March 14, 2006

Mr. Richard W. Boyle
Radioactive Materials Branch
Office of Hazards Material Technology
U.S. Department of Transportation
400 Seventh Street, S.W.
Washington, DC 20590

SUBJECT: REVALIDATION OF FRENCH CERTIFICATE OF APPROVAL NO. F/379/B(U)F-96 (Aa) FOR THE MODEL NO. TN-106 TRANSPORT PACKAGE

Dear Mr. Boyle:

This is in response to your letter dated December 2, 2004, as supplemented April 8, 2005, and January 9, 2006, requesting our assistance in evaluating the Model No. TN-106 transport package, authorized by French Certificate of Approval No. F/379/B(U)F-96 (Aa).

Based upon our review, the statements and representations contained in the French Safety Analysis Report (5573-Z, Rev. 2), as supplemented, and for the reasons stated in the enclosed Safety Evaluation Report, we recommend revalidation of the French Certificate of Approval No. F/379/B(U)F-96 (Aa), with the following conditions:

Condition No. 1: Authorization is limited to Contents No. 1, Appendix 1a, of the French Certificate with the following additional limitations.

The content is limited to uranium dioxide (UO_2). The maximum masses of UO_2 must comply with the values set out in the table below according to the enrichment in ^{235}U of the most highly enriched fuel element (or element part) present in the cavity and in accordance with the diameter (D) of the internal arrangement designed for criticality purposes:

Enrichment in U-235 (% by mass)	Mass UO_2 (kg) D = 6.0 cm	Mass UO_2 (kg) D = 12.0 cm	Mass UO_2 (kg) D = 20.3 cm
5 < E ≤10	No restriction	No restriction	14.2
4 < E ≤ 5	No restriction	No restriction	53.0
≤4	No restriction	No restriction	No restriction

Contents that include multiple enrichments shall adhere to the limits corresponding to the highest enrichment present. Other contents such as those containing other uranium compounds, plutonium oxides, mixed oxides, solid fuel elements containing uranium, or other solid non fissile radioactive material (except for the internal arrangement or any empty cladding having contained pellets from Content No. 1) are not authorized.

Condition No. 1: (continued) An inert matrices is a general expression that describes that UO_2 is in the presence of material (i.e., metal) under conditions that this material has no influence on the safety analysis (i.e., not a neutron moderator, not a neutron reflector, not radiolysable, not pyrophoric, and not susceptible to decomposing with heat).

Condition No. 2: For transport in the United States, trunnions shall not be used for tie-down attachments. A transport skid that cradles the package shell that is designed to meet the accelerations factors 2g in vertical direction, 5g in lateral direction, and 10g in longitudinal direction shall be used.

Condition No. 3: Transport by air is not authorized.

Condition No. 4: Radiation surveys must include neutron dose rate measurements after loading the contents and prior to transport in the specific areas identified in the French Certificate, Appendix 0a, Section 3, "Conditions for Use of the Packaging."

Condition No. 5: Any sealed capsules must be dry prior to transport.

Condition No. 6: Containment boundary seals shall be tested to show no leakage greater than 1×10^{-7} ref-cm^3/sec within the 12 month period prior to transport. Prior to each shipment, after loading, the package shall show no leakage when tested to a sensitivity of at least 1×10^{-3} ref-cm^3/sec.

Condition No. 7: The cooling time must be a minimum of 4 years.

Sincerely,

/RA/ Stewart Brown for

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3075
TAC No. L23838

Enclosure: Safety Evaluation Report

Condition No. 1: (continued) An inert matrices is a general expression that describes that UO_2 is in the presence of material (i.e., metal) under conditions that this material has no influence on the safety analysis (i.e., not a neutron moderator, not a neutron reflector, not radiolysable, not pyrophoric, and not susceptible to decomposing with heat).

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TAC No. L23838
Enclosure: Safety Evaluation Report
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DATE	3/9/06		3/10/06		3/13/06		3/9/06		3/13/06	

OFC	SFPO	E	SFPO	E	SFPO	E	SFPO		SFPO		SFPO	
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SAFETY EVALUATION REPORT
TN-106 Transport Package
French Certificate No. F/379/B(U)F-96 (Aa) (Content No. 1)
Docket No. 71-3075

SUMMARY

By letter dated December 2, 2004, the U.S. Department of Transportation (DOT) requested the U.S. Nuclear Regulatory Commission (NRC) staff's assistance in evaluating and providing a recommendation to revalidate the Model No. TN-106 Transport Package authorized by French Certificate No. F/379/B(U)F-96 (Aa) (French Certificate). The DOT provided additional information by supplement dated April 8, 2005. In this letter, only Content No. 1 in the French Certificate was requested for our review and recommendation. In response to a staff request for additional information (RAI) dated September 9, 2005, the DOT provided supplemental information on January 9, 2006.

Based upon our review, the statements and representations in the French Safety Analysis Report 5573-Z, Rev. 2 (SAR), as supplemented, and for the reasons stated in the enclosed Safety Evaluation Report, the staff agrees that the package meets the requirements of International Energy Atomic Agency (IAEA) TS-R-1, 1996 Edition, Amended 2003, (IAEA Transport Regulations). Staff recommends revalidation of French Certificate with the following conditions:

Condition No. 1: Authorization is limited to Contents No. 1, Appendix 1a, of the French Certificate with the following additional limitations.

The content is limited to uranium dioxide (UO_2). The maximum masses of UO_2 must comply with the values set out in the table below according to the enrichment in ^{235}U of the most highly enriched fuel element (or element part) present in the cavity and in accordance with the diameter (D) of the internal arrangement designed for criticality purposes:

Enrichment in U-235 (% by mass)	**Mass UO_2 (kg) D = 6.0 cm**	**Mass UO_2 (kg) D = 12.0 cm**	**Mass UO_2 (kg) D = 20.3 cm**
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≤ 4	No restriction	No restriction	No restriction

Contents that include multiple enrichments shall adhere to the limits corresponding to the highest enrichment present. Other contents such as those containing other uranium compounds, plutonium oxides, mixed oxides, solid fuel elements containing uranium, or other solid non fissile radioactive material (except for the internal arrangement or any empty cladding having contained pellets from content Number 1) are not authorized.

Condition No. 1: (continued) An inert matrices is a general expression that describes that UO_2 is in the presence of material (i.e., metal) under conditions that this material has no influence on the safety analysis (i.e., not a neutron moderator, not a neutron reflector, not radiolysable, not pyrophoric, and not susceptible to decomposing with heat).

Condition No. 2: For transport in the United States, trunnions shall not be used for tie-down attachments. A transport skid that cradles the package shell that is designed to meet the accelerations factors 2g in vertical direction, 5g in lateral direction, and 10g in longitudinal direction shall be used.

Condition No. 3: Transport by air is not authorized.

Condition No. 4: Radiation surveys must include neutron dose rate measurements after loading the contents and prior to transport in the specific areas identified in the French Certificate, Appendix 0a, Section 3, "Conditions for Use of the Packaging."

Condition No. 5: Any sealed capsules must be dry prior to transport.

Condition No. 6: Containment boundary seals shall be tested to show no leakage greater than 1×10^{-7} ref-cm^3/sec within the 12 month period prior to transport. Prior to each shipment, after loading, the package shall show no leakage when tested to a sensitivity of at least 1×10^{-3} ref-cm^3/sec.

Condition No. 7: The cooling time must be a minimum of 4 years.

1.0 GENERAL INFORMATION

1.1 Packaging

The Model No. TN-106 Transport Package is comprised of four basic components:

(1) A Cylindrical Body. The body bounds a cylindrical cavity, of variable length, that is made up of the following materials from the inside outwards:

- An internal stainless steel plate envelope,
- A primary biological shield (gamma shielding) made from lead,
- A secondary biological shield (neutron shielding) made from borated resin
- An external stainless steel envelope with a base plate and handling and tie-down devices.

(2) The Front of the Cylindrical Body. The front part is a stainless steel flange welded to the shell to which the following is fitted:

- A revolving plug made from lead which provides access to the cavity,
- Two screwed clamps which hold the revolving plug in place,
- A revolving plug control orifice upon which a protective plug is fitted,
- A front lid for revolving plug maintenance,
- A front closure plate for loading contents,
- A vent orifice.

(3) The Back of the Cylindrical Body. The back part is a stainless steel flange welded to the shell to which the following is fitted:

- A stainless steel pushing device with a tungsten shield disc,
- A back closure plate providing access to the pushing device,
- A fill and drainage orifice.

(4) Two Shock Absorbing Covers (Impact Limiters). Two removable shock absorbing covers made from balsa wood and plywood, covered by a stainless steel envelope are screwed into the ends to provide the packaging with shock absorbency in the event of a drop. These covers also prevent access to the openings during transportation.

Leaktightness of the six openings (front lid, front closure plate, two orifices, revolving plug control, and back closure plate) is ensured by double EPDM O-ring seals recessed in grooves.

1.2 Dimensions and Weights

Inner Cavity:
- Useful Length (UL): Variable from 3.3 ft (1,000 mm) to 8.2 ft (2,500 mm)
- Useful diameter: 8 in (203 mm)

Maximum mass of the loaded package as a function of the UL:
- 16,058 lb (7,284 kg) to 27,216 lb (12,345 kg)

Total Length: 7.9 ft (2,424 mm) to 12.9 ft (3,924 mm)
Length without shock-absorbing covers: 5.8 ft (1,778 mm) to 13.1 ft (3,978 mm)
Diameter with shock-absorbing covers: 4.8 ft (1,458 mm)
Diameter without shock-absorbing covers: 3.1 ft (958 mm)

1.3 Contents

The French Certificate, Appendix 1a, authorizes five “Contents.” By letter dated April 8, 2005, only Content No. 1 was requested for our review and recommendation.

The type of radioactive material in Contents No. 1 is UO_2 on its own or in inert matrices (excluding graphite and beryllium). It can be shipped as fuel elements, pieces of fuel elements, rods, or pieces of rods which may or may not be irradiated or pressurized. The maximum allowable mass is 254 kg per meter length of the inner cavity. The maximum uranium enrichment is 10 weight percent U-235. Authorization is limited to Contents No. 1 of the French Certificate with the additional limitations specified in Condition No. 1.

1.4 Criticality Safety Index

As described in Section 6.0 "Criticality" of this Safety Evaluation Report, the criticality safety of the package was evaluated with an infinite number of damaged and undamaged packages. Therefore the Criticality Safety Index (CSI) is 0.0, as described in para. 528 of IAEA Transport Regulations.

2.0 STRUCTURAL

The Model No. TN-106 package was designed to be transported in the horizontal position with fissile materials consisting of fuel pins or rods containing UO_2.

The package was tested using ½ scale model for normal conditions of transport (NCT) for a free drop of 0.9 meters and 0.6 meters (mass > 10,000 kg.), and the results presented in Calculation 5573-Z-1-4, Rev. 0, Chapter 1, Appendix 4, were found acceptable and in compliance with the requirements of para. 722 of IAEA Transport Regulations.

The package was tested using ½ scale model for hypothetical accident conditions of transport (HAC) for a free drop of 9 meters (axial, lateral, and corner drop orientations), and a 1 meter drop onto a puncture bar. The results presented in Calculation 5573-Z-1-4, Rev. 0, Chapter 1, Appendix 4, were found acceptable, and in compliance with the requirements of para. 727 of IAEA Transport Regulations

A condition for use of the package as presented in “Conditions for use of the Packaging,” in Appendix 0a of the French Certificate is to set the cavity to a negative pressure of 0.2 bar just after closing the packaging (the gas used to fill the cavity can be air or any other neutral gas). However, the cavity of the ½ scale model was fixed at atmospheric pressure during the drop tests. When the cavity pressure is set to a negative pressure of 0.2 bar, the difference between the cavity pressure and the atmospheric pressure is approximately 1.2 bar.

In the September 9, 2005, RAI, staff requested the applicant to determine if the regulatory drop test conclusions are still valid considering the regulatory drop tests were performed on a ½ scale model with the cavity at atmospheric pressure rather than at a negative cavity pressure as required as a condition for use in the French Certificate.

The applicant presented new calculations in response to the RAI. The staff noted that the applicant inadvertently used a resultant pressure of 0.8 bar which resulted in a stress of 4.2 MPa, instead of using the correct value of 1.2 bar. The 1.2 bar would result in an additional bending stress in the inner cylindrical shell of 6.4 MPa. This additional stress due to pressure is judged to be negligible compared to the level of stress due to impact loads during the drop tests. Calculation 5573-Z-1-6, Rev. 0, (page 3), and Calculation 5573-Z-1-4, Rev. 0, (page 19) states that the maximum stress in the packaging, for any effective length of the cavity, is 150 MPa at 1.05% deformation. For the case of the length of 3200 mm and a horizontal drop, the allowable is 5% deformation of the internal cavity and 380 MPa (tensile strength).

The staff concludes that the results of the ½ scale model tests performed for various regulatory drop scenarios is still valid and the requirements of para. 722 and para. 727 of IAEA Transport Regulations are met.

The acceleration factors used for design of the handling devices and the tie-downs for various modes of transport were not per those recommended in IAEA Safety Guide TS-G-1.1(ST-2). The staff generated RAIs requesting an explanation for this discrepancy. The responses to these RAIs were reviewed by the staff.

The evaluation presented in Calculation 5573-Z-1-3, Rev. 0, Chapter 1, Appendix 3, “Mechanical Strength of Tie-down and Handling Devices,” was reviewed by the staff. The handling device is made up of two trunnions and two lifting lugs. The acceleration factors used for the design were slightly different (1.5g in both the vertical and longitudinal direction instead

of an acceleration factor of 2g in these directions) than those recommended in the Safety Guide TS-G-1.1(ST-2). This difference of 0.5g for the design of the handling device was judged to be of minor significance because page 3 of the above mentioned calculation indicates that the maximum stress during handling is 149 MPa. This is considerably less than the allowable limit of 335 MPa.

The tie-down device is made up of four trunnions. An evaluation was performed by the applicant using accelerations of 3g in vertical downward direction, 1.5g in the lateral direction, and 2g in the longitudinal direction. These acceleration factors were different than those recommended in TS-G-1.1 (ST-2), Table V.2, for transport within the United States of America (USA). However, per the applicant the trunnions will not be the tie-down attachments when the package will be transported within the USA. Instead, a transport skid which cradles the package shell will be designed for transport within the USA in compliance with the accelerations recommended in TS-G-1.1 (ST-2), Table V.2 (2g in vertical direction, 5g in lateral direction, and 10g in longitudinal direction, for all transport modes). This would also envelope accelerations recommended for any sea/water transport. The additional evaluation, using these acceleration factors, performed in response to the RAI for the handling device and the tie-down attachment demonstrate that the tie-down attachment design is adequate for transporting the Model No. TN-106 package containing UO_2 within the USA. The design of tie-down complies with the accelerations recommended in TS-G-1.1 (ST-2) Table V.2. The staff determined that the applicant has met the requirements of para. 636 and para. 650 of IAEA Transport Regulations.

To ensure proper tie-down during transport, staff recommends the following condition be included in the DOT Certificate:

Condition No. 2:
For transport in the United States, trunnions shall not be used for tie-down attachments. A transport skid that cradles the package shell that is designed to meet the accelerations factors 2g in vertical direction, 5g in lateral direction, and 10g in longitudinal direction shall be used.

Although no test was performed for the water immersion test for the containment, the applicant verified by means of analytical calculations (Calculations 5573-Z-1-1, Rev. 0) that the package can adequately withstand external gauge pressure of at least 2 MPa or 20 bars. This would be equivalent to a pressure that would result from the package being immersed in water to a depth of 200 meters of water for a period of not less than 1 hour. Page 12 of these calculations demonstrated that under the normal conditions the shell is capable of withstanding an external pressure of 75 bars, and under exceptional conditions it is capable of withstanding an external pressure of 101 bars. The additional stress induced by the negative internal pressure of 0.2 bar in the shell was judged to be negligible when compared to the overall capacity of the shell to withstand external pressures. The staff concurs that the requirements of para. 630 and para. 730 of IAEA Transport Regulations are met.

2.1 Materials Evaluation

The UO_2 contents are placed in either sealed or unsealed austenitic stainless steel containers that are not considered part of the containment system. Gamma and neutron shielding is provided by layers of lead and a borated resin. The cask is made of austenitic stainless steel with welded end plates. The contents are loaded through a revolving port that is double sealed with EPDM O-rings seals recessed in grooves. Impact limitation is provided by balsa wood encased in stainless steel sheathing.

The following comments and conclusions are made with respect to the ability of the Model No. TN-106 packaging materials to meet IAEA Transport Regulations. Other review disciplines may apply to the same IAEA requirement.

Paragraphs 416 and 680: Air transport is not requested, therefore, these requirements are not applicable to this cask.

Because the French Certificate doesn't specifically specify that air transport is not authorized, the staff recommends the following condition be included in the DOT Certificate:

Condition No. 3
Transport by air is not authorized.

Paragraph 501a: A maintenance program is specified for casks that are 3 years or older or have been used for 20 or more cycles. The program includes inspection of the seal surfaces, the revolving plate, and other removable parts. A helium leak test is also prescribed on the containment boundary. Additional tests, such as replacement of all seals, and dye penetrant of all handling devices is required on all packaging over 6 years. The six openings in the package are double sealed with EPMD 0-rings in recessed grooves designed to be used at a minimum of -40 C. One seal is in the rotating plug and the second is in the closure plate. The gasket material has a useful range of -50 to 160 C and a maximum service temperature of 177 C.

For the purpose of determining the pressure pulse driving the release of radionuclides, 100% of the rod in accidents, three ruptured rods (>6%) for off-normal conditions and one rod (2%) for normal handling are considered ruptured. This is consistent with NRC guidance and practice. All the containment system and external envelope welds are butt-welded and full penetration. All are 100% x-rayed and dye penetrant tested.

No credit is taken for the containment properties of the cladding, therefore, the maximum cladding temperature limit of 465 C acceptable.

Paragraph 501b: The welds are specified in Chapter 0, Table 0.7, of the SAR. All welds are dye penetrant tested. Those welds involved in the containment system are full penetration, 100% dye penetrant, and x-ray tested.

All the movable seals in the containment system consist of double 0-rings made of EPDM polymer and have a useful operating range of -50 to 170 C. The polymer seals are tested to 1×10^{-7} Pa m^3 s^{-1}. The mechanical properties of the structural steels were checked against the ASTM B&PV code, Section 2, Part D, and the CAST1 guidebook to ASME and found to be correct.

Paragraph 501c: The neutron shielding is provided by a self extinguishing proprietary "Resin F" shell. The volume of the cavity for the neutron resin is well known. The total amount of resin used to fill the cavity is checked against the volume of the cavity to assure the fill material is in place. The applicant is relying on small pours to guarantee a uniform distribution of B-10 within the neutron shield. This only guarantees that a certain amount of B-10 is within that poured section. While this is better than just specifying global boron content, it still does not guarantee lack of striation and streaming within the individual pour.

Even though the French Certificate specifies dose rates be in compliance with statutory limits, the French Certificate does not specify that a neutron radiation survey be conducted.

Therefore, staff recommends the following condition be included in the DOT Certificate:

<u>Condition No. 4:</u>
Radiation surveys must include neutron dose rate measurements after loading the contents and prior to transport in the specific areas identified in the French Certificate, Appendix 0a, Section 3, "Conditions for Use of the Packaging."

Paragraph 613: The materials of the packaging include Type 304L stainless steel, lead, and Resin F.

The French Certificate, Appendix 1a, Contents No. 1, indicates that the UO_2 may be in an inert matrix but the certificate does not define the inert matrix. A definition for the inert matrix was given in the RAI response dated January 9, 2006, an "inert matrices is a general expression that describes that UO_2 is in the presence of material (example: metal) under conditions that this material has no influence on the safety analysis (example: not a neutron moderator, not a neutron reflector, not radiolysable, not pyrophoric, not susceptible to decomposing with heat...). One example of inert matrices is aluminum." If an inert material does not satisfy the stated definition, undesirable effects might occur. Therefore, staff recommends the following condition be included in the DOT Certificate as part of Condition No. 1:

An inert matrices as authorized in the French Certificate, Appendix 1a, Contents No. 1, is defined as "inert matrices is a general expression that describes that UO_2 is in the presence of material (i.e., metal) under conditions that this material has no influence on the safety analysis (i.e., not a neutron moderator, not a neutron reflector, not radiolysable, not pyrophoric, and not susceptible to decomposing with heat)."

Paragraph 607: The lifting trunnions are made of X2CrNiMoN22-5-3 austenitic stainless steel. The lifting trunnions are fillet welded to the cask outer shell. Each weld is 100% dye penetrant tested.

Paragraph 638: The body of the cask is composed of steel that is equivalent to ASTM A240 Type 304 L. The packaging is designed and used in accordance with IAEA Transport Regulations and other international codes specified in the SAR.

Paragraph 642: The drying procedure discussed under para. 651c should adequately remove moisture to the level where radiolytic decomposition of water is not important. The only other material subject to radiolytic decomposition is the Resin F shielding material. Specifications provided in the January 9, 2006, RAI response indicated that the resin has a critical gamma dose of 10^7 Gy. This is well above the expected dose since the resin is protected by an inner lead gamma shield.

Paragraph 651a: The maximum fuel temperature under any circumstance is 480°C. At this temperature, neither the UO_2 fuel nor zircaloy cladding will deform. The air in the maximum sized empty package cavity would allow at most 6.5g of UO_2 to oxidize (less then a ½ inch of fuel) to U_3O_8 powder. This is insignificant fuel relocation. The internal container or basket is made of stainless steel and will not deform under the imposed temperature limits.

Paragraph 651b: The gamma shielding is provided with 99.7% pure lead with a minimum density of 11.2 g/cc. Exterior to the lead is neutron shielding provided by the Resin F containing boron. The melting point of lead is well above the operating range of 150 C. The resin does not have a melting point but will have slight charring at high temperatures. The lead shield will incur -0.2% to +0.4% strain over the operating temperature range. Cracking will not

occur in the ductile lead. The resin will experience a similar range of strains over the operating temperature range.

Paragraph 651c: The drying criterion in the French SAR is not consistent with NRC guidance and practice, which led to the RAI No. 8 dated September 22, 2005. Adequate support for the proposed method was provided in the applicant's RAI response dated January 9, 2006. The response provided confidence to the staff that the proposed drying method and criteria will satisfactory remove the moisture to a level where corrosion due to the presence of moisture would be insignificant for the cask cavity and non-leaktight capsules.

The SAR also indicates that leak tight capsules might be used. The response to the RAI indicates that there is a different design leak tight capsule then shown in the SAR, that can be drained and dried. Therefore, staff recommends the following condition be included in the DOT Certificate:

Condition No. 5:
Any sealed capsules must be dry prior to transport.

Paragraph 655: There is no thermal shield in this cask.

Paragraph 656(b)(ii): The Resin F and lead shields are both contained by stainless steel. This shielding will perform to IAEA transport requirements under the required regulatory drop tests.

Paragraph 664: The materials used in this package consist of 304L stainless steel, lead, and Resin F. The stated thermal properties (conductivity, emissivity, specific heat, heat capacity) of the stainless steel and the lead were found to be in the acceptable -40 C to +38 C range.

Paragraph 679: Only addressing the material review, subcriticality is ensured by maintaining package and content configuration for normal and accident conditions.

The impact limiters are made from balsa wood sheathed in stainless steel. The compressive stress of the balsa wood must fall between 9 and 11 MPa. Staff verified the properties of the balsa wood.

The packaging consists mainly of 304L stainless steel and lead. The materials properties yield strength and ultimate strength were checked against the ASTM A340 for the structural stainless steel (ASTM B&PV code, Sec. 2 part D, and the CAST1 guidebook to ASME) and verified to be correct. In the temperature range of interest, the two austenitic stainless steels are not subject to brittle fracture.

The fuel is placed inside an internal container within the cask. The fuel can relocate within the container or fracture further but not relocate from the container.

3.0 THERMAL

The staff evaluated the applicant's thermal analyses for normal conditions of transport and hypothetical accident conditions following the requirements in the IAEA Transport Regulations. The applicant analyzed the package performance using the IDEAS MASTERS SERIES analysis program with a maximum decay heat of 500 W/m. Initial and boundary conditions and package pressures were consistent with the IAEA Transport Regulations.

Analysis results are presented in the following table:

THERMAL ANALYSIS RESULTS FOR NORMAL AND ACCIDENT CONDITIONS				
	Normal Conditions of Transport		Hypothetical Accident Conditions	
Component	Maximum Temperatures (C)	Temperature Limit (C)	Maximum Temperatures (C)	Temperature Limit (C)
Outer Shell	133 / < 85 [1]	85[2]	607	-
Resin	149	150	587	-
Lead	152	327	203	327
Inner Shell	153	-	204	-
Seals	149	160	167	180
Fuel Cladding	465	400[3]	480	570

[1] 133 C is the maximum surface temperature when exposed to solar insolation, <85 is the surface temperature for the package in the shade without the transport container.
[2] The 85 C temperature limit is for the package in the shade without the transport container.
[3] The 400 C limit is based on NRC Interim Staff Guidance No. 11. This limit has been established to minimize intact fuel cladding degradation. It is listed above for information only and is not applicable to this design because fuel cladding is not relied on for containment.

The staff finds that there is reasonable assurance that the Model No. TN-106 as presented in the SAR, with the contents as specified in Condition No. 1, and with a maximum decay heat of 500 W/m meets the thermal requirements of IAEA Transport Regulations.

4.0 CONTAINMENT

The containment analysis in the SAR postulates a radiological release based upon a specific design leakage rate. However, in the December 2, 2004, submittal and RAI response in the January 9, 2006, submittal, the applicant has proposed an exception to the SAR's analysis and proposes that the containment system be tested to the leaktight criteria of ANSI 14.5-1997 of 1×10^{-7} ref-cm^3/sec. The staff has unanswered issues with the methodology used to determine the radiological release, however, since the applicant is proposing a leaktight containment boundary, the SAR analysis for postulated radiological release is no longer needed in determining if the containment analysis meets IAEA Transport Regulations.

As a justification for invoking the leaktight criteria, the applicant referenced the initial leakage testing that was performed after fabrication and the leak testing performed after the hypothetical accident condition (HAC) tests. The staff reviewed the French fabrication leak test reports (with inserted translation) performed on the various parts of the containment boundary and was satisfied that it provides reasonable assurance that the fabrication leak test did meet the leaktight criteria. Additionally, the applicant stated in the forwarding letter, that the drop test report indicated that the leaktight criteria was met as shown in Chapter 1, Appendix 11, Test Report 5573-C-19. The staff reviewed the aforementioned test report and it showed that the scale model that was drop tested was indeed leak tested to leaktight criteria. However, leaktesting of scale models is not recommended by ANSI 14.5-1997, para. 7.2.2, and the staff can give no credit for this post drop leak test in this application.

The applicant further stated in the forwarding letter dated December 2, 2004, that the leak tests for the HAC tests, fabrication, maintenance, and periodic would be performed to 1×10^{-7} ref-cm^3/sec and the preshipment leak test would be performed to 1×10^{-3} ref-cm^3/sec in accordance with ANSI 14.5-1997. These values are in conflict with the SAR chapter for acceptance and maintenance test program and therefore staff recommends the following condition be included in the DOT Certificate:

Condition No. 6
Containment boundary seals shall be tested to show no leakage greater than 1×10^{-7} *ref-cm*^{3}*/sec within the 12 month period prior to transport. Prior to each shipment, after loading, the package shall show no leakage when tested to a sensitivity of at least* 1×10^{-3} *ref-cm*^{3}*/sec.*

The staff also reviewed the containment pressure information and noted that the applicant indicated that the normal pressure in the cask at thermal equilibrium is 0.37 bars and the maximum pressure under HAC in excess of 7.6 bars. These pressures were apparently derived from bulk gas temperatures of 553 K (536 F) and 586 K (615 F), respectively. The pressure calculations included consideration of the fuel plenum gases and fission gases, with a maximum of 6% of the tubes failing under normal conditions and 100% failing during HAC. The staff could not independently verify the internal pressure calculations because the applicant did not clarify the radiological basis of the source term as requested in the RAI dated September 9, 2005, and as a result the amount of fission gas could not be verified. Also, it was not apparent to the staff whether or not the normal vacuum condition of the cask (0.37 bars) was included in determining the maximum cask pressure. However, the staff did note that the cask shell and various lids are capable of withstanding internal pressures significantly greater than the HAC pressure of 7.6 bars (refer to Table 1.1-1 in Chapter 1, Appendix 1) assuming elastic behavior of the associated materials.

In conclusion, the staff could not verify that (1) the drop test report indicated that the leaktight criteria was met since the test was based on a scale model and (2) the contribution of the fission gas in the pressurization calculation. However, the original fabrication leaktest, the relative strength of the package during normal and accident conditions, and the verification of gasket temperature suitability by the thermal reviewer, provide reasonable assurance that the containment boundary will perform satisfactorily.

5.0 SHIELDING

The TN-106 is a cylindrical cask made up of a 23.5 mm shell of stainless steel, 145 mm of lead, 120 mm of resin, and a 20 mm stainless steel outer shell. The effective diameter of the internal cavity in 203 mm. The length of the cavity can vary but the maximum length is 3200 mm. The top of the cask has a revolving plug made from stainless steel and lead. This revolving plug is used to open the cask. The bottom of the cask is made of stainless steel and tungsten.

While this package has been designed to hold a variety of different fuels, the only fuel that was evaluated in this review was the uranium oxide (UOx) fuels with an enrichment less than or equal to 10 weight percent ^{235}U. The contents which can be loaded into this cask are UOx fuel pins or rods, irradiated or not, whole or in pieces. Additionally, the cladding may be pressurized or not. A maximum of 50 rods can be loaded into the package. The burnup will not exceed 100,000 Mwd/tU. The minimum cooling time requested in the application was 3 months, however, the supporting calculations were performed for fuel with a cooling time of 4 years.

Geometric properties such as the diameter of pellets, pitch of the rods, thickness and type of clad of the fuel were not provided in the application nor were they provided in response to staffs RAI dated September 22, 2005. This information was needed to perform confirmatory calculations to verify the source term and dose rate information.

In an attempt to verify the source term provided in the application, staff used the ORIGEN-ARP module from SCALE 5.0 to determine the source term for PWR and BWR fuel assemblies for enrichments up to 5 weight percent ^{235}U. The SAS2 module of SCALE 5.0 was used to evaluate the source term for fuel with enrichments up to 10 weight percent ^{235}U. SAS2 was also used to obtain an approximation of dose rates.

In the confirmatory calculations, using the properties for one 17x17 PWR and one 8x8 BWR fuel assembly, staff was able to replicate the gamma and neutron source terms for fuel with a minimum cooling time of 4 years for normal conditions of transport and for hypothetical accident conditions. The resulting dose rates from these calculations were less than the IAEA limits for normal conditions of transport of 2 mSv/hr on contact, 0.1 mSv/hr at one meter, and less than 10 mSv/hr at one meter for hypothetical accidents.

In confirmatory calculations, using a 3 month cooling time in the calculations, produced a significantly higher source term which would therefore result in higher dose rates.

With the specific information about the fuel to be shipped not being provided, staff had to use conservative assumptions to bound the fuel in the confirmatory calculations and could only confirm the source term and dose rates for fuel cooled for a minimum time of 4 years.

Therefore, staff recommends the following condition be included in the DOT Certificate:

Condition No. 7:
The cooling time must be a minimum of 4 years.

Based upon staff's review and confirmatory calculations, staff has reasonable assurance that the dose rates from the packages to be shipped will be within the limits of IAEA Transport Regulations paras. 530, 531, 532, 573(c), and 657(b)(ii)(i) as long as the minimum cool time is not less than 4 years. Additionally, the French Certificate, Appendix 0a, Section 3, "Conditions for Use of the Packaging" requires that dose rates be taken after loading the contents and prior to conveyance, and verified to be in compliance with statutory allowable limits.

6.0 CRITICALITY

This section presents the findings of the criticality safety review for the TN-106 package. The purpose of this review is to verify that the package design meets the criticality safety requirements of IAEA Transport Regulations under normal conditions of transport and hypothetical accident conditions.

6.1 Description of Criticality Design

The TN-106 package consists of a cylindrical body and two shock absorbing covers. The cylindrical body is of varying length and is composed of a series of concentric shells in the following order: stainless steel, lead, borated resin, and stainless steel. The two identical shock adsorbing covers are made from balsa wood and covered with a steel envelope.

The main criticality control feature of the TN-106 consists of the geometry control provided by either insert canisters or the inner diameter of the cavity. Mass limits are developed for each relevant diameter and enrichment combination. Additional criticality control is provided by the package design which prevents significant interaction between individual packages in an array.

The applicant performed an evaluation to show that the package design meets the criticality requirements of IAEA Transport Regulations. The applicant performed criticality analyses for a single package and an array of packages. These analyses both utilized a package model that included the loss of the outer 1.0 cm of the resin. This material is lost due to fire damage and is assumed to be replaced by water. The assumption of resin removal and subsequent replacement with full density water was analyzed by staff via confirmatory analyses and sensitivity studies and found to be acceptable.

Chapter 6 of the SAR was reviewed for completeness of information and consistency. The information, parameters, and dimensions provided were sufficient to perform a review and are consistent throughout the application. Chapter 6 presents the results of the applicant's criticality analyses. The criticality results were found to meet the applicable acceptance criteria.

6.2 Fissile Material Contents

The applicant requested authorization to transport irradiated UO_2 rods enriched up to 10% by mass in ^{235}U. The allowed UO_2 mass is a function of both enrichment and diameter as shown in Calculation No. 41299-02, Rev. 0, Table 2.2-1. This table dictates that for inner canister diameters of 6.0 cm and 12.0 cm there is no restriction on the total mass of UO_2 allowed. For an inner diameter of 20.3 cm the maximum allowed UO_2 mass are:

- 14.2 kg for enrichments greater than 5% and less than or equal to 10%
- 53.0 kg for enrichments greater than 4% and less than or equal to 5%
- unrestricted for enrichments less than or equal to 4%

Other contents such as those containing plutonium oxides, mixed oxides, solid fuel elements containing uranium, or other solid non-fissile radioactive material are not authorized. Contents that include multiple enrichments shall adhere to the limits corresponding to the highest enrichment present.

6.3 General Considerations

6.3.1 Model Configuration

The applicant evaluated a single package and package array under hypothetical accident conditions since these results conservatively bound the normal conditions of transport.

The applicant modeled the package as an inner cavity lined with stainless steel and surrounded by concentric cylinders of lead, resin, and an outer shell of steel. The outer 1.0 cm of resin on the inner surface of the outer steel shell burns under hypothetical accident conditions and is assume to be replaced with water.

The package impact limiters were neglected for both the normal and accident condition analyses. Because of the limited damage sustained in the accident tests, the nominal packaging dimensions were used for the model.

The rotating plug within the upper lid was modeled as water reflector to simplify the actual plug geometry. The modeling configuration and simplifications used in the criticality analysis were reviewed by the staff and were found to be acceptable.

6.3.2 Material Properties

The material specifications used in the criticality analysis are provided in Table 4.2-1 of Attachment 6. The applicant took credit for only 75% of the minimum acceptable ^{10}B content by reducing the isotopic abundance of ^{10}B from 18% to 13.5% (18* 75% =13.5%).

The material properties and specifications used in the criticality analysis were reviewed by the staff for completeness and correctness and were found to be acceptable.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant performed the criticality evaluation using the KENO V.a module from the SCALE5 package and the 44 group ENDF/B-V cross section library. Staff finds that the code and the selected cross sections were sufficiently documented and validated, and agrees that they are appropriate for this application.

To address the full range of fuel configures, calculations at an enrichment of 10% (bounding enrichments greater than 5% and #10%) modeled a homogenized fuel-moderator mixture, while calculations for enrichments ≤**5%** were performed assuming a heterogeneous fuel-moderator mixture. This was necessary to evaluate the increased reactivity of heterogeneous fuel lumps at enrichments less than 5%. The heterogeneous evaluations were conducted using a computational method where cell weighted cross-sections were generated considering heterogeneous fuel pellets and moderator and these cross sections were then applied to a homogeneous fuel/moderator volume. This methodology was evaluated by the staff and found to be acceptable for this application.

The applicant included a sufficient number of particle histories in its calculations to achieve a statistical standard deviation of 0.00040 to 0.00058 in the calculated values of k_{eff}. This was determined to be sufficient for this application.

6.3.4 Demonstration of Maximum Reactivity

The applicant performed sensitivity analyses on several parameter variations to identify the optimum set of conditions which maximize k_{eff} of the system.

The parameter variations considered were, H/X for specified uranium mass and contained within the package geometry, fuel pellet radius for **enrichments ≤5%, and** the density of interspersed moderator for array considerations.

The optimum conditions were used by the applicant in the final safety demonstration for each applicable configuration. Staff found the methods used to identify the optimum set of conditions to be appropriate and the set of parameter values to be acceptable.

6.3.5 Confirmatory Analyses

The NRC staff performed independent confirmatory criticality calculations for the single packages and packages array under hypothetical accident conditions. The staff's model assumptions were similar to the applicant's model.

The staff's calculations were performed with both SCALE5 (using KENOVI and the 44GROUNDF/B-V cross section set) and the Monte Carlo computer program MCNP5 with continuous energy cross sections primarily from the ENDF/B-VI data base. The staff's analyses included the single flooded package and flooded accident array. Staff’s maximum k_{eff} values have acceptable agreement with the applicant's results and are within the acceptable limits.

6.4 Single Package Evaluation

The applicant performed an analysis of a single reflected package for hypothetical accident conditions (HAC). These conditions bound the normal conditions of transport. The maximum k_{eff} values, adjusted for the statistical uncertainty, are shown in Table 6-1 below and are less than the applicant's upper subcritical limit.

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport (NCT)

The model geometry under NCT is assumed to be the same as that for HAC. Therefore, only HAC cases were evaluated.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The applicant's criticality evaluation for a package array under the hypothetical accident conditions considered an infinite number of flooded packages without impact limiters in a square lattice. Because water is free to flow throughout the interior of the package, the internal water density was assumed to be uniform for the study of the optimum moderator density condition. Reactivity was optimized for the case of full density water inside the package and no water between the packages. The maximum k_{eff}, including the statistical uncertainty, is shown in Table 6-1 below and is less that the applicant's upper subcritical limit.

Table 6-1
Applicant's Maximum Value of k_{eff} (10% Enrichment, D= 20.3 cm)

Hypothetical Accident Conditions (HAC)		
	ks (k_{eff}+2σ)	**Upper Subcritical Limit**
Single Package	0.93628	0.9408
Infinite Package Array	0.93583	0.9408

Since infinite arrays of packages were evaluated and found to be below the upper safety limit, the analysis resulted in a CSI of 0.

6.7 Benchmark Evaluations

The applicant performed a benchmark analysis to determine an Upper Subcritical Limit (USL) on its calculated values of k_{eff}. The benchmarks were taken from the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" published by the Organization for Economic Cooperation and Development/Nuclear Energy Agency. The applicant selected two sets of benchmark experiments, one for homogenous fuel/moderator mixtures and a second set for heterogeneous mixture. The experiments were chosen to reflect the enrichment and spectral characteristics of the transportation package. Experiments with neutron absorbers present were limited to only those containing boron poisons.

The applicant analyzed the benchmark data using the USLSTATS program developed by the Oak Ridge National Laboratory. For the homogenous cases the applicant tested for trends in the calculated values of k_{eff} as a function of three parameters: 1) enrichment, 2) moderator to fuel atom ratio (H/X), and 3) the energy of the average lethargy causing fission (EALF). For the heterogeneous cases the applicant tested for trends in the calculated values of k_{eff} as a function of four parameters: 1) enrichment, 2) moderator to fuel atom ratio (H/X), 3) the energy of the average lethargy causing fission (EALF), and 4) the water to fuel volume ratio. The applicant then determined a separate USL for both the heterogeneous and homogenous cases. The applicant established minimum USLs of 0.9408 and 0.9383, for the homogenous and heterogeneous cases respectively. These values include the bias administrative margin (0.05) and the 95% confidence band width of the data.

Staff reviewed the set of benchmark experiments and method for establishing the USL and found them to be acceptable.

6.8 Evaluation Findings

Based on its review of the representations and information supplied by the applicant, and the analyses performed by staff, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated by the applicant, and finds reasonable assurance that the package meets the criticality safety requirements of IAEA Transport Regulations.

7.0 PACKAGE OPERATIONS

Appendix 0a of the French Certificate includes “Conditions for use of the Packaging,” which states that the package operations must be in accordance with Chapter 6A.

Instructions in Chapter 6a include:
Section 2.2.1: Loading the package in a cell.
Section 2.2.2: Loading the package in a pool.
Section 2.3: Preparation and inspection prior to shipment.
Section 3.1: Drying the cavity.
Section 3.2: Leaktightness check prior to transport.[4]
Section 4.0: Decontamination after loading.

[4] In letter dated December 2, 2004, the applicant revised the preshipment leak test sensitivity specified in Chapter 6A of the French SAR and in the French Certificate from 6.65×10^{-5} Pa*m^3/sec to 1×10^{-3} ref-cm^3/sec for revalidation in the U.S. Therefore, as also stated in Chapter 4 of this Safety Evaluation Report, staff recommends the following condition be included in the DOT Certificate:

Condition No. 6:
Prior to each shipment, after loading, the package shall show no leakage when tested to a sensitivity of at least 1×10^{-3} ref-cm^3/sec.

The certificate further specifies to set the cavity to a negative pressure of 0.2 bar after closing the packaging, to perform radiation and contamination surveys, and to ensure both a tamper-indicating seal and statutory labels are affixed. However, since the French Certificate does not specify neutron dose rate surveys, staff recommends Condition No. 4 to be included as a condition in the DOT Certificate.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Appendix 0a of the French Certificate includes a "Maintenance Program," which states that the maintenance must be in accordance with Chapter 7A.

In letter dated December 2, 2004, the applicant revised the leak rate sensitivity specified in Chapter 7A of the French SAR for Fabrication, Maintenance, and Periodic tests to 1×10^{-7} ref-cm^3/sec. Therefore, as also stated in Chapter 4 of this Safety Evaluation Report, staff recommends the following condition be included in the DOT Certificate:

Condition No. 6:
Containment boundary seals shall be tested to show no leakage greater than 1×10^{-7} ref-cm^3/sec within the 12 month period prior to transport.

The certificate and instructions in Chapter 7a also include specifications for periodic maintenance. Two different levels of periodic maintenance are specified depending on when the packaging was manufactured and brought into service and the number of transport cycles the package has undergone. The periodic maintenance required at every 3 years or at the most 20 transport cycles includes a containment leaktightness check and inspections of seal bearing surfaces, tappings, screws, handling point welds, and the removable and moving parts of the packaging. The periodic maintenance required at every 6 years or at the most 60 transport cycles includes a visual inspection of the internal containment, dimensional inspections of the trunnions, leaktightness check of the shock-absorbing covers, dye-penetrate testing for the welds of handing devices, replacement of fusible plugs, and replacement of all seals.

CONCLUSIONS

Based upon our review, the statements and representations in the French TN-106 Safety Analysis Report (5573-Z, Rev. 2), as supplemented, and for the reasons stated in this Safety Evaluation Report, the staff agrees that the Model No. TN-106 transport package, authorized by the French Certificate No. F/379/B(U)F-96 (Aa), meets the requirements of IAEA TS-R-1, 1996 Edition (As amended 2003). The staff recommends revalidation of this package design with the conditions stated in the Safety Evaluation Report.

Issued with letter to R. Boyle, Department of Transportation,
on March 14, 2006.