August 9, 2010

Mr. Richard W. Boyle, Chief
Radioactive Materials Branch
Office of Hazardous Materials
Technology
U.S. Department of Transportation
1200 New Jersey Ave., S.E.
Washington, D.C. 20590

SUBJECT: REVALIDATION OF THE FRENCH CERTIFICATE OF APPROVAL NO. F/347/AF-96, REVISION CI, FOR THE MODEL NO. FCC-3 PACKAGE

Dear Mr. Boyle:

This is in response to your letter dated November 18, 2009, requesting our assistance in evaluating the Model No. FCC-3 package, authorized by the French Certificate of Approval No. F/347/AF-96, Revision Ci. On June 9, 2010, you provided the additional information that we requested on April 1, 2010, to complete our detailed technical review.

Based upon our review, the statements and representations contained in the Safety Analysis Report referenced TFX DC 2159 Rev. D and Rev. E, as supplemented, and for the reasons stated in the enclosed Safety Evaluation Report, we recommend revalidation of the French Certificate of Approval No. F/347/AF-96, Rev. Ci, for the Model No. FCC-3 package, limited to the following contents: two PWR 17x17 12 ft fresh fuel assemblies as described in Appendices 1, 5, and 9 of the French Certificate.

If you have any questions regarding this matter, please contact me or Pierre Saverot of my staff at (301) 492-3408.

Sincerely,

/RA/

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3083
TAC No. L24394

Enclosure: Safety Evaluation Report

Mr. Richard W. Boyle, Chief
Radioactive Materials Branch
Office of Hazardous Materials
Technology
U.S. Department of Transportation
1200 New Jersey Ave., S.E.
Washington, D.C. 20590

SUBJECT: REVALIDATION OF THE FRENCH CERTIFICATE OF APPROVAL NO. F/347/AF-96, REVISION CI, FOR THE MODEL NO. FCC-3 PACKAGE

Dear Mr. Boyle:

This is in response to your letter dated November 18, 2009, requesting our assistance in evaluating the Model No. FCC-3 package, authorized by the French Certificate of Approval No. F/347/AF-96, Revision Ci. On June 9, 2010, you provided the additional information that we requested on April 1, 2010, to complete our detailed technical review.

Based upon our review, the statements and representations contained in the Safety Analysis Report referenced TFX DC 2159 Rev. D and Rev. E, as supplemented, and for the reasons stated in the enclosed Safety Evaluation Report, we recommend revalidation of the French Certificate of Approval No. F/347/AF-96, Rev. Ci, for the Model No. FCC-3 package, limited to the following contents: two PWR 17x17 12 ft fresh fuel assemblies as described in Appendices 1, 5, and 9 of the French Certificate.

If you have any questions regarding this matter, please contact me or Pierre Saverot of my staff at (301) 492-3408.

Sincerely,

/RA/
Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3083
TAC No. L24394

Enclosure: Safety Evaluation Report
Distribution:
SFST r/f G:/SFST/Saverot/71-3083 AREVA FCC-3/Final Revalidation of the French Certificate of Approval for the FCC-3 package **ADAMS No.: ML102220064**

OFC	SFST	E	SFST		SFST		SFST		SFST		SFST	
NAME	PSaverot		JChang		ABarto		BTripathi		DTarantino		DJackson	
DATE	07/28/10		08/04/10		08/04/10		08/04/10		08/04/10		08/04/10	
OFC	SFST		SFST		SFST		SFST		SFST		SFST	
NAME	MRahimi		CRegan		MDeBose		EJBenner					
DATE	08/04/10		08/04/10		08/06/2010		08/09/2010					

C = COVER **E = COVER & ENCLOSURE** **N = NO COPY** **OFFICIAL RECORD COPY**

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

SUMMARY

By letter dated November 18, 2009, the U.S. Department of Transportation (DOT) requested that the Nuclear Regulatory Commission (NRC) staff review the Model No. FCC-3 package, as authorized by the French Certificate of Approval No. F/347/AF-96, Revision Ci. DOT requested that the staff provide a recommendation concerning the revalidation of the certificate for import and export use. DOT provided the Safety Analysis Report (SAR) referenced TFX DC 2159, and supplemental information, further to a request for additional information dated April 1, 2010.

The staff evaluated the Model No. FCC-3 package design against the standards in the International Atomic Energy Agency (IAEA) “Regulations for the Safe Transport of Radioactive Material,” Safety Standards Series No. TS-R-1, 2005 Edition. Based upon the statements and representations contained in the SAR and supplemental information, and for the reasons stated below, the staff recommends that DOT revalidate the French Certificate of Approval No. F/347/AF-96, Rev. Ci, limited to the following contents: two PWR 17x17 12 ft fresh fuel assemblies as described in Appendices 1, 5, and 9 of the French Certificate.

1.0 GENERAL INFORMATION

The Model No. FCC-3 packaging has a length of 4,923 mm (+/- 8 mm), an outside diameter of 1,048 mm (+/- 4 mm) and a maximum loaded weight of 4,300 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 22 rubber shock mountings, and an internal system (frame and doors) leaving space for two cavities (filled with a polymer composite resin – glass fiber and alumina) to accommodate two pressurized water reactor (PWR) 17x17 12 ft fresh fuel assemblies.

The authorized contents are limited to two unirradiated PWR 17x17 12 ft UO_2 fuel assemblies with the following specifications (multiple dimensions are listed in Appendices 1, 5, and 9):

Maximum mass heavy metal per assembly:	521 kg
Maximum number of fuel rods:	288
Cladding material:	Zirconium alloy
Maximum initial enrichment:	5.0 weight percent
Nominal active length:	365.8 cm
Nominal rod pitch:	1.26 cm
Minimum rod diameter:	0.946 cm / 0.940 cm
Minimum clad thickness:	0.052 cm
Maximum pellet diameter:	0.82 cm / 0.83 cm

2.0 STRUCTURAL EVALUATION

A summary of the applicable TS-R-1 requirements related to the structural performance of the package, and the ability of the package design to meet such requirements, is provided below.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
606	*Package* mass, volume and shape shall be such that it can be easily and safely transported. In addition, the *package* needs to be properly secured in or on the *conveyance* during transport.	Requirement met. See SAR in Ref. TFX DC 2159 Rev. E.
607	Any lifting attachments on the *package* will not fail when used in the intended manner and, if failure of the attachments should occur, the ability of the *package* to meet other requirements of these Regulations would not be impaired. Take account of appropriate safety factors to cover snatch lifting.	Requirement met. See Appendix 6 of the SAR.
608	Attachments and any other features on the outer surface of the *package* which could be used to lift it, shall be designed either to support its mass in accordance with the requirements of Para. 607 or shall be removable or otherwise rendered incapable of being used during transport.	Requirement met. See SAR Section 5.2.3.
609	As far as practicable, the *packaging* shall be so designed and finished that the external surfaces are free from protruding features and can be easily decontaminated.	Requirement met. See SAR Section 5.2.4.
610	As far as practicable, the outer layer of the *package* shall be so designed as to prevent the collection and the retention of water.	Requirement met. Outer enclosure of the packaging is cylindrical.
612	The package shall withstand the effects of any acceleration, vibration or vibration resonance which 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.	Requirement met. See SAR Section 5.2.7.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
613	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.	Requirement met. As the internal equipment system is exclusively stainless steel, there is no risk of corrosion.
614	All valves through which the *radioactive contents* could otherwise escape shall be protected against unauthorized operation.	Requirement met. The valve is protected by a bolted plate, prohibiting any unauthorized manipulation.
615	The design of the *package* shall take into account ambient temperatures and pressures that are likely to be encountered in routine conditions of transport.	Requirement met.
617, 618, 619	For *packages* to be transported by air…..	N/A
633	*Type A packages* shall be designed to meet the requirements specified in Paras. 606–616 and, in addition, the requirements of Paras. 617–619 if carried by air, and of Paras. 634–649.	Requirement met - as applicable. See SAR.
634	The smallest overall external dimension of the *package* shall not be less than10 cm.	Requirement met. Smallest dimension is 113 cm. See SAR Section 5.3.1.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
635	The outside of the *package* shall incorporate a feature such as a seal, which is not readily breakable and which, while intact, will be evidence that it has not been opened.	Requirement met. Security seals are provided. See SAR.
636	Any tie-down attachments on the *package* shall be so designed that, under normal and accident conditions of transport, the forces in those attachments shall not impair the ability of the *package* to meet the requirements of these Regulations.	Requirement met. See Appendix 12 of SAR.
637	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.	Requirement met. See Para. 5.4.4 of SAR.
638	The *design* and manufacturing techniques shall be in accordance with national or international standards, or other requirements, acceptable to the *competent authority*.	Requirement met. See Table 3-1 of SAR.
639	The *design* shall include a *containment system* securely closed by a positive fastening device which cannot be opened unintentionally or by a pressure which may arise within the *package*.	Requirement met. See Para 4.1.1 of SAR.
641	If the *containment system* forms a separate unit of the *package*, it shall be capable of being securely closed by a positive fastening device which is independent of any other part of the *packaging*.	Containment ensured by rod cladding.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
642	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.	The fuel rods contain only UO_2 pellets and inert gas. They do not contain liquids or other materials generating radiolysis.
643	The *containment system* shall retain its *radioactive contents* under a reduction of ambient pressure to 60 kPa.	Requirement met. Refer to FFDC01091 Rev. B.
644	All valves, other than pressure relief valves, shall be provided with an enclosure to retain any leakage from the valve.	Lower shell valve is a pressure-relief valve protected by a bolted plate.
645	A radiation shield which encloses a component of the *package* specified as a part of the *containment system* shall be so designed as to prevent the unintentional release of that component from the shield.	Requirement met by the resin included in the frames and doors.
646	A *package* shall be so designed that if it were subjected to the tests specified in Paras. 719–724, it would prevent: (a) Loss or dispersal of the *radioactive contents*; and (b) Loss of shielding integrity which would result in more than a 20% increase in the *radiation level* at any external surface of the *package*.	Requirement met. See Ref. FFDC05067 Rev. A.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
675	The packaging, after being subject to the tests specified in Paras. 719-724, must prevent the entry of a 10 cm cube.	The analysis in Section No. 5.5.2 of the SAR shows that construction would prevent the entry of a 10 cm cube.
701	Demonstration of compliance with the performance standards required in Section VI shall be accomplished by any of the methods listed below or by a combination thereof.	Requirement met.
702	After the specimen, prototype or sample has been subjected to the tests, appropriate methods of assessment shall be used to ensure that the requirements of this section have been fulfilled in compliance with the performance and acceptance standards prescribed in Section VI.	Requirement met.
704	Specimens that comprise or simulate *special form radioactive material* shall be subjected to the impact test, the percussion test, the bending test, and the heat test specified in Paras. 705–709. A different specimen may be used for each of the tests. For further details see TS-R-1.	Requirement met.
705	Impact test: The specimen shall drop onto the target from a height of 9 m. The target shall be as defined in Para. 717.	Requirement met.
716	After each of the applicable tests specified in Paras. 718–737: (a) Faults and damage shall be identified and recorded; (b) It shall be determined whether the integrity of the *containment system* and shielding has been retained to the extent required in Section VI for the *package* under test; and (c) For *packages* containing *fissile material*, it shall be determined whether the assumptions and conditions used in the assessments required by Paras. 671–682 for one or more *packages* are valid.	Requirements met. See SAR.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
717	The target for the drop test specified in Paras. 705, 722, 725(a), 727, and 735 shall be a flat, horizontal surface of such a character that any increase in its resistance to displacement or deformation upon impact by the specimen would not significantly increase damage to the specimen.	Requirements met. See SAR.
719	The tests are: the water spray test, the free drop test, the stacking test and the penetration test. Specimens of the *package* shall be subjected to the free drop test, the stacking test and the penetration test, preceded in each case by the water spray test. One specimen may be used for all the tests, provided that the requirements of Para. 720 are fulfilled.	Requirements met as applicable. See SAR.
722 (a), (b), (c)	Free drop test: The specimen shall drop onto the target so as to suffer maximum damage in respect of the safety features to be tested. *For details see TS-R-1.*	Requirement met. See SAR.
723	Stacking test: *For details see TS-R-1.*	Requirement met. See Section 5.5.2.3.
724	Penetration test: *For details see TS-R-1.*	See Section 5.5.2.4.
726	The specimen shall be subjected to the cumulative effects of the tests specified in Para. 727 and Para. 728, in that order. Following these tests, either this specimen or a separate specimen shall be subjected to the effect(s) of the water immersion test(s) as specified in Para. 729 and, if applicable, Para. 730.	Requirements met. See SAR Section 5.5.3. Response to RAI No. 3 is acceptable.
727	Mechanical test: The mechanical test consists of three different drop tests. Each specimen shall be subjected to the applicable drops as specified in Para. 656 or Para. 682. For details of Drop I, II, and III see TS-R-1.	Requirements met.
735	Puncture/tearing test: The specimen shall be subjected to the damaging effects of a solid probe made of mild steel. The orientation of the probe to the surface of the specimen shall be such as to cause maximum damage at the conclusion of the test sequence specified in Para. 734(a). See TS-R-1 for details.	N/A
737	Impact test: The specimen shall be subject to an impact on a target at a velocity of not less than 90 m/s, at such an orientation as to suffer maximum damage. The target shall be as defined in Para. 717, except that the target surface may be at any orientation as long as the surface is normal to the specimen path.	N/A

Additional Information Related to the Structural Evaluation

The Model No. FCC-3 packaging differs from the Model No. FCC-4 packaging only in its geometrical characteristics, e.g., length, weight, number of cradle cross-pieces and number of shock mounts. Both models have an identical design, except for handling points on the lower shell and the compliance of the Model No. FCC-3 package with IAEA requirements was demonstrated by subjecting a full-scale Model No. FCC-4 package to the regulatory mechanical tests (Article 701). The results of these tests are included in Appendix No. 11 to the SAR. No additional tests were performed for the Model No. FCC-3 package, and its behavior was analyzed by comparison with the Model No. FCC-4 package test results.

The test was performed with a fuel assembly with an inert supporting structure and a cladding made of zirconium alloy Zr4. Other zirconium alloy grades may also be used for the cladding material, as stated in Appendix 11 of the SAR. The results of the test on the Model No. FCC-4 package showed that (i) the two shells were still connected, (ii) the shock absorbers were still in place and undamaged, (iii) the relative positions and spacing of the assemblies within the packaging were unchanged, (iv) no rod failure was observed.

However, the frame-door assembly sustained damages from the combination of normal conditions of transport (NCT) and hypothetical accident conditions (HAC) tests which induced some modifications in the package geometry as well as in the position of the contents inside the cavity, i.e., a perforation of the enclosure with tears on the doors, with an overall reduction of the section of the cavity, but the resin's thickness was unchanged.

It was concluded that the package internals continued to fulfill their functions as (i) all the door, frame, top and bottom plate connections remained intact and (ii) the modifications in the package geometry were compatible with the assumptions taken for the criticality studies in Section No. 5.4 and Appendix No. 12 of the SAR.

Regarding the bolted connection strength, the applicant demonstrates that the behavior of the connection for the Model No. FCC-4 is at least equal to that of the flange of the Model No. FCC-3 package because, in identical conditions, the shear stress level in the Model No. FCC-4 bolted connection yields a 28% margin compared to that of the Model No. FCC-3, whose behavior was validated by a drop test. The risk of shell separation can therefore be ruled out.

3.0 MATERIALS EVALUATION

A summary of the applicable TS-R-1 requirements related to the materials performance of the package, and the ability of the package design to meet such requirements, is provided below.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
501 (b) 501(c)	Requirement before 1st shipment for packages containing fissile material, where, in order to comply with the requirements of paragraph 671, neutron poisons are specifically included as components of the package, checks shall be performed to confirm the presence and distribution of those neutron poisons.	This package includes neutron absorber resin. See response below.
507, 613, 642	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. 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.	See response below.
615, 618, 637, 653, 664, 651	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.	See response below.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
651(a)	Alter the arrangement, the geometrical form or the physical state of the radioactive contents or, if the radioactive material is enclosed in a can or receptacle (for example, clad fuel elements), cause the can, receptacle or radioactive material to deform or melt;	The results of destructive testing on prototype packaging are in Section 5.5 of the application TFX DC 2159 E0. The packaging is for transporting un-irradiated fresh fuel and the fuel assemblies are described in Section 2.2 of application TFX DC 2159 E0. The confinement of the fuel satisfies requirements per Section 4.1.1 of the application TFX DC 2159 E0.
651(b)	Lessen the efficiency of the packaging through differential thermal expansion or cracking or melting of the radiation shielding material;	The cask internal equipment system is constructed of similar stainless steel which houses the fuel assemblies and/or fuel rod channels. Thermal expansion requirements are satisfied.
733	The specimen shall be immersed under a head of water of at least 0.9-m for a period of not less than eight hours and in the attitude for which maximum leakage is expected.	The applicant's criticality analysis assumes water in-leakage into the package. Therefore, the result of this test does not impact the criticality evaluation.
806(a)	A package design for fissile material, which is also subject to paragraphs 812-814, shall require multilateral approval; and	This paragraph does not describe a requirement that is applied as part of the review of the criticality safety design of a package.
807(a), 807(f)	A detailed description of the proposed radioactive contents with reference to their physical and chemical states and the nature of the radiation emitted;…..	Detailed descriptions of the allowable contents and limitations on the contents are provided in Section 2 of the application TFX DC 2159 E0.
807(b)	A detailed statement of the design, including complete engineering drawings and schedules of materials and methods of manufacture;…..	The SAR provides detail drawings, manufacturing sequence, and material in Appendix 0 of the certificate. Drawing requirements are satisfied.

Response to Criterion 501(c)

The applicant subjected resin samples to various fire tests to determine the loss of hydrogen (4.65 wt.-%) and boron (2.1 wt.-%) which are responsible for the neutron absorption capacity of the resin. The thermal behavior of the neutron absorber resin was provided and the thermal tests performed on resin samples were described along with their corresponding damages. The staff finds that the non-standardized tests performed are reasonably appropriate and the conditions are conservative bounding values.

Response to Criteria 507, 613, and 642

The applicant provided information that there are no explosive, flammable, pyrophoric, chemically toxic, or corrosive materials in this package constructed of austenitic stainless steel. The zirconium (Zr) fuel elements are bulk Zr which is resistant to pyrophoric reactions. The absence of an electrolyte in the containment removes any potential for galvanic production of hydrogen, or chemical decomposition generated gases. For carbon steel parts (shell and cradle), the inspection program includes monitoring of the condition of the paint-coating per Sections 5.2.8 and 5.2.10 of the application referneced TFX DC 2159 E0. The staff finds that the material compatibility satisfies requirements.

Response to Criteria 615, 618, 637, 653, 664, and 651

The applicant stated the effects of brittle transition at a temperature of -40°C on the internal equipment system defined as safety components fabricated from austenitic stainless steel showed no brittle fracture. A design temperature upper limit of +38°C is acceptable for all packaging materials (steels and rubber) per Section 5.4.4 of the application TFX DC 2159 E0 (TS-R-1, 676). There are no liquids associated with this package, thus no effects from freezing. The staff finds that the materials performance under the temperature ranges satisfy the requirements of no brittle fracture or thermally induced degradation.

4.0 THERMAL EVALUATION

A summary of the applicable TS-R-1 requirements related to the thermal performance of the package, and the ability of the package design to meet such requirements, is provided below.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
501	Sets requirements that must be fulfilled before the first shipment of any package.	Requirement met.
501(b)	For …..each package containing fissile material, it shall be ensured that the effectiveness of its shielding and containment and, where necessary, the heat transfer characteristics and the effectiveness of the confinement system, are within the limits applicable to or specified for the approved design.	Requirement met. (thermal analyses are conservative).
502	Sets requirements that must be fulfilled prior to each shipment of any package.	Requirement met.
502(a)	For any package it shall be ensured that all the requirements specified in the relevant provisions of these Regulations have been satisfied.	Requirement met.
502(c)	For …….each package containing fissile material, it shall be ensured that all the requirements specified in the approval certificates have been satisfied.	Requirements met.
507	Sets the requirements of other dangerous properties of the package contents, such as explosiveness, flammability, pyrophoricity, chemical toxicity and corrosiveness…..	Requirement met.
637	The design of the package shall take into account temperatures ranging from -40ºC to +70ºC 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.	Requirements met.
642	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.	Requirement met. (No gas is generated by radiolysis).

TS-R-1	TS-R-1 SUMMARY	EVALUATION
651(a)	Alter the arrangement, the geometrical form or the physical state of the radioactive contents or, if the radioactive material is enclosed in a can or receptacle (for example, clad fuel elements), cause the can, receptacle or radioactive material to deform or melt;	N/A
651(b)	Lessen the efficiency of the packaging through differential thermal expansion or cracking or melting of the radiation shielding material;	N/A
652	A package shall be so designed that the temperature of the accessible surfaces of a package shall not exceed 50ºC, unless the package is transported under exclusive use.	N/A
653	The ambient temperature shall be assumed to be 38ºC.	N/A
654	The solar insulation conditions shall be assumed to be as specified in Table XI.	N/A
655	Requires that for a package which includes thermal protection in order to satisfy the 30 minute thermal test, the protection on the exterior of the package shall not be rendered ineffective by ripping, cutting, skidding, abrasion, or rough handling.	N/A
676	A package for fissile material shall be designed for an ambient temperature range of -40ºC to +38ºC.	Requirement met. See Section 5.4.4 of SAR
728(a)	Exposure of a specimen for a period of 30 minutes to a thermal environment which provides a heat flux at least equivalent to that of a hydrocarbon fuel/air fire in sufficiently quiescent ambient conditions to give a minimum average flame emissivity coefficient of 0.9 and an average temperature of at least 800ºC, fully engulfing the specimen, with a surface absorptivity coefficient of 0.8 or that value which the package may be demonstrated to possess if exposed to the fire specified, followed by:	Requirements met.
728(b)	Exposure of the specimen to an ambient temperature of 38ºC, subject to the solar insolation conditions specified in Table XI and subject to the design maximum rate of internal heat generation within the package by the radioactive contents for a sufficient period to ensure that temperatures in the specimen are everywhere decreasing and/or are approaching initial steady state conditions.	Requirements met. The solar insolation in Table XI is applied in NCT and post-fire cooldown conditions.
736	Enhanced thermal test: The conditions for this test shall be as specified in para. 728, except that the exposure to the thermal environment shall be for a period of 60 minutes.	N/A. No need for a 60-min. thermal test.
807(a)	A detailed description of the proposed radioactive contents with reference to their physical and chemical states and the nature of the radiation emitted;….	Requirement met. (Fresh UO_2 fuel)

Additional Information Related to the Thermal Evaluation

After completion of the structural tests, the applicant observed (i) a reduction of the clearance between the cladding of the fuel assembly, (ii) tears between the doors and the frame, and (iii) a perforation in the clamshell which could lead to the ingress of the flames as far as the internals of the package.

Therefore, the thermal tests evaluated the following: (1) cladding rupture because of visco-plastic creep due to the cumulated effect of temperature and internal pressure, and (2) resin performance.

The applicant performed both 2-D and 3-D analyses of the Model No. FCC-4 package with PWR 17x17 fuel rods to verify the thermal design of the Model No. FCC-3 package during NCT and HAC conditions of transport, in compliance with IAEA TS-R-1 requirements. The staff compared the package configurations and other items such as fuel rod pitch, fuel rod diameter, cladding thickness, cavity clearance and size, length of fuel rod, amounts of UO_2, and decay heat, and agreed that the Model No FCC-4 package can indeed be used to validate the thermal performance of the Model No. FCC-3 package.

The parameters and physical phenomena used by the applicant under NCT and HAC conditions are summarized below:

<u>NCT</u>

1) The thermal properties of the materials, e.g., Zircaloy, resin, stainless steel, helium and air are provided and their physical properties (thermal conductivity, air viscosity, density) are varying with temperature;
2) The helium layer separating the fuel pellets from their cladding is represented by a thermal resistance of $4x10^{-4} \sim 5x10^{-4}$ m^2K/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 ignored;
3) The initial temperatures of air and all components are conservatively set at 78ºC and the solar insolation conditions are in compliance with Table XI of IAEA article 728; and
4) The clamshell is cooled by natural convection and thermal radiation under the ambient air temperature of 38ºC of NCT.

<u>HAC</u>

It appears that the thermal loading on the fuel rods during the thermal test is the convection of the hot air entering the cavity of the internals through the frame/door clearance. The hot air enters the cavity and heats a certain number of fuel rods:

1) 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 a temperature of 800ºC during 30 minutes;
2) The clamshell is assumed to be present during the post-fire cooldown to minimize heat losses. The clamshell is cooled by natural convection and thermal radiation to the ambient air temperature of 38ºC, during the post-fire cooldown;
3) The package is rotated by 45º to lead to the most unfavorable position under HAC. The inclination of 45º maximizes the driving head (pressure), induced by buoyancy effects, to increase heat input onto the fuel rods;

4) The resin contained in the door and the frame is considered to be an inert, stable material, with unchanged thermal properties. The increase in the insulation capability of the resin, caused by the surface calcination effects and the endothermic reaction, is conservatively not taken into account;

5) The cladding outer face emissivity is modeled as 0.6 by taking into account 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 staff reviewed the descriptions and the model calculations of the Model No. FCC-4 package and evaluated the thermal performance of the Model No. FCC-3 package through its own confirmatory analyses. The applicant predicted that the maximum and the mean temperatures of the most stressed cladding are 642.8ºC and 624.2ºC, respectively, which correspond to a cumulated elongation of 14% that is much below the ultimate elongation (rupture) limit of 40% for the Zircalloy (Zr4) cladding. The resin, well contained within the stainless steel, has a temperature distribution ranging from 129.4ºC to 795.4ºC in the fire test, and is acceptable from the thermal perspective. The staff reviewed and validated the temperature distributions and determined that the thermal evaluation is acceptable for both models because the thermal analyses are very conservative: 1) the clamshell is assumed to be lost and therefore the fire gas ingresses into the package internals during the 30-minute fire phase, (2) the clamshell is back in position to minimize heat losses from the package during the post-fire cooldown, (3) the heat conduction through the resin to the fuel assembly is not reduced by assuming that the consumed resin is not replaced by air, and (4) all components, including fuel rods and fuel claddings, are assumed to have a high initial temperature of 78ºC.

5.0 CRITICALITY SAFETY EVALUATION

A summary of the applicable TS-R-1 requirements related to the criticality safety performance of the package, and the ability of the package design to meet such requirements, is provided below.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
501 (b)	For each …package containing fissile material, it shall be ensured that the effectiveness of its shielding and containment and, where necessary, the heat transfer characteristics and the effectiveness of the confinement system, are within the limits applicable to or specified for the approved design.	Characteristics related to shielding, containment, and heat transfer are inspected prior to first shipment according to Sections 4.1 and Table 7.1 of AREVA document TFX-DC-2159, "FCC3 SHIPPING CONTAINER in compliance with the IAEA Regulations regarding the Transportation of Radioactive Materials (2005 Edition)"
501 (c)	For packages containing fissile material, where, in order to comply with the requirements of para. 671, neutron poisons are specifically included as components of the package, checks shall be performed to confirm the presence and distribution of those neutron poisons.	Neutron absorbing resin in fuel assembly frame and doors inspected per Table 7.1 of AREVA document TFX-DC-2159
502(a)	For any package it shall be ensured	Relevant provisions satisfied by design,

TS-R-1	TS-R-1 SUMMARY	EVALUATION
	that all the requirements specified in the relevant provisions of these Regulations have been satisfied.	analyses, and procedures specified in AREVA document TFX-DC-2159.
502(c)	For each …package containing fissile material, it shall be ensured that all the requirements specified in the approval certificates have been satisfied.	Requirements in approval certificate are verified during the loading operations, as described in Sections 4.2 and 8 of AREVA document TFX-DC-2159.
502(g)	For packages containing fissile material, measurements of isotopic composition (if burnup credit is allowed) and tests of the closure of the package (if special features are used to avoid in-leakage of water) shall be performed.	N/A – package design does not credit fuel burnup or special features to avoid in-leakage of water.
528	The CSI for packages containing fissile material shall be obtained by dividing the number 50 by the smaller of the two values of N derived in paras. 681 and 682. The value of the CSI may be zero, provided that an unlimited number of packages is subcritical.	Criticality evaluations in the applicant's safety analysis demonstrate that an infinite array of undamaged packages under normal conditions of transport, or a 10x16x1 array (160 packages) of damaged packages under hypothetical accident conditions, are adequately subcritical. The limiting value of "N" is therefore 80, resulting in a CSI of 50/N = 0.63.
529	The CSI for each consignment shall be determined as the sum of the CSIs of all the packages contained in that consignment.	Consignment CSI verified by consignor/shipper.
530	The transport index of any package or overpack shall not exceed 10, nor shall the CSI of any package or overpack exceed 50 except for consignments under exclusive use.	Maximum CSI for the package is 0.63.
671	*Fissile material* shall be transported so as to:	-
671(a)	maintain subcriticality during normal and accident conditions of transport; in particular, the following contingencies shall be considered: water leaking into or out of packages; the loss of efficiency of built-in neutron absorbers or moderators; rearrangement of the contents either within the package or as a result of loss from the package; reduction of spaces within or between packages; packages becoming immersed in water or buried in snow; and temperature changes; and	Applicant evaluated individual packages and arrays of FCC3 packages under normal and hypothetical accident conditions of transport. The accident conditions model considered reduction in package spacing due to the regulatory drop and puncture tests, water in-leakage – including preferential flooding of the fuel channels and varying moderator density, fuel assembly pitch expansion due to a hypothetical end drop, and reduction in efficiency of the neutron absorber resin due to the effects of the thermal test. Individual packages and arrays of packages remained

TS-R-1	TS-R-1 SUMMARY	EVALUATION
		adequately subcritical under all conditions.
671(b)	meet the requirements: of para. 634 for fissile material contained in packages; prescribed elsewhere in these Regulations which pertain to the radioactive properties of the material; and specified in paras. 673-682, unless excepted by para. 672.	Smallest external dimension of the package is larger than 10 cm; package meets regulations pertaining to the radioactive properties of the contents; paras. 673 – 682 discussed in the following:
673	Contains requirements for fissile material for which the chemical or physical form, isotopic composition, mass or concentration, moderation ratio or density, or geometric configuration is not known.	N/A – fresh fuel chemical and physical form, isotopic composition, mass, concentration, and geometric configuration are well known.
674	For irradiated nuclear fuel the assessment of paras. 677-682 shall be based on an isotopic composition demonstrated to provide (a) the maximum neutron multiplication during the irradiation history, or (b) a conservative estimate of the neutron multiplication for the package assessments. After irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotopic composition.	N/A – contents are fresh fuel assemblies.
676	A package for fissile material shall be designed for an ambient temperature range of -40°C to +38°C.	Package is evaluated for this temperature range in Section 5.4.4 of AREVA document TFX-DC-2159.
677	For a package in isolation, it shall be assumed that water can leak into or out of all void spaces of the package, including those within the containment system. However, if the design incorporates, special features to prevent such leakage of water into or out of certain void spaces, even as a result of error, absence of leakage may be assumed in respect of those void spaces. Special features shall include the following:.	Package is evaluated for criticality safety assuming water in-leakage.
677(a)	Multiple high standard water barriers, each of which would remain watertight if the package were subject to the tests prescribed in para. 682(b), a high degree of quality control in the manufacture, maintenance and repair of packagings and tests to	N/A – see above.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
	demonstrate the closure of each package before each shipment;	
678	It shall be assumed that the confinement system shall be closely reflected by at least 20 cm of water or such greater reflection as may additionally be provided by the surrounding material of the packaging. However, when it can be demonstrated that the confinement system remains within the packaging following the tests prescribed in para. 682(b), close reflection of the package by at least 20 cm of water may be assumed in para 679(c).	Individual package is modeled with at least 20 cm of water external to the package outer shell.
679	The package shall be subcritical under the conditions of paras. 677 and 678 with the package conditions that result in the maximum neutron multiplication consistent with: (a) routine conditions of transport (incident free); (b) the tests specified in para. 681(b); (c), the test specified in para. 682(b)	The applicant has demonstrated that the maximum reactivity associated with the package under routine conditions of transport or hypothetical accident conditions, is adequately subcritical, per the criticality described in Appendix 12 of AREVA document TFX-DC-2159.
680	For packages to be transported by air: (a) the package shall be subcritical under conditions consistent with the tests prescribed in para. 734 assuming reflection by at least 20 cm of water but no water inleakage; and (b) allowance shall not be made for special features of para. 677 unless, following the tests specified in para. 734 and, subsequently, para. 733, leakage of water into or out of the void spaces is prevented.	The package has not been evaluated for transport by air.
681	A number "N" shall be derived, such that five times "N" shall be subcritical for the arrangement and package conditions that provide the maximum neutron multiplication consistent with the following…	The applicant's analysis of arrays of packages under normal conditions of transport demonstrates that an infinite array of packages remain subcritical.
681(a)	There shall not be anything between the packages, and the package arrangement shall be reflected on all sides by at least 20 cm of water, and..	The package array is modeled considering nothing between packages, and with an infinite number of packages.
681(b)	The state of the packages shall be their assessed or demonstrated condition if they had been subjected	The package under normal conditions of transport is not substantially altered from the initial design, and is modeled as such in

TS-R-1	TS-R-1 SUMMARY	EVALUATION
	to the test specified in paras. 719-724.	the criticality analysis.
682	A number "N" shall be derived, such that two times "N" shall be subcritical for the arrangement and package conditions that provide the maximum neutron multiplication consistent with the following:	The applicant has demonstrated that a 10x16x1 array (160 packages) of damaged packages under hypothetical accident conditions, are adequately subcritical. The limiting value of "N" is therefore 80, resulting in a CSI of 50/N = 0.63.
682(a)	Hydrogenous moderation between packages, and the package arrangement reflected on all sides by at least 20 cm of water; and	The criticality analysis under hypothetical accident conditions considers varying water density between packages, and a 20 cm water reflector external to the array.
682(b)	The tests specified in paras. 719-724 followed by whichever of the following if the more limiting: the tests specified in para. 727(b) and, either para. 727(c) for packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m^{33} based on the external dimensions, or para. 727(a) for all other packages; followed by the test specified in para. 728 and completed by the tests specified in paras. 731-733; or the test specified in para. 729; and	The criticality analysis under hypothetical accident conditions considers the results of the tests specified in Paras. 719 – 724, 726 – 729, and 731-733. The most significant changes due to these tests are the reduction in the package spacing in the array due to the drop and puncture tests, rod pitch expansion in part of the assembly length due to the end drop, and reduction in thickness and boron concentration in the neutron absorber due to the thermal test.
682(c)	Where any part of the fissile material escapes from the containment system following the tests specified in para. 682(b), it shall be assumed that fissile material escapes from each package in the array and all of the fissile material shall be arranged in the configuration and moderation that results in the maximum neutron multiplication with close reflection by at least 20 cm of water.	Fissile material does not escape the containment system under the tests specified in para. 682(b).
731	Packages for which water in-leakage or out-leakage to the extent which results in greatest reactivity has been assumed for purposes of assessment under paras. 677-682 shall be excepted from the test.	Water in-leakage is assumed in the criticality analysis for the package, therefore this test is not performed.
733	The specimen shall be immersed under a head of water of a least 0.9 m for a period of not less than eight hours and in the attitude for which maximum leakage is expected.	N/A – see above.
807 (a)	A detailed description of the proposed radioactive contents with reference to their physical and chemical states and	Detailed descriptions of the allowable contents and limitations on the contents are in the CoC.

TS-R-1	TS-R-1 SUMMARY	EVALUATION
	the nature of the radiation emitted;	
807 (b)	A detailed statement of the design, including complete engineering drawings and schedules of materials and methods of manufacture;	This information is included in AREVA document TFX-DC-2159.
807 (c)	A statement of the tests which have been done and their results, or evidence based on calculative methods or other evidence that the design is adequate to meet the applicable requirements;	Section 5.5 of AREVA document TFX-DC-2159, describes the results of the regulatory tests performed on the package.
807 (d)	The proposed operating and maintenance instructions for the use of the packaging;	Section 8 of AREVA document TFX-DC-2159 describes the operation of the package.
807 (f)	Wwhere the proposed radioactive contents are irradiated fuel, the applicant shall state and justify any assumption in the safety analysis relating to the characteristics of the fuel and describe any preshipment measurement required by para. 674(b);	N/A – package contents are unirradiated PWR fuel assemblies.
810 (b)	Any proposed supplementary operational controls to be applied during transport not regularly provided for in these Regulations, but which are necessary to ensure the safety of the package or to compensate for the deficiencies listed in (a) above;	N/A
810 (c)	A statement relative to any restrictions on the mode of transport and to any special loading, carriage, unloading or handling procedures;	Loading and unloading procedures are described in Section 8 of AREVA document TFX-DC-2159.
813	An application for approval shall include all information necessary to satisfy the competent authority that the design meets the requirements of para. 671, and a specification of the applicable quality assurance program as required in para. 310.	The applicant has provided all the necessary information to demonstrate that the package meets the requirements of para. 671.

Additional Information Related to Para. 671 Criticality Evaluation

The applicant performed all criticality analyses using the CRISTAL V0.2 code package. This code package uses APOLLO 2 version 2.4.3 to generate multi-group cross-sections based on the JEF 2.2 nuclear data, which are then used in the MORET 4 version 4.A.6 3-D Monte Carlo criticality code to determine k_{eff}. This code system has been validated for industry-wide use for fresh fuel evaluations, and is adequate provided the user add an uncertainty of 0.00682 to the calculated k_{eff}. The maximum k_{eff} of 0.937, including biases and uncertainties, was determined by the applicant for a 10 x 16 x 1 array of damaged packages, with preferential flooding of the fuel channels and no interstitial moderation.

The staff performed independent confirmatory evaluations of the FCC3 package with 17 x 17 12 ft fuel assemblies using the CSAS6 sequence of the SCALE 6.0 code package, with the KENO VI 3-D Monte Carlo code and ENDF/B-VII continuous energy cross-sections. Using assumptions similar to the applicant's for arrays of packages under hypothetical accident conditions, the staff's confirmatory calculations resulted in a maximum k_{eff} similar to what was reported in the application.

CONCLUSION

Based on the review of the statements and representations contained in the application, as supplemented, the staff agrees that the Model No. FCC-3 package meets the standards in IAEA Safety Standards Series No. TS-R-1, 2005 edition. The staff recommends that DOT revalidate the French Certificate of Approval No. F/347/AF-96, Revision Ci, for import and export use, limited to the PWR 17x17 12 ft fuel assemblies.

Issued with letter to R. Boyle, Department of Transportation,
On August 9, 2010.