RAI

2-1 Justify the use of a non-rigid surface for the fuel bundle end drop case.

Page 10 in the calculation package, FS1-0015328, Rev. 2, states “Except for the end drop case, the fuel bundle is dropped on the rigid surface.”

There does not appear to be a clear justification that quantifies the amount of surface non-rigidity to account for the collapsing nature of the inner container that is not modeled.

This information is needed to determine compliance with 10 CFR 71.41, 71.51, and 71.73.

AREVA Response:

As shown in AREVA licensing Drawing FS1-0014159, Rev 3, Page 48/403, the inner container has a protective layer between the fuel bundle and the container external shell. The protective layer consists of 1) End Face Lumber, Item #26 and 2) Thermal Insulator, Item #34. During the end drop simulation of the fuel bundles, the lumber and the thermal insulator are included in the finite element model. Therefore the fuel bundle is not impacting direction against the inner container shell, which is treated as a rigid body. During the side drop and corner drop simulation of the fuel bundles, the inner container is treated as rigid body, the lumber and the thermal insulator are omitted and not modeled because of conservatism.

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RAI

2-2 Clarify the calculation package's apparent relevance between the ASME B&PV Code Section F-1331.4 criteria and the application of a 1.4 scale factor to the acceleration time histories, as noted on page 11 of the calculation package, FS1-0015328, Rev. 2.

The calculation package did not explain the numerical equivalence between ASME B&PV Code Section F-1331.4 criteria and the 1.4 scale factor to the acceleration time histories. Therefore, an evaluation of the LS-DYNA results could not be performed.

This information is needed to determine compliance with 10 CFR 71.41, 71.51, and 71.73.

AREVA Response:

The intent of the ASME code to define a collapse load of an inelastic analysis is to establish a minimum factor of safety for structural stability. The structure is dynamically loaded (such as a seismic load or an impact load) progressively until the structure reaches a status of instability or total collapse. Then the load at total collapse is reduced 30 % to establish as the allowable safe dynamic load. The ratio of the allowable safe dynamic load to the total collapse load is therefore 1.4 (=1/0.7), that implies there is a minimum factor of safety of 1.4. In the dynamic analysis, the external load is applied as acceleration. The dynamic force withstood by the structural member is proportional to the magnitude of the acceleration. In our impact analysis, we did not progressively apply the load to the point of total fuel bundle collapse. Instead, we multiplied the measured acceleration by a factor of 1.4 to assure that the fuel bundle can withstand the magnified acceleration without collapse. The real collapse load could be even higher but we do not intend to fully explore the real ultimate collapse load. With the confirmation that the fuel bundle can withstand the magnified acceleration without collapse, we confirmed that the measure acceleration is less than the code allowable dynamic load. This also confirmed that the minimum factor of safety of 1.4 is established.

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RAI

2-3 Quantify the maximum deformation of the end plugs, especially at the end plug welds, and indicate whether or not the deformation is an inelastic deformation.

Page 34 of the calculation package FS1-0015328, Rev. 2, appears to show deformations of the end plugs. The maximum deformation of the end plugs is needed to perform the structural evaluation.

This information is needed to determine compliance with 10 CFR 71.41, 71.51, and 71.73.

AREVA Response:

Through material specification and welding qualification, including testing, the weld is ensured to be structurally stronger than the fuel rod cladding. For the purpose of analysis, the weld and the accompanying heat-affected zone are represented as unmodified cladding. Since the cladding structural analysis demonstrated margin to failure, the higher strength weld will also have margin to failure.

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RAI

2-4 With respect to the structural performance of the ATRIUM 11 fuel:

a) Justify that all ATRIUM 11 fuel to be shipped, irrespective of manufacturer, will be bounded by the assumed mechanical properties for Zircaloy-2 in the structural evaluation (Report No. FS1-0025122, Rev. 1). Clarify that alternative cladding types are not allowable.

b) Justify that the qualification program (parameters, tooling, process limits) provided in Document No. 127-9222576-001 for the Upset Shape Welding of ATRIUM 11 end cap fuel rod welds is adequate for fuel from alternative suppliers.

Report No. FS1-0024572 states that the proposed change for incorporating the limiting AREVA ATRIUM 11 fuel assembly design does not prevent shipment of fuel supplied by other manufacturers, as long as the design meets the requirements specified in the note in Table 3-5 of the application (related to Report No. FS1-0014159). The note in Table 3-5 states that the cladding thickness and diameters defined in the table are for example purposes, but alternatives are allowable if the cladding maximum hoop stress does not exceed the specified value. The staff is unclear if the structural evaluation and weld qualification program provided in the application will be bounding to ATRIUM 11 fuel from alternative manufacturers, particularly as the cladding serves as containment boundary in this package.

This information is required to ensure compliance with 10 CFR 71.33(b)(3), 71.55(d)(2), and 71.71(c)(7), and 71.73(c)(1).

AREVA Response

2-4.a) AREVA exclusively uses Zircaloy-2 as the base material in the manufacturing of BWR fuel cladding. Other material types are not being contemplated.

2-4.b) The qualification report, 127-9222576-001, is specific to the fuel rod design, equipment, tooling setup and weld parameters and limits detailed in the report. A verification or new qualification is required if there is a change to any aspect of the process. Therefore, the qualification report results are not transferable to other designs, equipment or suppliers.

This RAI, including items (a) and (b), discusses the potential for shipment of 11x11 fuel designs from other suppliers. AREVA concludes that shipment of 11x11 fuel designs from other suppliers is not possible in accordance with SAR and supporting documentation submitted. Our reasoning for this conclusion is as shown in the discussion below.

FS1-0024572 Section 2 “Summary of Results” reads in part as follows:

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"The thermal evaluation documented in the current SAR encompasses the TN-B1 container and the fuel assembly. Since the container design is unchanged, the existing thermal evaluation for the container in Section 3 of the SAR remains applicable. Therefore, only the fuel assembly designs were evaluated. The thermal evaluation for the AREVA fuel assembly designs was completed for the 9x9, 10x10 and 11x11 arrays, and the results are shown in Table 2. Fuel designs from other fuel fabricators were not evaluated. These designs, when shipped in the TN-B1, shall meet the current requirement defined by the product of the maximum pre-pressure and the maximum Inside Radius/Thickness of 10.18653MPa absolute (9.14 x 1.1145 MPa) or less."

A literal reading of this paragraph, when isolated from both the previous section of the document and the SAR could possibly lead the reader to include that 11x11 fuel designs manufactured by companies other than AREVA could be shipped in the TN-B1, which would be an incorrect conclusion. AREVA addressed this potential issue in section 1 "Introduction" of FS1-0024572, which reads, in part, as follows"

"The introduction of the ATRIUM™[1] 11 fuel assembly design in reload quantities has necessitated a resubmittal of the TN-B1 shipping container Safety Analysis Report (SAR) (Reference [1]) as this design is outside of the previously evaluated parameters. **The evaluations supporting the new submittal have been split into separate documents. This document specifically addresses the thermal evaluation documented in Section 3 of the SAR."** (emphasis added)

The above paragraph makes it clear that the document is to support the thermal evaluation only and should not be used as a "stand alone" document to determine if any particular fuel assembly can be shipped in the TN-B1 or not. Any such reading as such is clearly incorrect.

Any 11x11 fuel assembly to be shipped within the TN-B1 must meet all of the requirements detailed in the SAR, including, but not limited to, the requirements of Table 6-1 "TN-B1 Fuel Assembly Loading Criteria" and those detailed in section 6.12 "Appendix B: 11X11 Fuel Assembly Criticality Analysis". As such AREVA concludes that requirements of the SAR are sufficiently detailed as to prevent the shipment of other 11x11 fuel designs without additional SAR revisions.

AREVA also concludes that shipment of 8x8, 9x9 and 10x10 fuel designs from other fuel manufacturers, such as the GNF-10, would be permitted under the proposed TN SAR revision, consistent with what is authorized under the current TN-B1 SAR.

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3-1 Provide the analysis that demonstrates the fuel/cladding criteria listed in Table 3-5 of the application, for new and modified BWR fuel, that "all fuel to be shipped must have a maximum pre-pressure times the maximum Inside Radius/Thickness product of 9.14 x 1.1145 MPa = 10.18653 MPa or less. Thus, all products must meet the maximum product of allowed pressure multiplied by the Inside Radius/Thickness of 10.18653 MPa."

Section 4.1.1 of the application indicates that the fuel cladding is the containment boundary. Table 3-5 of the application provides the maximum pressure and cladding dimensions to attain an allowable stress and margin of safety after the hypothetical accident conditions (HACs); some of these parameters have values that have changed approximately 30% to 40% from an earlier application. However, the updated application did not address these changes and there was no analysis provided to show that the criteria described above was appropriate for determining the Table 3-5 parameters of the new and modified fuels.

This information is needed to determine compliance with 10 CFR 71.51 and 71.73.

AREVA Response

Section 3.5.3.2 of FS1-0014159-5.0 states that the maximum allowed cladding stress is limited to 31.1 MPa at HAC and is applicable to fuel shipped in the TN-B1 having zircaloy cladding. This stress limit was based on tests where zircaloy cladding samples filled with 1.1145 MPa at room temperature did not burst at a temperature of 800°C. The HAC condition is defined as a maximum temperature of 648°C (921 K) as determined by a transient analysis after thirty minutes from the start of the fire. The use of the 31.1 MPa allowable cladding stress limit at HAC is, therefore, conservative.

The criteria in the Table 3-5 note:

$$\text{maximum fuel rod pre-pressure} * \text{maximum inside radius / thickness} \leq 10.18653 \text{ MPa}$$

is a room temperature equivalent of the 31.1 MPa fuel rod cladding allowable stress limit at HAC. By applying the Ideal Gas Law and the empirical formula for calculating hoop stress in thin wall tubing, the room temperature allowable cladding stress limit can be derived from the limit at HAC:

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3-2 Provide a quantitative analysis to demonstrate that the package, with the cumulative effect of the 10 CFR 71.73 HACs, would not exceed the regulatory release.

Section 2.7.4 of the application stated that the maximum HAC testing temperature for an earlier-designed fuel assembly was 921 K (1198°F) and that the fuel rod pressure due to accident conditions does not exceed 508 psig (522.7 psia). The application also stated that the fuel rods have a rupture pressure in excess of 520 psi (pressure value was not provided). It was not evident that the condition (e.g., stress, strain) of the modified fuel assemblies' deformed fuel rods, after the HAC 30 ft drop (end drop, side drop, etc.) and puncture tests, was quantitatively considered when analyzing at the high temperatures and pressures (e.g., 1198°F and 508 psig) of the thermal HAC and during subsequent calculations that are used for input in the certificate of compliance fuel parameter tables and Table 3-5 of the application.

This information is needed to determine compliance with 10 CFR 71.51(a)(2) and 71.73.

AREVA Response

Section 2.7.4.1[1] of FS1-0014159 (Reference 3-2-1) states that the maximum predicted HAC temperature for any fuel type is 648°C (1198°F), which is below the allowable HAC temperature limits presented for the various fuel types. Section 2.7.4.1 of Reference 3-2-1 also states that the maximum predicted HAC pressures do not exceed the allowable limits listed in Section 3.0. Section 3.1.4 defines the maximum predicted pressure for BWR fuel as 508 psia, which is below its allowable pressure presented in Section 3.5.3.2. In summary, Reference 3-2-1 demonstrates that all fuel types remain below both temperature and pressure limits required to maintain containment.

Reference 3-2-1 discussions regarding fuel rod performance during a HAC event do not detail the impact of strain experienced by the fuel rods of the dropped certification test unit assemblies described in Table 2-11 of that reference. The impact of strain experienced by the fuel rods however is fully examined in FS1-0015328 (Reference 3-2-2). As shown in Table 7.9-2 of reference 3-2-2 the final pitch change is very small. Additionally reference 3-2-2 demonstrates that at no point does the load experienced by the fuel rods exceed 70% of the plastic instability load as required by ASME Code Subsection III, Appendix F, Section F-1341.4.

During fabrication, zirconium cladding is cold-worked and annealed, as seamless tubing manufactured by drawing, extruding, or pilgering. Tubes manufactured with these processes experience a high percentage of cold work and are subsequently annealed at temperatures typically less than 700°C. Since the manufacturing anneal temperature is below the estimated

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fuel rod rupture temperatures presented in Section 3 of Reference 3-1-1, the effects of cold working are further reduced.

This information is also discussed in more detail in Section 7 “Influence of Initial Stress State” of Reference 3-2-3. Specifically it documents that “The prior drop event does not create significant stresses and strains in the radial and circumferential directions of the transverse cross-section, which is the loading plane for the heating period. Therefore, the mechanical deformation during the heating period can be considered independent of the prior small axial bending during the drop event. This is the basis for the analysis in SAR Section 3.5, where results of the closed tube pressure ballooning tests have been used to assess the avoidance of clad rupture due to internal gas pressure.”

In summary, based on the small degree of cold work experienced by the fuel during the 9m drop, and because the subsequent fire event temperature of 800°C is above the anneal temperature for the zirconium alloy tubing, the effects of the strain discussed in Reference 3-2-2 are negligible on the thermal performance presented in Reference 3-2-1. This is confirmed by real events discussed in Reference 3-2-4. This reference discusses a severe accident, where a passenger vehicle traveling the wrong way down the road impacted a tractor trailer carrying 24 unirradiated nuclear fuel assemblies with zirconium alloy cladding. In this event, the car collided with the truck, and then the truck collided with a guardrail. A fire started due to the collision, and some fuel at the front of the trailer not only experienced extreme temperature conditions above HAC, but also fell from the trailer bed. Though this fuel incurred plastic deformation in excess of the RAJ-II 9m drop CTU-2 assembly, the fuel thermal performance remained consistent with that outlined in Reference 3-2-1.

AREVA would bring to your attention that our response is very similar to that provided by GNF to an identical RAI that they received. See Reference 3-2-5. The main difference between the GNF and the AREVA response is References 3-2-2 and 3-2-3, which provides a significantly more detailed analysis, yet consistent, of the impact of strain experienced by the fuel rods during HAC.

AREVA believes that Section 2.7.4 of the SAR, along with supporting documentation, is adequate as is to support determination of compliance with 10 CFR 71.51(a)(2) and 71.73.

References

3-2-1 FS1-0014159 “AREVA TN-B1 Docket No. 71-9372 Safety Analysis Report” Rev 5

3-2-2 FS1-0015328 “Structural Analyses of the AREVA Atrium-11 LTA Fuel Assembly in the RAJ-II Container during Normal and Accident Transport Conditions” Rev 2

3-2-3 FS1-0024572 “TN-B1 Container Thermal Analysis Applicability”

3-2-4 NUREG/CR-5892 “A Highway Accident Involving Unirradiated Nuclear Fuel in Springfield Massachusetts on December 16, 1991”

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3-2-5 ADAMS document M170085 “Draft Responses to the NRC RAIs for the RAJ-II Transportation Package”.

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HAC stress allowable = (fuel rod fill pressure absolute * HAC temperature / ambient temperature – 1 atmosphere) * inside cladding radius / cladding thickness

Substituting:

31.1 MPa = (1.1145 MPa * 921K / 293K – 0.101325 MPa) * r/t

Solving for the r/t necessary to achieve the maximum stress allowable:

r/t = 9.14

The maximum cladding stress allowable at room temperature (Table 3-5 note) based on the absolute initial fuel rod fill pressure is:

RT stress allowable = fuel rod fill pressure absolute * r/t

= 1.1145 MPa * 9.14

= 10.18653 MPa (absolute)

Table 3-5 of FS1-0014159-5.0 provides an example evaluation of one fuel design type per array size using the limits and methods described in Section 3.5.3. The maximum allowed cladding stress limit has remained unchanged between applications. The maximum allowed fuel rod initial fill pressure, however, is dependent on the fuel design. Design differences that can impact the maximum initial fuel rod pressure allowed between fuel design types include cladding ID and OD, and liner and non-liner cladding. Changes to these parameters will impact maximum allowed initial fuel rod fill pressure, and the initial fill pressure is specific to the cladding design used. The 11x11 fuel design is currently only offered with non-liner cladding. To provide alignment in the fuel designs presented in Table 3-5 for each array in the updated application as well as the most limiting in terms of maximum initial fill pressure and minimum cladding thickness, liner cladding was used to populate the table for the 9x9, 10x10 and 11x11 arrays. Section 3.5.3.2 of FS1-0014159-5.0 states that the thickness of the liner is not included in the minimum cladding thickness when determining the maximum internal pressure. The use of liner cladding in the evaluation for the 9x9 and 10x10 fuels, therefore, significantly reduces the allowed maximum initial fuel rod fill pressure for these designs. The fuel types shown in Table 3-5 of the updated application and other fuel designs were evaluated in FS1-0024572-2.0. AREVA transmitted this document to the USNRC on February 17, 2017, in letter number TJT:17:007 (Timothy J Tate to USNRC Document Control Desk, "Docket No. 71-9372,

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AREVA TN-B1 Shipping Container, Application For Approval For Incorporation of ATRIUM 11 Fuel Assemblies").

To improve the clarity of the application AREVA suggests enhancing Table 3-5 by providing the evaluation for both the liner and non-liner clad fuel designs transported in the TN-B1 shipping container. AREVA would also suggest adding a note to the Table 5 "Fuel Rod Parameters" of the CoC to read as follows: "Maximum product of allowed pressure multiplied by Inside Radius/Thickness, excluding the liner thickness (if applicable), may not exceed 10.18653 MPa." Addition of this note should also make the CoC more clear to users as well.

AREVA will also update FS1-0024572 to show liner and non-liner clad fuel designs transported in the TN-B1 shipping container. We will also submit revised text for Sections 2.7.4.1, 3.1.4, 3.4.2 and 3.5.3.2 of the SAR to specify maximum pressures for liner and non-liner cladding.

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RAI

4-1 Provide updated release and leak rate calculations under NCT and HAC for the new ATRIUM 11 fuel design which show the design will meet leak rate regulation requirements.

In Section 4.1.1 of the application, the applicant states that the fuel is leak tested to demonstrate that it is leak tight to 1x10-7 cm3/s. However, following one of the drop tests, fuel rods were leak tested and shown to have a leak rate of 5.5 x10-6 cm3/s, which is not leak tight. The applicant did not update the containment chapter to reflect the new ATRIUM 11 fuel design proposed in this amendment so the current release and leak rate calculations provided in Section 4.2.2 of the application are for the previously approved 10x10 fuel design. The applicant should provide updated release and leak rate calculations for NCT and HAC for the new ATRIUM 11 fuel design. Standard review plan guidance NUREG-1609 suggests that ANSI N14.5 provides an acceptable method to determine the maximum permissible volumetric leakage rates based on the allowed regulatory release rates under both normal conditions of transport and HACs.

This information is needed to determine compliance with 71.51(a) and (b).

AREVA Response

See the proposed revision to section 4.

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4. CONTAINMENT

4.1. DESCRIPTION OF THE CONTAINMENT SYSTEM

4.1.1. *Containment Boundary*

TN-B1 container is limited to use for transporting low enriched uranium, nuclear reactor fuel assemblies and rods. The radioactive material is bound in sintered ceramic pellets having very limited solubility and has minimal propensity to suspend in air. The pellets are sintered at temperatures greater than 1,600°C. These pellets are further sealed into zirconium alloy cladding to form the fuel rod portion of each assembly. The primary containment boundary for the TN-B1 package is the fuel cladding. Design and fabrication details for this cladding are provided in Section 1.2.3. The containment system includes the ceramic sintered pellet, clad in zirconium tubes which are contained in a stainless steel box which is contained in another stainless steel box.

There are no penetrations in the fuel cladding when shipped. The fuel cladding after loading with the pellets is pressurized with helium and end plugs are welded on to close the rod. These welds are designed to withstand the rigorous operating environment of a nuclear reactor. The fuel is leak tested to demonstrate that it is leak tight ($<1x\ 10^{-7}$ atm-cc/s).

4.1.2. *Special Requirements for Plutonium*

This section is not applicable since the package is not being used for plutonium shipments.

4.2. GENERAL CONSIDERATIONS

4.2.1. *Type A Fissile Packages*

The Type A fissile package is constructed, and prepared for shipment so that there is no loss or dispersal of the radioactive contents and no significant increase in external surface radiation levels and no substantial reduction in the effectiveness of the packaging during normal conditions of transport. The fissile material is bound as a ceramic pellet and contained in a zirconium fuel rod. These rods are leak tested prior to shipment to assure their integrity. Chapter 6.0 demonstrates that the package remains subcritical under normal and hypothetical accident conditions.

4.2.2. *Type B Packages*

The Type B fissile package is constructed, and prepared for shipment so that there is no loss or dispersal of the radioactive contents and no significant increase in external surface radiation levels and no substantial reduction in the effectiveness of the packaging during normal conditions of transport.

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The package satisfies the quantified release rate of 10 CFR 71.51 by having a release rate less than 10^{-6} A_2/hr as demonstrated below.

A_2 = 0.18 Ci, therefore$10^{-6}A_2$ = 1.8 x 10^{-7} Ci/hr

The mass density of UO_2 in an aerosol from NUREG/CR-6487, page 17 is 9 x 10^{-6} g/cm^3. Specific Activity of fuel material is 1.4 x 10^{-5} Ci/g UO_2 (7.89 Ci/562kg UO_2).

Leak rate at 1 x 10^{-7} atm-cm^3/s (3.6 x 10^{-4} cm^3/hr) is equal to 1 x 10^{-6} atm-cm^3/s (3.6 x 10^{-3} cm^3/h) when pressurized to 10 atm. Assuming that the pressure is further increased due to temperature the leak rate is assumed to increase by an additional factor of 10 so that it is equal to3.6 x 10^{-2} cm^3/h.

Release rate = 3.6 x 10^{-2} cm^3/hr x 1.4 x 10^{-5} Ci/g UO_2 x 9 x 10^{-6} g /cm^3

= 4.5 x 10^{-12} Ci/h

Much less than the 1.8 x 10^{-7} Ci/hr limit.

4.3. CONTAINMENT UNDER NORMAL CONDITIONS OF TRANSPORT (TYPE B PACKAGES)

The nature of the contained radioactive material and the structural integrity of the fuel rod cladding including the closure welds are such that there will be no release of radioactivity under normal conditions of transport. The welded close containment boundary is not affected by any of the normal conditions of transport as demonstrated in the previous chapters. The pressurization that could be seen by the containment boundary is far below the normal conditions the fuel experiences while in service.

4.4. CONTAINMENT UNDER FOR HYPOTHETICAL ACCIDENT CONDITIONS (TYPE B PACKAGES)

The sintered pellet form of the radioactive material and the integrity of the fuel rod cladding are such that there will be no substantial release of radioactivity under the Hypothetical Accident Conditions. Before and after the accident condition testing the rods were helium leak tested demonstrating leak tightness. Similar fuel rods have been tested at temperatures and resulting pressures that will be seen by fuel shipped in the TN-B1.

10 CFR 71.51 requires that no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package. The following qualitative assessment demonstrates that the performance requirement of 10 CFR 71.51(a)(2) will be satisfied.

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Table 1-4 shows the calculated A_2 for the mixture of the maximum radionuclide content in the package is 0.18 Ci. The total radioactivity in the package using the maximum isotopic values is 7.89 Ci. The mass of UO_2 equivalent to an activity of 7.89 Ci is 562kg (281kg UO_2/assembly x 2 assemblies) which yields a mass to activity ratio of 71.2kg UO_2/Ci. The mass equivalent A_2 is therefore 12.8 kg UO_2.

Following the drop test, fuel rods were leak tested and shown to have a very low leak rate of He at a rate of 5.5 x 10^{-6} cm^3/s. Over one week this is equal to 3.3 cm^3 (5.5E-6 cm^3/s x 6.05E5 s/wk = 3.3 cm^3). The tested assembly had 91 fuel rods while the ATRIUM 11 has 112 fuel rods. As a result a conservative assumption was made that the amount released would increase proportionately to the number of fuel rods. This was determined to be 4.1 cm^3 (3.3 cm^3 x 112 rods/91 rods). Conservatively assuming that the density of the radioactive material is 10g/cm^3 and using the A_2 mass above of 12.8 kg of UO_2, the UO_2 would have a volume of 1,280 cm^3. This is much greater than the volume leaked. This calculation is extremely conservative since the UO_2 would predominantly stay in a ceramic form and not be available for dispersion.

Test fuel rods as described in Section 2.0 have been baked at 800°C for over 30 minutes and did not leak.

Additionally, the large mass, 12.8 kg, of material required to exceed the A_2 would require a catastrophic failure of the rod, significant leak of the inner and outer container.

Dose rates are less than the 10mSv/hr under any condition because of the low specific activity and low abundance of gamma emitters in the fuel.

Based on this evaluation, it is demonstrated that the package meets the containment requirements of 10 CFR 71.51

4.5. LEAKAGE RATE TESTS FOR TYPE B PACKAGES

During manufacturing each fuel rod is He leak tested to demonstrate that it is leak tight (<1x 10^{-7}atm-cc/s). There are no leak rate requirements for the inner and outer packaging.

4.6. APPENDIX

None

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4-2 Specify the minimum cladding thickness of the fuel rods throughout the application.

Throughout the application, particularly in Table 6-2 and Table 6-58, the applicant requested changes that would allow the fuel to effectively have a cladding thickness of zero. Since the fuel cladding is part of the containment boundary, the possibility of a zero clad thickness is not appropriate; the fuel cladding thickness in the respective Fuel Rod Parameter and Fuel Assembly Parameter tables throughout the application and CoC should reflect the values that have been demonstrated to meet structural, thermal, and containment performance during NCT and HAC.

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

Response

AREVA will submit a revised Table 4 to the CoC which specifies the minimum cladding thickness.

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6.0 Criticality

6-1 Provide the exact material composition for the ATRIUM fuel, revise Section 1.2.3.1 or 1.2.3.2 of the application as necessary, and demonstrate that the current criticality safety analyses are appropriate for the proposed contents or provide new criticality safety analyses consistent with the material composition.

On page 33 of the application, the applicant states the following:

> "The nuclear fuel pellets located in rods and contained in the packaging are uranium oxides primarily as UO_2 and U_3O_8."

On page 34 of the application, the applicant states the following:

> "Where the contents of the packaging is enriched reprocessed uranium or other origin uranium not exceeding the values in Table 1-3, the packaging is considered to contain Type B quantities."

On page 35 of the application, the revision bar on the side of the text appears to indicate that the material composition is applicable to the ATRIUM fuel. Also, Table 1-3 indicates that the material composition includes transuranic isotopes such as NP-237, Pu-239, Pu–240, etc. However, in the criticality safety analyses, the applicant used UO_2 as the material composition for the fuel. In addition, the criticality safety analyses for the ATRIUM fuel assume UO_2 as the fuel. With this information, the staff is unable to determine the material composition for the new fuel to be transported in the package. The staff needs the exact material composition to determine if the package meets the regulatory requirements for criticality safety. The applicant needs to revise the application to clearly specify the material composition of the ATRIUM fuel. If the ATRIUM fuel is made from reprocessing of previously irradiated fuel, the applicant needs to provide criticality safety analyses consistent with the material composition or demonstrate that the current analyses are adequate for the proposed contents.

This information is needed to determine compliance with 10 CFR 71.33(b)(2).

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PROPOSED RESPONSE:

The SAR will be revised to remove the reference to U_3O_8. The ATRIUM-11 assemblies will contain only UO_2 or UO_2-Gd_2O_3 material.

The effect of the transuranic isotopes listed in Table 1-3, specifically Pu-239 which would have the greatest positive impact on the reactivity, is considered in Section 6.6.2.1 of the document.

6-2 Justify why the material density for $UO_2 + Gd_2O_3$ is used for pure UO_2 rods in the criticality safety analyses.

On page 407 of the application, the applicant calculated the material density for the fuel rods that are loaded with gadolinium burnable poison. The applicant used a density of 10.96 g/cm^3 for UO_2 and 7.407 g/cm^3 for Gd_2O_3 to determine the material density of the fuel rods containing gadolinium trioxide. The result shows that a density of 10.763 g/cm^3 is appropriate. However, it appears that the applicant used 10.763 g/cm^3 as the material density in the criticality safety analyses for non-poisoned fuel rods as well. The material specifications listed in Table 6-60 and the sample input file in Section 6.12.10.1 confirm this observation. The applicant needs to justify why it is appropriate to use the material density of $UO_2 + Gd_2O_3$ for pure UO_2 rods in the criticality safety analyses or revise the criticality safety analyses as necessary.

This information is needed to determine compliance with 10 CFR 71.55(a), 71.55(b), 71.55(d), 71.55(d), 71.59(a), 71.59(b), and 71.59(c)

PROPOSED RESPONSE:

On page 407 of the document, the theoretical density of UO_2 is given as 10.96 g/cm^3. This value is then multiplied by the % theoretical density of 98.2 to get the density for UO_2 of 10.763 g/cm^3 (ρ_{UO2}).

On page 407, the density of Gd_2O_3 is given as 7.407 g/cm^3 (ρ_{Gd})

The 10.763 g/cm^3 value is shown correctly in Table 6-60 as the density for the UO_2 material.

This value is also input correctly for both the pure UO_2 rods (material 1 of the sample input of Section 6.12.10.1) and for the UO_2 material within the $UO_2 + Gd_2O_3$ material (material 4 of the of the sample input of Section 6.12.10.1). The density of 7.407 g/cm^3 is input for the Gd_2O_3 material within the $UO_2 + Gd_2O_3$ material (material 4 of the of the sample input of Section 6.12.10.1). The density of the combined $UO_2 + Gd_2O_3$ material is calculated by the code during execution using the input densities and the volume fractions for the materials (also given on page 407 of the document). The density calculated by the code for $UO_2 + Gd_2O_3$ is 10.691 g/cm^3. This value is shown in Table 6-60.

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6-3 Clarify specifically what limitation of the SCALE code prevents modeling of the melt layer on the surface of fuel cladding and justify that the alternative treatment is appropriate and conservative.

On page 400 of the application, the applicant states the following:

> "In the HAC models, 10.2 kg of polyethylene is assumed to melt onto the fuel rods. Due to modeling limitations of the SCALE 6.1.3 software, the polyethylene is smeared into the cladding."

However, it is not clear what limitation of the SCALE code prevents modeling of the melt layer on the surface of fuel clad. The applicant needs to describe the limitation of the code that prevents one from modeling the polyethylene foam layer on the cladding. The applicant also needs to provide justification that the alternative treatment of smearing the polyethylene into the cladding is an appropriate and conservative treatment.

This information is needed to determine compliance with 10 CFR 71.55(a), 71.55(b), 71.55(d), 71.55(d), 71.59(a), 71.59(b), and 71.59(c).

PROPOSED RESPONSE:

The wording quoted above from page 400 of the application was unintentionally vague. A better description in the second quoted sentence would be: "Due to limitations in the lattice cell modeling of the SCALE 6.1.3 software, the polyethylene is smeared into the cladding" where the underlined section contains the change. This is the same method and limitation that was used for the previous analysis (see page 227, 250, or 268 of the document).

This page in the SAR will be updated to show the revised wording.

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