UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 15, 2017

Mr. Tim Tate
Environmental, Health, Safety,
and Licensing
AREVA Inc.
2101 Horn Rapids Road
Richland, WA 99354

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9372, FOR THE MODEL NO. TN-B1 PACKAGING (CAC NO. L25164)

Dear Mr. Tate:

By letter dated November 18, 2016, as supplemented on February 17, 2017, you submitted an application to revise Certificate of Compliance No. 9372 for the Model No. TN-B1 transportation package. You requested approval for the addition of ATRIUM 11 fuel assemblies as authorized content and changes to parameters related the 8 x 8, 9 x 9, and 10 x 10 fuel assemblies.

In connection with the staff's review, we need the information identified in the enclosure to this letter. We request you provide this information by June 16, 2017. Inform us at your earliest convenience, but no later than May 31, 2017, if a substantial date change is needed. To assist us in re-scheduling your review, you should include a new proposed submittal date.

If you have any questions regarding this matter, please contact me at 301-415-6999.

Sincerely,

/RA/

Norma García Santos
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9372
CAC No. L25164

Enclosure:
Requests for Additional Information

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9372, FOR THE MODEL NO. TN-B1 PACKAGING (CAC NO. L25164) – DATE: MAY 15, 2017

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Request for Additional Information
AREVA Inc.
Docket No. 71-9372
Model No. TN-B1 Package

By letter dated November 18, 2016, as supplemented on February 17, 2017, AREVA Inc. (AREVA) submitted an application for amendment of the Model No. TN-B1 transportation package, Certificate of Compliance No. 9372. AREVA requested approval of changes made to reflect the addition of ATRIUM 11 fuel assemblies and changes to the other fuel assemblies as identified in Table 3-5 of the application.

This requests for additional information (RAIs) identify information needed by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) in connection with its review of the application. The staff used NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," in its review of the application.

Each individual RAI describes information needed by the staff to complete its review of the application and to determine whether the applicant has demonstrated compliance with the regulatory requirements of 10 CFR Part 71.

2.0 Structural

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.

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.

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.

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).

3.0 Thermal

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.

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.

4.0 Containment

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 1×10^{-7} cm^3/s. However, following one of the drop tests, fuel rods were leak tested and shown to have a leak rate of $5.5\ \times10^{-6}$ cm^3/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).

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).

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).

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)

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).

6-4 Clarify if axial and horizontal variations of enrichment are allowed in the ATRIUM fuel assembly. If so, provide justification that the assumptions in the uniform axial enrichment analyses bound the axially varied enrichment configurations.

One of the requested allowable new contents for the TN-B1 package is the ATRIUM BWR fuel assembly. As a common design feature, BWR fuel assembly designs often include variation of fuel enrichment along the axial direction as well as across the planar direction. However, it is not clear from the application if the ATRIUM fuel has this feature. The applicant needs to clarify whether axial and horizontal variations of enrichment are allowed in the ATRIUM fuel assembly. If so, the applicant needs to provide updated criticality safety analyses for the case of axial varying enrichment or justify that the assumptions in the uniform axial and horizontal enrichment is bounding.

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).