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

SAFETY EVALUATION REPORT

Model No. FCC-4 Package
French Certificate of Approval No. F/348/AF-96
Docket No. 71-3097
Revision Fq

Enclosure 1

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SUMMARY

By letter dated January 14, 2020 [Agencywide Documents Access and Management System (ADAMS) Accession No. ML20022A001], and as supplemented on June 29, 2020 (ADAMS Package Accession No. ML20189A607), the U.S. Department of Transportation requested that the U.S. Nuclear Regulatory Commission (NRC) staff performs a review of the French Approval Certificate Number F/348/AF-96, Revision Fq, Model No. FCC-4 transport package, and make a recommendation concerning the revalidation of the package for import and export use.

The NRC reviewed the information provided to the DOT by Orano in its application for the Model No. FCC-4 package and its supplements against the regulatory requirements of International Atomic Energy Agency (IAEA) SSR-6, "Regulations for the Safe Transport of Radioactive Material," 2012 Edition (IAEA SSR-6 thereafter). Based on the statements and representations in the information provided by DOT and the applicant, the staff recommends the revalidation of French Approval Certificate Number (No.) F/348/AF-96, Revision Fq, Model No. FCC-4 package, for the contents included in Section 1.1.2, "Contents," of this safety evaluation report (SER).

1.0 GENERAL INFORMATION REVIEW

Document No. DOS-19-021166-000-NPV, Version 1.0, (ADAMS package accession No. ML20035D441) includes the list of documents related to this submittal with the documents' dates and revision numbers.

1.1 Package Description

1.1.1 Packaging

The Model No. FCC-4 packaging has a length of 5,748 millimeters (mm), width of 1,134 mm, and a height of 1,297 mm and a maximum loaded weight of approximately 5,550 kg. The packaging is composed of a lower shell (the base of the packaging), an upper shell (packaging cover), a cradle made up of two stringers and connected to the lower shell by rubber shock mountings, and an internal system (frame and doors) leaving space for two cavities to accommodate the authorized and proposed contents.

1.1.2 Contents

The design of the package is described in the approved French Competent Authority Certificate F/348/AF-96, Version Fq, dated August 14, 2019, and application for the Model No. FCC-4. The contents consist of pressurized water reactor's (PWR's) assemblies or rods in a box. The contents are allowed in either Version 1 or Version 2.

a) Version 1 - PWR 17x17 fuel assemblies, square fuel rod array (XL, XLR, EPR™, and GAIA) or for non-assembled fuel rods grouped in channels with a CSI = 0.625

b) Version 2 - PWR 16x16 or 18x18 fuel assemblies, square fuel rod array with a CSI = 8.33.

The FCC4, Version 1, packagings hold a single 17x17 EPR™ fuel assembly in a cluster configuration.

The assemblies and rods are all uranium dioxide (UO_2) PWR fuel with zirconium or M5® alloy cladding. The pellets may contain small amounts of chrome oxides in the form of chromium oxide (Cr_2O_3) (see Section 6.7 of this SER). All contents are enriched natural uranium. The supporting safety analyses mention enriched reprocessed uranium as contents; however, this content is not allowed by the French certificate and was not evaluated for revalidation by the staff.

The contents are discussed in more detail in Section 6.0 of this SER. Table 1 provides a road map of the package's authorized content contents and the document that includes the evaluation of listed on the same row as the evaluation.

1.1.3 Drawings

The applicant provided drawings for both versions of the FCC-4 packaging in Appendices 1.4-1 and 1.4-2 of the application. The drawings define the materials of construction for the packaging to include the following:

1) stainless steel,

2) low allow steel,

3) carbon steel, balsa (used for impact limiters), and

4) a polymer resin (used for criticality control) materials.

The drawings identify the minimum yield and tensile strength values for the metal material, as well as the crush strength (and associated tolerance), assuming moisture content and density for the balsa wood (impact limiter material) used in the design.

Table 1. Contents description and location of supporting analysis for the FCC-4 package.

Content	Certificate Appendix	Foreign Certificate Criticality Reference
a maximum of 2 new PWR 17x17 XL, PWR 17x17 XLR, PWR 17x17 GAIA fuel assemblies, in Version 1 of the packaging, as described in Appendix 1 at version q;	1	Chapter 2.5-1 DOS-12-00057682-501 Rev. 0
or, a maximum of 2 new PWR 16x16 fuel assemblies in Version 2 of the packaging, as described in Appendix 2 at version q;	2	Chapter 2.5-1 DOS-12-00057682-501 Rev. 0
or, a maximum of 2 new PWR 18x18 fuel assemblies in Version 2 of the packaging, as described in Appendix 3 at version q;	3	Chapter 2.5-1 DOS-12-00057682-501 Rev. 0
or, a maximum of 2 boxes containing new PWR 17x17 XL or PWR 17x17 XLR non-assembled fuel rods, in Version 1 of the packaging, as described in Appendix 4 at version q;	4	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0
or, a maximum of 2 boxes containing new PWR 17x17 non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 5 at version q;	5	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0
or, a maximum of 2 boxes containing new PWR 15x15 non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 6 at version q;	6	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0
or, a maximum of 2 boxes containing new PWR 14x14 8-foot non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 7 at version q;	7	Chapter 2.5-2 DOS-12-00057382-502 Rev. 0
or, a maximum of 2 boxes containing new PWR 14x14 10-foot non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 8 at version q;	8	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0
Content No. 9		
or, a maximum of 2 boxes containing new PWR 16x16 non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 10 at version q;	10	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0
or, a maximum of 2 boxes containing new PWR 18x18 non-assembled fuel rods in Version 1 of the packaging, as described in Appendix 11 at version q;	11	chapter 2.5-2 DOS-12-00057682-502 Rev. 0
or, a maximum of 2 new 17x17 EPR fuel assemblies or a single new 17x17 EPR fuel assembly fitted with a RCCA and a smooth walled dummy or a mock-up of the EPR assembly, in Version 1 of the packaging, as described in Appendix 12 at version q; and	12	Chapter 2.5-1 DOS-12-00057682-501 Rev. 0
or, a maximum of 2 boxes containing new 17x17 EPR fuel rods in version 1 of the packaging, as described in Appendix 13 at version q.	13	Chapter 2.5-2 DOS-12-00057682-502 Rev. 0

The staff notes that the drawings do not identify the standards and grades used for materials selection; instead, the information is reflected in Table 1.4-1, Chapter 1.4 of the application. The staff further notes that the drawings do not identify the specific weld requirements of the packaging; instead, the drawings identify locations that reference a separate legend in Table 1.4-5, Chapter 1.4 of the application. The latter table lists the packaging welds defined as important for safety (location, type of weld, type of inspection).

The staff finds that the drawings and the application contain adequate information to describe the design and fabrication of the package, and the package meets the requirements in Paragraph 640 of IAEA SSR-6.

2.0 STRUCTURAL EVALUATION

2.1 General Considerations

The purpose of the structural evaluation is to verify that the structural performance of the package meets the regulatory requirements of IAEA SSR-6. The applicant performed structural analyses to demonstrate that the strength of the FCC-4 package meets the requirements specified in IAEA SSR-6. Specifically, the applicant addressed the FCC-4 package under:

1) Routine conditions of transport (RCT) with package tie-down and lifting,

2) Normal conditions of transport (NCT) under regulatory tests relating to normal conditions of transport, and

3) Accident conditions of transport (ACT).

A summary of the staff's structural evaluation is provided below.

2.2 Structural Analysis under Routine Conditions of Transport

2.2.1 Package Tie Down

The applicant considered two tie-down configurations to evaluate the performance of the FCC-4 package under RCT. The two tie-down configurations were:

a) <u>Mode No. 1</u> - a container with two straps placed in proximity to the inner hole of the lifting box, and

b) <u>Mode No. 4</u> - a container with four straps, where the first two straps were placed on the upper half-shell reinforcement in proximity to the inner reinforcement angle bar of the lifting box, and the other two straps were placed on the upper half-shell reinforcement in proximity to the central reinforcement angle bar.

The applicant first performed numerical calculations to determine the stresses on the container due to the maximum weight of the FCC-4 package with the two tie-down configurations, and then compared the calculated stresses with the allowable stress of the material. The numerical calculations were carried out using the SYSTUS finite element (FE) computer program. The results of the numerical calculations are provided in Appendix 2.1-2 of the SAR. The results show that the calculated stresses in the tie-down are much lower than the allowable stress of the material (Document No. NEEL-F 2008 DC 118E1, Revision B).

Additionally, the applicant used these calculated stresses as input data for a fatigue analysis, which calculated the cumulative stresses associated with transportation, handling, and stacking. The results of the calculations are provided in in Appendix 2.1-3 of the SAR. The results show that the calculated minimum life is 8321 years for the tie-down Mode No. 1 and 37 years for tie-down Mode No. 4 at the rate of 15 road transports per year for the FCC-4 packaging. The applicant stated that the calculated minimum life is consistent with the life of the FCC4 packaging (Document No. NTC-08-00135891EN, Revision 1).

The NRC staff reviewed the results of the analysis and validates that the applicant demonstrated that the FCC-4 package meets the regulatory requirement in Paragraph 638 of IAEA SSR-6.

The staff reviewed the analysis results and finds the applicant has adequately demonstrated that the FCC-4 package meets the regulatory requirement of IAEA SSR-6 for RCT with package tie-down and lifting.

2.2.2 Package Lifting

The applicant considered the following three possible lifting modes:

a) by means of the 4 lifting lugs welded on the upper shell for handling the loaded or empty package and the lid alone during opening operations,

b) by means of 2 sets of 4 lifting boxes welded on the lower shell for handling the loaded or empty package, and

c) by means of the fork-lift pockets provided under the lower shell.

The applicant performed numerical calculations to demonstrate the acceptability of lifting the FCC-4 package via the lifting lugs located on the upper and lower shell lifting points. The numerical calculations were carried out using the SYSTUS FE computer program and a fatigue analysis was performed to determine a cumulative usage factor. The results of the numerical calculations are provided in Appendix 2.1-4 of the SAR (Document No. DOS-13-00081778-104, Revision 0). The results show that the cumulative usage factor is significantly less than the allowable limit of 1.0 for the structure and welded joints.

The NRC staff reviewed the results of the analysis and finds that the applicant demonstrated that the FCC-4 package meets the regulatory requirement in Paragraph 608 of IAEA SSR-6.

2.3 Structural Analysis under Normal Conditions of Transport

2.3.1 Water Spray and Immersion Tests

The applicant noted that the water spray and immersion tests were not applicable to the FCC-4 package because of the following reasons:

a) the package was not designed for water tightness and the in-leakage of water into the packaging is considered in the criticality safety studies as discussed in the package's criticality analysis, and

b) the enclosure formed by the fuel rod cladding and the welded end plugs provide water tightness.

Therefore, the staff finds that the water spray and immersion tests required by IAEA SSR-6 to demonstrate leak-tightness are not applicable to the FCC-4 package. Further, the staff's review of the criticality safety studies is documented in Section 6.0 of this SER.

2.3.2 Stacking Test

The applicant noted in Chapter 2.1 of the SAR that an analysis was performed to demonstrate compliance with the requirements of IAEA SSR-6 for the stacking test. The analysis was accomplished by calculations considering a compressive force, which is equal to five times the maximum weight of the package. This force was applied uniformly on the four bearing points of the packages stacked for storage, where the container should be able to support the weight of the five other packages without exceeding the allowable stress of the material. The applicant showed in Appendix 2.1-5 of the SAR (Document No. EVED DC 02 0144 E1, Revision B) that the container at the bottom of the stack is able to support the weight of other five packages without exceeding the allowable stress in the shell of the FCC-4 package.

The NRC staff reviewed the results of the analysis and finds that the applicant demonstrated that the FCC-4 package has adequate strength under NCT to meet the stacking requirement in Paragraph 723 of IAEA SSR-6.

2.3.3 Perforation (Penetration) Test

The applicant stated that an analysis was performed to demonstrate compliance with the requirements of IAEA SSR-6 for the penetration test. The applicant calculated the energy (capacity) necessary to penetrate the FCC-4 package and compared it with the energy (demand) exerted by the bar based on the drop test. The applicant demonstrated that the energy (capacity) required to perforate the FCC-4 package is larger than the energy (demand) from the dropped bar.

The NRC staff reviewed the analysis results and validates that the applicant demonstrated that the FCC-4 package has adequate strength under NCT to meet the penetration requirement in Paragraph 724 of IAEA SSR-6.

2.3.4 Free Drop Test

The applicant indicated in Chapter 2.1 of the SAR that a 0.9-m free drop physical model test under NCT was not performed because the performance of the package under NCT was bounded by a more severe 9-m free drop physical model test under ACT. Based on the results of the 9-m free drop test, the applicant reported that there was no significant damage on the package. Specifically, the applicant reported the following:

a) there was no dispersal of the radioactive content,

b) the relative positions of the assemblies did not change, and

c) deformation of the package remained localized.

The NRC staff reviewed the evaluations and test results provided in Chapter 2.1 of the SAR, and found that the applicant demonstrated that the FCC-4 package has adequate strength under NCT to meet the regulatory requirement in Paragraph 722 of IAEA SSR-6.

2.4 Structural Analysis under Accident Conditions of Transport

The applicant stated in Chapters 1.5 and 2.1 of the SAR that two model drop tests were carried out on representative full-scale prototypes of the packaging to demonstrate compliance of the FCC-4 packaging with the IAEA SSR-6. The two model drop tests were:

1) The first prototype model referred to as "Prototype 2", corresponding to the FCC-4 packaging, was used for the lateral drop test, and

2) The second prototype model referred to as "Prototype 1", corresponded to the FCC-3 packaging, which has a similar design to the FCC-4 packaging, was used for the vertical drop test.

The applicant selected the free-fall drop configurations that maximized the possible damage to the following package components:

1) the closing system on the internal fittings,

2) the bolted connectors on the top and bottom shells, and

3) the shell shock absorbers.

The applicant selected the following drop sequences for Prototype 2:

1) **1st drop** - a free-fall drop from 1-m onto a bar, near a door bolt on the internal fittings,

2) **2nd drop** - a free-fall drop from 1-m onto a bar, at the same impact point, but from a different drop angle to the 1st drop, and

3) **3rd drop** - a flat drop with ‘whiplash’ effect. Detailed information regarding the drop test program for Prototype 2 is provided in Appendices 2.1.7 and 2.1.8 of the SAR (Document Nos. TFXE DC 2104 E0, Revision E and TFX DC 2108 E0, Revision B).

The applicant reported the results of the model test for Prototype 2 in Appendix 2.1-10 of the SAR (Document No. TFXE DC 2132 E0, Revision E) and provided a summary of the results for the following safety components in Table 7.1 of the SAR:

1) top and bottom plate connections with the frame-doors unit,

2) doors - frame connection,

3) half-shell connector bolts, and

4) axial shock absorbers.

The applicant selected the following drop sequences for Prototype 1:

1) **1st drop** - a free-fall drop from 1-m onto a bar, against an edge of the door of the internal fittings,

2) **2nd drop** - a free-fall drop from 1-m onto a bar, against an upper face of the door of the internal fittings,

3) **3rd drop** - a free-fall drop from 9-m, in a vertical position along the centerline of the top end of the package, and

4) **4th drop** - a free-fall drop from 9-m with a 'whiplash' effect.

The applicant reported the results of the model test for Prototype 1 in Appendix 2.1-9 of the SAR (Document No. TFX/DC/2087, Revision A) and provided an extended analysis to the FCC-4 packaging in Appendix 2.1-7 of the SAR (Document No. TFXE DC 2104 E0, Revision E) supplemented by the information provided in Appendix 2.1-13 of the SAR (Document No. DOS-13-00081778-113, Revision 02).

Based on the results of the regulatory drop tests for ACT, the applicant concluded that the enclosure of the prototype package suffered localized deformations, but did not impact the safety of the package. Moreover, there was no dispersal of the radioactive contents after completing the regulatory tests for ACT.

The NRC staff reviewed the regulatory drop test results, including the supporting evaluations, and validates that the applicant demonstrated that the FCC-4 package has adequate strength to withstand ACT and meets the regulatory requirement in Paragraph 727 of IAEA SSR-6.

The FCC-4 package is also designed to transport non-assembled fuel rods in rod boxes. The applicant assessed the mechanical strength of the fuel rod boxes in Appendix 2.1-12, "FCC Packaging Rod Boxes - Proof of The Mechanical Strength of Box Equipment," of the SAR (Document No. DOS-13-00081778-112, Revision 01). The applicant analyzed the mechanical behavior of the rod boxes transported in FCC-4 package under ACT. The applicant's analysis focused on:

1) the radial spacers,

2) the compensation spacers,

3) the axial spacers,

4) the support plates between the radial spacer and the rod bundle, and

5) the end support plates for the fuel rods on the axial spacer side.

The applicant performed the analysis using the LS-DYNA FE computer program to determine the stresses and strains of the axial spacer and the fixed end support plate under ACT. Two models for the axial spacer and fixed end support were considered from the drawings in Appendix 1.3-2 (Document No. DOS-13-00081778-032, Revision 01):

1) **15x15** - 12 feet (ft) configuration (Drawing Nos. 47-0-01-99-01-02 and 47-0-01-99-01-16, and

2) **17x17** - 14 ft EPRTM configuration (Drawings Nos. PLA-16-00179264-201 and PLA-16-00179264-202).

The applicant provided the results of the analysis in Section 5.8 of Appendix 2.1-12 of the SAR. Figures 5 and 6 in Appendix 2.1-12 of the SAR show the deformed shapes and calculated plastic strains of the two models, respectively. Based on the results of the analysis, the applicant concluded that:

1) the deformed shape in the fixed end support plate (Figure 5 in Appendix 2.1-12 of the SAR) demonstrates that the gaps between the fixed end plate and the structure of the rod box are maintained, which guarantees that the rods are secured in the rod box, and

2) the calculated plastic strains in the axial spacer and the fixed end plate (Figure 6 in Appendix 2.1-12 of the SAR) are less than the true rupture elongation of the rod box, therefore, the rod box has adequate strength under ACT.

The NRC staff reviewed the analysis for the rod boxes used to transport non-assembled fuel rods in the FCC-4 package and validates that the applicant demonstrated that these rod boxes have adequate strength to withstand ACT and meets the regulatory requirement in Paragraph 727 of IAEA SSR-6.

2.5 Evaluation Findings

Based on the review of the statements and representations contained in the application, the staff concludes that the structural evaluations and the regulatory drop test program have been adequately described, and the FCC-4 package has adequate structural design to meet the requirements of IAEA SSR-6.

3.0 THERMAL EVALUATION

The purpose of the thermal review is to revalidate that the FCC-4 package (abbreviated as FCC-4), loaded with fresh fuel assemblies and fuel rods for PWR, including GAIA fuel, satisfies the thermal safety requirements of the IAEA SSR-6 (Ref. 3.8.1).

3.1 Description of Thermal Design

The FCC-4 package is categorized as a Type A fissile (AF) package to meet the IAEA requirements for normal transport conditions and accidental fire conditions, as described in the thermal requirements of IAEA SSR-6.

The applicant noted in Enclosure 1 to E-55607 of the application the following:

1) the maximum activity of the content shipped in FCC-4 is less than 1.0 A_2 and the heat load in FCC-4 is limited to 0.6 watts (W),

2) the materials being shipped are not in special form and are not permitted to have a greater hydrogen content than water as part of the packaging,

3) the package model is made in two versions of cavity cross-sections so the internal geometry can be adapted to the geometry of the contents, and

4) the FCC-4 is similar to the FCC-3 package (Ref. 3.8.2) but is a longer version to ship 14-foot-long PWR fuel assemblies.

The staff reviewed description and design drawings for the FCC-4 in Enclosure 1 to E-55607 and confirmed the information noted above by the applicant. Based on a review of the FCC-3 package (abbreviated as FCC-3), which was previously reviewed and recommended for revalidation to DOT by the NRC, the staff also confirmed that item No. (4), which mentions that FCC-4, similar to FCC-3 in package design and configuration, is a longer version than the FCC-3 (see Table 1 in Appendix 2.2-1, Document No.DOS-13-00081778-201 for reference). Therefore, the staff determined that the assumptions, parameters, and fire scenarios used in the thermal analyses developed for the FCC-3 package are also applicable for the thermal analyses of the FCC-4 package.

The applicant stated in Appendix 1.4 of the SAR (Document No.DOS-13-00081778-040), the two half-shells, the internal equipment system and the resin contained in the doors and frame provide fire protection for the FCC-4. Like the resin in the FCC-3, the resin in the FCC-4 protects the enclosure from the effects of the temperatures prevailing under the accident conditions of transport (ACT) fire. The applicant noted that similarity in designs between the FCC-3 and the FCC-4 allows partial use of the FCC-3 assumptions, evaluation, and test results in the FCC-4 SAR.

The staff referred to the documents related to the revalidation of the FCC-3 and confirmed the following:

1) the FCC-4 is identical to the FCC-3 in design and differs from the FCC-3 only in its geometrical characteristics (FCC-4 is a longer version), and

2) the resin in the FCC-4 will perform the same function, to limit the temperature rise in the package cavity, as it does in the FCC-3.

Based on these findings, the staff finds that the applicant demonstrated compliance of the FCC-4 with IAEA thermal requirements by the using applicable FCC-3 thermal evaluations, which were previously reviewed by the NRC.

3.2 Material Properties and Component Specifications

The applicant described the principal materials of construction for the FCC-4 packaging components in Table 1.4-1 of Appendix 1.4 of the SAR and the thermal properties for these materials, including steel and resin, in Tables 1 and 2 of Appendix 2.2-1 of the SAR. For the resin, the properties retained in Table 1 of Appendix 2.2-1 of the SAR are those corresponding to a temperature of 160 °C.

The staff reviewed Appendix 1.4 and Appendix 2.2-1 of the SAR and finds the use of these thermal properties in the thermal evaluations of the FCC-4 acceptable. The staff previously reviewed and recommended the revalidation of the FCC-3, which used these thermal properties.

3.3 General Considerations

The applicant described the thermal evaluations for the FCC-4 in Appendix 2.2-1 of the SAR, which included the following:

1) evaluation of the thermal material properties,

2) descriptions of the thermal model,

3) assumptions used in the thermal analyses, and

4) calculations provided by the thermal models for normal transport conditions and accidental fire conditions.

The staff reviewed Appendix 2.2-1 of the SAR and finds that the thermal model setup, including the thermal properties of packaging components and the assumptions, initial conditions, and boundary conditions used in the thermal analyses are acceptable for the revalidation for the FCC-4 loaded with the contents described in Section 1.1.2 of this SER. The staff also confirmed that, with the negligible decay heat and in the shade, the surface temperature of the FCC-4 would not exceed 50 °C and would meet Paragraph 654 of the IAEA SSR-6.

3.4 Thermal Evaluation under Normal Conditions of Transport

The applicant performed two-dimensional and three-dimensional analyses using fuel rods of 17x17 fuel assemblies to analyze the FCC-4 thermal design under normal transport conditions (NCT). The applicant described the thermal evaluation for the FCC-4 in Appendix 2.2-1 and Appendix 2.2-6 of the SAR (Document No. DOS-13-00081778-206), assuming the thermal conditions of ambient temperature of 38 °C and solar flux of 400 W/m^2 for thermal evaluation of NCT.

The staff reviewed the descriptions and the model calculations of the FCC-4. The parameters and physical phenomena used in the applicant's evaluation of FCC-4 under NCT are summarized below:

1) **Properties of the Materials.** The thermal properties of the materials are provided in SAR Appendix 2.2-1, including zirconium alloy fuel rods containing the fissile material UO_2, resin, stainless steel, helium and air. The physical properties of the materials (thermal conductivity, air viscosity, density) are varying with temperature.

2) **Helium Layer.** The helium layer separating the fuel pellets from their cladding is represented by a contact resistance of $4x10^{-4} \sim 5x10^{-4}$ m^2-K/W, equivalent to the pellet/cladding interface. The radiation heat transfer in the helium layer is very small relative to the heat conduction and is therefore ignored.

3) **Heat Transfer.** The clamshell is cooled by natural convection and thermal radiation under the ambient air temperature of 38 °C of normal transport conditions.

The staff reviewed the applicant's thermal evaluation for NCT as described in Appendix 2.2-1 of the SAR and confirmed that the maximum temperatures of 78 °C for the package shell and

66 °C for the resin would remain below their corresponding operating limits. Therefore, the FCC-4 meets the thermal requirements of Paragraph 679 of IAEA SSR-6.

3.5 Thermal Evaluation under Accident Conditions of Transport

3.5.1 Accident Conditions of Transport – Fire Scenario

The thermal test, as defined by the regulations in IAEA SSR-6, follows the mechanical tests. Therefore, the full set of tests is the most damaging for the package. Subsequent to the mechanical tests, the fuel cladding had not ruptured for the FCC-4. However, the applicant used the following features on a full-scale package after mechanical tests:

a) the reduction of the clearance between the claddings of the assemblies,

b) the opening or closing of a few clearances (openings) between the doors and the frame, and

c) the small perforation of the external `clamshell which could lead to the ingress of the flames as far as the internals.

The applicant performed two-dimensional and three-dimensional analyses using fuel rods of 17x17 assemblies to analyze the FCC-4 thermal design under ACT, in compliance with the requirements of IAEA SSR-6. The applicant described the ACT thermal evaluations for FCC-4 in Appendix 2.2-1 of the SAR, as follows:

a) flames at a temperature of 800 °C with an emissivity of 0.9 over all the exposed outer walls of the model,

b) an absorption coefficient of 0.8 at the exposed outer walls, and

c) a convective exchange coefficient of 10 W per square meter per degree Kelvin (W/m^2-K) between the flames and the walls.

The staff reviewed the descriptions and the model calculations of FCC-4. The staff finds that the parameters and physical phenomena shown below and used in the applicant's ACT fire evaluation of FCC-4 are acceptable because of their conservatism:

a) The initial temperatures for air and all components are conservatively set as 78 °C and the solar insolation conditions are in compliance with Table XI of IAEA article 728.

b) The clamshell is assumed to be lost after the mechanical tests. The entire outer skin of the internals is heated by forced convection and thermal radiation under the fire temperature of 800 °C (1475 °F) during the 30-minute fire phase,

c) The clamshell is assumed to be present during the post-fire cooldown to minimize the heat losses. The clamshell is cooled by natural convection and the thermal radiation to the ambient air temperature of 38 °C, during the post-fire cooldown,

d) The polyester resin, contained in the door and the frame, is considered as an inert, stable material, the thermal properties of which remain unchanged. The increase in the insulation capability of the resin, caused by the surface calcinations effects and the endothermic reaction, is conservatively not considered, and

e) The cladding outer face emissivity is modeled as 0.6 by accounting for the surface oxidation during the fire test. The emissivity of the inner walls of the door and the frame is modeled to be 0.6 by taking the surface oxidation into account.

The applicant also indicated in Appendix 2.2-1 of the SAR that if resin were deteriorated by fire, the consumed resin would be replaced by air and, the thermal conductivity of air being far lower than that of resin, the package would then be better insulated under the fire.

The staff finds the following thermal calculations acceptable:

a) the resin, well contained within the case made of stainless steel, has a temperature distribution ranging from 129.4 °C to 795.4 °C in the fire test. The use of resin properties at 160 °C is acceptable from the thermal perspective, and

b) the maximum and the mean temperatures of the most stressed cladding are 642.8 °C and 624.2 °C, which are below the design temperature limit.

As described in Appendix 2.2-3 of the SAR, the applicant evaluated the mechanical behavior of a fuel rod in the FCC-4 packaging, subjected to the ACT fire, and studied the risk of bursting of a cladding tube due to creep under the cumulative effect of internal temperature and pressure. The applicant calculated a cumulated elongation that is below the ultimate elongation (rupture) limit for the M5® cladding below the temperature for ACT; and a cumulated elongation that is below the ultimate elongation (rupture) limit for the Zircalloy-4 cladding below the temperature for ACT.

The staff reviewed the report titled “GAIA Fuel Assembly” (Ref. 3.8.3) and noted that the GAIA fuel design incorporates three intermediate GAIA mixers (IGMs) based on AFA-3G mid span mixing grid design and IGMs are fabricated with the M5® alloy. The staff finds that the GAIA fuel has similar elongations as the M5® cladding reported above. Therefore, the staff confirmed that the cumulated elongation analysis for the M5® cladding is also applicable to the GAIA fuel, and a cumulated elongation below the ultimate elongation (rupture) limit set for the GAIA fuel, is acceptable.

3.5.2 Impact of Glycerin Combustion

The applicant noted in Appendix 2.2-5 of the SAR (Document No. DOS-13-00081778) that glycerin is a hydrogen-containing material, with an auto-ignition temperature of 400 °C, which is less than the maximum temperature of 642.8 °C reached by the fuel rod under the regulatory fire test conditions. During manufacturing, the glycerin, which is soluble in water, is applied to each of the rods prior to them being inserted into the assembly grids. After rinsing the assemblies with water, the residual glycerin is therefore distributed among all rods in the array.

The applicant performed a thermal analysis of the risks associated with the combustion of the 5 grams (g) of glycerin during the ACT fire, as described in Appendix 2.2-5 of the SAR. The thermal analysis was carried out considering the combined phenomena of (a) combustion of the layer of glycerin and (b) overall energy input from the heated cavity gases. The bounding PWR 17x17 fuel assembly was used to cover all other types of fuel assemblies allowed for FCC-4 and the conclusions of the analysis on the impact of glycerin on PWR 17x17 assemblies are applicable to other types of fuel assembly. The applicant calculated temperature increases of 42.3 ºC by combustion of the layer of glycerin and of 26.3 °C by overall energy input of combustion.

The staff reviewed the calculations shown in Appendix 2.2-5 of the SAR and performed the confirmatory calculations. The staff finds the assumptions, equations, and calculations shown in Appendix 2.2-5 of the SAR acceptable for analysis of the glycerin combustion in the ACT fire. The staff confirmed that (a) the potential combustion of 5 g of glycerin causes a small transient increase of the temperature and (b) the impact upon the creep behavior of the rods is negligible and does not affect thermal analysis.

3.6 Evaluation Findings

Based on the review of the statements and representations contained in the application, the staff finds that the thermal design of FCC-4, loaded with the contents described in Section 1.1.2 of this SER, meets the requirements for thermal performance outlined in IAEA SSR-6, (Ref. 3.8.1) for the transportation of fuel assemblies. The staff has reasonable assurance that the package will perform as designed for shipments made in accordance with the French Approval Certificate Number F/348/AF-96, Revision Fq.

3.7 References

3.8.1 International Atomic Energy Agency (IAEA) SSR-6, “Regulations for the Safe Transport of Radioactive Material,” 2012 Edition.

3.8.2 Letter from John B. McKirgan, U.S. Nuclear Regulatory Commission, to Richard W. Boyle, U.S. Department of Transportation (DOT), “Revalidation Recommendation for the French Certificate of Approval No. F/347/AF-96, Revision Fs, Model No. FCC-3 Package (Docket No. 71-3083),” February 12, 2021, ADAMS Package Accession No. ML21042A037.

3.8.3 AREVA Inc., GAIA Fuel Assembly, AREVA Inc., ANP:U-395-V2-14-ENG, http://www.framatome.com/us_platform/liblocal/docs/Catalog/PWR/ANP_U_395_V2_14_ENG_GAIA.pdf

4.0 CONTAINMENT EVALUATION

The purpose of the containment review is to verify that the FCC-4 package design satisfies the requirements for the evaluation of the containment boundary as required in the IAEA SSR-6. The staff reviewed the application and confirmed that the containment system of the FCC-4 package is well described for revalidation. A summary of the staff’s review is provided below.

4.1 Description of the Containment System

Section 2.4, “Containment,” of Chapter 1.5, “Package Performance Characteristics,” of the application describes that the fuel rod cladding and the zirconium alloy welded end plugs form the containment for the fissile material in the FCC-4 package. Based on the staff’s review of Section 2.4 of Chapter 1.5 of the application, the staff finds this description to be acceptable.

The application further describes the fuel rod cladding performance during NCT and ACT in the following Sections of the SAR:

1) **Section 4.3.3**, “Representativity of the prototype 2 contents,” of Chapter 2.1, “Structural analysis,” of the application describes that the pre-oxidation surface treatment of the cladding has no impact on the mechanical behavior of the assembly in normal and accident conditions of transport.

2) **Section 3.2**, “Normal conditions of transport,” of Chapter 1.5 of the application and Section 4.2.4, “Conclusion,” of Chapter 2.1 of the application describe that there is no dispersal of the radioactive contents on completion of the regulatory tests for normal conditions of transport.

3) **Section 3.3**, “Accident conditions of transport,” of Chapter 1.5 of the application describes that the results of the satisfactory behavior of the Zircaloy-4 rods in the fuel assembly prototypes during drops are extended to the zirconium M5® alloy rods.

4) **Section 4.3.5**, “Drop test results for prototype 2,” of Chapter 2.1 of the application summarizes that the cladding remains intact and leak tight, and there is no dispersal of the radioactive contents after the tests representative of accident drop conditions.

5) **Section 10**, “Conclusion,” of Chapter 2.2, “Thermal,” of the application concludes that for the Zircaloy-4 and M5® cladding there is no dispersal of material after completion of the tests representative of accident conditions.

6) **Section 5**, “Conclusion,” of Appendix 2.2-3, “M5® or Zircaloy-4 fuel rod behaviour during IAEA thermal test in FCC packaging,” of the application concludes that the risk of ballooning and bursting of the M5® or Zircaloy-4 cladding under the effect of creep can be ruled out.

7) **Section 5**, “Conclusion,” of Appendix 2.2-5, “Analysis on the impact of glycerine on the thermal safety analysis,” of the application concludes that the presence of a residual 5 g of glycerin does not bring into question the mechanical behavior of the M5® or Zircaloy-4 fuel rod cladding.

Based on the staff’s review of the statements described above, the staff finds that the cladding will maintain containment during NCT as it meets the provisions of Paragraph 648 of IAEA SSR-6.

4.2 General Considerations

The approval certificate for the FCC-4 packaging describes that the physical state of each of the contents is fuel rod assemblies containing sintered pellets in a zirconium alloy cladding that is possibly pre-oxidized and meets the criteria (e.g., dimensions, weight, material, density, enrichment, and mass ratio) provided in Paragraph 1.1 for each of the contents in Appendices 1 through 13 of the approval certificate. The staff notes that there is no Appendix 9 of the approval certificate for the FCC-4 packaging; therefore, the staff cannot approve an Appendix 9 of the approval certificate for the FCC-4 packaging. The applicant informed the NRC that there is no intent to transport enriched reprocessed fuel in the FCC-4 packaging. The staff finds that this is consistent with the contents in Appendices 1 through 13 (excluding Appendix 9, which was not provided) of the approval certificate for the FCC-4 packaging which describes the pellets as enriched natural uranium (ENU).

For each of the contents, the approval certificate for the FCC-4 packaging describes that the maximum activity level per packaging is less than 1 A_2; therefore, the staff finds this to be acceptable for a Type A(F) package as it meets the provisions of Paragraph 429 of IAEA SSR-6. The material classification for each of the contents in Appendices 2 through 13 of the approval certificate for the FCC-4 packaging is an activity that is less than 1 A_2. However, the material classification was not described for the contents in Appendix 1 of the approval certificate for the FCC-4 packaging. As noted above, it is specified in Appendix 1 of the approval certificate for the FCC-4 packaging that the maximum activity of the content per packaging is less than 1 A_2; therefore, the staff finds this to be acceptable for a Type A(F) package as it meets the provisions of Paragraph 429 of IAEA SSR-6.

Section 8.1, “Article 429,” of Chapter 1.6, “Compliance with regulatory requirements,” of the application describes that the A_2 value of ENU under 20% is unlimited. Therefore, the staff finds it to be acceptable that activity of an FCC-4 packaging transporting enriched natural uranium is less than 1 A_2. Section 8.2, “Article 430,” of Chapter 1.6 of the application describes that the A_2 value for ENU that is under 20% enriched is unlimited. Therefore, the staff accepts that the activity is equal to 0 A_2.

This is also shown in Table 1.3-5, “Maximum activity of ENU assemblies,” of the application which shows that the total activity for the types of fuel assemblies and rods is equal to 0 A_2, which is less than 1 A_2. Therefore, the staff finds that the assemblies (including GAIA fuel) and rods can be transported in a Type A(F) package as it meets the provisions of Paragraph 429 of IAEA SSR-6. Section 8.11, “Article 644,” of Chapter 1.6 of the application describes that the containment system consists of the fuel rod cladding and the rods only contain UO_2 pellets and inert gas. The applicant describes that rods do not contain liquids or materials which may generate gas by chemical reaction or radiolysis; therefore, the staff finds this to be acceptable as it meets the provisions of Paragraph 644 of IAEA SSR-6.

4.3 Evaluation Findings

Based on review of the statements and representations in the FCC-4 package application, the staff concludes that the applicant adequately described and evaluated the containment system for the FCC-4 package and that the package meets the containment requirements of the IAEA SSR-6. The staff recommends revalidation of the French Certificate of Approval No. F/348/AF-96, Revision Fq.

5.0 MATERIALS EVALUATION

The FCC-4 shipping package is designed to transport unirradiated commercial PWR fuel assemblies (14-foot long, AFA3G 17x17 design)) or individual fuel rods. The staff conducted a revalidation review for the French Competent Authority Certificate Number F/348/AF-96, Revision Fq, Model No. FCC-4, Type A(F) package, per the requirements in Section VI of the IAEA SSR-6. The pertinent IAEA SSR-6 requirements are listed below:

1) **501.** Before a packaging is first used to transport radioactive material, it shall be confirmed that it has been manufactured in conformity with the design specifications to ensure compliance with the relevant provisions of these Regulations and any applicable certificate of approval.

2) **502.** Before each shipment of any package, it shall be ensured that the package contains neither: (a) radionuclides different from those specified for the package design; nor (b) contents in a form, or physical or chemical state, different from those specified for the package design.

3) **613.** The package shall be capable of withstanding the effects of any acceleration, vibration or vibration resonance that may arise under routine conditions of transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole. In particular, nuts, bolts, and other securing devices shall be so designed as to prevent them from becoming loose or being released unintentionally, even after repeated use.

4) **614.** The materials of the packaging and any components or structures shall be physically and chemically compatible with each other and with the radioactive contents. Account shall be taken of their behavior under irradiation.

5) **639.** The design of the package shall take into account temperatures ranging from -40 °C to +70 °C for the components of the packaging. Attention shall be given to freezing temperatures for liquids and to the potential degradation of packaging materials within the given temperature range.

6) **640.** The design and manufacturing techniques shall be in accordance with national or international standards, or other requirements, acceptable to the competent authority.

7) **644.** The design of any component of the containment system shall take into account, where applicable, the radiolytic decomposition of liquids and other vulnerable materials and the generation of gas by chemical reaction and radiolysis.

8) **679.** The package shall be designed for an ambient temperature range of -40 °C to +38 °C unless the competent authority specifies otherwise in the certificate of approval for the package design.

5.1 Materials

The staff reviewed the application to determine the adequacy of the materials of construction of the package and the associated technical discussions in justification of compliance with the requirements in IAEA SSR-6.

The application (DOS-18-016472, Revision 1) defined two versions for the FCC-4 packaging, which vary depending on the length of the fuel rods or full fuel assemblies. The first design version applies to fuel assemblies composed of a 17x17 square fuel rod array (XL, XLR, EPR, and GAIA) or for non-assembled fuel rods grouped in channels. The second design version applies to fuel assemblies composed of a 16x16 or 18x18 square fuel rod array. The rods in either of these versions can be made from ENU or from enriched reprocessed uranium. The full fuel assemblies have a length of 14 ft, while the non-assembled rods have a length from 8 to 14 ft. The characteristics of the fuel assemblies are defined in the Certificate of Compliance from the French Competent Authority (Appendix 1, of the French Certificate of Competent Authority No. F/348/AF-96, Revision Fq).

5.2 Materials Standards

Chapter 1.4 of the application states that the packaging will be fabricated per specification values defined in Table 1.4.-1, Chapter 1.4 of the application. These values are defined per the latest revision of the cited Association Française de Normalisation (AFNOR) standards. The application also clarified that AFNOR grades defined in Table 1.4.-1, Chapter 1.4 of the application, may be replaced by grades with at least equivalent mechanical properties.

The low-alloy steel materials and associated grades are defined per AFNOR standards NF EN 10025-3 and NF EN 10025-4. The carbon steel material (and associated grade) is defined per NF EN A 36-601. The stainless-steel material is defined per AFNOR standards NF EN 10008-3, NF EN 10028-7, and NF A 35-557. The staff confirmed the mechanical properties defined in the applicant per those in the AFNOR standards.

The low alloy steel bolting used in the packaging are defined per AFNOR standards NF-EN-898.1 and NF-EN-898.2. The general tolerances for bolting materials are defined per AFNOR standards NF-E-86-050 and NF-E-02-350. In addition, package fabrication requires minimum guaranteed values for bolt toughness at -40 °C, in accordance with AFNOR standard NF EN 10113-1, and consistent with the requirements in Paragraphs 639 and 679 of IAEA SSR-6.

Based on the evaluation above, the staff finds that the design of the packaging materials are in accordance with applicable standards, and the package meets the requirements in Paragraph 640 of IAEA SSR-6.

5.3 Weld Design and Inspection

Table 1.4-5, Chapter 1.4 of the application lists the packaging welds defined as important for safety (location, type of weld, type of inspection). Weld inspection activities include visual inspection and dye penetrant testing, which is to be completed per CODAP (Code Français de Construction des Appareils à Pression; French construction Code for pressurized vessels), Section I or another equivalent code. The applicant provided English translations of the Translation of CODAP 2005 Code defining the acceptance criteria for visual inspection (Division 1, Part I, Controls and Inspection, Appendix I1.A1) and dye penetrant (Appendix I1. A2).

Chapter 1.7 of the application requires that these inspection activities be performed by qualified personnel, consistent with the applicant's Quality Assurance Program. The staff considers the methods and acceptance criteria for packaging welds to be acceptable.

Based on the evaluation above, the staff finds that the manufacture of the package is in accordance with applicable standards, and the package meets the requirements in Paragraphs 640 and 648 of IAEA SSR-6.

5.4 Mechanical Properties

5.4.1 Carbon Steels

Chapter 1.4 of the application states that the mechanical properties of low-alloy steel materials (including bolting), carbon steel and stainless-steel materials used in the packaging will be fabricated per the specification values in the latest revision of the AFNOR standards listed in Table 1.4-1 of Chapter 1.4 of the application. The staff confirmed the temperature-dependent mechanical properties of the material grades in the pertinent AFNOR standards with those defined in Chapter 1.4 of the application.

5.4.2 Low-Alloy and Carbon Steels

Appendix 1.4-3 of the application documents the toughness tests that were carried out on the low-alloy steel material used in the package shells. The staff confirmed that the test results demonstrate that the minimum toughness values conform with those defined in Chapter 1.4 of the application and the referenced AFNOR standard. Chapter 1.4 of the application also states that the procurement requirements for low-alloy steel bolting will provide for minimum guaranteed toughness values at -40 °C (-40 °F) by means of specific product tests, in accordance with AFNOR standard NF EN 10113-1. The staff confirmed that the minimum guaranteed toughness values defined in Chapter 1.4 are consistent with AFNOR standard NF EN 10113-1.

5.4.3 Stainless Steels

Chapter 1.4 of the application states that the stainless-steel materials used in the package design do not exhibit brittle fracture at low service temperatures consistent with the requirements in Paragraphs 639 and 679 of IAEA SSR-6. The staff agrees with this conclusion.

5.4.4 Wood (Impact Limiter)

Chapter 1.4 of the application states that the axial shock absorbers are fabricated of a stainless-steel enclosure containing balsa wood. Table 1.4-1 of the applications specifies the density, moisture content, and crush strength of the impact limiter wood material. Table 1.4-1 of the application also defines that the wood grain direction must adhere to the specification in the design drawing. The staff reviewed these material properties used in the applicant's mechanical calculations and confirmed that the properties are either conservative or consistent with those in the technical literature.

Based on the evaluations above, the staff finds that the mechanical properties of materials used in the structural analysis are consistent with applicable standards and values in the technical literature, and the package meets the requirements in Paragraphs 616, 639, 640, 648 and 679 of IAEA SSR-6.

5.5 Criticality Control Materials

The doors and frame of the packaging encase a neutron-absorbing polymer resin used to maintain subcriticality. Chapter 1.4 of the application states that a qualified process (i.e., controlled injection procedure) is used to fill the packaging cavities with the resin. Table 1.4-1 of Chapter 1.4 of the application defined the properties of the resin material used in the packaging. More specifically, the applicant defined property specifications for composition (hydrogen and boron content), density, thermal conductivity and heat capacity, which are not specific to a French standard.

Chapter 2.5 of the application justified the suitability of the resin specifications in Table 1.4-1 of Chapter 1.4 of the application to demonstrate compliance with the thermal test per SSR-6, Requirement 728. More specifically, Appendix 2.2-4 of the application described test results to characterize the loss of hydrogen and boron under conservative fire test conditions. The tests demonstrated that the resin remained largely intact when exposed to a direct flame, forming a surface char layer that protects the material beneath. Further, the tests demonstrate that the resin retains adequate thickness to ensure the sub-criticality of the package array during a fire accident. The applicant used the test data to adjust the neutron absorption performance of the resin in the criticality analysis to account for potential degradation during a fire accident.

The staff reviewed the thermal test results and verified that that the applicant applied appropriate penalties to the neutron absorption performance of the resin in the criticality analysis to account for material changes in a fire accident. The staff also concluded that the resin is expected to remain intact and not relocate under accident conditions of transport.

Based on the evaluations above, the staff finds that the properties of the neutron absorbing materials used in the criticality analysis are acceptable, and the package meets the requirements in Paragraphs 673 and 728 of IAEA SSR-6.

5.6 Corrosion and Chemical Reactions

Chapter 1.6 of the application states that the constituent materials of the packaging and all internal components or structures were chosen to be physically and chemically compatible with each other and with the intended radioactive contents of the package. The staff reviewed the packaging materials and service environments to verify that adverse reactions will not prevent the package from performing its safety functions.

Chapter 1.6 of the application further states that the internal equipment of the package is made exclusively of stainless steel, which protects these parts against corrosion risks. The staff reviewed the materials of the packaging and concluded that the stainless steel is compatible with the air environments to which the surfaces of the packaging are exposed. Based on this evaluation, the staff finds that the applicant adequately considered the corrosion resistance of the internal equipment.

For low-alloy and carbon steel components, Table 1.4-1 of the application indicates that a "corrosion inhibitor + paint" coating is applied to the entirety of the upper and lower shells (inside and out). With respect to these components, Chapter 1.7 of the application specifies that, before each shipment, there is a provision to check for the absence of flaking paintwork on the uninterrupted sections of the packaging to detect and remedy incipient corrosion. Further, Chapter 1.8 of the application details a periodic maintenance program, which specifies that the FCC-4 packaging is required to undergo an inspection and maintenance operation that includes a check for paint defects and rework deficient areas on the internal and external surfaces of the shells if missing paintwork is identified.

Regarding the aluminum packing shims, in Appendix 1.3-1 of the application, the applicant notes that shims are used when the FCC-4 packaging transports unassembled fuel rods in the accompanying fuel rod boxes. The staff reviewed the design drawings in Appendix 1.3-2 of the application and applicable sections of the application to evaluate the effects, if any, of intimate contact between aluminum packing shims and the materials in the FCC-4 package. The staff finds that galvanic corrosion between the aluminum packing shims and the stainless steel components of the FCC-4 package is not expected because water is effectively sealed off under NCT. Visual inspections are to be performed of the payload cavity prior to loading and following off-loading, which provide reasonable assurance that any corrosion will be detected in a timely manner.

Regarding potential radiolytic gas generation, the staff reviewed the content of the fuel rods and the environmental conditions. The authorized content for the FCC-4 is Type A(F), including ^{232}U, ^{234}U, ^{235}, U ^{236}U, and gadolinium oxide (Gd_2O_3), which are dominantly alpha emitters. Some daughter products decay via beta emission, but they do not emit significant gamma rays. Inside unirradiated fuel rods, there is no water and, therefore, there is no radiolytic gas generation. Very little/low strength gamma may be present outside cladding where very small amounts of water are present. Therefore, the staff finds that gas generation from radiolysis is not an issue with unirradiated fuel.

Based on the evaluations 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 Paragraphs 614 and 644 of IAEA SSR-6.

5.7 Content Integrity: Fresh (Unirradiated) Fuel Cladding

Chapter 1.5 of the application states that drop tests were carried out on full-scale prototypes with dummy fuel assemblies with rods cladded with Zircaloy-4 and filled with a material which had mechanical characteristics representative of the fuel pellets. The applicant clarified that the structural and geometric characteristics of the dummy assemblies were identical to those of the production assemblies in the allowable contents of the FCC-4 package.

Appendix 2.1-6 of the application provided an evaluation of the bending strength of fuel rods cladded with M5®. The applicant conducted bending tests on M5®-clad fuel rods. The results of the deformation capacity of the M5® rods were compared to and exceeded the calculated bending loads during a 9-meter drop, which was used to define the minimum acceptable mechanical properties for M5® cladding to ensure containment is maintained during a drop accident.

Appendix 2.2-2 of the application provided an evaluation of the behavior of the package contents during the thermal test (fire accident) scenario, which demonstrated that the fuel cladding retained its integrity. Appendix 2.2-3 of the application also provided an evaluation of the thermal-mechanical behavior of the fuel rods during the thermal test, which accounted for the maximum temperatures of the hottest rods for the M5® and Zircaloy-4 claddings as per Appendix 2.2-2, the azimuthal thermal gradient, and the presence of pellets inside the rods. The evaluation demonstrated that the risk of ballooning and bursting of the cladding under the effect of creep can be ruled out for both cladding alloys.

Based on the staff's review of the mechanical and thermal tests of the packaging and fuel contents, the staff finds that the fuel cladding is capable of maintaining the fuel in its analyzed configuration during normal and accident conditions of transport, and the package meets the requirements in Paragraphs 673, 682, and 726 of IAEA SSR-6.

5.8 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 FCC-4 package and that the package meets the requirements of IAEA SSR-6.

6.0 CRITICALITY SAFETY EVALUATION

6.1 Criticality SER for FCC-4 Package

The FCC-4 has two different versions of the design, Version 1 and Version 2. The difference between Versions 1 and 2 are geometrical differences in the size of the cavity. The Version 2 has reduced cavity dimensions and a thicker resin within the doors such that the external dimensions remain consistent with Version 1. Section 1.0 of this SER includes a description of the package's contents.

The French authority determined that the Model No. FCC-4 meets the regulations for the safe transport of radioactive material in the IAEA SSR-6, 2012 Edition. The staff's basis for revalidation is based on the package meeting the paragraphs that apply to criticality safety regulations within these regulations: 501(c), 526, 673, 682, 684, 685, 686, 716, 814, 815, and 816. The package was not evaluated for air transport, therefore, the package is not authorized for air transport.

6.2 Design Description

The packaging consists of a horizontal cylindrical casing consisting of two connected half-shells that hold a metallic cradle for shock absorption. Inside the casing is a support frame, which is supported by the metallic cradle when the package is in the horizontal position, that holds the contents and this frame contains a neutron absorbing resin for criticality control.

The staff determined that the application includes all information necessary for the staff to determine that the design meets the requirements in Paragraph 673 of the IAEA SSR-6. This application meets the requirement in Paragraph 815 of the IAEA SSR-6. This includes drawings and a description of the package features and the tests performed to the package.

Drawings of the package referenced in the certificate are the following:

1) Drawing No. 229K0400,

2) Drawing No. 229K0600

3) Drawing No. 229K0500

The staff specifically reviewed Drawing Nos. 220K0402, 229K0502, and 229K0602, which are referenced by the above drawings, to verify the dimensions of the neutron absorber resin.

The applicant included the specifications of the resin in Chapter 1.4, “Specification Relating to the Packaging,” of the application. Table 1.4-1 of Chapter 1.4 of the application contains the composition and density of the resin, while Table 1.4-4 of Chapter 1.4 of the application contains the inspection criteria for this material. This information shows that the presence and distribution of the neutron poison will be appropriately verified per the requirement in Paragraph 501(c) of the IAEA SSR-6.

6.3 Contents

The applicant has grouped the contents to reduce the number of criticality evaluations such that a single evaluation represents multiple contents. The staff has summarized the contents in Table 1 as well as the application and the certificate appendix that defines each content. The certificate appendices contain the detailed description of each content. For assemblies this includes: type of array, grid pitch, maximum weight of assembly, maximum UO_2 weight of the assembly, nominal active length, maximum number of fuel rods, cladding material, minimum thickness and minimum diameter, pellet maximum diameter, maximum UO_2 density, maximum enrichment, and maximum mass ratio of other uranium isotopes (^{232}U, ^{234}U, ^{236}U). For rods, the certificate contains information about the type of array, length of rods, maximum mass of rods per box, maximum mass of UO_2 per box and per package, nominal active length, maximum number of rods per box, cladding material, minimum thickness and minimum diameter, pellet maximum diameter, maximum UO_2 density, maximum enrichment, maximum mass ratio of other uranium isotopes (^{232}U, ^{234}U, ^{236}U), and minimum mass of Gd_2O_3 in gadolinium fuel rods.

6.4 Package Model Configuration

The applicant replaced the assembly hardware with moderator (water). The staff found this acceptable because fuel assemblies are under moderated, therefore, adding extra moderator increases reactivity and it is conservative. The applicant modeled the cladding as zirconium and the staff found this acceptable because the cladding is zirconium based and any other additives are likely to have negligible effect on reactivity.

6.5 Criticality Safety Evaluation under Normal Transport Conditions

The applicant describes its model for normal transport conditions in Appendix 2.5-1 and 2.5-2, Section 4.1.1, of the application. The staff found that Section 4.1.1 adequately reflects the condition of the package under routine and under normal conditions of transport as defined in Paragraphs 719 to 724 of the IAEA SSR-6, and evaluated in Chapter 2.5 of the application.

6.5.1 Normal Transport Conditions – Content Nos. 1, 2, 3 and 12

The applicant provided the characteristics of the assemblies for content Nos. 1, 2, 3 and 12, PWR 16x16, 17x17, and 18x18 and EPR 17x17 assemblies, in Table 2 of Appendix 2.5-1 of the application. The applicant only modeled the fissile height of the fuel and cladding and replaced the rest of the fuel hardware with water. The staff considers this assumption conservative because the applicant demonstrated that the rods are under moderated and adding additional moderator will increase reactivity. The applicant made other simplifying assumptions that the staff found acceptable and conservative. The applicant used an enrichment and fuel/assembly dimensions from Table 2 of Appendix 2.5-1 of the application that are bounding with respect to the allowable content specifications within the French certificate.

6.5.2 Normal Transport Conditions – Content Nos. 4 through 8, 10, 11 & 13

These contents can contain a minimum of 2% Gd_2O_3 content for gadolinium rods. Gadolinium is a neutron absorber and decreases the reactivity of the rods. The criticality models assume that no gadolinium is present and this is conservative. The applicant has stated that they modeled the Gd_2O_3 rods in an infinite array with optimum moderation that shows that they are subcritical. This substantiates the staff's finding that neglecting the gadolinium is conservative.

The applicant modeled two rod diameters as discussed in Section 4.4 in Appendix 2.5-1 of the application to cover all of these contents within the criticality evaluation. The rod diameter of greater than 10 mm covers Content Nos. 6, 7, 8, 10 while the rod diameter smaller than 10 mm covers Content Nos. 4, 5, 11 and 13.

For "small quantities of rods", the applicant assumed there would be enough material for 10 rods that would form the most reactive geometry (a sphere). In this configuration there are no constraints that keep the rods in an organized geometry and they can reconfigure, therefore, the staff found that assuming the most reactive geometry is conservative and acceptable.

6.6 Criticality Safety Evaluation under Accident Conditions of Transport

The applicant describes its model for accident transport conditions for Content Nos. 1, 2, 3 and 12 in Section 4.1.2 of Appendix 2.5-1 and Section 5.4.2 of Appendix 2.5-2 for Content Nos. 4 – 8, 10, 11, and 13.

The applicant modeled the maximum crushing of less than 10-inches due to the 9 m drop and assumed that the package takes on a rectangular shape to simulate the crushing of the cylindrical surface due to the drop conditions. The staff found that this is bounding of the results of the drop tests documented in Chapter 2.1 of the application.

For all of the allowable contents, the applicant's evaluations considered water ingress within the package and partial and preferential flooding to the extent that it produced maximum reactivity.

6.6.1 Accident Transport Conditions – Content Nos. 1, 2, 3, and 12

For content Nos. 1 and 2, the applicant modeled maximum pitch expansion for 1/3 of the height of the assembly to simulate the consequences from a drop event. A larger pitch increases reactivity due to assemblies being under moderated. However, based on staff experience (Refs. 6.13.1 and 6.13.2), the pitch of a fuel assembly does not uniformly expand as a result of a drop accident. The assembly may experience some pitch expansion, but also have areas of pitch contraction or experience a sinusoidal deformation. Therefore, the staff found that a maximum pitch expansion for 1/3 of the height is a reasonably bounding assumption to simulate the effects from the drop on the assembly.

6.6.2 Accident Transport Conditions – Content Nos. 4 through 8, 10, 11, and 13

Since the applicant determined and modeled optimal pitch for the normal condition of transport, the applicant used the same model for the accident condition. The staff found that this is conservative and acceptable, as it is unlikely that rods will reconfigure in a way that is more reactive than the optimal pitch condition.

6.7 Material Properties

Under NCT, the applicant modeled the neutron absorbing resin according to the properties in Table 3 of Appendices 2.5-1, and 2.5-2 of the application. The staff verified that the applicant analysis matched with Table 1.4-1 of Chapter 1.4 of the application. The applicant uses a conservative density with respect to the required mean measurement density of the resin. Under accident transport conditions the applicant changed the modeling of the resin consistently with the tests performed in Appendix 2.2-4 of the application, and, based on the discussion in Section 5.5 of this SER, the staff found this acceptable.

The applicant neglected the presence of the chromium oxides within the UO_2 fuel pellets. Because the amount is small and the staff is aware of studies (Refs. 6.13.3 and 6.13.4) to support criticality safety analyses for other packages involving UO_2 fuel with chromium oxides that show that the reactivity effect of a small quantity of chromium oxide is statistically insignificant, the staff found this acceptable.

6.8 Analysis Methods and Nuclear Data

The applicant states that it uses the CRISTAL V1.1 calculation code for all criticality safety analyses and the JEF2.2 cross section library. The applicant provided the validation of this code in Appendix 2.5-6 of the application.

The staff reviewed this report. The staff does not necessarily agree with the method used to determine the bias, as it appears that the applicant has set the k_{eff} of all critical experiments equal to one. The staff found this unusual as it would be rare for a critical experiment to be exactly critical and realistically are slightly subcritical or slightly supercritical and the staff does not have enough information about the actual critical experiments to determine that assuming they are all exactly critical is conservative or not. Still, the staff expects that critical experiments should have a k_{eff} close to one and acknowledges that the CRISTAL V1.1 code in all but one case predicts a k_{eff} over one which likely means that the code would have a positive bias, which the applicant has truncated to be a zero bias, as this is conservative. The applicant also showed results of the CRISTAL V1.1 code as compared to other widely used criticality codes (i.e., MCNP, KENO, etc.) which showed that the CRISTAL V1.1 code calculates overall higher k_{eff} than other criticality codes which further supports the assertion that the code has a positive bias. In addition, the applicant also combined uncertainties due to the following:

1) calculation statistical uncertainty (due to the statistical nature of the Monte Carlo method),

2) experiment uncertainty, and

3) uncertainty due to the manufacturing tolerances.

The applicant added these to the uncertainty that is used to determine the upper subcritical limit (USL). Although the applicant calculates a USL, the applicant does not use it for the analyses related to the Model No. FCC-4. Instead, the k_{eff} values reported by the applicant include the bias and bias uncertainty and the resulting k_{eff} is shown to remain below 0.95.

6.9 Demonstration of Maximum Reactivity

The FCC-4 is not leaktight. Therefore, the applicant assumes that water leaks into the system to the extent that it produces maximum reactivity per Paragraphs 680, 682, and 731 of the IAEA SSR-6. The applicant must demonstrate that the system is subcritical under these conditions for both a single package and an array of packages if the applicant requests to ship multiple packages per Paragraphs 682, 684, and 685 of the IAEA SSR-6.

In Section 5.2 and 5.3 of Appendix 2.5-1 of the application, the applicant states that the system under normal and accident conditions for both single packages and arrays (where the array is not infinite) is reflected by 20 cm of water. This is consistent with the requirements in Paragraph 681 of the IAEA SSR-6.

The applicant noted in Section 5 of Appendices 2.5-1 and 2.5-2, of the application that the various normal and accident conditions studied to determine maximum reactivity include the following:

1) low density water (mist conditions),

2) variations in pellet diameter,

3) presence of ribs on the doors,

4) tolerances of package components,

5) partially immersed assemblies, and

6) movements of the rods.

The applicant found that for the fuel assemblies, Content Nos. 1, 2, 3, and 12, in the fully flooded condition, rods at maximum pellet diameter and nominal dimensions produces the maximum reactivity condition. For the loose rods, Content Nos. 4 through 8, 10, 11, and 13, the applicant used the APOLLO 2 code to determine the number of rods and moderation. The applicant calculated the maximum material buckling and spread the rods uniformly over the cross section of the rod box. Although staff is less familiar with the APOLLO code, the staff found information documenting that it has been benchmarked for calculating the maximum material buckling (Ref. 6.13.5), and the staff found its use acceptable for that purpose in this application. To determine optimum moderation, the applicant varied the number of rods as distributed throughout the rod box space to determine the maximum material buckling and used this to determine maximum reactivity conditions. For Content Nos. 4 through 8, 10, 11, and 13, the applicant studied the effect of replacing the spacers (referred to as “wedges” in the application) with water and found that the most reactive configuration is with these spacers represented as water. The staff found that this provides a reasonable demonstration that the applicant is modeling the FCC-4 package using assumptions that produce maximum reactivity.

The applicant performed additional studies for ruptured rods and the impact of the cavity being off-center from the shell as discussed in Section 5 of Appendix 2.5-1 of the application. The staff found this further demonstrates these effects do not increase reactivity and the applicant's assumptions remain conservative.

For arrays of packages, the applicant also calculates k_{eff} for the configuration where the space between the cavity and the shell is empty which it refers to as the differentially flooded or differentially drained condition, which is the most limiting condition for the arrays of packages. The applicant states in Reference 6.13.6 (which the applicant states was applicable to the FCC-4 in Reference 6.13.7) that arrays of packages are modeled with no space between them. This is conservative as it increases neutron communication between packages and meets the intent of Paragraph 685(a) of the IAEA SSR-6, which requires hydrogenous moderation between packages. Since there is no space between the packages, the intent of this regulation is satisfied by modeling the array conditions in a more reactive way.

6.10 Single Package Evaluation

For the individual package, the applicant calculated the following k_{eff} values for the allowable contents. The k_{eff} values for all fuel assemblies represented by content Nos. 1, 2, 3, and 12 are less than 0.9. For all loose rods transported in rod boxes represented by content Nos. 4 through 8, 10, 11, and 13 the k_{eff} values are all less than 0.8.

6.11 Package Arrays

The number of packages that can be shipped is limited by the Criticality Safety Index (CSI). The method of calculation of the CSI is defined in Paragraph 686 of the IAEA SSR-6, which involves the size of the array used to perform the criticality safety evaluations. Separate evaluations are needed for each content to determine that each content meets the criticality safety regulations and, therefore, the applicant has determined a unique CSI for each content, or group of contents if they used a single criticality evaluation to represent multiple contents. The CSI of each content is listed in Table 4.

Radial and axial support spacers are required for rod boxes with mass over a certain quantity of UO_2 as described in Chapter 1.3-1 of the application and referenced in the French certificate for the FCC-4 package.

Table 2. CSI for the allowable contents in the Model No. FCC-4.

Content Number	**CSI**
1, 12	0.625
2, 3	8.33
4, 5, 6, 7, 8, 10, 11, 13	0 for UO_2 rods with radial and axial support 0 for UO_2-Gd_2O_3 rods with or without axial support 50 for small quantities of UO_2 rods without radial or axial support

6.11.1 Package Arrays Under Normal Transport Conditions

For Content Nos. 1, 2, 3, and 12 in Section 5.3.1 of Appendix 2.5-1 and Content Nos. 4 through 8, 10, 11, and 13 in Section 5.4.1 of Appendix 2.5-2 of the application, the applicant explains that it assumed for the array modeling under normal conditions that the package has total reflection on all faces of the package. This simulates an infinite array of packages, which is appropriate for or conservative with respect to the CSI of the respective contents and acceptable to the staff.

For arrays of packages under normal conditions of transport, the applicant calculated the following k_{eff} values for the allowable contents. The k_{eff} values for all fuel assemblies represented by content Nos. 1, 2, 3, and 12 are less than 0.95. For all loose rods transported in rod boxes represented by content Nos. 4 through 8, 10, 11, and 13 the k_{eff} values are all less than 0.85.

For small quantities of rods allowed for Content Nos. 4 through 8, 10, 11, and 13, as discussed in Section 6.5.2 of this SER, the applicant modeled a single package. The applicant did not state if it had modeled arrays of packages. However, the applicant's CSI is 50, which implies, per Paragraphs 684 through 686 of the regulations, a normal condition array of 5 packages and an accident condition array of 2 packages. As stated in Section 6.5.2 of this SER, the modeling of this content is very conservative and it is therefore the staff's judgment that array analyses would also be able to demonstrate that these arrays of packages meet subcriticality requirements. To investigate the staff's assumption, the staff performed independent calculations using the KENO-VI code as part of the SCALE 6.2.3 package and a very conservative representation of the package with this content. The staff modeled the content as a homogenous sphere of 5% enriched uranium and water using the allowable mass of uranium and the optimal moderation ratio found by the applicant documented in Section 5.2 of Appendix 2.5-2 of the application. The staff used the reduced package spacing from the accident condition evaluations and assumed no moderator between packages to increase neutron communication in an infinite array and significantly less neutron absorber resin material than the applicant justified for the accident conditions (discussed in Section 6.7 of this SER). The staff's calculations show that k_{eff} is significantly below 0.95 substantiating its expectation that for small quantities of rods for Content Nos. 4 through 8, 10, 11, and 13 array conditions (normal and accident), the package meets requirements in Paragraphs 684 and 685.

6.11.2 Package Arrays Under Accident Transport Conditions

For Content Nos. 4 through 8, 10, 11, and 13 in Section 5.4.2 of Appendix 2.5-2 of the application, the applicant explains that, for the array modeling under accident conditions, it assumed that the package has total reflection to all faces of the package. This simulates an infinite array of packages, which is appropriate for the CSI of these contents and, therefore, acceptable to the staff. Small quantities of rods are discussed in the previous section of this SER.

For Content Nos. 1, 2, 3, and 12 the applicant modeled the array sizes discussed in Section 5.3.2 of Appendix 2.5-1 of the application.

For the arrays of packages under accident conditions of transport the applicant calculated the following k_{eff} values for the allowable contents. The k_{eff} values for all fuel assemblies represented by content Nos. 1, 2, 3, and 12 are less than 0.95. For all loose rods transported in rod boxes represented by content Nos. 4 through 8, 10, 11, and 13 the k_{eff} values are all less than 0.85.

The staff found that the array sizes and the values of N are appropriate for the allowable CSI value for each content.

In Reference 6.13.6, which the applicant stated was applicable to the FCC-4 in Reference 6.13.7, the applicant notes that the statistical uncertainty for all criticality safety calculations of the FCC-4 from the MORET monte carlo code is equal to or less than the values in Section 4.2.2.2 of Appendix 2.5-6 of the application and is accounted for within the code (CRISTAL V.1.1) uncertainty determined in App. 2.5-6 of the application. This deviates from the recommendations used for domestic certificates as described in Section 6.4.1.3 of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," and NUREG/CR-6648, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology." In these documents, the staff recommends considering the uncertainty of the calculational method when evaluating the code bias and bias uncertainty. It does not state that this can also be used to account for the calculational method uncertainty for the safety calculations used to support licensing. As both of these NUREGs state, the results of the safety evaluations should also include an additional 2 sigma (2σ) to these results. Although the staff does not necessarily endorse the way the calculational uncertainties related to the Monte Carlo method have been treated within this application, it recognizes that these uncertainties are very low and even if 2σ was added to all of the k_{eff} results for the FCC-4 package, the package would still be subcritical (i.e., k_{eff} below 0.95, considering a 0.05 administrative margin). Therefore, the staff found the results of the FCC-4 criticality safety analyses adequately demonstrate that the package is subcritical.

6.12 Evaluation Findings

The staff finds that the French Certificate for the FFC-4 conforms to the criticality safety requirements within these regulations: 501(c), 526, 673, 682, 684, 685, 686, 716, 814, 815, and 816 of the IAEA SSR-6. Based on the statements in the safety evaluation above, the staff recommends revalidation of French Certificate of Competent Authority F/437/AF-96, Revision Fq, for the FCC-4 package.

6.13 References

6.13.1 U.S. Nuclear Regulatory Commission, "Certificate of Compliance No. 9380, Revision No. 0, for the Model No. Traveller STD & XL Package," November 2019, ADAMS Accession No. ML19311C542.

6.13.2 NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages," September 2015, ADAMS Accession No. ML15266A413.

6.13.3 Letter from the NRC to W. Stilwell, Westinghouse Electric Company, "Special Authorization for a One Time Shipment of the Model No. Traveller Package," December 20, 2018, ADAMS Accession No. ML18354B136.

6.13.4 Letter from T. Tate, U.S. Nuclear Regulatory Commission (NRC), to Framamtome Inc., "Certificate of Compliance No. 9319, for the Model Nos.

MAP-12 and MAP-13 Transportation Packages, Revision No. 11," January 10, 2019, ADAMS Accession No. ML19011A013.

6.13.5 Santamarina, D. Bernard, P. Blaise, L. Erradi, P. Leconte, R. Le Tellier, C. Vaglio, J-F. Vidal, CEA-Cadarache, "APOLLO2.8: A validated code package for PWR neutronics calculations," Conference Paper, April 2009, Advances in Nuclear Fuel Management IV (ANFM 2009, Hilton Head Island, South Carolina, USA, April 12-15, 2009).

6.13.6 Letter from R. Boyle, U.S. Department of Transportation (DOT) to N. García Santos, U.S. Nuclear Regulatory Commission (NRC), "Request for Additional Information, French Approval Certificate Number F/348/AF-96, Revision Fq, Model No. FCC-4 Package," October 29, 2020, ADAMS Accession No. ML20309A844.

7.0 QUALITY ASSURANCE

The purpose of the quality assurance (QA) review is to verify that the package design meets the requirements of the IAEA SSR-6. The staff reviewed the description of the QA program for the Model No. FCC-4 package against the standards in the IAEA SSR-6.

7.1 Evaluation of the Quality Assurance Program

The applicant developed and described a QA program for activities associated with transportation packaging components important to safety. Those activities include design, procurement, fabrication, assembly, testing, modification, maintenance, repair, and use. The applicant's description of the QA program (i.e., management system and compliance assurance programs in IAEA SSR-6, 2012 Edition) meets the requirements of the applicable IAEA SSR-6. The staff finds the QA program description acceptable, since it allows implementation of the associated QA program for the design, procurement, fabrication, assembly, testing, modification, maintenance, repair, and use of the Model No. FCC-4 transportation package.

The staff finds, with reasonable assurance, that the QA program for the FCC-4 transportation packaging meets the requirements in IAEA SSR-6 by encompassing the following:

1. design controls,
2. materials and services procurement controls,
3. records and document controls,
4. fabrication controls,
5. nonconformance and corrective actions controls,
6. an audit program, and
7. operations or programs controls, as appropriate.

These controls are adequate to ensure that the package will allow safe transport of the radioactive material authorized in this approval.

7.2 Evaluation Findings

Based on review of the statements and representations in the Model No. FCC-4 package application and as discussed in this SER section, the staff has reasonable assurance that the FCC-4 package meets the requirements in IAEA SSR-6. The staff recommends revalidation of French Competent Authority Certificate of Approval F/348/AF-96, Revision Fq.

CONDITIONS

The staff recommends the revalidation of French Competent Authority Certificate of Approval F/348/AF-96, Revision Fq, for the Model No. FCC-4 package, with the following additional condition:

Transport by air is not allowed.

CONCLUSION

Based on the statements and representations contained in the documents referenced above, and the conditions listed above, the staff concludes that the changes to the Model No. FCC-4 package meet the requirements of IAEA SSR-6.

Issued with letter to R. Boyle, U. S. Department of Transportation,
on April 22, 2021.