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NUCLEAR REGULATORY COMMISSION
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August 31, 2015

Mr. Glenn Mathues, Licensing Engineer
AREVA TN
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION – APPLICATION FOR AMENDMENT NO. 14 TO THE STANDARDIZED NUHOMS® SYSTEM

Dear Mr. Mathues:

By letter dated April 16, 2015, AREVA, Inc. (AREVA) submitted an application to amend Certificate of Compliance (CoC) No. 1004 for the Standardized NUHOMS® System.

The staff has determined that further information is needed to complete its technical review. The request for additional information (RAI) is in the enclosure. Your response should be provided by November 6, 2015. If you are unable to meet this deadline, please notify us in writing, at least one week in advance, of your new submittal date and the reasons for the delay. The staff will then assess the impact of the new submittal date and notify you of a revised schedule.

Please reference Docket No. 72-1004 and TAC No. L25011 in future correspondence related to this request. If you have any questions regarding this matter, I may be contacted at (301) 415-0606.

Sincerely,

/RA/

Jose R. Cuadrado, Project Manager
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1004
TAC No. L25011

Enclosure: Request for Additional Information

Mr. Glenn Mathues, Licensing Engineer
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Request for Additional Information

Docket No. 72-1004
Certificate of Compliance No. 1004
Amendment No. 14 to the Standardized NUHOMS® System

By letter dated April 16, 2015, AREVA, Inc. (AREVA) submitted an application to amend Certificate of Compliance (CoC) No. 1004 for the Standardized NUHOMS® System.

This request for additional information (RAI) identifies additional information needed by the NRC staff in connection with its review of the amendment application. The requested information is listed by topic and/or page number in the application and associated documentation. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" was used by the staff in its review of the application.

Each individual RAI section 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.

3.0 Structural Evaluation

3-1 Clarify failed fuel can drawing details used for the NUHOMS 32PTH1F.

Section B-B of drawing NUH32PTH1-1007-SAR (page 536 of 755 of document E-38701 CoC) depicts the bottom lid of the fuel can (item 2) adjacent to the liner (item 1), but it is unclear how these two items are connected. In addition, it is unclear how the over sleeve (item 6) is attached to the remainder of the failed fuel can, or the weld size used to connect the top cover plate (item 8) to the tool socket (item 9) and the tool socket (item 9) to the tool socket closure plate (item 10).

This information is required by the staff to determine compliance with the regulatory requirements of 10 CFR 72.236(b).

3-2 Clarify how digital radiographic images or real time radioscopy images will be equivalent to film radiography.

Change No. 13 of the application requests the use of digital radiographic examination as an alternative to film images. Describe how digital radiographic examination will be equivalent to film radiography in terms of image detail (mega pixel count, frames per second etc.). Staff is concerned that details captured by film may not be captured using digital radiographic examination.

This information is required by the staff to determine compliance with the regulatory requirements of 10 CFR 72.236(l).

3-3 Describe the effect that reduced concrete density would have on tornadic, seismic, and flood based analyses on the HSM-H module.

It is unclear what effect reduced density of concrete will have on analyses that consider sliding and tip-over analyses due to seismic, tornadic, and flood induced loading.

This information is required by the staff to determine compliance with the regulatory requirements of 10 CFR 72.236(b) and 10 CFR 72.236(l).

4.0 Thermal Evaluation

4-1 Clarify in Section T.4.6.10.2 of the application how the design basis values were determined to be bounding. A revision to the sensitivity study may be necessary.

Compared to Tables T.4-12 and T.4-14 of the application, it does not appear that the design basis values provided in Section T.4.6.10.2 of the application are bounding. The temperatures for the dry shielded canister (DSC) in the transfer cask (TC) at 120°F ambient in vertical transfer appear to be higher than the design basis values provided in Section T.4.6.10.2 of the application.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f).

4-2 Clarify in Section Y.4.6.11 of the application how Load Case #S3 was determined to be bounding. A revision to the sensitivity study may be necessary.

Compared to Tables Y.4-9 and T.4-10 of the application, it does not appear that Load Case #S3 described in Table Y.4-9 of the application is bounding. The temperatures for the DSC in Load Case #T6 for normal transfer at 100°F with a transient at operation time equal to 15.75 hours appear to be higher than the Load Case #S3 values provided in Section Y.4.6.11 of the application.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f).

4-3 Clarify the use of the fuel assembly material properties in the ANSYS file, “Mat69BTH.inp” for the 69BTH thermal evaluation.

The fuel assembly material properties in the ANSYS file Mat69BTH.inp appear to be slightly higher at temperatures greater than or equal to 400°F compared to the thermal conductivity values in Table 2, “Bounding Effective Properties for BWR Fuel Assembly” of Section Y.4.2 of the application.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f).

4-4 Provide a thermal analysis that addresses the 61BTH DSC with 61 damaged fuel assemblies as rubble during hypothetical accident conditions.

The application stated in Section T.4.6.11 of the application that high burnup damaged fuel assemblies can experience further damages during the postulated drop accident, in the worst case turning into rubble at the bottom of the DSC. The application did not address in a thermal analysis the effect on structures, systems, and components (SSCs) important to safety in this rubblized fuel scenario.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f) and 72.236(l).

4-5 Provide a thermal analysis, or show how a previous analysis is bounding, that shows the 61BTH DSC maximum peak cladding temperature for the intact fuel assemblies for normal, off-normal, and accident conditions.

Table 1-1t (page T-31) of the Technical Specifications states that there could be up to 61 damaged fuel assemblies in the 61-BTH DSC. If less than 61 damaged fuel assemblies are stored, the balance may be intact or dummy assemblies. The thermal analysis assumes the high burnup damaged fuel assemblies can experience further damages during the postulated drop accident, in the worst case turning into rubble at the bottom of the DSC. The applicant did not address the predicted peak cladding temperature for the intact fuel assemblies for the bounding scenario where there could be anywhere from 1 to 60 intact fuel assemblies with up to 61 damaged fuel assemblies (such that the total of intact and damaged fuel assemblies adds up to 61). Technical Specifications Figure 1-25 describes further that up to four of the damaged fuel assemblies in the corner locations could be failed fuel assemblies with the remaining intact in a 61BTH basket, the effect of this on the 61BTH intact fuel assemblies was also not addressed thermally. The failed fuel assemblies also should be considered in determining the bounding scenario above.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f).

4-6 Provide a discussion, and if necessary a thermal analysis, that addresses the effect of the rubblized fuel scenario during accident conditions for the four failed fuel cans (FFCs) in the 32PTH1F DSC on SSCs important to safety.

The application stated in Section U.4.9 of the application that the four failed fuel assemblies in FFCs in the 32PHT1F DSC are assumed to transform into rubble during accident conditions. While a discussion was provided to address the effect on the intact fuel, the application did not address the effect on SSCs important to safety in this rubblized fuel scenario.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f) and 72.236(l).

4-7 Provide a discussion on the thermal analysis of the 37PTH to include poison rod assemblies (PRAs).

The addition of PRAs could impact the thermal effective conductivity (Keff), which could also impact the predicted peak cladding temperature and component temperatures. Any potential thermal impact, the reasoning for no thermal impact, or reference to a bounding analysis, should be discussed in the application.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(f).

4-8 Address the following in the Technical Specifications (T.S.):

a. On page F-26, Figure 1-25, Note 1, it is not clear if the "remaining intact" refers to only corner "A" locations if less than four damaged or failed fuel assemblies are loaded, or if "remaining intact" refers to all remaining "A," "B," and "C" locations being loaded with intact assemblies.
b. On page F-26, Figure 1-25, Note 1, it is not clear if there could be a combination of up to four failed and damaged assemblies, or up to four only failed assemblies, or up to four only damaged assemblies.
c. On page F-26, Figure 1-25, Note 3, it is not possible to put the first 57 damaged assemblies in the interior/edge "C" locations where there is space for 45 assemblies. This note should be additionally clarified to address the "B" locations.
d. On page F-32, Figure 1-28a, staff suggests changing Note (2) to the following, "Adjust payload to maintain total canister heat load within the specified limit." This change is consistent with similar notes in other T.S. figures (e.g. Figures 1-25a, 1-25b, 1-38). Based on the suggested change, it is clearer that the heat load can be below 31.2 kW and does not have to be maintained at 31.2 kW.
e. On page F-42, Figure 1-38, Note 4, based on pages Y.4-28 and Y.4-29 of the application, it appears the note on the LaCrosse fuel should not only apply to the maximum decay heat per DSC (kW), but it should also apply to the entire row, "Max Decay Heat (kW/FA)" due to the LaCrosse fuel assembly shorter active fuel length. For example, this was done appropriately in T.S. Figure 1-31.

This information is necessary to determine compliance with the regulatory requirements of 10 CFR 72.236(a).

7.0 Criticality Evaluation

7-1 Revise the application to clarify the materials, geometry, and criticality model of failed fuel and the rod storage basket in the 32PTH1 DSC.

Section U.6 states that: “Failed fuel is defined as fuel debris or fuel rods that have been removed from a fuel assembly and placed in a secondary container, such as a rod storage basket (RSB).” However, there is no description of the materials of construction or geometry of the RSB. Given that failed fuel may consist of fuel debris with essentially no structural capability, the fuel material can be expected to fill whatever container it is placed in. Therefore, the geometry of the RSB defines the expected geometry of the fuel material, and may be very different from the assembly-like geometry assumed in the criticality model, unless otherwise constrained. For example, if the RSB consists of an array large diameter steel tubes, fuel material may consolidate in the bottom of the tubes, essentially creating an array of shorter, larger diameter fuel rods, which may be more reactive than the configuration modeled. The criticality safety chapter of the SAR should be modified to detail the material and geometric features of the RSB that are important for criticality safety, and to model configurations that are representative of the actual RSB geometry.

Section U.6 also states that each RSB in the 32PTH1 DSC is modeled as a 10 x 10 array of various types of failed fuel rods, without cladding. The proposed mass limit in Table 1-1aa of the Technical Specifications is 250 kg uranium. Modeling 100 rods of any fuel type will result in significantly less than the proposed 250 kg uranium limit (e.g., for a WE 17x17 fuel assembly, 100 rods is roughly equivalent to 180 kg uranium). The criticality analysis should be revised to consider the maximum fuel mass that is allowed in Table 1-1aa, or Table 1-1aa should be revised to limit the failed fuel mass to the maximum considered in the criticality analysis. Also note that the mass limit in Table 1-1aa should be stated as kg *initial* uranium.

Additionally, Section U.6 states that “loose fuel debris not contained in an RSB may also be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at least 10 in. above the top of the bottom shield plug of the DSC.” Table 1-1aa of the Technical Specifications does not include this configuration, and limits failed fuel to fuel material in an RSB, placed in a failed fuel can. Either revise Table 1-1aa to include the loose fuel debris description and evaluate for this contents configuration in the criticality analysis, or remove this description from Section U.6.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

7-2 Revise the application to clarify the specification for poison rod assemblies (PRAs) used for criticality control for WE 17x17 fuel assemblies stored in the 37PTH DSC, and revise the Technical Specifications to provide the minimum ^{10}B requirement for PRA rods, and minimum number and configuration of rods per PRA.

Appendix Z contains several discrepancies regarding the specification for PRAs used in the 37PTH DSC. First, Figure Z.1-1 shows that PRAs for the WE 17x17 fuel assembly should have 24 rods, while Table Z.2-5a states a minimum of 8 rods. Also, Table Z.2-5a states that the minimum required B_4C content per rod is 0.60 g/cm (0.45 g/cm modeled), while Table Z.6-45 states that the minimum required is 0.61 g/cm (0.46 g/cm modeled). The application should be revised to include a single, uniform specification for PRAs used in the 37PTH DSC.

Also, Section Z.6 should be revised to show how the eight PRA rods are arranged in the criticality model. A WE 17x17 fuel assembly has 24 guide tube locations where PRA rods could potentially be located. Section Z.6 should state which of these locations contain a PRA rod.

Additionally, the description of the PRA criticality model in Section Z.6.2 states that the B_4C radius used is 0.38 cm, which conflicts with the nominal dimension in Figure Z.1-1 (diameter of 0.362 inches equivalent to radius of 0.414 cm). The PRA specification should provide a single nominal radius with a tolerance (or a minimum and maximum), and the associated criticality analysis should evaluate sensitivity to this dimension to determine the limiting configuration.

Finally, since the PRAs are relied on for criticality safety, the minimum ^{10}B requirement for individual PRA rods should be included in the Technical Specifications, as it is for neutron absorber plates.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

7-3 Revise the application to clarify damaged and intact fuel location and initial enrichment requirements for the 61BTH DSC.

Figure 1-25 of the Technical Specifications provides instructions for locations of damaged, failed, and intact fuel assemblies in the 61BTH DSC, and Table 1-1x provides maximum initial enrichments for these fuel types. It is not clear how more than 12 but less than 57 damaged fuel assemblies would be loaded into the DSC. Note C of Figure 1-25 states that: “If loading more than 12 damaged assemblies, place first 57 damaged assemblies in the interior/edge "C" locations.” However, it does not state where intact fuel should be loaded when loading between 12 and 57 damaged fuel assemblies (other than in the four corner locations), or what the enrichment limits are for intact fuel assemblies in “C” locations in this case. The notes for Figure 1-25, and any associated text in Section T.6, should be revised to clarify the loading of between 12 and 57 damaged fuel assemblies, and should be supported by the criticality analysis.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

7-4 Revise Sections T.6 and U.6 to provide an explicit validation of SCALE 6.0 for the 61BTH DSC 57 damaged fuel assembly and 32PTH1 failed fuel assembly analyses, respectively. Also, clarify which cross section library was used with these analyses.

Both the 61BTH 57 damaged fuel assembly and 32 PTH1 failed fuel assembly analyses include new criticality calculations using SCALE 6.0. These calculations are not accompanied by an explicit code benchmarking analysis. Rather, these sections attempt to show applicability of the existing SCALE 4.4a validation to calculations using SCALE 6.0. American Nuclear Society (ANS) Standard 8.24, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, states that: “The calculational methods and analysis techniques used to analyze the set of benchmarks shall be the same as those used to analyze the system or process to which the validation is applied.” Code-to-code comparison is not considered an acceptable criticality safety method validation approach in industry standards, and is not considered acceptable by staff. Note that Table U.6-53 shows that SCALE 6.0 predicts lower k_{eff} values with a Δk significantly greater than a standard deviation. Sections T.6 and U.6 should be revised to include an explicit benchmarking analysis of SCALE 6.0 as used in the 61BTH and 32PTH1 criticality analyses.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

7-5 Revise the apparent typographical error in Section U.6.4.2 regarding the version of SCALE used for 32PTH1 failed fuel assembly analysis.

This section states that “representative values from Table U.6-29 are selected to be evaluated in SCALE *6.6* to determine a bias …” As the latest released version of SCALE is 6.1, this section should be revised to state which version of SCALE was used for the 32PTH1 failed fuel assembly analysis.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

7-6 Revise Section Z.9 to include testing to demonstrate that the minimum required amount of ^{10}B is present and uniformly distributed within the poison rod assembly.

NUREG-1536 states: "Neutron absorber materials require both qualification and acceptance testing to provide assurance that the control of criticality by absorbing thermal neutrons will be assured in systems designed for nuclear fuel storage, transportation or both." Since the ^{10}B content of the poison rod assemblies is relied on for criticality safety, qualification and acceptance testing requirements should be provided in Section Z.9.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

8.0 Materials Evaluation

8-1 Provide additional detailed information regarding the sampling methods to be used for the proposed B-10 volume density testing, which support the basis of 95% probability, 95% confidence level or better in the areal density (i.e., uniformity) of boron carbide (B-10) neutron absorber for acceptance testing. Provide the details of the sample coming from a normal population, or alternative methods to be used.

The application proposes to add an option of using B-10 volume density measurement, in addition to the already authorized neutron attenuation method, as an approved method for acceptance testing for measuring boron density and uniformity of neutron absorber materials. Although American Society for Testing and Materials (ASTM) Standard Practice C1671-15 recommends both methods as acceptable for acceptance testing, and the volumetric method is specifically recommended in ASTM Standard Test Methods, C791-12, these ASTM standards do not specifically address the sampling methodology required to establish the uniformity requirement for neutron absorber materials.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.

8-2 Provide detailed information on the adequacy of vacuum drying for the 61BTH DSC loaded with 61 damaged BWR assemblies. Alternatively, provide additional detail on the description of damaged BWR assemblies to be stored in the 61BTH.

The application seeks to increase the number of damaged fuel assemblies allowed for storage in the 61BTH DSC by up to 61 assemblies. Depending on the extent and nature of the damage in the fuel, the increase in the amount of damaged fuel assemblies authorized for dry cask storage may have an effect on the effectiveness of vacuum drying operations. The applicant did not provide detailed analysis information on drying adequacy for this increase in damaged spent fuel assemblies, nor did the applicant provide detailed descriptions of the characteristics of fuel damage for the 61 damaged assemblies within the 61BTH DSC, that would allow the staff to conduct confirmatory analyses of vacuum drying effectiveness.

This information is needed to determine compliance with the requirements of 10 CFR 72.234(f) and 72.236(a).

9.0 Operating Procedures Evaluation

9-1 Revise the operating procedures for the Standardized NUHOMS® 61BTH, 32PTH1, and 37PTH DSCs to address storage of additional damaged fuel assemblies, additional failed fuel assemblies, and poison rod assembly requirements, respectively.

Appendixes T, U, and Z have been revised to allow new contents and to require additional criticality controls. The operating procedures in Section 8 of each appendix should be revised to reflect these changes, as necessary.

This information is needed to determine compliance with the criticality safety requirements of 10 CFR 72.124.