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

July 10, 2019

Mr. Richard W. Boyle
Radioactive Materials Branch
U.S. Department of Transportation
1200 New Jersey Avenue SE
Washington, D.C. 20590

SUBJECT: CERTIFICATE OF APPROVAL NO. F/357 /B(U)F-96, FOR THE MODEL NO. TN-MTR PACKAGE – REVALIDATION RECOMMENDATION

Dear Mr. Boyle:

This is in response to your letter dated September 25, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18292A563), as supplemented on October 17, 2018 (ADAMS Accession No. ML18345A235), October 24, 2018 (ADAMS Accession No. ML19177A067), and February 7, 2019 (ADAMS Accession No. ML19115A243), requesting our assistance in evaluating the Model No. TN-MTR transport package, authorized by the French Certificate of Approval No. F/357 /B(U)F-96, Revision Eaf.

Based upon our review, the statements and representations contained in the application, in the Package Design Safety Report No. DOS-06-00032593 Rev. 15, as supplemented, and for the reasons stated in the enclosed safety evaluation report, we recommend revalidation of French Certificate of Approval No. F/357 /B(U)F-96 (REVISION Eaf) transport package, with the following additional conditions:

1. MTR fuel elements loaded in a MTR-52SV2 basket as described in the French Certificate, Appendix 9, Content No. 9, specifically Sub-Content Number 1, intact elements, fuel type U_ySi_2 in Table 9.2.1 of the French certificate, excluding U_3O_8 alloy fuel assemblies, including:
   - Maximum of 22 spent fuel assemblies plus 16 fresh fuel assemblies per shipment,
   - Minimum enrichment of 19 weight percent U-235 and maximum enrichment of 21 weight percent U-235,
   - Maximum mass of uranium per assembly of 2 kg,
   - Assembly burnup less than 45,000 MWd/MTU,
   - Assembly cooling time greater than 964 days,
   - Maximum decay heat of 5 watts/assembly, and
   - Assemblies are to be loaded symmetrically in the center of the MTR-52SV2 basket with the fresh fuel assemblies on the exterior of the spent fuel assemblies for added shielding

2. The Mechanical Criterion for cladding in Table 9.2.1 of the French Approval Certificate must use aluminum yield strength values for fully-annealed material. Data from Table 0A-9.10 of the package design safety report that reflect a fully-annealed condition are those for aluminum type 1050A, 3003, 5052, 5454, 5086, and 5083.

3. Prior to each shipment, the seals of the TN-MTR must demonstrate no leakage when tested to a sensitivity of at least 1E-4 Pa-m^3/sec standardized leakage rate (or 1E-3-ref-cm^3/sec).

4. The lid ethylene propylene diene monomer seals and the two cover plate EPDM seals are to be replaced at an interval not to exceed 1 year.

If you have any questions regarding this matter, please contact me or Bernard White of my staff at (301) 415-6877.

Sincerely,

/RA Christian Jacobs Acting for/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3052
EPID L-2018-RNW-0022

Enclosure:
Safety Evaluation Report

SUBJECT: CERTIFICATE OF APPROVAL NO. F/357 /B(U)F-96, FOR THE MODEL NO. TN-MTR PACKAGE, DOCUMENT DATE: July 10, 2019

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

SAFETY EVALUATION REPORT
Docket No. 71-3052
Model No. TN-MTR
French Certificate of Approval No. F/357 /B(U)F-96
Revision Eaf

SUMMARY

By letter dated September 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18292A563), as supplemented on October 17, 2018 (ADAMS Accession No. ML18345A235), October 24, 2018 (ADAMS Accession No. ML19177A067), and February 7, 2019 (ADAMS Accession No. ML19115A243), the U.S. Department of Transportation (DOT) requested that the U.S. Nuclear Regulatory Commission (NRC) review and provide a recommendation regarding revalidation of French Certificate of Approval No. F/357 /B(U)F-96, Revision Eaf, for the Model No. TN-MTR package. Specifically, the DOT requested that the NRC only review the intact material test research (MTR) spent fuel assemblies in the MTR-52SV2 basket (as described in Appendix 9 attached to the French certificate as content number 9, sub-content 1, excluding U_3O_8 alloy fuel elements.

In support of this request the DOT provided the following documents with its letter dated September 26, 2018:

1. French Certificate of Approval No. F/357 /B(U)F-96, Revision Eaf, dated July 31, 2018.

2. Package Design Safety Report No. DOS-06-00032593 Rev. 15.

The NRC previously reviewed and recommended revalidation of this package to the DOT on March 19, 2004 (ADAMS Accession No. ML040820100). Based on the statements and representations in the information provided by DOT and TN Americas LLC, the NRC staff recommends that French Certificate of Approval No. F/357 /B(U)F-96, Revision Eaf, dated July 31, 2018, be revalidated for the contents listed below (see Section 1.2, “Contents”).

1.0 GENERAL INFORMATION

1.1 Package Description

The TN-MTR package is designed to transport irradiated MTR-type research reactor fuel. The packaging consists of three main components: the package body, the impact absorbing cover, and the lid. The package body is constructed of stainless steel and incorporates lead gamma shield and thermal insulation for protection against fires. Fins are welded to the external shell of the packaging. The top part of the body includes a flange that has tapped holes for the lid and the lid screws. Two trunnions are screwed onto the top flange. Tie-down lugs are welded to the outer shell. The impact absorbing cover, or impact limiter, consists of wood blocks encased in a steel skin, and is attached to the package body by screws.

Enclosure

The package is closed by a stainless steel lid that also incorporates lead for radiation shielding and is equipped with two ports for draining and drying. The lid is fixed to the cask body by 36 screws and has double elastomeric O-ring seals for containment and to facilitate leak-testing.

A removable fuel basket within the package cavity accommodates 52 MTR spent fuel elements. The package has two basket designs. The second basket design, the MTR-52SV2, is composed of a stack of 10 stainless steel disks and 9 aluminum disks, assembled with 7 tightening rods. The steel disks have 52 square openings for fuel elements. The aluminum discs are composed of an outer aluminum ring that supports borated aluminum plates that are assembled to form 52 square housings. The borated aluminum plates act as neutron absorbers. The 52 openings for the fuel elements are lined with square stainless steel sleeves.

The package has the following approximate dimensions and weights:

Overall package diameter, with impact limiter 2080 mm
Overall package height, with impact limiter 2008 mm
Package cavity diameter 960 mm
Package cavity height 1080 mm
Packaging mass, without basket and contents 20,600 kg
Maximum package mass, with contents 23,400 kg

1.2 Contents

The DOT requested the NRC review and recommend revalidation on the uranium silicide and uranium alloy fuel in Table 9.2.1 of the French certificate, as shown below in Table 1, with a maximum decay heat of 5 Watts (W) per fuel location for a total decay heat of 260 W.

Table 1 –TN-MTR Fuel Element Characteristics

Type of fuel elements		Assembled flat plate elements	
Fuel core		U_3Si_2 alloy, possibly mixed with aluminum	UAl_x or U_ySi_z alloy, possibly mixed with aluminum
Cladding	Material	Aluminum cladding	
	Mechanical criterion	$1{,}475.5 \times \frac{\rho_{eq}\, b^2}{t} \le R_e$	
Number of elements in Basket		≤51[2]	40[3]
Enrichment by weight of ^{235}U (%)		≤19.95	≤95
V/^{235}U[4] (g/cm^3)		≥2.5	≥2.37
Mass of ^{235}U per compartment before irradiation (g)		≤470	≤460
Mixes[5]		Prohibited	
Presence of solid aluminum end pieces		authorized	
Height of active part of plates (mm)		≤1.030	
Protrusion of the active part outside the basket (mm)		0	

(1) Where:
b, nominal width of the plate (mm),
t, nominal thickness of the plate (mm),
ρ_{eq}, the equivalent density of the plate (fissile core and cladding) (kg/mm^3),
Re, the yield strength of the fuel cladding (MPa) considered at the maximum temperature under normal conditions of transport depending on the power actually transported, as given in Table 0A-9.10 in Chapter 0A-9 of the Package Design Safety Report DOS-06- 00032593-013 Rev. 5, depending on the grade of cladding used.

(2) The loading plan must comply with Figure OA-9.4 of Chapter 0A-9 of the Package Design Safety Report ref. DOS-06-00032593-013 Rev. 5.

(3) The loading plan must comply with Figure OA-9.3 of Chapter 0A-9 of the Package Design Safety Report ref. DOS-06-00032593-013 Rev. 5.

(4) V/^{235}U: ratio calculated for the flat plates:
where V is the product of the nominal thickness of a plate multiplied by the nominal fissile height and by the nominal fissile width,
where ^{235}U is the total mass of ^{235}U of the element divided by the number of plates.

(5) Elements which satisfy the characteristics of one of the columns in this table, and are "authorized", as indicated, can be mixed with other "authorized" elements in this table.

(C) Minimum Transport Index for Criticality Control: 0.

2.0 STRUCTURAL EVALUATION

2.1 Structural Evaluation

The TN-MTR safety analysis report (SAR) Revision 15 review was conducted, in part, by comparing the previously approved version (Rev. 2) of the SAR from 2004. The certificate of approval for the TN-MTR issued by the Autorité de Sûreté Nucléaire (ASN), the French competent authority, states that only the MTR-52SV2 basket is used for shipments. Thus, other basket types that are described in Revision 15 of the SAR (such as the MTR-44 basket which makes use of a different lid) have not been evaluated. As a result, this review only focuses on components related to the use of the MTR-52SV2 basket.

According to the applicant, the proposed contents include fewer fuel assemblies for the planned shipments, then authorized by the French certificate. In addition, the structural performance of the fuel is bounded by the previously approved package design (Rev. 2 of the SAR (ADAMS Accession No. ML033210332) and the proposed contents in this application generate less decay heat.

Structurally, the package has not significantly changed in overall dimensions, center of gravity, weight, and construction from the previous version except as noted below; therefore, the TN-MTR package will experience the same g-loads as observed in the previously approved version (Rev. 2) of the SAR from 2004. Only those changes that affect the structural performance of the package have been examined as part of this review.

2.1.1 Comparison of Material Changes and Effect on Testing

Several components of the TN-MTR have changed with respect to material properties. This includes the addition of J steel as an alternative to A steel, and K steel which is an alternative to B steel for certain components. Both J and K steels have higher yield and tensile strengths than A and B steels, respectively, according to Table 0.7 of the application and do not have rupture elongation values listed in that table. However, communication from the applicant indicates that as-built components made from J steel have elongation values that exceed those listed for A steel. The following parts from a structural point of view are made from J or K steel:

- The Flange (Part 105 on Drawing No. PLA-15-00166811-000, Revision 00) made from J steel rather than A steel
- The lid (Part 307 on Drawing No. PLA-15-00166811-000, Revision 00) made from K steel rather than B steel

The previously approved package was certified using drop testing. The drop testing report indicated that the flange did suffer some inelastic deformation around the trunnion cavity. The engineering drawings of the current version indicate that the flange is to be made of J steel rather than A steel. No drop testing was performed for the current version of the package. Since J steel has higher yield, tensile strength, and rupture elongation values than previously approved A steel, the staff concludes that less damage would be observed by the flange made of J steel if a similar drop test on the trunnions were conducted.

With respect to the lid (Part 307), previous drop testing indicated that it did not suffer any damage during any of the drops. Since the lid is now constructed of K steel with higher yield and rupture strength than previously approved B steel, the lid will continue to maintain its structural integrity.

The package also makes use of class 10.9 material screws rather than class 8.8 material screws (Part 150 on Drawing No. PLA-15-00166811-000, Revision 00) which secure the lid to the package. Since class 10.9 material screws have higher yield and rupture strength, the package will continue to function as intended.

The orifice cover plugs are secured by screws (Part 350 on Drawing No. PLA-15-00166811-001, Revision 00). The previously approved package made use of class 12.9 material screws, which have a higher yield and rupture strength than currently called for class 10.9 material screws; however, the applicant provided an analysis that demonstrated class 10.9 material screws will continue to maintain the structural integrity of the package during normal conditions of transport and hypothetical accident conditions.

2.2 Materials Evaluation

The staff's review focused on the materials changes that occurred since the NRC's previous recommendation to revalidate the TN-MTR package in 2004. The review was also limited to the changes related to the specific fuel shipments for which the revalidation request is intended. The fuel shipments include intact aluminum-clad U_3Si_2, UAl_x, or U_ySi_z alloy (possibly mixed with aluminum) flat plate fuel elements shipped in the MTR-52SV2 fuel basket.

Significant materials changes include additional steel grades for structural components, a new insulation material, revised material temperature limits for the elastomeric containment seals, and additional detail on the mechanical properties of the MTR fuel cladding. The fuel contents have not changed.

2.2.1 Structural Material Properties

The applicant added two types of steels for the structural components of the packaging. "Type J" austenitic stainless steel is equivalent to the Association Française de Normalisation (AFNOR) standard Z3 CN 18-10 (similar to unified numbering system (UNS) S30403) and the "Type K" duplex stainless steel is equivalent to AFNOR standard Z3 CND 22 05 Az (similar to UNS S32205).

In its review of the stainless steel mechanical properties in SAR Table 0.7, the staff noted that the properties do not correspond to the minimum requirements of the AFNOR steel standards. In response to a staff question, the applicant clarified that the properties in SAR Table 0.7 represent the minimum required properties the package (rather than the AFNOR standard). Further, SAR Section 10.3 states that the yield strength, ultimate tensile strength, and elongation values in SAR Table 0.7 are the minimum values to which the package must comply.

Based on the package characteristics defined in SAR Section 10.3, which establish clear requirements for mechanical properties, the staff finds the use of the cited structural material properties to be acceptable.

2.2.2 Chemical, Galvanic, and Other Reactions

The staff reviewed the materials changes to the package and verified that they do not introduce any adverse corrosive or other reactions that were not previously considered in the staff's prior revalidation of the package. The added stainless steel grades are similar to the stainless steels previously evaluated, and the staff has found these materials to be compatible with the service environment. In addition, the staff evaluated the Vyal B thermal insulation and found that the insulation's vinylester-based matrix and mineral fillers are chemically compatible with the adjacent lead and stainless steel components within the enclosed service environment.

2.2.3 Thermal Insulation

The applicant added an option to use the Vyal B resin for the thermal insulation. The insulation is also credited in the dose rate analyses. The staff notes that the NRC has approved the use of Vyal B for thermal insulation and neutron shielding in other transportation packages (e.g., Model No. NUHOMS-MP197HB in Certificate of Compliance No 9302 and Model No. TN-LC in Certificate of Compliance No. 9358, NRC, 2014a and 2012, respectively).

The staff reviewed technical data for the Vyal B resin (Issard, 2009; Issard and Abadie, 2011) and the staff's prior reviews of this material in other packages (NRC, 2012 and 2014a) and verified that the thermal conductivity, specific heat, density, and maximum operating temperature are consistent with the values in SAR Table 0.8. In addition, the applicant performed thermal testing on a full-scale packaging to validate the assumptions in the thermal calculations. The applicant found that the test results confirm that the values for the insulation thermal conductivity are appropriate.

Based on data from the technical literature and the confirmatory thermal testing, the staff finds the use of the Vyal B resin to be acceptable.

2.2.4 Elastomer Seals

The containment seals are manufactured from an ethylene propylene diene monomer (EPDM) rubber. The specification of EPDM did not change since the previous NRC recommendation to revalidate this package. However, the current SAR now identifies specific allowable EPDM seals manufactured by STACEM and Le Joint Francais. The SAR also identifies a higher short-term maximum operating temperature compared to the value in the SAR previously evaluated by the NRC (220 °C vs. 170 °C).

The NRC previously had concerns about the validity of a 220 °C short-term operating temperature limit for EPDM seal materials (NRC, 2014b). For example, the staff notes that the Le Joint Francais seal catalogue (Hutchinson, 2015) states that the maximum operational temperature limit for EPDM materials is 165 °C. The applicant's thermal analysis in Chapter 2 of the SAR, which assumes a 5500 W decay heat, shows that the seal temperatures can reach 201 °C in a fire accident.

The staff notes that the specific fuel shipments associated with the current revalidation request limit the decay heat in each basket cell to only 5W, totaling a maximum of 260 W for the package (5W x 52 cells in the MTR-52SV2 basket). Given the significantly lower actual decay heat compared to the applicant's bounding analysis (5500 W), the staff has reasonable assurance that the EPDM seals will maintain their function in a fire accident. The staff recommends an addition to Condition No. 1 to ensure that the decay heat is limited to 5W per basket cell to limit the temperature of the seals.

Condition No. 1: *The decay heat must not exceed 5W per fuel basket cell.*

Based on the condition that establishes a conservatively-low decay heat, the staff finds the use of the elastomeric seal material to be acceptable.

2.2.5 Fuel Cladding Mechanical Properties

The applicant added SAR Table 0A-9.10 to provide temperature-dependent yield strengths of the various grades of aluminum that may be used as fuel cladding. These values are used in the calculation that must be performed for each shipment to ensure that the plates do not deform in an accident, per Table 9.2.1 of the French approval certificate.

The staff reviewed the properties in Table 0A-9.10 against data in the technical literature (e.g., ASM, 1998), and notes that, in many cases, the cited aluminum yield strengths may not be conservative. Although the strength values for many aluminum grades conservatively reflect those of an annealed alloy (i.e., Temper O), the strengths of other grades assume higher-strength tempers (e.g., T6). The basis for choosing higher-strength tempers for some grades was unclear to the staff, given that the aluminum cladding is subject to several heating and rolling steps in the fuel manufacturing process and long-term elevated temperature operation in the research reactor. As a result, the staff recommends Condition No. 2 to ensure that conservative values of yield strength are used in the accident analyses for cladding prior to shipment.

Condition No. 2: *The Mechanical Criterion for cladding in Table 9.2.1 of the French Approval Certificate must use aluminum yield strength values for fully-annealed material. Data from SAR Table 0A-9.10 that reflect a fully-annealed condition are those for aluminum type 1050A, 3003, 5052, 5454, 5086, and 5083.*

Based on the condition that establishes the conservative use of annealed aluminum properties, the staff finds the structural calculation methodology for the cladding to be acceptable.

2.3 Conclusion

Staff evaluated the amendment request and concludes that provisions 719, 722, 727, and 730 of the International Atomic Energy Agency (IAEA) Specific Safety Requirements, No. SSR-6, “Regulations for the Safe Transport of Radioactive Material,” 2012 edition will continue to be met.

References

ASM, “Aluminum and Aluminum Alloys: Properties of Wrought Aluminum Alloys”, Metals Handbook, Desk Edition, 2nd Edition, ASM International, Materials Park, Ohio, 1998.

Hutchinson, Precision Sealing Catalogue (https://www.oring.hutchinson.fr/en/media/catalogue/), March 2015.

Issard, H. “Development of Neutron Shielding Materials for Nuclear Fuel Storage Facilities,” 20th International Conference on Structural Mechanics in Reactor Technology, Espoo Finland. 2009.

Issard, H. and Abadie, P. “Aging Tests of Neutron-Shielding Materials for Transport of Storage Casks,” Nuclear Technology, Vol. 176, No. 1, pp. 2-8, 2011.

NRC, Revalidation of the French Certificate of Approval No. F/357/B(U)F-96 for the TN-MTR Package, ADAMS Accession No. ML040820100, Letter dated March 19, 2004.

NRC, Safety Evaluation Report for the Model No. TN-LC Package, Docket No. 71-9358, Certificate of Compliance No. 9358, Revision No. 0, ADAMS Accession No. ML12366A082, 2012.

NRC, Safety Evaluation Report for the Model No. NUHOMS-MP197HB Package, Docket No. 71-9302, Certificate of Compliance No. 9358, Revision No. 7, ADAMS Accession No. ML14114A132, 2014a.

NRC, Safety Evaluation Report for the Revalidation of the French Certificate of Approval No. F/379/B(U)F-96 for the Model No. TN-106 Package, Docket No. 71-3075, ADAMS Accession No. ML14162A339, 2014b.

3.0 THERMAL EVALUATION

3.1 Thermal Design

The approval request for the Model No. TN-MTR package is for two specific shipments of the TN-MTR package. The content of the TN-MTR package includes the MTR-52SV2 basket, and the specific contents to be shipped is specified in the French Approval Certificate (Number F/357/B(U)F-96, Revision Eaf), and specifically Table 9.2.1 for Content 9.

The evaluation of the thermal performance of the package components is provided in the thermal section of the application, specifically Chapter 2, and Chapter 2, Appendices 1, 1.1, and 2.

The maximum decay heat for the contents requested is 5 W per fuel assembly, total maximum package decay heat of 260 W. The analysis in the application is for 5500 W; therefore, the analyses referred to are conservative.

3.2 Thermal Evaluation

Given that the decay heat of the proposed contents is within previously analyzed parameters (42 W per fuel location), there is reasonable assurance that none of the package components or contents will exceed allowable temperature limits, nor will the package exceed the prescribed pressure limits.

In accordance with the issuing country certificate of compliance, if packages are transported inside a closed conveyance, the contents decay heat must be such that any of the package components evaluated for the regulatory atmospheric conditions must remain within their temperature limits as defined in the package design safety report.

As discussed in Section 2.2.4 above “Elastomer Seals,” the staff identified a concern regarding the performance of the package seals during the fire test for hypothetical accident conditions. In order to address the staff’s concern for this application Condition No. 1, to ensure that the decay heat is limited to 5 W per basket cell to limit the temperature of the seals, has been proposed.

3.3 Conclusion

Based on the statements and representations in the application, the staff finds that the thermal evaluation has been adequately described, and the thermal performance of the package meets the requirements of the IAEA Specific Safety Requirements, No. SSR-6, 2012 edition.

4.0 CONTAINMENT EVALUATION

Except for slight design changes discussed in the staff's structural evaluation and the materials evaluation, the package design has not changed from the previous revalidation. Although the applicant submitted a revised containment analysis, limited comments on this analysis are provided below. In addition, the content associated with this revalidation is a reduced subset of the content and activity that was approved in the previous revalidation. Although Chapter 1 (page 11/34 and subsequent pages) states that the package is in compliance with the 1996 IAEA requirements, the French Approval Certificate (Number F/357/B(U)F-96, Revision Eaf) certified the TN-MTR using the IAEA Safety Standards Series, No. SSR-6, 2012 edition. The French Approval Certificate indicated that the content could be shipped by road, rail, or sea.

4.1 Description of Containment System

As described in Chapter 0 (pages 3/38 and 13/38) and Chapter 3A (page 6/32), the containment boundary includes the bottom of containment body, shell, flange, lid, inner seal of lid, orifice A plug cover and inner seal, and orifice B plug cover and inner seal; a pictorial of the containment boundary is shown in Figure 0.1 (Chapter 0, page 38/38). It is noted in Chapter 0 (page 9/38) that Orifice A and Orifice B are composed of self-closing, quick connect couplings used for the vent and drain, respectively. However, these components do not comprise the containment boundary; rather, plug covers are placed over the quick connect couplings and are tightened via screws. Details of the Orifice A and Orifice B quick connects and cover plates and their arrangement relative to the entire TN-MTR package are provided on pages 10/12 and 11/12 of Chapter 00. According to Chapter 00 and Chapter 3A, the lid is attached to the flange using 36 M30 screws and each of the orifice A and orifice B cover plates are attached to the lid using four M12 screws; the torque values for these screws were provided in Chapter 6A (page 20/26) and Chapter 0 (page 30/38).

The containment boundary is constructed of stainless steel (properties provided in Chapter 0, page 31/38); the lid seal and orifice A and orifice B cover plate seal material is EPDM (Chapter 0, page 28/38). Chapter 0 (page 15/38) provides the properties of the EPDM O-rings. Chapter 0 (page 15/38) states that the weld filler material has mechanical characteristics at least equal to the base metal and Chapter 0 (page 16/38) indicates that the package is constructed according to Code Français de Construction des Appareils à Pression (CODAP) 95 rules (French pressure vessel code). According to Chapter 0 (page 15/38), the EPDM seals have a steady-state maximum operational temperature of 160 °C and a short period maximum operational temperature of 220 °C. As noted in Chapter 2 (page 4/23), lid and orifice plug inner seal temperatures were 114.5 °C and 117.1 °C, respectively, at normal conditions and 201.1 °C and 190.6 °C during hypothetical accident conditions. These temperatures were based on an analysis that assumed a decay heat of 5500 W. This thermal analysis decay heat would bound the decay heat associated with the present revalidation because, as noted below, this revalidation has a reduced subset of the original content. Therefore, inner seal temperatures would be lower than those reported in Chapter 2.

4.2 Description of Content

The content consists of flat plate MTR fuel. For this revalidation, the transport basket for the MTR fuel elements is the MTR-52SV2. The fuel core is either a U_3Si_2, UAl_x or U_ySi^z alloy and the cladding, which is bonded to the fuel core, is aluminum. The content associated with this revalidation is limited to Sub-content No. 1 (intact elements) as denoted in the French Approval Certificate Table 9.1, which prohibits the loading and transporting of failed/damaged elements. In addition, the content for this revalidation is a subset of the content approved in the previous revalidation. For example, the leakage rate acceptance criterion provided in Chapter 6A and Chapter 7A was based on MTR bounding fuel with a burnup of 450,000 MegaWatt days/ton U (MWd/t U) and cooling time of 1 year. However, the content for this revalidation is based on a burnup of 45,000 MWd/t U and a cooling time of 2 years. Therefore, the activity of releasable material for this revalidation is expected to be less than the quantity of releasable material for the application's bounding fuel. This is because the burnup is reduced by an order of magnitude, recognizing that gamma-related activity of fission products is proportional to burnup and the neutron-related activity is proportional to the burnup raised to the fourth power. Likewise, the doubling of the cooling time results in an additional year for radioactive isotopes to decay to a ground state. Finally, the number of spent fuel assemblies to be transported for this revalidation is reduced from 52 from the previous revalidation to either 18 irradiated standard fuel assemblies with four irradiated control fuel assemblies for one package or 14 irradiated standard fuel assemblies for another package. The reduced number of fuel assemblies further reduces the content and available activity from that previously revalidated.

4.3 Description of Containment System Performance

Chapter 3A and Chapter 3A-Appendix 1 provided an updated containment analysis that the applicant used to determine leakage rate test acceptance criteria. The analysis included considerations of activity of fission gases (including Kr-85 and tritium) and aerosols (which also includes CRUD, (see October 24, 2018 supplement)); the analysis did not explicitly address volatiles (Cs, Ru, Sr). In addition, the analysis provided in Appendix 3A-1.1 did not explain the rationale for assuming 200 grams (g) of U as being a representative quantity of "sheared" fuel under normal conditions of transport. It is noted that the presence of volatiles and the amount of failed fuel at normal conditions of transport and hypothetical accident conditions are inputs in determining a release activity. However, results of the updated containment analysis showed that the activity release at normal conditions of transport (1.57E-7 A_2/hr) is less than the maximum activity release at normal conditions (3.3E-7 A_2/hr) that was reported in the previously approved revalidation (2004) with the same calculated content. It is noted that the hypothetical accident condition activity release for both the updated calculation (1.76E-1 A_2/week) and in the previously approved revalidation (6.8E-2 A_2/week) were below the regulatory limit of A_2/week. Therefore, recognizing that the content associated with this revalidation is a subset of the earlier approved revalidation, the activity release would be even less than the reported values of the current revalidation and the previously approved revalidation with the same calculated content.

Section 2.5 of Chapter 7A (page 4/12) indicated that the containment lid inner seal, inner seals of the two orifice covers, and containment system welds undergo a fabrication leakage rate test with an acceptance criterion of 3.5E-5 Pa-m^3/sec standardized leakage rate (SLR). An additional consideration of the fabrication leakage rate test is that the welds undergo a leakage rate test and must satisfy an acceptance criterion of 1.1E-7 Pa-m^3/sec SLR. In addition to the fabrication leakage rate test, Chapter 7A describes the pre-shipment and maintenance/periodic leakage rate test acceptance criteria for each shipment cycle (Table 7A.1), every 15 shipment cycles (or 3 years, whichever is shorter, per Table 7A.2), and 60 shipment cycles (Table 7A.3). Chapter 7A (page 10/12) and Chapter 6A (page 22/26) indicated that, prior to shipment, the space between the seals of the lid and two orifice cover plates of a package are leakage rate tested such that the sum of the leakage rates (including all uncertainties) does not exceed the leakage rate criterion of 4.7E-4 Pa m^3/sec SLR. In addition, prior to each shipment the seals are replaced if necessary. As part of the 15 cycle program, in addition to inspection of the package welds (white dye penetration) and the seals being replaced, the seals are tested for leakage to a 3.5E-5 Pa m^3/sec SLR criterion. For the 60 cycle program, in addition to the seals being replaced and tested to the 3.5E-5 Pa m^3/sec SLR criterion, the welds are inspected (white dye penetration) and tested for leakage to a 1.1E-7 Pa m3/sec SLR criterion. It is noted that the above mentioned leakage rate test criteria are associated with the previously revalidated content; the criteria have not been relaxed for the reduced subset of content described above. Finally, Chapter 6A (page 22/26) indicated that the leakage rate tests are to comply with the International Organization for Standardization (ISO) Standard 12807 "Safety of radioactive material transport – leak testing of packages" and the tests are to be performed by qualified personnel in accordance with the COFREND II quality assurance system.

According to Chapter 3A Appendix 1 (page 3/27), the maximum potential activity release is 1.57E-7 A_2/hr under normal conditions of transport and 1.76E-1 A_2/week under hypothetical accident conditions. These release rates, which are nearly an order of magnitude less than those required by the regulations, were based on the full MTR-52SV2 content; therefore, the potential release rates for the reduced subset of content would be less.

According to Chapter 1 Appendix 1 (page 3/16), calculations show that the package would withstand an external gauge pressure of at least 1.5 bars (water immersion test) and would resist a more severe pressure of 20 bars gauge. Likewise, according to Chapter 3A (page 6/32), the containment boundary is tested to an internal pressure of 7 bars. This test pressure is larger than the 0.853 bar pressure at normal conditions of transport and 6.85 bar pressure at hypothetical accident conditions (Chapter 3A, Appendix 1, page 3/27); these pressures are associated with previous bounding fuel content and its higher decay heat. It is noted that, for any potential leak paths, the normal conditions pressure in the package, being lower than ambient pressure, would tend to drive flow into the package cavity rather than outward into the ambient.

The integrity of the containment boundary after normal conditions of transport tests and hypothetical accident conditions tests was discussed in Chapter 1. According to Chapter 1 (pages 17/34 and 18/34), the normal conditions of transport tests (water spraying, free drop, stacking, and penetration) would not reduce the effectiveness of the containment. Likewise, pages 17/34 through 30/34 indicated that hypothetical accident conditions tests would not impact the effectiveness of containment. In addition, Chapter 2 (page 4/23) indicated that the maximum temperature of the lid inner seal and the orifice plug inner seals under normal conditions of transport and hypothetical accident conditions were below their respective limit values. Section 2.0 of the structural evaluation in this SER provides further discussion associated with this revalidation.

According to SER Section 2.2.2, there were no material changes that would introduce corrosion or other reactions and no changes in the drying criteria that would affect radiolysis. According to SER Section 2.2.2.2, there were no material changes that would introduce corrosion or other reactions and no changes in the drying criteria that would affect radiolysis.

As discussed above, except for slight design changes discussed in the staff's structural evaluation and the materials evaluation, the package design has not changed from the previous revalidation. In addition, the content and activity associated with this revalidation is a reduced subset of that approved in the previous revalidation. It was noted earlier that as part of the revalidation request, the application included changes related to package containment analyses. Because the package and content were previously revalidated, these updated containment analyses were not extensively reviewed; staff is not expressing any view related to this material at this time beyond the earlier discussion about volatiles (Cs, Ru, Sr) and the assumed 200 g U value for sheared fuel.

4.4 Conclusions

Based on the representations in the application, the staff evaluated the revalidation request and concludes that containment-related provisions of the IAEA Specific Safety Requirements, No. SSR-6, 2012 edition will continue to be met. The staff recommends, however, that the revalidation be conditioned as follows:

1. Prior to each shipment, the seals of the TN-MTR must demonstrate no leakage when tested to a sensitivity of at least 1E-4 Pa-m^3/sec SLR (or 1E-3 ref-cm^3/sec).

2. The lid EPDM seals and the two cover plate EPDM seals are to be replaced at an interval not to exceed one year.

5.0 SHIELDING EVALUATION

The staff reviewed the application to ensure that the shielding is adequate to meet the radiation level requirements within the IAEA Safety Standards in SSR-6, 2012 Edition, for this package. Specifically, for exclusive use packages, Paragraph 573(a) of SSR-6 requires that the surface radiation level not exceed 2 mSv/hr (200 mrem/hr) unless the provisions in subpart (a) (i), (ii), and (iii) are met; then the surface radiation level shall not exceed 10 mSv/hr (1 rem/hr). Paragraph 573(b) requires that the maximum radiation level at the surface of the vehicle not exceed 2 mSv/hr (200 mrem/hr) and Paragraph 573(c) requires that the radiation level not exceed 0.1 mSv/hr (10 mrem/hr) at any point 2 m from the vertical planes represented by the outer lateral surfaces of the vehicle, or, if the load is transported in an open vehicle, at any point 2 m from the vertical planes projected from the outer edges of the vehicle. For Type B(U) packages, Paragraph 652 requires that the package meet the requirement in Paragraph 648 of SSR-6. Paragraph 648(b) states that under the tests for normal conditions of transport, the package cannot experience more than a 20% increase in the maximum radiation level. Paragraph 659(b)(1) of the SSR-6 requires that the package does not exceed 10 mSv/hr (1000 mrem/hr) at 1 m under hypothetical accident conditions.

The revalidation request was limited to intact MTR fuel assemblies in an MTR-52SV2 basket. The contents are described in Appendix 9 to the French certificate as content number 9, sub-content 1, excluding U_3O_8 fuel elements. The applicant further limited the contents to a maximum of 22 spent fuel assemblies plus 16 fresh fuel assemblies, minimum enrichment of 19 weight percent (w/o) U-235 and maximum enrichment of 21 w/o U-235, maximum mass of uranium per assembly of 2 kg, burnup less than 45,000 MWd/MTU, cooling time greater than 964 days and a maximum decay heat of 5 watts/assembly. These assemblies are to be loaded symmetrically in the center of the MTR-52SV2 basket with the fresh fuel assemblies in the outer basket locations, surrounding the spent fuel assemblies for added shielding.

The TN-MTR packaging (including the lid) is cylindrical in shape with a height of 2.008 meters and a diameter of 2.080 meters. The cavity has a height of 1080 mm and a diameter of 960 mm. The body is composed of lead, surrounded by thermal protection made of resin to protect from fire and is enveloped in two containments made of stainless steel. It has a lid that is composed of lead surrounded by a stainless steel casing. The package includes a shock absorbing cover made of wood enclosed in stainless steel.

5.1 Source Term

The applicant calculated the source term of the MTR fuel using the CESAR 5.3 code. The depletion characteristics used to generate the source term are in Section 2.3 of Chapter 4A-11 and include a maximum burnup of 150,000 MWd/MTU, minimum cooling time of 1 year, enrichment of 21 w/o, and maximum fuel mass of 2.564 kg Uranium. With the exception of enrichment, these parameters are conservative with respect to the contents being requested within the revalidation request. The applicant depleted the MTR fuel using a continuous operating cycle which is conservative as the MTR reactor is, in reality, operated intermittently which would allow for further decay.

The staff performed an independent evaluation of the source term. The staff simulated the depletion using the 1D depletion sequence, T-DEPL-1D for the TRITON code using the ENDF/B-VII cross section library. This code is part of the SCALE 6.2.3 code package. The staff modeled the fuel assembly using the requested parameters (rather than the bounding parameters in the SAR). This includes minimum enrichment of 19 w/o, burnup of 45,000 MWd/MTU and a cooling time of 964 days. The staff used the most limiting of the fuel assembly parameters within the October 17, 2018, submittal with respect to specific power as well as assembly mass. Although this combination of parameters is not representative of an actual assembly within the October 17, 2018, submittal, the parameters encompass the requested contents and are conservative as they do not exist simultaneously in any one of the assemblies requested for shipment. The staff also assumed a continuous irradiation cycle. The staff compared the results of its calculations to the source term from Tables 4.A-11.2 and 4.A-11.3 of the application for the neutron and gamma source term, respectively, with the source term normalized to gammas or neutrons per second per MTU. The analysis comparison shows that the applicant's source term is more conservative by about an order of magnitude for the gamma source and about 1-2 orders of magnitude for the neutron source term. The staff determined that this is expected as it is generally known that for spent fuel the gamma source term is proportional to burnup and the neutron source term is proportional to burnup to the fourth power. This gives the staff reasonable assurance that the source term used by the applicant in its calculations to determine external radiation levels is conservative.

5.2 Package Model

The applicant evaluated the shielding capability for the TN-MTR package by calculating the external radiation level with the MTR fuel assemblies within the MTR-52 basket and states that this is representative of the MTR-52S and MTR-52SV2. Based on the drawing "TN MTR 52SV2 Basket Concept Drawing," (Drawing Reference No. 4466-104) and the description of the modeled basket of the MTR-52 within Chapter 4A-11 of the application, the staff found that the model of the MTR-52 basket would adequately represent the characteristics important for shielding performance of the MTR-52SV2. The drawings of the MTR-52SV2 basket do not contain manufacturing tolerances and the information within Chapter 4A-11 does not contain specific information on the basket dimensions. The staff based its determination on the fact that the number and arrangement of the assemblies and the materials of the basket construction are the same as the actual basket and the staff's engineering judgement that any uncertainties in basket dimensions are insignificant with respect to their effects on radiation levels. The staff also found it acceptable based on the conservatism in the source term. In addition, the applicant is requesting approval of a maximum of 22 spent fuel assemblies, and assuming 52 assemblies is a conservative assumption.

The applicant homogenized the MTR plate fuel in each basket cell and modeled it as U-Al. Although the requested contents also contain uranium-silicon alloys, the staff compared the density used within the model as described in Chapter 4A-11 of the application to that of the contents submitted to the staff within the supplement dated October 17, 2018, and found that the modeled fuel would bound the requested contents because the assumed density is lower than that of an actual fuel assembly per Enclosure 3 of the supplemental submittal dated October 17, 2018. The staff found that uncertainties in the modeling of the fuel would be bounded by the conservatisms in the source term and number of irradiated assemblies.

The staff compared the representation of the package body in Chapter 4A-11 of the application to that of the drawings “TN-MTR Packaging – Safety Drawing (General View),” (Drawing Reference No. PLA-15-00166811-000). The staff found that the dimensions used by the applicant within the shielding model are equivalent to the dimensions in the licensing drawing. The applicant did not consider manufacturing tolerances as identified within the drawings. Although this is a non-conservative assumption, the staff found that the conservatism in the source term as well as the number of irradiated assemblies would bound this uncertainty.

Accident Conditions

Under hypothetical accident conditions the applicant calculated the external radiation level assuming the resin is replaced by air, neglecting the cover, and implementing a reduction in lead height and thickness due to lead slump. The staff found that these considerations would adequately represent the package under hypothetical accident conditions and that they were consistent with, or bounding for, the drop and fire tests’ effects discussed in Chapters 1 and 2 of the application. The staff did not evaluate if the amount of lead slump assumed was appropriate, nor the adequacy of the geometry of the spent fuel (i.e. same as the intact) and used its judgment that the uncertainty in these modeling assumptions would not cause the package to exceed regulatory radiation level due to the conservatism in the source term and reduced number of irradiated assemblies.

5.3 Evaluation Method

The applicant calculated the external radiation level of the package using the TRIPOLI 4.7 Code coupled with the CEAV5 cross section library. This code is internationally recognized and available through Nuclear Energy Agency of the Organization for Economic Co-operation and Development. It is also distributed by the Radiation Safety Information Computational Center managed by Oak Ridge National Laboratory. The staff finds that the code is capable of performing the necessary radiation level calculations. The staff also took into account the conservatism within the evaluation in assessing that the code is acceptable for the evaluation.

Flux-to-Dose Rate Conversion Factors

The applicant used the flux-to-dose-rate conversion factors from International Commission on Radiological Protection (ICRP) Publication 60, “1990 Recommendations of the International Commission on Radiological Protection,” (ICRP-60). ICRP-60 is an update to ICRP-26, “Recommendations of the International Commission on Radiological Protection,” which provided the basis for ICRP-51, “Data for Use in Protection against External Radiation.” ICRP-51 was used as a basis for the ANSI/ANS 6.1.1-1991, “American National Standard for Neutron and Gamma-Ray Fluence-To-Dose Factors,” standard that was subsequently withdrawn because a formal review to determine its accuracy was not performed. The NRC has continued to accept analyses using ANSI/ANS-6.1.1-1977, “American National Standard for Neutron and Gamma-

Ray Flux to Dose Rate Factors," flux-to-dose rate conversion factors (NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2000). The staff did not investigate the quantitative change in radiation level when using ANSI/ANS-6.1.1-1977 over ICRP-60 for flux-to-dose-rate conversion factors. However, the staff's experience with the ANSI/ANS-6.1.1-1991 flux-to-dose-rate conversion factors show that they are non-conservative but by less than the amount of conservatism within the applicant's source term. In comparing the differences between ICRP-26 and ICRP-60 (Nuclear Energy Agency of the Organization for Economic Co-operation and Development, ISBN 978-92-64-99153-8, "Evolution of ICRP Recommendations 1977, 1990 and 2007," 2011) the staff did not find that the updates incorporated within ICRP-60 would produce an increase in radiation level so significant that the analysis conservatisms would not be able to adequately compensate for the increase. Therefore, the staff found the ICRP-60 flux-to-dose-rate conversion factors acceptable for this evaluation based on the conservative source term and number of irradiated assemblies. It should be noted that the dose rates calculated with ICRP-60 conversion factors do not result in dose rates that are the same, in terms of their meaning, as the radiation level limits in the regulations. However, as the staff has described above, given the amount of conservatisms in the applicant's analysis for the requested package content, the staff found the use of dose rates determined with ICRP-60 conversion factors to be acceptable for this particular evaluation.

5.4 Radiation Levels

The applicant evaluated the radiation level at the top, bottom and several locations on the side of the package. The applicant found that the most limiting surface location is at the bottom of the package. From drawing "TN MTR 52 SV2 Basket Concept Drawing," (Drawing Reference No. 4466-104), the side and the bottom of the package have the same amount of shielding; however, since the basket walls would provide extra radial shielding, the staff found that higher dose rates at the axial ends would be expected. The applicant shows the radiation level results for the MTR-52 basket for Content 1 in the tables in Section 5.2.4 of Chapter 4A-11 of the application. These tables show that the highest surface radiation level is 1.193 mSv/hr. This is below the limit of 2 mSv/hr from SSR-6 Paragraph 573(a) and 573(b).

The applicant evaluated the maximum radiation level at 2 meters from the surface of the vehicle which is conservatively assumed as being the same as the overpack. This value is shown for the MTR-52 basket for Content 1 in the tables in Section 5.2.4 of Chapter 4A-11 of the application and is 0.065 mSv/hr. This is below the limit of 0.1 mSv/hr from SSR-6 Paragraph 572(c).

The applicant evaluated the accident condition radiation level for Content 1 in the MTR-68 basket. The routine condition radiation levels for this basket are higher than that of the MTR-52 basket for this content as shown in Section 5.2.1 of Chapter 4A-11 of the application. Therefore, the staff found that using the MTR-68 basket in the analysis would be conservative for estimating the hypothetical accident condition radiation level. As shown in Section 5.3.1 of Chapter 4A-11 of the application, the applicant calculated the radiation level at the top, bottom and various locations on the side at 1 meter from the surface and calculated the maximum radiation level under accident conditions to be 7.8 mSv/hr. This is below the limit of 10 mSv/hr from SSR-6 Paragraph 659(b)(1).

In Appendix 4A.1 of the application, the applicant justifies how the package complies with the criterion in Paragraph 648(b) that under normal conditions of transport that the package cannot experience more than a 20% increase in the maximum radiation level. The staff reviewed the information in this appendix. This information shows that under the normal conditions of transport drop tests specified in Paragraph 722 of SSR-6, there is some deformation to the shock absorbing cover that would cause radiation levels to increase slightly due to the reduced distance to the detector. The applicant evaluated the difference in radiation levels and showed that they do not increase by more than 20%. The staff found the applicant's evaluation demonstrates that the package meets Paragraph 648(b) of SSR-6.

5.5 Conclusion

As discussed in the above paragraphs, the staff has reasonable assurance that the TN-MTR package with intact MTR fuel assemblies in an MTR-52SV2 basket and further limited in quantity and irradiation parameters, as discussed above, meets the requirements in Paragraphs 573, 648(b), 659(b)(1) in SSR-6. The staff recommends revalidation of French Certificate of Approval No. F/357/B(U)-96 Rev. Eaf for the TN-MTR package with the following conditions:

- Contents limited to those described in Appendix 9 to the French certificate as content number 9, sub-content 1, excluding U_3O_8 fuel elements,
- Maximum of 22 spent fuel assemblies plus 16 fresh fuel assemblies,
- Minimum enrichment of 19 w/o U-235 and maximum enrichment of 21 w/o U-235,
- Maximum mass of uranium per assembly of 2 kg,
- Assembly burnup less than 45,000 MWd/MTU,
- Assembly cooling time greater than 964 days,
- Maximum decay heat of 5 watts/assembly, and
- Assemblies are to be loaded symmetrically in the center of the MTR-52SV2 basket with the fresh fuel assemblies on the exterior of the spent fuel assemblies for added shielding.

6.0 CRITICALITY EVALUATION

This criticality safety review focuses on the specific fuel shipments requested by the applicant and how they are integrated into the previous revalidation of the TN-MTR package from 2004, specifically, regarding use of the MTR-52SV2. The proposed fuel shipments consist of a limited number of intact aluminum-clad fuels and are approved for shipment under IAEA regulations SSR-6, Section VI, 2012 Edition.

6.1 Description of Criticality Design

The package design and criticality safety features are unchanged from the previously approved MTR-52SV2 fuel basket transported in the TN-MTR package. The MTR-52SV2 is constructed using a peripheral non-borated aluminum ring supporting neutron absorbing plates made up of a mix of aluminum and either boron or boron carbide and is designed to hold up to 52 MTR fuel assemblies of varying compositions, uranium loadings, and enrichments. The applicant utilized a volume (V) to ^{235}U ratio to determine the maximum mass and optimum moderation of the various fuel assembly types allowed in the TN-MTR package. Since the fuel elements allowed in the TN-MTR package can be either flat, curved, or cylindrical, for conservatism they are considered as homogeneous mixtures of either U-Al_x, U_3O_8-Al, or U_3Si_2-Al, mixed with H_2O. Using this method allows for the most reactive fuel configurations to be determined regardless of the geometry of the fuel plates, and uses the mass of ^{235}U per fuel plate, the ^{235}U enrichment, and the ratio of V/^{235}U to define the most reactive fuel configurations. The fuel elements are modeled in position within the fuel basket and are fixed in place such that they cannot travel between basket compartments. Damaged fuel was analyzed independently as 20 failed fuel elements placed in a canister in various configurations to determine the most reactive failed fuel configuration. The borated aluminum alloys used in the criticality analysis take 75% credit of the nominal ^{10}B content, which is consistent with industry practice.

6.2 Spent Nuclear Fuel Contents

This revalidation considers only two specific fuel types. The first consists of 14 U_3Si_2-Al spent flat plate fuel assemblies enriched to a maximum of 19.75 w/o, with a maximum mass of ^{235}U per assembly before irradiation of $\leq$ 470 grams. The second fuel type consists of 22 spent fuel assemblies and 16 fresh fuel assemblies, also composed of U_3Si_2-Al flat plate fuel assemblies enriched to a maximum of 19.75 w/o with a maximum mass of ^{235}U per assembly before irradiation of $\leq$ 460 grams. Both the fuels are considered sub-content no. 1 as specified in Table 9.2.1 of the F/357/B(U)F-96 approval certificate. All fuels to be transported are considered as fresh fuel within the criticality safety analyses for the purposes of this revalidation and are intact assemblies. Failed, degraded, or materially damaged fuel assemblies are not allowable contents under this revalidation

6.3 General Considerations for Criticality Safety

The original revalidation of the MTR-52SV2 fuel basket allowed for up to 51 U_3SI_2-Al alloy fuel assemblies enriched up to 19.95 w/o, or up to 40 U_ySi_z or UAl_x alloy fuel assemblies enriched up to 95 w/o. All of the fuel assemblies proposed in this revalidation are enriched to less than 19.75 w/o, which is well bounded by the original analysis. In addition, the limited number of fuel assemblies requested in this revalidation are far below the maximum permissible number of fuel assemblies allowed in the MTR-52SV2 fuel basket (ref. Table 9.2.1) and do not fully load any individual MTR-52SV2 basket. Array configurations and the criticality safety index are unchanged from the previous analysis, and bound the contents requested in this application. Air transport of package is not allowed. The criticality safety analysis code and benchmarking for this code are unchanged from the previous revalidation and apply to the fuel types proposed by the applicant.

6.4 Demonstration of Maximum Reactivity

Staff evaluated the proposed fuel types as allowable contents and found the MTR fuel types to be bounded by the original revalidation for the MTR-52SV2 fuel basket in the TN-MTR transportation package. In all instances, the proposed fuel types were bounded by the uranium enrichment and uranium loadings per assembly for U_3SI_2-Al alloy fuel assemblies as provided in the original revalidation. The previous revalidation considered normal conditions of transport and hypothetical accident conditions, optimum moderation, water intrusion, and fuel assembly shifting, all resulting in k_{eff}s that were below the upper subcritical limit of 0.93 in the fully loaded configuration at the given fuel parameters. The limited number of fuel assemblies requested in this revalidation are well below the currently approved basket loading parameters, and subsequently possess a much lower overall uranium mass when loaded into an MTR-52SV2 basket, which is conservatively bounded by the original analysis.

6.5 Evaluation Findings

The staff found that the proposed contents within the TN-MTR package and MTR-52SV2 basket will remain subcritical for all routine, normal and accident conditions of transport. Specifically, these contents are:

- 14 U_3Si_2-Al spent flat plate fuel assemblies enriched to a maximum of 19.75 w/o, with a maximum mass of ^{235}U per assembly before irradiation of $\leq$ 470 grams; and

- 22 spent fuel assemblies and 16 fresh fuel assemblies, composed of U_3Si_2-Al flat plate fuel assemblies enriched to a maximum of 19.75 w/o with a maximum mass of ^{235}U per assembly before irradiation of ≤ 460 grams.

The staff based its finding on its verification of adequate system modeling performed by the applicant in the previous approval and the limited requested contents in this application that are bounded by that analysis. The acceptance standard of a maximum k_{eff} of 0.93 was maintained for all analyzed scenarios and meets the requirement that the package maintain subcriticality under all conditions of routine, normal and accident conditions as required by SSR-6 2012 edition Paragraphs 637(a) and 682.

7.0 OPERATING PROCEDURES EVALUATION

Chapter 6A of the safety analysis report and the French approval certificate include sections on package acceptance, loading, unloading, and pre- and post-shipment requirements. The operating procedures have specific measures to be taken prior to each shipment, including drying the package cavity after loading the fuel elements in a wet environment, installing the package closures and confirming that the lid screws are properly torqued, leak testing the lid and containment penetration seals, taking radiation measurements, and affixing tamper indicating seals.

The operating procedures specify that the pre-shipment leakage test should confirm that the leakage from all the seals is less than 4.7 10^{-4} Pa-m^3 /s SLR. The staff agrees that for undamaged fuel elements this leakage testing is adequate for the pre-shipment leak test, and no additional conditions are recommended regarding leakage testing. However, it is recommended that the approval be conditioned to specify that the fuel elements must be undamaged.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION

The acceptance test program includes: a) document examination; b) visual examination; c) leak testing; d) shielding integrity tests; e) thermal tests; f) packaging and basket operating tests; and g) packaging markings to ensure that the package is fabricated in accordance with the design approved in the French certificate. The maintenance program includes requirements for each shipment, after 15 cycles (or 3 years whichever is less), and after 60 cycles. the maintenance tests for each cycle, 15 cycles and 60 builds upon one another to ensure continued efficacy of the package design.

CONDITIONS

1. MTR fuel elements loaded in a MTR-52SV2 basket as described in the French Certificate, Appendix 9, Content No. 9, specifically Sub-Content Number 1, intact elements, fuel type U_ySi_2 in Table 9.2.1 of the French certificate, excluding U_3O_8 alloy fuel assemblies, including:
    - Maximum of 22 spent fuel assemblies plus 16 fresh fuel assemblies,
    - Minimum enrichment of 19 w/o U-235 and maximum enrichment of 21 w/o U-235,
    - Maximum mass of uranium per assembly of 2 kg,
    - Assembly burnup less than 45,000 MWd/MTU,
    - Assembly cooling time greater than 964 days,
    - Maximum decay heat of 5 watts/assembly, and

- Assemblies are to be loaded symmetrically in the center of the MTR-52SV2 basket with the fresh fuel assemblies on the exterior of the spent fuel assemblies for added shielding

2. The Mechanical Criterion for cladding in Table 9.2.1 of the French Approval Certificate must use aluminum yield strength values for fully-annealed material. Data from Table 0A-9.10 of the package design safety report that reflect a fully-annealed condition are those for aluminum type 1050A, 3003, 5052, 5454, 5086, and 5083.

3. Prior to each shipment, the seals of the TN-MTR must demonstrate no leakage when tested to a sensitivity of at least 1E-4 Pa-m^3/sec standardized leakage rate (or 1E-3-ref-cm^3/sec).

4. The lid ethylene propylene diene monomer seals and the two cover plate EPDM seals are to be replaced at an interval not to exceed 1 year.

CONCLUSION

Based on the statements and representations contained in the documents referenced above (see SUMMARY), and the conditions listed above (see Contents), the staff concludes that the Model No. TN-MTR package meets the requirements of International Atomic Energy Agency Regulations for the Safe Transport of Radioactive Material, IAEA Safety Standards Series, No. SSR-6, 2012 edition.

Issued with letter to R. Boyle, Department of Transportation,
on ____________.