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

SAFETY EVALUATION REPORT

Model No. FCC-3 Package
French Certificate of Approval No. F/347/AF-96, Revision Fs
Docket No. 71-3083

Enclosure 1

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SUMMARY

By letter dated December 12, 2019 [Agencywide Documents Access and Management System (ADAMS) Accession No. ML20052D258], as supplemented on October 9, 2020, and October 29, 2020 (ADAMS Package Accession No. ML20309A842), the U.S. Department of Transportation requested that the U.S. Nuclear Regulatory Commission (NRC) staff perform a review of the French Approval Certificate Number (No.) F/347/AF-96, Revision Fs, Model No. FCC-3 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-3 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. 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/347/AF-96, Revision Fs, Model No. FCC-3 package, for the contents included in Section 1.1.2, "Contents," of this safety evaluation report (SER).

1.0 GENERAL INFORMATION REVIEW

The staff provided a recommendation to DOT on the Model No. FCC-3 package on August 9, 2010 (ADAMS Accession No. ML102220064), under the IAEA Safety Standards Series No. TS-R-1, "Regulations for The Safe Transport of Radioactive Material," 2009 Edition. The staff focused the review of this submittal on the changes made to the design of the Model No. FCC-3 since the revalidation recommendation provided to DOT in 2010. Document No. DOS-19-021165-000-NPV, Version 1.0, includes the list of documents related to this submittal with the documents' dates and revision numbers. The changes to each document are briefly described in the documents submitted as part of the application.

1.1 Package Description

1.1.1 Packaging

The Model No. FCC-3 packaging has a length of 4,923 millimeters (mm), an outside diameter of 1,048 mm and a maximum loaded weight of approximately 4,000 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 contents consist of pressurized water reactor's (PWR's) assemblies or rods in a box. There are 8 different contents allowed in either Version 1 or Version 2.

a. Version 1 - PWR 17x17 12-foot (ft) assemblies or rods, and 15x15 assemblies or rods; PWR 14x14 8- and 10-ft rods.

b. Version 2 - PWR 14x14 8- and 10-ft assemblies.

A single package can ship up to 2 assemblies or assembly boxes.

The assemblies and rods are all uranium dioxide (UO_2) PWR fuel with zirconium alloy cladding. The pellets may contain chrome oxides in the form of chromium oxide (Cr_2O_3). As referenced by the applicant in Reference 6.13.6, this is stated in ANP-10340P, Revision 0, "Incorporation of chromia-doped fuel properties in AREVA approved methods." All contents are enriched natural uranium. The supporting safety analyses mention enriched reprocessed uranium as contents; however, this is not allowed by the French certificate and was not evaluated for revalidation by the staff.

Table 1. Contents description and location of supporting analysis for the FCC-3 package.

Content Description in the French Certificate	Certificate Appendix	Application's Criticality Evaluation Reference
Maximum of 2 new PWR 17x17 twelve-foot fuel assemblies, in Version 1 of the packaging	1	Appendix 2.5-1
Maximum of 2 new PWR 15x15 fuel assemblies, in Version 1 of the packaging	2	Appendix 2.5-1
Maximum of 2 new PWR 14x14 8-ft fuel assemblies, in Version 2 of the packaging	3	Appendix 2.5-3
Maximum of 2 new PWR 14x14 10-ft fuel assemblies, in Version 2 of the packaging	4	Appendix 2.5-3
Maximum of 2 boxes, containing new PWR 17x17 twelve-foot nonassembled fuel rods, minimum 2 percent (%) gadolinium oxide (Gd_2O_3) content of gadolinium rods, in Version 1 of the packaging (rods containing less than 2% Gd_2O_3 are considered to be UO_2 rods)	5	Appendix 2.5-2
Maximum of 2 boxes, containing new PWR 15x15 non-assembled fuel rods, minimum 2% Gd_2O_3 content of gadolinium, in Version 1 of the packaging	6	Appendix 2.5-2
Maximum of 2 boxes, containing new PWR 14x14 8-ft non-assembled fuel rods, minimum 2% Gd_2O_3 content of gadolinium, in Version 1 of the packaging	7	Appendix 2.5-2
Maximum of 2 boxes, containing new PWR 14x14 10-ft nonassembled fuel rods, minimum 2% Gd_2O_3 content of gadolinium, in Version 1 of the packaging (rods containing less than 2% Gd_2O_3 are considered to be UO_2 rods)	8	Appendix 2.5-2

To summarize the above table, there are 8 different contents and 3 different analyses that cover these contents. Contents Nos. 1 and 2 are represented by the criticality analysis in Chapter 2.5-1 of the application, Contents 3 and 4 are represented by the criticality analyses in Appendix 2.5-3 of the application and Contents 5, 6, 7, and 8 are represented by the criticality analyses in Appendix 2.5-2 of the application.

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, 2012 Edition. The applicant performed structural analyses to demonstrate that the strength of the FCC-3 package meets the requirements specified in IAEA SSR-6, 2012 Edition. Specifically, the applicant addressed the FCC-3 package under:

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

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

3) Accident conditions of transport.

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

2.2 Strength Analysis under Routine Conditions of Transport

Chapter 2.1, "Structural Analysis," of the application (document No. DOS1300081779-100, Revision 1) describes the strength analyses performed to demonstrate the behavior of the FCC-3 package in routine conditions of transport with the tie-downs in place. Specifically, the applicant considered the following tie-down configurations:

1) ***Tie-down No. 1:*** container with two straps placed in proximity to the inner hole of the lifting box;

2) ***Tie-down No. 2:*** container with two slings placed on the upper half-shell reinforcement, in proximity to the inner reinforcing angle of the lifting box; and

3) ***Tie-down No. 4:*** container with four straps, two straps placed on the upper half-shell reinforcement, in proximity to the inner reinforcement angle bar of the lifting box, and two straps placed on the upper half-shell reinforcement, in proximity to the central reinforcement angle bar.

The applicant performed calculations to determine the stress levels reached for these three tie-down configurations under the maximum authorized weight of the FCC-3 package. The applicant then used these calculated stresses of the analysis as input data for the fatigue analysis, which considered the cumulative stresses associated with transportation, handling, and stacking. This fatigue analysis demonstrates that the estimated minimum life of each tie-down configuration is larger than the required life of the FCC-3 package.

Chapter 2.1 of the application describes the following possible lifting modes:

1) 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; and

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

The applicant demonstrated the acceptability of lifting the FCC-3 package via the lifting lugs located on the upper shell with a conservative maximum package weight. The applicant's numerical calculations show that the FCC-3 package meets the criteria for excessive strain and plastic instability. Additionally, the applicant performed a fatigue analysis using a conservative assumption of round-trip transports and concluded that the cumulative usage factor was significantly less than the allowable limit of 1.0 for structural and the welds.

In regard to the bolts for the FCC-3 package, the fatigue strength is determined based on the allowable number of tightening-lifting-loosening cycles (i.e., one round-trip transport being equivalent to two tightening-lifting-loosening cycles). The applicant determined that the allowable number of tightening-lifting-loosening cycles was significantly greater than the acceptance criterion (document No. DOS1300081779080, Revision 1).

The staff reviewed the analysis results and finds the applicant has adequately demonstrated that the Model No. FCC-3 package meets the regulatory requirement of IAEA SSR-6, 2012 Edition, for routine conditions of transport with package tie-down and lifting.

2.3 Strength 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-3 package because the following reasons:

1) the package was not designed for water tightness and the in-leakage of water into the packaging is considered in the criticality safety studies discussed in Chapter 2.5, "Criticality Safety Analysis," of the application (document No. DOS-13-00081778-500, Revision 3); and

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

Based on its review of the criticality safety studies discussed in Chapter 2.5 of the application, the staff confirmed that these analyses consider that the FCC-3 package is not watertight, and the presence of water is accounted for. Therefore, the staff finds the water spray and immersion tests required by IAEA SSR-6 to demonstrate leak-tightness are not applicable to the FCC-3 package. Further, the staff's review of the criticality safety studies is documented in Section 6.0 of this safety evaluation report (SER).

2.3.2 Stacking Test

In Chapter 2.1 of the application, the applicant indicates that analyses were performed to demonstrate compliance with the requirements of IAEA SSR-6, 2012 Edition for the stacking test. This was accomplished by calculation considering a compressive force, which equals to five times the maximum weight of the package. This force is 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 yield stress. The applicant showed that the container at the bottom of the stack is able to support the weight of the five other packages without exceeding the yield strength in the shell and welds of the FCC-3 package. The staff reviewed the results of the analysis and finds that the applicant adequately demonstrated that the FCC-3 package has adequate strength to meet the stacking requirement of IAEA SSR-6, 2012 Edition.

2.3.3 Perforation (Penetration) Test

In Chapter 2.1 of the application, the applicant describes the analyses performed to address the perforation (penetration) test required by IAEA SRR 6. The applicant first conservatively calculated the energy (capacity) necessary to penetrate the FCC-3 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 penetrate the FCC-3 package is larger than the energy (demand) from the dropped bar. The staff reviewed the analysis results and finds the applicant adequately demonstrated that the FCC-3 package has adequate strength to meet the penetration requirements of the IAEA SSR6, 2012 Edition.

2.3.4 Free Drop Test

In Chapter 2.1 of the application, the applicant indicates that the FCC-3 package was loaded with an assembly and a ballast weight, and subjected to a 9-meter (9-m) free drop, which is a more severe test than the 1.2 m free drop test required by the IAEA, SSR-6, 2012 Edition. Based on the results of the free drop test, the applicant indicated that the condition of the packaging after free drop test showed no significant damage. Specifically, it indicated that: (i) there was no dispersal of the radioactive content, (ii) the relative positions of the assemblies did not change, and (iii) deformation of the packaging remained localized. The staff reviewed the analysis and test results provided in the application and finds the applicant adequately demonstrated that the FCC-3 package has adequate strength to meet the regulatory requirement of IAEA SSR-6, 2012 Edition, and conservatively withstands the free-fall test.

2.4 Strength Analysis Under Accident Conditions of Transport

Chapter 1.5, “Package Performance Characteristics,” of the application (document No. DOS-19-021165-003, Version 1) indicates that the drop tests were carried out on representative full-scale prototypes of the packaging to demonstrate compliance of the FCC-3 package with the IAEA SSR-6, 2012 Edition. The applicant explained that the prototype fuel assemblies used during the drop tests were dummy assemblies with cladding made of Zircaloy-4. This was done to use identical structural and geometric characteristics of the fuel assemblies as assemblies transported in the FCC-3 package.

The applicant studied the possible free-fall drop configurations from 9 meters (m) and from 1 m onto a bar and selected configurations that maximized the possible damage to the following package components:

1) the closing system on the internal fittings (e.g., doors, frame, top and bottom plates, door-frame connections, top/bottom plate connections to the frame or doors);

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

3) the shell shock absorbers.

The applicant explained that the following drop sequences were performed for the prototype representative of the FCC-3 package:

1) ***first drop*** - a free-fall drop from 1 m onto a bar, against an edge of the door of the internal fittings (Places loads on the door-frame connections);

2) ***second drop*** - a free-fall drop from 1 m onto a bar, against an upper face of the door of the internal fittings (maximum load on the door of the internal fittings);

3) ***third drop*** - a free-fall drop from 9 m, in a vertical position along the centerline of the top end of the package (testing the shock-absorber of the shell and the bolted connections in the shells); and

4) ***fourth drop*** - a free-fall drop from 9 m, flat, with a 'whiplash' effect (initial impact at the top end to hit the bottom end target with maximum velocity).

Following the sequence of regulatory drop tests for the accident conditions of transport, the applicant indicated 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 the accident conditions of transport. The staff confirmed that the applicant considered the results and condition of the prototype package following the regulatory drop tests in the thermal analysis described in Chapter 2.2, "Thermal Analysis," of the application and the criticality analysis in Chapter 2.5 of the application. Sections 4.0 and 6.0 of this SER include the staff's thermal and criticality safety evaluations, respectively.

The staff reviewed the regulatory drop test results, including the supporting analyses, and finds the applicant adequately demonstrated that the FCC-3 package has adequate strength to meet the regulatory requirement of IAEA SSR-6, 2012 Edition, and has the ability to withstand the accident conditions of transport.

Additionally, the FCC-3 package is 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 application (document No. DOS-13-00081778-112, Revision 1). The applicant evaluated the mechanical behavior of the rod boxes transported in FCC-3 package under the accident drop conditions of transport. Specifically, the applicant's evaluation focused on: (i) the radial spacers, (ii) compensation spacers, (iii) axial spacers, (iv) support plates between the radial spacer and the rod bundle, and (v) end support plates for the fuel rods on the axial spacer side to demonstrate that their geometrical properties are not affected by stresses on the packaging due to accidents. The applicant demonstrated that plastic strain outside of the compression zone in the axial spacer and the fixed end plate is less than the true rupture elongation of the rod box; therefore, the applicant concluded that the rod box has adequate strength under the accident conditions of transport. The staff reviewed the supporting analyses for the rod boxes used to transport non-assembled fuel rods in the FCC-3 package and finds that the applicant adequately demonstrated that these rod boxes have adequate strength to withstand the accident conditions of transport.

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-3 package has adequate structural design to meet the requirements of IAEA SSR-6, 2012 Edition.

3.0 THERMAL EVALUATION

The current application requests additional contents that have no direct impact on the thermal performance of the package. Therefore, the package is evaluated against the thermal performance requirements in the IAEA SSR-6, 2012 Edition.

3.1 Description of Thermal Design

The design of the package is described in the approved French Competent Authority Certificate F/347/AF-96, Version Fs, dated September 24, 2010, and application for the Model No. FCC-3 (Ref. 3.8.2).

3.1.1 Design Features

The applicant did not propose changes to the design features of the FCC-3 package as part of this request for revalidation.

3.1.2 Contents Decay Heat

The addition of the proposed new contents did not result in changes to the decay heat of the package.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The applicant did not propose changes to the properties of the materials used to fabricate the FCC-3 package as part of this request for revalidation.

3.2.2 Component Specifications

The applicant did not propose changes to the specifications of the component for the FCC-3 package as part of this application for revalidation.

3.3 General Considerations

3.3.1 Evaluation by Analysis

The applicant provided descriptions of the analysis models and results of analyses completed to demonstrate the thermal performance of the FCC-3 package. Table 1 of this SER includes the documents that include the analyses related to the thermal performance of the package.

Table 2. Documents related to the thermal performance of the Model No. FCC-3 package.

Document No.	Application's Section	Title
DOS-13-00081778-201 (Ref. 3.8.3)	Appendix 2.2-1	"Thermal behaviour of FCC container during regulatory thermal test"
DOS-13-00081778-202 (Ref. 3.8.4)	Appendix 2.2-2	"Additional information for justification of the thermal behaviour of the FCC container"
DOS-13-00081778-206 (Ref. 3.8.5)	Appendix 2.2-6	"Thermal behaviour of the FCC packaging under normal conditions of transport"

The applicant used the results of these analyses as part of the demonstration that, for the content requested and already authorized by the French competent authority under Certificate F/347/AF-96, Version Fs, the FCC-3 package meets the thermal requirements in SSR-6, 2012 Edition.

3.3.2 Evaluation by Test

The applicant calculated fuel clad burst probabilities based on testing of cladding (both, M5® and Zircaloy-4) that was described in Appendix 2.2-3, "M5® or Zircaloy-4 Fuel Rod Behaviour During IAEA Thermal Test in FCC Packaging," of the application (Ref. 3.8.6). The conclusion of the report is that the "risk of ballooning and bursting of the cladding under the effect of creep can be ruled out for both alloys." Therefore, a cladding rupture for the temperatures possible during transport, including accidents, is not possible.

3.3.3 Margins of Safety

The thermal-mechanical analysis carried out on fuel rods that would exceed a temperature of 600°C show that the risk of bursting of the Zircaloy-4 or M5® cladding, due to creep, can be ruled out with a minimum safety factor of 3.3.

3.4 Thermal Evaluation under Normal Conditions of Transport

3.4.1 Heat and Cold

The additional new contents requested do not change the thermal performance of the package under the conditions stated for normal conditions of transport in paragraphs 654, 656, and 657 of the IAEA SSR-6, 2012 Edition. Therefore, the package continues to meet these requirements.

3.4.2 Maximum Normal Operating Pressure

The applicant did not propose changes to the maximum normal operating pressure of the FCC-3 package as part of this request for revalidation.

3.4.3 Maximum Thermal Stresses

No changes in the maximum thermal stresses in the package have been observed as part of this application for revalidation.

3.5 Thermal Evaluation under Hypothetical Accident Conditions

3.5.1 Initial Conditions

Initial conditions for the hypothetical accident conditions (HAC) evaluation are defined in paragraph 728(b) of the IAEA SSR-6, 2012 Edition. No changes in the initial conditions for the package evaluation have been observed as part of this application for revalidation.

3.5.2 Fire Test Conditions

The fire test conditions for the HAC evaluation are defined in paragraph 728(a) of the IAEA SSR-6, 2012 Edition. No changes in the fire test conditions for the package evaluation have been observed as part of this application for revalidation.

3.5.3 Maximum Temperatures and Pressure

No changes in the maximum temperatures and pressures for the package have been observed as part of this application for revalidation.

3.5.4 Maximum Thermal Stresses

No changes in the maximum thermal stresses in the package have been observed as part of this application for revalidation.

3.5.5 Applicable Supporting Documents or Specifications

A list of the documents referenced for this review are provided at the end of this section.

3.5.6 Analyses Details

No changes in the thermal analysis under hypothetical accident conditions.

3.6 Impact of Glycerine Combustion

The applicant performed a thermal analysis of the risks related to possible combustion of 5 grams (g) of glycerine during the HAC fire, which would potentially be present in a residual form on rods in the array, as it is applied to the rods prior to insertion into the assembly grids. Glycerine is a hydrogen-containing material, with an auto-ignition temperature of 400°C as described in Appendix 2.2-5 of the application. The applicant performed the thermal analysis considering the combined phenomena of (a) combustion of the layer of glycerine 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 transport in the FCC-3 package and the conclusions of the analysis on the impact of glycerine on PWR 17x17 assemblies are applicable to other types of fuel assembly. The applicant calculated temperature increases of 42.3°C by combustion of the layer of glycerine and of 26.3°C by overall energy input of combustion.

The staff validated that the assumptions, equations, and calculations shown in Appendix 2.2-5 of the application are acceptable for analysis of the glycerine combustion in the HAC fire. The staff confirmed that (a) the potential combustion of 5 g of glycerine 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.

3.7 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the FCC-3 package conforms to the requirements for Type A packages loaded with fissile materials found in the regulations for the safe transport of radioactive material, IAEA Safety Standards series, No. SSR-6, 2012 Edition.

3.8 References

3.8.1 AREVA TN, Document No. DOS-13-00081778-200, Chapter 2.2, "Thermal analysis," ADAMS Accession No. ML20052D341.

3.8.2 AREVA TN, Document No. DOS-19-021165-000, "Table of Contents," Version 1.0, ADAMS Accession No. ML20052D296.

3.8.3 AREVA TN, DOS-13-00081778-201, Appendix 2.2-1, "Thermal behaviour of FCC container during regulatory thermal test," ADAMS Accession No. ML20052D345.

3.8.4 AREVA TN, DOS-13-00081778-202, Appendix 2.2-2, "Additional information for justification of the thermal behaviour of the FCC container," ADAMS Accession No. ML20052D345.

3.8.5 AREVA TN, DOS-13-00081778-206, Appendix 2.2-6, "Thermal behaviour of the FCC packaging under Normal Conditions of Transport," ADAMS Accession No. ML20052D352.

3.8.6 AREVA TN, DOS-13-00081778-203, Appendix 2.2-3, "M5® or Zircaloy-4 Fuel Rod Behaviour During IAEA Thermal Test in FCC Packaging," ADAMS Accession No. ML20052D347.

3.8.7 AREVA TN, DOS-13-00081778-205, Appendix 2.2-5, "Analysis of the impact of glycerine on the thermal safety analyses," ADAMS Accession No. ML20052D351.

4.0 CONTAINMENT EVALUATION

The staff performed the containment review of the FCC-3 package application. The purpose of the containment review is to verify that proposed changes to the FCC-3 package design satisfies the requirements for the evaluation of the containment boundary as required in the IAEA SSR-6, 2012 Edition. The staff reviewed the application and confirmed that the FCC-3 package containment system had a suitable description for revalidation.

4.1 Description of the Containment System

The fuel cladding along with the zirconium alloy welded end plugs form the containment boundary for the fissile material in the FCC-3 package. This information is located in Chapter 1.5 "Package Performance Characteristics," Section 2.4, "Enclosure," of the application. After reviewing the description, the staff finds the description to be adequate.

Given that the fuel cladding is critical to the containment boundary, the staff reviewed the cladding performance during normal and accident conditions of transport. In Chapter 2.1, "Structural Analysis," Section 4.3.3, "Representativity of the prototype 2 contents," of the application, the applicant describes that the pre-oxidation surface treatment of the cladding has no impact on the mechanical behavior of the assembly in both normal and accident conditions of transport.

In Chapter 1.5, Section 3.2, "Normal conditions of transport," and Chapter 2.1, Section 4.2.4, "Conclusion" of the application, the applicant describes that there is no dispersal of radioactive contents on completion of the regulatory tests for normal conditions of transport. Furthermore, in Chapter 1.5, Section 3.3, "Accident conditions of transport," of the application, the applicant provides results of satisfactory behavior of the Zircaloy-4 rods in the fuel assembly prototypes during the regulatory drop tests are extended to the Zirconium M5® alloy rods.

In Chapter 2.1, Section 4.3.5, "Drop test results for prototype 2," of the application, the applicant summarizes that the fuel cladding remains intact, is "leak tight," and there is no dispersal of the radioactive contents after the tests representative of accident drop conditions. In Chapter 2.2, "Thermal Analysis," Section 10, "Conclusion," of the application, the applicant concludes that for the Zircaloy-4 and M5® cladding there is no dispersal of material after completion of the tests representative of accident conditions.

In Appendix 2.2-5, "Analysis of the Impact of Glycerine on the Thermal Safety Analysis," Section 5, "Conclusion," of the application, the applicant describes the presence of a residual amount of 5 g of glycerine within the FCC-3 package. However, this amount does not bring into question the mechanical behavior of the M5® or Zircaloy-4 fuel rod cladding. Based on the information listed above, the staff finds that the fuel cladding will maintain containment during both normal and accident conditions of transport.

4.2 General Considerations

In paragraph 1.2 of the approval certificate for the FCC-3 packaging, the applicant describes the physical state of the contents as 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 8 of the approval certificate. After reviewing the appendices and the overall approval certificate, the staff finds the description of the contents of the package in the approval certificate for the FCC-3 packaging to be acceptable.

In Appendices 1 through 8 of the certificate of approval, the applicant describes that the maximum activity level per packaging is less than 1 A_2. In Chapter 1.6, "Compliance with Regulatory Requirements," Section 8.1, "Article 429," of the application, the applicant describes the A_2 value for enriched natural uranium (ENU) under 20% enriched uranium-235 (^{235}U)as unlimited. Therefore, staff finds that the certificate of approval adequately classifies the Model No.FCC-3 as a Type A(F) package since the activity of its contents is less than 1 A_2. In Section 8.2, "Article 430," of Chapter 1.6, of the application, the applicant describes the A_2 value for ENU under 20% ^{235}U is unlimited; therefore, the staff finds the activity being equal to 0 A_2 to be acceptable.

In Section 8.11, "Article 644," of Chapter 1.6 of the application, the applicant 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 the rods do not contain liquids or materials, which may generate gas by chemical reaction or radiolysis. Therefore, the staff finds this description to be acceptable.

4.3 Evaluation Findings

Based on review of the statements and representations in the FCC-3 application, the staff concludes that the containment design has been adequately described and evaluated and that the package design meets the containment requirements of the IAEA SSR-6 2012 Edition. The staff recommends revalidation of the French Certificate of Approval No. F/347/AF-96, Revision Fs.

5.0 MATERIALS EVALUATION

5.1 Drawings

The staff reviewed the drawings and Chapter 1.4, "Specification Relating to the Packaging," of the application and verified that the applicant provided an adequate description of the component safety functions, materials of construction, dimensions and tolerances, and fabrication specifications. The staff notes that the carbon steels and stainless steels used in the fabrication of the FCC-3 packaging are specified in Chapter 1.4 of the application as conforming to the applicable Association Française de Normalisation (AFNOR) standards.

Based on the evaluation above, the staff finds that the drawings contain sufficient information to describe the design and manufacture of the package, and the package meets the requirements in paragraph 640 of IAEA SSR-6, 2012 Edition.

5.2 Materials Standards

Table 1.4-1 of the application provides the materials standards for the carbon steel and stainless steel packaging components. The carbon steel materials are specified to meet AFNOR material standards NF EN 10025, NF EN 10028-3, or NF A 36-601; and the stainless steel materials are specified to meet AFNOR material standard NF EN 10088-3. In addition, the fasteners that link the two shells of the FCC-3 packaging are fabricated and surface treated in accordance with applicable AFNOR standards. The staff reviewed the package's materials standards and verified that these cite the appropriate French construction standards.

Based on the evaluation above, the staff finds that the design of the packaging materials meet the applicable standards, and the package meets the requirements in paragraph 640 of IAEA SSR-6, 2012 Edition.

5.3 Weld Design and Inspection

The staff reviewed the drawings and Table 1.4-5 of the application and verified that the applicant identified the weld materials, weld type, and post-weld examination requirements. The applicant notes that all welds are inspected visually or with dye penetrant by a qualified inspector. In Chapter 1.4 of the application, the applicant notes that the verification of appropriate weld construction is completed per French Code for Construction of Unfired Pressure Vessel (CODAP) or another equivalent code, and the application identifies the specific inspection criteria that must be met. Furthermore, Chapter 1.7, "Operation," of the application provides quality assurance provisions and requirements specific to the manufacturing activity and indicates that inspections must be carried out by qualified personnel. The staff reviewed the packaging weld design and verified that the welding and associated inspections are consistent with the applicable French construction code.

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, 2012 Edition.

5.4 Mechanical Properties

5.4.1 Carbon Steels

Chapter 1.4 of the application includes a description of the mechanical properties of the carbon steel packaging materials and specifies that the absence of brittle fracture at -40°C must be demonstrated for each grade of steel. Appendix 1.4-3, “Summary of Impact Strength Tests,” of the application documents the toughness tests that were carried out on the S355 high strength structural steel grade used in the package shells. The staff reviewed the property requirement of the structural steel materials and verified that the tensile properties conform to the applicable AFNOR standards and that the low-temperature toughness was adequately demonstrated.

5.4.2 Stainless Steels

In Appendix 1.3-1, “Description of Fuel Rod Boxes for FCC-3 Containers,” of the application, the applicant notes that the components of the rod box are made of Type 304L stainless steel and the mechanical properties conform to AFNOR NF EN 10088-3. The staff reviewed the temperature-dependent mechanical properties used in the applicant’s mechanical calculations and confirmed that the properties are consistent with those in the technical literature. The staff also confirmed that the applicant adequately considered fracture behavior in the package design because the austenitic stainless steels used in the FCC-3 package are resistant to brittle fracture at low service temperatures.

5.4.3 Wood (Impact Limiter)

In Chapter 1.4 of the application, the applicant notes 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. 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, and 648 of IAEA SSR-6, 2012 Edition.

5.5 Thermal Properties of Materials

The staff reviewed the thermal calculations in Chapter 2.2, “Thermal Analysis,” of the application and verified that the thermal properties (i.e., density, thermal conductivity, and heat capacity) of the materials are consistent with values available in the technical literature.

Based on the evaluation above, the staff finds that the properties of materials used in the thermal analysis are acceptable, and the package meets the requirements in paragraphs 616, 639, and 640 of IAEA SSR-6, 2012 Edition.

5.6 Criticality Control Materials

A polymeric neutron-absorbing resin is included in the support frames and the doors to maintain subcriticality. Table 1.4-1 of the application specifies the material composition and properties of the resin. A batch chemical analysis is performed to check the neutron absorbing properties of the resin (i.e., hydrogen and boron contents) and a qualified process (i.e., controlled injection procedure) is used to fill the packaging cavities with the resin. Based on the description of design and the manufacturing criteria provided by the applicant, the staff finds the resin properties used in the thermal and criticality analyses to be acceptable.

In addition, as described in Appendix 2.2-4, "Note on the Characterization of Resin FS 69," of the application, the applicant subjected resin samples to thermal tests to characterize the performance of the resin in a fire accident. The thermal tests included a direct-flame test where resin samples were exposed to an 800ºC flame and the resin material beneath the flame-exposed surface was measured for temperature (during the test) and post-test hydrogen and boron concentrations. The applicant used the test data to adjust the neutron absorption performance of the resin in the criticality analysis to account for potential degradation in a fire accident. The tests also demonstrated that the resin remained largely intact when exposed to the flame, forming a surface char layer that protected the material beneath.

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, 2012 Edition.

5.7 Corrosion and Chemical Reactions

In Chapter 1.6, "Compliance with Regulatory Requirements," of the application, the applicant describes 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.

Also, in Chapter 1.6 of the application, the applicant notes 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 carbon steels, 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, the periodic maintenance program specifies that the FCC-3 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-3 packaging transports unassembled fuel rods in the accompanying fuel rod boxes. The staff reviewed the design drawings 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-3 package. The staff finds that galvanic corrosion between the aluminum packing shims and the stainless steel components of the FCC-3 package is not expected because water is effectively sealed off under normal conditions of transport and 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.

Based on the evaluations above, the staff finds that the package design, inspections, and maintenance activities adequately prevent against adverse reactions that may affect the ability of the package to perform its safety functions, and the package meets the requirements in paragraph 614 of IAEA SSR-6, 2012 Edition.

5.8 Content Integrity: Fresh (Unirradiated) Fuel Cladding

In Chapter 1.5, "Package Performance Characteristics," of the application, the applicant notes that drop tests were carried out on full-scale prototypes with dummy fuel assemblies with cladding made of zircaloy-4. Therefore, the structural and geometric characteristics were identical to those of the production assemblies. In addition, Appendix 2.1-6, "Transportation in FCC Container Mechanical Aspects Related to a Change in the Fuel Assembly Materials," of the application includes the evaluation of the mechanical strength of the FCC-3 package transporting fuel rods made of M5® cladding. The regulatory tests of the behavior of the package during a fire was also conducted, and those tests demonstrated that the fuel cladding retained its integrity.

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.

5.9 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-3 package and that the package meets the requirements of IAEA SSR-6.

6.0 CRITICALITY SAFETY EVALUATION

6.1 Criticality SER for FCC-3 Package

The FCC-3 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-3 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, 2012 Edition. This meets the requirement in paragraph 815 of the IAEA SSR-6, 2012 Edition. 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. 229K0100,

2) Drawing No. 229K0200, and

3) Drawing No. 229K0700 for Version 1 and 229K0300 for Version 2.

The staff specifically reviewed Drawing Nos. 220K0102, 229K0202, and 229K0702, 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) the IAEA SSR-6, 2012 Edition.

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, max 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 Normal Transport Conditions

The applicant describes its model for normal transport conditions in Appendix 2.5-1, 2.5-2, and 2.5-3, 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, 2012 Edition, and evaluated in Chapter 2.1 of the application.

6.5.1 Normal Transport Conditions – Content Nos. 1 and 2

The applicant provided the characteristics of the assemblies for content Nos. 1 and 2, PWR 15x15 and 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. 3 and 4

Contents 3 and 4 are for transport in Version 2 of the FCC-3. The modeling assumptions discussed for Content Nos. 3 and 4 are consistent with those for Content Nos. 1 and 2.

6.5.3 Normal Transport Conditions – Content Nos. 5, 6, 7, and 8

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 to cover all four of these contents within the criticality evaluation included in Appendix 2.5-2 of the application. The rod diameter of greater than 10 mm covers Content Nos. 6, 7, and 8, while the rod diameter smaller than 10 mm covers Content No. 5.

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 Accident Transport Conditions

The applicant describes its model for accident transport conditions for Content Nos. 1 and 2 in Section 4.1.2 of Appendix 2.5-1; Section 5.4.2 of Appendix 2.5-2 for Content Nos. 5, 6, 7 and 8; and Section 5.3.2 of App. 2.5-3 of the application for Content Nos. 3 and 4.

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 and 2

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. 3 and 4

The applicant modeled a case with infinite length with pitch expansion to fill the entire cavity with water to demonstrate that the system with these contents remains subcritical under accident conditions considering these effects. Since the difference between Content Nos. 3 and 4 is the length of the assemblies, the staff found the applicant's analysis of infinite length conservative and applicable to both contents.

6.6.3 Accident Transport Conditions – Content Nos. 5, 6, 7, and 8

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 most optimal for reactivity.

6.7 Material Properties

Under normal conditions of transport, the applicant modeled the neutron absorbing resin according to the properties in Table 3 of Appendices 2.5-1, 2.5-2, and 2.5-3 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 5.6 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-3. 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.

The applicant provided a benchmarking report for the CRISTAL V1.1 code, while the criticality safety evaluations performed to support the transport of Content Nos. 3 and 4 in Appendix 2.5-3 of the application for the FCC-3 use the CRISTAL V0.2 code. The applicant provided additional benchmarking information in Reference 6.13.6 stating that the V0.2 version of this code also has a positive bias making the benchmarking performed for the CRISTAL V1.1 version applicable to the V0.2 version, at least for the bias. With respect to the other uncertainties that are treated separately from the benchmarking, as discussed above, the applicant showed that the uncertainties related to V0.2 of the code are the same as that of the V1.1. Therefore, the staff found that using the same bias and bias uncertainty is acceptable.

6.9 Demonstration of Maximum Reactivity

The FCC-3 is not leaktight. Therefore, the applicant assumes that water leaks into the system to the extent that it produces maximum reactivity per paragraph 680, 682, and 731 of the IAEA SSR-6, 2012 Edition. 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, 2012 Edition.

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, 2012 Edition.

The applicant noted in Section 5 of Appendices 2.5-1, 2.5-2, and 2.5-3 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 to 4, in the fully flooded condition, rods at maximum pellet diameter and nominal dimensions produces the maximum reactivity condition. For the rods, Content Nos. 5 to 8, 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. 5 to 8, 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-3 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, further demonstrating these effects do not increase reactivity and its 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 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 SSR6, 2012 Edition, 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 various allowable contents:

Table 3. K_{eff} values for allowable contents in the Model No. FCC-3.

Content No.	Description	Maximum k_{eff}
1	17x17 5% enriched ^{235}U assemblies	0.8< x < 0.9
2	15x15 5% enriched ^{235}U assemblies	0.8< x < 0.9
3,4	14x14 5% enriched ^{235}U assemblies	Depends on the array
5,6, 7, or 8	Loose pellets	0.7 < x < 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, 2012 Edition, 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-3 package.

Table 4. CSI for the allowable contents in the Model No. FCC-3.

Content Number	**CSI**
1	0.625
2	0.625
3	0
4	0
5	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	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
7	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
8	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 and 2 in Section 5.3.1 of Appendix 2.5-1; Content Nos. 5, 6, 7, and 8 in Section 5.4.1 of Appendix 2.5-2; and Content Nos. 3 and 4 in Section 5.3.1 of Appendix 2.5-3 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 the arrays of packages under normal conditions of transport, the applicant calculated the following k_{eff} values for the various allowable contents:

Table 5. Arrays of packages under normal conditions of transport.

Content No.	**Description**	**Array size**	**Maximum k_{eff}**
1	17x17 5% enriched ^{235}U assemblies	Infinite	0.9 < x < 1
2	15x15 5% enriched ^{235}U assemblies	Infinite	
3, 4	14x14 5% ^{235}U enriched assemblies	Infinite	0.8 < x < 0.9
5,6, 7, 8	Loose pellets	Infinite	

For small quantities of rods allowed for Content Nos. 5 through 8, as discussed in Section 6.5.3 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 IAEA SSR-6, 2012 Edition, 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% wt. enriched ^{235}U 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. 5 through 8 array conditions (normal and accident), the package meets requirements in Paragraphs 684 and 685 of the IAEA SSR-6, 2012 Edition.

6.11.2 Package Arrays Under Accident Transport Conditions

For Content Nos. 5, 6, 7, and 8 in Section 5.4.2 of Appendix 2.5-2, and Content Nos. 3 and 4 in Section 5.3.2 of Appendix 2.5-3 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. For Content Nos. 1 and 2, the applicant modeled an array size 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 various allowable contents:

Table 6. Arrays of packages under accident conditions of transport.

Content No.	Description	Array size	Maximum k_{eff}
1	17x17 5% enriched ^{235}U assemblies	See Section 5.3.2 of Appendix 2.5-1	0.9 < x < 1
2	15x15 5% enriched ^{235}U assemblies	See Section 5.3.2 of Appendix 2.5-1	
3, 4	14x14 5% enriched ^{235}U assemblies	Infinite	0.8 < x < 0.9
5,6, 7, 8	Loose pellets	Infinite	

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, the applicant notes that the statistical uncertainty for all criticality safety calculations of the FCC-3 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 or V0.2) 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-3 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-3 criticality safety analyses adequately demonstrate that the package is subcritical.

6.12 Evaluation Findings

The staff finds that the French Certificate for the FFC3 conforms the criticality safety regulations within these regulations: 501(c), 526, 673, 682, 684, 685, 686, 716, 814, 815, and 816 of the IAEA SSR-6, 2012 Edition. Based on the statements in the safety evaluation above, the staff recommends revalidation of French Certificate of Competent Authority F/437/AF-96, Revision Fs, for the FCC-3 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/347/AF-96, Revision Fs, Model No. FCC-3 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 proposed changes to the package design meet the requirements of the IAEA SSR-6, 2012 Edition. The staff reviewed the description of the QA program for the Model No. FCC-3 package against the standards in the IAEA SSR-6, 2012 Edition.

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, 2012 Edition. 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-3 transportation package.

The staff finds, with reasonable assurance, that the QA program for the FCC-3 transportation packaging meets the requirements in IAEA SSR-6, 2012 Edition 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-3 package application and as discussed in this SER section, the staff has reasonable assurance that the FCC-3 package meets the requirements in IAEA SSR-6, 2012 Edition. The staff recommends revalidation of French Competent Authority Certificate of Approval F/347/AF-96, Revision Fs.

CONDITIONS

The staff recommends the revalidation of French Competent Authority Certificate of Approval F/347/AF-96, Revision Fs, for the Model No. FCC-3 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-3 package meet the requirements of IAEA SSR-6, 2012 Edition.

Issued with letter to R. Boyle, U. S. Department of Transportation,
on February 12, 2021.