April 9, 2014

Mr. Richard W. Boyle, Chief
Sciences Branch
Division of Engineering and Research
Office of Hazardous Materials Safety
U.S. Department of Transportation
1200 New Jersey Ave., S.E.
Washington, D.C. 20590

SUBJECT: SECOND REQUEST FOR ADDITIONAL INFORMATION FOR THE MODEL NO. TN-106 PACKAGE

Dear Mr. Boyle:

By letter dated June 14, 2013, the Department of Transportation submitted a request for review of the French Certificate of Approval No. F/379/B(U)F-96, Revision Ct, for the Model No. TN-106 package, with Content No. 26 only. We acknowledged receipt of your request on July 25, 2013, and issued a first request for additional information (RAI) letter on November 26, 2013.

In connection with the staff's detailed technical review of the RAI responses, dated February 11, 2014, we need the information identified in the enclosure to this letter.

The applicant should notify the Department of Transportation when it can provide the requested information.

Please reference Docket No. 71-3075 and TAC No. L24762 in future correspondence related to this revalidation action. If you have any questions regarding this matter, you may contact me at (301) 287-0759.

Sincerely,

/RA/

Pierre M. Saverot, Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3075
TAC No. L24762

Enclosure: Request for Additional Information

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Request for Additional Information
Docket No. 71-3075
Model No. TN-106 Package

By letter dated June 14, 2013, the Department of Transportation submitted a request for review of the French Certificate of Approval No. F/379/B(U)F-96, Revision Ct, for the Model No. TN-106 package, with Content No. 26 only.

On February 11, 2014, the U.S. Nuclear Regulatory Commission staff received responses to a first request for additional information (RAI) letter dated November 26, 2013.

This second RAI letter identifies information needed by the staff in connection with its review of the Model No. TN-106 package to determine whether the applicant has demonstrated compliance with the International Atomic Energy Agency, "Regulations for the Safe Transport of Radioactive Material," TS-R-1, 1996 Edition (as amended 2003).

2.0 Structural and Materials Evaluation

2-1 Provide a reference for the maximum allowable temperature of 220°C for the EPDM seals. The reference should confirm that the seals can withstand 179°C (maximum seal temperature listed in the application) and 209°C (maximum seal temperature listed in RAI response 4-3) for the period of time the seals are at elevated temperatures during accident conditions.

The SAR claims the maximum temperature exposure of 179°C to the seals is acceptable, due to the "thermal criterion" of 220°C. The response provided to RAI 2-2 states this criterion was reached with a "level of damage formula," located in paragraph 11 of Chapter 2. However, as provided in the application, this formula does not explicitly address the maximum allowable temperature of the seal. In fact, Enclosure 6, a materials compatibility guide provided in the RAI response, indicates an EDPM maximum allowable temperature range of 150°-200°C. Even at these temperatures, the seals will only hold for "short term use." A reference should be provided that clearly states the maximum allowable temperature for EPDM, and confirms that the period of time the seals are at these elevated temperatures, during accident conditions, is acceptable.

This information is required to determine compliance with the requirements of paragraphs 651 and 656 of IAEA TS-R-1.

2-2 Provide additional information on the cladding material:

a) Data demonstrating the stress of the cladding, as a function of temperature, during transportation for both normal conditions and hypothetical accident conditions of transport.

b) Yield stress values of the material as a function of temperature, including references used.

The response to RAI 2-3 references the IAEA-TECDOC-1083 “Status of liquid metal cooled fast reactor technology,” when stating that “the temperature of the cladding of the pins irradiated in the core of the Phenix reactor can reach 650°C.” Enclosure 5, “Irradiation Report of the Futurix-FTA Metal Capsule,” also referenced in RAI response 2-3, was said to “show that the temperature reached in the Phenix reactor by the DOE1 and DOE2 pins is higher than 500°C.” While these enclosures address the cladding material, they do so in reactor conditions, which are significantly different from transportation conditions.

This information is required to determine compliance with the requirements of paragraphs 651(a) and 807(f) of IAEA TS-R-1.

2-3 Analyze the potential for degradation due to sodium residues, if present outside of the fuel rods, and the release of sodium if rod failure were to occur. Discuss the possibility of liquid metal embrittlement.

Page 5/35 of Chapter 0A stated that up to 50 grams of sodium may be present in the cavity. The response to RAI 2-6 appears to indicate that sodium is limited to 2 grams and that it is contained within the pin by the cladding.

a) Confirm that there is no sodium residue external to the cladding.

b) Considering that package temperatures are above the sodium melting point, discuss if sodium, whether in the form of residue outside the rods, or released from inside a rod due to rod failure, comes into contact with the basket/shell material or stainless steel welds. If so, adverse reactions, such as liquid metal embrittlement of steels and welds in contact with sodium, should be addressed.

c) Clarify the adverse effects of sodium in contact with the EPDM seals. The materials chemical compatibility guide in Enclosure 6 did not address pure sodium.

This information is required to determine compliance with the requirements of paragraphs 651 and 656 of IAEA TS-R-1.

2-4 Provide a condition to the specifications to ensure that the required minimum yield strength of the “Stainless Steel Type B” used in the construction of the trunnions will be maintained, or provide a condition that prohibits the use of the trunnions for tiedown.

According to the standard cited in Enclosure 6, the minimum yield strength for “Stainless Steel Type B” at 85°C is 377MPa. This is insufficient to meet the 436MPA yield strength requirement of the trunnions, at 85°C, from the analysis presented in the application. The real yield strength figures, from the cited manufacturing report, are insufficient to ensure the acceptability of future procurements of the material.

The response to RAI 2-1 notes that the applicant does not have any plans to fabricate new TN106 packagings; however, this is insufficient to ensure that new packages can never be fabricated under this revalidation.

Provide a condition to the specifications to either ensure that the minimum yield strength of the trunnions will be maintained or to prohibit the use of the trunnions for tiedown.

This information is required to determine compliance with the requirements of paragraph 606 of IAEA TS-R-1.

3.0 Thermal Evaluation

3-1 Clarify the revolving plug closure plate ACT temperature of 209°C provided in the RAI 4-3 response (Enclosure 1 to TN E-37141).

Chapter 2, Appendix 2, page 15/19, stated that the maximum seal temperature in the package was the Orifice B seal at 179°C. The table provided in the RAI 4-3 response appears to indicate that the maximum seal temperature is the revolving plug closure plate seal at 209°C. The reason for the change in the maximum seal temperature from that found in the application and any updated thermal analysis that shows revised component and content (i.e., cladding) temperatures should be provided.

This information is required to determine compliance with the requirements of paragraph 660 of IAEA TS-R-1.

4.0 Containment Evaluation

4-1 Provide a brief description of the type of leakage tests performed for fabrication, maintenance, periodic, and pre-shipment.

This RAI is a follow-up to a similar RAI dated November 26, 2013. Chapter 6A and 7A provide leakage rate criteria, but there are no descriptions of the leakage tests. A brief description of the tests should include the type of test (evacuated envelope, gas pressure rise, etc.). The pre-shipment test description should explain how the leakage criterion is satisfied considering that the package is not backfilled with helium, per page 12/16 of Chapter 6A.

This information is required to determine compliance with the requirements of paragraph 807 of IAEA TS-R-1.

4-2 State the maximum activity of content #26 that is to be shipped as part of the revalidation review.

This RAI is a follow-up to a similar RAI dated November 26, 2013. The containment release calculations for content #26 in Chapter 3A appear to indicate an activity much less than 1 PBq. However, page 3/4 of document F/379/B(U)F-96-26t states that the package's activity can approach 180 PBq. A clear statement of the total content #26 activity, that is supported by the release calculations in Chapter 3A (i.e., an analyzed condition), should be provided.

This information is required to determine compliance with the requirements of paragraphs 656 and 807 of IAEA TS-R-1.

4-3 Provide updated release calculation and release fractions (including valid references that verify the appropriateness of the release fractions) that reflect the content's non-Light Water Reactor (LWR) fuel.

This RAI is a follow-up to a similar RAI dated November 26, 2013. The RAI 4-1 response (Enclosure 1 to TN E-37141) used release fractions based on LWR fuel. However, the containment calculations and release fractions should reflect the TN-106 content's non-LWR fuel. Note that containment calculations do not have to be provided if the fabrication, maintenance, and periodic leakage tests have a leakage rate criterion less than 1E-7 ref cm3/sec (per ANSI N14.5-1997).

This information is required to determine compliance with the requirements of paragraph 656 of IAEA TS-R-1.

5.0 Shielding Evaluation

5-1 Provide the source term calculation input for the content #26 of the package.

In Ref. DOS-08-00126114-401, Rev. 2, the overall shielding methodology for the calculation of the dose equivalent for gamma and neutron sources, using Mercury 6 and SND1, was provided. However, the methodology used to determine the actual source terms and the resulting neutron and gamma source terms in these calculations was not provided.

It is only stated that the source term is calculated with Origen 2.1. Provide the inputs for neutron and gamma source terms that were used in the dose equivalent calculations corresponding to content #26.

This information is required to determine compliance with the requirements of paragraphs 530, 532, and 656(b)(ii)(I) of IAEA TS-R-1.

6.0 Criticality Evaluation

6-1 Provide confirmation of the assertion that filling the cavity and all other void spaces with steel is the most reactive configuration.

In Section 6.1 of Ref. DOS-08-00126114-511, the reactivity effects of filling the cavity and other void space with steel were analyzed and compared with the results obtained when water is considered. The criticality analyses, provided in Calculation 21220-0600 for the single cask and infinite array calculation, included the assumption that the cask cavity and remaining void spaces were filled with steel.

Although the analysis performed in Ref. DOS-08-00126114-511 determined that using steel would result in higher reactivity for the system, staff identified that both numbers obtained for water and steel were higher than those yielded for the steel case when using SCALE. Staff also identified no comparative cases using SCALE with water being considered.

Confirm that assuming steel instead of water is more reactive when calculated using SCALE.

This information is required to determine compliance with paragraphs 671 through 682 of IAEA TS-R-1.