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

July 19, 2017

Mr. Richard W. Boyle
Radioactive Materials / Research and Development
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: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. F-522 PACKAGE

Dear Mr. Boyle:

By letter dated March 15, 2017, the U.S. Department of Transportation (DOT) requested the NRC staff to perform a review of the Canadian Certificate of Approval No. CDN/2094/B(U)-96, Revision 1, for the Model No. F-522 package, and make a recommendation concerning the revalidation of the package for import and export use.

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 August 25, 2017. Inform us at your earliest convenience, but no later than August 11, 2017, if a substantial date change is needed. To assist us in re-scheduling the review, you should include a new proposed submittal date.

If you have any questions regarding this matter, please contact me at 301-415-5790.

Sincerely,

/RA/

John Vera, Project Manager
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3091
CAC No. L25212

Enclosure: Request for Additional Information

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. F-522 PACKAGE – DOCUMENT DATE: JULY 19, 2017

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OFC	DSFM		DSFM		DSFM		DSFM		DSFM	
NAME	JVera		SFigueroa*		JChang*		VWilson*		TAhn*	
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DATE	7/6/2017		7/10/2017		7/6/2017		7/19/2017			

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Request for Additional Information
Docket No. 71-3091
Model No. F-522 Package

This request for additional information (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 the International Atomic Energy Agency (IAEA) Specific Safety Requirements No. SSR-6, 2012 edition (SSR-6).

MATERIALS

Mt-1. Revise package drawings to include: (i) tolerances of dimensions and (ii) material specifications of receptacles.

The applicant provided a few drawings of package components. The staff requests (i) tolerances of dimensions and (ii) material specifications of receptacles, for the staff's uncertainty reviews of component functional requirements.

This information is needed to determine compliance with the requirements of paragraph 501 of IAEA SSR-6.

Mt-2. Provide information on the potential susceptibility to caustic stress corrosion cracking (SCC) of leak proof insert made of stainless steel with alkaline solutions based on NaOH or NH_4OH.

The applicant did not assess the potential susceptibility of austenitic stainless steels (e.g., 300 series) to caustic SCC at above ~100 ˚C in alkaline solution (e.g., NaOH) (Materials Technology Institute). With the SCC, the leak proof insert may not remain intact for the containment. The alkaline liquid radioactive content is contained within the receptacle inside the leak proof insert. The leak proof insert is fabricated with stainless steel. If damage to the receptacle occurred, the leak proof insert would maintain containment. The leak proof insert of stainless will be in contact with alkaline solution. The susceptible temperature of stainless steel SCC varies with alkalinity strength (e.g., NaOH weight percent from above 0 to 80, see *Materials Technology Institute, Technical Awareness Bulletin, Caustic Stress Corrosion Cracking, No. 13*). The application did not include potential susceptibility to SCC of stainless steel in General Provision, 618 and 620 of IAEA SR-6. The staff requests information on the potential susceptibility of stainless steel to SCC for containment safety of liquid radioactive contents.

This information is needed to determine compliance with the requirements of paragraphs 501, 614, 618 and 620 of IAEA SSR-6.

Mt-3. Clarify very high temperature rise of remaining nitrogen after pyrophoric reaction of Rubidium does not have a significant effect in terms of package radiological safety such as content form (e.g., vaporization).

Radioactive Contents contain pyrophoric target materials of Rubidium (Rb-82, 83, 84). The applicant assessed very high temperature rise of remaining nitrogen during the pyrophoric reaction. The applicant assessed the amount of hydrogen generation for the pyrophoricity with three potential chemical reactions of Rubidium with water and oxygen coming from assumed air intrusion with moisture (the hazard analysis in Safety Analysis Report (SAR), Appendix 13). The extremely conservative assessment made by the applicant is not meaningful, unless there is an assessment that the shows the package continues to perform its safety function during and after this transient.

This information is needed to determine compliance with the requirements of paragraphs 305 and 618 of IAEA SSR-6.

Mt- 4. Clarify that the assessment of chemical reactions made by the applicant for Rubidium is equally used for the pyrophoricity assessment of Sr-82.

The applicant assessed the pyrophoricty of the content of Sr-82 targets (SAR, Section 2.5). The applicant stated that the chemical reactions would result in a temperature increase of 2 °C to the leakproof insert, which would not cause damage. The staff requests how the applicant determined this temperature increase. If it came from the assessment of Rubidium, the applicant needs to explain utilizing proper chemical reaction data.

This information is needed to determine compliance with the requirements of paragraphs 305 and 618 of IAEA SSR-6.

Mt-5. Provide assessments on post-fire thermal stability for mechanical strength of Foam. The applicant assessed the effect on the density of the Foam and the depth of char during regulatory fire conditions based on tests done by General Plastics (SAR, Appendix 14). The applicant concluded that the density change is minimal and would not affect drop tests. Foam (polymer) is thermally stable in terms of strength (such as elastic modulus) at less than approximately 120 °C (*Sanchez, E.M.S., C.A.C. Zavaglia, and M.I. Felisberti. "Unsaturated Polyester Resins: Influence of the Styrene Concentration on the Miscibility and Mechanical Properties." Polymer 41 (2000): 765-769*). The applicant provided appropriate data on mechanical properties within 121 °C (250 ˚F) (General Plastics). However, the applicant did not address post-fire mechanical strength of Foam.

This information is needed to determine compliance with the requirements of paragraph 614 of IAEA SSR-6.

THERMAL

Th-1. Provide more information of the thermal model used in the post-fire cooldown analysis.

The applicant performed hypothetical accident conditions (HAC) thermal analysis with the initial component temperatures imported from the normal conditions of transport analysis (with solar insolation), the emissivity of 0.8 for all package outer surfaces, and the flame temperature of 800°C applied directly to the package outer walls during the 30-minute fire as described in the IS/TR-2650-F522. The applicant simplified the model to an internal flow analysis.

For the post-fire cooldown, the applicant performed the transient analysis using temperature data from the end of 30-minute fire, 38°C ambient and solar insolation. To clarify the model for the post-fire cooldown, the applicant should provide the package surface emissivities (0.8?), the convection flow type (natural/forced convection?) and the convection heat transfer coefficients used in the Flow Simulation code for the post-fire cooldown analysis.

This information is needed to determine compliance with the requirements of paragraph 728(a) of IAEA SSR-6.

Th-2. Provide transient temperatures of the leak proof insert (LPI) cavity for F-522 loaded with (a) liquid Mo-99 and (b) solid Mo-99 during the HAC fire, including both 30-minute fire and its post-fire cooldown.

The applicant presented the after-fire O-ring temperatures in IS/TR 2650 F522, Figure 29 for the load case of liquid Mo-99 (4 watts) and Figure 31 for the load case of solid Mo-99 (40 watts) for the HAC transient temperature variations.

Beside the O-ring transient temperatures, the applicant should also provide the LPI-cavity temperature history each for (a) solid Mo-99, 40 watts and (b) liquid Mo-99, 4 watts during the HAC fire, including both 30-minute fire transient and its post-fire cooldown for justification of potential auto-ignition in the LPI-cavity.

This information is needed to determine compliance with the requirements of paragraph 728(a) of IAEA SSR-6.

CONTAINMENT

Co-1. Clarify whether the vacuum liquid bubble test is suitable as a leak test method for the F-248 LPI.

The applicant stated in Section 4.4.2 of the IS/DS-2651-F522 that each F-248 LPI shall be leak tested using a method sensitive to 10^{-8} std cc/s according to a procedure prepared in accordance with ISO 12807. The leak rate shall be less than 10^{-7} std cc/s. The applicant also stated in Section 5.3.3 that a Vacuum Liquid Bubble Test shall be performed on the LPI. There shall be no visible sign of leakage.

Per reference to Table A-1 of ANSI N14.5 (2014), the staff finds that the vacuum bubble test method (the method involves immersing the test item in a liquid) has nominal test sensitivity of 10^{-3} std cc/s and therefore may not be suitable for F-248 LPI to which the sensitivity of leak test is required to be 10^{-8} std cc/s. The applicant should clarify use of the vacuum liquid bubble test is appropriate for a leak test to F-248 LPI.

This information is needed to determine compliance with the requirements of paragraphs 510 and 511 of IAEA SSR-6.

SHIELDING

SH-1. Provide additional information and justification on the modeling of the F-522 using MICROSHIELD.

The applicant calculated the external radiation level of the package using the MICROSHIELD code. There are approximations to geometry that need to be handled in a conservative way as MICROSHIELD is not capable of modeling complex geometries. Another approximation comes in the form of the buildup factor, which is used to account for generation of secondary gammas within the shield medium and depending on how this factor is derived and which material it is applied can change the results of the radiation level calculation significantly. Calculations can also vary widely based on the flux-to-dose-rate conversion factors used.

The applicant needs to provide additional information on the geometry approximations used to model the F-522 with the MICROSHIELD code and justify that it is conservative. The applicant needs to state the materials that were used within the F-522 MICROSHIELD model, including density, and justify that the materials were specified in a conservative way. The applicant needs to provide information on the buildup factor, including how it was derived and which material it was applied to, and justify that it was applied in a conservative manner. The applicant needs to state which version of MICROSHIELD and which flux-to-dose-rate conversion factors it applied.

This information is needed to determine compliance with the requirements of SSR-6 relating to allowable maximum radiation level in paragraphs 526, 527, 648(b), 659(b)(i).

Sh-2. Provide additional information on the activity for the Mo-99 content and the modeling of this source in MICROSHIELD.

Section 4.1 of the SAR (page A4-10 of the SAR) states that the applicant used MICROSHIELD to model 370 TBq of Mo-99, and that the resulting surface radiation level was 0.15 mSv/hr with a temperature increase (TI) of 0.5. The applicant then states that the external radiation level from the I-132 is 1.2 mSv/h with a TI of 4 and that the total external radiation level (Mo-99 plus I-132) is 1.9 mSv/h with a TI of 6. Given the external radiation level of the separate Mo-99 and I-132 components, the total is higher than the sum of the two sources. The applicant needs to discuss this discrepancy or provide any additional source contributions to the total radiation level.

This information is needed to determine compliance with the requirements of SSR-6 relating to allowable maximum radiation level in paragraphs 526, 527, 648(b), 659(b)(i).

Sh-3. Provide additional information on the impurities allowed for the Mo-99 content and how “equivalency” is determined.

The certificate for the F-522 states that for the “Mo-99/Tc-99m and associated impurities” content that an “impurity level equivalent to 1850 GBq of I-132” is allowed. Section 4.1 of the SAR (page A4-10) states that “The extraction process removes most of the significant isotopes. Of the remaining isotopes, I-132 has the most significant impact on shielding.” The specification of this content within the certificate is ambiguous. The applicant needs to discuss which other isotopes are allowed as part of the “associated impurities” and how the equivalency to I-132 is determined. The applicant also needs to discuss if daughter products of the allowable nuclides has been considered as these can also significantly contribute to external radiation levels.

This information is needed to determine compliance with the requirements of SSR-6 relating to allowable maximum radiation level in paragraphs 432, 526, 527, 648(b), 659(b)(i).

Sh-4. Provide additional information on the activation of rubidium and justify that the source term was modeled conservatively.

For the activated rubidium content, the F-522 certificate states: "Proton irradiated rubidium based target material and associated target shells." The applicant gives isotopic content for a "typical" irradiation history but also states that the irradiation can be unpredictable. The staff does not have enough information about this content to determine that the applicant has modeled a bounding source. The applicant needs to provide the distribution of nuclides among the 6 TBq. Currently the certificate states: "Sr-82, Sr-83, Sr-85, Rb-82, Rb-83, Rb- 84, Co-55, V-48, Mn-52 and other radionuclides," in an amount of 6 TBq total. The use of the phrase "other nuclides" and quantity "6 TBq total" are ambiguous. The applicant also needs to discuss if daughter products of the allowable nuclides have been considered as these can also significantly contribute to external radiation levels.

This information is needed to determine compliance with the requirements of SSR-6 relating to allowable maximum radiation level in paragraphs 432, 526, 527, 648(b), 659(b)(i).