SAFETY EVALUATION REPORT
U.S. Department of Transportation
French Approval Certificate No. F/313/B(U)F (Mbs)
Docket No. 71-3034
Certificate of Compliance No. 3034
Model No. TN-BGC-1

SUMMARY

By letter dated July 23, 2024 (Agencywide Documents Access and Management System Accession No. ML24283A024), as supplemented on January 21, 2025 (ML25119A286), TN Americas LLC, on behalf of Orano NPS, requested a U.S. Department of Transportation (DOT) revalidation of the French Approval Certificate F/313/B(U)F (Mbs) for the Model No. TN-BGC-1.

In this safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC) has reviewed the safety analysis report (SAR) and accompanying documentation associated with the TN-BGC-1 transportation package in accordance with the requirements in International Atomic Energy Agency SSR-6 (IAEA SSR-6), "Regulations for the Safe Transport of Radioactive Material," 2018 Edition.

Based on the statements and representations in the application, as supplemented, staff recommends revalidation of the French Package Design Certificate No. F/313/B(U)F-96, for use in the United States, with the conditions stated below. The staff notes, however, that the application does not provide any calculation to demonstrate that the package meets the tie-down system design requirement using acceleration factors in the Specific Safety Guide (SSG)-26, table IV.2.

- Regarding component fatigue evaluations for tie-down during road transport, a weld inspection (as identified in the Major Maintenance penetrant exam scope in chapter 8 of the TN-BGC-1 SAR) should be performed prior to the first use of each package in the U.S., unless the package is new or has undergone this type of inspection within 5 transports or one year of its first use in the U.S. If a package meeting the description above is then used exclusively in the U.S., resumption of the normal maintenance program is acceptable (see section 2.5 of this SER).
- If pyrophoric contents, including metal powders, are transported, inerting of the primary packaging boxes should be performed, internal fitting, and the TN-BGC-1 cavity (see SAR chapter 3 and SAR chapter 16, section 5).
- For Content 11, the presence of transuranic elements up to a maximum of 1 gram (g) is permitted.
- The mixing of different sub-contents within the same package is prohibited (see SAR chapter 3 section 2.5.1).
- The transport period between package closure and opening at the destination facility will not exceed 1 year (see section 3 of French certificate).
- The presence of polyethylene is not authorized for air transport, regardless of the sub-content (Content 11, see chapter 3). In addition, for overland transport, the presence of polyethylene:
  - is not authorized in sub-contents No. 11b and 11d and
  - is authorized up to 1000 g in sub-contents No. 11a and 11c.
- The condition in chapter 3 section 2.5.1 (Content 11) and section 2.10.1 (Content 26) stated that the time between closure of the internal fittings at the shipping facility and the

Enclosure

opening of the package at the consignee facility must be less than 1 year when polymer materials are present during transport.

- A package shipped in a CB9-type overpack is transported under exclusive use.
- For Content 11, if polymeric materials are present and if there is no Poral-type device (described in SAR chapter 3) on the lids of the internal fitting, a minimum free volume of 2 liters in the cavity of the internal fitting is required and the storage time of the closed internal fitting does not exceed 20 years.
- Based on a review of the package handling and maintenance criteria described in chapters 7 and 8 of the SAR, the staff recommends the applicant provide DOT with the following:
    - Specific provisions for inspection of stainless steel components to detect and evaluate indications of localized corrosion, stress corrosion cracking (SCC), and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.
    - Specific provisions for inspection of aluminum components to detect and evaluate indications of localized corrosion and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.
    - A bounding analysis to show that loss of boron atoms over the 40-year period of use will not result in a loss in effectiveness of the borated resin, or specific provisions to manage loss of neutron attenuation and shielding, such as through radiation monitoring and subcriticality testing, to ensure the borated resin's continued effectiveness.

1.0 GENERAL INFORMATION

The applicant requested approval to ship unirradiated uranium-bearing content (designated as Content 11) and unirradiated uranium TRIGA fuel content (designated as Content 26), described in detail in appendix 11 and appendix 26, respectively, of the French certificate. The French certificate stated that Content 11 can contain up to a maximum 1 (g) gram of transuranic elements. The French certificate also stated that TRIGA fuel does not come from reprocessing and no transuranic content was listed. Decay heat was small because Content No. 11 and Content No. 26 were based on unirradiated uranium.

1.1. Content No. 11

The applicant proposed modifications to the previously approved Content No. 11 – solid uranium-bearing materials. Section 2.5.1 describes the chemical forms of the authorized contents and added a new uranium fluoride form: UO_2F_2. Section 2.5 also defines the isotopic composition and masses by defining sub-contents. The applicant proposed leaving sub-contents A and C unchanged, modifying B and D, and removing E, F, G and H. For sub-contents B, contents with any uranium enrichment, the maximum mass of U-235 was changed to 8.5 kilogram (kg). For sub-contents D, contents with enrichment less than or equal to 20 percent, the maximum mass of U-235 was changed to 40 kg.

1.2. Content No. 26

The applicant proposed modifications to the previously approved Contents No. 26 – TRIGA Fuel. In SAR section 2.10.1, the applicant proposed recategorizing the characteristic groups of fuel elements from groups of standard and thin fuel elements to sub-contents families (F1, F2, F3, F4, Marvel, and Miscellaneous). The applicant provided new tables that define the chemical composition and maximum transportable quantities of uranium for each sub-contents family.

Based on a review of the statements and representations in the application, the staff concludes the contents have been adequately described and evaluated to meet the requirements in IAEA SSR-6, 2018 Edition.

2.0 STRUCTURAL EVALUATION

The objective of the NRC structural evaluation is to verify that the applicant has adequately analyzed the structural performance of the transportation package per French approval certificate F/313/B(U)F (Mbs) so that it meets the requirements in IAEA SSR-6.

2.1. Description of Package and Changes

In the U.S., the TN-BGC-1 Type B(U)F transportation package is employed for the transport of TRIGA fuel assemblies and certain solids uranium-bearing materials (i.e., fuel type Nos. 11 and 26). As described and illustrated in SAR chapter 4, CEA-DES-DDSD-DTEL-SGPE-DSEM-07600, Revision C, the package is comprised of four major structural components: a confinement body and closure system, top impact limiter, and transport/lifting cage. The body consists of a cylindrical shape packaging that has overall nominal dimensions of 1.8 meters (m) long, 0.3 m in diameter at the bottom and 0.5 m in diameter at the cover. The cage is an aluminum member frame that has overall nominal dimensions of 1.82 m in length and 0.6 m square in depth and width. The empty and maximum loaded masses of the packaging are 280 kg and 396 kg, respectively.

The applicant lists all the principal components of the package and associated materials in SAR chapter 4, table 1, “List of Main Packaging Components.” The applicant provides a tabulation of IAEA SSR-6 regulatory requirements and where they are addressed in SAR chapter 6 and additional package illustrations in SAR chapter 10. The structural analyses performed for the package are summarized by the applicant in SAR chapter 11.

The proposed Revision 22 of Competent Authority Certification USA/0492/B(U)F-96 contains the following changes: 1) renumbers, reorganizes and reformats the entire SAR, 2) eliminates fuel types 11e to 11g and creates new groupings of fuel type 26, 3) updates the material for several screws attaching the body to the cage, and 4) addresses selected component aging effects on the package due to the change in edition of the IAEA SSR-6 referenced, from 2012 to 2018.

The NRC staff notes that, despite the changes made to the contents and package noted above, the Revision 22 of Certificate of Approval maintains the same package design, dimensions, lifting devices, maximum gross weight and maximum content weight limits as specified for the transportation package in Revision 21.

The NRC staff reviewed the updated description for the general design of the package for completeness and accuracy and finds that the applicant adequately incorporated information related to the geometry, dimensions, materials, components, and relevant details to describe major structural components of the transportation package. The applicant provided supplemental information related to the package in responses to staff questions, contained in ML25119A286. Therefore, the NRC staff finds that the information provided for the transportation package includes sufficient detail to demonstrate compliance with the design requirements of the SSR-6 Regulation, and as required in paragraphs 809 and 810 of the Regulation for the proposed TRIGA fuel contents.

2.2. Screw material change for body attachment to cage

Per SAR chapter 11, section 3.4.1.2, due to a fastener failure during the 1.2 m drop test associated with normal conditions of transport, the material for the four middle screws attaching the package body to the cage are revised to a higher-strength material, as well having their threads removed from the shear plane. Based on the material properties of A4-80 or A4-100 stainless steel at the ambient temperature of 20 degrees Celsius (°C), the evaluation concludes that, as these new screws have a larger shear area and higher energy absorption capacity, they are adequate to survive the 1.2 m drop, even if whiplash effects are considered. The applicant's evaluation approach has been reviewed by staff and is determined to be acceptable.

2.3. SSR-6 Edition change

The revalidation request is for the TN-BGC-1, a Type B(U)F transportation package, in accordance with the requirements of the 2018 version of IAEA SSR-6, whereas the previous revalidation employed the 2012 version. The chief differences between these two versions in that the 2018 version invokes the requirements to manage the effects of aging of the package per paragraph 613A of the SSR. To address this new requirement for the expected service life of 40 to 50 years for the TN-BGC-1, the applicant provided quantitative and qualitative evaluations for the reusable package components, as noted in section 4 of SAR chapter 5 and the responses provided in ML25119A286.

Per the guidance of paragraph 613A.1 of IAEA SSG-26, "Advisory Material for the International Atomic Energy Agency (IAEA) Regulations for the Safe Transport of Radioactive Material," Revision 1, the applicant has evaluated and determined that fatigue is neither an aging concern for the screws and brackets that attach the package to the tie-down/lifting cage, nor the tie-down/lifting cage members and welded connections, during normal conditions of road transport or lifting and handling conditions. The aging effects on the remaining components of the package are qualitatively determined by the applicant to either not be applicable or to be negligible, as documented either in the SAR and responses in ML25119A286.

In the package structural evaluation in chapter 11, attachment 7 of the SAR, the applicant has determined tie-down design loads for normal conditions of transport via road per the accelerations provided in table IV.1 of IAEA SSG-26, as referred in paragraph 638.3. The applicant takes the bounding cage member stress results, which had the highest stress ratio between member, bracket and screw components of the cage, and determines the allowable number of cycles corresponding to this tie-down configuration to be 141,962. In the fatigue evaluation, the applicant employed a random vibration distribution to the component being studied. The applicant did not specifically quantify the expected number of cycles the component would experience over the 40- to 50-year service life of the package, or for the number of cycles for the period before the component would be replaced, however, the applicant determined 141,962 cycles to be a sufficient upper limit for the cage members to satisfy the fatigue criteria.

The applicant determined that no aging analysis was required for the following components, or that aging concerns were negligible, as stated in either section 4 of SAR chapter 5 and the responses provided in ML25119A286:

- Gaskets are replaced during minor maintenance operations, which occurs every 15 uses or 3 years, whichever comes first.

- Neutron-absorbing resins, aluminum bronze, bronze, and aluminum are not sensitive to thermal aging effects at the temperatures reached during normal conditions of transport (NCT).
- Wood inside the covers is not subject to any thermal or mechanical aging effects.
- The number of cycles and magnitude of pressurization during NCT conditions for the transport of fuel type Nos. 11 and 26, (i.e., 1.2 bar absolute for Family A1) has a negligible aging effect on the package.
- The aging effects are negligible on the package closure system of the bayonet and clamping rings, as they are visually and dimensionally inspected during minor maintenance operations, (i.e., 15 uses or 3 years, whichever comes first) and are replaced if they fail inspection.
- The aging effects are negligible for the lifting and handling of the package and its lifting/handling cage, as the conservatively-determined maximum stress on components is not significant, and a subsequent fatigue analysis that determined 39,718 cycles to be the upper limit for the critical component was deemed sufficient by the applicant.

The applicant also emphasized that their experience during the 30-year operation and maintenance of this package has not revealed any external indications of aging effects.

The NRC staff reviewed the allowable number of cycles determined in the fatigue analysis for lifting and handling and determined that, even though the applicant did not quantify the expected number of cycles over the anticipated 40- to 50-year service life of the package, due to the high magnitude, it is unlikely that the TN-BGC-1 will approach these limits over the package service life. However, for tie-down during road transport, since the applicant did not quantify the expected number of cycles over the service life of the package, or the number of cycles for the period before that component would be inspected or replaced, it is not possible for staff to conclusively determine that their aging management evaluations are fully in accordance with the requirements of the 2018 version of IAEA SSR-6. Furthermore, the road transport fatigue analyses were performed using tie-down accelerations that do not comply with those prescribed in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 71.45(b)(1), as explained in the following paragraph, which, if considered, would reduce the allowable number of cycles.

2.4. Tie-down considerations

In the package structural evaluation in chapter 11, attachments 7 and 8 of the SAR, the applicant has determined tie-down design loads for normal conditions of transport via road and air, respectively, per table IV.1 of IAEA SSG-26, as referred in paragraph 638.3. However, according to the IAEA SSG-26, paragraph IV.17, the package designer is responsible for ensuring that the package tie-down attachment points are designed in compliance with values acceptable to the relevant competent authorities and defined in modal requirements. This refers the user to acceleration factors published in table IV.2 of the IAEA SSG-26 for U.S. transportation, which is consistent with the U.S. regulatory requirement 10 CFR 71.45(b)(1). The applicant’s response provided in ML25119A286, confirmed that the SAR does not provide any calculation to demonstrate that the package meets the U.S. tie-down system design requirement using acceleration factors in the SSG-26, table IV.2.

2.5. Evaluation Findings

Based on the review of the relevant structural component aging management assessments of the package for the proposed contents of fuel type Nos. 11 and 26, staff finds the conclusions of the evaluations to be in accordance with the requirements of the 2018 version of IAEA SSR-6, except for the component fatigue evaluations for tie-down during road transport. Since the applicant has not determined the anticipated number of transport-induced stress cycles expected on the most vulnerable component over the life of the package, or between inspection or replacement of the component, the NRC staff recommends that a condition be added requiring that a weld inspection be performed prior to the first use of each package in the U.S. The NRC staff finds that a liquid penetrant examination (LPE) of the cage member welds for fatigue cracking, per CODAP[1] 2005, Annex I1.A2, per qualified personnel, as cited in the section 3.2.4 of SAR chapter 8, "Maintenance Instructions," to be an acceptable measure to reasonably ensure the package safety from fatigue effects during transport. If the package is new or has undergone an LPE within 5 transports or 1 year prior to its first use in the U.S., then no LPE is required. If either of the packages described above is then exclusively employed in the U.S., then the resumption of the normal maintenance program is acceptable.

Based on the staff's review of the structural evaluation and related sections of the application, the TN-BGC-1 package has adequate structural capacity to meet the requirements of IAEA SSR-6, with exception of the tie-down system. The tie-down system design does not fulfill the requirement using acceleration factors for the U.S. transport per table IV.2 of the IAEA SSG-26.

3.0 THERMAL EVALUATION

3.1. Review Scope and Objective

TN-BGC-1 is a Type B(U)(F) package designed to transport various forms of fissile material. Although there have been previous revalidation reviews of the TN-BGC-1 package, the focus of this revalidation dealt with the slight changes associated with the transport of unirradiated uranium-bearing content (designated as Content 11) and unirradiated uranium TRIGA fuel content (designated as Content 26). The French certificate stated that the TRIGA fuel does not come from reprocessing. Staff notes that chapter 6 of the SAR provided a table containing IAEA regulations found in "IAEA Regulations for the safe transport of radioactive materials," SSR-6 (2018 edition) and the corresponding SAR chapters that demonstrated those regulations were met.

There were no major changes to the packaging thermal design and content (e.g., decay heat of Content 11 and Content 26) as part of this revalidation submittal, as indicated in document NTE-24-000591-000-1.0 "TN-BGC-1 SAR Summary Changes" (Enclosure 5 to E-63632). Decay heat was small because Content 11 and Content 26 were based on unirradiated uranium. For example, SAR section 4.3.2 of chapter 16 noted that the bounding uranium-based content was Content 11b with a decay heat of 20 mW.

3.2. Package Description and General Considerations

As described in SAR section 2.5.1 of chapter 3, the packaging can consist of content containers placed within various types of internal fittings and the TN-BGC-1 cavity. The TN-BGC-1 packaging, which includes an aluminum cage structure, package body consisting of inner and outer shells and bottom, closure system, and shock-absorbing cover (according to SAR section 4 of chapter 4), can be transported individually or as multiple packages within a type CB9 physical protection overpack (similar to an International Organization for Standardization [ISO] 20 feet [ft] stainless steel container with an insulator between its two walls, per document

[1] Code Français de Construction des Appareils à Pression; French Construction Code for Pressurized Vessels

RE21078000010, Revision 2 “TN-BGC-1 packaging – Thermal analysis”). SAR table 2 of chapter 3 and subsequent figures within the chapter provided the cavity volumes and diagrams associated with internal fittings and spacers. SAR section 4 and section 5.2 of chapter 4 noted that the resin between the inner and outer shells provides neutron and thermal shielding. Section 9.4 of chapter 4 indicated that the resin has a steady-state allowable temperature range between -40°C and approximately 160°C. In addition, plastic fuse plugs allow the removal of burnt gases during an accident condition of transport (ACT) fire to prevent overpressure. SAR section 9.7.1 noted that the plastic fuse plugs are made of polyethylene or any other material with a melting point of less than 150°C. Although SAR section 9.6 mentioned that the gaskets have a steady-state allowable temperature range of -20°C and 200°C, SAR section 3.5.2 of chapter 13 discussed that the gaskets in a Type B(U) package have a -40°C cold allowable temperature.

3.3. Description of Thermal Analysis

As noted earlier, the revalidation review was focused on the unirradiated uranium-bearing and unirradiated TRIGA fuel contents that have small decay heat (i.e., a few mW). SAR chapter 12 summarized the thermal model geometry and boundary conditions for both NCT and the fire ACT for all TN-BGC-1 package contents that were described in detail in document RE21078000010, “TN-BGC 1 packaging – Thermal analysis.” SAR chapter 9 indicated that computer codes used in the analyses were qualified, per quality assurance management.

SAR chapter 12 noted that the packaging was modeled (and is transported) vertically, although it can be transported horizontally when loaded with uranium-bearing contents of small decay heat (i.e., a few mW). Chapter 12, section 6, listed the different configurations of the package that were modeled, including decay heat, types of internal fittings and spacers, and whether an overpack was present or not. SAR table 1 of chapter 3 as well as table 3 and table 6 of chapter 4 provided mechanical and thermal properties (e.g., thermal conductivity, specific heat, wall emissivity) of internal fittings and spacer materials and the physical protection overpack, including stainless steel and aluminum. In addition, section 3.4 of document RE21078000010, “TN-BGC 1 packaging – Thermal analysis,” provided the thermal properties of other package materials.

The thermal analyses were based on a three-dimensional finite element model, which consisted either of a single TN-BGC-1 package or configuration of TN-BGC-1 packages within the overpack; symmetry was applied to the middle of the overpack model. SAR chapter 12, section 4.1, indicated that the unirradiated uranium decay heat was not modeled due to its small value (e.g., 20 mW). Section 3.6 of document RE21078000010 mentioned that axial and radial air gaps were modeled, and that conduction and radiative heat transfer were considered across the gaps. Radiative heat transfer within a package cavity considered the view factors between elements.

According to document RE21078000010, the NCT calculation at 38°C ambient temperature assumed no damage to packaging or overpack and both geometries were modeled in the vertical position. In addition, SAR section 8 of chapter 12 also considered the effect of a 55°C ambient temperature for air transport. Solar flux was applied as a transient condition (12 hour [hr.] on/12 hr. off) when modeling NCT to achieve a cyclical convergence. The insolation was applied to the package surfaces when not modeling an overpack; similarly, it was applied to the overpack surfaces when modeling the transport of multiple packages. Regulatory flux values (e.g., 200 W/m^2 for vertical surfaces) were applied to the package or overpack surfaces exposed to the ambient environment. Package and overpack solar absorptivity values were 0.4 and 0.33, respectively.

According to document RE21078000010, the fire ACT test was modeled by an 800°C fire for 30 minutes followed by the cooldown period modeled by a 12 hr. on/12 hr. off solar flux transient. Radiative heat transfer input was considered by modeling the fire with an 800°C temperature and an emissivity value of 0.9; the outer surface of the package or overpack had an emissivity of 0.8. A convection heat transfer coefficient of 10 W/m^2K was applied to surfaces in contact with the fire to model the convective component of the fire. The fire ACT thermal model assumed a package in the horizontal position and in a damaged condition, including reduced thickness of the wood in the shock-absorbing cover and the bottom of the packaging body, reduced radial thickness of the resin along the package's body, no axial air gaps, and modified wood properties; the resin was considered charred after reaching 300°C.

3.4. Thermal Results at NCT and ACT

SAR section 6 of the chapter 12 thermal analysis summary indicated that package temperatures (e.g., gaskets), when transporting content with small decay heat (e.g., 20 mW) at NCT, were within allowable values and had surface temperatures less than 50°C (i.e., meeting non-exclusive use transport requirements). SAR section 7.2 noted that 24 packages containing uranium-bearing content could be transported in the overpack. Similarly, section 7.2 indicated that package temperatures were within allowable temperatures for the fire ACT (e.g., approximately 123°C calculated gasket temperature compared to 200°C allowable temperature), except that resin filled between the package body's dual shell can reach temperatures that result in a degraded 10 millimeters (mm) thick layer of char. However, section 7.2 noted that the remaining resin retained its neutron-absorbing properties. In addition, the effects of thermal gradients were discussed in SAR chapter 11, section 3.2, and showed that stresses were below the yield stress of the package's inner and outer shell.

The calculation for determining pressure within the package for uranium-bearing content was presented in SAR chapter 13, section 4.4.3.1. The calculation was based on an initial package pressure of 1.04 bar (i.e., after closing of lids) and considered the increase in pressure due to higher temperatures that occur during NCT and the fire ACT as well as the increased pressure due to increased moles of gas from radiolysis during NCT and the increased moles of gas from radiolysis and thermolysis during the fire ACT; further discussion about radiolysis and thermolysis is presented in the staff's Containment Evaluation in chapter 4 of this SER. In addition, the fire ACT pressure calculation, which showed internal package temperatures greater than 100°C, included the water vapor pressure component from absorbed water within the content and package. Results of the calculations showed a NCT pressure of 1.195 bar (absolute) and an ACT pressure of 6.05 bar (absolute). These pressures are less than the 10 bar (gauge) values that formed the bounding assumption for the package's internal pressure structural calculations performed in SAR chapter 11, section 3.1, and less than the internal fitting maximum allowable pressures reported in section 4.1. These pressures are also less than the maximum allowable pressure of 7.8 bar (absolute) and 6.8 bar (gauge) to accommodate a hypothetical ignition event and its resulting dynamic pressure spike, as reported in SAR chapter 11, section 3.1.8, and chapter 16, section 3. Staff notes that chapter 3 and chapter 16, section 5, indicated that transport of pyrophoric contents, including metal powders, would require inerting of the content's primary packaging boxes, internal fittings, and the packaging cavity (e.g., 20 kPa absolute inert package cavity pressure according to Containment request for additional information [RAI] 1 response). Procedure information for inerting the packaging cavity was provided in SAR chapter 7; the response to Containment RAI 1 (ML25119A286) noted that inerting procedures of primary packaging boxes and internal fittings are site-dependent.

3.5. Evaluation Findings

This licensing action is a revalidation recommendation to DOT of the French approval certificate (Number F/313/B(U)F found in Enclosure 1 to E-63632) for the TN-BGC-1 package that was approved according to "IAEA Regulations for the safe transport of radioactive materials," SSR-6 (2018 edition). Based on a review of the relevant portions of the French certificate and the representations in the application, the NRC staff has reasonable assurance that the TN-BGC-1 package transporting content designated as unirradiated uranium-bearing Content 11 and unirradiated TRIGA fuel Content 26 with small decay heat (e.g., 20 mW) meets the thermal requirements found in IAEA SSR-6 based on the following conditions:

- Appendix 11 mentions the need for inerting of the primary packaging boxes, internal fitting, and the TN-BGC-1 cavity if pyrophoric contents, including metal powders, are transported (see SAR chapter 3 and SAR chapter 16, section 5).
- For Content 11, the presence of transuranic elements up to a maximum of 1 (g) is permitted.
- Chapter 3, section 2.5.1 said that mixing of different sub-contents within the same package is prohibited.
- The condition in chapter 3, section 2.5.1 (Content 11) and section 2.10.1 (Content 26) stated that the time between closure of the internal fittings at the shipping facility and the opening of the package at the consignee facility must be less than 1 year when polymer materials are present during transport.
- Chapter 3 noted the presence of polyethylene is not authorized for air transport (Content 11). In addition, the SAR mentioned "For overland transport, the presence of polyethylene:
    - is not authorized in sub-contents No. 11b and 11d,
    - is authorized up to 1000 g in sub-contents No. 11a and 11c.
- For transport by air, the presence of polyethylene is not authorized, regardless of the sub-content."

4.0 CONTAINMENT EVALUATION

4.1. Review Scope and Objective

TN-BGC-1 is a Type B(U)(F) package designed to transport various forms of fissile material. Although there have been previous revalidation reviews of the TN-BGC-1 package, the focus of this revalidation dealt with the changes associated with the previously revalidated transport of unirradiated uranium-bearing content (designated as Content 11) and unirradiated uranium TRIGA fuel content (designated as Content 26). The French certificate stated that the TRIGA fuel does not come from reprocessing. Staff notes that SAR chapter 6 provided a table relating the IAEA regulations ("IAEA Regulations for the safe transport of radioactive materials", SSR-6, 2018 edition) and the corresponding SAR chapters that demonstrated those regulations were met.

There were no major changes to the package's containment design and content (e.g., activity and decay heat of Content 11 and Content 26) as part of this revalidation submittal, as indicated in document NTE-24-000591-000-1.0 "TN-BGC-1 SAR Summary Changes" (Enclosure 5 to E-63632).

4.2. Description of Containment System

As described in SAR chapter 4 and section 2.5.1 of chapter 3, the packaging can consist of content containers placed within various types of internal fittings and the TN-BGC-1 cavity. According to SAR figures 1 and 2 of chapter 3 as well as appendix 11 and appendix 26 of the French Competent Authority Certificate, the Content 11 and Content 26 radioactive material can be placed within primary packaging boxes (pyrophoric content) and gasketed and threaded cap internal fitting enclosures designated as TN90, AA203, AA203c, AA204; Content 11 also can use the gasketed and threaded cap AA41 internal fitting. The TN-BGC-1 packaging, which includes an aluminum cage structure as well as the package body consisting of inner and outer shells and bottom, closure system, and shock-absorbing cover (according to SAR section 4 of chapter 4), can be transported individually or as multiple packages within a type CB9 physical protection overpack (similar to an ISO 20 feet stainless steel container with an insulator between its two walls, per document RE21078000010, Revision 2, "TN-BGC-1 packaging – Thermal analysis"). SAR table 2 of chapter 3 and subsequent figures within the chapter provided the cavity volumes and diagrams associated with internal fittings and spacers.

SAR section 6 and figure 2 of chapter 4 stated that the containment boundary included the package's stainless steel internal shell, bottom of the body, closure plate, flange, main plug, and quick-connect coupling cap as well as the O-ring gaskets associated with the plug and quick-connect coupling orifice cap; details of the closure system were provided in SAR section 5.3. SAR chapter 9, section 8, listed gaskets and the containment boundary metal components (shell, bottom, closure plate, flange) with a Level 1 safety classification. Details of the gaskets (e.g., Viton grade 67GXX15 material, geometry of gasket) were provided in SAR chapter 4, section 9.6 and table 5; chapter 13, section 3.5, provided the containment boundary's groove dimensions for the gaskets. SAR table 8 of chapter 4 provided the tightening torques of the plug and quick-connect coupling plug.

4.3. Description of Content

As noted above, the content associated with the revalidation includes unirradiated uranium-bearing content (designated as Content 11) and unirradiated uranium TRIGA fuel content (designated as Content 26), described in detail in appendix 11 and appendix 26, respectively, of the French certificate. The French certificate stated that Content 11 can contain up to a maximum 1 g of transuranic elements. The French certificate also stated that TRIGA fuel does not come from reprocessing and no transuranic content was listed. Decay heat was small because Content 11 and Content 26 were based on unirradiated uranium. For example, SAR section 4.3.2 of chapter 16 noted that the bounding uranium-based content was Content 11b with a decay heat of 20mW.

4.4. Description of Containment System Performance

The SAR chapter 11 structural analysis summary noted that the structural strength of the TN-BGC-1 package meets requirements for a fissile Type B package under all transport conditions. Similarly, SAR section 6 and section 7.2 of the chapter 12 thermal analysis summary indicated that temperatures of package components (e.g., gaskets) were within allowable values when transporting content with small decay heat (e.g., 20 mW) during NCT and fire ACT.

SAR chapter 8 provided the minor and major maintenance schedules relating to the containment boundary, including dimensional inspection of closure parts with threads and replacement of O-ring gaskets, as well as leak testing of the closure system (5E-5 Pa m^3/sec standard leak rate [SLR] conditions) for minor maintenance and leak testing of the containment boundary and closure system (1E-7 Pa m^3/sec standard helium conditions) for major maintenance (i.e., including initial fabrication test according to Containment RAI 2 response) by certified personnel, based on ISO standards. In addition, SAR chapter 13, section 4.4, and chapter 7, section 3.3, stated that a pressure rise leakage rate test is performed prior to all transports by certified personnel to an acceptance value of 5E-4 Pa m^3/sec (SLR, according to Containment RAI 2 response) using the package's test plug.

SAR chapter 13 provided the release calculations for determining the allowable content specific activity to be transported to meet the 1E-6 A2/hr NCT regulatory release limit and an A2/week ACT regulatory release limit. According to SAR section 4.2.1, the analysis for determining the initial leakage size parameter was based on the 5E-4 Pa m^3/sec (SLR) allowable test leakage rate (i.e., at standard conditions), provided in section 4.4, for uranium-bearing Content 11 and Content 26; subsequent calculations were based on the gas properties, temperatures, and pressures within the package during NCT and ACT. Using the leakage rate equations provided in SAR chapter 13, section 4.5.1, calculated a 70 A2/g and 50 A2/g specific activity limit for content transported by ground and air, respectively, to meet the above-mentioned regulatory release limit criteria. In addition, SAR section 4.5.1 concluded that the intrinsic specific activity of Content 11 and Content 26 transported in the TN-BGC-1 package would be less than the calculated 70 A2/g and 50 A2/g specific activity limits. This indicates that transported unirradiated uranium content with up to 1 g of transuranic elements would not include high specific activity transuranic radionuclides (or they would have low mass quantities).

The staff notes that calculations presented in ANSI N14.5, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," and ISO 12807, "Safe transport of radioactive materials – Leakage testing on packages," use the regulatory A2 release rate limit at transport conditions (bounding conditions at NCT or ACT) for determining the capillary equivalent diameter, from which subsequent allowable leakage rate test criterion at standard conditions (e.g., SLR) are determined. Using a consensus standard procedure based on descriptions and inputs provided in the application (e.g., no more than 1 g of transuranic elements of content, no substantial high activity transuranic radionuclides, 2 liter (L) minimum content volume, and transported content specific activity less than 70 A2/g, with SAR chapter 13, section 4.5.1, noting the calculated specific activity values "… are higher than the theoretical maximum specific activity of any uranium-bearing contents, even in the presence of up to 1 gram of transuranium elements"), staff finds that the 1E-6 Pa m^3/sec (SLR) and 1E-7 Pa m^3/sec (SHeLR) leakage test rates provided in SAR chapter 8 are reasonable acceptance criteria for containment performance, especially considering that the available free volume can be greater than 2 L (per SAR section 4.3.2.3 of chapter 16) as well as the release calculations did not take credit for the additional confinement functions for content placed within an internal fitting (i.e., closed via a gasketed and threaded cap), packaging primary boxes (i.e., for pyrophoric material), and, if used, the package protection overpack; these additional confinements would reduce the content's release fraction and, hence, further reduce release.

4.5. General Considerations

Chapter 16, which is the SAR subsidiary risk analysis, presented radiolysis and thermolysis analyses for unirradiated uranium-based content; staff notes that many of the equations and parameter values were similar to those presented in document DSEM 0609 (Revision 01) that was an earlier TN-BGC-1 package SAR subsidiary risk analysis. A calculation assumption was that the moles of gaseous components from the content were based on the amounts emanating from a 20-year interim storage period, 1 year of transport under NCT, and 7 days under an ACT. The analyses assumed that the polymer material portion of the content was polyethylene because it resulted in greater radiolytic yields; this can be a conservative assumption because SAR chapter 3, section 2.5.2, noted that polyethylene is not authorized for transport by air or overland transport for sub-contents 11b and 11d. In addition, SAR chapter 16, section 4.3.2.1 noted that temperatures associated with the analyses considered a 60 W decay heat; this is a conservative input because, as noted in section 4.3.2, the bounding content decay heat for the uranium-bearing analyses was 20 mW for Content 11b (which does not include polyethylene). SAR section 4.3.2.2 noted that interior package temperatures during NCT were less than sufficient to generate gases from thermolysis. The SAR markings indicated that certain inputs of the radiolysis analysis, such as radiolytic hydrogen yields (i.e., G_{H2} values) and alpha free path in uranium, as well as the results of the radiolysis and thermolysis calculations of polyethylene found in section 4.3.2.2 and section 4.3.2.3, were not revised; however, calculations were updated in the response to Thermal RAI 1. The results showed that the thermolysis component of hydrogen generation and gas generation during thermal ACT could be greater than radiolysis values. Calculations of hydrogen concentration from radiolysis and thermolysis indicated the concentration value was less than a hydrogen lower flammability limit of 4 percent. Staff notes that the above-mentioned assumption of 1 year transport is supported by the condition in SAR chapter 3, section 2.5.1 (Content 11) and section 2.10.1 (Content 26), which stated that the time between closure of the internal fittings at the shipping facility and the opening of the package at the consignee facility must be less than 1 year when polymer materials are present during transport.

SAR chapter 3 and chapter 16 discussed the conditions for ensuring hydrogen concentration within the package volumes (e.g., internal fittings, package cavity) was less than the lower flammability limit and that corresponding pressures were less than 6.8 bar (gauge) during NCT and ACT. For example, chapter 3, section 2.5.1, stated that there must be a minimum free volume of 2 L in the cavity of an internal fitting (and no more than 20 years of a closed internal fitting storage period) if there is no Poral filter type ventilation device on the internal fitting lid. Section 2.5.2 noted that polyethylene is not permitted for air transport and is not permitted for sub-content 11b and sub-content 11d for overland transport. In addition, Sub-Content 11a and sub-content 11c are limited to 1000 g polyethylene for overland transport, which, according to SAR chapter 16, section 4.3.2, would have less than the 20mW decay heat that formed the basis for the radiolysis calculation. Similarly, chapter 16, section 4.3, stated that the cavity of the package is to have a minimum free volume of 2 L; this is a conservative assumption because section 4.3.2.3 indicated that the minimum free volume in the cavity of the TN-BGC-1 is 7.4 L when conveying a TN90 internal fitting (which is the internal fitting with the largest external volume, according to SAR table 2 in chapter 3, section 3).

4.6. Evaluation Findings

This licensing action is a revalidation of the French Competent Authority Certificate for the TN--BGC-1 package that was approved according to “IAEA Regulations for the Safe Transport of Radioactive Materials”, SSR-6, 2018 edition. Based on a review of the relevant portions of the French Competent Authority Certificate and the representations in the application, the staff has reasonable assurance that the TN-BGC-1 package transporting content designated as unirradiated uranium-bearing Content 11 and unirradiated TRIGA fuel Content 26 with small decay heat (e.g., 20mW) meets the containment requirements found in “IAEA Regulations for the Safe Transport of Radioactive Materials,” SSR-6 (2018 edition) with the following conditions:

- The condition in chapter 3 section 2.5.1 (Content 11) and section 2.10.1 (Content 26) stated that the time between closure of the internal fittings at the shipping facility and the

opening of the package at the consignee facility must be less than 1 year when polymer materials are present during transport.
- A package shipped in a CB9-type overpack is transported under exclusive use.
- For Content 11, if polymeric materials are present and if there is no Poral-type device (described in SAR chapter 3) on the lids of the internal fitting, a minimum free volume of 2 liters in the cavity of the internal fitting is required and the storage time of the closed internal fitting does not exceed 20 years.

5.0 SHIELDING EVALUATION

This shielding evaluation aims to verify the proposed revalidation request for Model No. TN--BGC-1, as it pertains to shielding, provides adequate protection to immediate area workers and public members against direct radiation above the regulatory limits stated in IAEA SSR-6, 2018 edition, “Regulations for the Safe Transport of Radioactive Material.”

The applicant is requesting a revalidation for the TN BGC-1 container (Competent Authority Certificate USA/0492/B(U)F) and issue USA/0492/B(U)F, Revision 22, to revalidate the approval of content No.11(solid, nonirradiated uranium-bearing materials contained within the secondary container) and content No.26 (TRIGA fuel) with special conditions as authorized in USA/0492/B(U)F-96, Revision 21.

5.1. Cask Contents

This package will consist of Content No.11, solid, nonirradiated uranium-bearing materials not from reprocessing sources, and Content No. 26, solid, nonirradiated bars of TRIGA fuel in cylindrical form not from reprocessing sources.

5.2. Design Criteria

The package contains a parallelepiped cage within a cylindrical body fitted with a cover and closure system. The cage is a tubular aluminum structure with internal frames to connect the cage to the packaging. The packaging body has two concentric stainless-steel shells. Between these shells is a defined, resin-filled annulus, a neutron absorber, and a heat insulator. The package uses resin to reinforce the bottom of the packaging: shock-absorbing wood disk and stainless metal sheet metal. A welded, machined stainless steel flange supports the closure system. This package has Content No.11 and Content No. 26 validated.

5.3. Shielding Evaluation

The staff reviewed French Approval Certificate F/313/B(U)F and Chapter 14, “Radiation Protection Analysis,” in the SAR. This report modeled contents from three separate cases consisting of irradiated materials. The routine conditions of transport, ACT, and NCT met the regulatory transport requirements for exclusive use. The packaging material will shield the nonirradiated, which are primarily alpha-emitting materials. Therefore, the shielding evaluation provided in this SAR is bound by the analysis of these contents.

The staff also reviewed TN-BGC-1 against IAEA SSR-6-1, 2018 edition, to ensure the shielding transport conditions meet the regulatory limits for this package. The staff finds the shielding material provided adequate for shielding nonirradiated uranium-bearing materials.

5.4. Evaluation Findings

Based on the discussion above, the NRC concludes that TN-BGC-1 sufficiently meets IAEA SSR-6-1, 2018 edition, to minimize the dose to any actual individual, and previous analyses bound the doses from the package.

6.0 CRITICALITY EVALUATION

The applicant requested adding No.11 (solid, nonirradiated uranium-bearing materials contained within the secondary container) and content No.26 (TRIGA fuel) with special conditions as authorized in USA/0492/B(U)F-96 Revision 21 in various forms of highly enriched uranium bearing materials as allowable contents of the TN-BGC-1 package. Staff reviewed chapter 15 of the SAR (ref. CER/DES/DDSD/DTEL/SGPE/DSEM07615 Rev. B), Technical Note 00048550.01A, “Criticality Safety Analysis in Air Transport Configuration of the TN-BGC-1 Package Model Loaded with Metallic Uranium in Air Transport”, Technical Note 00064690.01C, “Criticality Study of the TN-BGC 1 Package Model Loaded with TRIGA Fuel or Uranium Metal,” and Technical Note 00046711.01, “Criticality Study Package Model Loaded with 3.5 kg of ^{239}Pu Under Metallic Form in ACT,” as well as the relevant criticality safety chapters and appendices as part of this revalidation review.

The applicant provided criticality safety analyses for the TN-BGC-1 package with the requested contents. The applicant used the SCALE 6.0 computer code and 238-group ENDF/B-V cross section library in these analyses. The applicant provided code benchmarking analyses and determined the Upper Subcriticality Limit (USL) which included code biases for criticality safety analyses associated with these types of material compositions. The selected critical experiments included systems with various enrichment of uranium homogeneously mixed with water. The USL for the content was 0.9384 and the applicant used 0.9370 as its acceptance criterion to include additional safety margin.

The applicant calculated the effective neutron multiplication factor, k_{eff}, for a single package as loaded with these various contents to demonstrate that the package met the requirements of paragraph 677 of SSR-6, 2018 edition. The applicant searched for the optimal moderator/fuel concentration with the given container geometry. The applicant calculated the k_{eff} of a single package under ACT to demonstrate compliance with the requirements of paragraphs 671-673 of IAEA SSR-6, 2018 edition. Based on the applicant’s calculation, the maximum k_{eff} of a single package as loaded and flooded with water is below the USL for the contents, and staff finds this acceptable.

Staff also performed confirmatory calculations for transport. Staff used SCALE 6.3, 238 group ENDF/B-VII cross section library. In the models, optimally moderated,100% enriched uranium/water solutions were evaluated at varying heights of cylindrical columns limited by the constraints of the confinement diameters specified in the criticality safety evaluation report. The confirmatory analyses showed that the applicant’s calculations, results, and conclusions are acceptable.

The staff reviewed the application and the applicant’s responses to the staff’s requests for additional information following the regulations of IAEA SSR-6, 2018 edition as well as the guidance provided in SSG-26, 2018 edition. Based on the information provided in the SAR, the applicant’s additional information request responses, as well as staff’s confirmatory calculations, staff finds with reasonable assurance that the addition of Contents No. 11 and 26, as requested, are allowable contents in the TN-BGC-1 package and continues to meet the regulatory requirements of SSR-6 for transportation.

7.0 MATERIALS EVALUATION

The purpose of this materials evaluation is to verify the materials performance of the TN-BGC-1 package to ship Contents No. 11 and No. 26, ensuring the regulatory requirements of IAEA SSR-6 are met. The staff's material review focused on the changes since the staff's most recent recommendation to revalidate the TN-BGC-1 package (ML16109A252), including changes to the contents, mechanical properties used in the structural analysis, thermal properties used in the thermal analysis, gasket materials, bolting materials, and an evaluation of aging effects on the package performance. The staff used NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material", to guide the review of the proposed packaging changes.

7.1. Materials of Construction

As described in SAR chapter 4, section 9.6 and table 1, the applicant modified the fluorocarbon elastomer O-rings from Viton GLT to Viton 67GXX15 for the main plug and quick-connect coupling orifice cap internal seals. These O-rings may still be alternatively fabricated from the previously evaluated Viton VHT silicone.

As described in SAR chapter 4, section 9.1 and table 1, the applicant added the option of using A4-100 austenitic stainless steel for the cage to body middle screws. These screws may still be alternatively fabricated from the previously evaluated A4-80 austenitic stainless steel.

Per the above discussion, the staff finds that the applicant's description of the materials of construction is acceptable, and the package meets the requirements in paragraph 640 of IAEA SSR-6.

7.2. Mechanical Properties

As shown in SAR chapter 4, table 2, the application added mechanical properties for the A2--70 and A4-100 austenitic stainless steel used in the cage to body screws. The staff reviewed the new material properties and verified they are consistent with the values in ISO 3506-1, "Fasteners - Mechanical properties of corrosion-resistant stainless steel fasteners - Part 1: Bolts, screws and studs with specified grades and property classes."

Additionally, the applicant modified the temperature ranges for the previously evaluated temperature dependent mechanical properties provided for bronze, Z2 CN 18.10 type stainless steel, and 6082 T6 aluminum to be consistent with the temperature ranges in the structural and thermal analysis provided in the SAR. The staff reviewed the modified material properties and verified that they are consistent with the previously cited international standards, American Society for Testing and Materials standards, and technical literature.

Based on the evaluation above, the staff finds that the mechanical properties of materials used in the structural analysis are acceptable, and the package meets the requirements in paragraph 652 of IAEA SSR-6.

7.3. Thermal Properties of Materials

As shown in SAR chapter 4, table 3, the applicant added thermal properties for balsa wood, poplar wood, and carbon steel type 39 CD 4. The staff reviewed the new thermal properties and verified that they are consistent with the supplied technical references[2,3,4].

As shown in SAR chapter 4, table 3, the applicant modified the surface emissivity value for stainless steel and modified the thermal conductivity and heat capacity for the resin.

As described in SAR chapter 12, section 4.2, the stainless steel emissivity value was modified to account for the polymer material wrappings around the contents. The staff reviewed the thermal analysis in SAR chapter 12 and the modified thermal properties, and verified that they are consistent with expected values in ASME Boiler and Pressure Vessel Code, section II, part D and/or technical literature.

Based on the evaluation above, the staff finds that the properties of materials used in the thermal analysis are acceptable, and the package meets the requirements in paragraph 652 of IAEA SSR-6.

7.4. Content Reactions

As described in SAR chapter 3, section 2.5, the applicant added a new uranium fluoride form: UO_2F_2 to the list of allowable chemical forms for Content No. 11. In its response, the applicant provided an evaluation of content reactions between UO_2F_2 and the packaging components. The applicant stated that UO_2F_2 is stable at temperatures up to 300°C, at which point highly corrosive hydrofluoric acid generation is possible. The applicant noted that damage to packaging components is unlikely because the highest cavity temperature during accident conditions is 136°C. The staff evaluated the new contents and package design and determined that the contents will not produce significant adverse reactions amongst the packaging components at the temperatures and humidity levels experienced in the package cavity.

Based on the evaluation above, the staff finds that the package design, inspections, and maintenance activities adequately prevent adverse reactions that may affect the ability of the package to perform its safety functions, and the package meets the requirements in paragraph 652 of IAEA SSR-6.

7.5. Seals

As stated in section 7.1 of this SER, the applicant added the option of using fluorocarbon elastomer Viton 67GXX15 for the main plug and quick-connect coupling orifice cap internal seals. The applicant provided material properties in chapter 13, section 3.5 and Attachment 13-4 of the SAR, from manufacture data sheets, to include minimum and maximum operating temperatures, coefficient of thermal expansion, tensile strength, elongation, heat resistance, cold temperature resistance, and compression set. The staff reviewed the specified material properties and the applicant's thermal analysis in chapter 3 of the SAR and determined that these properties are adequate for the range of temperatures and operating environments for

[2] Report on the compression tests at temperature of the various species of wood used in the shock-absorbing covers of the Alternative Energies and Atomic Energy Commission (CEA's) transport packaging, ref. DSN/STMR/EMBAL/LEPE/NOT0007 Rev. 02 of 23 July 2018.
[3] RCC-M code (Design and Construction)
[4] Rules for Mechanical Components of Pressurized-water Reactor Nuclear Islands

NCT and HAC. Based on the above discussion, the staff finds that the applicant's sealing material is acceptable, and the package meets the requirements in paragraph 659 of IAEA SSR-6.

7.6. Package Contents

In chapter 3, section 2.5 of the SAR, the applicant proposed modifications to the previously approved Content No. 11 – solid uranium-bearing materials. Section 2.5.1 describes the chemical forms of the authorized contents and added a new uranium fluoride form: UO_2F_2. Section 2.5 also defines the isotopic composition and masses by defining sub-contents. The applicant proposed leaving sub-contents A and C unchanged, modifying B and D, and removing E, F, G and H. For sub-contents B, contents with any uranium enrichment, the maximum mass of U235 was changed to 8.5 kg. For sub-contents D, contents with enrichment less than or equal to 20 percent, the maximum mass of U235 was changed to 40 kg. In section 2.5.2, the description of internal fittings and packaging used to wedge the contents in place, the applicant added tetrafluoroethylene to the list of authorized polymer materials.

In chapter 3, section 2.10 of the SAR, the applicant proposed modifications to the previously approved Contents No. 26 – TRIGA Fuel. Section 2.10.1 describes the chemical and physical form of the non-irradiated TRIGA fuel elements. The applicant proposed recategorizing the characteristic groups of fuel elements from groups of standard and thin fuel elements to sub-contents families (F1, F2, F3, F4, Marvel, and Miscellaneous). The applicant provided new tables that define the chemical composition and maximum transportable quantities of uranium for each sub-contents family. In section 2.10.2, the description of internal fittings and packaging used to wedge the contents in place, the applicant removed the E7 wedge.

Per the above discussion, the staff finds that the applicant provided an adequate description of the chemical and physical form of the contents, consistent with the guidance in section 7.4.12 of NUREG-2216, and the package meets the requirements in paragraphs 607 and 614 as required by paragraph 652 of IAEA SSR-6.

7.7. Management of Aging Degradation

This section covers the staff's materials review on the updates to the package application to evaluate aging of package components and materials to satisfy the new regulatory requirements in the 2018 Edition of IAEA SSR-6.

7.7.1. Evaluation of Changes to Comply with New IAEA SSR-6 Requirements Regarding Aging

The staff's technical review, documented below, addresses the applicant's identification and evaluation of package component materials, loading conditions, service environments, material aging mechanisms, and description of package inspection and maintenance activities for managing effects of credible aging mechanisms. To assist in performing its review, the staff applied technical knowledge and insights, as appropriate, from the NRC guidance in NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report", July 2019 (ML19214A111), for managing component aging in spent fuel storage systems.

7.7.1.1. Applicant's Evaluation of Material Aging Mechanisms to Comply with IAEA SSR-6, 2018 Edition

To address the new IAEA SSR-6 paragraph 613A requirement and associated SSG-26 guidance related to component aging, Revision C of the SAR contains a new chapter 5, which provides an evaluation of material aging mechanisms for the TN--BGC-1 package components. Additional information on material aging mechanisms was provided in a January 21, 2025, RAI response (ML25119A286). These new sections specify that the planned period of use of the packaging is 40 years.

SAR Chapter 5 and the RAI response provided an evaluation of the material aging mechanisms associated with the transportation and handling of the loaded package. The applicant's evaluation of material aging mechanisms cites the relevant provisions of the maintenance campaigns in chapter 8 to address detection of aging effects and repairs and replacements for the TN--BGC-1 package components.

SAR Chapter 8 describes the applicant's maintenance program, and it is comprised of minor and major intervals. Minor maintenance operations are performed every 3 years or 15 transports (whichever occurs first) and include O-ring replacement, closure system thread inspections, visual inspections of the body, closure system, shock-absorbing cover, and the cage, and a leak test of the closure system. Major maintenance operations are performed every 6 years or 40 transports (whichever occurs first) and include an overload test of cage handling points, a leak test of fuse plugs, a leak test of the containment system, and a penetrant test of cage welds relevant to lifting/handling and tie-down operations. More details on the maintenance program are described in section 7.7.2 below.

7.7.1.2. Package Components and Materials

To identify and evaluate the potential aging mechanisms, chapter 5 of the SAR includes a list of package components and materials that are included in the scope of the aging evaluation. The scope of components subject to aging evaluation includes reusable components of the packaging that perform a safety function. The application identifies the following materials for package components (that receive aging evaluation):

Function	Packaging Components	Material
Containment	Internal shell	Stainless steel
	Plug	Stainless steel
	Quick-connect cap	Stainless steel
	Internal O-rings	Elastomer
Radiological shielding	Internal and external shells	Stainless steel, resin
	Cavity bottom	Stainless steel, carbon steel, resin, wood
	Shock-absorbing cover	Stainless steel, resin, wood
Dissipation of the internal thermal power	Internal and external shells	Stainless steel, resin
Protection against impacts	Cage	Aluminum
	Shock-absorbing cover	Stainless steel, wood
Protection against the effects of fire	Body	Stainless steel, resin
	Shock-absorbing cover	Stainless steel, wood
Sub-criticality maintenance	Cage	Aluminum
	Internal and external shells	Stainless steel, Borated resin

The staff confirmed that the scope of the applicant's aging evaluation includes all long-lived reusable packaging components that perform a safety function. The materials for these components include stainless steel, carbon steel, aluminum, wood, and resin.

The application states that replacement parts such as elastomer gaskets are appropriately inspected in accordance with package handling procedures and replaced in accordance with the maintenance criteria described in chapter 8 of the SAR. Based on the replacement criteria for gaskets, the application identified that it is not necessary to consider aging of the gaskets, and accordingly, these are excluded from the scope of the aging evaluation. The staff agrees that an evaluation of potential aging mechanisms for the gaskets is not needed since they are required by the package handling procedures and maintenance criteria to be replaced during each minor maintenance interval. Per the above discussion, staff finds that the applicant's identification of package components and materials that are subject to aging evaluation is acceptable because it ensures that materials for long-lived reusable package components are appropriately evaluated for potential aging mechanisms over the service life of the package.

7.7.1.3. Licensee's Evaluation of Package Aging Mechanisms

The application states that the planned period of use of the TN-BGC-1 package is 40 years and identifies the environmental and loading conditions that are considered in the evaluation of potential aging mechanisms during this 40-year period. These conditions include highest analyzed temperature (heat) for normal conditions of transport, radiation emitted by the radioactive contents, chemical reactions in materials that may lead to corrosion of components, and fatigue of package structural components caused by cyclical stress in components due to mechanical lifting cycles, package enclosure pressurization cycles, thermal cycles, and vibration during transport. The applicant evaluated each of these conditions to determine aging mechanisms that could potentially lead to package component degradation during the 40-year period of use of the package.

7.7.1.4. Evaluation of Potential Aging Due to Heat

For the evaluation of potential aging due to heat, the applicant determined that the metal components (stainless steel, carbon steel, and aluminum) covered in the aging evaluation would not be susceptible to thermal aging from exposure to heat at the highest analyzed temperature for normal conditions of transport. For the wood components covered in the aging evaluation, the applicant determined that they would not likely have degradation of thermal or mechanical properties from exposure to heat at the highest analyzed temperature for normal conditions of transport. For the neutron-absorbing resins, the applicant provided test reports to demonstrate that they are not sensitive to thermal aging from exposure to heat at the highest analyzed temperature for normal conditions of transport.

The staff reviewed the applicant's evaluation of potential aging due to heat and determined that the highest analyzed temperature for normal conditions of transport is not a concern with respect to aging degradation for any of the reusable package component materials. The staff's determination is based on confirming that susceptibility to the adverse effects of aging mechanisms caused by steady-state high temperatures in the subject materials—such as dimensional changes due to thermal creep in structural alloys, unacceptable reduction in the yield and tensile strength of structural alloys, thermal embrittlement of structural alloys, and changes to the structure and properties of organic materials—occur at significantly higher temperatures for the subject materials than the maximum temperature for normal conditions of transport. Furthermore, the staff determined that the applicant's evaluation is consistent with the guidance in section 7.4.4.3 of NUREG-2216 for creep behavior of aluminum alloy components.

The staff also noted that the aging evaluation appropriately considers the most adverse condition (e.g., highest temperature) for normal conditions of transport since material aging occurs due to long term exposure to normal operating and environmental conditions rather than the more extreme short-duration accident conditions (e.g., fire) that have a low probability of occurrence. Therefore, based on these considerations, the staff finds that the applicant's evaluation of potential aging due to heat exposure, and its determination that there would be no adverse changes to the subject materials from exposure to heat, is acceptable.

7.7.1.5. Evaluation of Potential Aging Due to Radiation

For the evaluation of potential aging due to radiation, the applicant determined that all the packaging materials covered in the aging evaluation would not be susceptible to adverse changes from exposure to radiation since the contents of the TN-BGC-1 package are not irradiated.

The staff reviewed the applicant's evaluation of potential aging due to radiation and notes that the non-irradiated uranium-bearing materials and non-irradiated TRIGA fuel elements contain uranium isotopes that emit predominantly alpha radiation and only very small amounts of gamma and neutron radiation. These alpha particles would be blocked by the inner shell of the packaging and have no effect on the other packaging components. The staff confirmed that adverse changes to mechanical properties such as neutron embrittlement and loss of fracture toughness are not a concern for the metallic and wood package components since the accumulated neutron fluence over 40 years is at least several orders of magnitude lower than the lowest neutron fluence threshold at which adverse changes to the material microstructure and mechanical properties are expected to occur. This is consistent with the guidance in sections 3.2.2.9 and 3.2.3.8 of NUREG-2214.

The staff reviewed the applicant's evaluation of resin aging due to radiation and notes that the boron concentration in the borated resin will decrease as the boron atoms absorb neutrons from the package contents. Significant depletion of the boron atoms may occur over the 40-year period if exposed to sufficient neutron fluence. This is consistent with the guidance in section 3.3.1.1 of NUREG-2214.

The staff determined that aging due to radiation is a credible aging mechanism for the borated resin during the 40-year period of use and requires that the applicant either 1) provide a bounding analysis to show that loss of boron atoms over the 40-year period of use will not result in a loss in effectiveness of the borated resin or 2) manage the loss of neutron attenuation and shielding such as through radiation monitoring and subcriticality testing to ensure the borated resin's continued effectiveness.

7.7.1.6. Evaluation of Potential Aging Due to Chemical Reactions

For the evaluation of potential aging of metal components due to chemical reactions, the applicant stated that 30 years of visual inspections performed in accordance with their maintenance program have not shown any corrosion of packaging component materials, such as stainless steel and aluminum. Additionally, the applicant provided technical literature to demonstrate the compatibility and lack of corrosion with the resin in contact with the stainless steel. The applicant stated there is no risk of galvanic corrosion between the different packaging component materials due to being in a dry environment.

For the evaluation of potential aging of wood due to chemical reactions, the applicant stated that the wood for shock absorption is covered with a stainless steel casing. The applicant stated that these casings are leak tight and would provide an oxygen and moisture free environment that will prevent degradation of the thermal or mechanical properties of wood. The staff notes that the visual inspections of the body and shock-absorbing cover and the leak testing of the containment system, as described in section 7.7.2 below, will verify the integrity of the casings.

For the evaluation of potential aging due to generation of decomposition gases, the applicant stated that gas release during normal conditions of transport would be limited to very small quantities of hydrogen and oxygen. The applicant determined that these small quantities of gas would have no impact on aging of materials.

The staff reviewed the applicant's corrosion evaluation for the stainless steel components in outdoor environments and noted that stainless steel passivity may adequately inhibit general corrosion. This is consistent with the guidance in section 3.2.2.1 of NUREG-2214. But, as described in section 3.2.2.2. of NUREG-2214, stainless steel is susceptible to localized corrosion effects, including loss of material due to pitting and crevice corrosion when exposed to aqueous outdoor air environments. Over extended operating periods, in particular during numerous package transport operations over a 40-year period, these chemical species may gradually degrade the protective passive oxide film on stainless steel surfaces leading to the formation of pits and crevice corrosion. As described in section 3.2.2.3, dissimilar metals in physical contact, as is the case between the aluminum cage and the stainless steel body, are susceptible to galvanic corrosion in the presence of conducting solutions, such is the case in aqueous outdoor air environments. As described in section 3.2.2.5 of NUREG-2214, stainless steel components under high tensile stress (such as weld residual stress) exposed to aqueous outdoor air environments are also susceptible to the formation of cracks due to chloride-induced SCC. For interior sheltered environments, such as those associated with the interior of the packaging enclosure, the staff noted that localized corrosion and SCC of stainless steel surfaces are not a concern provided that the interior surfaces do not undergo sustained or frequent exposure to inleakage of aqueous electrolytes from outdoor rainwater mixed with halide-bearing chemical compounds present in the outdoor environment.

Per the above discussion, the staff determined localized corrosion, SCC, and galvanic corrosion are credible aging mechanisms for stainless steel components in outdoor environments, during the 40-year period of use, and require that adequate visual inspections performed by qualified personnel using qualified techniques are needed in order to detect and evaluate indications of corrosion so that personnel can reliably determine the need for remedial action, such as repair or replacement of components that show unacceptable indications.

Although not covered in the applicant's corrosion evaluation, the staff assessed the susceptibility of the carbon steel distribution plate in its embedded environment and determined, as described in section 3.2.1 of NUREG-2214, that galvanic corrosion, localized corrosion, galvanic corrosion, and SCC are not credible aging mechanisms over the 40-year period of use due to limited exposure to water and oxygen.

The staff reviewed the applicant's corrosion evaluation for aluminum alloy components in outdoor environments and notes that aluminum alloys exhibit good corrosion resistance due to the formation of a passive protective layer. This is consistent with the guidance in section 3.2.3.1 of NUREG-2214. But, as described in section 3.2.3.2. of NUREG-2214, aluminum alloys are susceptible to localized corrosion effects, including loss of material due to pitting and crevice corrosion, when exposed to aqueous outdoor air environments. Additionally, as described above, the aluminum cage and the stainless steel body are susceptible to galvanic corrosion. Per the above discussion, the staff determined localized corrosion and galvanic corrosion are credible aging mechanisms for aluminum components in outdoor environments, during the 40-year period of use, and require that adequate visual inspections performed by qualified personnel using qualified techniques are needed in order to detect and evaluate indications of corrosion so that personnel can reliably determine the need for remedial action, such as repair or replacement of components that show unacceptable indications.

The staff evaluated the aging of wood components due to chemical reactions and determined that the stainless steel casings surrounding the wood provided adequate protection from moisture and oxygen that could lead to rot, or moisture induced degradation such as warping or cracking. Additionally, the staff evaluated aging due to off gassing of packaging materials (resin and wood) and determined that the rate of generation of these gases (hydrogen and oxygen) would be negligible at the temperatures associated with normal conditions of transport.

7.7.1.7. Evaluation of Structural Fatigue Due to Stress Cycles in Package Components

The applicant included in their RAI response a fatigue evaluation for demonstrating that the package components will not be susceptible to fatigue failure due to accumulated stress cycles in components.

The only components of the package that the applicant found required a study of fatigue strength were the tie-down components. This fatigue study showed that these components have a significant service life with 141,692 occurrences of acceleration. The staff notes that these tie-down components would also be visually inspected during minor maintenance operations and any damage repaired or replaced as necessary.

The staff's evaluation of structural fatigue is contained in section 2.3 of this SER.

7.7.2. Criteria for Managing Effects of Aging Mechanisms on Package Components

Chapter 7 of the SAR describes package handling procedures that cover loading operations, inspections before shipment, and unloading methods. Section 3 provides pre-shipment inspections, including visual inspections to verify:

- The condition of the package and document any defects.
- The condition of the gaskets and replaced if necessary.
- The condition of the contents, the internal diameter of the cavity, and wall thicknesses of internal fittings.

After closing the package, the leakage rate is measured to ensure it is less than the value indicated in the transport authorization for the contents transported. Additionally, dose rates are measured radially at level with the material and axially at the bottom of the package.

Chapter 8 of the SAR provides maintenance criteria, including criteria for periodic visual inspection of the packaging components, to ensure components with defects that significantly lower safety are replaced and any package that fails to meet maintenance program criteria is removed from service until the package is brought into compliance. The maintenance criteria specifies that minor maintenance operations are performed every 15 transports or every

3 years, whichever occurs first. Additionally major maintenance operations are performed every 40 transports or every 6 years, whichever occurs first.

The maintenance criteria for the minor maintenance intervals includes the following:

- Replacement of the O-ring gaskets.

- Inspections of closure system parts of the containment system:
    - Go-no-go dimensional checks of threads are performed in accordance with the standard “NF ISO 1502: ISO general-purpose metric screw threads — Gauges and gauging”.
    - Visual checks of all closure system parts, with appropriate lighting, for the presence of foreign materials, scrapes, nicks, blows, impacts, and scratches on the gasket bearing surfaces. Defective components will be repaired or replaced as applicable.

- Visual inspection of the body, shock-absorbing cover and cage:
    - Performed in accordance with standard "NF EN 13018 /2016: Non-destructive testing - Visual testing - General principles” and with appropriate lighting.
    - Inspection personnel must meet the requirements of the standard “NF EN ISO 9712 / 2012 – Non-destructive testing – Qualification and certification of NDT personnel.”
    - Components surfaces are cleaned prior to inspection.
    - Visual inspections will look for the presence of foreign material, scratches, nicks, notches, dents, impact, indentions, and deformations on package components and hardware. Additionally, welds will be inspected for cracks and crazing. The packaging cavity will be inspected for traces of corrosion.
    - Defective components will be repaired or replaced, or the package removed from service if repair or replacement is not possible.
    - Repairs of upper oblique cage bars are followed by a load test and a penetrant test of welds.
    - Indentations (5 mm or greater) on parts of the body and cover containing resin will undergo neutronics testing to ensure compliance. Likewise, indentions (5mm or greater) on parts of the body and cover containing wood will be removed from service and corrective action determined.

- Leak testing of the closure system:
    - Criterion: 10^{-6} $Pa \cdot m^3/s$ standardized leak rate.
    - Personnel are certified in accordance with Standard ISO 12807: Safe transport of radioactive materials - Leakage testing on packages.

The maintenance criteria for the major maintenance intervals includes the following:

- Overload testing of the handling points (upper oblique bars of the cage):
    - Performed by an approved inspection body.
    - 15-min duration.
    - No permanent deformation allowed.

- Leak testing on the fuse plugs of the cover.

- Helium leak testing of the containment system (closure system and body):

- Criterion: 10^{-7} Pa·m^3/s standardized helium leak rate.
- Personnel are certified in accordance with Standard ISO 12807: Safe transport of radioactive materials - Leakage testing on packages.

- Penetrant testing of the accessible welds in the upper and lower parts of the cage (areas under stress during lifting/handling and tie-down). Acceptance criteria per Annex I1.A2 of CODAP 2005 for construction category A.

Based on a review of the package handling and maintenance criteria described in chapters 7 and 8 of the SAR, the staff recommends the applicant provide DOT with the following:

1. Specific provisions for inspection of stainless steel components to detect and evaluate indications of localized corrosion, SCC, and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.

2. Specific provisions for inspection of aluminum components to detect and evaluate indications of localized corrosion and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.

3. A bounding analysis to show that loss of boron atoms over the 40-year period of use will not result in a loss in effectiveness of the borated resin, or specific provisions to manage loss of neutron attenuation and shielding, such as through radiation monitoring and subcriticality testing, to ensure the borated resin's continued effectiveness.

Per the above discussion, the staff finds the Aging Management Program, with the additional conditions provided above, is adequate for managing the aging mechanisms identified per the aging management review.

7.8. Evaluation Findings

Based on a review of the statements and representations in the application, the staff concludes that the applicant adequately described and evaluated the materials used in the TN-BGC-1 package, with the conditions described in section 7.7.2 above, and that the package meets the requirements of IAEA SSR-6 (2018).

CONDITIONS

The NRC recommends revalidation of French Competent Authority Certificate F/313/B(U)F-96, for use in the United States with the conditions stated below. The staff notes, however, that the application does not provide any calculation to demonstrate that the package meets the tie-down system design requirement using acceleration factors in the SSG-26, table IV.2.

- Regarding component fatigue evaluations for tie-down during road transport, a weld inspection (as identified in the Major Maintenance penetrant exam scope in chapter 8 of the TN-BGC-1 SAR) should be performed prior to the first use of each package in the U.S., unless the package is new or has undergone this type of inspection within 5 transports or one year of its first use in the U.S. If a package meeting the description above is then used exclusively in the U.S., resumption of the normal maintenance program is acceptable (see section 2.5 of this SER).

- If pyrophoric contents, including metal powders, are transported, inerting of the primary packaging boxes should be performed, internal fitting, and the TN-BGC-1 cavity (see SAR chapter 3 and SAR chapter 16, section 5).
- For Content 11, the presence of transuranic elements up to a maximum of 1 g is permitted.
- The mixing of different sub-contents within the same package is prohibited (see SAR chapter 3 section 2.5.1).
- The transport period between package closure and opening at the destination facility will not exceed 1 year (see section 3 of French certificate).
- The presence of polyethylene is not authorized for air transport, regardless of the sub-content (Content 11, see chapter 3). In addition, for overland transport, the presence of polyethylene:
    - is not authorized in sub-contents No. 11b and 11d,
    - is authorized up to 1000 g in sub-contents No. 11a and 11c.
- The condition in chapter 3 section 2.5.1 (Content 11) and section 2.10.1 (Content 26) stated that the time between closure of the internal fittings at the shipping facility and the opening of the package at the consignee facility must be less than 1 year when polymer materials are present during transport.
- A package shipped in a CB9-type overpack is transported under exclusive use.
- For Content 11, if polymeric materials are present and if there is no Poral-type device (described in SAR chapter 3) on the lids of the internal fitting, a minimum free volume of 2 liters in the cavity of the internal fitting is required and the storage time of the closed internal fitting does not exceed 20 years.
- Based on a review of the package handling and maintenance criteria described in chapters 7 and 8 of the SAR, the staff recommends the applicant provide DOT with the following:
    - Specific provisions for inspection of stainless steel components to detect and evaluate indications of localized corrosion, SCC, and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.
    - Specific provisions for inspection of aluminum components to detect and evaluate indications of localized corrosion and galvanic corrosion to ensure that components with unacceptable corrosion are repaired or replaced.
    - A bounding analysis to show that loss of boron atoms over the 40-year period of use will not result in a loss in effectiveness of the borated resin, or specific provisions to manage loss of neutron attenuation and shielding, such as through radiation monitoring and subcriticality testing, to ensure the borated resin's continued effectiveness.

CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, the staff concludes the Model No. TN-BGC-1 package design, with the above stated conditions, meets the IAEA requirements of SSR-6, 2018 Edition. Therefore, the staff recommends revalidation of French Competent Authority Certificate F/313/B(U)F-96, for use in the U.S. with the above stated conditions.

Issued with letter to R. Boyle, DOT.