application.

This information is needed to determine compliance with 10 CFR 71.33(b), 10 CFR 71.43(d), 10 CFR 71.47, 10 CFR 71.51(a), 10 CFR 71.55(d), 10 CFR 71.55(e) and 10 CFR 71.73(c)(4).

2-7 Provide the fuel basket material qualification documentation and acceptance testing criteria for the required mechanical properties, thermal properties, and neutron attenuation performance.

SAR Section 8.1.5.1 states that the Al-B_4C MMC basket material does not conform to the requirements of the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code. The SAR also states that:

> "These materials and their requirements are either specified by other standards, or they are tested individually for the use in the CASTOR® packages and their qualification is documented."

The SAR does not provide information on the qualification or acceptance testing requirements for the basket material structural, thermal, or neutron attenuation performance. As a result, the staff requests this information for:

- mechanical properties (e.g., yield strength, tensile strength, elongation, fracture performance), including effects of temperature,
- inspection requirements to verify that the material is free of defects that could affect mechanical performance,
- thermal properties (e.g., thermal conductivity, density, specific heat)
- neutron attenuation performance, and
- porosity, considering implications on corrosion

NUREG-2216, Section 7.4.7 provides guidance on the staff's review of neutron absorber materials, which includes reference (with some exceptions) to ASTM International C1671-15, "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorber Materials for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging."

This information is needed to determine compliance with 10 CFR 71.33(a)(5), 10 CFR 71.43(d), 10 CFR 71.47, 10 CFR 71.51(a), 10 CFR 71.55(d) and (e), and 10 CFR 71.85(a).

3.0 Thermal Evaluation

3-1. Provide the thermal analyses and results associated with the transfer and loading operations discussed in SAR Section 7.1 “Package Loading”.

There was no discussion in Chapter 3, “Thermal Evaluation,” about the thermal analyses or results for the various operations mentioned in the SAR’s Section 7.1. Relevant thermal issues during operations (e.g., loading, unloading, transfer) include, for example, time limits of transient operations, flushing water flow rate associated with loading the fuel assemblies into the canister, the basis of the time limits associated with the vacuum drying process (e.g., Step E2.7 and the remedy if the time limit is not met), and measures to ensure that the water used for cooling purposes during loading operations does not boil.

This information is needed to determine compliance with 10 CFR 71.31(a) and 10 CFR 71.35.

3-2 Provide Section 3.5, “Appendix” in the SAR Thermal chapter so that a review can be performed.

The SAR’s Table of Contents and SAR page 3.5-1 mention a thermal-related Appendix. However, there is no information provided.

This information is needed to determine compliance with10 CFR 71.31(a) and 10 CFR 71.35(a).

3-3. Provide the fin performance calculations and the relevant portion(s) of Reference [2] “VDI Heat Atlas” listed in Section 3.3, in English, so that a review of the finned surface enlargement factor can be performed.

SAR Section 3.3.1.3 mentioned that a finned surface enlargement factor was calculated analytically according to the “VDI Heat Atlas”. A review of the calculations and the relevant portion of the reference is needed because the finned surface contributes to the thermal performance of the package.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

5.0 Shielding Evaluation

5-1 Provide the following specifications for the identified packaging components:

a. The minimum areal density of the boron-10, the maximum weight percent of boron carbide (B_4C), and the tolerances for the thickness of the {proprietary information removed} borated aluminum MMC basket components, and

b. The dimensions and tolerances of the {proprietary information removed} for both the top and bottom impact limiters.

This information should also be included in the package drawings because the drawings will be incorporated by reference as part of the certificate.

This information is needed for staff review to confirm compliance with 10 CFR 71.33(a)(5), 10 CFR 71.47, 10 CFR 71.51(a), 10 CFR 71.55(b), 10 CFR 71.55(d), 10 CFR 71.55(e), and 10 CFR 71.59.

8.0 Acceptance Tests and Maintenance Program

8-1 Provide descriptions of the acceptance tests, maintenance tests, and acceptance criteria for the fabricated packaging items that are relied on to perform a shielding function.

Sections 8.1.5.1 and 8.1.6 of the application include statements that acceptance tests of shielding components will be performed to ensure that certain characteristics of the components that are relevant for shielding are met by the fabricated components. However, the application does not include any kind of description of those tests, nor does it include the criteria that will be used in those tests to determine whether the fabricated components comply with the necessary characteristics (e.g., minimum density, composition and dimensional uniformity, minimum hydrogen content, minimum boron content, minimum thickness, lack of internal voids) as defined in the package design and needed to perform the shielding function as designed and evaluated in the application. This is particularly important for those components which are fabricated from non-standard materials, such as the borated aluminum MMC basket plates and the high molecular weight polymer components. For components made from standard materials, ensuring conformance to the material standards and the dimensions and tolerances (all of which are specified in the package drawings) is sufficient to ensure their shielding function.

The staff notes that Table 8.2-1 of the application indicates that shielding components may be replaced as needed. Thus, the replacement items would also need to undergo the same acceptance tests with the same acceptance criteria as should be described Sections 8.1.5.1 and 8.1.6 of the application. Sections 5.4.1.1, 5.4.3.2, and 9.4.1.7 of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," provide information that may be helpful in understanding the kind of information that should be provided. For items like the borated aluminum MMC basket plates, which also serve a criticality safety function, qualification and acceptance tests that demonstrate the basket plates' performance of that criticality safety function may be sufficient for demonstrating the shielding function of those plates as well (see item 2-7, above in Section 2, "Structural Evaluation" regarding neutron absorber qualification and acceptance tests).

This information is needed to confirm compliance with 10 CFR 71.85 and to ensure the package will be fabricated consistent with the design to meet the requirements in 10 CFR Part 71 Subparts E and F.

Observations:

2.0 Structural Evaluation

2-1 Revise the application to evaluate the package and describe the condition of the package to show that the most damaging package drop orientations have been evaluated.

Section 2.6.7.1 of the SAR states that the side drop is the only "credible" package orientation evaluated for free drop. Credible package orientations are not defined in the regulations, and all package orientations or "positions" that could produce maximum damage are required to be evaluated for the NCT drop test in 10 CFR 71.71(c)(7). Although not meant to be exhaustive, drop orientations that should be considered in addition to the side drop include end drops, center of gravity over corner, slap down, etc. and any other orientation expected to cause maximum damage.

This information is required to determine compliance with 10 CFR 71.71(c)(7).

2-2 Clarify the condition of the package after being subjected to a sequential application of tests for HAC and justify that the drop orientations evaluated are the most damaging.

The applicant has examined the tests specified for HAC on an individual basis but did not evaluate those tests sequentially, as required by 10 CFR 71.73 for cumulative damage. As an example, a 9m drop on one impact limiter followed by a puncture drop intended to remove it could cause the package to be more susceptible to the ensuing fire test that follows in the sequence. The configuration of the package should be selected to maximize damage in a cumulative sense. In addition, it is unclear how the drop orientations thus provided are indeed the most damaging, since package drop orientations between a side drop and center of gravity drops (e.g., slap down scenario) could result in more damage to the package.

This information is required to determine compliance with 10 CFR 71.33 and 10 CFR 71.73.

2-3 Provide the input and output files used to carry out drop simulations and stress analysis in ANSYS and LS-DYNA for NCT and HAC to support benchmarking efforts.

The applicant provided drop test results for NCT and HAC conditions that were based solely on simulation by ANSYS and LS-DYNA. As discussed in Section 2.4.10 and Attachment 2A of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," providing the staff this information to verify that the aforementioned analyses provide reasonable assurance that the regulations for NCT and HAC drops are being met would enhance the efficiency of the NRC review.

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

3.0 Thermal Evaluation

3-1 Provide input and output files of the bounding thermal analyses (NCT, HAC, steady-state, transient) that result in the highest thermal loads (e.g., highest ITS component temperatures) for the CASTOR® geo69 transport package.

Although SAR Section 3.3 and Section 3.4 described thermal models, the submittal did not provide input and output files used in the NCT and HAC thermal analyses. These files are used to verify the model inputs (e.g., boundary conditions, material properties, geometry) are consistent with the assumptions described in the SAR and that model outputs are consistent with the modeled input. Providing the analysis files before the start of the review will result in a more efficient review.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

3-2 Provide a detailed surface energy balance calculation and other technical discussion that would justify the external surface temperature of approximately 715 °F during the 1475 °F HAC fire.

SAR Table 3.4-2 indicated a surface temperature of approximately 715 °F during the fully engulfing 1475 °F HAC fire. Typically, the large radiant energy from the fire, in addition to the convection heat transfer coefficient associated with the engulfing fire, imparts a higher temperature to a package outer surface. A higher surface temperature could indicate higher temperatures for the important to safety (ITS) components within the package.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

3-3 Provide the canister and cask pressure calculation(s) so that an evaluation of the analysis can be performed.

SAR Sections 4.2.1 and 4.3.1 indicated canister and cask pressures and only briefly discussed the methodology of determining the pressures. However, in SAR Table 4.2-1 and Table 4.3-1 the references for several items' states "Calc." which NRC staff presumes is a calculation reference used to determine the canister and cask pressures. These references will help in evaluating the considerations used when determining pressure to ensure pressure loads are adequately addressed.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

3-4 Provide the calculation(s) that determined the effective thermal properties of the homogenized fuel assembly zones.

Although SAR Section 3.3.1.4 provided some discussion about the homogenized fuel assembly zones, a more detailed review is needed because effective thermal properties can have a large impact on ITS package component temperatures.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

3-5 Describe and list the fuel basket gaps and discuss how those gaps were modeled in the ANSYS thermal analyses.

Gaps and contact resistances in a thermal model can have a large effect on ITS component temperatures. Although SAR Section 3.3.1.7 briefly mentioned a bulk sensitivity analysis of the radial gaps between the basket, canister, and cask, there was no discussion of how the interconnecting plates (and their corresponding gaps and contact resistances) that make up the basket (Drawing No. 1014-DD-30984) were connected during fabrication and then modeled.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

3-6 Provide a discussion on grid sensitivity analysis to ensure the thermal model has sufficient spatial resolution for determining ITS component temperatures.

SAR Sections 3.3 and 3.4 provided thermal results for NCT and HAC. However, there was no discussion of a grid sensitivity analysis to ensure the ITS component temperatures listed in the SAR were from a thermal model with sufficient resolution.

This information is needed to determine compliance with 10 CFR 71.31(a), 10 CFR 71.35(a), and 10 CFR 71.41(a).

4.0 Containment Evaluation

4-1 Show that the package meets the containment criteria in 10 CFR 71.51 for both NCT and HAC.

The application is proposing to show the package meets the containment regulations in 10 CFR 71.51 by adhering to the American National Standards Institute N14.5, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment.” Staff noted that the applicant is claiming that the ”monolithic cask body and the lids can be considered as leak-tight, so the containment analysis can be reduced to the gasket sealing system, ” thereby not leak testing the entire cask body. However, in order to meet the containment criteria in American National Standards Institute N14.5, the entire containment boundary must undergo fabrication leak testing and meet the leak rate acceptance criteria specified in the SAR, as there is no recognized standard that allows for the assumption of monolithic materials to be leak-tight without being leak tested.

This information is needed to determine compliance with 10 CFR 71.51, 10 CFR 71.55(d), 10 CFR 71.55(e) and 10 CFR 71.59.

5.0 Shielding Evaluation

5-1 Provide the axial burnup profile data (source), moderator density profiles, and peaking factors used in the shielding evaluation for the radiation source term and shielding calculations. Also, provide information to show that the selected profiles are appropriate or bounding for U.S. commercial reactors.

The application indicates that a number of axial burnup profiles and moderator density profiles were evaluated and that a bounding profile was selected for burnup and for moderator density, including a burnup profile peaking factor. However, it appears that all this information was based on data for non-U.S. commercial reactors. Therefore, it is not clear that the evaluated and selected data and profiles are appropriate for commercial U.S. power reactors.

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

5-2 Explain how the normalized burnup profile is determined for the fuel assemblies that contain partial length rods.

The applicant states: "{proprietary information removed}." However, it is not clear how the normalized burnup profile is determined for the fuel assemblies that contain partial length rods. It is also not clear how the axial variation of the uranium load is accounted for. Because the axial uranium distribution is a key parameter for both radiation source terms and decay heat distribution along the axial direction of the fuel assembly, it should be treated explicitly in the normalization of the burnup profile or the source distribution in the shielding model.

This information is needed for the staff to confirm compliance with 10 CFR 71.33(b), 10 CFR 71.47, and 10 CFR 71.51(a).

5-3 Provide the specifications for the proposed package contents' grid spacers and top handles and information about their impact on package radiation levels.

While the application includes a number of the proposed spent fuel contents' specifications, specifications about their grid spacers and top handles, which are activated with the rest of the assembly and contribute to radiation source terms, appear to be missing. With this information missing, it also appears that the shielding analyses are missing evaluation of the radiation source term and the contributions to the package radiation levels from these components of the spent fuel contents.

This information is needed for the staff to confirm compliance with 10 CFR 71.33(b), 10 CFR 71.47, and 10 CFR 71.51(a).

5-4 Ensure that the SAR provides the maximum package radiation levels for all appropriate configurations of the contents and properties of the packaging components with temperature variation for all relevant surfaces and locations.

While the application's shielding evaluation chapter includes a number of package radiation level results, it is not clear that the reported results include all relevant combinations of the following conditions and the different configurations of the contents described in the application for the package:

- Undamaged fuel and reconfigured fuel (see item 2-6, above, in "Structural Evaluation" of the "Request for Supplemental Information," for reconfigured fuel) for NCT and HAC,
- Hot condition and cold condition of the packaging components for NCT,
- Top axial end, bottom axial end, and radial side of the package for NCT and HAC, and
- Appropriate package surfaces for NCT, 1-meter distance for NCT and HAC; and 2-meter distance, and normally occupied spaces for NCT.

The application should include sufficient information to demonstrate that the maximum axial and radial package radiation levels have been identified and provided.

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

5-5 Provide information to demonstrate that the identified maximum axial (both top and bottom) package radiation levels are in fact the maximum radiation levels for the axial ends of the package.

The shielding of the package's axial ends is not uniform across the full surface of the top and bottom axial ends of the package. For example, {proprietary information removed}. While Figure 5.3-8 of the application appears to indicate that mesh tallies were used for determining the package radiation levels at the axial ends of the package, the discussion of the tallies in Section 5.3.1.9 does not appear to describe the mesh tallies for the package axial ends (e.g., mesh division size along the package radius). Additionally, no information appears to be provided that shows the variation of the axial radiation levels with radial position of the axial mesh tallies on the surfaces of the impact limiters and at 1 and 2 meters from the impact limiters' surfaces. Thus, it is not clear how the applicant's shielding analysis accounts for variation in radiation levels on the impact limiter surfaces (and 1 and 2 meters away) due to radial variations in package shielding in the lid and base and tally location when identifying the maximum axial package radiation levels. This consideration is important for radiation levels at the package surface and the necessary distances from the package surface. Something similar to what was provided for the radial package radiation level variations in Figures 5.4-1 through 5.4-5 would be a possible method to show maximum axial radiation levels have been identified.

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

5-6 Clarify the intent of the proposed contents.

The shielding evaluations appear to be very specific and limited in terms of the contents and the evaluation of radiation source terms and subsequent evaluation of package radiation levels. For example, only certain assembly types are evaluated for high burnups and only a limited set of cooling, or decay, times are evaluated as well, which are uniquely set for the different assembly types described in the application. Also, Table 5.2-2, "Minimum Cooling Times (in Years) Needed to Reach Certain Decay Heat," of the application doesn't include evaluations of source terms for some assemblies because the minimum cooling time in Table 1.2-11 exceeds the cooling time needed to result in a particular decay heat. Other cooling times are not specified for certain decay heats because the needed cooling time is a "unreasonably long." Thus, the staff would anticipate conditioning the certificate such that entries in Table 5.2-2 which are not analyzed would not be authorized contents in the package.

Further, the shielding analyses for high burnup fuel assumes a minimum 20-year decay time; therefore, the staff anticipates conditioning the certificate for all fuel with high burnup having an absolute minimum cooling time of 20 years.

If the intent is to have a more flexible description of the authorized contents in the package certificate of compliance (e.g., maximum burnup limits, minimum enrichment limits, minimum cooling times that apply to all assemblies), either show that the existing analyses bound the additional content descriptions or modify the shielding analysis to demonstrate that radiation levels for a package containing the additional contents comply with the regulatory limits in 10 CFR Part 71.

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

7.0 Operating Procedures Evaluation

7-1 Clarify the loading and unloading operations for the package as described below, providing missing information, if any.

Based on the procedures in Section 7.1.2, "Loading of Contents" of the SAR, canister loading will only occur using a transfer cask. However, it appears, based on the procedures in Section 7.2.2, "Removal of Contents" that removal of the contents will be accomplished by putting the CASTOR® geo69 directly into the spent fuel pool, rather than transferring the canister to a transfer cask for unloading. If unloading of the contents will be accomplished using a transfer cask, revise either Section 7.2.1 or 7.2.2, as appropriate, to provide descriptions of movement of the canister from the package into the transfer cask and the preparation of the canister and transfer cask to go into the spent fuel pool. Also note that it appears that Step M3 in Table 7.2-2, "Required steps for the unloading of contents," appears to be incomplete in that there are not any sub-steps (e.g., M3.1, M3.2, etc), as there are for the rest of the table.

Clarify whether the information is not needed or is missing from the application.

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

Editorial Information:

1. Clarify whether 'transport hood' (step J7 on page 7.1-9) is a personnel barrier and its potential impacts on operation (e.g., thermal)? If not, please clarify whether it is a different enclosure that attaches to the package?