U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-0383
Volume 2
Revision 28

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

Office of Nuclear Material Safety and Safeguards

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United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-0383
Volume 2
Revision 28

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

Manuscript Completed: October 2013
Date Published: October 2013

Office of Nuclear Material Safety and Safeguards

ABSTRACT

The purpose of this directory is to make available a convenient source of information on package designs approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packages, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volume 2. The report includes all package designs approved prior to the publication date of the directory as of September 2013.

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NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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0361	10	71-0361	USA/0361/B(U)F-96	1	OF	6

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

National Nuclear Security Administration
P.O. Box 5400
Albuquerque, NM 87185-5400

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NUREG-0361; Safety Analysis Report for the Plutonium Air Transportable Package Model No. PAT-1, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PAT-1

(2) Description

A stainless steel containment vessel (designated TB-1, SAR Drawing 1017) surrounded by a stainless steel and redwood overpack (designated AQ-1, SAR Drawing 1002).

- For plutonium oxide shipments, the contents are sealed within a stainless steel product can (designated PC-1, SAR Drawing 1024) inside the containment vessel.
- For plutonium metal shipments, the contents are sealed within a titanium vessel (designated T-Ampoule, SAR Addendum Drawing 2A0261) inside of the containment vessel.

The AQ-1 overpack is a right circular cylinder, approximately 42-1/2 inches long by 24-1/2 inches outside diameter. The walls of the overpack consist of approximately 8 inches of grain oriented redwood encased within double stainless steel drums. The ends of the drums are doubly closed. A copper heat conducting element and an aluminum load distributer are encased within the redwood.

The TB-1 containment vessel is approximately 8-1/2 inches outside length by 6-3/4 inches outside diameter. The minimum wall thickness of the vessel is approximately 1/2 inch. The interior cavity of the vessel is a right circular cylinder, 4-1/4 inches diameter, with hemispherical ends. For oxide shipments, the vessel is closed by 12, 1/2-inch diameter bolts and doubly sealed with a copper gasket and knife edges and an elastomer O-ring. For plutonium metal shipments the vessel is closed by 12, 1/2-inch diameter bolts and sealed with a copper gasket and knife edges only.

The weight of the package is approximately 500 pounds. The weight of the TB-1 containment vessel, when loaded with 4.4 pounds of contents is approximately 41.7 pounds. The PAT-1 with T-Ampoule configuration is limited to the current certified TB-1 gross payload weight of 2100 g (4.7 lbs).

(3) Drawings and Specifications

The Model No. PAT-1 packaging is fabricated in accordance with the drawings and specifications in Section 9.0 of the *Safety Analysis Report for the Plutonium Air Transportable Package, Model PAT-1*, NUREG-0361, hereafter identified as SAR, as supplemented by Issue B of Drawing Nos. 1004, 1009, 1013, 1016, 1017, 1019, 1020 and 1022 and with drawings and specifications in Section 1.0 of the *PAT-1 Safety Analysis Report Addendum*, SAND2010-6109 Revised, hereafter identified as SAR Addendum, as supplemented by Issue A of Drawings R99794, 2A0259, 2A0260, 2A0262, 2A0264, 2A0266, 2A0267, 2A0269, and 2A0385, and Issue B of Drawings 2A0263, 2A0261, 2A0265, and 2A0268.

(b) Contents

(1) Type and form of material A

Plutonium oxide and its daughter products, in any solid form. The plutonium oxide may be mixed with uranium oxide and its daughter products, in any solid form.

(2) Maximum quantity of material A per package and additional permissible contents

(i) Maximum 2.0 kg total radioactive material, plus: maximum 16 grams of water and 10 grams of polyethylene or polyvinylchloride bagging material. The maximum decay heat load of the contents may not exceed 25 watts.

(ii) Maximum 200 grams total radioactive material, plus: maximum one gram of water, maximum 200 grams of metal canning material (in addition to the PC-1 product can, Drawing No. 1024), maximum 64 grams of aluminum foil or honeycomb (in addition to the top spacer, Drawing No. 1015), maximum 175 grams of glass and maximum 35 grams polyethylene or polyvinylchloride bagging material. The maximum decay heat load of the contents may not exceed 25 watts.

(3) Type and form of material B

Plutonium metal (alloyed or non alloyed) in various isotopic compositions and composite (Pu and beryllium (Be) separated by a titanium layer) material. The maximum decay heat load of the plutonium metal contents may not exceed 25 watts.

(4) Maximum quantity of material B per package and additional permissible contents:

(a) 731 to 831 gram alpha or delta phase plutonium metal hollow cylinder in T-Ampoule per package; or,
(b) 338 gram maximum plutonium metal in each of two SC-2 Sample Container (SAR Addendum Drawing 2A0265), alloyed plutonium, maximum content per package is 676 grams; or,
(c) 174 gram maximum plutonium metal in each of three SC-1 Sample Container (SAR Addendum Drawing 2A0268), alloyed plutonium, maximum per package is 522 grams; or,
(d) 60 gram maximum plutonium composite in each SC-1 or SC-2 Sample Container, 120 gram maximum for SC-2 and 180 gram maximum for SC-1 shipments.
(e) One type of plastic labeling or tagging material of those listed below per shipment not to exceed these limits:

Material Type	Quantity (gram)
Polyethylene terephthalate (such as Metalized PET, Mylar™)	6.9
Polyethylene	3.5
Polyvinyl chloride (PVC)	12.2
Polytetrafluoroethylene (PTFE, such as Teflon™)	12.5

All previous labels must be removed prior to application of new labels.

(f) Tantalum foil may be used as packing material within the T-Ampoule and SC-1 or SC-2 Sample Containers.
(g) Neutron emission from the Pu/Be source is limited to 363 n/s/cm^2.
(h) Pu/Be sources are limited to a contact surface of 91 cm^2 or less.

(c) Criticality Safety Index

Minimum transport index to be shown on label material A content package for nuclear criticality control:	0.4
Minimum transport index to be shown on label of material B content package for nuclear criticality control:	0.1

6. Prior to first use, each packaging shall meet the acceptance tests and standards specified in Subsection 8.1 and Section 9.0 of the SAR.

   In addition to the packaging requirements above, for plutonium metal shipments, prior to first use, the acceptance tests and standards for the components listed in Section 1 and Subsection 8.1 of the SAR Addendum apply.

7. Prior to each shipment, the package shall meet the tests and criteria specified in Subsection 8.2 of the SAR for plutonium oxide shipments and the SAR and SAR Addendum for components specific to plutonium metal shipments.

8. The TB-1 O-ring is removed from the PAT-1 design for the plutonium metal shipment.

9. The electro-refined plutonium metal as defined by the isotopic composition in SAR Addendum Section 1.2.2 must be shipped within one year of manufacture.

10. Each TB-1 with T-Ampoule configuration shall be leak tested to leaktight prior to shipment. The leak rate tests shall be conducted in accordance with ANSI N14.5 using calibrated equipment as described in the SAR Addendum. A radiological survey must be performed on each PAT-1 package prior to shipment. Validation that the survey was performed must be communicated to the package destination and retained by the shipper as part of shipment records. Surface contamination on any accessible part of the package must not exceed the limits specified in 49 CFR 173.443, Table 9. Emanations must not exceed the limits specified in 49 CFR 173.441. Measurement equipment used for surveys must be calibrated and of sufficient accuracy.

11. For the Department of Energy, the packaging shall be designed, procured, fabricated, accepted, operated, maintained, and repaired in accordance with the Quality Assurance requirements of Chapter 9 of the SAR Addendum. For other applicants, the packaging shall be designed, procured, fabricated, accepted, operated, maintained, and repaired in accordance with an NRC approved Quality Assurance program.

12. In addition to the requirements in Subsection G of 10 CFR Part 71:

    a. Packages containing plutonium oxide and its daughter products in any solid form or containing plutonium oxide mixed with uranium oxide and its daughter products, in any solid form shall be prepared for shipment in accordance with the operations specified in the SAR, Section 7, *Operating Procedures.*

    b. Packages containing plutonium metal shall be prepared for shipment in accordance with the operations specified in SAR Addendum, Section 7, *Package Operations.*

13. The systems and components of each packaging shall meet the periodic tests and criteria specified in Subsection 8.3 of the SAR for plutonium oxide shipments and Subsection 8.2 of the SAR Addendum for plutonium metal shipments.

14. Repair and maintenance of the packaging shall be in accordance with Sections 8.0 and 9.0 of the Safety Analysis Report and with Subsection 8.2 of the SAR Addendum.

15. For plutonium oxide shipments, the PC-1 product can and the top spacer need not be used when the contents include 20 curies or less of plutonium oxide.

16. Through special arrangement with the carrier, the shipper shall ensure observance of the following operational controls for each shipment of plutonium by air:

    (a) The package(s) must be stowed aboard aircraft on the main deck in the aft-most location that is possible for cargo of its size and weight. No other type cargo may be stowed aft of the package(s).

    (b) The package(s) must be securely cradled and tied-down to the main deck of the aircraft. The tie-down system must be capable of providing package restraint against the following inertia forces acting separately relative to the deck of the aircraft: Upward, 2g; Forward, 9g; Sideward, 1.5g; Downward, 4.5g.

    (c) In commercial transport, cargo which bears one of the following hazardous material labels may not be transported aboard an aircraft carrying a package(s):

Explosive A	Non-Flammable Gas
Explosive B	Flammable Liquid
Explosive C	Flammable Solid
Spontaneously Combustible	Flammable Gas
Dangerous When Wet	Oxidizer
Organic Peroxide	Corrosive

    This restriction does not apply to hazardous material cargo labeled solely as:

Radioactive I	Poison
Radioactive II	Poison Gas
Radioactive III	Irritant
Magnetized Materials	Etiologic Agent

17. Packagings must be marked with Package Identification Number USA/0361/B(U)F-96.

18. The package authorized by this certificate is hereby approved for transportation of plutonium by air (10 CFR 71.64).

19. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

20. Expiration date: December 31, 2015.

REFERENCES

Safety Analysis Report for the Plutonium Air Transportable Package Model Number PAT-1, NUREG-0361, June 1978 and PAT-1 Safety Analysis Report Addendum, SAND2010-6109 Revised submitted on December 15, 2010.

Sandia Laboratories application dated February 20, l980.
Supplements dated: July 27, 1990, July 20, 1993, September 21, 2009, and September 20, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert K. Johnson

Robert K. Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 12/23/10

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Babcock & Wilcox Nuclear Operations Group, Inc.
P.O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWXT Nuclear Products Division application dated December 23, 2003, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UNC-2600

(2) Description

The inner container is an 11-gauge steel box with inside dimensions of 2-5/8" high x 7" wide x 96" long. The inner container is supported in a 22-1/2" ID by 102-1/2" long, 14-gauge steel drum by an insertable cage formed by nine 21-1/2" diameter by 3/8" thick steel plates, spaced approximately 12" apart, with a channel formed through the center of the plates by angle irons. The outer container closure is made with a 14-gauge drum lid with 12-gauge bolt locking ring with drop forged lugs, one of which is threaded, having a 5/8" diameter bolt.

(3) Drawings

The packaging is constructed in accordance with Thomas Gutman Consultant Drawing No. B-2600-2, Sheets 1 through 6, Rev. 3.

(b) Contents

(1) Type and form of material

Unirradiated, uranium-zirconium, fuel elements. The uranium may be enriched to any degree, up to 100%, in the U-235 isotope.

(2) Maximum quantity of material per package

375 grams of U-235 per package as clad fuel element. The net weight of the contents shall not exceed 265 pounds.

(3) Contents are limited to one A_2 quantity based on the actual isotopic values for all constituent nuclides of the loaded contents.

(c) Criticality Safety Index (CSI): 100

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with Chapter 7 of the application dated October 16, 2009, with the exception that the CSI must be in accordance with Condition No. 5(c).

(b) The package must be maintained in accordance with Chapter 8 of the application dated October 16, 2009, with the exception that welding repairs are not authorized.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. The fabrication of new packages is not authorized after April 1, 1999.

9. Expiration date: November 30, 2014.

REFERENCES

BWXT Nuclear Products Division application dated December 23, 2003.
Supplements dated: May 30, 2008; January 29 and October 16, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Jennifer Davis, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 24, 2009.

NRC FORM 618
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U.S. NUCLEAR REGULATORY COMMISSION

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
U.S. Department of Energy
application dated May 30, 1991,
as supplemented

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Inner HFIR Unirradiated Fuel Element Shipping Container, and Outer HFIR Unirradiated Fuel Element Shipping Container

(2) Description

Packaging for unirradiated fissile radioactive material as fuel elements for the High Flux Isotope Reactor (HFIR). The containers are right circular cylinders with an 11-gauge carbon steel shell. The lid is attached to the container with sixteen 3/8-16x1-inch steel bolts. The steel shell is filled with stacked fir plywood rings. The plywood rings form a central cavity which is lined with 1-inch thick polyethylene foam.

The packaging for the inner HFIR fuel element has overall dimension of 25 inches OD by 45 inches high, a 10-7/8-inch diameter by 30-1/4-inch deep cavity, and a 660 pound gross weight.

The packaging for the outer HFIR fuel element has overall dimensions of 31.5 inches OD by 45.75 inches high, a 17-3/8-inch diameter by 31-1/8-inch deep cavity, and a 1,050 pound gross weight.

(3) Drawings

(i) The packaging for the inner HFIR fuel is constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-20978-EL-003E, Rev. F, and M-20978-EL-008E, Rev. C.

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5. (a) (3) Drawings (continued)

(ii) The packaging for the outer HFIR fuel is constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-20978-EL-002E, Rev. E, and M-20978-EL-008E, Rev. C.

(b) Contents

(1) Type and form of material

Uranium as U_3O_8-Al cermet, enriched up to 95% in the U-235 isotope, and clad in aluminum, 10-mils thick, and:

(i) For the packaging described in 5(a)(3)(i), the contents are described in ORNL/RRD/INT-37-V3, "Specification for High Flux Isotope Reactor Fuel Elements RRD-FE-3," Revision 4, and in the following Oak Ridge National Laboratory Drawing Nos.: E-42118, Rev. R; E-42112, Rev. H; D-42113, Rev. G; D-42114, Rev. K; and E-42117, Rev. H.

(ii) For the packaging described in 5(a)(3)(ii) the contents are described in ORNL/RRD/INT-37-V3, "Specification for High Flux Isotope Reactor Fuel Elements RRD-FE-3," Revision 4, and in the following Oak Ridge National Laboratory Drawing Nos.: E-42126, Rev. N; E-42120, Rev. H; D-42121, Rev. H; D-42122, Rev. K; and E-42125, Rev. J.

(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i) not more than 2.63 kg of U-235.

(ii) For the contents described in 5(b)(1)(ii) not more than 6.88 kg of U-235.

(c) Criticality Safety Index 0.4

6. The lid lifting attachments must be blocked as shown on Martin Marietta Energy Systems, Inc., Drawing No. M-20978-EL-009E, Rev. 2, to prevent inadvertent use of the attachments during transport.

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7. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) Each package shall be maintained in accordance with the Maintenance Program in Chapter 8 of the application;

   (b) Each package shall be operated and prepared for shipment in accordance with the Operating Procedures in Chapter 7 of the application; and

   (c) The fuel element shall meet the fabrication inspection requirements of ORNL/RRD/INT-37-V3, "Specification for High Flux Isotope Reactor Fuel Elements RRD-FE-3," Revision 4.

8. Use of packaging fabricated after December 31, 1976, is not authorized.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Transport by air of fissile material is not authorized.

11. Expiration date: October 31, 2017.

REFERENCES

U.S. Department of Energy Application dated May 30, 1991.

Supplements dated: February 26, 1992; April 2, 1993; September 23, 1996; September 2, 1998; February 24, 2000; February 4, 2002; August 20, 2007; and October 29, 2007; June 28, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Bernard H White IV

for Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 25, 2012.

NRC FORM 618
(8-2000)
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for 235R001 Shipping Container dated August 11, 1970, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 235R001

(2) Description

The 235R001 shipping container structure is horizontal, having an oblong cross section and is fabricated from 0.104-inch thick carbon sheet steel. The container is 313 inches long and has a maximum weight of 4,640 pounds, empty. The oblong cross-section dimensions are approximately 35.5 inches high by 33.0 inches wide. The container was originally designed to ship unirradiated fuel modules of the AIG/A4W type. Subsequently, the container has been adapted to ship standard size or partial S8G fuel modules by use of a special frame assembly and cradle clamps, S3G-3 refueling modules using cell support assemblies, rodded or unrodded D1G fuel modules, rodded or unrodded D2W fuel cells, rodded S9G fuel cells and rodded A1B fuel cells. The loaded container maximum weight is 17,200 pounds.

(3) Drawings

The packaging is constructed in accordance with Container Research Corporation Drawing No. 235R001, Rev. BC, and Bettis Atomic Power Laboratory Drawing No. 6292E98, Rev. B.

5.(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies of the following types:

(i) A1G reactor cell without upper mechanism and with control rod, leadscrew and shipping fixture installed on rodded type modules.

(ii) Standard size S8G reactor cluster with regular or substitute support adapters and regular control rods. If only one cell is shipped per container, a dummy load shall be installed for balance.

(iii) Partial size S8G reactor cluster with regular or substitute support adapters and regular control rods. If only one cell is shipped per container, a dummy load shall be installed for balance.

(iv) S3G-3 refueling cells, with a maximum of one 0-1 reactor cell assembly per container.

(v) D2W side or central fuel cell and shear block with control rod inserted in rodded fuel cell.

(vi) D2W corner fuel cell, with shear block, unrodded.

(vii) S9G type fuel cell with control rod inserted.

(viii) A1B type fuel cell with control rod inserted.

(2) Maximum quantity of material per package

(i) One fuel assembly as described in 5(b)(1)(i), 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), or 5(b)(1)(viii).

(ii) Two fuel assemblies as described in 5(b)(1)(ii), 5(b)(1)(iii), or 5(b)(1)(iv).

5.(c) Criticality Safety Index

(1) For contents described in 5.(b)(1)(i), 5.(b)(1)(ii), 5.(b)(1)(iii), 5.(b)(1)(iv), 5.(b)(1)(v), and 5.(b)(1)(vi) and limited in 5.(b)(2)(i) and 5.(b)(2)(ii): 25.0

(2) For contents described in 5(b)(1)(vii), 5(b)(1)(viii), and limited in 5(b)(2)(i): 100.0

6. The contents as described in 5.(b)(1)(i) through 5.(b)(1)(vi) and limited in 5.(b)(2) shall be designated as B(U)F.

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7. Transport by air of fissile material is not authorized.

8. Expiration date: April 30, 2015.

REFERENCES

Safety Analysis Report for 235R001 Shipping Container, WAPD-OP(R)RD-357 dated August 11, 1970.

Supplements: Knolls Atomic Power Laboratory letter A1G 25-159, dated October 2, 1970. Bettis Atomic Power Laboratory letters WAPD-0P(R)RD-444, dated October 9, 1970; WAPD-OP(R)RD-476, dated October 26, 1970; and WAPD-OP(R)RD-488, dated October 30, 1970. Knolls Atomic Power Laboratory letters AlG 25-181, dated April 9, 1971; and A1G 25-191, dated May 11, 1971. Bettis Atomic Power Laboratory letters WAPD-OP(R)C-94, dated May 16, 1972; WAPD-OP(R)C-199, dated December 13, 1972; and WAPD-OP(R)C-229, dated March 6, 1973. Naval Reactors letters G#5078, dated January 26, 1976; G#5776, dated September 8, 1977; G#5905, dated January 23, 1978; G#5923, dated February 22, 1978; G#6095, dated August 17, 1978; G#6208, dated March 8, 1979; G#6373, dated September 4, 1979; G#68l3, dated October 17, 1980; G#C85-0467, dated July 17, 1985; G#C88-8112, dated October 18, 1988; G#90-03655, dated August 10, 1990; G#92-03560, dated June 15, 1992; G#96-03371, dated March 15, 1996, G#C97-03444 dated April 8, 1997; G#C99-03514, dated June 1, 1999; G#C99-03688, dated December 30, 1999; G#C02-0750, dated April 8, 2002; G#C03-00273, dated January 24, 2003; G#C03-01695, dated July 14, 2003; G#C07-02462, dated December 18, 2007; G#09-02803, dated June 11, 2009; G#09-04797, dated November 20, 2009; G#C10-00794, dated March 31, 2010; and G#C10-03819, dated September 30, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert K. Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 1/19/11

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global Inc., application dated
August 16, 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: 702

(2) Description

The Model No. 702 is composed of a stainless steel cylinder containing a depleted uranium shield and a cover assembly sealed by a neoprene gasket. The cover assembly flange is anchored to the cask with six bolts. The overall dimensions of the Model No. 702 are 19 ¾" x 21" x 19" (502 mm x 533 mm x 483 mm) and the maximum weight is 410 pounds (186 kg) including contents. The Model No. 702 is mounted on a rectangular carbon steel skid and secured to the skid by a tie-down system. A protective carbon steel cage, placed over the Model No. 702, is also bolted to the skid at each corner.

There is no locking assembly on the Model No. 702. Sources are secured in the shielded position by the cover assembly and two of the six securing bolts of the cover assembly are seal-wired with a tamper indicator seal. Metallic canisters and inserts used for holding special form sources are limited to non-pyrophoric metals with a melting temperature at or above 800°C.

(3) Drawings

The Model No. 702 and other system components are constructed in accordance with QSA Global Drawing No. R70290, sheets 1 to 9, Revision W.

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5.
(b) Contents

(1) Type and form of material

Iridium-192, Selenium-75, Cesium-137, and Ytterbium-169 as special form sealed sources.

(2) Maximum quantity of material per package:

Isotopes	Output Activity
Cs-137	500 Ci (18.5 TBq)
Ir-192	15,000 Ci (555 TBq)
Se-75	10,000 Ci (370 TBq)
Yb-169	10,000 Ci (370 TBq)

Output curies are determined by measuring the source output at 1 meter from the device and expressing its activity in curies. (Procedures reference: American National Standards Institute N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

(3) Maximum decay heat per package:

130 watts

(4) Maximum weight of contents:

0.44 pounds (200 grams)

6. The name plate must be fabricated of material capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

(b) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Expiration Date: June 30, 2018

REFERENCES

QSA Global Inc., application dated August 16, 2010.

Supplements dated: September 21, October 28, 2010, November 9, 2012, and January 16, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 3/22/2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global Inc., application dated August 30, 2010, Revision No. 11, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 741-OP

(2) Description

The Model No. 741-OP consists of a gamma ray projector within a protective carbon steel container. The protective container is of welded steel construction and is approximately 32 inches (813 mm) long, 19 inches (483 mm) wide, and 18.5 inches (470 mm) high. Polyurethane foam and wood inserts locate the Model No. 741 series projectors in the center of the container and provide impact protection.

The 741 series projectors include the Model Nos. 741, 741A, 741B, 741E, 741AE, and 741BE. The primary components of the projector consist of an outer steel shell, internal bracing, polyurethane foam, depleted uranium shield, and an "S" tube. The radioactive contents are securely positioned in the "S" tube by a source cable locking device and shipping plug. A ¼-inch thick steel shipping plate is bolted over the source locking mechanism for additional protection during transport. Tamper-proof seals are provided on the outer steel container. The dimensions of the projector are approximately 19 1/8 inches (486 mm) long, 13 7/8 inches (352 mm) wide, and 11 3/8 inches (289 mm) in height. The maximum weight of the package is 510 pounds (231 kg), and the maximum weight of the projector is 360 pounds (162 kg).

(3) Drawings

The package is constructed in accordance with QSA Global Inc., Drawing Nos. R74190, Rev. M, sheets 1-7; R741-OP, Rev. J, sheets 1-7.

5.
(b) Contents

(1) Type and form of material

Cobalt-60 as sealed source which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package:

Co-60: 33 curies (1.22 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 1.30 R/(h-Ci) (Ref: American National Standards Institute N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

(3) Maximum weight of contents: 0.09 pounds (40 grams)

The content weight value is based on the weight of the full source wire assembly that can be transported in the package

(4) Maximum decay heat: 0.55 watts

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The source assembly lock, lock cap and safety plug must be fabricated of materials capable of resisting a 1475°F fire environment for one half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplate shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application; and

(b) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. No welding repair or no new fabrication of the projector is authorized.

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U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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11. Revision No. 20 of this certificate may be used until October 31, 2011.

12. Expiration date: October 31, 2015.

REFERENCES

QSA Inc., application dated August 30, 2010, Revision No. 11.

Supplement dated: September 28, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 12, 2010.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

General Atomics
P.O. Box 85608
San Diego, CA 92186-5608

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated October 4, 1995, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TRIGA-I

(2) Description

TRIGA fuel element shipping container. The outer packaging is a steel drum, approximately 22.5 inches in diameter by 39-1/4 inches high. The inner vessel is a 5-inch Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 31 inches in height with a 1/4-inch thick wall and a 5-inch inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-inch thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-inch diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lbs/ft^3. Maximum gross weight including contents is approximately 235 pounds.

(3) Drawing

The packaging is constructed in accordance with General Atomic Company Drawing No. TOS396C160, Rev. G.

5. (b) Contents

(1) Type and form of material

TRIGA fuel elements containing uranium-zirconium-hydride or erbium-uranium-zirconium-hydride with nominal fuel composition (excluding erbium content) as described in Table A.1-1 of the October 4, 1995 application, and clad with stainless steel, aluminum or incoloy. Uranium enriched to a maximum 93.5 w/o in the U-235 isotope. The H to Zr atomic ratio within the fuel meat must not exceed 1.65.

(2) Maximum quantity of material per package

U-235 content not to exceed 1.39 kg, contained in a maximum of 7 1.5-inch diameter fuel elements, or a maximum of 25 0.5-inch diameter fuel elements, with nominal fuel composition (excluding erbium content) as described in Table A.1-2 (Rev. 1) of the October 4, 1995, application. For enrichments greater than 5 weight percent U-235, uranium content not to exceed an A_2 quantity.

(c) Criticality Safety Index 0.4

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 8 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 9 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Expiration date: December 31, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

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REFERENCES

General Atomic Company application dated October 4, 1995.

Supplements dated: December 5, 1995, October 16, 2000, November 16, 2005, and November 22, 2010. |

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Johnson

Robert Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 12/14/10

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9035	22	71-9035	USA/9035/B(U)-96	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., application dated August 30, 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 680-OP

(2) Description

The Model No. 680-OP consists of a gamma ray projector within a protective steel container. The protective container is of welded steel construction and is approximately 32 inches (813 mm) long, 19 inches (483 mm) wide, and 18-1/2 inches (470 mm) high. Polyurethane foam and wood inserts locate the Model 680 series projectors in the center of the container and provide impact protection.

The 680 series projectors include the Model Nos. 680, 680E, 680A, 680AE, 680B and 680BE. The primary components of the projector consist of an outer steel shell, internal bracing, polyurethane foam, depleted uranium shield, and an "S" tube. The radioactive contents are securely positioned in the "S" tube by a source cable locking device and shipping plug. A 1/4-inch thick steel shipping plate is bolted over the source locking mechanism for additional protection during transport. Tamper-proof seals are provided on the outer steel container. The dimensions of the projector are approximately 21 inches (530 mm) long, 14-5/8 inches (372 mm) wide, and 11-13/16 inches (300 mm) high. The maximum weight of the package is 615 pounds (279 kg), and the maximum weight of the projector is 465 pounds (211 kg).

(3) Drawings

The packaging is constructed in accordance with QSA Global Inc., Drawing No. R68090, sheets 1-7, Rev. M, and R680-OP, sheets 1-7, Rev. M.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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5.(b) Contents

(1) Type and form of material:

Cobalt-60 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package:

110 curies (4.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 1.30 R/h-Ci cobalt-60 at 1 meter (Ref: American National Standards Institute, N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography").

(3) Maximum weight of contents: 0.09 pounds (40 grams)

6. The source shall be secured in the shielded position of the packaging by the source assembly lock; lock cap and safety plug assembly. The source assembly lock, lock cap and safety plug assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application; and

(b) Each package shall be operated and prepared for shipment in accordance with Section 7 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. No support for replacement of "non-Posilock style" assemblies is allowed after September 2010. However, future manufacture and production of all package components, including the inner device, is authorized.

11. Revision No. 21 of this certificate may be used until October 31, 2011.

12. Expiration date: October 31, 2015.

REFERENCES

QSA Global, Inc., application dated August 30, 2010.

Supplements dated: September 21 and 28, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 12, 2010.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc. application dated February 28, 2001, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: C-1

(2) Description

The packaging consists of a steel inner unit inside an outer overpack. The inner unit is a rectangular box approximately 9" high x 7.5" wide x 7.5" deep around a depleted uranium shield. All fittings and source locking components are protected and enclosed within the 1/8" carbon steel outer shell. The inner receptacle consists of a uranium shield equipped with two closed bottom Zircalloy or titanium "J" tubes, each of which may house one "pigtail type" special form source. The overpack is a 12-gallon, 20- or 22-gage steel drum partially filled with foam. The weight of the inner unit is 51 to 70 lbs. The weight of the overpack is 19 to 22 lbs. Up to 8 lbs. of ancillary equipment may be included within the overpack. The maximum gross weight of the package is 100 lbs.

(3) Drawings

The packaging is constructed in accordance with Source Production and Equipment Company, Inc., Drawing Nos. B322000, Rev. (3); B311000, Rev. (2); B311001, Rev. (1); and B311002, Rev. (0).

5.(b) Contents

(1) Type and form of material

Iridium-192, Selenium-75, and Ytterbium-169 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

Two sealed sources with a combined activity not to exceed 300 curies (11.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/(h-Ci) Iridium-192.

6. Tungsten shield pads, with dimensions up to approximately 2-inches diameter and ½-inch thick, may be welded to the inside surface of the source changer housing.

7. The nameplate shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application, as supplemented; and

(b) The package shall meet the Acceptance Tests and be maintained in accordance with the Maintenance Program of Section 8.0 of the application, as supplemented.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Revision No. 11 of this certificate may be used until October 31, 2012.

11. Expiration date: October 31, 2016.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9036	12	71-9036	USA/9036/B(U)-96	3	OF	3

REFERENCES

Source Production and Equipment Company, Inc., applications dated September 27, 2000, and February 28, 2001.

Supplements dated: April 11 and May 11, 2001; May 1, June 14 and June 23, 2006; and May 26, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: July 13, 2011.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9037	14	71-9037	USA/9037/AF	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

General Atomics
P.O. Box 85608
San Diego, CA 92186-5608

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated October 4, 1995, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TRIGA-II

(2) Description

TRIGA fuel element shipping container. The outer packaging is a steel drum, approximately 22.5 inches in diameter by 57.5 inches high. The inner vessel is a 5-inch Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 50 inches in height with a 1/4-inch thick wall and a 5-inch inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-inch thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-inch diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lbs/ft^3. Maximum gross weight including contents is approximately 330 pounds.

(3) Drawing

The packaging is constructed in accordance with General Atomic Company Drawing No. TOS396C161, Rev. F.

5. (b) Contents

(1) Type and form of material

Special function TRIGA fuel elements containing uranium-zirconium-hydride or erbium-uranium-zirconium-hydride whose fuel portion has nominal compositions (except erbium content) as described in Table A.1-1 of the October 4, 1995, application, and clad with stainless steel, aluminum or incoloy. Uranium enriched to a maximum 93.5 w/o in the U-235 isotope. The H to Zr atomic ratio within the fuel meat must not exceed 1.65.

(2) Maximum quantity of material per package

U-235 content not to exceed 1.39 kg, contained in a maximum of 7 1.5-inch diameter fuel elements, or a maximum of 25 0.5-inch diameter fuel elements, whose fuel portion has nominal compositions (except erbium content) as described in Table A.1-2 (Rev. 1) of the October 4, 1995, application. For enrichments greater than 5 weight percent U-235, uranium content not to exceed an A_2 quantity.

(c) Criticality Safety Index 0.4

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 8 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 9 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Expiration date: December 31, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9037	14	71-9037	USA/9037/AF	3 OF	3

REFERENCES

General Atomic Company application dated October 4, 1995.

Supplements dated: December 5, 1995, October 16, 2000, November 16, 2005, and November 22, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Johnson

Robert Johnson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 12/14/10

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9056	13	71-9056	USA/9056/B(U)	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Source Production and Equipment Company, Inc. application dated March 24, 2000, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: SPEC 2-T

(2) Description

A steel encased, uranium shielded Gamma Ray Projector. Primary components consist of an outer steel shell, internal bracing, depleted uranium shield, and a Zircalloy "S" tube. The contents are securely positioned in the Zircalloy "S" tube by a source cable locking device and shipping plug. The unit resembles a rectangular box approximately 13-3/8" long by 4-11/16" high by 4-3/8" wide with a maximum gross weight of 56 pounds.

(3) Drawings

The packaging is constructed in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 12688-1, Rev. (2); 788-1, Rev. (4); and 788-2, Rev. (0).

The packaging may also be as shown in Source Production and Equipment Company Drawing No. 1000, Rev. (0), provided fabrication was completed prior to June 8, 1989.

The overpack is a 12 gallon open head 20 or 22 gauge National Motor Freight Classification 100-H, or succeeding issues, Item 260 steel drum constructed in accordance with Source Production and Equipment Company, Inc. Drawing No. 53189-2, Rev. (2).

5.(b) Contents

(1) Type and form of material

Iridium 192 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

225 curies (8.3 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/(h-Ci) Iridium-192.

6. The source must be secured in the shielded position of the packaging by the shipping plug, source assembly, and locking device. The shipping plug and source assembly used must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The source assembly ball stop must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplates must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. For transportation of more than 1.7 TBq (45 curies) (output) per package in private carriage the shipment must be in accordance with 49 CFR 173.441(b).

9. For transportation of more than 1.7 TBq (45 curies) (output) per package by a common carrier, the package must be within a protective overpack as described and constructed in accordance with 5(a)(3).

10. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application, as supplemented; and

(b) The package shall be maintained in accordance with the Maintenance Program of Section 8.0 of the application, as supplemented.

11. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17.

12. Fabrication of the package must have been completed by April 1, 1999.

13. Revision No. 12 of this certificate may be used until January 30, 2011.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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14. Expiration date: April 30, 2015.

REFERENCES

Source Production and Equipment Company, Inc. application dated March 24, 2000.

Supplements dated: March 30, 2000, March 14, 2005, and December 16, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: January 12, 2010.

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9099	11	71-9099	USA/9099/B(U)F-85	1	OF 3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

ATR Fresh Fuel Shipping Container
Safety Analysis Report, INEL-94/0275
Application dated January 27, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: ATR

(2) Description

The inner container is a right parallelepiped, 69-7/16 inches x 26-13/16 inches x 6-15/16 inches, constructed of 3/4-inch plywood, covered with 16-gauge steel. The top and bottom are lined with high density polyethylene foam and with a 0.020-inch cadmium plate. Wood spacers covered with sponge rubber and with a 0.020-inch thick cadmium plate provide separation for four fuel assemblies. Positive closure is provided by a continuous hinge, and two wire sealed hinge pins provide access.

The inner container is enclosed within an overpack, 73-15/16 inches x 31-3/4 inches x 11-3/16 inches, constructed of 1-inch plywood, framed by steel angle members and covered with 18-gauge steel. Aluminum, honeycomb impact limiters are fixed to the ends of the overpack. Positive closure of the overpack is provided by four hinge pins which are secured in place using 1/16-inch diameter cotter pins. The package weight is approximately 853 pounds.

(3) Drawings

The packaging is fabricated in accordance with EG&G Idaho, Inc., Drawing No. 445721, Sheets 1, 2, and 3; and EG&G Idaho, Inc., Drawing No. 445722, Sheets 1 and 2.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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5. (b) Contents

(1) Type and form of material

Unirradiated ATR fuel elements. Each element contains 19 formed fuel plates, clad in Aluminum 6061. Each element contains a maximum of 1,100 grams of U-235 in uranium that is enriched to a maximum of 94 wt% in the U-235 isotope.

(2) Maximum quantity of material per package

Up to four (4) unirradiated ATR fuel elements. Total U-235 content not to exceed 4,400 grams per package.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on
label for nuclear criticality control: 4.2

6. The contents must be maintained within its compartment and the active fuel length must be completely within the region of the cadmium covered spacers. Wood spacers may be used to accomplish this.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

8. Air transport of fissile material is not authorized.

9. Revision No. 10 of this certificate may be used until January 7, 2014.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration date: January 31, 2014.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9099	11	71-9099	USA/9099/B(U)F-85	3	OF	3

REFERENCES

ATR Fresh Fuel Shipping Container Safety Analysis Report, INEL-94/0275, January 27, 1999.

Supplements dated: February 18, 1999, April 27, 2000, December 5, 2003, and November 19, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 1/6/09

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9148	9	71-9148	USA/9148/B(U)	1	OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., application dated October 1, 2012, Revision No. 9, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 770

(2) Description

A steel encased uranium shielded source changer for radiographic sources in special form. The source changer measures 23 inches long, 24 inches wide, and 19.75 inches high. The radioactive source assembly is housed in a titanium "S" tube. The "S" tube is surrounded by depleted uranium metal shield. The depleted uranium shield assembly is encased in two steel containers. The void space between the depleted uranium shield assembly and the inner container is filled with a rigid polyurethane foam. The gross weight of the container is 970 pounds.

(3) Drawings

The packaging is constructed in accordance with QSA Global, Inc., Drawing No. R77091 - sheets 1 through 6, Rev. A.

(b) Contents

(1) Type and form of material

(i) Sources which meet the requirements of special form radioactive material. Authorized isotopes include Ir-192, Co-60, Sc-46, and Cs-137

(2) Maximum quantity of material per package

Isotope	Output Curies
Ir-192	1,000
Co-60	800
Sc-46	800
Cs-137	1,000

(3) Maximum decay heat per package:

14 watts.

6. Name plates must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the operating procedures in the application; and

(b) The package shall be maintained in accordance with the maintenance program in the application.

8. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.17.

9. No new fabrication of the package is authorized.

10. Expiration date: March 31, 2018.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9148	9	71-9148	USA/9148/B(U)	3	OF	3

REFERENCES

QSA Global Inc., application dated October 1, 2012, Revision No. 9.

Supplements dated: March 21 and 25, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 4, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9157	13	71-9157	USA/9157/B(U)-96	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Industrial Nuclear Company
14320 Wicks Blvd.
San Leandro, CA 94577

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Industrial Nuclear Company application
dated June 8, 1999, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: IR-100

(2) Description

The Model No. IR-100 package is approximately 8.87 inches long, 4.5 inches wide, and 8.5 inches high. The radioactive material contents consist of iridium-192 in source assemblies that meet the requirements for special form material. The source assemblies are positioned within a zircalloy or titanium "S" tube within the IR-100. The "S" tube is surrounded by a shield assembly made of depleted uranium. The uranium shield assembly is encased in a stainless steel housing. The space between the uranium shield assembly and the stainless steel casing is filled with a rigid polyurethane foam. The maximum weight of the IR-100 exposure device is 53 pounds and the maximum shield weight is 38 pounds.

(3) Drawings

The packaging is constructed in accordance with Industrial Nuclear Company Drawing Nos.: IR 100-1A, Rev. 5 and IR 100-1B, Rev. 2.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed sources that meet the requirements of special form radioactive material.

CERTIFICATE OF COMPLIANCE
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5. (b) Contents (continued)

    (1) Maximum quantity of material per package

    120 (output) curies

    Output curies are determined in accordance with American National Standard N432-1980, “Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.”

6. The source must be secured in the shielded position of the packaging by the shipping plug, source assembly lock, and lock cap. The shipping plug, source assembly lock, and lock cap used must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The ball stop of the source assembly lock must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate on the exposure device must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) The package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application; and

    (b) Each package shall be operated and prepared for shipment in accordance with the operating procedures in accordance with Section 7 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Revision No. 12 of this certificate may be used until October 31, 2010.

11. Expiration date: October 31, 2014.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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REFERENCES

Industrial Nuclear Company application dated June 8, 1999.

Supplements dated: June 9, August 6 and September 14, 1999; October 24, 2003; August 20, 2004; and March 22, 2007.

Renewal dated: August 20, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

B. White for

Steven Baggett, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 30, 2009

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9168	19	71-9168	USA/9168/B(U)-96	1	OF	4

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Energy*Solutions* Services, Inc.
Suite 100, Center Point II
100 Center Point Circle
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Energy*Solutions* application, Revision No. 3, dated July 20, 2012, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. 8-120B

(2) Description

A cylindrical carbon steel, lead shielded, packaging designed for the transport of radioactive waste materials. The packaging has four tie-down and two removable lifting devices and is transported in the upright position with cylindrical foam-filled impact limiters, 102 inches outside diameter (OD), installed at each end of the packaging. The overall height of the package with the impact limiters attached is 132 ¼ inches. The maximum gross weight of the package is approximately 74,000 pounds (lbs), as follows:

Packaging Body	42,220 lbs
Lid	7,080 lbs
Payload	14,430 lbs
Impact Limiters	4,860 lbs (each)
Thermal Shield	250 lbs

The cavity of the packaging is a right circular cylinder with an internal diameter of 61 13/16 inches and a height of 74 7/8 inches. The package body consists of two shells, both fabricated of ASTM A516, Grade 70 steel. The annular space between the 1½ inch thick external shell and the ¾ inch thick internal shell is filled with 3.35 inch thick lead. The primary lid is attached to the packaging body with twenty equally spaced 2-inch diameter bolts. A supplemental 14 gauge stainless steel sheet is welded to the inside surface of the primary lid.

5(a)(2) Packaging Description (Continued)

The centered secondary lid is attached to the primary lid with twelve equally spaced 2-inch diameter bolts. A thermal shield, consisting of two polished stainless-steel plates separated by a thin air gap, is attached to the secondary lid lifting lugs with hitch-pins. A 12 gauge stainless steel liner is welded to the cavity of the package and the lid surface to protect all accessible areas from contamination.

The containment boundary consists of the inner shell, the upper baseplate, the bolting ring, the inner O-rings of the lids, and the lids. Test ports for leak testing of the package are located between the twin O-ring seals for both the primary and secondary lids.

There are three configurations of the packaging; Configuration 1 includes a drain port, sealed with the insertion and welding of a rod in the drain port; Configuration 2 does not have a drain port; Configuration 3 does not have a drain port and the packaging's base plate is fabricated differently than for Configurations 1 and 2.

(3) Drawings

The packaging is constructed and assembled in accordance with Energy*Solutions* Drawing Nos. C-110-E-0007, sheets 1-6, Revision No. 18.

The secondary lid thermal shield is constructed in accordance with Energy*Solutions* Drawing No. DWG-CSK-12CV01-EG-0001-01, Rev. 3.

(b) Contents

(1) Type and form of material

(i) Byproduct, source, or special nuclear material in the form of dewatered resins, solids, including powdered or dispersible solids, or solidified material, contained within secondary containers; or

(ii) Radioactive material in the form of activated metals or metal oxides in solid form contained within secondary containers.

(2) Maximum quantity of material per package

(i) Activity not to exceed 3,000 times a Type A quantity along with the following limits:

(1) The limit determined per the procedure in Attachment 1 to Chapter No. 7 of the application for beta and gamma emitting radionuclides.

(2) The mass limits for fissile materials as prescribed by 10 CFR 71.15 for exempting materials from classification as fissile material.

(3) A maximum total package neutron source of $1x10^5$ neutrons/second for materials that produce neutrons (other than fissile materials) through any means, including spontaneous fission, alpha-neutron reactions, and gamma-neutron reactions.

(ii) Maximum decay heat: 200 Watts.

(iii) Maximum weight of contents: 14,430 lbs including shoring and secondary containers.

(iv) Powdered or dispersible solid materials must have a mass of at least 60 grams or a specific activity of 50 A_2/g or less.

(v) Explosives, corrosives, and non-radioactive pyrophorics are prohibited. Pyrophoric radionuclides may be present only in residual amounts below 1 weight per cent.

(vi) Materials that may auto-ignite or change phase at temperatures below 350°F, not including water, shall not be included in the contents. Also, contents shall not include any materials that may cause any significant chemical, galvanic, or any other reaction.

(vii) Powdered radioactive materials shall not include radioactive forms of combustible metal hydrides or combustible element metals, i.e., magnesium, titanium, sodium, potassium, lithium, zirconium, hafnium, calcium, zinc, plutonium, uranium, and thorium, or combustible non-metals, e.g., phosphorus.

(viii) Contents may only include quantities of boron, lithium, or beryllium such that these materials do not constitute quantities sufficient to be considered as a bulk material for a payload item or a portion of that payload item.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(i) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application,

(ii) The packaging must be tested and maintained in accordance with the acceptance tests and maintenance program described in Chapter 8 of the application.

7. Two independent physical verifications of the secondary container's closure system shall be performed as part of the package loading operations to ensure proper closing so as to prevent release of material from the secondary container.

8. Shipments of powdered radioactive materials shall be performed only when the most recent periodic leak test meets the requirements of Section 4.8 of Chapter 4 of the application.

9. Except for close fitting contents, shoring must be placed between the secondary containers, or activated components, and the package cavity's walls to prevent both radial and axial movement during transport.

10. Flammable gas (hydrogen) concentration is limited to less than 5% in volume. Compliance with this concentration limit is determined by the methodology used in NUREG/CR-6673.

11. A pre-shipment leak test is required before each shipment of Type B quantities.

12. The package may be used until August 31, 2013, with the seals authorized in Revision No. 17 of the Certificate, in accordance with the Addendum of the July 20, 2012, application.

13. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

14. Expiration date: August 31, 2017.

REFERENCES

Energy*Solutions* application, Revision No. 3, and Addendum dated July 20, 2012.

Supplements dated July 26 and August 10, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 23, 2012

NRC FORM 618 (8-2000) 10 CFR 71					U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9184	9	71-9184	USA/9184/B(U)	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA Federal Services, LLC
505 S. 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Packaging, Inc., consolidated application dated March 31, 1989, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PAS-1

(2) Description

The packaging consists of a primary containment vessel (20.5" OD x 23.4" OH) enclosed inside a secondary containment vessel and radiation shield (32.5" OD x 39.0" OH). The 15 milliliter water sample is contained within a undefined sample cask. Additionally, four iodine collection cartridges and four offgas vials are maintained inside the foam shoring above the sample cask. Loose vermiculite surrounds the perimeter of the sample cask to absorb the water sample should leakage occur. Completely surrounding the secondary containment vessel and radiation shield is a foam filled steel encased overpack (48.0" OD x 66.0" OH) which provides impact and thermal protection.

The primary containment vessel, which is constructed of 304 stainless steel varying in thickness from 3/4" to 1.25", is provided with double Viton O-ring seals and a sealed test port between the seals for leak testing. The assembly is secured with eight, 3/8"-16 UNC x 8" long screws.

The secondary containment vessel and radiation shield provides 0.75" thick steel and 5.1" thick lead shielding in the radial direction, 2.0" thick steel and 5.1" thick lead shielding on the bottom, and 3.5" thick steel and 4.8" thick lead shielding on the top. The lid is secured with eight, 1.0"-8 UNC x 3.0 long bolts. The lid is sealed with two Viton O-rings with a sealed test port between the seals for leak testing.

5.(a) Packaging Continued

(2) Description continued

The overpack provides about 7.25" thick foam on the sides and about 13" on the top and bottom. The two halves of the overpack are held together by eight, 3/4"-10 UNC x 1.5" long bolts. A Neoprene gasket prevents rain water from entering the overpack.
The weight of the package including a maximum sample cask weight of 1,375 pounds, is about 12,800 pounds.

(3) Drawings

The package is constructed in accordance with Nuclear Packaging, Inc. Drawing No. X-20-218D, Sheets 1 and 2, Rev. C.

(b) Contents

(1) Type and form of material

(i) Radioactive material in form of liquid or gaseous samples in sample casks, cartridges and vials.

(ii) Byproduct and activation materials as solids and process solids or resins, either dewatered, solid, or solidified in secondary containers.

(2) Maximum quantity of material per package

50 Ci of mixed fission and activation products, 15 milliliters of liquid, one sample cask or secondary container and four cartridges and four vials.

6. In addition to the requirements of Subpart G of 10 CFR Part 71, each package prior to first use must meet the acceptance tests and criteria specified in Section 8.1, must be maintained in accordance with Section 8.2, and must be prepared for shipment in accordance with Chapter 7.0 of the application, and the supplement dated July 8, 1994.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17, provided the fabrication of the packaging was satisfactorily completed by April 1, 1999.

8. Transport by air of fissile material is not authorized.

9. Expiration date: August 31, 2014.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Nuclear Packaging, Inc., consolidated application dated March 31, 1989.

Supplement dated: April 7, 1989.

VECTRA Technologies, Inc., supplements dated: July 8, 1994 and January 30, 1998.

Transnuclear, Inc., supplement dated January 30, 1998.

Packaging Technology, Inc., supplements dated: April 30, 1999, March 16, 2004, and November 26, 2007.

AREVA Federal Systems, LLC, supplements dated: June 12, 2009 and October 31, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 18, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Industrial Nuclear Company, Inc.
14320 Wicks Blvd.
San Leandro, CA 94577

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Industrial Nuclear Company application dated July 1, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging
(1) Model No.: OP-100

(2) Description

The Model No. OP-100 package consists of either an IR-50 source changer, or an IR-100 exposure device, which is positioned within a 10 gallon drum. The drum is made of 20 gauge steel, and is closed with a 12 gauge closure ring and a 5/8 inch diameter steel bolt. Plywood members are used to position and support either the IR-50 or IR-100 within the steel drum.

The IR-50 source changer and the IR-100 exposure device are approximately 8.87 inches long, 4.5 inches wide, and 8.5 inches high. The radioactive material contents consist of iridium-192 in source assemblies that meet the requirements for special form material. The source assemblies are positioned within a zircalloy or titanium "S" tube within the IR-50 or IR-100. The "S" tube is surrounded by a shield assembly made of depleted uranium. The uranium shield assembly is encased in a stainless steel housing. The space between the uranium shield assembly and the stainless steel casing is filled with a rigid polyurethane foam. The maximum weight of the IR-50 source changer is 55 pounds, the maximum weight of the IR-100 exposure device is 53 pounds, and the maximum gross weight of the Model No. OP-100 package is 77 pounds.

(3) Drawings

The packaging is constructed in accordance with Industrial Nuclear Company Drawing Nos.: OP 100-1, Rev. 5, IR 50-1A, Rev. 3, IR 50-1B, Rev. 1, IR 100-1A, Rev. 5, and IR 100-1B, Rev. 2.

5. (b) Contents

(1) Type and form of material

Iridium-192 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

120 Ci (4.44 TBq) (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap, and the shipping plug (IR-100 only). The source assembly lock, lock cap, and the shipping plug (IR-100 only), must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The ball stop of the source assembly must engage the source assembly lock. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate on the overpack must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintain its legibility. The two vent holes in the side of the overpack must be covered with tape or rubber (plastic) plugs to prevent entry of rain water.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment in accordance with the Operating Procedures of Chapter 7 of the application and

(b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

NRC FORM 618 (8-2000) 10 CFR 71			U.S. NUCLEAR REGULATORY COMMISSION	
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
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10. Revision No. 7 of this certificate may be used until February 28, 2010.

11. Expiration date: February 28, 2014.

REFERENCES

Industrial Nuclear Company application dated July 1, 1999.

Supplements dated: September 14 and December 29, 1999; October 24, 2003; March 22, and July 12, 2007, and November 25, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: February 9, 2009

NRC FORM 618 (8-2000) 10 CFR 71				U.S. NUCLEAR REGULATORY COMMISSION
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20858

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis for Shipping S8G Power Units in the S-6213 Container, Rev. 7, dated June 16, 1975, as supplemented; and Safety Analysis for Shipment of S6W Shipboard Power Units in the Model 2 S-6213 PUSC, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: Model 1, S-6213 Power Unit Shipping Container

Model 2, S-6213 Power Unit Shipping Container

(2) Description

A power unit shipping container (PUSC) for shipment of a power unit complete with control rods and control rod drive mechanisms installed.

The Model 1 S-6213 PUSC consists of a carbon steel cylindrical shell approximately 9-1/4 feet in outside diameter by 39-1/2 feet long, including hemispherical steel end impact limiters, with 10-3/4-foot outside diameter central flanges joining the barrel and cover halves. The Model 2 S-6213 PUSC is of the same design as the Model 1, except that the primary container material is HY-80 steel. A power unit is supported in the PUSC by a centrally located thick circular steel plate (PU head) which is clamped between the central mating flanges of the PUSC and fastened by 94, 2-inch diameter high strength studs. The upper and lower extremities of the power unit cantilever into the barrel and cover halves without additional support. A lower support adapter is installed in the barrel end of the container during shipment of the S6W shipboard power unit. A shipping/lifting ring, a flange adapter, and a lower support adapter are installed in the container during shipment of the S9G shipboard power unit.

5.(a) Packaging (Continued)

(2) Description (Continued)

The PUSC is shipped in the horizontal position on a support frame which is secured to a specially built flatbed rail car.

The weight of the PUSC, including frame and contents, is approximately 429,900 pounds for shipment of the S6W shipboard power unit, and 329,000 pounds for shipment of the S9G shipboard power unit.

(3) Drawings

The Model 1 and Model 2 S-6213 PUSC are constructed in accordance with the Drawings included in the applications (see references, below).

5.(b) Contents

(1) Type and form of material

(i) Unirradiated S6W advanced fleet reactor shipboard power unit as described in Chapter 6 of "S6W Prototype Power Unit in S-6213 Power Unit Shipping Container Safety Analysis Report" WAPD-REO(c)1219, Revision 1, and containing uranium enriched in the U-235 isotope.

(ii) Unirradiated S9G shipboard power unit, as described in Chapter 6 of "S9G Shipboard Power Unit in S-6213 Power Unit Shipping Container Safety Analysis Report For Packaging," Revision 2, and containing uranium enriched in the U-235 isotope.

(2) Maximum quantity of material per package

For the Model 1 S-6213 PUSC:

One S9G Shipboard Power Unit.

For the Model 2 S-6213 PUSC:

One S6W Advanced Fleet Reactor Shipboard Power Unit, or
One S9G Shipboard Power Unit.

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5.(c) Criticality Safety Index (CSI):

Minimum CSI to be shown on
label for nuclear criticality control: 100

6. All control rods shall be restrained in the power unit fuel cells by the control rod holddown latches.

7. Transport by air of fissile material is not authorized.

8. For the Model 1 S-6213 PUSC, a nondestructive examination of the entire length of both inner and outer surfaces of the four tie-down support bracket-to-container wall butt welds shall be conducted prior to each loaded shipment.

(a) The nondestructive examination in accordance with a written procedure may be by either:

(1) The liquid penetrant method in accordance with:

(i) Article 6, Section V, ASME Code, or

(ii) MIL-STD-271E, "Nondestructive Testing Requirements for Metals," Section 5, October 31, 1973, or

(iii) NAVSHIPS 250-1500-1, "Welding Standard," Section 12.5

(2) or the magnetic particle method in accordance with:

(i) Article 7, Section V, ASME Code (Yoke Technique; Dry Particle Method; direct or rectified current), or

(ii) MIL-STD-271E, Section 4; specifically 4.3.1 (General) and 5.6.1(coatings), 4.3.3 (Dry Powder), 4.3.3.3.6 (Continuous), and 4.3.3.3 (Procedure) as excepted by using direct or rectified current, 4.3.3.3.3 (Yoke Technique), 4.3.2.5 (sensitivity and cleaning), and 4.3.1.3 (smoothness), or

(iii) NAVSHIPS 250-1500-1, Section 12.4, 12.4.1 (General), 12.4.3 (Dry powder), 12.4.3.3.2.1 (Yoke Technique) using direct or rectified current.

(b) If any indications, as defined in accordance with either:

(1) Paragraph UA-93(a), Appendix VIII, Division 1, Section VIII, ASME Code (with 7(b)(2)(i), above), or

(2) Paragraphs UA-72 and UA-73, Appendix VI, Division 1, Section VIII, ASME Code (with 7(b)(2)(i), above), or

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(3) Class 1 acceptance criteria of NAVSEA 0900-LP-003-8000, "Surface Inspection Acceptance Standards for Metal," with Change 2, July 1, 1974 (with 7(b)(1)(ii) or 7(b)(2)(ii), above), or

(4) NAVSHIPS 250-1500-1, Section 10.3.2 (with 7(b)(1)(iii) or 7(b)(2)(iii), above), as noted,

are detected, the packaging shall be repaired and reinspected prior to use and shall be inspected prior to each shipment thereafter. Any defects shall be reported in accordance with 10 CFR §71.95.

9. Expiration date: March 31, 2017.

REFERENCES

For the Model 1 S-6213 PUSC:

U.S. Naval Reactors application dated July 24, 1975.

Supplements dated: June 3, 1977; July 24, 1978; Naval Reactors letter G#C89-2838, dated May 22, 1989; Naval Reactors letter G#C90-03664, dated September 5, 1990; Naval Reactors letter G#92-03563, dated June 17, 1992; and Naval Reactors letter G#C92-03714, dated October 2, 1992; Naval Reactors letter G#97-03425, dated February 7, 1997; Naval Reactors letter G#C97-03614, dated September 29, 1997; Naval Reactors letter G#01-03619, dated December 11, 2001; Naval Reactors letter G#06-04833, dated December 18, 2006; Naval Reactors letter G#C08-00667, dated March 13, 2008; and Naval Reactors letter G#11-04084, dated September 20, 2011.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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For the Model 2 S-6213 PUSC:

U.S. Naval Reactors application G#C91-11165, dated December 19, 1991.

Supplements dated: Naval Reactors letter G#92-03563, dated June 17, 1992; and Naval Reactors letter G#C92-03714, dated October 2, 1992; Naval Reactors letter G#97-03425, dated February 7, 1997; Naval Reactors letter G#C97-03614, dated September 29, 1997; Naval Reactors letter G#01-03619, dated December 11, 2001; Naval Reactors letter G#06-04833, dated December 18, 2006; Naval Reactors letter G#C08-00667, dated March 13, 2008; and Naval Reactors letter G#11-04084, dated September 20, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Dated December 6, 2011

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9187	9	71-9187	USA/9187/B(U)-96	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc. application dated March 6, 2006, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 865

(2) Description

A steel encased, uranium shielded radiographic exposure device 5" OD x 12.25" long. The device is provided with 0.88" OD x 9.25" long handle and two 1.38" x 5.5" long triangular shaped legs. Primary components consist of an outer steel shell, internal bracing, depleted uranium shield, and a source tube. The contents are securely positioned in the source tube by a source holder assembly and actuator and locking assembly. Tamper-indicating seals are provided on the packaging and a 0.12-inch thick steel outer cover is bolted over the source actuator and locking assembly for additional protection during transport. The total weight of the package is approximately 59 pounds.

(3) Drawings

The packaging is constructed in accordance with QSA Global Drawing No. R86590, Sheets 1 through 8, Rev. J.

5.(b) Contents

(1) Type and form of material:

Iridium-192 as sealed source must meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package:

240 curies (8.9 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/(h-Ci) Iridium-192 at 1 meter, (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography").

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application, as supplemented.

(b) Each packaging shall be maintained in accordance with the Maintenance Program in Section 8 of the application, as supplemented.

(c) Fabrication of new packagings is not authorized.

(d) Repair or replacement of welds on existing packagings is not authorized.

7. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.17.

8. Revision No. 8 of this certificate may be used until August 30, 2014.

9. Expiration date: March 31, 2019.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

QSA Global, Inc., application dated March 6, 2006.

Supplement(s) dated: August 24, 2006; September 28, 2006; July 14, 2008; February 9, 2009 and March 13, 2009; and July 16, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 22, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech
1802 Fairfax Road
Greensboro, North Carolina 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Duratek, Inc., application dated June 9, 2005, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UX-30

(2) Description

Overpack for 30-inch uranium hexafluoride (UF_6) cylinders. The overpack is a right circular cylinder constructed of two stainless steel shells with the volume between the shells filled with 6-inch thick foam (7.8 - 9.8 PCF). A stepped and gasketed horizontal joint permits the top half of the overpack to be removed from the base. The package "halves" are secured with ten indexed, cross-locking "ball lock" pins. The overpack is 43.5" in diameter by 96" long. The maximum gross weight of the package is 8270 lbs.

Two types of 30 inch uranium hexafluoride cylinders may be carried in the UX-30 overpack. These are (1) an ANSI N14.1 Standard 30B cylinder, or (2) an ANSI N14.1 Standard 30C cylinder.

The ANSI N14.1 Standard 30C cylinder is essentially a 30B cylinder equipped with a Valve Protective Cover (VPC) that bolts over and protects the cylinder valve during transport. The VPC is a special design feature that provides additional assurance against the inleakage of water to the containment system and is an enclosure that retains any leakage.

5.(a) Packaging (continued)

(3) Drawings

The Model No. UX-30 packaging is fabricated in accordance with Energy*Solutions* Drawing No. C-110-B-57922-0002, sheets 1 through 3, Rev. 4.

(b) Contents

(1) Type and form of material

A. Unirradiated uranium, in the form of UF_6, with a U-235 mass percentage not to exceed 5 weight percent.

B. Reprocessed uranium, in the form of UF_6, with a U-235 mass percentage not to exceed 5 weight percent. The fission product gamma activity shall not exceed 4.4×10^5 MeV Bq/kgU. The alpha activity from neptunium and plutonium shall be less than 3.3×10^3 Bq/kgU.

(2) Maximum quantity of material per package

5,020 pounds UF_6 contained in an ANSI Standard N14.1 30B or 30C cylinder.
The maximum H/U atomic ratio for the UF_6 is 0.088.
The total activity in the package may not exceed 10^5 A_2.

(c) Criticality Safety Index (CSI)

Criticality safety index for the UX-30 overpack containing a standard ANSI N14.1 30B cylinder	5.0
Criticality safety index for the UX-30 overpack containing a standard ANSI N14.1 30C cylinder	0.0

Criticality safety index for the UX-30 overpack is not applicable to non-fissile or fissile-excepted contents.

6. The ANSI standard 30B, 30-inch diameter UF_6 cylinder, must be fabricated, inspected, tested and maintained in accordance with a) American National Standard N14.1-2001 or an earlier version of ANSI N14.1 in effect at the time of fabrication or b) American National Standard N14.1-2001 or an earlier version of ANSI N14.1 in effect at the time of fabrication and ISO 7195:1993(F). Cylinders must be fabricated in accordance with Section VIII, Division I, of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code and be ASME Code stamped.

7. The ANSI N14.1 Standard 30C cylinder (new or retrofitted cylinders) must be fabricated, inspected, tested, and maintained in accordance with ANSI N14.1-2001 Addendum 2-2004.

8. When the optional 4 lid lifting clips are used instead of the top lugs, the top lid (cover) must be lifted with a spreader bar (saddle).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) Prior to each shipment, the weather/dust seal gasket between the upper and lower shells must be inspected and must be replaced if inspection shows excessive wear or any defects to the gasket.

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

   (c) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

   (d) Prior to each shipment, the stainless steel components of the packaging, which include the ball-lock pins, must be visually inspected. Packagings in which stainless steel components show pitting, corrosion, cracking, or pinholes are not authorized for transport.

10. The 30-inch diameter UF_6 cylinder valve and plug threads may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

11. Transport by air is not authorized.

12. Packagings may be marked with Package Identification Number USA/9196/AF-96 until February 28, 2011, and must be marked with Package Identification Number USA/9196/B(U)F-96 after February 28, 2011. Any package transporting greater than a Type A quantity of UF_6 must be marked with Package Identification Number USA/9196/B(U)F-96.

13. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

14. Revision No. 25 of this certificate may be used until April 30, 2012.

15. Expiration date: February 28, 2016.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9196	26	71-9196	USA/9196/B(U)F-96	4	OF	4

<u>REFERENCES</u>

Duratek Inc., application dated: June 9, 2005.

Duratek Inc., supplements dated: June 30 and September 9, 2005.

Energy*Solutions* supplements dated: October 22, 2007, September 25, October 29, November 6, and December 16, 2008, and February 24, March 9 and 27, 2009, March 29, 2011.

Columbiana Hi Tech supplements dated: March 29, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 14, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9204	20	71-9204	USA/9204/B(U)F-96	1	OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Energy*Solutions* Services, Inc.
Suite 100, Center Point II
100 Center Point Circle
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Energy*Solutions* application dated January 24, 2011, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 10-160B

(2) Description

A cylindrical carbon steel and lead shielded shipping package, designed to transport radioactive waste material. The package is transported in the upright position and is equipped with steel encased, rigid polyurethane foam impact limiters on the top and bottom. The package has approximate dimensions, shielding, and weight as follows:

Package height	88 inches
Package outer diameter	78-1/2 inches
Package cavity height	77 inches
Package cavity diameter	68 inches
Overall package height, with impact limiters	130 inches
Overall package diameter, with impact limiters	102 inches
Lead shielding thickness	1-7/8 inches
Gross weight (packaging and contents)	72,000 lbs
Maximum total weight of contents, shoring, secondary containers, and optional shield insert	14,250 lbs

5.(a)(2) Description (Continued)

The packaging body consists of a 1-1/8-inch thick carbon steel (ASME SA516 or SA537) inner shell, a 1-7/8-inch thick lead gamma shield, and a 2-inch thick carbon steel outer shell (ASME SA516). The inner and outer shells are welded to a 5-1/2-inch thick carbon steel bottom plate. The packaging cavity has an optional 11-gage stainless steel liner. A 12-gage stainless steel thermal shield surrounds the cask outer shell in the region between the impact limiters. The impact limiters are secured to each other around the cask by eight ratchet binders.

The packaging lid is a 5-1/2-inch thick carbon steel plate, and has a 31-inch diameter opening equipped with a secondary lid. The primary lid is sealed with a double elastomer O-ring and 24 equally spaced 1-3/4-inch diameter bolts. The secondary lid is 46 inches in diameter, is centered within the primary lid, and is sealed to the primary lid by a double elastomer O-ring and 12 equally spaced 1-3/4-inch diameter bolts. The space between the double O-ring seals is provided with a test port for leak testing the primary and secondary lid seals.

The secondary lid is protected by a thermal shield which consists of two polished stainless steel plates separated by a thin air gap. The thermal shield is attached to the secondary lid lifting lugs with hitch-pins. The optional drain and vent ports are sealed with a plug and an O-ring seal.

The package is equipped with four tie-down lugs welded to the cask outer shell. Two lifting lugs and two redundant lifting lugs are removed during transport. The lid is equipped with three lifting lugs which are covered by the top impact limiter and rain cover during transport.

An optional carbon steel shield insert may be used within the cask cavity for contents as specified in Condition No. 5(b)(1)(i) through (v). For contents specified in Condition No. 5(b)(1)(vi), a source insert shall be used. The source insert design weight is 8,000 lbs; it has side walls consisting of 6.0-inch thick lead, sandwiched between an inner 8 inches nominal SCH 40 steel pipe and an outer 24.0-inch SCH 60 steel pipe. The bottom of the source insert also consists of lead supported by a 0.75-inch thick steel base plate. The lid includes a steel encased lead plug, steel bolting plate and flat silicone rubber gasket.

(3) Drawings

The packaging is constructed and assembled in accordance with Energy*Solutions* Drawing No. C-110-D-29003-010, sheets 1 through 5, Rev. 16.

The Secondary Lid Thermal Shield is constructed in accordance with Energy*Solutions* Drawing No. DWG-CSK-12CV01-EG-0002-01, Rev. 3.

An optional shield insert is constructed in accordance with Chem-Nuclear Systems Drawing No. C-119-B-0018, Rev. 2.

The Source Insert Assembly is constructed in accordance with Energy*Solutions* Drawing No. C-038-145083-004, Rev.0.

The Source Insert Steel Cribbing is constructed in accordance with Energy*Solutions* Drawing No. C-038-145083-005, Rev. 0.

5.(b) Contents

(1) Type and form of material

(i) Byproduct, source, and special nuclear material, non-fissile or fissile-excepted, as special form, or non-special form in the form of process solids or resins, either dewatered, solid, or solidified waste in secondary containers; or

(ii) Dewatered, solid or solidified transuranic-containing wastes (TRU), fissile or non-fissile or fissile-excepted, in secondary containers; or

(iii) Plutonium-239 (Pu-239) as Pu-Be neutron sources meeting the requirements of special form sources; or

(iv) Neutron activated metals or metal oxides in solid form in secondary containers; or

(v) Miscellaneous radioactive solid waste materials, including special form materials and powdered solids, in secondary containers.

(vi) Byproduct material as Co-60 loaded into the source insert.

(2) Maximum quantity of material per package

(i) The maximum quantity of radioactive materials must be the lesser of the quantity determined by the methodology described in Attachment 1 to Chapter No. 7 of the application or 3000 A_2, except for contents specified in Condition No. 5(b)(1)(vi) for which the limit is 10,000 Ci of Co-60.

(ii) Fissile contents must be limited to the fissile gram equivalent of 325 grams of Pu-239, as determined using the conversion factors in Table 9.1.3, in Chapter No. 4, Appendix 4.10.2, of the application. Plutonium content exceeding 0.74 TBq (20 Ci) must be in solid form.

(iii) TRU exceeding the fissile limits of 10 CFR 71.15 must not be machine-compacted and must have no more than 1% by weight of special reflectors and no more than 25% by volume of hydrogenous material.

(iv) Neutron sources as described in 5(b)(1)(iii) are limited to a maximum emission rate of 1.1E+8 n/sec.

(v) Maximum decay heat: 200 watts.

(vi) Maximum weight of contents: 14,500 pounds including shoring, secondary containers, and either optional shield insert or mandatory source insert. The contents of the source insert have a maximum weight of 500 pounds.

(vii) Explosives, corrosives, non-radioactive pyrophorics, and compressed gases are prohibited. Pyrophoric radionuclides may be present only in residual amounts less than 1 weight percent.

(viii) The total amount of potentially volatile organic compounds present in the headspace of a secondary container is restricted to 500 parts per million.

(ix) Powdered solid radioactive materials shall not include radioactive forms of combustible metal hydrides or combustible elemental metals, i.e., magnesium, titanium, sodium, potassium, lithium, zirconium, hafnium, calcium, zinc, plutonium, uranium, and thorium, or combustible non-metals, i.e., phosphorus.

(x) Powdered solids contents with neutron emitters are not permitted.

5.(c) Criticality Safety Index 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter No. 7 of the application.

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter No. 8 of the application.

7. Transport by air of fissile material is not authorized.

8. Flammable gas (hydrogen) concentration is limited to less than 5% in volume. For contents other than TRU waste, inerting is not allowed to limit the concentration of flammable gases. For TRU waste, compliance with the 5% hydrogen concentration limit is determined by the methods discussed in Appendix 4.10.2 of the application. For contents with a radioactivity concentration not exceeding that for Low Specific Activity material, the hydrogen concentration can be assumed to be less than 5% provided the package is shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers.

9. Payload containers authorized for shipment of TRU waste are the 30-gallon and the 55-gallon drums. TRU waste characteristics are determined and limited in accordance with Appendix 4.10.2 of the application.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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10. The non-homogeneity of the package contents may lead to elevated levels of radiation on the package surfaces. Radiation surveys must be performed to obtain measurements from all surfaces of the package, and from the outer surfaces of the vehicle enclosure, unless process knowledge or survey history indicates that elevated radiation levels are not likely to be encountered.

11. Appropriate devices or measures must secure contents in the secondary container, if necessary.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. Expiration date: October 31, 2015.

REFERENCES

Energy*Solutions* application dated January 24, 2011.

Supplements dated April 6, September 9, and October 28, 2011; July 20, July 26 and August 10, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 23 , 2012.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9212	6	71-9212	USA/9212/B(M)F-96	1	OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Washington TRU Solutions LLC consolidated application dated February 12, 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No: RH-TRU 72-B

(2) Description

A stainless steel, lead-shielded cask designed to provide double containment for shipment of transuranic waste materials. The packaging consists of a cylindrical stainless steel and lead cask body, a separate inner stainless steel vessel, and foam-filled impact limiters at each end of the cask body.

The cask body (outer cask) consists of a 1 1/2-inch thick, 41 5/8-inch outer diameter stainless steel outer shell, and a 1-inch thick, 32 3/8-inch inside diameter stainless steel inner shell, with 1 7/8 inches of lead shielding between the two shells. The cask bottom is 5-inch thick stainless steel plate. The cask is closed by a 6-inch thick stainless steel lid, and 18, 1 1/4-inch diameter bolts. The main closure lid has a double bore-type O-ring seal. The containment seal is the inner butyl O-ring seal, which is leak testable. The cask lid has a single vent/sampling port that is sealed with leak testable butyl O-ring seals.

The separate inner vessel consists of a 3/8-inch thick, 32-inch outside diameter stainless steel shell, and a 1 1/2-inch thick stainless steel bottom plate. The inner vessel is closed by a 6 1/2-inch thick stainless steel lid, and eight, 7/8-inch diameter bolts. The inner vessel closure lid has three bore-type O-ring seals. The containment seal is the middle butyl O-ring seal, which is leak testable. The inner vessel lid has a helium backfill port and a combination vent/sampling port that are sealed with leak-testable butyl O-ring seals.

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5.(a) (2) Description (Continued)

A polyurethane foam-filled stainless steel impact limiter is attached to each end of the cask body using six, 1 1/4-inch diameter bolts. The radioactive contents are packaged within a stainless or carbon steel waste canister that is placed in the inner vessel.

The approximate dimensions and weights of the package are as follows:

Overall package length	187 3/4 inches
Impact limiter diameter	76 inches
Cask length	141 3/4 inches
Cask outer diameter (OD)	41 5/8 inches
Inner vessel length	130 inches
Inner vessel OD	32 inches
Cask lead shield thickness	1 7/8 inches
Maximum package weight (including contents)	45,000 pounds
Maximum weight of contents (including waste canister)	8,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-500-SNP, sheets 1-8, Rev. 5.

The fixed lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-501-SNP, Rev. 4. The removable lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-502-SNP, Rev. 2. The neutron shielded waste canister is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-503-SNP, Rev. 0.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and waste, within the stainless or carbon steel waste canister described in Item 5(a)(3). Explosives, corrosives (pH less than 2 or greater than 12.5), and compressed gases are prohibited. Within a waste canister radioactive and non-radioactive pyrophorics must not exceed 1 weight percent. Flammable volatile organics are limited along with hydrogen to ensure the absence of flammable gas mixtures in RH-TRU waste payloads as described in RH-TRAMPAC (Rev. 1).

(2) Maximum quantity of material per package.

Not to exceed 8,000 pounds, including the weight of the waste canister.

5.(b) (2) Contents (Continued)

Fissile material not to exceed limits described in Section 3.1, "Nuclear Criticality" of RH-TRAMPAC (Rev. 1). Pu-239 equivalent is determined in accordance with RH-TRAMPAC (Rev. 1). Low enriched uranium is authorized for waste containers containing material that is primarily uranium (in terms of heavy metal component) and the waste matrix material must be distributed within the canister in such a manner that the maximum enrichment does not exceed 0.96% uranium (U-235) fissile equivalent mass in any location of the waste material.

Maximum decay heat per package not to exceed 50 watts per canister for all payloads in accordance with RH-TRAMPAC (Rev. 1).

(c) Criticality Safety Index: 0.0

6. Waste content codes and classification, physical form, chemical properties, chemical compatibility, gas generation, fissile content, decay heat, isotopic inventory, weight, and radiation dose rate must be determined and limited in accordance with RH-TRAMPAC (Rev. 1).

7. Each waste canister must not exceed the decay heat limits determined as specified in RH-TRAMPAC (Rev. 1), or must be tested for gas generation in accordance with RH-TRAMPAC (Rev. 1), Section 5.0, "Gas Generation Requirements."

8. A RH-TRU waste canister may be comprised of inner containers with different content codes provided that the hydrogen gas generation rate limit or decay heat limit for all of the inner containers within the payload is assumed to be the same as the content code with the lowest hydrogen gas generation rate limit or decay heat limit.

9. The waste canister and any sealed secondary containers greater than 4 liters in size overpacked in the waste canister must be vented in accordance with the minimum specifications in Section 2.4, Filter Vents, of RH-TRAMPAC (Rev. 1).

10. Shipments must not exceed the 60 or 10 day maximum shipping period requirement specified in RH-TRU Payload Appendices (Rev. 1), Sections 2.3 and 2.4, respectively.

11. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented.

(b) Each packaging must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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(c) Each containment O-ring seal material formulation and each batch of containment O-ring seal material must be qualified and tested in accordance with the procedures described in Section 3.6.4 of RH-TRU 72-B (Rev. 5).

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13 Packages must be marked with Package Identification Number USA/9212/B(M)F-96.

14. Revision No. 5 of this certificate may be used until June 30, 2012.

15. This package may not be used for transport by aircraft.

16. This package is for exclusive use shipments.

17. Expiration date: February 28, 2015.

REFERENCES

Washington TRU Solutions LLC consolidated application dated February 12, 2010.

Washington TRU Solutions LLC supplements dated April 19, 2010; August 30, 2010; February 16, 2011; April 14, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 17, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9215	13	71-9215	USA/9215/B(U)	1 OF	4

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation other applicable regulatory agencies, including the government of any country through or into which the package will be transporte

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Neutron Products, Inc.
22301 Mt. Ephraim Road
P.O. Box 68
Dickerson, MD 20842

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Neutron Products, Inc., application dated September 14, 1992, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

( ) Packaging

(1) Model No.: NPI-20WC-6 MkII

(2) Description

A steel encased, lead shielded cask contained within a wooden overpack with a steel outer shell. The cask is 24 inches in diameter with a 3/8-inch thick steel spherical shell and a cavity formed by an 8-1/4-inch ID by 3/16-inch thick steel tube. Positive closure of the shielded cask is accomplished by bolted end covers at each end of the cavity. The overpack is approximately 49 inches in diameter and 59 inches high, including the lid lifting eye and the base support structure. The maximum package gross weight is 6,000 pounds.

(3) Drawings

The Model No. NPI-20WC-6 MkII packaging is constructed in accordance with Neutron Products, Inc., Drawing Nos. 240116, Rev. G; and 240122, Sheet 1 of 2, Rev. H, Sheet 2 of 2, Rev. H, except as noted in Condition No. 9 below.

(b) Contents

(1) Type and form of material

(i) Cobalt-60 as sealed sources which meet the requirements of special form radioactive material.

5.(b) Contents (Continued)

(ii) Cesium-137 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

(i) For contents described in 5(b)(1)(i) and 5(b)(1)(ii):

For sources contained within drum assembly shown as Item 5 on Neutron Products, Inc., Drawing No. 240122, Sheet 1 of 2, Rev. H:

For contents described in 5(b)(1)(i):

Maximum activity not to exceed 15,000 curies, maximum decay heat not to exceed 240 watts.

For contents described in 5(b)(1)(ii):

Maximum activity not to exceed 20,600 curies, maximum decay heat not to exceed 97 watts.

(ii) For contents described in 5(b)(1)(i) and 5(b)(1)(ii):

For sources contained within drum assembly shown as Item 4 on Neutron Products, Inc., Drawing No. 240122, Sheet 2 of 2, Rev. H:

For contents described in 5(b)(1)(i):

Maximum activity not to exceed 9,500 curies, maximum decay heat not to exceed 150 watts.

For contents described in 5(b)(1)(ii):

Maximum activity not to exceed 20,600 curies, maximum decay heat not to exceed 97 watts.

(iii) For contents described in 5(b)(1)(i) and 5(b)(1)(ii):

For sources contained within drum assembly shown as Item 2 on Neutron Products, Inc., Drawing No. 240122, Sheet 2 of 2, Rev. H:

For contents described in 5(b)(1)(i):

Maximum activity not to exceed 6,300 curies, maximum decay heat not to exceed 100 watts.

5.(b) Contents (Continued)

For contents described in 5(b)(1)(ii):

Maximum activity not to exceed 20,600 curies, maximum decay heat not to exceed 97 watts.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be maintained in accordance with Maintenance and Storage Procedure for USA/9215/B(U) Package, R-2019-G, Revision 2, provided in the supplement dated March 29, 2013.

(b) The package shall be prepared for shipment and operated in accordance with Unloading and Loading Procedure for USA/9215/B(U) Package, R-2014-G, Revision 2, provided in the supplement dated March 29, 2013.

7. The contents must be secured in the drum assembly so as to restrict movement in any direction to less than 0.25 inch, by lead, steel, or tungsten full diameter plugs and spacers.

8. The gross weight of the package must not exceed 6,000 pounds, and the inner shield cask shall be snug-fitting within the wooden overpack.

9. The two permanent package identification labels and the single temporary package identification holder are attached with 3/16 inch aluminum pop rivets. The two manufacturer's stamped name and date labels are attached with 1/8 inch aluminum pop rivets. The temporary identification labels are held in their holder with a single 1/4 - 20 stainless steel screw. The eight one-quarter inch holes remaining from previous permanent package identification labels and the twelve half inch vent holes are covered by waterproof tape.

10. Contents described in 5(b)(1)(i) and 5(b)(1)(ii) may not be shipped together in the same package.

11. Fabrication of new packagings is not authorized.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. Revision No. 12 of this certificate may be used until May 31, 2013.

14. Expiration date: May 31, 2018.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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REFERENCES

Neutron Products, Incorporated, application dated September 14, 1992.

Supplements dated: October 29, 1992; November 17, 1993; September 8, 1997; September 5, 2002; May 1 and October 7, 2003, and February 16, and March 15, 2007; March 12, 2008; April 8, 2010; electronic correspondence dated April 15, 2010; February 9, and July 5, 2012; and March 29, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 16, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9217	16	71-9217	USA/9217/AF	1 OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Transnuclear, Inc.
7135 Minstrel Way, Ste. 300
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Siemens Power Corporation application dated January 26, 2000, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: ANF-250

(2) Description

A uranium oxide powder/pellet shipping container. The packaging consists of a 16-gauge steel inner vessel, approximately 11-1/2 inches ID by 57 inches long, with a bolted and gasketed top flange closure and steel welded bottom plate. The inner vessel is centered and supported in a 22-1/2-inch ID by 68-3/8-inch long, 16-gauge steel drum by twelve 1/4-inch diameter spring steel rods welded to the inner vessel at the top and the bottom of the vessel. A 3/8-inch thick steel flange and a 16-gauge inner band position and support the top of the inner vessel within the outer container. The annulus between the inner vessel and outer container is filled with vermiculite.

The inner vessel is closed by six 1/2-inch square shank studs with hex head nuts at each end. The outer container is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt and lock nut. A "half circle" ("U") type closure ring is used. A product container insert is positioned within the inner vessel.

The maximum gross weight of the packaging and contents is 616 pounds.

5.(a) (3) Drawings

(i) The ANF-250 shipping container is constructed in accordance with Siemens Power Corporation Drawing No. EMF-306,175, Rev. 16.

(ii) The pellet shipping suit case is constructed in accordance with Siemens Power Corporation Drawing No. EMF-304,306, Rev. 8.

(iii) The powder and pellet product container inserts are constructed in accordance with Siemens Power Corporation Drawing No. EMF-306,176, Rev. 6, Sheets 1 and 2.

(b) Contents

(1) Type and form of material

(i) Dry uranium oxide powder enriched to a maximum 5.0 w/o in the U-235 isotope with or without burnable absorbers.

(ii) Dry uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope with or without burnable absorbers.

(iii) Dry uranium oxide pellet scrap enriched to a maximum 5.0 w/o in the U-235 isotope with or without burnable absorbers.

(iv) Uranium oxide pellets enriched to a maximum of 1 w/o in the U-235 isotope with or without burnable absorbers.

(v) Uranium oxide pellet scrap enriched to a maximum of 1 w/o in the U-235 isotope with or without burnable absorbers.

(vi) Uranium oxide powder enriched to a maximum of 1 w/o in the U-235 isotope with or without burnable absorbers.

(2) Maximum quantity of material per package

Not to exceed 310 pounds and:

(i) For the contents described in 5(b)(1)(i):

5.(b)(2)(i) (continued)

The contents not to exceed the following:

Maximum Enrichment (wt% U-235)	Maximum Uranium Mass (kg U)	Maximum U-235 Mass (kg U-235)
3.4	62.4	2.12
3.8	41.0	1.56
4.6	31.2	1.44
5.0	27.7	1.38

Not to exceed a maximum mass of 1149 g H, considering all sources of hydrogenous material within the inner vessel. The contents must be contained in product container described in 5(a)(3)(iii).

(ii) For the contents described in 5(b)(1)(ii):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 6 kg. Not to exceed a maximum mass of 1149 g H, including a maximum mass of 600 g polyethylene, considering all sources of hydrogenous material within the inner vessel. The contents must be contained in product container described in 5(a)(3)(ii).

(iii) For the contents described in 5(b)(1)(iii):

The total contents not to exceed 61.7 kg U, with the U-235 content not to exceed 3.08 kg. Not to exceed a maximum mass of 1149 g H, including a maximum mass of 600 g polyethylene, considering all sources of hydrogenous material within the inner vessel. The contents must be contained in product container described in 5(a)(3)(ii).

(iv) For the contents described in 5(b)(1)(iv):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 1.2 kg. The contents must be contained in product container described in 5(a)(3)(ii).

(v) For the contents described in 5(b)(1)(v):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 1.2 kg. The contents must be contained in product container described in 5(a)(3)(ii).

(vi) For the contents described in 5(b)(1)(vi):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 1.2 kg. The contents must be contained in product container described in 5(a)(3)(iii).

5.(c) Criticality Safety Index

Minimum criticality safety index to be shown on label for nuclear criticality control:

For contents described in 5(b)(1)(i) and limited in 5(b)(2)(i):	1.8
For contents described in 5(b)(1)(ii) and 5(b)(1)(iii), and limited in 5(b)(2)(ii) and 5(b)(2)(iii):	0.9
For contents described in 5(b)(1)(iv), 5(b)(1)(v) and 5(b)(1)(vi), and limited in 5(b)(2)(iv), 5(b)(2)(v) and 5(b)(2)(vi):	0.4

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   a. The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

   b. The packaging must meet the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Revision Nos. 13 and 14 may be used until June 30, 2010.

9. Revision No. 15 may be used until July 1, 2011.

10. Transport by air of fissile material is not authorized.

11. Expiration date: June 30, 2015.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9217	16	71-9217	USA/9217/AF	5 OF	5

REFERENCES

Siemens Power Corporation application dated January 26, 2000.

Supplements dated: January 31, June 6, June 15 and September 29, 2000; February 6 and August 21, 2001; December 16, 2004; November 25, 2009; December 21, 2009; December 23, 2009; and April 8, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

.te: May 27, 2010

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9218	21	71-9218	USA/9218/B(U)F-96	1	OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Waste Partnership, LLC application dated March 27, 2013.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TRUPACT-II

(2) Description

A stainless steel and polyurethane foam insulated shipping container designed to provide single containment for shipment of contact-handled transuranic waste. The packaging consists of an unvented, 1/4-inch thick stainless steel inner containment vessel (ICV), positioned within an outer confinement assembly (OCA) consisting of an unvented 1/4-inch thick stainless steel outer confinement vessel (OCV), a 10-inch thick layer of polyurethane foam and a 1/4 to 3/8-inch thick outer stainless steel shell. The package is a right circular cylinder with outside dimensions of approximately 94 inches diameter and 122 inches height. The package weighs not more than 19,250 pounds when loaded with the maximum allowable contents of 7,265 pounds.

The OCA has a domed lid which is secured to the OCA body with a locking ring. Although not part of the containment boundary, the OCV confinement seal is provided by an optional butyl rubber O-ring (bore seal). The OCV is equipped with a seal test port and a vent port.

The ICV is a right circular cylinder with domed ends. The outside dimensions of the ICV are approximately 73 inches diameter and 98 inches height. The ICV lid is secured to the ICV body with a locking ring. The ICV containment seal is provided by a butyl rubber O-ring (bore seal). The ICV is equipped with a seal test port and vent port. Aluminum spacers are placed in the top and bottom domed ends of the ICV during shipping. The cavity available for the contents is a cylinder of approximately 73 inches diameter and 75 inches height.

5.(a)(3) Drawings

The packaging is constructed in accordance with Washington TRU Solutions, LLC, Drawing No. 2077-500 SNP, sheets 1-11, Rev. Y. The contents are positioned within the packaging in accordance with the Contact-Handled Transuranic Waste Authorized Methods for Payload Control (CH-TRAMPAC), Rev. 4, Section 2.9, "Payload Container/Assembly Configuration Specifications." The standard pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-001, sheets 1-3, Rev. 7. The S100 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-002, sheets 1 and 2, Rev. 5. The S200 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-003, sheets 1 and 2, Rev. 4. The S300 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-004, Rev. 2. The compacted puck drum spacers needed for the purpose of maintaining subcriticality in 55-, 85-, and 100-gallon drums are constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-006, Rev. 1. The criticality control overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-009, sheets 1 and 2, Rev. 0.

(b) Contents

(1) Type and form of material

Dewatered, solid or solidified transuranic and tritium-contaminated materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, an 85-gallon drum, a 100-gallon drum, a standard waste box (SWB), a standard pipe overpack, an S100 pipe overpack, an S200 pipe overpack, an S300 pipe overpack, ten-drum overpack (TDOP) or criticality control overpack (CCO). The payload containers are described in CH-TRAMPAC, Rev. 4, Section 2.9, "Payload Container/Assembly Configuration Specifications." Materials must be restricted to prohibit explosives, corrosives, nonradioactive pyrophorics and compressed gases. Within a payload container, radioactive pyrophorics must not exceed 1 percent by weight, and residual liquids must not exceed 1 percent by volume. Flammable organics and methane are limited along with hydrogen to ensure the absence of flammable gas mixtures in TRU waste payloads as described in Chapter 5.0 of CH-TRAMPAC, Rev. 4. For payloads of content code LA 154 and SQ 154, the absence of flammable gas mixtures is ensured as described in Appendix 6.12 of the CH-TRU Payload Appendices, Rev. 3. For payload configurations with unvented heat-sealed bag layers, the absence of flammable gas mixtures is ensured as described in Appendix 6.13 of the CH-TRU Payload Appendices, Rev. 3. For Analytical Category payload containers containing puck drums, the absence of flammable gas mixtures is ensured as described in Appendix 6.14 of the CH-TRU Payload Appendices, Rev. 3.

(2) Maximum quantity of material per package

Contents not to exceed 7,265 pounds including shoring and secondary containers. The maximum gross weight for a payload container not to exceed the following:

(I) 1,000 pounds per 55-gallon drum,
(ii) 328 pounds per 6-inch standard pipe overpack,
(iii) 547 pounds per 12-inch standard pipe overpack,
(iv) 550 pounds per S100 pipe overpack,
(v) 547 pounds per S200 pipe overpack,
(vi) 547 pounds per S300 pipe overpack,
(vii) 1,000 pounds per 85-gallon drum,
(viii) 1,000 pounds per 100-gallon drum,
(ix) 4,000 pounds per SWB,
(x) 6,700 pounds per TDOP, or
(xi) 350 pounds per CCO.

Maximum number of payload containers per package and authorized packaging configurations are as follows:

(I) 14 55-gallon drums,
(ii) 14 standard pipe overpacks,
(iii) 14 S100 pipe overpacks,
(iv) 14 S200 pipe overpacks,
(v) 14 S300 pipe overpacks,
(vi) 8 85-gallon drums,
(vii) 6 100-gallon drums,
(viii) 2 SWBs,
(ix) 1 TDOP, or
(x) 14 CCOs

Fissile material not to exceed the limits specified in CH-TRAMPAC, Rev. 4, Section 3.1, "Nuclear Criticality." Fissile material in the CCCs shall not be machine compacted and shall not exceed 380 fissile gram equivalent of Pu-239 containing less than or equal to 1% by weight Be/BeO.

All payloads shall meet the activity limits specified in CH-TRAMPAC, Rev. 4, Section 3.3, "Activity Limits." The payload is limited to 10^5 A_2 quantities.

Maximum decay heat per package not to exceed 40 watts. Decay heat per payload container not to exceed the values given in CH-TRAMPAC, Rev. 4, Table 5.2-1, "List of Approved Alpha-numeric Shipping Categories, Maximum Allowable Hydrogen Gas Generation Rates, and Maximum Allowable Wattages," or calculated for approved shipping categories in accordance with the methodology specified in Section 5.2.3 of CH-TRAMPAC, Rev. 4. For content code LA 154 and SQ 154 payloads, decay heat per payload container

5.(b)(2) Maximum quantity of material per package (continued)

not to exceed the values determined as specified in Appendix 6.12 of CH-TRU Payload Appendices, Rev. 3.

5.(c) Criticality Safety Index: 0.0

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight, center of gravity, and radiation dose rate must be determined and limited in accordance with CH-TRAMPAC, Rev. 4.

7. Each payload container must be assigned to a shipping category in accordance with CH-TRAMPAC, Rev. 4, Section 5.1, "Payload Shipping Category." For a payload assembly made up of payload containers with the same shipping categories, each payload container and payload assembly must not exceed the allowable wattage in accordance with CH-TRAMPAC, Rev. 4, Section 5.2.3, "Hydrogen Gas Generation Rate and Decay Heat Limits for analytical category," or must be tested for gas generation in accordance with CH-TRAMPAC, Rev. 4, Section 5.2.5, "Unified Flammable Gas Test Procedure." For a payload made up of payload containers with different (nonequivalent) shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with CH-TRAMPAC, Rev. 4, Section 6.2.4, "Mixing of Shipping Categories," and Appendix 2.4 of the CH-TRU Payload Appendices, "Mixing of Shipping Categories and Determination of the Flammability Index." For Analytical Category payload containers containing puck drums, the absence of flammable gas mixtures is ensured as described in Appendix 6.14 of the CH-TRU Payload Appendices, Rev. 3. Each content code LA 154 and SQ 154 payload container must be assigned to a shipping category in accordance with Appendix 6.12 of CH-TRU Payload Appendices. Content code LA 154 and SQ 154 payload containers may only be assembled with other payload containers belonging to content code LA 154 and SQ 154, respectively, or dunnage in accordance with Appendix 6.12 of CH-TRU Payload Appendices. For a payload of content code LA 154 or SQ 154 containers with different shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with Appendix 6.12 of CH-TRU Payload Appendices.

8. Payload containers within a package shall be selected in accordance with CH-TRAMPAC, Rev. 4, Section 6.0, "Payload Assembly Requirements." Payload containers of content code LA 154 and SQ 154 shall be assembled in accordance with Appendix 6.12 of CH-TRU Payload Appendices, Rev. 3.

9. Each payload container must be vented in accordance with Section 2.5, "Filter Vents," of the CH-TRAMPAC, Rev. 4. Payload containers which were not equipped with filtered vents during storage must be aspirated in accordance with CH-TRAMPAC, Rev. 4, Section 5.3, "Venting and Aspiration."

10. For close-proximity and controlled shipments meeting the conditions specified in Appendices 3.5 and 3.6, respectively, of CH-TRU Payload Appendices, shipping periods of 20 days and 10 days may be applicable. The shipping period for any mode of transport is not to exceed 60 days. For content code LA 154 and SQ 154 shipments, the shipping period as defined in Appendix 6.12 of the CH-TRU Payload Appendices is not to exceed 5 and 10 days, respectively.

11. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented. For content code LA 154 and SQ 154 payloads, each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0 of the application, as modified by Appendix 6.12 of CH-TRU Payload Appendices.

(b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

(c) All free standing water must be removed from the inner containment vessel cavity and the outer confinement vessel cavity before shipment.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. Revision No. 20 of this certificate may be used until June 30, 2014.

14. Expiration date: August 31, 2014.

REFERENCES

Nuclear Waste Partnership, LLC, application dated March 27, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William C. Allen for

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 19, 2013

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9221	7	71-9221	USA/9221/B(M)F-96	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Safety Analysis for Radioactive Material Shipping Cask NRBK-41 dated November 2, 1995, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NRBK-41

(2) Description

Top loading cylindrical lead shielded 304L stainless steel clad casks for the shipment of irradiated test specimens. The cask has an outside diameter of 27.16 inches and is 40 inches high. The outer shell is 1/2-inch thick stainless steel. The cask cavity is 5 inches in diameter by 16 inches deep and is provided with a bottom drain. The cavity shell is 1/4-inch thick stainless steel and is shielded by 10 inches of lead. The cask is closed by a lead-filled flanged plug fitted with an elastometer O-ring gasket and bolted closure. The cask has a seal-welded, 1/4-inch thick, stainless steel outer thermal shield which provides a 1/16-inch air gap between the outer surface of the cask outer shell and the inside surface of the thermal shield. A one-inch thick stainless steel plate is welded to the bottom of cask. A second one-inch thick stainless steel plate with a 1/8-inch deep, 25.5-inch diameter recess is welded to the first plate to provide a thermal shield for the bottom surface of the cask. The cask is bolted to a 48-inch square, all welded, "I" beam skid. Gross weight of the package is approximately 9,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Battelle Memorial Institute Drawing No. 41-0001, Sheet 1, Rev. D, and Sheet 2, Rev. E, and Westinghouse Electric Corporation Drawing No. 1755E01, Rev. D.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9221	7	71-9221	USA/9221/B( )F	2	OF	3

5. (b) Contents

   (1) Type and form of material

   Byproduct and special nuclear material in solid form, contained within either the MIN-41 or the HIP-41 product containers. The MIN-41 container is constructed in accordance with Westinghouse Electric Corporation, Drawing No. 2D77456 Rev. F. The HIP-41 product container is constructed in accordance with Westinghouse Electric Corporation Drawing No. 5D06622, Rev. B.

   (2) Maximum quantity of material per package

   The fissile contents of the package must be limited to a maximum of 350 equivalent grams of U-235. The number of equivalent grams of U-235 is determined by the equation: 1.0 x grams U-235 + 1.4 x grams U-233 + 1.6 x grams plutonium. The maximum decay heat load per package must not exceed 240 Btu/hr.

   Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy or reactor fuel elements.

5. (c) Criticality Safety Index: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be operated in accordance with the Operating Procedures in Section 7.0 of the application, as supplemented.

   (b) The package must be maintained in accordance with the Maintenance Procedures in Section 8.2 of the application, as supplemented.

7. The NRBK-41 shipping container may be covered with a wrapping of polyvinyl chloride (PVC) during shipment provided the shipment is made in a closed vehicle. The applicable requirements of 10 CFR §71.87 must be satisfied prior to wrapping the shipping container.

8. Transport by air of fissile material is not authorized.

9. Expiration date: April 30, 2018.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9221	7	71-9221	USA/9221/B(M)F-96	3 OF 3

REFERENCES

Safety Analysis for Radioactive Material Shipping Cask No. NRBK-41 dated November 2, 1995.

Supplements: Naval Reactors letters S#96-11965 dated August 28, 1996, S#01-10827 dated March 16, 2001, S#06-01881 dated May 31, 2006, S#06-03403 dated September 7, 2006, S#07-04076 dated November 5, 2007, S#08-00664 dated February 22, 2008, S#08-04186 dated November 14, 2008, and S#12-03559 dated August 10, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele M. Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 28, 2012

UNITED STATES NUCLEAR REGULATORY COMMISSION

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International, Inc.
3930 East Jones Bridge Road
Norcross, GA 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated June 18, 2010.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: NAC-LWT

(2) Description

The LWT is a steel-encased, lead-shielded shipping cask. The cask is designed to transport various radioactive contents as listed in 5.(b)(1). The overall dimensions of the package, with impact limiters, are 232 inches long by 65 inches in diameter. The cask body is approximately 200 inches in length and 44 inches in diameter. The cask cavity is 178 inches long and 13.4 inches in diameter. The volume of the cavity is approximately 14.5 cubic feet.

The cask body consists of a 0.75-inch-thick stainless steel inner shell, a 5.75-inch-thick lead gamma shield, a 1.2-inch-thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded to a 4-inch-thick stainless steel bottom end forging. The cask bottom consists of a 3-inch-thick, 20.75-inch-diameter lead disk enclosed by a 3.5-inch-thick stainless steel plate and bottom end forging. The cask lid is 11.3-inch-thick stainless steel stepped design, secured to a 14.25-inch-thick ring forging with twelve 1-inch diameter bolts. The cask seal is a metallic O-ring. A second teflon O-ring and a test port are provided to leak test the seal. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and O-rings.

The neutron shield tank consists of a 0.24-inch-thick stainless steel shell with 0.50-inch-thick end plates. The neutron shield region is 164 inches long and 5 inches thick. The neutron shield tank contains an ethylene glycol/water solution that is 1% boron by weight.

The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 65.25 inches and a maximum thickness of 27.8 inches. The bottom impact limiter has an outside diameter of 60.25 inches and maximum thickness of 28.3 inches. Both impact limiters extend 12 inches along the side of the cask body.

5.(a)(2) Description (continued)

The maximum weight of the package is 52,000 pounds and the maximum weight of the contents and basket is 4,000 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with the following Nuclear Assurance Corporation Drawings:

LWT 315-40-01, Rev. 7	Cask Assembly
LWT 315-40-02, Rev. 24 (Sheets 1-2)	Body Assembly
LWT 315-40-03, Rev. 22 (Sheets 1-7)*	Transport Cask Body
LWT 315-40-04, Rev. 12	Cask Lid Assembly
LWT 315-40-05, Rev. 10	Upper Impact Limiter
LWT 315-40-06, Rev. 10	Lower Impact Limiter
LWT 315-40-08, Rev. 18 (Sheets 1-5)	Cask Parts Detail

* Packaging Unit Nos. 1, 2, 3, 4, and 5 are constructed in accordance with Drawing No. LWT 315-40-03, Rev. 6 (Sheets 1-6).

(ii) The fuel assembly baskets are constructed in accordance with the following Nuclear Assurance Corporation and NAC International Drawings:

LWT 315-40-09, Rev. 2	PWR Basket Spacer
LWT 315-40-10, Rev. 8 (Sheets 1-2)	PWR Basket
LWT 315-40-11, Rev. 2	BWR Basket Assembly
LWT 315-40-12, Rev. 3	Metal Fuel Basket Assembly
LWT 315-40-045, Rev. 6	42 MTR Element Base Module
LWT 315-40-046, Rev. 6	42 MTR Element Intermediate Module
LWT 315-40-047, Rev. 6	42 MTR Element Top Module
LWT 315-40-048, Rev. 3	42 MTR Element Cask Assembly
LWT 315-40-049, Rev. 6	28 MTR Element Base Module
LWT 315-40-050, Rev. 6	28 MTR Element Intermediate Module
LWT 315-40-051, Rev. 6	28 MTR Element Top Module
LWT 315-40-052, Rev. 3	28 MTR Element Cask Assembly
LWT 315-40-070, Rev. 6	7 Cell Basket TRIGA Base Module
LWT 315-40-071, Rev. 6	7 Cell Basket TRIGA Intermediate Module
LWT 315-40-072, Rev. 6	7 Cell Basket TRIGA Top Module
LWT 315-40-079, Rev. 6	Transport Cask Assembly, 120 TRIGA Fuel Elements or 480 Cluster Rods
LWT 315-40-080, Rev. 4	7 Cell Poison Basket TRIGA Base Module
LWT 315-40-081, Rev. 4	7 Cell Poison Basket TRIGA Intermediate Module
LWT 315-40-082, Rev. 4	7 Cell Poison Basket TRIGA Top Module
LWT 315-40-083, Rev. 0	Spacer, LWT Cask Assembly TRIGA Fuel
LWT 315-40-084, Rev. 4	LWT Transport Cask Assy, 140 TRIGA Elements

5.(a)(3)(ii) Drawings (continued)

LWT 315-40-085, Rev. 1	Axial Fuel and Cell Block Spacers, MTR, and TRIGA Fuel Baskets
LWT 315-40-090, Rev. 4	35 MTR Element Base Module
LWT 315-40-091, Rev. 4	35 MTR Element Intermediate Module
LWT 315-40-092, Rev. 4	35 MTR Element Top Module
LWT 315-40-094, Rev. 4	35 MTR Element Cask Assembly
LWT 315-40-096, Rev. 3	Fuel Cluster Rod Insert, TRIGA Fuel
LWT 315-40-098, Rev. 6 (Sheets 1-3)	PWR/BWR Rod Transport Canister Assembly
LWT 315-40-099, Rev. 3 (Sheets 1-3)	Can Weldment, PWR/BWR Transport Canister
LWT 315-40-100, Rev. 4 (Sheets 1-5)	Lids, PWR/BWR Rod Transport Canister
LWT 315-40-101, Rev. 0	4 x 4 Insert, PWR/BWR Transport Canister
LWT 315-40-102, Rev. 2	5 x 5 Insert, PWR/BWR Transport Canister
LWT 315-40-103, Rev. 0	Pin Spacer, PWR/BWR Transport Canister
LWT 315-40-104, Rev. 6 (Sheets 1-3)	LWT Cask Assembly, PWR/BWR Rod Transport Canister
LWT 315-40-105, Rev. 3 (Sheets 1-2)	PWR Insert, PWR/BWR Transport Canister
LWT 315-40-106, Rev. 2 (Sheets 1-3)	MTR Plate Canister, LWT Cask
LWT 315-40-108, Rev. 1 (Sheets 1-3)	7 Cell Basket, Top Module, DIDO Fuel
LWT 315-40-109, Rev. 1 (Sheets 1-3)	7 Cell Basket, Intermediate Module, DIDO Fuel
LWT 315-40-110, Rev.1 (Sheets 1-3)	7 Cell Basket, Base Module, DIDO Fuel
LWT 315-40-111, Rev. 2	LWT Transport Cask Assy DIDO Fuel
LWT 315-40-113, Rev. 0	Spacer, Top Module DIDO Fuel
LWT 315-40-120, Rev. 2 (Sheets 1-3)	Top Module, General Atomics IFM, LWT Cask
LWT 315-40-123, Rev. 1 (Sheets 1-2)	Spacer, General Atomics IFM, LWT Cask
LWT 315-40-124, Rev. 1	Transport Cask Assembly, General Atomics IFM, LWT Cask
LWT 315-40-125, Rev. 3 (Sheets 1-3)	Transport Cask Assembly, Framatome/EPRI, LWT Cask
LWT 315-40-126, Rev. 2 (Sheets 1-2)	Weldment, Framatome/EPRI, LWT Cask
LWT 315-40-127, Rev. 2 (Sheets 1-2)	Spacer Assembly, TPBAR Shipment
LWT 315-40-129, Rev. 2	Canister Body Assembly, Failed Fuel Can, PULSTAR
LWT 315-40-130, Rev. 2	Assembly, Failed Fuel Can, PULSTAR
LWT 315-40-133, Rev. 2 (Sheets 1-2)	Transport Cask Assembly, PULSTAR Shipment, LWT Cask
LWT 315-40-134, Rev. 2	Body Weldment, Screened Fuel Can, PULSTAR Fuel
LWT 315-40-135, Rev. 1	Assembly, Screened Fuel Can, PULSTAR Fuel
LWT 315-40-139, Rev. 1	Transport Cask Assembly, ANSTO Fuel
LWT 315-40-140, Rev. 1 (Sheets 1-2)	Weldment, 7 Cell Basket, Top Module, ANSTO Fuel

5.(a)(3)(ii) Drawings (continued)

LWT 315-40-141, Rev. 1 (Sheets 1-2)	Weldment, 7 Cell Basket, Intermediate Module, ANSTO Fuel
LWT 315-40-142, Rev. 1 (Sheets 1-2)	Weldment, 7 Cell Basket, Base Module, ANSTO Fuel
LWT 315-40-145, Rev. 0 (Sheets 1-2)	Irradiated Hardware, Lid Spacer, LWT Cask
LWT 315-40-148, Rev. 0	LWT Transport Cask Assembly, ANSTO-DIDO Combination Basket
LWT 315-40-156, Rev. 3 (Sheets 1-4)	Canister Assembly SLOWPOKE Fuel
LWT 315-40-158, Rev. 0	Legal Weight Truck Transport Cask Assy, SLOWPOKE Fuel
LWT 315-40-170, Rev. 1	LWT Transport Cask Assy,. AECL NRU/NRX Components
LWT 315-40-172, Rev. 0 (Sheets 1 - 2)	Lid Assembly, NRU/NRX
LWT 315-40-173, Rev. 0 (Sheets 1 - 2)	Basket Weldment, NRU/NRX
LWT 315-40-174, Rev. 0	Basket Spacer, NRU/NRX
LWT 315-40-175, Rev. 1	Caddy Assembly, NRU/NRX

5.(b) Contents

(1) Type and form of material

All contents listed include both unirradiated and irradiated conditions.

(i) PWR fuel assemblies. The maximum fuel assembly weight is 1650 pounds, the maximum average burnup is 35,000 MWd/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zirconium alloy type cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-5, of the application.

Fuel Type	No. Fuel Rods	Max. Initial Uranium Enrichment (wt % U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
B&W 15x15	208	3.5	0.4750	144.0
B&W 17x17	264	3.5	0.4658	143.0
CE 14x14	176	3.7	0.4037	137.0
CE 16x16	236	3.7	0.4417	150.0
WE 14x14 Std	179	3.7	0.4144	145.2
WE 14x14 OFA	179	3.7	0.3612	144.0
WE 15x15	204	3.5	0.4646	144.0
WE 17x17 Std	264	3.5	0.4671	144.0

5.(b)(1)(i) PWR fuel assemblies. (continued)

WE 17x17 OFA	264	3.5	0.4282	144.0
Ex/ANF 14x14 WE	179	3.7	0.3741	144.0
Ex/ANF 14x14 CE	176	3.7	0.3814	134.0
Ex/ANF 15x15 WE	204	3.7	0.4410	144.0
Ex/ANF 17x17 WE	264	3.5	0.4123	144.0

(ii) BWR fuel assemblies. The maximum fuel assembly weight is 750 pounds, the maximum average burnup is 30,000 MWd/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zirconium alloy type cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-6, of the application.

Fuel Type	No. Fuel Rods	No. Water Rods	Max. Initial Uranium Enrichment (wt % U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
GE 7x7	49	0	4.0	0.1923	146
GE 8x8-1	63	1	4.0	0.1880	146
GE 8x8-2	62	2	4.0	0.1847	150 (1)
GE 8x8-4	60	4	4.0	0.1787	150 (1,2)
GE 9x9	74	2	4.0	0.1854	150 (1,3,4)
	79	2	4.0	0.1979	150 (1,4)
Ex/ANF 7x7	49	0	4.0	0.1960	144
Ex/ANF 8x8-1	63	1	4.0	0.1764	145.2
Ex/ANF 8x8-2	62	2	4.0	0.1793	150
Ex/ANF 9x9	79	2	4.0	0.1779	150
	74	2	4.0	0.1666	150 (3)

(1) Six-inch natural uranium blankets on top and bottom.
(2) One large water hole - 3.2 cm ID, 0.1 cm thickness.
(3) Two large water holes occupying seven fuel rod locations - 2.5 cm ID, 0.07 cm thickness.
(4) Shortened active fuel length in some rods.

5.(b)(1) Type and form of material (continued)

(iii) Deleted.

(iv) MTR fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_x-Al positioned within the MTR fuel basket specified in 5.(a)(3)(ii). Loose fuel plates must meet the requirements of the MTR fuel element content tables and must be loaded into an MTR plate canister prior to shipment. The fuel elements are composed of aluminum clad plates, with initial uranium enrichment up to 94.0 weight percent U-235. The maximum burnup and the minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(iv) and shall be determined using the operating procedures in Section 7.1.5 of the application.

NISTR MTR fuel elements specifications are listed in Item 5.(b)(1)(iv)(a), generic MTR fuel elements are listed in Item 5.(b)(1)(iv)(b), and expanded fuel specifications applicable to LEU MTR fuel (up to 25.0 wt %^{235}U) are listed in Items 5.(b)(1)(iv)(c) and 5.(b)(1)(iv)(d).

(a) NISTR MTR Fuel Content Description

Parameter	Plate	Plate (cut in half)
Enrichment, wt % ^{235}U	≤94	
Number of fuel plates	≤17	≤34
^{235}U content per plate	≤22	≤11
Plate thickness (cm)	≥0.115	
Clad Thickness (cm)	≥0.02	
Active fuel width (cm)	≤6.6	
Active fuel height (cm)	≥54 cm	27 to 30
Maximum ^{235}U content per element (g)	≤380	

5.(b)(1) Type and form of material (continued)

(iv) (b) Generic MTR Fuel Content Description

Parameter	Limiting Values[2]					
Enrichment, wt. % ^{235}U	≤94					
Number of fuel plates	≤23	≤19	≤23[1]	≤17	≤19	≤23
^{235}U content per plate	≤18	≤20	≤20[1]	≤21	≤21	≤16.5
Plate thickness (cm)	≥0.115	≥0.115	≥0.123[1]	≥0.115	≥.200	≥0.115
Clad Thickness (cm)	≥0.02					
Active fuel width (cm)	≤6.6	≤6.6	≤6.6	≤6.6	≤6.6	≤7.3
Active fuel height (cm)	≥56					
^{235}U content per element (g)	≤380[2]					

Notes:

1. HEU (>90 wt% ^{235}U enriched) MTR fuel having 23 plates with up to 20 g of ^{235}U per plate, with a minimum plate thickness of 0.123 cm, must have at least 2.0 cm of non-fuel material at the ends of each element. This fuel may also be loaded up to 460 g ^{235}U per element.

2. At enrichments ≤25 wt% ^{235}U, MTR fuel elements with extended fuel characteristics may be loaded with the specifications defined in 5.(b)(1)(iv)(c).

5.(b)(1) Type and form of material (continued)

(iv) (c) Expanded LEU MTR Fuel Content Description

Parameter	Base	≤7.0 cm Active Fuel Width			≤7.1 cm Active Fuel Width		≤7.15 cm Active Fuel Width		
Enrichment, wt. % ^{235}U	≤25	≤25			≤25		≤25		
Number of fuel plates	≤23	≤23			≤17	≤23	≤22	≤23	≤23
^{235}U content per plate	≤22	≤22	≤22	≤21.5	≤22		≤22	≤21.5	≤22
Plate thickness (cm)	≥0.115	≥0.119	≥0.115	≥0.115	≥0.115	≥0.200	≥0.119		
Clad Thickness (cm)	≥0.02								
Active fuel width (cm)	≤6.6	≤7.0			≤7.1		≤7.15		
Active fuel height (cm)	≥56	≥56	≥63	≥56	≥56		≥56	≥56	≥61
^{235}U content per element (g)	≤420	≤470			≤470		≤470		

5.(b)(1) Type and form of material (continued)

(iv) (d) Expanded LEU MTR Fuel Content Description for High Fissile Material Mass

Parameter	Limiting Value
Enrichment, wt.% ^{235}U	≤25
Number of fuel plates	≤23
^{235}U content per plate (g)	≤32
Plate thickness (cm)	≥0.115
Clad thickness (cm)	≥0.02
Active fuel width (cm)	≤7.3
Active fuel height (cm)	≥56
^{235}U content per element (g)	≤640

(v) Metallic fuel rods containing natural enrichment uranium pellets with aluminum cladding 0.080-inches thick. The fuel pellet diameter is 1.36 inches and the maximum fuel rod length is 120.5 inches. The maximum weight of uranium per rod is 54.5 kg with a maximum average burnup of 1,600 MWd/MTU and a minimum cooling time of one year.

(vi) TRIGA damaged and undamaged fuel elements. TRIGA fuel elements that have a cladding breach that allows the escape of gas or intrusion of water are considered damaged and will be loaded and transported in a sealed damaged fuel can.

5.(b)(1) Type and form of material (continued)

(vi) (a) TRIGA fuel elements acceptable for loading in the poisoned TRIGA basket and meeting the following specifications:

	TRIGA HEU (Notes 1, 2, 6, & 7)	TRIGA LEU (Notes 1, 2, 6, & 7)	TRIGA LEU (Notes 1, 2, 6, & 7)
Fuel Form	Clad U-ZrH rod	Clad U-ZrH rod	Clad U-ZrH rod
Maximum Element Weight, lbs	13.2	13.2	13.2
Maximum Element Length, in	47.74	47.74	47.74
Element Cladding	Stainless Steel	Stainless Steel	Aluminum
Clad Thickness, in	0.02	0.02	0.03
Active Fuel Length, in	15	15	14-15 (Note 4)
Element Diameter, in	1.478 max.	1.478 max.	1.47 max.
Fuel Diameter, in	1.435 max.	1.435 max.	1.41 max.
Maximum Initial U Content/Element, kilograms	0.196	0.845	0.205
Maximum Initial ^{235}U Mass, grams	137	169	41
Maximum Initial ^{235}U Enrichment, weight percent	70	20	20
Zirconium Mass, grams (Note 5)	2060	1886 – 2300	2300
Hydrogen to Zirconium Ratio, max. (Note 5)	1.6	1.7	1.0
Maximum Average Burnup, MWd/MTU	460,000 (80% ^{235}U)	151,100 (80% ^{235}U)	151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)	90 days (Note 3)	90 days (Note 3)

Notes:
1. Mixed TRIGA LEU and HEU contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Aluminum clad fuel with 14 inch active fuel is solid and has no central hole with a zirconium rod.
5. Zirconium mass and H/Zr ratio apply to the fuel material (U-Zr-H_x) and do not include the center zirconium rod.
6. Listed TRIGA fuel elements have a 0.225-inch diameter zirconium rod in the center.
7. Dimensions listed are as-fabricated (unirradiated) nominal values.

5.(b)(1) Type and form of material (continued)

(vi) (b) TRIGA fuel elements acceptable for loading in the nonpoisoned TRIGA basket and meeting the following specifications:

	TRIGA HEU (Notes 1, 2, & 6)	TRIGA LEU (Notes 1, 2, & 6)	TRIGA LEU (Notes 1, 2, & 6)
Fuel Form	Clad U-ZrH rod (Note 4)	Clad U-ZrH rod (Note 4)	Clad U-ZrH rod (Note 4)
Maximum Element Weight, lbs	13.2	13.2	13.2
Maximum Element Length, in	47.74	47.74	47.74
Element Cladding	Stainless Steel	Stainless Steel	Aluminum
Minimum Clad Thickness, in	0.01	0.01	0.01
Maximum Element Diameter, in	1.5 max.	1.5 max.	1.5 max.
Active Fuel Length, in	15	15	15
Maximum Initial U Content/Element, kilograms	0.196	0.845	0.205
Maximum Initial ^{235}U Mass, grams	137	169	41
Maximum Initial ^{235}U Enrichment, weight percent	70	20	20
Hydrogen to Zirconium Ratio, max. (Note 5)	2.0	2.0	2.0
Maximum Average Burnup, MWd/MTU	460,000 (80% ^{235}U)	151,100 (80% ^{235}U)	151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)	90 days (Note 3)	90 days (Note 3)

Notes:
1. Mixed TRIGA LEU and HEU contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Element may contain a zirconium rod in the center.
5. H/Zr ratio applies to the fuel material (U-Zr-H_x) and does not include the center zirconium rod.
6. Dimensions listed are as-fabricated (unirradiated) nominal values.

5.(b)(1) Type and form of material (continued)

(vi) (c) General Atomics TRIGA fuel elements acceptable for loading in the nonpoisoned TRIGA basket and meeting the following specifications:

	TRIGA HEU (Notes 1, 2, & 6)		TRIGA LEU (Notes 1, 2, & 6)		TRIGA LEU (Notes 1, 2, & 6)
Fuel Form	Clad U-ZrH rod (Note 4)		Clad U-ZrH rod (Note 4)		Clad U-ZrH rod (Note 4)
Maximum Element Weight, lbs	13.2		13.2		13.2
Maximum Element Length, in	47.74		47.74		47.74
Element Cladding	Stainless Steel		Stainless Steel		Aluminum
Minimum Clad Thickness, in	0.01		0.01		0.01
Maximum Element Diameter, in	1.5 max.		1.5 max.		1.5 max.
Active Fuel Length, in	15		15		15
Maximum Initial U Content/Element, kilograms	0.198	0.186	0.845	1.447	0.205
Maximum Initial ^{235}U Mass, grams	138	175[7,8]	169	275[7,8]	41
Maximum Initial ^{235}U Enrichment, weight percent	71	95[7,8]	25	25[7,8]	25
Hydrogen to Zirconium Ratio, max. (Note 5)	2.0		2.0		2.0
Maximum Average Burnup, MWd/MTU	460,000 (80% ^{235}U)	583,000 (80% ^{235}U)	151,100 (80% ^{235}U)		151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)		90 days (Note 3)		90 days (Note 3)

Notes:

1. Mixed TRIGA LEU and HEU fuel elements and LEU and HEU TRIGA fuel cluster rod contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Element may contain a zirconium rod in the center.
5. H/Zr ratio applies to the fuel material (U-Zr-H_x) and does not include the center zirconium rod.
6. Dimensions listed are as-fabricated (unirradiated) nominal values.
7. Limited to loading in top and bottom basket modules only.
8. Limited to a maximum of three elements per basket module cell.

5.(b)(1) Type and form of material (continued)

(vii) (a) TRIGA fuel cluster rods. TRIGA HEU fuel cluster rods have a maximum average burnup of 600,000 MWd/MTU (80% ^{235}U depletion) and a minimum cooling time of 90 days. TRIGA LEU fuel cluster rods have a maximum average burnup of 140,000 MWd/MTU (80% ^{235}U depletion) and a minimum cooling time of 90 days. TRIGA fuel cluster rods must meet the following specifications prior to irradiation:

	TRIGA Fuel Cluster Rods	
	HEU	LEU
Fuel Form	Clad U-ZrH rod	
Maximum Rod Weight, lbs	1.5	
Maximum Rod Length, in	31	
Rod Cladding	Incoloy 800	
Minimum Clad Thickness, in	0.015	
Maximum Active Fuel Length, in	22.5	
Maximum Fuel Pellet Diameter, in	0.53	
Maximum U Content/Rod, grams	48.6	289.5
Maximum ^{235}U Mass, grams	45.4	55.0
Maximum ^{235}U Enrichment, weight percent	93.3	20
Maximum Zirconium Mass, grams	421	357
Hydrogen to Zirconium Ratio, max.	1.7	

NOTE: TRIGA fuel cluster rods that have a cladding breach that allows the escape of gas or intrusion of water are considered damaged and will be loaded and transported in a sealed damaged fuel can.

5.(b)(1) Type and form of material (continued)

(vii) (b) General Atomics TRIGA fuel cluster rods. TRIGA HEU fuel cluster rods have a maximum average burnup of 600,000 MWd/MTU (80% ^{235}U depletion) and a minimum cooling time of 90 days. TRIGA LEU fuel cluster rods have a maximum average burnup of 140,000 MWd/MTU (80% ^{235}U depletion) and a minimum cooling time of 90 days. TRIGA fuel cluster rods must meet the following specifications prior to irradiation:

	TRIGA Fuel Cluster Rods	
	HEU	LEU
Fuel Form	Clad U-ZrH rod	
Maximum Rod Weight, lbs	1.5	
Maximum Rod Length, in	31	
Rod Cladding	Incoloy 800	
Minimum Clad Thickness, in	0.015	
Maximum Active Fuel Length, in	22.5	
Maximum Fuel Pellet Diameter, in	0.53	
Maximum U Content/Rod, grams	48.6	289.5
Maximum ^{235}U Mass, grams	46.5	55.0
Maximum ^{235}U Enrichment, weight percent	93.3	20
Maximum Zirconium Mass, grams	421	357
Hydrogen to Zirconium Ratio, max.	1.7	

NOTE: TRIGA fuel cluster rods that have a cladding breach that allows the escape of gas or intrusion of water are considered damaged and will be loaded and transported in a sealed damaged fuel can.

(viii) High burnup PWR rods, consisting of uranium dioxide pellets within zirconium alloy type cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.3765 inches. The maximum burnup is 80,000 MWd/MTU, and the minimum cool time is 150 days.

5.(b)(1) Type and form of Material (continued)

(ix) High burnup BWR rods, consisting of uranium dioxide pellets within zirconium alloy type cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.490 inch. The maximum burnup is 80,000 MWd/MTU and the minimum cool time is between 150 - 270 days, as specified in the table below:

BWR Fuel Type Array Size	Burnup, b (GWd/MTU)	Minimum Cool Time (days)
7 x 7	$b \leq 60$ $60 < b \leq 70$ $70 < b \leq 80$	210 240 270
8 x 8[1]	$b \leq 80$	150

Note 1: Includes rods from all larger BWR assembly arrays (e.g., 9 x 9, 10 x 10)

(x) Intact or degraded clad DIDO fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_x-Al plates fabricated into four concentric tubes of varying diameters. The fuel elements have an initial enrichment up to 94.0 weight percent U-235. Maximum degraded clad allowable per element is ≤ 5% surface area. Degraded clad DIDO fuel elements are to be loaded into an aluminum damaged fuel can (DFC) per Figure 1.2.3-18 of the application. The fuel elements shall have the specifications listed below:

Parameter	LEU[(1)]	MEU[(1)]	HEU[(1)]
Maximum ^{235}U content per Element	≤ 190 g	≤ 190 g	≤ 190 g
Maximum Uranium content per Element	≤ 1000 g	≤ 475.0 g	≤ 211.1g
Minimum Fuel Tube Thickness	0.130 cm	0.130 cm	0.130 cm
Minimum Clad Thickness	0.025 cm	0.025 cm	0.025 cm
Maximum Outer Diameter	9.535 cm	9.535 cm	9.535 cm
Minimum Inner Diameter	5.88 cm	5.88 cm	5.88 cm
Minimum Initial Enrichment	19 wt% ^{235}U	40 wt% ^{235}U	90 wt% ^{235}U

1 The maximum burnup and minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(xi)(a) and (b) and shall be determined using the operating procedures in Section 7.1.4 of the application.

5.(b)(1) Type and form of material (continued)

(xi) General Atomics (GA) Irradiated Fuel Material (IFM) consisting of two separate types of fuel materials: (a) High Temperature Gas Cooled Reactor (HTGR); and (b) Reduced-Enrichment Research and Test Reactor (RERTR) type TRIGA fuel entities.

(a) GA HTGR IFM comprised of four forms: fuel particles (kernels), fuel particles (coatings), fuel compacts (rods), and fuel pebbles. Fuel particles (kernels) are solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of UO_2, UCO_2, $(Th,U)C_2$, or $(Th,U)O_2$. Fuel particles (coatings) are solid , spheridized, isotropic, discrete multi-layered fuel particle coatings with chemical composition including pyrolitic-carbon (PyC) and silicon carbide (SiC). Fuel compacts (rods) are multi-coated ceramic fuel particles, bound in solid, cylindrical, injection molded, high-temperature heat-treated compacts which are composed of carbonized graphite shim, coke, and graphite powder. Fuel pebbles are multi-coated fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles composed of carbonized graphite shim, coke and graphite powder. Initial enrichment of the HTGR IFM varies from 10.0 to 93.15 wt% ^{235}U.

(b) GA RERTR IFM comprised of irradiated TRIGA fuel elements which contain three distinct mass loadings of uranium of 20, 30, and 45 wt% U. The average mass of the fuel portion of the elements is 551 g with a maximum initial enrichment of 19.7 wt% U-235.

GA IFM content description:

	GA HTGR IFM	GA RERTR IFM
Fuel material	UC_2, UCO, UO_2 $(Th,U)C_2$, $(Th,U)O_2$	U-ZrH metal alloy
Maximum fuel weight, lbs	23.52	23.73
Maximum overall length, in	n/a	29.92
Maximum active fuel length, in	n/a	22.05
Fuel rod cladding	n/a	Incoloy 800
Maximum Uranium, kg U	0.21	3.86
Maximum initial ^{235}U, wt%	93.15	19.7
Maximum Activity, Ci	483	2920

5.(b)(1) Type and form of material (continued)

(xii) Tritium-producing burnable absorber rods (TPBARs), as described in Section 1.2.3.6 of the application. Each TPBAR is approximately 153 inches in length and 0.381 inches in diameter and is stainless steel clad. The TPBARs contain lithium aluminate annular pellets, with an inner zircaloy liner and an outer nickel-plated zircaloy tube. Each TPBAR contains a maximum of 1.2 grams tritium. The minimum cool time is 30 days.

(xiii) Intact or damaged PULSTAR fuel elements, including fuel debris, pieces and nonfuel components of PULSTAR fuel assemblies as specified below.

Description	Value
Maximum Pellet Diameter (inch)	0.423
Minimum Element (Rod) Cladding Thickness (inch)	0.0185
Minimum Element (Rod) Diameter (inch)	0.470
Maximum Active Fuel Height (inch)	24.1
Nominal Element (Rod) Length (inch)	26.2
Nominal Assembly Length (inch)	38
Maximum Assembly or Loaded Can Weight (lb)	80
Maximum PULSTAR Can Content Weight (lb)	39.6
Maximum Enrichment (wt % ^{235}U)	6.5
Maximum ^{235}U Content per Element (g)	33
No. of Elements (Rods) per Assembly	25
No. of Elements (Rods) per Can[1]	≤25
Maximum Depletion (% ^{235}U)	45
Minimum Cooling Time (yrs)	1.5
Maximum Heat Load per Assembly (W)	30
Maximum Heat Load per Element (W)	1.2

[1] Damaged PULSTAR fuel elements, including fuel debris, pieces and nonfuel components of PULSTAR fuel assemblies must be loaded into a PULSTAR can. The contents of a PULSTAR can are restricted to the equivalent of the fuel material in 25 intact PULSTAR fuel elements and of the displaced volume of 25 intact PULSTAR fuel elements.

5.(b)(1) Type and form of material (continued)

(xiv) Intact or degraded clad ANSTO fuel consisting of spiral fuel assemblies and MOATA plate bundles. Maximum degraded clad allowable per element is ≤ 5% surface area. Degraded clad ANSTO fuel elements are to be loaded into an aluminum damaged fuel can (DFC) per Figure 1.2.3-18 of the application.

Spiral fuel assemblies consist of 10 curved uranium-aluminum alloy fuel plates between an inner and an outer aluminum shell, with the following fuel parameters:

Parameter	Limiting Values
Number of fuel plates per assembly	10
Maximum ^{235}U content per assembly (g)	160
Maximum enrichment (wt % ^{235}U)	95
Maximum assembly weight (lb)	18
Minimum plate thickness (cm)	0.124
Minimum active fuel height (cm)	59.075

MOATA plate bundles consist of uranium-aluminum alloy fuel plates with aluminum cladding, with the following specifications:

Parameter	Limiting Values
Maximum number of fuel plates per assembly	14
Maximum ^{235}U content per plate (g)	22.3
Maximum enrichment (wt % ^{235}U)	92
Maximum plate spacer thickness (cm)	0.18
Maximum active fuel width (cm)	7.32
Maximum bundle weight (lb)	18

(xv) Segmented TPBARs and associated segmentation debris resulting from post-irradiation examination, as described in Section 1.2.3.6 of the application. Each equivalent TPBAR contains a maximum of 1.2 grams of tritium. The minimum cool time is 90 days.

(xvi) Solid, irradiated and contaminated fuel assembly structural or reactor internal component hardware, which may include fissile material, provided the quantity of fissile material does not exceed a Type A quantity and qualifies as an exempt quantity under 10 CFR 71.15.

5.(b)(1) Type and form of material (continued)

(xvii) PWR MOX (mixed oxide) undamaged fuel rods consisting of uranium and plutonium and plutonium dioxide pellets within zirconium alloy type cladding. The plutonium enrichment is 7.0 weight percent maximum and 2.0 weight percent minimum, the maximum active fuel rod length is 153.5 inches, and the maximum pellet diameter is 0.3765 inch. The maximum burnup is 62,500 MWd/MTU and the minimum cool time is 90 days.

(xviii) Damaged or undamaged SLOWPOKE fuel rods, including fuel pieces and debris as specified below:

Parameter	Limiting Values
Maximum Cask Heat Load (W)	5
Maximum Canister Heat Load (W)	0.625
Payload Limit (lb/canister)	25
Maximum ^{235}U per rod (g)	2.800
Maximum U per rod (g)	3.111
Maximum Enrichment (wt% ^{235}U)	95
Minimum cool time (yr)	14
Maximum burnup (GWd/MTU) or wt% ^{235}U Depletion	30 4.5

(xix) Undamaged NRU or NRX fuel assemblies as specified below:

Parameter	NRU (HEU)	NRU (LEU)	NRX
Maximum Heat Load per Package (W)	640.0		
Maximum Heat Load per Tube (W)	35.6		
Maximum Weight of Contents per Tube (g)	20.0		
Maximum Mass ^{235}U per Rod (g)	43.24	43.68	79.05
Maximum Mass U per Rod (g)	48	230	87
Minimum Cool Time (yr)	19	3	18
Maximum Burnup (MWd/assembly or wt% ^{235}U Depletion)	364.0 87.4	363.0 83.6	375.0 85.1

5.(b)(2) Maximum quantity of material per package

Not to exceed 4,000 pounds, including contents and fuel assembly basket or other internal support structure.

(i) For the contents described in Item 5.(b)(1)(i): one PWR assembly positioned within the PWR fuel assembly basket. Maximum decay heat not to exceed 2.5 kilowatts per PWR assembly.

(ii) For the contents described in Item 5.(b)(1)(ii): two BWR assemblies positioned within the BWR fuel assembly basket. Maximum decay heat not to exceed 1.1 kilowatts per BWR assembly.

(iii) Deleted.

5.(b)(2) Maximum quantity of material per package (continued)

(iv) For MTR fuel elements as described in Item 5.(b)(1)(iv):

Up to 42 fuel elements positioned within the MTR fuel assembly basket (7 fuel elements per basket module). Each of the MTR basket cell openings may contain a loose plate canister. The contents of each loose plate canister are limited to the number of fuel plates, dimensions, and masses that are equivalent to an intact MTR fuel element, as specified in Item 5.(b)(1)(iv).

(a) The maximum decay heat is not to exceed 1.26 kilowatts per package, with each MTR fuel assembly basket module not to exceed 210 watts.

(b) HEU, MEU, and LEU MTR fuel elements with decay heat not exceeding 30 watts per element may be loaded in any basket position.

(c) Mixed HEU, MEU, and LEU MTR contents, with decay heat limits as specified above, are authorized.

(d) MTR fuel elements with degraded or mechanically damaged cladding are authorized, provided the total surface area of through-clad corrosion and/or mechanical damage does not exceed 5% of the total surface area of the damaged element.

(e) For HEU-MTR fuel elements only, the center fuel element in any basket module is not to exceed 120 watts. The two exterior fuel elements vertically in-line with the center assembly for transport are not to exceed 70 watts.

(f) MTR fuel elements containing more than 470 g ^{235}U (more than 22 g ^{235}U per plate) are limited to up to four elements loaded in basket positions 4, 5, 6, and 7 of a seven-element basket per Figure 7.1-1 of the application. Basket positions 1, 2, and 3 are to be blocked by spacer hardware.

(v) For the contents described in Item 5.(b)(1)(v): up to 15 intact metallic fuel rods positioned within the appropriate basket. Maximum decay heat not to exceed 0.036 kilowatts per rod. Total weight of all rods not to exceed 1,805 pounds.

(vi) For failed metallic fuel rods of the type described in Item 5.(b)(1)(v):

(a) Up to six canisters containing one defective metallic fuel rod per canister. The canisters are 2.75-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D2, Rev. 10, and are placed in a six-hole liner as shown on Nuclear Assurance Corporation Drawing No. 315-040-43, Rev. 1. The maximum decay heat load for a defective metallic fuel rod is limited to 5 watts; or

5.(b)(2) Maximum quantity of material per package (continued)

(b) Up to three canisters containing either up to three defective metallic fuel rods per canister or up to 10 failed fuel filters per canister. The canisters are 4.00-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D1, Rev. 10, and are placed in a three-hole basket as shown on Nuclear Assurance Corporation Drawing No. 315-40-12, Rev. 3. The weight of the filters is limited to 125 pounds per canister. For canisters containing fuel rods, the maximum decay heat load is 15 watts per canister; and for canisters containing filters, the maximum decay heat load is 5 watts per canister.

(vii)(a) For TRIGA fuel elements as described in Item 5.(b)(1)(vi)(a):

Up to 140 intact fuel elements in the TRIGA fuel package with poisoned baskets. Up to four fuel elements per basket cell and up to seven cells per basket may be loaded. Damaged TRIGA fuel elements or fuel element debris (up to a total of two equivalent elements) shall be transported in a sealed damaged fuel can (one damaged fuel can per cell). The sealed cans are to be in accordance with NAC International Drawing Nos. 315-40-086, 315-40-087, and 315-40-088.

Mixed intact and damaged fuel contents and fuel debris are authorized. Base and top fuel basket modules may contain intact fuel elements or sealed damaged fuel cans containing damaged fuel and fuel debris. A maximum of seven damaged fuel cans is authorized per top and base basket modules with a maximum of 14 per package. Intermediate fuel basket modules may contain only intact TRIGA fuel elements.

The maximum decay heat shall not exceed 7.5 watts per TRIGA fuel element (or equivalent for damaged fuel) and 1050 watts per package. The cask and baskets must be configured as shown in NAC International Drawing Nos. 315-40-084, 315-40-080, 315-40-081, and 315-40-082.

(vii)(b) For TRIGA fuel elements as described in Item 5.(b)(1)(vi)(b):

Up to 120 intact fuel elements in the TRIGA fuel package with non-poisoned basket. Up to four fuel elements per basket cell only loaded in the six periphery cells. TRIGA fuel elements or sealed cans may not be loaded in the center cell of the non-poisoned basket. Damaged TRIGA fuel elements or fuel debris (up to two equivalent elements) shall be transported in a sealed damaged fuel can (one damaged fuel can per cell). The sealed cans are to be in accordance with NAC International Drawing Nos. 315-40-086, 315-40-087, and 315-40-088.

Mixed intact and damaged fuel contents and fuel debris are authorized. Base and top fuel basket modules may contain intact fuel elements or sealed damaged fuel cans containing damaged fuel or fuel debris. A maximum of six damaged fuel cans is authorized only in the periphery cells per top and base basket modules with a maximum of 12 per package. Intermediate fuel basket modules may contain only intact TRIGA fuel elements.

5.(b)(2) Maximum quantity of material per package (continued)

Maximum decay heat not to exceed 7.5 watts per TRIGA fuel element (or equivalent for damaged fuel) and 900 watts per package. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket. The cask and baskets must be configured as shown in NAC International Drawing Nos. 315-40-070, 315-40-071, and 315-40-072, and 315-40-079.

(vii)(c) For General Atomics TRIGA fuel elements as described in Item 5.(b)(1)(vi)(c):

Up to 120 intact fuel elements in the TRIGA fuel package with non-poisoned basket. Up to four fuel elements per basket cell only loaded in the six periphery cells. TRIGA fuel elements or sealed cans may not be loaded in the center cell of the non-poisoned basket. Damaged TRIGA fuel elements or fuel debris (up to two equivalent elements of maximum 1.5 inch diameter) shall be transported in a sealed damaged fuel can (one damaged fuel can per cell). The sealed cans are to be in accordance with NAC International Drawing Nos. 315-40-086, 315-40-087, and 315-40-088.

Loading of TRIGA HEU and LEU fuel elements having >138 g and >169 g initial ^{235}U mass contents, respectively, are limited to top and bottom basket modules and up to three rods per basket cell. A minimum of one TRIGA dummy rod per NAC Drawing No. 315-40-085 shall be installed in place of a TRIGA fuel element to limit the maximum number of rods per cell to three.

Mixed loading in separate cells of TRIGA fuel elements and TRIGA fuel cluster rods [per 5.(b)(1)(vii)(b)] is authorized in fuel basket modules with the content quantities limited in accordance with the other conditions and limitations of 5.(b)(2)(vii)(c) and 5.(b)(2)(viii)(b).

Maximum decay heat not to exceed 7.5 watts per TRIGA fuel element (or equivalent for damaged fuel) and 900 watts per package. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket. The cask and baskets must be configured as shown in NAC International Drawing No. 315-40-079.

(viii)(a) For TRIGA fuel cluster rods as described in Item 5.(b)(1)(vii)(a):

Maximum decay heat not to exceed 1.875 watts per TRIGA fuel cluster rod (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel cluster rods must be positioned in either the non-poisoned TRIGA fuel basket or in the poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket. The non-poisoned basket must be configured as shown in NAC International Drawing Nos. 315-40-070, 315-40-071, and 315-40-072, and the poisoned basket must be configured as shown in NAC International Drawing Nos. 315-40-080, 315-40-081, and 315-40-082.

Up to 480 intact cluster rods per package in the non-poisoned TRIGA fuel baskets (up to six periphery cells loaded with 16 cluster rods each), and up to 560 intact cluster rods per package in the poisoned TRIGA fuel baskets (up to 7 total cells loaded with 16 cluster rods each). TRIGA fuel cluster rods must be positioned within the fuel rod inserts as shown on NAC International Drawing No. 315-40-096.

5.(b)(2) Maximum quantity of material per package (continued)

Damaged TRIGA fuel cluster rods or cluster rod debris (up to six equivalent rods) shall be transported in a sealed damaged fuel can. The sealed cans are to be in accordance with NAC International Drawing Nos. 315-40-086, 315-40-087, and 315-40-088.

Mixed intact and damaged fuel contents and fuel debris are authorized. Base and top fuel basket modules may contain intact fuel cluster rods or sealed DFCs. Intermediate fuel basket modules may contain only intact fuel cluster rods.

(viii)(b) For TRIGA fuel cluster rods as described in Item 5.(b)(1)(vii)(b):

Maximum decay heat not to exceed 1.875 watts per TRIGA fuel cluster rod (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel cluster rods must be positioned in the non-poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket. The non-poisoned basket must be configured as shown in NAC International Drawing Nos. 315-40-070, 315-40-071, and 315-40-072.

Up to 480 intact cluster rods per package in the non-poisoned TRIGA fuel baskets (up to six periphery cells loaded with 16 cluster rods each), and up to 560 intact cluster rods per package in the poisoned TRIGA fuel baskets (up to 7 total cells loaded with 16 cluster rods each). TRIGA fuel cluster rods must be positioned within the fuel rod inserts as shown on NAC International Drawing No. 315-40-096.

Damaged TRIGA fuel cluster rods or cluster rod debris (up to six equivalent rods) shall be transported in a sealed damaged fuel can. The sealed cans are to be in accordance with NAC International Drawing Nos. 315-40-086, 315-40-087, and 315-40-088.

Mixed intact and damaged fuel contents and fuel debris are authorized. Base and top fuel basket modules may contain intact fuel cluster rods or sealed DFCs. Intermediate fuel basket modules may contain only intact fuel cluster rods.

Mixed loading in separate cells of TRIGA fuel elements [per 5.(b)(1)(vi)(c)] and TRIGA fuel cluster rods [per 5.(b)(1)(vii)(b)] is authorized in fuel basket modules with the content quantities limited in accordance with the other conditions and limitations of 5.(b)(2)(vii)(c) and 5.(b)(2)(viii)(b).

(ix) For high burnup PWR fuel rods, as described in Item 5.(b)(1)(viii): up to 25 fuel rods. Maximum decay heat not to exceed 2.3 kilowatts per package.

Intact individual rods may be placed either in an irradiated or unirradiated fuel assembly lattice (skeleton) or in a fuel rod insert. The PWR fuel assembly lattice must be transported in the PWR basket.

5.(b)(2) Maximum quantity of material per package (continued)

Up to 14 of the 25 fuel rods may be classified as damaged. Damaged fuel rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert. Damaged fuel rods may also be placed in individual failed fuel rod capsules, as shown in Figure 1.2.3-11 of the application, prior to placement in the fuel rod insert. Guide/instrument tubes and tube segments may be placed in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket.

(x) For high burnup BWR fuel rods, as described in Item 5.(b)(1)(ix): up to 25 fuel rods. Maximum decay heat not to exceed 2.1 kilowatts per package.

Intact individual rods may be placed either in a fuel assembly lattice or in a fuel rod insert. The BWR fuel assembly lattice must be transported in the PWR insert in the PWR basket.

Up to 14 of the 25 fuel rods may be classified as damaged. Damaged fuel rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert. Damaged fuel rods may also be placed in individual failed fuel rod capsules, as shown in Figure 1.2.3-11 of the application, prior to placement in the fuel rod insert. Water rods and inert rods may be placed in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket.

(xi) For DIDO fuel as described in Item 5.(b)(1)(x):

(a) Up to 42 DIDO fuel elements with a maximum decay heat not to exceed 25 watts per DIDO fuel element, provided the top basket fuel element active fuel region is spaced a minimum 3.7 inches from the bottom of the cask lid. Spacing of the active fuel may be accomplished by fuel element hardware, lid spacer, or a combination thereof. Maximum decay heat is 1.05 kilowatts per package. At a top basket active fuel region to cask lid spacing of less than 3.7 inches, the maximum decay heat not to exceed 18 watts per DIDO fuel element and a total of 756 watts per package. The DIDO fuel elements are to be loaded into a DIDO basket configured as shown in NAC International Drawing No. 315-40-111.

(b) A mixed fuel load of up to 42 DIDO fuel elements and spiral and MOATA fuel assemblies [per item 5.(b)(1)(xiv)] in an ANSTO-DIDO combination basket configured as shown in NAC International Drawing No. 315-40-148 consisting of a top ANSTO basket module per NAC International Drawing No. 315-40-140; four intermediate DIDO basket modules per NAC International Drawing No. 315-40-109; and one bottom DIDO basket module per NAC International Drawing No. 315-40-110. DIDO fuel elements loaded into intermediate and bottom basket modules are limited to ≤18 Watts. Up to seven degraded clad DIDO, spiral, and/or MOATA fuel assemblies in DFCs per Figure 1.2.3-18 of the application, or intact DIDO, spiral, and/or MOATA assemblies may be loaded in the top ANSTO module. The per element or DFC heat load limits for the top ANSTO module are: DIDO fuel element with or without DFC is 10 Watts; spiral fuel element in DFC is 10 Watts and 15.7W without DFC; and MOATA fuel element in DFC is 1 Watt and 3 Watts without DFC. Maximum heat load per package is 753 Watts.

5.(b)(2) Maximum quantity of material per package (continued)

(xii) For GA IFM as described in Item 5.(b)(1)(xi):

(a) Mixture of fuel particles (kernels and coatings), fuel compacts (rods), and fuel pebbles, packaged in its own Fuel Handling Unit (FHU).

GA HTGR FHU consists of two redundant canisters. GA HTGR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032237, Rev. B, "HTGR Primary Enclosure." The primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032231, Rev. A, "HTGR Secondary Enclosure."

GA HTGR FHU total maximum decay heat not to exceed 2.05 watts, and maximum loaded weight not to exceed 71.5 lbs.

(b) Twenty irradiated TRIGA fuel elements; 13 of the elements are intact, and the remaining 7 are sectioned. GA RERTR IFM is packaged in its own FHU.

GA RERTR FHU consists of two redundant canisters. GA RERTR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032236, Rev. B, "RERTR Primary Enclosure." The GA RERTR IFM primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032230, Rev. A, "RERTR Secondary Enclosure."

GA RERTR FHU total maximum decay heat not to exceed 11 watts, and maximum loaded weight not to exceed 76.0 lbs.

(xiii) For TPBARs as described in Item 5.(b)(1)(xii):

Up to 300 TPBARs, including a maximum of 2 damaged rods, positioned within a consolidation canister, as shown in Figure 1.2.3-10 of the application. The consolidation canister is transported in a TPBAR basket assembly. The maximum decay heat is 2.31 watts per rod and 693 watts per package. The maximum weight of the TPBARs and the consolidation canister is 1,000 pounds. Consolidation canisters with fewer than 300 TPBARs may also contain stainless steel spacers of various geometries. The total weight and volume of the reduced TPBAR contents plus the spacers must be less than or equal to the weight and volume of 300 TPBARs.

Up to 25 TPBARS, including a maximum of 2 prefailed rods, positioned within a PWR/BWR Rod Transport Canister. The PWR/BWR Rod Transport Canister is transported in a TPBAR basket assembly. The maximum decay heat is 2.31 watts per rod and 58 watts per package.

(xiv) For PULSTAR fuel as described in Item 5.(b)(1)(xiii):

Up to 700 intact or damaged PULSTAR fuel elements in either assembly or element form, including fuel debris, pellets, pieces and nonfuel components of PULSTAR fuel assemblies. The contents of a PULSTAR can are restricted to the equivalent of the fuel material in 25 intact PULSTAR fuel elements and of the displaced volume of 25 intact PULSTAR fuel elements.

5.(b)(2) Maximum quantity of material per package (continued)

(xv) For ANSTO fuel as described in Item 5.(b)(1)(xiv):

(a) Up to 42 spiral fuel assemblies, MOATA plate bundles, or any combination of spiral fuel assemblies and MOATA plate bundles. ANSTO fuel must be loaded within ANSTO basket modules. Spiral fuel assemblies may be cropped by removing nonfuel-bearing hardware to fit the ANSTO basket modules. Fuel assemblies that are cropped, but are otherwise intact, may be considered intact. For spiral fuel assemblies, the maximum decay heat per assembly is 15.7 watts. The minimum cool time as a function of burnup shall be consistent with the maximum decay heat limit and shall be determined using the procedures for medium enriched DIDO fuel in Section 7.1.4 of the application; the minimum cool time may not be less than 270 days. For MOATA plate bundles, the maximum heat load per bundle is 3 watts, and the minimum cool time is 10 years.

(b) A mixed fuel load of up to 42 spiral and MOATA fuel assemblies and DIDO fuel elements [per item 5.(b)(1)(x)] in an ANSTO basket configured as shown in NAC International Drawing No. 315-40-139. Degraded clad elements placed in DFCs per Figure 1.2.3-18 of the application or intact DIDO fuel elements are limited to loading in the top ANSTO basket module. Maximum heat load per DIDO element is 10W. Degraded clad spiral and MOATA fuel assemblies in DFCs are also limited to loading in the top ANSTO basket module. Spiral fuel assemblies placed into DFCs are limited to a maximum of 10W and MOATA plate bundles loaded in DFCs are limited to 1W. Spiral fuel elements not placed in DFCs are limited to 15.7W and MOATA plate bundles not placed in DFCs are limited to a maximum of 3W with a minimum cool time of 10 years.

(xvi) For segmented TPBARs as described in Item 5.(b)(1)(xv):

Up to 55 equivalent TPBARs as segments and segmentation debris, placed within a welded waste container, as shown in Figure 1.2.3-16 of the application. The waste container is transported in a TPBAR basket assembly. The maximum decay heat is 2.31 watts per equivalent TPBAR and 127 watts per package. The maximum weight of the segmented TPBARs and the TPBAR waste container is 700 pounds.

(xvii) For solid irradiated hardware as described in Item 5.(b)(1)(xvi):

Up to 4,000 pounds, including spacers, dunnage and containers, and meeting the gamma source defined in Table 1.2-13 of the application. An irradiated hardware spacer source, per NAC Drawing No. 315-40-145, shall be installed.

5.(b)(2) Maximum quantity of material per package (continued)

(xviii) For intact PWR MOX fuel rods as described in Item 5.(b)(1)(xvii):

Up to 16 undamaged irradiated PWR MOX rods or a combination of PWR MOX and high burnup PWR fuel rods as described in Item 5.(b)(1)(viii). Maximum decay heat not to exceed 2.3 kW per package. Individual PWR MOX and PWR UO_2 fuel rods shall be placed in a 5x5 insert loaded into a screened or free flow rod canister in accordance with NAC International Drawing No. 315-40-104, for transport. Up to nine nonstainless burnable poison rods (BPRs) may be loaded in the spare locations in the 5x5 insert. The PWR/BWR fuel rod canister shall be transported in the PWR basket and the PWR insert installed in the cask cavity.

(xix) For the SLOWPOKE fuel described in Item 5.(b)(1)(xviii):

Up to 100 SLOWPOKE fuel rods (or the equivalent quantity of damaged material) may be loaded per SLOWPOKE canister in accordance with NAC Drawing No. 315-40-156 utilizing either a 4x4 or 5x5 tube array or any combination thereof. Up to 4 SLOWPOKE canisters may be loaded within a 28 MTR fuel basket module with the three center fuel cells blocked. Only the top and top intermediate fuel basket modules may be loaded with SLOWPOKE fuel. Cask configuration is to be in accordance with NAC Drawing No. 315-40-158.

(xx) For NRU/NRX fuel described in item 5.(b)(1)(xix):

Up to 18 undamaged NRU or NRX fuel assemblies (or the equivalent number of loose rods) may be loaded per NRU/NRX fuel basket in accordance with NAC Drawing Nos. 315-40-172, 315-40-173, 315-40-174 and 315-40-175. Package configuration to be in accordance with NAC Drawing No. 315-40-170. NRX fuel shall be placed into the fuel caddy. Placement of NRU fuel into the fuel caddy is optional. NRU and NRX fuel may not be comingled within a single package.

5(c) Criticality Safety Index (CSI)

For PWR fuel assemblies described in 5(b)(1)(i) and limited in 5(b)(2)(i)	100
For BWR fuel assemblies described in 5(b)(1)(ii) and limited in 5(b)(2)(ii)	5.0
For MTR fuel elements described in 5(b)(1)(iv) and limited in 5(b)(2)(iv)	0.0
For metallic fuel rods described in 5(b)(1)(v) and limited in 5(b)(2)(v) and (vi)	0.0
For TRIGA fuel elements (in poisoned TRIGA fuel baskets) described in 5(b)(1)(vi)(a) and limited in 5(b)(2)(vii)(a)	0.0

5(c) Criticality Safety Index (CSI)

For TRIGA fuel elements (in nonpoisoned TRIGA fuel baskets) described in 5(b)(1)(vi)(b) and 5(b)(1)(vi)(c) and limited in 5(b)(2)(vii)(b) and 5(b)(2)(vii)(c), respectively	12.5
For mixed loads of TRIGA fuel elements described in 5(b)(1)(vi)(c) and limited in 5(b)(2)(vii)(c), and TRIGA fuel cluster rods described in 5(b)(1)(vii)(b) and limited in 5(b)(2)(viii)(b)	12.5
For TRIGA fuel cluster rods described in 5(b)(1)(vii) and limited in 5(b)(2)(viii)	0.0
For high burnup PWR rods described in 5(b)(1)(viii) and limited in 5(b)(2)(ix)	0.0
For high burnup BWR rods described in 5(b)(1)(ix) and limited in 5(b)(2)(x)	0.0
For DIDO fuel elements described in 5(b)(1)(x) and limited in 5(b)(2)(xi)	12.5
For General Atomic Irradiated Fuel Material (GA IFM) described in 5(b)(1)(xi) and limited in 5(b)(2)(xii)	0.0
For TPBARS and segmented TPBARS described in 5(b)(1)(xii) and 5(b)(1)(xv) and limited in 5(b)(2)(xiii) and 5(b)(2)(xvi)	0.0
For intact (uncanned) PULSTAR fuel described in 5(b)(1)(xiii) and limited in 5(b)(2)(xiv)	0.0
For (canned) PULSTAR fuel described in 5(b)(1)(xiii) and limited in 5(b)(2)(xiv) – for a package with any number of PULSTAR cans	33.4
For ANSTO fuel described in 5(b)(1)(xiv) and limited in 5(b)(2)(xv)	0.0
For solid irradiated hardware described in 5(b)(1)(xvi) and limited in 5(b)(2)(xvii)	0.0
For PWR MOX rods described in 5.(b)(1)(xvii) and limited by 5(b)(2)(xviii)	0.0

5(c) Criticality Safety Index (CSI)

For a mixed fuel load of DIDO and ANSTO fuel elements described in 5(b)(1)(x) and 5(b)(1)(xiv) and limited by 5(b)(2)(xi)(b) and 5(b)(2)(xv)(b)	0.0
For (canned) SLOWPOKE fuel described in 5(b)(1)(xviii) and limited by 5(b)(2)(xix)	0.0
For the NRU/NRX fuel described in 5(b)(1)(xix) and limited by 5(b)(2)(xx)	100.0

6. Known or suspected damaged fuel assemblies (rods) or elements, and fuel with cladding defects greater than pin holes and hairline cracks are not authorized, except as described in Items 5.(b)(1)(x); 5.(b)(1)(xiv); 5.(b)(1)(xviii); 5.(b)(2)(iv)(d); 5.(b)(2)(vi); 5.(b)(2)(vii)(a); 5.(b)(2)(vii)(b); 5.(b)(2)(viii); 5.(b)(2)(ix); 5.(b)(2)(x); 5.(b)(2)(xi); 5.(b)(2)(xiv); 5.(b)(2)(xv); and 5.(b)(2)(xix).

7. The cask must be dry (no free water) when delivered to a carrier for transport.

8. Bolt torque: The cask lids bolts must be torqued to 260 +/- 20 ft-lbs. The bolts used to secure the alternate vent and drain port covers must be torqued to 100 +/- 10 inch-lbs. The bolts used to secure the Alternate B port covers must be torqued to 285 +/- 15 inch-lbs.

9. Prior to each shipment, the package must be leak tested to 1 x 10^{-3} std cm^3/sec, except that replaced seals must be leak tested to 2.0 x 10^{-7} std cm^3/sec (He). Prior to first use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 2.0 x 10^{-7} std cm^3/sec (He).

10. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) The metallic O-ring lid seal must be replaced prior to each shipment; and

    (b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application; and

    (c) The package shall be prepared for shipment and operated in accordance with the Package Operations of Chapter 7 of the application. If the cask is loaded under water or water is introduced into the cask cavity, the cask must be vacuum dried as described in Chapter 7 of the application. The cask cavity must be backfilled with 1.0 atm of helium when shipping PWR or BWR assemblies, individual PWR and BWR rods, or TPBAR contents.

11. When shipping PWR, BWR, PWR MOX, MTR, DIDO assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, high burnup PWR or BWR rods, GA IFM, PULSTAR fuel elements, spiral fuel assemblies, and MOATA plate bundles, the neutron shield tank must be filled with a mixture of water and ethylene glycol which will not freeze or precipitate in a temperature range from -40 °F to 250 °F. The water and ethylene glycol mixture must contain at least 1% boron by weight.

12. A personnel barrier must be used when shipping PWR or BWR assemblies. Shipments of MTR, DIDO fuel assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, high burnup PWR or BWR rods, PWR MOX rods, TPBAR contents, PULSTAR fuel elements, spiral fuel assemblies, MOATA plate bundles, or irradiated hardware must use the ISO container or a personnel barrier.

13. Packages used to ship metallic fuel rods may be shipped in a closed shipping container provided that the closed container, the cask tie-down and support system and transport vehicle (trailer) meet the applicable requirements of the Department of Transportation. When the cask is shipped in a closed shipping container, the center of gravity of the combined cask, closed shipping container and trailer must not exceed 75 inches.

14. For shipment of TPBAR contents:

    (a) Prior to first use for shipment of TPBAR contents, each packaging must be hydrostatic pressure tested to 450 +15/-0 psig, as described in Section 8.1.2 of the application;

    (b) The package must be marked with Package Identification Number USA/9225/B(M)-96;

    (c) The package must be configured as shown in NAC International Drawing No. 315-40-128, Rev. 4 (Sheets 1-2), for the applicable TPBAR contents; and

    (d) Prior to each shipment, after loading, each cask containment seal must be tested to show no leakage greater than 2×10^{-7} std-cm^3/s (helium).

15. For shipment of PULSTAR fuel:

    (a) Intact fuel elements may be configured as PULSTAR fuel assemblies, may be placed into a TRIGA fuel rod insert (a 4 x 4 rod holder), or may be loaded into PULSTAR fuel cans. Intact PULSTAR fuel assemblies and PULSTAR fuel elements in a TRIGA fuel rod insert may be loaded in any module of the 28 MTR basket assembly. PULSTAR fuel cans may only be loaded into the top or base module of the 28 MTR basket assembly.

    (b) Damaged PULSTAR fuel elements and nonfuel components of PULSTAR fuel assemblies must be loaded into PULSTAR cans. Damaged PULSTAR fuel, including fuel debris, pellets or pieces, may be placed in an encapsulating rod prior to loading into a PULSTAR fuel can. PULSTAR fuel cans may only be loaded into the top or base module of the 28 MTR basket assembly.

    (c) Loading of modules with mixed PULSTAR payload configuration is allowed.

16. For shipment of non-fissile contents, with fissile content in the package not exceeding Type A quantity, and qualifying as a fissile exempt quantity under 10 CFR 71.15, the Model No. NAC-LWT shall be designated as Type B(U)F-96, with package identification number USA/9225/B(U)-96.

17. Transport by air is not authorized.

18. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

19. Revision 55 of this certificate may be used until December 31, 2013.

20. Revision 57 of this certificate may be used until February 28, 2014.

21. Expiration Date: February 28, 2015.

REFERENCES

NAC International, Inc., application dated June 18, 2010.

NAC International, Inc., supplements dated February 3, March 2, and May 24, October 26, and December 5, 2012; January 14, and February 14, 2013.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Michele M. Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 2/28/13

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9226	4	71-9226	USA/9226/B(U)F-85	1	OF	9

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

General Atomics
3550 General Atomics Court
San Diego, California 92121-1122

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated January 6, 2009

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a. Packaging

(1) Model No.: GA-4

(2) Description

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the packaging (cask and impact limiters) and the radioactive contents. The packaging is designed to transport up to four intact pressurized-water reactor (PWR) irradiated spent fuel assemblies as authorized contents. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The overall dimensions of the packaging are approximately 90 inches in diameter and 234 inches long.

The containment system includes the cask body (cask body wall, flange, and bottom plate); cask closure; closure bolts; gas sample valve body; drain valve; and primary O-ring seals for the closure, gas sample valve, and drain valve.

<u>Cask Assembly</u>
The cask assembly includes the cask, the closure, and the closure bolts. Fuel spacers are also provided when shipping specified short fuel assemblies to limit the movement of the fuel. The cask is constructed of stainless steel, depleted uranium, and a hydrogenous neutron shield. The cask external dimensions are approximately 188 inches long and 40 inches in diameter. A fixed fuel support structure divides the cask cavity into four spent fuel compartments, each approximately 8.8 inches square and 167 inches long. The closure is recessed into the cask body and is attached to the cask flange with 12 1-inch diameter bolts. The closure is approximately 26 inches square, 11 inches thick, and weighs about 1510 lbs.

5.a. (2) (continued)

The cask has two ports allowing access to the cask cavity. The closure lid has an integral half-inch diameter port (hereafter referred to as the gas sample valve) for gas sampling, venting, pressurizing, vacuum drying, leakage testing, or inerting. A 1-inch diameter port in the bottom plate allows draining, leakage testing, or filling the cavity with water. A separate drain valve opens and closes the port. The primary seals for the gas sample valve and drain valve are recessed from the outside cask surface as protection from punctures. The gas sample valve and the drain valve also have covers to protect them during transport.

Cask

The cask includes the containment (flange, cask body, bottom plate and drain valve seals); the cavity liner and fuel support structure; the impact limiter support structure; the trunnions and redundant lift sockets; the depleted uranium gamma shield; and the neutron shield and its outer shell. The cask body is square, with rounded corners and a transition to a round outer shell for the neutron shield. The cask has approximately a 1.5 inch thick stainless steel body wall, 2.6 inch thick depleted uranium shield (reduced at the corners), and 0.4 inch thick stainless steel fuel cavity liner.

The cruciform fuel support structure consists of stainless steel panels with boron-carbide (B_4C) pellets for criticality control. A continuous series of holes in each panel, at right angles with the fuel support structure axis, provides cavities for the B_4C pellets. The fuel support structure is welded to the cavity liner and is approximately 18 inches square by 166 inches long and weighs about 750 lbs.

The flange connects the cask body wall and fuel cavity liner at the top of the cask, and the bottom plate connects them at the bottom. The gamma shield is made up of five rings, which are assembled with zero axial tolerance clearance within the depleted uranium cavity, to minimize gaps. The impact limiter support structure is a slightly tapered 0.4 inch thick shell on each end of the cask. The shell mates with the impact limiter's cavity and is connected to the cask body by 36 ribs.

The neutron shield is located between the cask body and the outer shell. The neutron shield design maintains continuous shielding immediately adjacent to the cask body under normal conditions of transport. The details of the design are proprietary. The design, in conjunction with the operating procedures, ensures the availability of the neutron shield to perform its function under normal conditions of transport.

Two lifting and tie-down trunnions are located about 34 inches from the top of the cask body, and another pair is located about the same distance from the bottom. The trunnion outside diameter is 10 inches, increasing to 11.5 inches at the cask interface. Two redundant lift sockets are located about 26 inches from the top of the cask body and are flush with the outer skin.

5.a. (2) (continued)

Materials
All major cask components are stainless steel, except the neutron shield, the depleted uranium gamma shield, and the B_4C pellets contained in the fuel support structure. All O-ring seals are fabricated of ethylene propylene.

Impact Limiters
The impact limiters are fabricated of aluminum honeycomb, completely enclosed by an all-welded austenitic stainless steel skin. Each of the two identical impact limiters is attached to the cask with eight bolts. Each impact limiter weighs approximately 2,000 lbs.

(3) Drawings
The packaging is constructed and assembled in accordance with the following GA Drawing Number:

Drawing No. 031348,
sheets 1 through 19, Revision D (Proprietary Version)
GA-4 Spent Fuel Shipping Cask Packaging Assembly

5.(b) Contents

(1) Type and Form of Material:

(a) Intact fuel assemblies. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

(b) The fuel authorized for shipment in the GA-4 package is irradiated 14x14 and 15x15 PWR fuel assemblies with uranium oxide fuel pellets. Before irradiation, the maximum enrichment of any assembly to be transported is 3.15 percent by weight of uranium-235 (^{235}U). The total initial uranium content is not to exceed 407 Kg per assembly for 14x14 arrays and 469 Kg per assembly for 15x15 arrays.

c) Fuel assemblies are authorized to be transported with or without control rods or other non-fuel assembly hardware (NFAH). Spacers shall be used for the specific fuel types, as shown on sheet 17 of the Drawings.

(d) The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U or 45,000 MWd/MTU with a minimum cooling time of 15 years (no minimum enrichment).

(e) The maximum assembly decay heat of an individual assembly is 0.617 kW. The maximum total allowable cask heat load is 2.468 kW (including control components and other NFAH when present).

5.b. (1) (continued)

(f) The PWR fuel assembly types authorized for transport are listed in Table 1. All parameters are design nominal values.

(2) Maximum Quantity of Material per Package

(a) For material described in 5.b.(1): four (4) PWR fuel assemblies.

(b) For material described in 5.b.(1): the maximum assembly weight (including control components or other NFAH when present) is 1,662 lbs. The maximum weight of the cask contents (including control components or other NFAH when present) is 6,648 lbs., and the maximum gross weight of the package is 55,000 lbs.

Table 1 - PWR Fuel Assembly Characteristics

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-15x15 (Std/ZC)	469	204	0.563	0.3659	0.0242	144
W-15x15 (OFA)	463	204	0.563	0.3659	0.0242	144
BW-15x15 (Mk.B,BZ,BGD)	464	208	0.568	0.3686	0.0265	142
Exx/A-15x15 (WE)	432	204	0.563	0.3565	0.030	144
CE-15x15 (Palisades)	413	204	0.550	0.358	0.026	144
CE-14x14 (Ft.Calhoun)	376	176	0.580	0.3765	0.028	128
W-14x14 (Model C)	397	176	0.580	0.3805	0.026	137
CE-14x14 (Std/Gen.)	386	176	0.580	0.3765	0.028	137
Exx/A-14x14 (CE)	381	176	0.580	0.370	0.031	137

5.b(2)(b)(continued)

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-14x14 (OFA)	358	179	0.556	0.3444	0.0243	144
W-14x14 (Std/ZCA,/ZCB)	407	179	0.556	0.3674	0.0225	145.5
Exx/A-14x14 (WE)	379	179	0.556	0.3505	0.030	142

5.c. Criticality Safety Index (CSI): 100

6. Fuel assemblies with missing fuel pins shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.

7. In addition to the requirements of Subpart G of 10 CFR 71:

   a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed using the specifications contained within the application. At a minimum, those procedures shall require the following provisions:

      (1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

      (2) That before shipment the licensee shall:

         (a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.
         (b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates: 3.4 x (peak neutron dose rate at any point on cask surface at its midlength) + 1.0 x (gamma dose rate at that location) $\leq$ 1000 mR/hr.
         (c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.

7.a.(2) (continued)

(d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1 x10^{-3} standard cubic centimeters per second of air (std-cm^3/sec) and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in Condition 7.b.(5), below.

(3) Before leak testing, the following closure bolt and valve torque specifications:

(a) The cask lid bolts shall be torqued to 235 ± 15 ft-lbs.
(b) The gas sample valve and drain valve shall be torqued to 20 ± 2 ft-lbs.

(4) During wet loading operations and prior to leak testing, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:

(a) Cask evacuation to a pressure of 0.2 psia (10 mm Hg) or less for a minimum of 1 hour.
(b) Verifying that the cask pressure rise is less than 0.1 psi in 10 minutes.

(5) Before shipment, independent verification of the material condition of the neutron shield as described in SAR Section 7.1.1.4 or 7.1.2.4.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed using the specifications contained within the application and shall include the following provisions:

(1) All containment boundary welds, except the final fabrication weld joint connecting the cask body wall to the bottom plate, shall be radiographed and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB. Examination of the final fabrication weld joint connecting the cask body wall to the bottom plate may be ultrasonic and progressive liquid penetrant examined in lieu of radiographic and liquid penetrant examination.

(2) The upper lifting trunnions and redundant lifting sockets shall be load tested, in the cask axial direction, to 300 percent of their maximum working load (79,500 lbs. minimum) per trunnion and per lifting socket, in accordance with the requirements of ANSI N14.6. The upper and lower lifting trunnions shall be load tested, in the cask transverse direction, to 150 percent of their maximum working load (20,625 lbs. minimum) per trunnion, in accordance with the requirements of ANSI N14.6.

7.b.(continued)

(3) The cask containment boundary shall be pressure tested to 1.5 times the Maximum Normal Operating Pressure of 80 psig. The minimum test pressure shall be 120 psig.

(4) All containment seals shall be replaced within the 12-month period prior to each shipment.

(5) A fabrication leakage test shall be performed on all containment components including the O-ring seals prior to first use. Additionally, all containment seals shall be leak tested after the third use of each package and within the 12-month period prior to each shipment. Any replaced or repaired containment system component shall be leak tested. The leakage tests shall verify that the containment boundary leakage rate does not exceed the design leakage rate of 1×10^{-7} std-cm^3/sec. The leak tests shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

(6) The depleted uranium shield shall be gamma scanned with 100 percent inspection coverage during fabrication to ensure that there are no shielding discontinuities. The neutron shield supplier shall certify that the shield material meets the minimum specified requirements (proprietary) used in the applicant's shielding analysis.

(7) Qualification and verification tests to demonstrate the crush strength of each aluminum honeycomb type and lot to be utilized in the impact limiters shall be performed.

(8) The boron carbide pellets, fuel support structure and fuel cavity dimensions, and ^{235}U content in the depleted uranium shall be fabricated and verified to be within the specifications of Table 2 to ensure criticality safety.

Table 2

Specified Parameter	Minimum	Maximum
B_4C boron enrichment	96 wt% ^{10}B	N/A
Diameter of each B_4C pellet	0.426 in	0.430 in
Height of each B_4C pellet stack	7.986 in	8.046 in
Mass of ^{10}B in each B_4C pellet stack	31.5 g	N/A
Mass of each B_4C pellet stack	43.0 g	45.0 g
Diameter of each fuel support structure hole	0.432 in	0.44 in
Fuel support structure nominal hole pitch	N/A	0.55 in
Fuel support structure hole depth minus B_4C pellet-stack height (at room temperature)	0.009 in	0.129 in
Thickness of each fuel support structure panel	0.600 in	0.620 in
Fuel cavity width	N/A	9.135 in
^{235}U content in depleted uranium shielding material	N/A	0.2 wt%

8. Transport of fissile material by air is not authorized.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Fabrication of new packagings is not authorized.

11. Expiration Date: October 31, 2018.

REFERENCES

General Atomics Application for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, January 6, 2009.

As supplemented: September 11, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date 9/27/2013

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9228	25	71-9228	USA/9228/B(U)F-96	1 OF 10

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Electric Company* application dated December 12, 2000, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 2000

(2) Description

A steel encased lead shielded shipping cask. The cask is within a double-walled overpack with toroidal shell impact limiters at each end. The overall dimensions are approximately 131.5 inches in height and 72.0 inches in diameter. The cask is transported in the upright or horizontal position. The gross weight of the package is approximately 33,550 lbs.

The cask is constructed of two concentric 1-inch thick 304 stainless steel cylindrical shells (ASTM A 240) joined at the bottom end to a 6-inch thick 304 stainless steel forging (ASTM A 182). The annulus between the two shells is filled with lead approximately 4 inches thick. The cask is approximately 71.0 inches in height and has an outer diameter of 38.5 inches. The cask cavity is approximately 26.5 inches in diameter and 54.0 inches deep.

The cask lid is 304 stainless steel and lead, has a stepped design, and is fully recessed into the cask top flange. The lid is secured to the cask body by 15, 1.25-inch diameter socket head screws. The cask is sealed by elastomeric O-rings bonded to a thin aluminum disc-shaped ring. The cask is equipped with a seal test port on the side of the cask body, a vent port in the cask lid, and a drain port near the bottom of the cask.

The cask is positioned within an overpack constructed from two 0.5-inch thick concentric 304 stainless steel cylindrical shells (ASTM A 240). The shells are separated radially by eight

*This license was transferred from General Electric Company to GE-Hitachi Nuclear Energy Americas, LLC, in 2007.

equally spaced tubes and horizontally by two tube sections. A 304 stainless steel toroidal shell impact limiter is attached to each end of the overpack. The overpack opens just above the lower impact limiter for access to the cask. The top of the overpack is joined to the base by 15, 1-3/8-inch diameter shoulder screws.

5(a) (2) Description (Continued)

Gussets on the top and bottom impact limiters provide tie-down points for the package. The cask body is equipped with attachment plates for lifting devices. The cask lifting devices are detached during transport.

(3) Drawings

(i) The packaging is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 105E9520, Rev. 4; and 105E9521, Rev. 5.

(ii) Packaging Serial No. 2001 is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 101E8718, Rev. 12; and 101E8719, Rev. 12.

(iii) The HFIR fuel basket and liner are constructed and assembled in accordance with General Electric Company Drawing No. 105E9523, Rev. 3.

(iv) The multifunctional rack is constructed and assembled in accordance with General Electric Company Drawing No. 105E9555, Rev. 2.

(v) The barrel rack is constructed and assembled in accordance with General Electric Company Drawing No. 166D8066, Rev. 2.

(vi) The material basket is constructed in accordance with General Electric Company Drawing No. 183C8356, Rev. 2. The material basket may be used with the multifunctional rack and the barrel rack.

(vii) The TSR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9560, Rev. 2.

(viii) The MTR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9557, Rev. 9.

(ix) The optional lead liner is constructed and assembled in accordance with General Electric Company Drawing No. 129D4922, Rev. 2.

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5.(b) Contents

(1) Type and form of material

(i) Irradiated fuel rods, which may be cut or segmented.

(ii) Byproduct, source, or special nuclear material in solid form.

(iii) Irradiated High Flux Isotope Reactor (HFIR) fuel assembly, positioned within the HFIR fuel basket and liner as specified in 5(a)(3). The HFIR fuel assembly is fabricated in accordance with Oak Ridge National Laboratory Drawing Nos. M-11524-OH-101-D, Rev. 0, and M-11524-OH-102-D, Rev. 0.

(iv) Irradiated Tower Shielding Reactor (TSR) fuel elements, positioned within the TSR fuel basket specified in 5(a)(3).

5.(b)(1) Type and form of material (continued)

(v) Irradiated MTR-type fuel assemblies, positioned within the MTR fuel basket specified in 5(a)(3). The fuel assemblies may be sectioned only in the non-fuel bearing region of the assembly. The fuel assemblies are composed of aluminum clad plates, and are limited as follows:

Fuel material	U_3O_8	UAl_x	U_{METAL}
Max. uranium enrichment (w/o U-235)	94.0	94.0	95.0
Max. active fuel thickness (in)	0.023	0.020	0.020
Min. clad thickness (in)	0.014	0.015	0.015
Max. U-235 per fuel assembly (g)	355	290	110
Max. U-235 mass per fuel basket cell (g)	710	580	220
Max. burnup (GWd/MTU)	568	568	568
Min. cool time (days)	120	120	120

Fuel material	U_3Si_2	UAl_x
Max. uranium enrichment (w/o U-235)	20.0	20.0
Max. active fuel thickness (in)	0.020	0.100

Min. clad thickness (in)	0.015	0.010
Max. U-235 per fuel assembly (g)	347	150
Max. U-235 mass per fuel basket cell (g)	694	300
Max. burnup (GWd/MTU)	122	122
Min. cool time (days)	120	120

Note: The enrichments, masses, and dimensions shall be based on values prior to irradiation.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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5.(b) (1) Type and form of material (Continued)

(vi) Irradiated TRIGA fuel elements, positioned with the MTR fuel basket specified in 5(a)(3). The fuel material consists of $UZrH_x$ in cylindrical elements, with aluminum, stainless steel, or inconel cladding. The H to Zr ratio in the fuel ranges from approximately 1.0 to 1.7. Some fuel elements contain graphite reflectors in each end of the fuel element. The fuel elements are limited as follows:

Approximate rod diameter (in)	1-1/2	½	1-1/2	1-1/2	½
Graphite reflectors	With or without reflectors	With or without reflectors	With reflectors	With reflectors	Without reflectors
Uranium concentration in fuel (w/o U)	8 - 45	10 - 45	8.5 min.	8.5 min.	10 min.
Max. rod length (in)	30	30	30	30	30
Max. active fuel length (in)	15	22	15	15	22
Min. clad thickness (in)	0.02	0.016	0.02	0.02	0.016
Max. uranium enrichment (w/o U-235)	20.0	20.0	70.0	94.0	94.0
Max. active fuel diameter (in)	1.435	0.51	1.435	1.435	0.51
Max. U-235 per rod (g)	165	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)	140	220	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)
Max. U-235 mass per fuel basket cell (g)	560	660	560	660	660
Max. burnup (GWd/MTU)	427	427	427	568	568
Min. cool time (days)	120	120	120	120	120

Note: The enrichments, masses, and dimensions shall be based on values prior to irradiation.

5.(b) (2) Maximum quantity of material per package

Not to exceed 5,450 lbs, including fuel baskets, carrier racks, shoring, secondary containers, and shielding liner.

(i) For the contents described in 5(b)(1)(i):

600 watts decay heat; and

Fissile contents not to exceed 1175 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 45 GWd/MTU, and minimum cooling time of 120 days; or

Fissile contents not to exceed 1750 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.35 inch, maximum burnup of 38 GWd/MTU, and minimum cooling time of 120 days. Fuel rods must be contained in closed, 5-inch schedule 40 pipe, with a maximum of 437.5 grams U-235 equivalent per pipe; or

Fissile contents not to exceed 242 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 52 GWd/MTU, and minimum cooling time of 180 days.

(ii) For the contents described in 5(b)(1)(ii):

2000 watts decay heat. Fissile contents not to exceed 500 grams U-235 equivalent mass. Carrier racks specified in 5(a)(3)(iv) or 5(a)(3)(v) must be used for contents exceeding 600 watts decay heat per package.

(iii) For the contents described in 5(b)(1)(iii):

One HFIR fuel assembly. The fuel assembly is composed of one inner fuel element, with up to 2628 grams U-235, and one outer fuel element, with up to 6872 grams U-235. The maximum uranium enrichment is 93.2 weight percent U-235. The maximum burnup per assembly is 2300 MWd, the minimum cool time is two years. Decay heat not to exceed 600 watts per package.

5.(b) (2) Maximum quantity of material per package (Continued)

(iv) For the contents described in 5(b)(1)(iv):

A maximum of 4393 grams U-235 per package. The maximum uranium enrichment is 94.0 weight percent U-235. Decay heat not to exceed 35 watts per package. The TSR fuel elements must be positioned and limited within the TSR fuel basket as follows:

Lower fuel basket section - Up to 4 upper or lower fuel elements, or a combination of upper and lower fuel elements, for a total U-235 mass of 1412 grams.

Middle fuel basket section - Up to 4 fuel cover (lune) plates, for a total U-235 mass of 304 grams.

Upper fuel basket section - Up to 6 annular fuel elements plus one cylindrical fuel element, for a total U-235 mass of 2677 grams.

(v) For the contents described in 5(b)(1)(v):

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

Decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

Failed fuel elements are permitted provided the damage is limited to cladding defects due to corrosion, nicks, and scratches. Failed fuel elements must be structurally and geometrically intact.

(vi) For the contents described in 5(b)(1)(vi):

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

For stainless steel and inconel clad fuel, decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

For aluminum clad fuel, decay heat not to exceed either of the following: 630 watts per package, 30 watts per cell.

5.(c) Criticality Safety Index

For the contents described in 5(b)(1)(i), 5(b)(1)(ii) (except byproduct material), and 5(b)(1)(iii); and limited in 5(b)(2)(i), 5(b)(2)(ii), and 5(b)(2)(iii): 100

For the contents described in 5(b)(1)(iv), 5(b)(1)(v), 5(b)(1)(vi), and byproduct material from 5(b)(1)(ii); and limited in 5(b)(2)(iv), 5(b)(2)(v), 5(b)(2)(vi), and 5(b)(2)(ii): 0.0

6. Plutonium in excess of twenty curies per package must be in the form of metal, metal alloy or reactor fuel elements.

7. The U-235 equivalent mass is determined by U-235 mass plus 1.66 times U-233 mass plus 1.66 times Pu mass.

8. Bolt torque:

The cask lid bolts must be torqued to 690 ft-lbs (lubricated).

The bolts used to secure the top of the overpack to the overpack base must be torqued to 100 ft-lbs (dry).

9. (a) For any package containing organic or inorganic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(i) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with a radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

10. Prior to each shipment (except for contents meeting the requirements of special form radioactive material), the package must be leak tested to 1 x 10^{-3} std cm^3/sec. Prior to first use, after the third use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 1 x 10^{-7} std cm^3/sec.

11. The cask must be vacuum dried prior to shipment if contents are loaded under water, or if water is introduced into the cask cavity. During shipments for which vacuum drying is performed, the cask cavity must be filled with helium.

12. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) Prior to each shipment the cask seal must be inspected. The seal must be replaced with a new seal if inspection shows any defects or every 12 months, whichever occurs first; and

    (b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except that inspections in Section 8.2 of the application must be performed at least once within the 12-month period prior to each use; and

    (c) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.

13. Appropriate carrier racks or shoring must be provided to minimize movement of contents during accident conditions of transport.

14. Each batch of ethylene propylene seals must be tested in accordance with Section 8.1.4.2 of the application.

15. Fissile mass limits for reactor fuel are based on fissile mass prior to irradiation.

16. For the contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(v), and 5(b)(1)(vi), the package may be transported horizontally. For horizontal transport, the package must be secured to the truck bed with the top end of the package (closure end) facing the front (cab) of the truck. For horizontal transport of irradiated fuel and byproduct material contents described in 5(b)(1)(i) and 5(b)(1)(ii), the maximum decay heat is limited to 600 watts per package and the lead liner described in 5(a)(3)(ix) must be used.

17. Packagings may be marked with Package Identification Number USA/9228/B(U)F-85 until May 31, 2006, and must be marked with Package Identification Number USA/9228/B(U)F-96 after May 31, 2006.

18. Air transport of fissile material is not authorized.

19. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

20. Revision No. 24 of this certificate may be used until May 31, 2012.

21. Expiration date: May 31, 2016.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9228	25	71-9228	USA/9228/B(U)F-96	10	OF	10

REFERENCES

General Electric Company application dated December 12, 2000.

Supplements dated: December 20, 2000; March 16 and 27, 2001; March 22, 2002; and March 25, May 4, 5, and 23, July 28, 2005, January 25, 2006, January 19, 2007, and February 14, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

P.O.

Michael Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 4, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9233	10	71-9233	USA/9233/B(U)-96	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc
7135 Minstrel Way, Suite 300
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. application
dated March 8, 2005, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TN-RAM

(2) Description

The package is a steel encased lead shielded cask with wood impact limiters attached at both ends. The cask is a right circular cylinder. The overall dimensions of the packaging are approximately 178 inches long and 92 inches diameter with the impact limiters installed. The cask body is approximately 129 inches long with an outer diameter of 51 inches. The cask cavity has a length of approximately 111 inches and an inside diameter of 35 inches. The cask body is made of a 0.75-inch stainless steel inner shell, a 5.88-inch thick lead annulus, a 1.5-inch thick stainless steel outer shell, a 0.5-inch thick inner bottom plate and a 2.5-inch thick outside bottom plate. The lead shielding is approximately 6 inches thick in the bottom end of the cask. The outer shell of the cask body is covered with a stainless steel thermal shield. The closure lid consists of a 2.5-inch thick outer stainless steel plate and a 0.5-inch thick inner stainless steel plate separated by approximately 6 inches of lead shielding. An optional lid, with the lead shielding in the form of a separate shielding disk, can also be used. The lid is secured by sixteen 1.5-inch diameter closure bolts. Two concentric silicone O-rings are installed in grooves on the underside of the lid. The cask is equipped with a sealed leak test port between the O-rings, a vent port in the closure lid and a sealed drain port in the bottom of the cask. Each impact limiter is attached to the cask by eight 1.75-inch diameter bolts. The cask is equipped with 6 trunnions, four at the top and two at the bottom. The gross weight of the package is approximately 80,000 pounds, including maximum contents of 9,500 pounds.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9233	10	71-9233	USA/9233/B(U)-96	2	OF	3

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with Transnuclear, Inc. Drawing Nos. 990-701, Rev. 8; 990-702, Rev. 7; 990-703, Rev. 9; 990-704, Rev. 5; 990-705, Rev. 6; 990-706, Rev. 4; 990-707, Rev. 4; 990-708, Rev. 7; 990-709, Rev. 2; and 990-710, Rev. 1.

(b) Contents

(1) Type and Form of Material

Dry irradiated and contaminated non-fuel-bearing solid materials contained within a secondary container.

(2) Maximum quantity of material per package

Greater than Type A quantities of radioactive material which may include fissile material provided that the fissile material does not exceed the mass limits of 10 CFR 71.15. The contents may not exceed 1,272 times an A_2 quantity. The decay heat of the contents may not exceed 300 watts. The maximum gross weight of the contents, secondary container, and shoring is limited to 9,500 pounds.

6. As appropriate, shoring must be used in the secondary container sufficient to prevent significant movement of the contents under accident conditions.

7. Both the inner cask cavity and the secondary container must be free of water when the package is delivered to a carrier for transport.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Prior to each shipment, the lid seals must be inspected. The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first;

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application; and

(c) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration date: April 30, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9233	10	71-9233	USA/9233/B(U)-96	3	OF	3

REFERENCES

Transnuclear, Inc., application dated March 8, 2005.

Supplements dated: May 4, 2007; October 19, 2007; September 30, 2008; February 16, 2009 and March 15, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: April 14, 2010

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	12	71-9235	USA/9235/B(U)F-96	1	OF	15

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
NAC International, Inc., application dated February 19, 2009.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: NAC-STC

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has

5.(a)(2) Description (Continued)

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

The cask is closed by two steel lids which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominantly balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Connecticut Yankee-multi-purpose canister (MPC) configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC).

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks

5.(a)(2) Description (Continued)

and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

The Yankee Class MPC and Connecticut Yankee MPC TSC assemblies include a vessel shell, bottom plate, and welded shield and structural lids that are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC. The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter. The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister. The structural lid is a 3-inch thick steel plate. The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

The Yankee Class MPC TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick, 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The Connecticut Yankee MPC fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The LaCrosse boiling water reactor multi-purpose canister MPC-LACBWR TSC assembly consists of a vessel shell, a bottom plate and a welded closure lid/closure ring assembly that are fabricated from stainless steel. The MPC-LACBWR TSC bottom stainless steel thickness is 1.25 inches. The shell is ½-inch thick rolled steel plate and 70.6 inches in diameter. The closure lid is a 7.0-inch thick steel plate/forging. The closure lid redundant welded closure is provided by a closure ring. The closure lid is provided with vent and drain penetrations to access the TSC cavity and they are closed by redundant welded port cover

5.(a)(2) Description (Continued)

plates. The MPC-LACBWR TSC fuel basket is designed to hold up to 68 irradiated LACBWR fuel assemblies, including up to 32 damaged fuel assemblies contained in DFCs and up to 36 intact fuel assemblies.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14-inches high and 70-inches in diameter, and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-3, Rev. 15
423-802, sheets 1-7, Rev. 21
423-803, sheets 1-2, Rev. 9
423-804, sheets 1-3, Rev. 9
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-811, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

5.(a)(3) Drawings (Continued)

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 4
455-820, sheets 1-2, Rev. 3
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

[1]Drawing defines the alternate configuration that accommodates the Yankee-MPC damaged fuel can.

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060, Rev. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev. D4
YR-00-062, sheet 2, Rev. D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos.:

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 3
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets 1-2, Rev. 2
414-820, Rev.0
414-870, Rev.3
414-871, sheets 1-2, Rev.6
414-872, sheets 1-3, Rev.6
414-873, Rev.2
414-874, Rev.0
414-875, Rev.0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev.4

5.(a)(3) Drawings (Continued)

414-887, sheets1-4, Rev. 4
414-889, sheets 1-3, Rev. 7
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets 1-2, Rev. 3
414-894, Rev. 0
414-895, sheets 1-2, Rev. 4

(vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 1
414-902, sheets 1-3, Rev. 3
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

(viii) For the Dairyland Power Cooperative LaCrosse BWR transport package and TSC configuration, the TSC, fuel basket, and DFCs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

630045-800, sheets 1-2, Rev. 0
630045-870, Rev. 2
630045-872, sheets 1-2, Rev. 1
630045-877, Rev. 1
630045-881, sheets 1-2, Rev. 1
630045-894, Rev. 1
630045-901, Rev. 0
630045-820, Rev. 0
630045-871, sheets 1-4, Rev. 2
630045-873, Rev. 1
630045-878, Rev. 1
630045-893, Rev. 1
630045-895, sheets 1-3, Rev. 1
630045-902, sheets 1-2, Rev. 1

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264[1]
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:
[1] - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

Uranium Enrichment (wt% U-235)	Fuel Assembly Burnup (BU): BU ≤ 30 GWD/MTU				30 < BU ≤ 35 GWD/MTU				35 < BU ≤ 40 GWD/MTU				40 < BU ≤ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7≤E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9≤E<2.1	7	7	5	7	9	9	7	8	12	13	9	11	--	--	--	--
2.1≤E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3≤E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5≤E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7≤E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9≤E<3.1	6	5	5	5	7	7	5	6	9	8	6	8	11	11	8	10
3.1≤E<3.3	5	5	5	5	7	6	5	6	8	8	6	7	10	10	8	9
3.3≤E<3.5	5	5	5	5	6	6	5	6	8	7	6	7	10	10	7	9
3.5≤E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7≤E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9≤E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1≤E≤4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5(1)	--	--	--	6(1)	--	--	--	7(1)	--	--	--	9(1)
4.3≤E≤4.5	--	--	--	5(1)	--	--	--	6(1)	--	--	--	7(1)	--	--	--	8(1)

Notes:
(1) - Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950
Maximum Burnup (MWD/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

Notes:
1. Combustion Engineering (CE) fuel with a maximum burnup of 32,000 MWD/MTU, a minimum enrichment of 3.5 wt. percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.
2. Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.
3. Stated enrichments are nominal values (fabrication tolerances are not included).

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity, and does not exceed the mass limits of 10 CFR 71.15.

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397.1	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤ 1,500	≤ 1,500	≤ 1,500	≤ 1,600	≤ 1,600
Maximum Burnup (MWD/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:

1. Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.
2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.
3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.
4. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
6. Enrichment of the fuel within each DFC or RFA is limited to that of the basket configuration in which it is loaded.

5.(b)(1) Contents - Type and Form of Material (Continued)

(v) Irradiated undamaged and damaged Dairyland Power Cooperative LACBWR fuel assemblies based on design nominal or operating history record values listed below. Fuel assemblies may contain zirconium alloy shroud compaction debris.

Parameter	Units	Allis Chalmers	Exxon
Number of Assemblies per Canister[1]	---	32	68
Maximum Assembly Weight[6]	lbs	400	400
Assembly Length	in	103	103
Fuel Rod Cladding	---	Stainless Steel	Stainless Steel
Maximum Initial Uranium Mass[2]	kgU	121.4	111.9
Maximum Initial Enrichment	Wt% ^{235}U	3.64/3.94[5]	3.71[3]
Minimum Initial Enrichment	Wt% ^{235}U	3.6	3.6
Maximum Burnup	MWd/MTU	22,000	21,000
Maximum Assembly Decay Heat	W	63	62
Minimum Cool Time	yr	28	23
Assembly Array Configuration	---	10X10	10X10
Number of Fuel Rods	---	100	96
Maximum Active Fuel Length	in	83	83
Rod Pitch	in	0.565	0.557
Rod Diameter	in	0.396	0.394
Pellet Diameter	in	0.350	0.343
Clad Thickness	in	0.020	0.0220
Number of Inert Rods[4]	---	0	4
Inert Rod OD	in	N/A	0.3940

1. Maximum 68 assemblies per canister. Allis Chalmers fuel is restricted to Damaged Fuel Cans (DFCs). Therefore, Allis Chalmers fuel is limited to 32 assemblies per canister.
2. DFCs have been evaluated for 2% additional fuel rod mass.
3. Represents planar average enrichment.
4. Inert rods comprised of stainless steel clad tube containing zirconium alloy slug. Inert rods not required for fuel assemblies located in DFC.
5. Two Allis Chalmers fuel types: Type 1 at an enrichment of 3.64 wt% ^{235}U and Type 2 at 3.94 wt% ^{235}U.
6. Not including weight of DFC. DFCs may contain optional inner container subject to maximum weight and fissile material limits in this table.

5.(b)(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i): 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii): Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs. with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod. Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

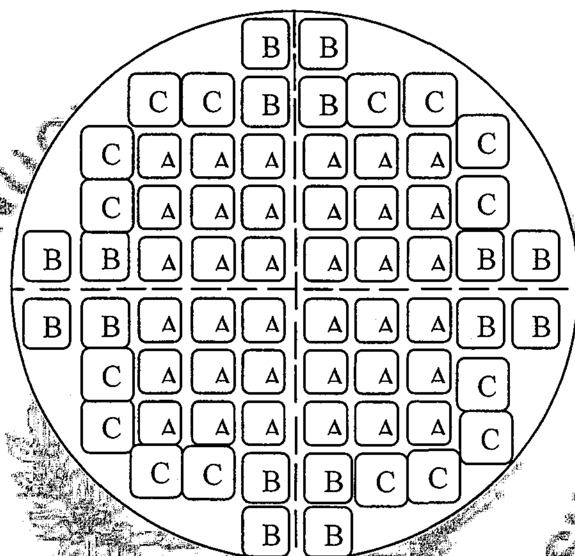
(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 12,340 lbs. with a maximum decay heat of 2.9 kW.

(v) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

5.(b)(2) Maximum quantity of material per package (Continued)

(vi) For the contents described in 5.(b)(1)(v): Up to 68 LACBWR assemblies, including up to 32 damaged fuel assemblies contained in DFCs, may be transported in the MPC-LACBWR TSCs.

Total weight of contents within the MPC-LACBWR TSC is 28,870 lbs., including the weight of 32 DFCs. The maximum decay heat is 4.5 kW per package. LACBWR undamaged fuel assemblies and LACBWR DFCs must be loaded in accordance with the following loading pattern:



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U, up to four slots maximum, B and C combined. Damaged Allis Chalmers fuel maximum enrichment 3.64 wt% ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U, up to four slots maximum, B and C combined. Damaged Allis Chalmers fuel maximum enrichment 3.94 wt% ^{235}U.

LACBWR DFCs are allowed to contain an additional 2% fissile material to account for loose pellets, not necessarily associated with the as-built fuel assembly.

NOTE: The above sketch is not to scale. It is a depiction of the loading pattern.

5.(c) Criticality Safety Index (CSI):

(1) CSI=0.0 for contents described in 5.(b)(1)(i), 5.(b)(1)(ii), 5.(b)(1)(iii), and 5.(b)(1)(iv) (i.e., Yankee Class and CY Fuel and GTCC Waste).

(2) CSI=100 for contents described in 5.(b)(1)(v) (i.e., LACBWR fuel).

6. Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii).

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii), and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0 X 10^{-8} cm^3/sec (helium) and shown to have a leak rate no greater than 8.0 X 10^{-8} cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least 1.0 X 10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0 X 10^{-7} cm^3/sec (helium).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report.

(c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	12	71-9235	USA/9235/B(U)F-96	15	OF	15

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

12. Transport by air is not authorized.

13. Packagings must be marked with Package Identification Number USA/9235/B(U)F-96.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15. Revision No. 11 of this certificate may be used until October 31, 2011.

16. Expiration date: May 31, 2014.

REFERENCES

NAC International, Inc., application dated: February 19, 2009.

As supplemented June 3 and December 17, 2009; February 3, April 28, June 3, August 19, and September 1, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Chris Staab

Chris Staab, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 5, 2010

NRC FORM 618 (8-2000) 10 CFR 71				U.S. NUCLEAR REGULATORY COMMISSION
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9239	18	71-9239	USA/9239/AF	1 OF 5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company, LLC
Columbia Fuel Site
P.O. Drawer R
Columbia, SC 29250

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company, LLC, application, Revision No. 13, dated October 2011, as supplemented.

CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: MCC-3, MCC-4, and MCC-5

(2) Description

The MCC packages are shipping containers for unirradiated uranium oxide fuel assemblies. The packagings consist of a steel fuel element cradle assembly equipped with a strongback and an adjustable fuel element clamping assembly. The cradle assembly is shock mounted to a 13-gauge carbon steel outer container by shear mounts. The MCC-3 container is closed with thirty ½-inch T-bolts. The MCC-4 and MCC-5 containers are closed with fifty ½-inch T-bolts.

The MCC-3 and MCC-4 containers are permanently equipped with vertical Gd_2O_3 neutron absorber plates that are mounted on the center wall of the strongback. Additional horizontal Gd_2O_3 neutron absorber plates, mounted on the underside of the strongback, are required for the contents as specified.

The MCC-5 container is permanently equipped with both the vertical and horizontal Gd_20_3 neutron absorber plates. Additional vee-shaped, guided Gd_20_3 neutron absorber plates are required for the contents as specified.

Approximate dimensions of the MCC-3 packaging are 44½ inches O.D. by 194½ inches long. The gross weight of the packaging and contents is 7,544 pounds. The maximum weight of the contents is 3,300 pounds.

5. (a) (2) Packaging (continued)

Approximate dimensions of the MCC-4 packaging are 44½ inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,870 pounds.

Approximate dimensions of the MCC-5 packaging are 44½ inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,700 pounds.

(3) Drawings

The MCC-3 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL301, Sheets 1, 2, 3, and 4, Rev. 6.

The MCC-4 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL401, Sheets 1, 2, 3, 4, and 5, Rev. 9.

The MCC-5 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL501, Sheets 1 through 10, Rev. 6.

(b) Contents

(1) Type and form of material

Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent with the following exceptions: 15x15 BW fuel assemblies have a maximum enrichment of 4.65 wt%, and VVER-1000 fuel assemblies have a maximum enrichment of 4.80 wt%.

5. (b) (1) Contents (continued)

The fuel assemblies shall meet the specifications given in Westinghouse Drawing No. 6481E15, Rev. 6, and in the following tables of Appendix 1-5 of the application:

Table 1-5.1, Rev. 13	Fuel Assembly Parameters 14x14 Type Fuel Assemblies†
Table 1-5.2, Rev. 13	Fuel Assembly Parameters 15x15 Type Fuel Assemblies‡
Table 1-5.3, Rev. 13	Fuel Assembly Parameters 16x16 Type Fuel Assemblies**
Table 1-5.4, Rev. 13	Fuel Assembly Parameters 17x17 Type Fuel Assemblies**
Table 1-5.5, Rev. 13	Fuel Assembly Parameters VVER-1000 Type Fuel Assembly***

** 16x16 CE fuel assemblies and the 17x17 W-STD/XL fuel assemblies shall be shipped only in the Model No. MCC-4 package.

*** VVER-1000 fuel assemblies shall be shipped only in the Model No. MCC-5 package.

† 14x14 Type fuel assemblies' annular pellet zone length is not restricted and may exceed 6-inches.

‡ 15x15 (Type B) OFA fuel assemblies may be modified by replacing seven fuel rods in locations O10 through O15 and N15 with solid stainless steel.

(2) Maximum quantity of material per package

Two (2) fuel assemblies

(c) Criticality Safety Index 0.4

6. (a) For shipments of 14x14, 15x15, 16x16, and 17x17 OFA fuel assemblies with U-235 enrichments of over 4.65 wt% and up to 5.0 wt%, horizontal Gd_2O_3 neutron absorber plates shall be positioned underneath each assembly. The horizontal absorber plates shall be placed horizontally on the underside of the strongback, as specified in the respective drawings in Condition 5(a)(3) for the MCC-3 and MCC-4 models.

6. (b) For shipments of 17x17 STANDARD lattice fuel assemblies (17x17 STD and 17x17 XL) with U-235 enrichments of over 4.85 wt% and up to 5.0 wt%, horizontal Gd_2O_3 neutron absorber plates shall be positioned underneath each assembly. The horizontal absorber plates shall be placed horizontally on the underside of the strongback, as specified in the respective drawings in Condition 5(a)(3) for the MCC-3 and MCC-4 models.

7. Shipments of VVER-1000 fuel assemblies are authorized with U-235 enrichments up to 4.80 wt%.

8. Each fuel assembly must be unsheathed or must be enclosed in an unsealed plastic sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent flow of liquids into or out of the sheathed fuel assembly.

9. The dimensions, minimum Gd_2O_3 loading and coating specifications, and acceptance testing of the neutron absorber plates shall be in accordance with the "Gd_2O_3 Neutron Absorber Plates Specifications," Appendix 1-7, Rev. 12, of the application, as supplemented. The minimum Gd_2O_3 coating areal density on the vertical and horizontal neutron absorber plates shall be 0.054 g-Gd_2O_3/cm^2. The minimum Gd_2O_3 coating areal density on guided neutron absorber plates shall be 0.027 g-Gd_2O_3/cm^2.

10. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) Each package shall be prepared for shipment and operated in accordance with the "Routine Shipping Container Utilization Summary Operating Procedures," in Chapter 7 of the application; and

    (b) Each package shall be tested and maintained in accordance with the "Acceptance Tests, Maintenance Program, and Recertification Program," in Chapter 8 of the application, and as specified in the respective drawings in Condition 5(a)(3) for the MCC-3, MCC-4, and MCC-5 models.

11. Transport by air of fissile material is not authorized.

12. Fabrication of new packagings is not authorized.

13. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

14. Revision No. 17 of this certificate may be used until April 30, 2014.

15. Expiration date: March 31, 2017.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9239	18	71-9239	USA/9239/AF	5	OF	5

REFERENCES

Westinghouse Electric Company, LLC, "Application For Approval of Packaging Of Fissile Radioactive Material (MCC Shipping Containers)", Revision No. 13, dated October 2011.

Supplement dated March 28, 2013, Revision No. 14.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 15, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9246	8	71-9246	USA/9246/AF	1	OF	2

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
National Institute of Standards and Technology
Gaithersburg, MD 20899

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
National Institute of Standards and Technology application dated October 17, 2011.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: ST

(2) Description

A closed steel pipe for the transport of an unirradiated research reactor fuel element. The pipe is a 5-1/2-inch OD carbon steel pipe, approximately 71 inches in length, with a closed bottom end and flanged top end. The top end is closed by a cover plate, which is 1/4-inch thick, and 6-1/2 inches in diameter, and a gasket. The cover plate is secured to the pipe flange by 8 cap screws. A wooden nozzle support, bottom support, and top support position the fuel element within the pipe. The package weighs approximately 75 pounds, including the fuel element.

(3) Drawing

The packaging is constructed and assembled in accordance with National Institute of Standards and Technology Drawing No. D-04-048, Sheet 1, Rev. 4, and Sheet 2, Rev. 4.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9246	8	71-9246	USA/9246/AF	2	OF	2

5. (b) Contents

(1) Type and form of material

Unirradiated NBSR fuel element composed of enriched uranium and aluminum.

(2) Maximum quantity of material per package

One fuel element containing not more than 360 grams U-235. The total quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Criticality Safety Index 50.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71, the package shall be prepared for shipment, operated, and maintained by written procedures prepared to meet the requirements and make the determinations specified in Chapter 7 of the package application. Additionally, the acceptance tests and maintenance program shall comply with Chapter 8 of the application.

7. Transport by air is not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Revision No. 7 of this certificate may be used until February 28, 2013.

10. Expiration date: January 31, 2017.

REFERENCES

National Institute of Standards and Technology application dated October 17, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine A. Lipa

Christine A. Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: February 29, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9248	22	71-9248	USA/9248/AF	1	OF	7

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
7135 Minstrel Way
Suite 300
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome ANP, Inc. application dated September 5, 2003, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: SP-1, SP-2, and SP-3

(2) Description

Fuel assembly and fuel rod shipping containers. The packages consist of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

The metal inner container is approximately 11-1/2 inches by 18 inches by 179-1/2 inches long and is positioned within a wooden outer container approximately 30 inches by 31 inches by 207 inches long. The SP-1 and SP-2 packagings differ in the length of the metal inner container and end piece. The SP-3 packagings have a reduced spacing between the fuel assembly channels and the outer surface of the metal inner container. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and ethafoam, or equivalent. Closure of the metal inner container and the wooden outer container is accomplished by bolts. A pressure relief (breather) valve is provided on the inner container, and is set for 0.5 psi differential. The maximum weight of the packaging and contents is 2,800 pounds.

5.(a) (3) Drawings

The packagings are fabricated and assembled in accordance with the following Framatome ANP, Inc., and Siemens Nuclear Power Corporation/Advanced Nuclear Fuels Corporation Drawing Nos.:

EMF-304,416, Rev. 14.
EMF-306,272, Rev. 10.
EMF-309,141, Rev. 1.

5.(a) (4) Product Containers

(i) Five-inch Schedule 40 stainless steel pipe fitted with screw type or flange closure. The product container shall be vented if it contains materials which decompose at less than 1475 °F.

(ii) Rod shipping container as shown on Siemens Power Corporation Drawing No. EMF-309,141, Rev. 1.

5.(b) Contents

(1) Type and form of material

(i) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel cross-section area of 25 in^2, maximum fuel length of 174 inches, and maximum average enrichment of 3.3 wt.% U-235. Minimum Zircaloy clad thickness is 0.025 inches; maximum pellet diameter is 0.555 inches. Any number of water rods in any arrangement is permitted.

(ii) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel length of 174 inches, and a maximum average enrichment between 3.3 to 4.0 wt.% U-235. The maximum pellet diameter is 0.555 inch, and the minimum clad thickness is 0.025 inch. Any number of water rods in any arrangement is permitted, including part length rods. Each assembly contains at least 4 rods with nominal 2 wt.% Gd_2O_3, which are in non-perimeter locations and are symmetric about the diagonal.

(iii) UO_2 fuel assemblies with a maximum U-235 enrichment of 5.0 wt.%, and a maximum average U-235 planar enrichment of 4.0 wt.%. Each fuel assembly is made up of fuel rods in a 10 x 10 square array, with a maximum fuel cross section area of 25.221 in^2, a nominal pitch of 0.511 inch, and a maximum fuel length of 174 inches. The maximum pellet diameter is 0.3356 inch, the minimum clad thickness is 0.0225 inch, and the maximum U-235 enrichment in any edge rod is 4.0 wt.%. Each assembly contains at least 6 rods with nominal 2 wt.% Gd_2O_3, which are symmetric about the diagonal, and each assembly contains at least 4 water rods in the 4 central rod positions.

5.(b) (1) Type and form of material (Continued)

(iv) UO_2 fuel rods with a maximum U-235 enrichment of 5.0 wt.%, and a minimum Gd_2O_3 content of 1.0 wt.%. The rods may be clad with Zircaloy, steel, or aluminum. The rods have a maximum fuel pellet diameter of 0.5 inch, and a maximum fuel length of 169 inches.

(v) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section area of 25.0 in^2, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 wt.%, the maximum U-235 enrichment for all edge rods is 4.0 wt.%, and the maximum average planar enrichment, excluding perimeter rods and rods containing gadolinia (Gd_2O_3), is 4.0 wt.% U-235. The maximum pellet diameter is 0.35 inch, and the minimum clad thickness is 0.018 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least twelve rods with a minimum nominal content of 2.0 wt.% gadolinia (Gd_2O_3), in a pattern symmetric about one of the assembly diagonals. At least eight of the twelve gadolinia rods must be located in rows 2 and 9, and in columns 2 and 9 of the assembly.

(vi) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 25.0 in^2, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 wt.%. The maximum pellet diameter is 0.35 inch, and the minimum clad thickness is 0.018 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 wt.% in all axial regions with enriched pellets. Additional gadolinia rod specifications are included in supplement dated April 30, 1996.

(vii) UO_2 fuel assemblies composed of fuel rods in a 9 x 9 square array, with a maximum fuel cross section of 25.0 in^2, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 wt.%. The maximum pellet diameter is 0.40 inch, and the minimum clad thickness is 0.015 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 wt.% in all axial regions with enriched pellets. Additional gadolinia rod specifications are included in supplement dated April 30, 1996.

5.(b) (1) Type and form of material (Continued)

(viii) UO_2 fuel assemblies composed of fuel rods in a 9 x 9 square array, with a maximum fuel cross-section area of 25.0 in^2, a maximum fuel length of 174 inches, and a maximum average uranium enrichment of 4.0 wt.% U-235. The nominal pellet diameter is 0.370 inch. At least the center 3 x 3 rod locations must be a water channel. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 wt. % in all axial regions with enriched pellets. The eight gadolinia rod locations are shown in Figure 1 of the supplement dated July 27, 1999.

(ix) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section area of 25.0 in^2, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 wt.%, the maximum U-235 enrichment for all edge rods is 4.75 wt.%, the maximum U-235 enrichment for the four (4) corner edge rods is 3.05 wt. %, and the maximum U-235 enrichment for the eight (8) edge rods immediately adjacent to the four corner edge rods is 3.55 wt.%. The pellet diameter is between 0.30 and 0.3957 inch. Each assembly must have a water channel in a central 3 x 3 position. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least 10 rods with a minimum nominal content of 2.0 wt.% gadolinia (Gd_2O_3) in all axial regions with the enriched pellets, and in a pattern symmetric about one of the assembly diagonals. At least 10 gadolinia rods must be located in rows 2 and 9, and in columns 2 and 9 of the assembly and cannot be immediately adjacent to another one of the 10 gadolinia rods; however, diagonally adjacent is permitted. An additional upper tie plate (UTP) shipping shim may be added between the UTP and the fueled region. This UTP shim may consist of a maximum of 345 g plastic or plastic composite.

(x) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section area of 25.0 in^2 and a maximum fuel length of 174 inches. The maximum uranium enrichment is 2.3 wt.% U-235. The pellet diameter is between 0.30 and 0.3957 inch. Each assembly must have a water channel in a central 3 x 3 position. Any number of additional water rods in any arrangement is permitted, including part length rods. An additional upper tie plate (UTP) shipping shim may be added between the UTP and the fueled region. This UTP shim may consist of a maximum of 345 grams of plastic or plastic composite.

5.(b) (2) Maximum quantity of material per package

Total weight of contents (fuel assemblies, or fuel rods and rod shipping containers) not to exceed 1265 pounds. Total quantity of radioactive material within a package may not exceed a Type A quantity.

(i) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), 5(b)(1)(viii), 5(b)(1)(ix), and 5(b)(1)(x):

Two full length fuel assemblies. Two short fuel assemblies may be substituted for each full length fuel assembly provided the two short assemblies are shipped end-to-end and the total fuel length does not exceed 174 inches.

(ii) For the contents described in 5(b)(1)(iv):

Two product containers specified in 5.(a)(4). Each product container may contain any number of loose fuel rods.

5.(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control:

(1) For contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(iv), and 5(b)(1)(viii), and limited in 5(b)(2)(i) and 5(b)(2)(ii): 0.4

(2) For contents described in 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), 5(b)(1)(ix), 5(b)(1)(x), and limited in 5(b)(2)(i): 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assembly.

7. Polyethylene shipping shims may be inserted between rods within fuel assemblies as follows:

(a) For contents described in 5(b)(1)(i) and 5(b)(1)(ii), up to a maximum of 0.20 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

7. Polyethylene shipping shims may be inserted between rods within fuel assemblies as follows (Cont.):

   (b) For contents described in 5(b)(1)(v), up to a maximum of 0.25 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

   (c) For contents described in 5(b)(1)(viii), up to a maximum volume fraction of 0.13 averaged over the void volume of the assembly.

   (d) For contents described in 5(b)(1)(iii), 5(b)(1)(vi), and 5(b)(1)(vii), polyethylene shipping shims are not permitted.

   (e) For contents described in 5(b)(1)(ix) and 5(b)(1)(x), up to a maximum volume fraction of 0.14 averaged over the void volume of the assembly.

8. Only contents described in 5(b)(1)(viii) and 5(b)(1)(ix) are authorized for transport in Model No. SP-3 packages.

9. Maximum average enrichment means the highest average enrichment through any cross sectional plane of the assembly.

10. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application dated September 5, 2003, as supplemented.

    (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application dated September 5, 2003, as supplemented.

11. The package authorized by this certificate is hereby authorized for use under the general license provisions of 10 CFR §71.17.

12. Transport by air of fissile material is not authorized.

13. Fabrication of new packagings is not authorized.

14. Revision 21 of this certificate may be used until April 30, 2014.

15. Expiration date: April 30, 2014.

REFERENCES

Framatome ANP, Inc., application dated September 5, 2003.

Supplements dated: September 24, November 6, 2003, June 28, 2012, March 26, 2013, and June 27, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 10, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9250	11	71-9250	USA/9250/B(U)F-85	1	OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Babcock and Wilcox
Nuclear Operations Group
P.O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Babcock and Wilcox Nuclear Operations Group application dated June 13, 2005.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: 5X22

(2) Description

A shipping container for unirradiated uranium of any enrichment. The outer packaging is a 16-gauge steel drum, approximately 22-1/2 inches in diameter and 34-3/4 inches high, with a heavy-duty clamp ring and forged lugs. The inner vessel (containment vessel) is a Schedule 40S stainless steel pipe with a welded bottom cap and a top weldneck flange. The inner vessel lid is a blind flange which is bolted to the weldneck flange with eight hex-head bolts. The closure includes double silicone O-ring seals and a leak-test port. The dimensions of the inner vessel are approximately 5 inches ID by 22 inches high. The inner vessel is centered within the outer drum by fiberboard and supported by plywood disks. The maximum weight of the package, including contents, is 300 pounds.

(3) Drawings

The packaging is constructed in accordance with Babcock and Wilcox, Drawing Nos. 1220276 E, Rev. 5, and 1220277 E, Rev. 9.

5.(b) Contents

Type and form of material, maximum quantity of material per package, and Criticality Safety Index.

The weight of the contents, including secondary containers, inserts, and other materials in the inner vessel, shall not exceed 50 pounds.

(1) Unirradiated uranium as solid compounds or alloys which do not decompose at temperatures up to 250 degrees Fahrenheit, and uranium oxides as powder or pellets. The uranium may be of any enrichment. Carbide compounds are not authorized. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

(2) Unirradiated solid uranyl nitrate in the form of uranyl nitrate dihydrate crystals, which may have small amounts of uranyl trihydrate crystals interspersed. The uranyl nitrate crystals shall have a uranium content that is from 52.5 to 56.0 percent by weight. The uranyl nitrate shall be packaged in Teflon primary containers that will not melt at temperatures up to 94 degrees Celsius. The uranium may be of any enrichment. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

5.(b) Contents (continued)

(3) Unirradiated uranium as solid metal. The uranium may be of any enrichment. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.5
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

(4) Unirradiated uranium as solid metal. The uranium may be of any enrichment. The packaging must include a solid aluminum disk insert positioned in the inner vessel, as shown on Babcock and Wilcox, Drawing No. 1220277 E, Rev. 9 (Part No. 6). The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0

(5) Unirradiated liquid uranyl nitrate solution in sealed glass containers or screw top plastic vials, each within one or more additional plastic vials with taped lids, and within a sealed product can or polyethylene bottle containing a sufficient amount of vermiculite to absorb twice the liquid contents present. The uranium may be of any enrichment. The quantity of uranyl nitrate shall not exceed 1000 mL of solution.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	N/A	0.4	0.4

6. The vent holes on the outer steel drum shall be capped or taped closed during transport and storage to preclude entry of rain water into the packaging.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

   (b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of10 CFR 71.17.

9. The package is subject to the provisions of 10 CFR 71.19(c), which requires that all fabrication of this packaging must have been completed by December 31, 2006.

10. Transport by air of fissile material is not authorized.

11. Expiration date: October 31, 2014.

REFERENCES

Babcock and Wilcox Nuclear Operations Group, application dated June 13, 2005

Supplements dated February 15, 2008; April 18, 2008; August 29, 2008; December 4, 2008; February 16, 2009; and July 30, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: September 11, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9251	14	71-9251	USA/9251/AF	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA NP, Inc.
2101 Horn Rapids Rd
Richland, WA 99354

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA NP, Inc., application dated
September 13, 2007.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: BW-2901

(2) Description

A shipping container for low-enriched uranium oxide powder and pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner cross sectional dimensions of the inner container are a maximum 11.15-inch square by 29.5-inch long. The inner container is constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum with 16-gauge head and DOT Specification 17H or an equivalent DOT UN1A2/Y1.5/100 closure by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum is approximately 22-1/2 inches in diameter and either 34-1/4 inches or 35-1/2 inches in overall height. The drum lid is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt. In addition to the locking ring, three lid clamps are installed to secure the drum lid. The uranium oxide is packaged in boxes, and wood boards position the boxes within the inner container. Three borated aluminum plates (approximately 25 inches by 9.25 inches by 0.375 inch) are positioned within the inner container. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed in accordance with B&W Fuel Company Drawing Nos. 1215597D, Rev. 5; 1215598B, Rev. 1; 1215599E, Rev. 5; and AREVA NP, Inc., Drawing No. 12155600, Rev. 7.

5. (b) Contents

(1) Type and form of material

(i) Sintered uranium oxide pellets enriched to a maximum 5.05 weight percent U-235. The minimum pellet diameter is 0.315 inch, and the maximum pellet diameter is 0.400 inch.

(ii) Uranium dioxide as powder, pellets, or any combination thereof, enriched to a maximum 5.05 weight percent U-235.

(2) Maximum quantity of material per package

370 pounds, with the U-235 content not to exceed 7.47 kg. The maximum weight of the uranium oxide, pellet boxes, and all packaging materials within the inner container is 427 pounds. Uranium oxide must be packaged in accordance with B&W Fuel Company Drawing Nos. 1215597D, Rev. 5, and AREVA NP, Inc., Drawing No. 1215600, Rev. 7. The maximum mass of polyethylene within the inner container shall not exceed 1000 grams per package. Maximum quantity of radioactive material within a package may not exceed a Type A quantity.

5. (c) Criticality Safety Index (CSI) 0.7

6. Each package must be shipped with borated aluminum plates positioned within the inner container, on the top of, between, and on the bottom of the rows of pellet boxes. The three borated plates must have dimensions and boron concentration, and must be positioned in accordance with B&W Fuel Company Drawing No. 1215597D, Rev. 5.

7. For packages with fewer than six pellet boxes, solid aluminum or wood pellet box spacers must be substituted for pellet boxes. The pellet boxes, pellet box spacers, borated plates, and wood boards must provide a snug axial and cross sectional fit in the inner container.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each packaging must be maintained and acceptance tested in accordance with Chapter 8 of the application;

(b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application; and

(c) Prior to each shipment the insert (containment vessel) gasket shall be inspected. The gasket shall be replaced if it is damaged, defective, or degraded.

9. Transport of fissile material by air is not authorized.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17, provided that fabrication of the package was satisfactorily completed by April 1, 1999.

11. Expiration date: January 31, 2018.

REFERENCES

AREVA NP, Inc., application dated September 13, 2007.

AREVA NP, Inc., supplement dated December 6, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material
Safety and Safeguards

Date: December 26, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9252	7	71-9252	USA/9252/AF	1	OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA NP, Inc.
3315 Old Forest Road,
P.O. Box 10935
Lynchburg, VA 24506-0935

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA NP, Inc., consolidated application dated October 28, 2008, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 51032-2

(2) Description

A steel shipping container for fuel bundles, consisting of a strong-back and fuel bundle clamping assembly, shock mounted to a steel outer container. Nine separator blocks, which are 6" x 8" x 8-1/2" long and have a 3/8" thick wall and a rectangular gusset plate welded inside, are bolted between fuel bundles. The outer container is composed of an 11 gauge steel shell approximately 43" diameter by 216" long. The maximum weight of the package, including contents, is 7,500 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with the following AREVA NP Inc. Drawing Nos.: 02-1215926C-002; 02-1215929D-003; 02-1215930D-003; 02-1215931D-003; 02-1215932D-003; 02-1215933D-003; 02-1215934C-002; 02-1215935D-003; 02-1216010D-001.

5.(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies, composed of uranium dioxide fuel pellets clad in zircaloy tubes. Uranium is enriched to a maximum of 5.0 weight percentage U-235. The fuel assemblies may contain inserted control rod assemblies. The fuel assemblies have the following specifications:

Type	15x15	15x15	17x17	17x17
Maximum Number of Fuel Rods Per Assembly	208	204	264	264
Minimum Number of Non-Fuel Rods Per Assembly	17	21	25	25
Nominal Rod Pitch (in.)	0.568	0.563	0.501	0.496
Maximum Pellet Diameter (in.)	0.3742	0.3671	0.3252	0.3232
Maximum Density of Active Fuel Stack Length (%TD)	97.5	97.5	97.5	97.5
Nominal Cladding Maximum OD (in.)	0.430	0.422	0.379	0.374
Nominal Cladding Minimum OD (in.)	0.377	0.370	0.332	0.326
Nominal Fuel Assembly Envelope (in.)*	8.520	8.445	8.517	8.432
Nominal Active Fuel Stack Length (in.)	144	144	144	144

* The nominal fuel assembly envelope is defined as the product of the nominal rod pitch and the number of rods per edge.

5.b(1)(continued)

Type	15x15	17x17	GEN1 14x14, 15x15 16x16	L1 15x15	L2 15x15	L4 17x17
Maximum Number of Fuel Rods Per Assembly	204	264	256	208	208	264
Minimum Number of Non-Fuel Rods Per Assembly	21	25	0	17	17	25
Nominal Rod Pitch (in.)	0.563	0.496	0.501-0.590	0.568	0.568	0.496
Maximum Pellet Diameter (in.)	0.384	0.334	0.454	0.3707	0.3742	0.3232
Maximum Density of Active Fuel Stack Length (%TD)	95.0	95.0	97.5	97.5	97.5	97.5
Nominal Cladding Maximum OD (in.)	0.430	0.380	0.500	0.430	0.430	0.374
Nominal Cladding Minimum OD (in.)	0.410	0.355	0.260	n/a	n/a	n/a
Nominal Fuel Assembly Envelope (in.)*	8.445	8.432	*	8.520	8.520	8.432
Nominal Active Fuel Stack Length (in.)	196	196	196	196	196	196

* The nominal fuel assembly envelope is defined as the product of the nominal rod pitch and the number of rods per edge.

5.(b)(continued)

(2) Maximum quantity of material per package

Two fuel assemblies. Total weight of fuel assemblies, including control rod assemblies, not to exceed 3300 pounds.

Maximum quantity of radioactive material within a package may not exceed a Type A quantity.

5. (c) Criticality Safety Index (CSI): 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed polyethylene sheath which will not extend beyond the ends of the fuel assemblies. The ends of the sheaths must not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assemblies.

7. Hydrogenous shims are not permitted within the fuel assemblies.

In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with Chapter 7.0 of the application.

(b) Each packaging shall be maintained in accordance with Section 8.2 of the application.

(c) Each packaging shall meet the acceptance tests in Section 8.1 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Transport by air of fissile material is not authorized.

11. Revision No. 5 of this certificate may be used until October 31, 2009.

12. Expiration date: October 31, 2013.

REFERENCES

AREVA NP Inc. consolidated application dated October 28, 2008.

Supplement dated: November 4, 2008; July 7, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: September 9 , 2009

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for the TN-FSV Package, dated March 31, 1993, as supplemented; Addendum A for the Oak Ridge Container in the TN-FSV Packaging, dated June 15, 2001, as supplemented; Addendum B for the PWR Fuel Rod Shielded Basket dated December 28, 2007, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TN-FSV

(2) Description

A steel and lead shielded shipping cask for irradiated nuclear fuel. The cask has three shipping configurations: Configuration 1 for shipping irradiated Fort St. Vrain high temperature gas cooled reactor (HTGR) fuel elements, Configuration 2 for shipping irradiated fuel parts and intact irradiated Peach Bottom Unit 1 fuel elements within a secondary containment vessel, and Configuration 3 for shipping irradiated Pressurized Water Reactor (PWR) fuel rods within a shielded basket. The cask is a right circular cylinder, with a balsa and redwood impact limiter at each end. The package has approximate dimensions and weights as follows:

Cavity diameter	18 inches
Cavity length	199 inches
Cask body outer diameter	31 inches
Lead shield thickness	3.44 inches
Package overall outer diameter, including impact limiters	78 inches
Package overall length, including impact limiters	247 inches
Packaging weight (Configuration 1)	42,000 pounds
Gross package weight, including contents (Configurations 1 and 2)	47,000 pounds

5.(a) (2) Description (Continued)

The cask body is made of two concentric shells of Type 304 stainless steel, welded to a bottom plate and a top closure flange. The inner shell has an ID of 18 inches and is 1.12 inches thick. The outer shell has an OD of approximately 30 inches and is 1.5 inches thick. The annular space between the inner and outer shells is filled with lead. The bottom plate is 5.5-inch thick Type 304 stainless steel. The closure lid is 2.5-inch thick Type 304 stainless steel, and is fully recessed into the cask top flange. The lid is fastened to the cask body by 12, 1-inch diameter closure bolts. The lid is sealed with double O-ring seals with a leak test port. A vent port and drain port are sealed with single O-rings and cover plates. Configuration 1 uses silicone O-ring seals and Configurations 2 and 3 use butyl O-ring seals. The cask body is covered with a stainless steel thermal shield composed of 0.25-inch thick stainless steel plate over a wire wrap. The impact limiters are constructed of balsa and redwood encased in stainless steel shells.

The cask has two lifting sockets bolted to the cask top flange. Two rear trunnions are provided for cask tie-down.

<u>For Configuration 1:</u>

Irradiated hexagonal HTGR fuel elements are shipped in Configuration 1. The fuel elements are stacked in a carbon steel fuel storage container which has an OD of approximately 17.6 inches and an overall length of 195 inches. The fuel storage container has a 0.5-inch thick shell, a 2.0-inch thick bottom plate, and a 1.5-inch thick lid. The lid accommodates a removable depleted uranium plug.

<u>For Configuration 2:</u>

Irradiated fuel parts and intact Peach Bottom Unit 1 fuel elements are shipped in Configuration 2. Canisters, containing either fuel parts or a single intact Peach Bottom fuel element, are loaded into a separate, secondary containment vessel, the Oak Ridge Container. The Oak Ridge Container is composed of a right circular cylindrical vessel and a basket assembly. The stainless steel vessel has a 10-gage (0.135-inch) wall thickness, an overall length of approximately 198 inches, and an outside diameter of approximately 20 inches at the lid end. The lid is approximately 7 inches thick and is closed by 12, 1/2-inch diameter bolts and two butyl O-ring seals. There is a single penetration through the lid which is closed by a bolted port cover and two butyl O-ring seals. The basket is composed of a series of discs, tie rods, and support tubes, with five fuel compartment tubes arranged in a star-like configuration. The basket incorporates fixed borated aluminum neutron poison plates. Flux trap spacers are positioned axially between stacked fuel parts canisters, and the canisters and spacers are positioned within a stainless steel sleeve that forms the fuel compartment. Canisters containing fuel parts (called Oak Ridge Canisters) and canisters containing intact Peach Bottom fuel elements may be shipped together.

5.(a) (2) Description (Continued)

For Configuration 3

Irradiated PWR fuel rods are shipped in Configuration 3. The fuel rods are loaded into a PWR fuel rod shielded basket. The basket has an overall length of 166 inches and an overall diameter of 17.5 inches, and fits closely within the TN-FSV cask cavity. The basket consists of a bottom end spacer, a cylindrical body with an inner diameter of 4 inches and an outer diameter of 10-1/2 inches with lateral support discs, and an 11-inch thick top lid. The basket is constructed of stainless steel. Up to 7 PWR fuel rods are loaded into individual stainless steel tubes within the basket.

(3) Drawings

The TN-FSV packaging is constructed and assembled in accordance with the following drawings.

Transnuclear, Inc., Drawing Nos.:

1090-SAR-1, Rev. 3
1090-SAR-2, Rev. 3
1090-SAR-6, Rev. 3
1090-SAR-7, Rev. 3
1090-SAR-8, Rev. 3
1090-SAR-9, Rev. 3
1090-SAR-10, Rev. 2

AREVA Federal Services LLC, Drawing Nos.:

1090-SAR-3, Rev. 4
1090-SAR-4, Rev. 5
1090-SAR-5, Rev. 5

The Oak Ridge Container and internals are constructed and assembled in accordance with the following Transnuclear, Inc. Drawing Nos.:

3044-70-1, Rev. 5
3044-70-2, Rev. 3
3044-70-3, Rev. 2
3044-70-4, Rev. 2
3044-70-5, Rev. 2
3044-70-6, Rev. 2
3044-70-7, Rev. 2
3044-70-8, Rev. 1
3044-70-9, Rev. 0

The Oak Ridge Canister is constructed and assembled in accordance with the following Lockheed Martin Energy Systems, Inc. Drawing No.:

X3E020566A175, Rev. 0

5.(a) (3) Drawings (Continued)

The PWR Fuel Rod Shielded Basket is constructed and assembled in accordance with the following AREVA Federal Services LLC, Drawing Nos.:

P03FM108-SAR, Sheets 1-3, Rev. 2

5.(b) Contents

(1) Type and form of material

(i) For Configuration 1:

Irradiated HTGR fuel elements within a fuel storage container. Each fuel element consists of a graphite block containing fuel rods. The fuel is composed of thorium/uranium carbide and thorium carbide fuel particles within the fuel rods. The graphite block is hexagonal in cross section and is approximately 14.2 inches across the flats and 31.2 inches long. Each fuel element contains a maximum of 1.4 kg of uranium enriched to a maximum of 93.5 weight percent U-235 and approximately 11.3 kg of thorium. The maximum burnup is approximately 70,000 MWd/MTIHM, and the minimum cool time is 1600 days.

(ii) For Configuration 2:

Irradiated intact Peach Bottom Unit 1, Core 2, fuel elements within aluminum canisters with steel liners. Each fuel element consists of stacked graphite annular rings, or compacts, with an inner diameter of approximately 1.75 inches and an outer diameter of approximately 2.75 inches. The fuel is composed of coated thorium/uranium carbide particles within the graphite. The active fuel length is approximately 90 inches. The fuel element may include associated hardware such as top plug/reflector apparatus, grappling hook, etc. Each fuel element contains a maximum of 0.25 kg of uranium enriched to a maximum of 93.15 weight percent U-235 and approximately 1.5 kg of thorium prior to irradiation. The maximum burnup is approximately 73,000 MWd/MTIHM and the minimum cool time is 27 years.

5.(b) (1) (iii) For Configuration 2:

Irradiated fuel parts within Oak Ridge Canisters, as described in Item No. 5(a)(3), above. The minimum fuel cool time is 15 years. The maximum fissile mass prior to irradiation per Oak Ridge Canister is limited as shown below:

Canister Group	Maximum mass U-235 per canister (grams)	Maximum mass Pu-239 + Pu-241 per canister (grams)
1	475	0
2	865	191
3	200	415
4	275	160
5	910	0

(iv) For Configuration 3:

Undamaged irradiated PWR fuel rods. The fuel rods are composed of uranium oxide pellets within zirconium-alloy cladding. The maximum uranium enrichment is 5.0 weight percent U-235. The maximum fuel rod length is 156.6 inches. For rods categorized as high uranium loading, the maximum uranium mass is 2.36 kgU/rod. For rods categorized as medium uranium loading, the maximum uranium mass is 1.78 kgU/rod. Known or suspected damaged fuel rods, and fuel rods with cladding defects greater than pin holes and hairline cracks are not authorized. Irradiated guide tubes may be substituted for fuel rods.

(2) Maximum quantity of material per package

Total weight of contents and packaging material within the TN-FSV cavity not to exceed 5,000 pounds. For Configuration 1 this includes fuel elements, fuel storage container, and depleted uranium shield plug. For Configuration 2 this includes fuel materials, Oak Ridge Container, basket, Oak Ridge Canisters, Peach Bottom fuel canisters, flux trap spacers, and other packaging materials. For Configuration 3 this includes fuel rods and PWR fuel rod shielded basket.

(i) For the contents described in Item 5(b)(1)(i):

Six fuel elements, with decay heat not to exceed 60 watts per fuel element.

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5.(b) (2) Maximum quantity of material per package (Continued)

(ii) For the contents described in Item 5(b)(1)(ii) and 5(b)(1)(iii):

Total weight of fuel materials, canisters, and flux trap spacers within the Oak Ridge Container not to exceed 1,789 pounds. Decay heat not to exceed 120 watts per package. The maximum decay heat per Oak Ridge Canister is 35 watts, except that the maximum decay heat per Oak Ridge Canister in the position next to the lid is 7 watts. The maximum decay heat in any cross sectional region corresponding to the axial length of an Oak Ridge Canister is 55 watts, except that the maximum decay heat in the cross sectional region next to the lid is 35 watts.

(ii) For the contents described in Item 5(b)(1)(ii) and 5(b)(1)(iii): (continued)

Canisters containing intact Peach Bottom fuel elements and Oak Ridge Canisters containing irradiated fuel parts must be loaded into the Oak Ridge Container fuel compartments as follows:

Loading Pattern	One Fuel Compartment	Other Four Fuel Compartments
1	Four Group 2 Canisters	Four Group 1 Canisters
2	Four Group 5 Canisters	Four Group 1 Canisters
3	One Peach Bottom Element and One Group 4 Canister	One Peach Bottom Element and One Group 4 Canister
4	Two Group 3 Canisters and Two Group 4 Canisters	One Peach Bottom Element and One Group 4 Canister

Flux trap spacers, as shown in Transnuclear, Inc. Drawing No. 3044-70-3, must be positioned axially between any two Oak Ridge Canisters.

5.(b) (2) (iii) For the contents described in Item 5(b)(1)(iv):

Total decay heat not to exceed 360 watts per package. The maximum number of rods per package is determined based on the maximum burnup and minimum decay time of any single (most limiting) rod in the package. The maximum number of rods per package is as follows:

Rod Type	Maximum Burnup (GWD/MTU)	Minimum Decay Time (days)	Maximum Number of Rods Per Package
High Uranium Loading (Max. 2.36 kgU/rod)	60	265	7
		210	6
		180	5
	72	450	7
		365	6
	80	1500	7
		900	6
		450	5
		365	4
Medium Uranium Loading (Max. 1.78 kgU/rod)	60	180	7
	72	365	7
	80	500	7
		365	6
Zirconium-alloy guide tubes	NA	180	7

5.(c) Criticality Safety Index (CSI) 100

6. The package must be leak tested as follows:

(a) <u>For Configuration 1</u>:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal must be tested to show a leak rate no greater than 1×10^{-3} ref-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-4} ref-cm^3/sec.

(2) Prior to each shipment, the package seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of the Safety Analysis Report. The acceptance criterion is a leak rate no greater than 1×10^{-3} ref-cm^3/sec. The test must have a sensitivity of at least 1×10^{-3} ref-cm^3/sec. The drain seal must also be tested if the drain port cover has been removed since the seal was last leak tested.

6. (b) For Configuration 2:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal of the outer cask and the Oak Ridge Container must be tested to show a leak rate no greater than 1 x 10^{-7} ref-cm^3/sec. The leak test must have a sensitivity of at least 5 x 10^{-8} ref-cm^3/sec.

(2) Prior to each shipment, the Oak Ridge Container containment seals (main seal and vent seal) and the outer cask containment seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of Addendum A. The seals must show no leakage greater than 1 x 10^{-7} ref-cm^3/sec or no leakage when tested to a sensitivity of at least 1 x 10^{-3} ref-cm^3/sec. The drain seal of the outer cask must also be tested if the drain port cover has been removed since the seal was last leak tested.

(c) For Configuration 3:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal of the outer cask must be tested to show a leak rate no greater than 1 x 10^{-7} ref-cm^3/sec. The leak test must have a sensitivity of at least 5 x 10^{-8} ref-cm^3/sec.

(2) Prior to each shipment, the outer cask containment seals (main seal and vent seal) must be leak tested in accordance with Section B7.1.2 of Addendum B. The seals must show no leakage greater than 1 x 10^{-7} ref-cm^3/sec or no leakage when tested to a sensitivity of at least 1 x 10^{-3} ref-cm^3/sec. The drain seal of the outer cask must also be tested if the drain port cover has been removed since the seal was last leak tested.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with Chapter 7 of the Safety Analysis Report for Configuration 1; Chapter 7 of Addendum A for Configuration 2; and Chapter B7 of Addendum B for Configuration 3.

(b) Each packaging must meet the acceptance tests and must be maintained in accordance with Chapter 8 of the Safety Analysis Report for Configuration 1; Chapter 8 of Addendum A for Configuration 2; and Chapter B8 of Addendum B for Configuration 3.

(c) Prior to each shipment for Configurations 1, 2, and 3, the cask main closure seal and vent seal must be inspected. The drain seal must be inspected if the drain port cover has been removed during preparation for shipment. In addition, prior to each shipment for Configuration 2, the Oak Ridge Container main closure seal and vent seal must be inspected. For Configurations 1, 2, and 3, all seals must be replaced within the 12-month period prior to shipment, or earlier if inspection shows any defect.

8. Transport of fissile material by air is not authorized.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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9. Fabrication of additional impact limiters (balsa and redwood encased in stainless steel shells) is not authorized.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Packagings may be marked with Package Identification Number USA/9253/B(U)F-85 until June 30, 2010, and must be marked with Package Identification Number USA/9253/B(U)F-96 after June 30, 2010.

12. Revision No. 10 of this certificate may be used until June 30, 2010.

13. Expiration date: June 30, 2014.

REFERENCES

Public Service Company of Colorado application dated March 31, 1993; as supplemented February 24, June 2, and June 14, 1994; and September 11 and December 7, 1995.

U.S. Department of Energy supplements dated: March 24, 1997; March 24, 1999; June 15, September 18, October 2, 2001; April 22, 2004; December 28, 2007; April 23, and June 11, 2009; and July 13, 2009.

Transnuclear, Inc., supplements dated September 19, 2001; and March 1, May 17, June 14 and 21, 2002; June 3, and July 21, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: July 14, 2009

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. consolidated application dated August 4, 2003, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NUHOMS® MP187 Multi-Purpose Cask

(2) Description

The NUHOMS® MP187 Multi-Purpose Cask (package) consists of an outer cask, into which one of the four different dry shielded canisters (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection.

Cask

The purpose of the cask is to provide containment and shielding of the radioactive materials contained within the DSC during shipment. The cask is constructed of stainless steel and lead with a neutron shield of cementitious material. The inside cavity of the cask is a nominal 68 inches in diameter and 187 inches long. The bottom access closure is approximately 5 inches thick and 17 inches in diameter, secured by 12 1-inch diameter bolts. The top closure is approximately 6.5 inches thick and is secured by 36 2-inch diameter bolts. Both closures are sealed by redundant O-rings.

Containment is provided by a stainless steel closure lid bolted to the stainless steel cask. The containment system of the NUHOMS® MP187 transportation cask consists of (a) the inner shell, (b) the bottom end closure plate, (c) the top closure plate, (d) the top closure inner O-ring seal, (e) the ram closure plate, (f) the ram closure inner O-ring seal, (g) the vent port screw, (h) the vent port O-ring seal, (i) the drain port screw, and (j) the drain port O-ring seal. No credit is given to the DSC as a containment boundary.

Shielding is provided by 4 inches of stainless steel, 4 inches of lead, and approximately 4.3 inches of neutron shielding. The overall length of the cask is approximately 200 inches; the outer diameter is approximately 93 inches. The maximum gross weight of the package, with impact limiters, is approximately 282,000 lbs. The total length of the package with the impact limiters attached is approximately 308 inches. Four removable trunnions (two upper and two lower) are provided for handling and lifting.

Dry Shielded Canisters (DSCs)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The approximately 5/8-inch thick shell has an outside diameter of about 67 inches and an external length of about 186 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. The basket is composed of circular spacer discs machined from thick carbon steel plates. Axial support for the DSC basket is provided by four high strength steel support rod assemblies. Carbon steel components of each DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion.

On the bottom of each DSC is a grapple ring, which is used to transfer a DSC horizontally from the cask into and out of dry storage modules. Because of the nature of the fuel that is to be transported, four different types of DSCs are designed for the package. Variations in the DSC configurations are summarized below:

- Fuel-Only Dry Shielded Canisters (FO-DSC)

  The FO-DSC has a cavity length of approximately 167 inches and has solid carbon steel shield plugs at each end. The FO-DSC is designed to contain up to 24 intact Babcock and Wilcox (B&W) pressurized water reactor (PWR) spent fuel assemblies. The FO-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies.

- Fuel/Control Components Dry Shielded Canister (FC-DSC)

  The FC-DSC has an internal cavity length of approximately 173 inches to accommodate fuel with the B&W control components installed. To obtain the increased cavity length, the shield plugs are fabricated from a composite of lead and steel. The FC basket is similar to the FO-DSC except that the support rod assemblies and guide sleeves are approximately 6-inches longer. The FC-DSC is also designed to contain up to 24 intact B&W PWR spent fuel assemblies with control components.

- Failed Fuel Dry Shielded Canister (FF-DSC)

  The FF-DSC has an internal cavity length of approximately 173 inches to accommodate 13 damaged B&W PWR spent fuel assemblies. Because the cladding has been locally degraded, individual (screened) fuel cans are provided to confine any gross loose material, maintain the geometry for criticality control, and facilitate loading and unloading operations. The FF-DSC is similar to FC-DSC in most respects with the exception of the basket assembly. The FF-DSC basket may be fabricated from austenitic stainless steel.

- 24PT1 Dry Shielded Canister (24PT1-DSC)

  The 24PT1-DSC has an internal cavity length of approximately 167 inches with a solid carbon steel shield plug at each end. The 24PT1-DSC will accommodate 22 to 24 Westinghouse (WE) 14 x14 PWR spent fuel assemblies, including control components. Control components authorized that are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only. Fuel assemblies may be damaged or intact as described in 5.b(2)(a). The 24PT1-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies. Up to four screened individual failed fuel cans are provided for storage of damaged fuel within the guide sleeve assemblies. These failed fuel cans are similar in configuration to the FF-DSC failed fuel cans.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell are closed-cell polyurethane foam and aluminum honeycomb material. The impact limiter is attached to the cask by carbon steel bolts. Each impact limiter is bolted to the cask body through the neutron shield top and bottom support rings. The weight of each impact limiter is approximately 15,800 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear West Drawing Numbers:

NUH-05-4000NP, Revision 9,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 15,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 5
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 10,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 16,
Sheets 1 through 5
NUHOMS® FO-DSC & FC-DSC
PWR Fuel Main Assembly

NUH-05-4005, Revision 14,
Sheets 1 through 5
NUHOMS® FF-DSC
PWR Fuel Main Assembly

NUH-05-4006NP, Revision 7,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

NUH-05-4010, Revision 2,
Sheets 1 through 6
NUHOMS® - 24PT1-DSC
Main Assembly

(b) Contents of Packaging

(1) Type and Form of Material

(a) Intact fuel assemblies Intact fuel assemblies - Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the FO-DSC, FC-DSC, or 24PT1-DSC.

(b) Damaged fuel assemblies - Assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding are authorized when contained in a failed fuel can in the FF-DSC or the 24PT1-DSC. Spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies are not authorized. Damaged fuel assemblies may be shipped with or without control components.

(c) (i) The fuel authorized for shipment in the NUHOMS®-MP187 FO, FC, or FF DSC is B&W 15x15 uranium oxide PWR fuel assemblies with a maximum initial pellet enrichment of 3.43% by weight of U235, and a total uranium content not to exceed 466 Kg per assembly.

(ii) The fuel authorized for shipment in the NUHOMS®-MP187 24PT1-DSC is WE 14x14 stainless steel clad (SC) or zircaloy clad mixed oxide (MOX) PWR fuel assemblies as described in Table 2.

(d) Intact B&W 15x15 fuel assemblies without control components shall be shipped only in the FO-DSC. Intact B&W 15x15 fuel assemblies with control components shall be shipped only in the FC-DSC.

(e) Intact WE 14x14 fuel assemblies with or without control components shall be shipped only in the 24PT1-DSC. Control components authorized are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only.

(f) (i) The maximum burn-up and minimum cooling times for the individual B&W 15x15 assemblies shall meet the requirements of Table 1. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(g) and (h). The maximum total allowable cask heat load is 13.5 kW.

(ii) The maximum enrichment, burn-up and minimum cooling times for the individual WE 14x14 fuel assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b.(1)(g) and (h). The maximum total allowable cask heat load for the 24 PT1-DSC is per Table 2.

(g) (i) The maximum assembly decay heat (including control components when present) of B&W 15x15 individual fuel assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

(ii) The maximum assembly decay heat (including control components when present) of WE 14x14 individual fuel assembly is per Table 2.

(h) (i) Control components for B&W 15x15 fuel assemblies stored in the FO, FC and FF-DSCs shall be cooled for at least 8 years.

(ii) Control components for WE 14x14 fuel assemblies stored in the 24PT1-DSC shall be cooled for at least 10 years.

(2) Maximum quantity of material per package

(a) (i) For material described in 5.b(1) to be stored in the FO, FC or FF-DSCs: 24 PWR intact fuel assemblies or 13 damaged fuel assemblies, with no more than 15 damaged fuel rods per assembly. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(ii) For material described in 5.b(1) to be stored in the 24PT1-DSC: 22 to 24 PWR fuel assemblies of which up to four may be damaged WE 14x14 SC fuel assemblies with the balance intact WE 14x14 SC or MOX fuel assemblies. No more than one damaged WE 14x14 MOX fuel assembly can be stored per 24PT1-DSC with the balance intact WE 14x14 SC fuel assemblies. The damaged fuel assemblies shall have no more than 14 damaged fuel rods per assembly and shall be stored in the four outer corner fuel assembly locations along the 45°, 135°, 225°, 315°azimuth of the 24PT1-DSC. A DSC may include two empty slots if they are located on symmetrically opposite locations with respect to the 0° - 180° and 90°-270° DSC axes. Any additional empty fuel slots shall be loaded with dummy fuel assemblies that displace the same or greater amount of volume and with the same nominal weight as a standard fuel assembly. Fuel spacers shall be located at the bottom and top of each fuel assembly to center the fuel assemblies within the DSC. Failed fuel cans require only bottom spacers since a top spacer is integral to each failed fuel can.

(b) For material described in 5.b(1): the approximate maximum payload (including control components when present) is 81,100 lbs.

Table 1- FO, FC and FF-DSC Fuel Assembly Burn-up vs. Cooling Time

Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)	Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)
<23,200	n/a	5	5	33,000	2.90	7	10
23,200	2.38	5	5	34,000	2.95	7	11
24,000	2.43	5	6	35,000	2.67	7	14
25,000	2.49	5	6	35,000	2.99	7	11
26,000	2.55	5	7	36,000	3.03	8	13
27,000	2.61	5	7	37,000	3.00	8	14
28,000	2.66	5	8	37,000	3.07	8	14
29,000	2.00	6	10	38,000	3.11	9	15
29,000	2.71	5	8	39,000	3.15	9	16
30,000	2.76	5	8	40,000	3.19	9	17
31,000	2.81	6	9				
32,000	2.86	6	10	* Megawatt Days per Metric Ton of Initial Heavy Metal			

Table 2 - 24PT1-DSC Fuel Assembly Burnup vs. Cooling Time

Fuel Type	Maximum Enrichment (Weight %)	Minimum Enrichment (Weight %)	Maximum Burnup (MWD/ MTU)	Minimum Cooling Time / Max Heat Load Per Cask / Max Assembly Heat Load (Incl. Control Components[1])
WE 14x14 Stainless Steel Clad (SC) (May include Integral Fuel Burnable Absorber, boron coated fuel pellets)	4.05 ^{235}U	3.76 ^{235}U	45,000	38 years/14 kW/ 0.583 kW
		3.36 ^{235}U	40,000	
		3.12 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U 2.84 fissile Pu (64 rods) 3.10 fissile Pu (92 rods) 3.31 fissile Pu (24 rods)	2.78 fissile Pu (64 rods) 3.05 fissile Pu (92 rods) 3.25 fissile Pu (24 rods)	25,000	30 years/13.706 kW/ 0.294 kW

Notes:
1 Control component cooling time must be a minimum of 10 years.

(c) Criticality Safety Index 0

6. Type I fuel assemblies shall be loaded only into the four innermost cells of a DSC, while Type II assemblies may be loaded into any cell when using the FO-DSC or the FC-DSC. The FF-DSC has no Type I or II placement restrictions. The 24PT1-DSC has restrictions on the location of damaged fuel assemblies per Section 5.b.(2).

7. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) All fabrication acceptance tests and maintenance shall be performed in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 , as supplemented. In addition, this shall include:

(1) With the exception of the weld between the inner shell and top forging, all longitudinal and circumferential inner shell welds, which form the containment boundary of the cask, shall be radiographically inspected (RT) with acceptance standards in accordance with the ASME Code, Section III, Division 1, NB-5320. The weld between the inner shell and top forging shall be verified by RT or ultrasonically inspected (UT). The substitution of UT for the examination of the completed weld may be made provided the examination is performed using detailed written procedures, proven by actual demonstration to the satisfaction of the inspector as capable of detecting and locating defects described in ASME Code, Section III, Division 1 Subsection NB.

(2) Verification of the DSC outer top cover plate weld by either volumetric or multilayer PT examination. If PT is used, at a minimum, it must include the root, each successive 1/4 inch weld thickness, and the final layer. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PVC Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(3) The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches.

(4) The neutron shield shall have a minimum thickness of 4.31 inches.

8. This package is approved for exclusive use by rail, truck, or marine transport. Transport by air of fissile material is not authorized.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Fabrication of new packagings is not authorized.

11. Expiration Date: November 30, 2018.

REFERENCES

Transnuclear, Inc. application dated August 4, 2003.

Supplement(s) dated September 16, 2008; July 26, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele M. Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 9, 2013.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Holtec International
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report on the HI-STAR 100 Cask System, Revision No. 15, dated October 11, 2010.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below. The HI-STAR 100 System includes the HI-STAR 100 Version HB (also referred to as the HI-STAR HB).

Multi-Purpose Canister

There are seven Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32,MPC-68, MPC-68F, and the MPC-HB. All MPCs are designed to have identical exterior dimensions, except 1) MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design; and 2) MPC-HBs custom-designed for the Humboldt Bay plant, which are approximately 6.3 feet

5.(a)(2) Description (continued)

shorter than the generic Holtec MPC designs. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to 32 intact PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. The MPC-HB is designed to contain up to 80 Humboldt Bay BWR fuel assemblies.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. The outer diameter of the Humboldt Bay MPCs is the same as the generic MPC, but the height is approximately 6.3 feet shorter than the generic MPC design. The Humboldt Bay MPCs are transported in a shorter version of the HI-STAR overpack, designated as the HI-STAR HB. The fuel basket designs vary based on the MPC model.

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report on the HI-STAR 100 Cask System*, Revision No. 15:

5.(a)(3) Drawings (continued)

(a) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 10
(b) MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. 25
(c) MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 9
(d) MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev. 11
(e) MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 14
(f) HI-STAR 100 Impact Limiter	Drawing C1765, Sheet 1, Rev. 6; Sheet 2, Rev. 4; Sheet 3, Rev. 5, Sheet 4, Rev. 5; Sheet 5, Rev. 2; Sheet 6, Rev. 5; and Sheet 7, Rev. 1.
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 2
(h) Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i) Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j) Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k) Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 7
(l) MPC-32 Fuel Basket Assembly	Drawing 3927, Sheets 1-4, Rev. 16
(m) HI-STAR HB Overpack	Drawing 4082, Sheets 1-7, Rev. 7
(n) MPC-HB Enclosure Vessel	Drawing 4102, Sheets 1-4, Rev. 1
(o) MPC-HB Fuel Basket	Drawing 4103, Sheets 1-3, Rev. 6
(p) Damaged Fuel Container HB	Drawing 4113, Sheets 1-2, Rev. 2

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(i) below are authorized for transportation.

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5.(b)(1) Type, Form, and Quantity of Material (continued)

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.

Damaged Fuel Containers (or Canisters) (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates.

The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10, 1.2.11, and 1.I.1 of the HI-STAR 100 System Safety Analysis Report, Rev. 15.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage, including containers and structures supporting these parts. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s). Trojan fuel assemblies not loaded into DFCs or FFCs are classified as intact assemblies.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TPDs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

5.(b)(1)(b) Definitions (continued)

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System Safety Analysis Report, Rev. 15.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System Safety Analysis Report, Rev. 15.

Undamaged Fuel Assemblies are fuel assemblies where all the exterior rods in the assembly are visually inspected and shown to be intact. The interior rods of the assembly are in place; however, the cladding of these rods is of unknown condition. This definition only applies to Humboldt Bay fuel assembly array/class 6x6D and 7x7C.

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more

5.(b)(1) Type, Form, and Quantity of Material (continued)

restrictive of the decay heat limits for the damaged fuel assemblies or the intact fuel assemblies.

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR, and the values shall be compared against the limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC-32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.2 of the application.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the provisions provided in Chapter 7 of the application.

(b) All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for acceptance testing and maintenance shall be developed and shall include the provisions provided in Chapter 8 of the application.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds, except for the HI-STAR HB, where the gross weight shall not exceed 187,200 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

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9. The personnel barrier shall be installed at all times while transporting a loaded overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Transport by air of fissile material is not authorized.

12. Revision No.7 of this certificate may be used until October 31, 2011.

13. Expiration Date: March 31, 2014

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report on the HI-STAR 100 Cask System*, Revision 15, dated October 11, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 12, 2010

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 8

MODEL NO. HI-STAR 100 SYSTEM

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Page:	Table:	Description:
Page A-1 to A-23	Table A.1	Fuel Assembly Limits
Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3 with or without Zircaloy channels.
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6X6C, 7x7A, or 8x8A.
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-9		MPC-68F: Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.

A-10	Table A. 1 (Cont'd)	MPC-68F: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-11		MPC-68F: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-12		MPC-68F: Mixed Oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-13		MPC-68F: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters.
A-15		MPC-24E: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-16		MPC-24E: Trojan plant damaged fuel assemblies.
A-17		MPC-24EF: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-18		MPC-24EF: Trojan plant damaged fuel assemblies.
A-19		MPC-24EF: Trojan plant Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris.
A-20 to A-21		MPC-32: Uranium oxide, PWR intact fuel assemblies in array classes 15X15D, E, F, and H and 17X17A, B, and C as listed in Table A.2.
A-22 to A-23		MPC-HB: Uranium oxide, intact and/or undamaged fuel assemblies and damaged fuel assemblies, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C.
A-24 to A-27	Table A.2	PWR Fuel Assembly Characteristics
A-28 to A-33	Table A.3	BWR Fuel Assembly Characteristics
A-34	Table A.4	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-24/24E/24EF PWR Fuel with Zircaloy Clad and with Non-Zircaloy In-Core Grid Spacers.

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Table A.1 (Page 1 of 23)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (Page 2 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, except assembly classes 6x6D and 7x7C, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, and a minimum initial enrichment $\geq$ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge $\geq$ 16 years, an average burnup $\leq$ 22,500 MWD/MTU, and a minimum initial enrichment $\geq$ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	≤272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat ≤183.5 Watts.
a. SS Clad:	≤83 Watts
f. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
g. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
h. Fuel assembly weight:	$\leq$ 700 lbs, including channels

Table A.1 (Page 3 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 4 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 5 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers.

Table A.1 (Page 6 of 23)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System application, Revision 15) and meeting the following specifications:

a. Cladding type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table A.1 (Page 7 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 8 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 9 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 10 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 11 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 12 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 13 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System application, Revision 15) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1 (Page 14 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 15 of 23)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1 (Page 16 of 23)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 17 of 23)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class.
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A.1 (Page 18 of 23)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

Table A.1 (Page 19 of 23)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 20 of 23)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference)	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly:	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel assembly weight:	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A.1 (Page 21 of 23)
Fuel Assembly Limits

VI. MP C MODEL: MPC-32 (continued)

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.1 (Page 22 of 23)
Fuel Assembly Limits

VII. MPC MODEL: MPC-HB

A. Allowable Contents

1. Uranium oxide, INTACT and/or UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C and the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 29 years, an average burnup ≤ 23,000 MWD/MTU, and a minimum initial enrichment ≥ 2.09 wt% ^{235}U.
e. Fuel assembly length:	≤ 96.91 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels and DFC
h. Decay heat per assembly:	≤ 50 W
h. Decay heat per MPC:	≤ 2000 W

Table A.1 (Page 23 of 23)
Fuel Assembly Limits

VII. MPC MODEL: MPC-HB (continued)

B. Quantity per MPC-HB: Up to 80 fuel assemblies

C. Damaged fuel assemblies and fuel debris must be stored in a damaged fuel container. Allowable Loading Configurations: Up to 28 damaged fuel assemblies/fuel debris, in damaged fuel containers, may be placed into the peripheral fuel storage locations as shown in SAR Figure 6.I.3, or up to 40 damaged fuel assemblies/fuel debris, in damaged fuel containers, can be placed in a checkerboard pattern as shown in SAR Figure 6.I.4. The remaining fuel locations may be filled with intact and/or undamaged fuel assemblies meeting the above applicable specifications, or with intact and/or undamaged fuel assemblies placed in damaged fuel containers.

NOTE 1: The total quantity of damaged fuel or fuel debris permitted in a single damaged fuel container is limited to the equivalent weight and special nuclear material quantity of one intact assembly.

NOTE 2: Fuel debris includes material in the form of loose debris consisting of zirconium clad pellets, stainless steel clad pellets, unclad pellets, or rod segments up to a maximum of one equivalent fuel assembly. A maximum of 1.5 kg of stainless steel clad is allowed per cask.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	14x14E
Clad Material (Note 2)	ZR	ZR	ZR	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide Tubes	17	17	5 (Note 4)	16	0
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.015	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. ZR Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	$\leq$ 110	$\leq$ 110	$\leq$ 110	$\leq$ 100	$\leq$ 195	$\leq$ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	$\leq$ 2.7	$\leq$ 2.7 for the UO_2 rods. See Note 4 for MOX rods	$\leq$ 2.7	$\leq$ 2.7	$\leq$ 4.2	$\leq$ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 5.5	$\leq$ 5.0	$\leq$ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	$\geq$ 0.5550	$\geq$ 0.5625	$\geq$ 0.5630	$\geq$ 0.4860	$\geq$ 0.5630	$\geq$ 0.4120
Fuel Clad I.D. (in.)	$\leq$ 0.5105	$\leq$ 0.4945	$\leq$ 0.4990	$\leq$ 0.4204	$\leq$ 0.4990	$\leq$ 0.3620
Fuel Pellet Dia. (in.)	$\leq$ 0.4980	$\leq$ 0.4820	$\leq$ 0.4880	$\leq$ 0.4110	$\leq$ 0.4910	$\leq$ 0.3580
Fuel Rod Pitch (in.)	$\leq$ 0.710	$\leq$ 0.710	$\leq$ 0.740	$\leq$ 0.631	$\leq$ 0.738	$\leq$ 0.523
Active Fuel Length (in.)	$\leq$ 120	$\leq$ 120	$\leq$ 77.5	$\leq$ 80	$\leq$ 150	$\leq$ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	$\geq$ 0	$\geq$ 0	N/A	N/A	N/A	$\geq$ 0
Channel Thickness (in.)	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.120	$\leq$ 0.100

Table A.3 (Page 2 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**8x8B**	**8x8C**	**8x8D**	**8x8E**	**8x8F**	**9x9A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6D**	**7x7C**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.)(Note 3)	≤ 78	≤ 78
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.6	≤ 2.6
Initial Maximum Rod Enrichment (wt.% ^{235} U)	≤ 4.0 (Note 14)	≤ 4.0
No. of Fuel Rod Locations	36	49
Fuel Clad O.D. (in.)	≥ 0.5585	≥ 0.486
Fuel Clad I.D. (in.)	≤ 0.505	≤ 0.426
Fuel Pellet Dia. (in.)	≤ 0.488	≤ 0.411
Fuel Rod Pitch (in.)	≤ 0.740	≤ 0.631
Active Fuel Length (in.)	≤ 80	≤ 80
No. of Water Rods (Note 11)	0	0
Water Rod Thickness (in.)	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060

Table A.3 (Page 6 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. ZR designates cladding material made from Zirconium or Zirconium alloys.

3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full-length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

8. This assembly class contains 92 total fuel rods; 78 full-length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

14. Only two assemblies may contain one rod each with an initial maximum enrichment up to 5.5 wt%.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 11	≤ 39,500	≥ 3.2
≥ 14	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 5	≤ 10,000	≥ 0.7
≥ 7	≤ 20,000	≥ 1.35
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% ^{235}U)
≥16	≤42,000	≥3.09
≥16	≤37,500	≥2.6
≥16	≤30,000	≥2.1

NOTES:

1. Each fuel assembly must only meet one set of limits (i.e., one row)

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCES COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (Years)
BPRAs	≤15,998	≥24
TPDs	≤118,674	≥11
RCCAs	≤125,515	≥9
Cf neutron source	≤15,998	≥24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤45,361	≥19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤88,547	≥9

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥12	≤24,500	≥2.3
≥14	≤29,500	≥2.6
≥16	≤34,500	≥2.9
≥19	≤39,500	≥3.2
≥20	≤42,500	≥3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% U-235)
≥8	≤24,500	≥2.3
≥9	≤29,500	≥2.6
≥12	≤34,500	≥2.9
≥14	≤39,500	≥3.2
≥19	≤44,500	≥3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	Configuration (Note 2)	Maximum Enrichment (wt.% U-235)	Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)
15x15D, E, F, H	A	4.65	$B = (1.6733)*E^3-(18.72)*E^2+(80.5967)*E-88.3$
	B	4.38	$B = (2.175)*E^3-(23.355)*E^2+(94.77)*E-99.95$
	C	4.48	$B = (1.9517)*E^3-(21.45)*E^2+(89.1783)*E-94.6$
	D	4.45	$B = (1.93)*E^3-(21.095)*E^2+(87.785)*E-93.06$
17x17A,B,C	A	4.49	$B = (1.08)*E^3-(12.25)*E^2+(60.13)*E-70.86$
	B	4.04	$B = (1.1)*E^3-(11.56)*E^2+(56.6)*E-62.59$
	C	4.28	$B = (1.36)*E^3-(14.83)*E^2+(67.27)*E-72.93$
	D	4.16	$B = (1.4917)*E^3-(16.26)*E^2+(72.9883)*E-79.7$

NOTES:

1. E = Initial enrichment (e.g., for 4.05 wt.%, E = 4.05).
2. See Table A.13.
3. Fuel Assemblies must be cooled 5 years or more.

Table A.13

LO ADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report on the HI-STAR 100 Cask System*, Revision 15, dated October 11, 2010.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9263	8	71-9263	USA/9263/B(U)-96	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc. application dated February 14, 2011, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SPEC-150

(2) Description

A welded titanium-encased, uranium-shielded, radiographic exposure device. Primary components consist of an outer titanium shell, internal supports, depleted uranium shield, and a titanium, titanium alloy or zircalloy S-tube. The contents are securely positioned in the S-tube by a source cable lock assembly and source safety plug assembly. The unit resembles a rectangular box approximately 5.4 inches wide, 5.6 inches high and 14.5 inches long. The maximum weight of the package is 53.5 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 15B000, Rev. 10; 15B001-3, Rev. 3; 15B002-A, Rev. 9; 15B008, Rev. 7; 19B005, Rev. 3; 19B006, Rev. 3; and 190909, Rev. 0.

(b) Contents

(1) Type and form of material

Iridium-192, Selenium-75, and Ytterbium-169 as encapsulated sealed sources meeting the requirements of special form radioactive material.

(2) Maximum quantity of material per package

5.55 TBq (150 Ci) (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography".

6. The source shall be secured in the shielded position of the packaging by the source assembly lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly used must be fabricated of materials capable of resisting a 1475 degrees Fahrenheit fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

Packagings must be marked with Package Identification Number USA/9263/B(U)-96.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7, of the application, as supplemented, and

(b) Each packaging must meet the Acceptance Tests and Maintenance Program in Section 8, of the application, as supplemented.

(c) The packaging will be fabricated and inspected in accordance with the 2007 or later edition of the ASME Code, Section VIII, Division 1. Alternatively, the 2007 or later edition of the AWS D1.9 Welding Code may be used for fabrication and inspection. Regardless of which construction code is used, any single package must be entirely fabricated and inspected in accordance with only a single edition of the referenced construction code. No mixing of codes or editions is permitted for a single package.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9263	8	71-9263	USA/9263/B(U)-96	3 OF	3

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Revision No. 7 of the certificate may be used until February 28, 2012.

11. Expiration date: June 30, 2015.

REFERENCES

Source Production and Equipment Company, Inc., application dated February 14, 2011.
Supplement dated February 18, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Spent Fuel Storage and Transportation Division
Office of Nuclear Material Safety
and Safeguards

Date: February 28, 2011

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9269	6	71-9269	USA/9269/B(U)-96	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., application dated August 3, 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 650L

(2) Description

A welded stainless steel encased, uranium shielded, Iridium-192 or Selenium-75 source changer. Primary components consist of a welded carbon steel shell, internal supports, depleted uranium shield, and a titanium "U" tube. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly. The Model No. 650L has two source locking assemblies, mounted on the top cover plate, that are used to secure the radioactive source in a shielded position during transport. The packaging measures approximately 10-inches (254 mm) wide, 13.25-inches (337 mm) high and 8.25-inches (210 mm) deep. The maximum weight of the packaging is 90 pounds (41 kg).

(3) Drawings

The packaging is constructed in accordance with QSA Global, Inc., Drawing No. R65006, Rev. J, sheets 1-5.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed sources which meet the requirements of special form radioactive material.

5. (b) Contents (continued)

Selenium-75 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

Ir-192: 240 curies (8.9 TBq) (output)

Se-75: 300 curies (11.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/(h-Ci) Iridium-192 at 1 meter (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography") and 0.2 R/(h-Ci) Selenium-75 at 1 meter (Ref: U.S. Public Health Service, Bureau of Radiological Health, 1970: Radiological Health Handbook, Rockville, MD).

(3) Maximum weight of contents

0.08 pounds (36 grams), including the mass of radioactive material and the weight of the source capsule handling wire assembly for a shipment containing two source wire assemblies.

(4) Maximum decay heat

Ir-192: 4.8 Watts

Se-75: 1.52 Watts

6. The source shall be secured in the shielded position of the packaging by the source assembly. The source assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining its positioning function. The cable of the source assembly must engage the source hold-down assembly. The flexible cable of the source assembly must be of sufficient length and diameter to provide positive positioning of the source at the crimp of the "U" tube.

7. The nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7 of the application, and

(b) The packaging shall be maintained in accordance with the Maintenance Program in Chapter 8 of the application.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGE
9269	6	71-9269	USA/9269/B(U)-96	3	OF	3

9. Fabrication of new packagings is not authorized. However, fabrication of replacement components needed to support shipment of existing packages is authorized, except for the depleted uranium shield and the inner carbon steel shell.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Revision No. 5 of this certificate may be used until September 30, 2011.

12. Expiration date: November 30, 2015.

REFERENCES

QSA Global, Inc., application dated August 3, 2010.
Supplements dated August 11 and 25, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: September 14, 2010

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9270	4	71-9270	USA/9270/B(U)F-96	1	OF	20

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International, Inc.
3930 East Jones Bridge Rd.
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc. application dated April 30, 1997, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UMS Universal Transport Cask Package

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

The UMS is a canister-based system for the storage and transportation of spent nuclear fuel. The transportation component of the UMS system designated the Universal Transport System, consists of a Universal Transport cask body with a closure lid and energy absorbing impact limiters loaded with a Transportable Storage Canister (TSC) containing either spent Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) nuclear fuel or Maine Yankee site specific contents including Greater than Class C (GTCC) waste.

The NAC-UMS is designed to transport up to 24 intact PWR spent fuel assemblies, 56 intact BWR spent fuel assemblies, GTCC waste, or site specific spent nuclear fuel with associated component hardware. Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Class 4 and 5 BWR fuel assemblies include the zirconium alloy channels. The loading of site specific fuels that include control component hardware may require the use of a TSC that is longer than if the hardware were excluded. The spent fuel is loaded into a TSC which contains a stainless steel grid work referred to as a basket.

The cask body of the UMS is a right-circular cylinder of multi wall construction which consists of 304 stainless steel inner and outer shells separated by lead gamma radiation shielding which is poured in place. The inner and outer shells are welded to a 304 stainless steel top forging which mates to the cask lid. The inner shell is also welded to a 304 stainless steel bottom forging and the outer shell is welded to the bottom plate. The cask bottom consists of the bottom forging and bottom plate with

5.(a)(2) Description (Continued)

neutron shield material sandwiched between them. Layers of 4.5 inches thick 304 stainless steel ring and two 0.75 inch stainless steel disks are located at the bottom lead annulus between the bottom forging and the outer shell.

Neutron shield material is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates.

The containment boundary of the UMS consists of the inner shell; bottom forging; top forging; cask lid and lid inner O-ring; vent port cover plate and vent port cover plate inner O-ring; and drain port cover plate and drain port cover plate inner O-ring.

There are five TSCs of different lengths, each to accommodate a different class of PWR or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and the lengths vary from about 175 to 192 inches long. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The TSC contains the basket and fuel assemblies or GTCC waste. Spacers are placed below each Class 1, 2, 4 or 5 canisters to locate and support the canister in the cask cavity.

The spacers are free standing structures that are confined in place by the bottom of the canister and the cask bottom inner surface. The spacer(s) ensure that the canister lid is laterally supported by the cask top forging when the cask is horizontal and minimizes axial movement of the canister. Each Class 1 PWR canister is positioned by a stainless steel spacer that is 16.75 inches in length. Each Class 2 PWR canister is positioned by a stainless steel spacer that is 7.65 inches in length. No spacers are used with the Class 3 PWR canister. The Class 4 BWR canister is located by four 1.5 inch aluminum spacers and the Class 5 BWR canister is located with a 1.5 inch aluminum spacer.

The spent fuel basket design uses a series of high strength stainless steel PWR or carbon steel BWR support disks to support the fuel assemblies in stainless steel tubes. The PWR fuel tubes contain neutron absorber on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing neutron absorber on two sides of the tubes; (2) tubes containing neutron absorber on one side; and (3) tubes containing no neutron absorber. Aluminum heat transfer disks are provided in both the PWR and BWR fuel baskets to enhance thermal performance of the basket. The heat transfer disks are supported by stainless steel tie rods and split spacers that maintain the basket assembly configuration.

The GTCC waste canister is essentially identical to the Class 1 TSC, except for the placement of lifting lugs and the placement of a key way within the canister. The GTCC basket is constructed of Type 304 stainless steel and consists primarily of a cylinder with a 3-inch thick wall closed at the bottom end with a 3-inch thick plate. The cylinder is centered in the GTCC waste canister by 14 Type 304 stainless steel support plates along its length. A 3-inch thick 304 stainless steel separator fixture divides the cylinder into two vertically stacked components, each 77 inches deep with a diameter of 47.8 inches.

5.(a)(2) Description (Continued)

The package has impact limiters at each end of the cask body. The impact limiters consist of a combination of redwood and balsa wood encased in Type 304 stainless steel. The impact limiters limit the g-loads acting on the cask during a transport drop load condition due to crushing of the redwood and balsa wood. The upper and lower impact limiters are bolted to the cask body by 16 equally spaced attachment rods with nuts.

The approximate dimensions and weights of the package are as follows:

Overall length (with impact limiters, in)	273.3
Overall length (without impact limiters, in)	209.3
Impact Limiter Outside diameter (in)	124.0
Outside diameter (without impact limiters, in)	92.9
Cavity diameter (in)	67.6
Cavity length (in)	192.5
Cask lid thickness (in)	6.5
Bottom thickness (in)	10.3
Inner shell thickness (in)	2.0
Outer shell thickness (in)	2.75
Gamma shield thickness (in)	2.75
Radial neutron shield thickness (in)	4.50

Transportable Storage Canister

Shell thickness (in)	0.625
Shell bottom (in)	1.75
Shield lid thickness (in)	7
Structural lid thickness (in)	3
Outer diameter (in)	67
Internal cavity diameter (in)	65.8
Internal fuel cavity length (in), depending on class	163-180
Overall length (in), depending on class	175-192

Fuel Basket

Basket assembly length (in), depending on class	162-180
Basket assembly diameter (in)	65.5
Number of support disks, depending on class	30-41
Number of heat transfer disks, depending on class	17-33

Total weight (pounds) including cask, basket, impact limiters, fuel, canister with lids, cask lid, and spacers for each fuel class is approximately:

Class 1 (PWR)	251,000
Class 2 (PWR)	252,000
Class 3 (PWR)	249,000
Class 4 (BWR)	256,000
Class 5 (BWR)	255,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with NAC drawings:

790-209, Rev. 1	790-210, Rev. 1	790-500, Rev. 4	790-501, Rev. 3
790-502, Rev. 7	790-503, Rev. 3	790-504, Rev. 2	790-505, Rev. 2
790-508, Rev. 2	790-509, Rev. 3	790-516, Rev. 3	790-519, Rev. 2
790-520, Rev. 2	790-570, Rev. 4	790-571, Rev. 3	790-572, Rev. 4
790-573, Rev. 7	790-574, Rev. 3	790-575, Rev. 10	790-581, Rev. 9
790-582, Rev. 12	790-583, Rev. 8	790-584, Rev. 19	790-585, Rev. 19
790-587, Rev. 1	790-591, Rev. 6	790-592, Rev. 8	790-593, Rev. 7
790-594, Rev. 2	790-595, Rev. 10	790-605, Rev. 11	790-611, Rev. 6
790-612, Rev. 9	412-501, Rev. 4	412-502, Rev. 6	

5.(b) Contents

(1) Type and Form of Material

The package is designed to transport four types of contents as listed below:

i. 24 intact irradiated PWR fuel assemblies within a TSC;
ii. 56 intact irradiated BWR fuel assemblies within a TSC;
iii. 24 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee within a TSC; or
iv. GTCC waste from Maine Yankee within a TSC.

Each type of package contents is described in detail below.

(i) Intact PWR assemblies

The package is designed to transport 24 irradiated intact PWR fuel assemblies within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. An empty fuel rod position must be filled with a solid filler rod, fabricated from either zirconium alloy or Type 304 stainless steel, which displaces an equal or greater volume than that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications of Table 5.(b)(1)(i)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(i)-2 below. PWR fuel assemblies may include standard inserts such as guide tube thimble plugs and burnable poison rods.

The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.2 wt% ^{235}U respectively. Unenriched fuel assemblies are not authorized for loading into the TSC. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is 5 years. The maximum weight of UO_2 is 11.53 MTU per cask.

Table 5.(b)(1)(i)-1, Intact PWR Fuel Assembly Characteristics

TSC Class[1]	Vendor[2]	Array	Max. Length (in)	Max. Width (in)	Max. Assembly Weight	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	Min Guide Tube Thickness (in)
1	CE	14x14	157.3	8.11	1292	0.404	176[4]	0.590	0.438	0.024	0.380	137.0	0.040
1	Ex/ANF	14x14	160.2	7.76	1271	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14x14	159.8	7.76	1177	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14x14	159.8	7.76	1302	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15x15	159.8	8.43	1472	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17x17	159.8	8.43	1348	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17x17	159.8	8.43	1482	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17x17	160.1	8.43	1373	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15x15	165.7	8.54	1515	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17x17	165.8	8.54	1505	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16x16	178.3	8.10	1430	0.442	236[4]	0.506	0.382	0.023	0.3255	150.0	0.035
1	Ex/ANF[3]	14x14	160.2	7.76	1215	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15x15	147.5	8.20	1360	0.432	216	0.550	0.418	0.026	0.358	132.0	---
1	Ex/ANF[3]	15x15	148.9	8.25	1339	0.431	216	0.550	0.417	0.030	0.358	131.8	---
1	CE[3]	16x16	158.2	8.10	1300	0.403	236[4]	0.506	0.382	0.023	0.3255	136.7	0.035

[1] Minimum and Maximum initial Enrichments are 1.9 wt% ^{235}U and 4.2 wt% ^{235}U, respectively. All fuel rods are zirconium alloy type clad.

[2] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[3] 14x14, 15x15, and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively, These are not generic fuel assemblies provided to multiple reactors.

[4] Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 5.(b)(1)(i)-2, Loading Table for Intact PWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)					30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	6	8	8	7	8	8	10	11	9	10
2.1 ≤ E < 2.3	6	7	8	6	7	7	10	10	8	10
2.3 ≤ E < 2.5	6	7	7	6	7	7	9	10	8	9
2.5 ≤ E < 2.7	6	7	7	6	7	7	9	9	7	8
2.7 ≤ E < 2.9	6	7	7	6	7	6	8	9	7	8
2.9 ≤ E < 3.1	5	7	7	6	6	6	8	8	7	8
3.1 ≤ E < 3.3	5	6	7	6	6	6	8	8	7	7
3.3 ≤ E < 3.5	5	6	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	5	6	6	6	6	6	7	7	6	7
3.7 ≤ E ≤ 4.2	5	6	6	6	6	6	7	7	6	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)					40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	11	15	15	13	15	18	20	21	20	20
2.1 ≤ E < 2.3	10	13	14	12	13	15	19	19	18	19
2.3 ≤ E < 2.5	9	12	13	11	12	14	17	19	17	17
2.5 ≤ E < 2.7	9	12	12	10	11	12	16	18	15	17
2.7 ≤ E < 2.9	8	11	11	9	11	11	15	18	14	17
2.9 ≤ E < 3.1	8	10	10	9	10	10	14	18	13	15
3.1 ≤ E < 3.3	7	10	10	9	10	10	13	17	13	15
3.3 ≤ E < 3.5	7	9	10	8	9	9	12	17	13	15
3.5 ≤ E < 3.7	7	9	10	8	9	8	11	17	12	15
3.7 ≤ E ≤ 4.2	7	8	10	8	8	8	11	15	12	14

5.(b)(1)(ii) Intact BWR assemblies

The package is designed to transport 56 irradiated intact BWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks.

For BWR fuel, the initial enrichment limit (the enrichment of the as-delivered fresh fuel assembly) represents the maximum peak planar-average enrichment allowed for loading into the TSC. The peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 5.(b)(1)(ii)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(ii)-2.

BWR intact fuel assemblies are authorized with or without channels based on a maximum channel width of 120 mils. The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.0 wt% ^{235}U respectively. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is six years. The maximum weight of UO_2 is 11.08 MTU per cask. Unenriched fuel assemblies are not authorized for loading into the TSC. BWR fuel assemblies with unenriched axial blankets must have an enriched central fuel region and are acceptable for loading into a TSC if the minimum fuel enrichment of the central region is 1.9 wt% ^{235}U. Any empty fuel position must be filled with a solid filler rod fabricated from either zirconium alloy or Type 304 stainless steel.

Table 5.(b)(1)(ii)-1, Intact BWR Fuel Assembly Characteristics												
Canister Class [1,5]	Vendor[4]	Array	Max. Length (in)	Max. Assembly Width (in)[5]	Max. Assembly Weight (lb)[6]	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)[2]
4	Ex/ANF	7x7	171.3	5.51	620	0.196	48	0.738	0.570	0.036	0.490	144
4	Ex/ANF	8x8	171.3	5.51	563	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9x9	171.3	5.51	557	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7x7	171.1	5.51	681	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7x7	171.2	5.51	681	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8x8	171.1	5.51	639	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8x8	176.1	5.51	588	0180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9x9	176.1	5.51	576	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[5]	Ex/ANF	9x9	176.1	5.51	576	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7x7	175.9	5.51	683	0.198	49	0.738	0.563	0.032	0.487	144.0
5	GE	8x8	176.1	5.51	665	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9x9	176.1	5.51	646	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9x9	176.1	5.51	646	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

1 Maximum Peak Planar Average Enrichment 4.0 wt% ^{235}U. Minimum enrichment is 1.9 wt% ^{235}U. All fuel rods are zirconium alloy type clad.

2 150 inch active fuel length assemblies contain 6 inch natural uranium blankets on top and bottom.

3 Shortened active fuel length in some rods.

4 Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

5 Assembly width including channel. Unchanneled or channeled may be loaded based on a maximum channel thickness of 120 mils.

6 Exxon/ANF assembly weights are listed without channel.

Table 5.(b)(1)(ii)-2, Loading Table for Intact BWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)			30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	8	8	8	14	13	15
2.1≤ E < 2.3	7	7	8	12	12	13
2.3≤ E < 2.5	7	7	7	11	10	11
2.5 ≤ E < 2.7	7	6	7	9	9	10
2.7≤ E < 2.9	6	6	6	9	8	9
2.9≤ E < 3.1	6	6	6	8	8	8
3.1≤ E < 3.3	6	6	6	7	7	8
3.3 ≤ E < 3.5	6	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	7	7	7
3.7 ≤ E≤ 4.0	6	6	6	7	7	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)			40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	24	23	25	34	33	35
2.1≤ E < 2.3	21	20	22	31	30	32
2.3≤ E < 2.5	19	18	20	29	28	29
2.5 ≤ E < 2.7	17	16	17	26	25	27
2.7≤ E < 2.9	14	14	15	24	23	24
2.9≤ E < 3.1	13	12	13	21	20	22
3.1≤ E < 3.3	11	11	12	19	18	20
3.3 ≤ E < 3.5	10	10	11	17	16	18
3.5 ≤ E < 3.7	10	9	10	15	14	16
3.7 ≤ E≤ 4.0	10	9	10	14	13	15

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The package is designed to transport 24 irradiated intact or damaged PWR fuel assemblies, canistered fuel debris, and GTCC waste within the TSC from the Maine Yankee Reactor. The standard Maine Yankee fuel assembly is the intact PWR CE 14x14 (see section 5.(b)(1)(i)).

In the course of reactor operations, some of the 14x14 assemblies were modified to change the standard configuration. These modifications included a) the removal of fuel rods without replacement; b) the replacement of removed fuel rods or burnable poison rods with rods of a different material, such as stainless steel, or with fuel rods of a different enrichment; and c) the insertion of control elements, or instruments or plug thimbles, in guide tube positions. In addition to the modified fuel assemblies, there are fuel assemblies that were designed with variable enrichment and axial blankets. These fuel assemblies are not modified, but differ from the cask design basis fuel assemblies.

Stainless steel spacers may be used in canisters to axially position PWR intact fuel assemblies that are shorter than the available cavity length. The minimum length of the PWR intact fuel assembly internal structure and bottom end fitting and/or spacers will ensure that the minimum distance to the fuel region for the base of the canister is 3.2 inches.

Unenriched fuel assemblies are not authorized for loading.

The following are the allowable Maine Yankee site specific contents:

5.(b)(1)(iii)(A) Maine Yankee's site specific contents not requiring preferential loading patterns:

(1) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly characteristics in Table 5.(b)(1)(i)-1. The maximum fuel assembly weight, including other associated hardware is 1,515 pounds. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) Irradiated Maine Yankee CE 14 x 14 PWR intact fuel assemblies may contain inserted control element assemblies (CEA), in-core instrument (ICI) thimbles or CEA plugs. CEAs or CEA plugs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or CEA plug inserted must be loaded in a Class 2 canister and cannot be loaded in a Class 1 canister. Fuel assemblies without an inserted CEA or CEA plug, including those with inserted ICI Thimbles, must be loaded in a Class 1 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 except for those assemblies containing ICI thimbles which must meet the specifications of Table 5.(b)(1)(iii)(A)-2.

(3) PWR intact fuel assemblies with fuel rods replaced with stainless steel or zirconium alloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt%. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-3.

(4) PWR intact fuel assemblies with fuel rods having variable enrichments with a maximum rod enrichment up to 4.21 wt% ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(5) PWR intact fuel assemblies with annular axial end blanket enrichments up to 2.6 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(6) PWR intact fuel assemblies with burnable poison rods or solid filler rods may occupy up to 16 of 176 fuel rod positions. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(7) PWR intact fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

5.(b)(1)(iii)(B) Maine Yankee site-specific allowable contents requiring preferential loading based on shielding, criticality, or thermal constraints (Maine Yankee CE 14 x 14 intact PWR fuel assemblies). A PWR basket fuel diagram can be found on Figure 5.(b)(1)(iii)(B)-1.

(1) Maine Yankee CE 14 x 14 PWR intact fuel assemblies with a burn up between 45,000 and 50,000 MWD/MTU meeting the following requirements for verification of the oxide layer thickness and high burn up fuel requiring preferential loading in the peripheral PWR fuel basket positions:

A verification program is required to determine the oxide layer thickness on high burn up fuel by measurement or by statistical analysis. A fuel assembly having a burn up between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burn up. The verification program shall be capable of classifying high burn up fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

I. A HIGH BURN UP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

II. A HIGH BURN UP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) PWR intact fuel assemblies with up to 176 fuel rods missing from the fuel assembly lattice. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner loading position in the PWR fuel basket.

(3) PWR intact fuel assemblies with burnable poison rods replaced by hollow zirconium alloy rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(4) Intact fuel assemblies with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 canister. Only one start-up source may be loaded in any fuel assembly or any canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(5) PWR intact fuel assemblies with CEA ends (fingertips) and/or an ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a PWR fuel basket corner position and must be loaded in a Class 2 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. CEA fingertips are not considered as CEAs for determination of minimum cool times.

5.(b)(1)(iii)(C) Maine Yankee CE 14 x 14 PWR fuel enclosed in a Maine Yankee Fuel Can (MYFC).

All Maine Yankee CE 14 x 14 PWR fuel enclosed in an MYFC must be loaded in a Class 1 fuel canister in a corner position of the PWR fuel basket. Up to 4 MYFC may be loaded into a TSC. Intact Maine Yankee CE 14 x 14 PWR fuel may be loaded into a MYFC. The contents that must be loaded in the MYFC are:

(1) PWR fuel assemblies with up to two intact or damaged fuel rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(2) A damaged fuel assembly with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A damaged fuel assembly cannot have an inserted CEA or other non-fuel component. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(3) Individual intact or damaged fuel rods in a rod type structure, which may be a guide tube, to maintain configuration control. The combined maximum average burn up,

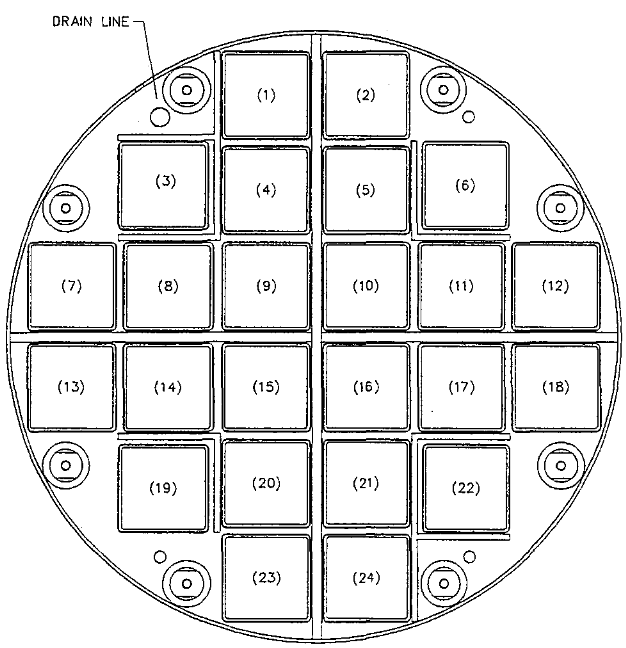
minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(4) Fuel debris consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(5) Consolidated Fuel lattice and structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight less than or equal to 2,100 pounds. A consolidated fuel lattice cannot have an inserted CEA or other non-fuel component. Only one consolidated fuel lattice may be stored in any TSC. Fuel must be cooled a minimum of 24 years.

(6) High burn-up fuel assemblies not meeting the oxide layer thickness criteria previously defined in Section 5.(b)(1)(iii)(B)(1). The combined maximum average burn-up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

PWR Basket Fuel Loading Position Diagram, Figure 5.(b)(1)(iii)(B)-1



1. Basket corner positions are positions 3, 6, 19, and 22. Corner positions are also periphery positions.
2. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24. Periphery positions include the corner positions.

Table 5.(b)(1)(iii)(A)-1, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 30 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	6	6	7	6	6	6
2.1≤ E < 2.3	6	6	7	6	6	6
2.3≤ E < 2.5	6	6	6	6	6	6
2.5 ≤ E < 2.7	6	6	6	6	6	6
2.7≤ E < 2.9	6	6	6	6	6	6
2.9≤ E < 3.1	5	6	6	6	6	6
3.1≤ E < 3.3	5	5	6	6	6	5
3.3 ≤ E < 3.5	5	5	6	6	5	5
3.5 ≤ E < 3.7	5	5	6	5	5	5
3.7 ≤ E≤ 4.2	5	6	5	5	5	5

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 35 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	8	8	9	8	8	8
2.1≤ E < 2.3	7	7	9	8	8	8
2.3≤ E < 2.5	7	7	8	7	7	7
2.5 ≤ E < 2.7	7	7	8	7	7	7
2.7≤ E < 2.9	6	7	7	7	7	7
2.9≤ E < 3.1	6	6	7	7	6	6
3.1≤ E < 3.3	6	6	7	6	6	6
3.3 ≤ E < 3.5	6	6	7	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6	6
3.7 ≤ E≤ 4.2	6	6	6	6	6	6

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9270	4	71-9270	USA/9270/B(U)F-96	15 OF	20

Table 5.(b)(1)(iii)(A)-1, continued , Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 40 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	11	12	14	13	12	12
2.1≤ E < 2.3	10	10	13	11	11	11
2.3≤ E < 2.5	9	9	12	10	10	10
2.5 ≤ E < 2.7	9	9	10	9	9	9
2.7≤ E < 2.9	8	8	10	9	8	8
2.9≤ E < 3.1	8	8	9	8	8	8
3.1≤ E < 3.3	7	7	8	8	8	8
3.3 ≤ E < 3.5	7	7	8	7	7	7
3.5 ≤ E < 3.7	7	7	8	7	7	7
3.7 ≤ E≤ 4.2	7	7	7	7	7	7

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 45 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	18	18	21	19	18	18
2.1≤ E < 2.3	15	16	19	17	17	16
2.3≤ E < 2.5	14	14	18	16	15	15
2.5 ≤ E < 2.7	12	13	16	14	14	13
2.7≤ E < 2.9	11	12	14	13	12	12
2.9≤ E < 3.1	10	11	13	12	11	11
3.1≤ E < 3.3	10	10	12	11	10	10
3.3 ≤ E < 3.5	9	9	11	10	10	10
3.5 ≤ E < 3.7	9	9	10	10	10	10
3.7 ≤ E≤ 4.2	9	9	10	10	10	10

Table 5.(b)(1)(iii)(A)-1, continued, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 50 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	27	27	29	27	27	27
2.1≤ E < 2.3	24	24	27	25	24	24
2.3≤ E < 2.5	22	22	25	23	22	22
2.5 ≤ E < 2.7	19	19	23	21	20	20
2.7≤ E < 2.9	17	17	21	19	18	18
2.9≤ E < 3.1	15	16	19	18	18	18
3.1≤ E < 3.3	15	15	18	17	17	17
3.3 ≤ E < 3.5	15	15	17	17	17	17
3.5 ≤ E < 3.7	14	14	15	15	15	15
3.7 ≤ E≤ 4.2	14	14	15	15	15	15

Table 5.(b)(1)(iii)(A)-2, Loading Table (Years) for Maine Yankee CE 14x14 fuel containing ICI Thimbles

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 ≤ Burnup ≤ 35 GWD/MTU	35 ≤ Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	6	8	11	18	27
2.1≤ E < 2.3	6	7	10	16	24
2.3≤ E < 2.5	6	7	9	14	22
2.5 ≤ E < 2.7	6	7	9	13	19
2.7≤ E < 2.9	6	6	8	11	17
2.9≤ E < 3.1	5	6	8	10	15
3.1≤ E < 3.3	5	6	7	10	15
3.3 ≤ E < 3.5	5	6	7	9	15
3.5 ≤ E < 3.7	5	6	7	9	14
3.7 ≤ E≤ 4.2	5	6	7	9	14

Table 5.(b)(1)(iii)(A)-3, Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assy Number	Burnup (GWD/MTU)	Enrichment (wt %)	SSR Source (g/s/assy)	Cool Time (years)	Earliest Transportable
N420	45	3.3	2.1602E+13	10	Jan 2001
N842	35	3.3	3.1396E+12	6	Jan 2001
N868	40	3.3	5.2444E+12	7	Jan 2001
R032	45	3.5	1.4550E+13	9	Jan 2005
R439	50	3.5	1.3998E+13	14	Jan 2010
R444	50	3.5	5.5993E+13	19	Jan 2015

Table 5.(b)(1)(iii)(A)-4, Cool time (years) for Maine Yankee CE 14x14 damaged fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	7	11	19	28	37
2.1≤ E < 2.3	6	9	16	26	34
2.3≤ E < 2.5	6	8	14	23	32
2.5 ≤ E < 2.7	6	8	12	21	30
2.7≤ E < 2.9	6	7	11	19	27
2.9≤ E < 3.1	6	7	10	17	25
3.1≤ E < 3.3	5	7	9	15	23
3.3 ≤ E < 3.5	5	6	8	13	21
3.5 ≤ E < 3.7	5	6	8	12	19
3.7 ≤ E≤ 4.2	5	6	7	11	17

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee

The package is designed to transport Maine Yankee Greater Than Class C Waste within a TSC. Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.15. The maximum curie inventory shall not exceed the values shown in Table 5.(b)(1)(iv)-1.

Table 5.(b)(1)(iv)-1, Maine Yankee GTCC Curie Inventory Limits per TSC

Radionuclide	Curie Inventory (Ci)/ TSC
H-3	3.00E+02
C-14	1.50E+02
Mn-54	3.50E+02
Fe-55	2.00E+05
Co-58	1.00E+01
Co-60	2.90E+05
Ni-59	8.20E+02
Ni-63	9.00E+04
Nb-94	1.00E+01
Tc-99	1.00E+01

5.(b)(2) Maximum quantity of material per package

The maximum weight of the contents shall not exceed 77,500 pounds.

(i) For the contents described in 5.(b)(1)(i) and 5.(b)(1)(iii): 24 PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, with a maximum weight of 38,500 pounds and a maximum decay heat limit per package not to exceed the values in Table 5.(b)(2)-1. The individual PWR assembly decay heat is limited to 0.83 kW.

Table 5.(b)(2)-1, PWR Decay Heat Limits				
Cool Time (Years)	PWR Decay Heat Limit (kW) Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000[1]
5	20.0	20.0	19.9	19.3
6	19.5	19.3	19.2	18.7
7	17.8	17.8	17.7	17.2
10	17.4	17.3	17.2	16.8
15	16.8	16.8	16.7	16.5

[1]Maine Yankee PWR fuel assemblies

(ii) For the contents described in 5.(b)(1)(ii): 56 BWR assemblies with a maximum weight of 39,000 pounds and a maximum decay heat limit per package of 16 kW. The individual BWR assembly decay heat is limited to 0.29 kW.

(iii) For the contents described in 5.(b)(1)(iv): GTCC waste with a maximum weight per package of 20,000 pounds in total or 10,000 pounds per compartment. The maximum decay heat for the GTCC is 4.5 kW per package.

5.(c) Criticality Safety Index 0.0

6. The package must be transported as exclusive use in a closed transport vehicle as per 10 CFR 71.47(b).

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Transport by air of fissile material is not authorized.

10. Expiration date: October 31, 2017.

REFERENCES

NAC International, Inc., Application dated April 30, 1997.

NAC International, Inc., Supplements dated June 18, 1999, May 31, June 29, August 8, and September 20, 2000; February 28, March 14, March 31, June 1, and November 16, 2001; January 31, March 13, August 12, September 27, and October 21, 2002; March 31, and September 28, 2004; May 4, and June 6, 2005; September 25, 2007, and September 19, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 10/26/12

FORM 618
))
.³71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9274	9	71-9274	USA/9274/AF	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company, LLC
Columbia Fuel Fabrication Facility
5801 Bluff Road
Hopkins, SC 29061

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company application dated May 15, 2003, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: ABB-2901

(2) Description

A shipping container for low-enriched uranium oxide pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner container is 10.75 ± 1/4 inches square and approximately 30-inches long, constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has a 16-gauge closure lid. The drum lid is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt. In addition to the locking ring, three lid clamps are installed to secure the drum lid. The drum has approximate dimensions of 22.5-inch ID by 36-inch height. The uranium oxide pellets are packaged in boxes positioned within a steel insert. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC, Drawing Nos.

10004E01, Rev. 2;
10004E02, Sheets 1 and 2, Rev. 2; and
10004E03, Rev. 2.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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5. (b) Contents

(1) Type and form of material

Sintered uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope. The maximum pellet diameter is 0.969 cm, and the minimum pellet diameter is 0.818 cm.

(2) Maximum quantity of material per package

227 pounds of pellets, with the U-235 content not to exceed 4.54 kg. The pellets must be packaged on corrugated stainless steel trays, within shipping container boxes and a shipping container insert in accordance with ABB Combustion Engineering Nuclear Systems Drawing Nos. L-9274-02, Sheets 1 and 2, Rev. 0, and L-9274-03, Rev. 0.

Maximum weight of contents within the inner container is 427 pounds, including radioactive material, secondary containers, and other packaging material.

(c) Criticality Safety Index (minimum index to be shown on label): 0.5

6. Corrugated stainless steel trays must be positioned between each layer of pellets, and on the top and bottom of the pellet stack. Spacers must be inserted in partially filled pellet shipping boxes to provide a snug fit.

7. The package may also contain stainless steel pellets, depleted uranium pellets, and neutron poisons such as gadolinia, erbium, and boron carbide.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Prior to each shipment the insert (containment vessel) gasket shall be inspected. This gasket shall be replaced if inspection shows any defects.

(b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 and the Maintenance Program of Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Transport by air of fissile material is not authorized.

11. Expiration date: September 30, 2017.

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0)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Westinghouse Electric Company application dated May 15, 2003.

Supplements dated November 21, 2003; July 23, 2007; and July 20, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John M. Goshen

John M. Goshen, P.E.
Acting Branch Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 6 2012

UNITED STATES NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

EnergySolutions Spent Fuel Division
2105 S. Bascom Ave., Suite 230
Campbell, CA 95008

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BNFL Fuel Solutions application dated April 20, 2001, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. FuelSolutions™ TS125 Transportation Package

(2) Description

The FuelSolutions™ TS125 Transportation Package consists of a TS125 Transportation Cask and impact limiters, together with a FuelSolutions™ W21 or W74 canister and its payload. The FuelSolutions™ canister and its payload are contained inside the TS125 Transportation Cask cavity. The TS125 Transportation Cask cavity is sized to accommodate one FuelSolutions™ long canister, or alternatively, one FuelSolutions™ short canister with a cask cavity spacer. The approximate dimensions and weights of the package are as follows:

Package Length: .... 342.4 inches
Package Outside Diameter: .... 143.5 inches
Cask Length (w/o impact limiters): .... 210.4 inches
Cask Outside Diameter (w/o impact limiters): .... 94.2 inches
Cask Cavity Length: .... 193.0 inches
Cask Cavity Diameter (section at rails): .... 66.88 inches
Canister Outside Diameter: .... 66.0 inches
Maximum Long Canister Length: .... 192.25 inches
Maximum Short Canister Length: .... 182.25 inches
Cask Cavity Spacer Length: .... 10.0 inches

Max. Package Weight: .... 285,000.0 pounds
Max. Cask Payload Weight (incl. canister and cavity spacer):......... 85,000.0 pounds

The TS125 Transportation Cask body is an assembly composed of stainless steel components of an inner shell, an outer shell, a top ring forging, a closure lid with a seal test port and a cavity vent port, a bottom plate forging, and a cavity drain port. The inner and outer shells are welded to the bottom plate forging and the top ring forging. The cask body also includes an annular lead gamma shield; an annular neutron shield with cask tie-down rings, support angles, and jacket; a bottom end neutron shield with a support ring and jacket; a longitudinal shear block; and lifting trunnion mounting bosses. The inner and outer shells form the annular cavity for the lead gamma shield. The outer shell and the neutron shield jacket form the annular cavity for the solid neutron shield. The neutron shield support angles facilitate heat rejection through the solid neutron shielding material to the outer surface of the cask body. The cask closure lid includes a thick recessed plate with two concentric "Helicoflex" silver-jacketed metallic o-ring seals, the cavity vent port, and the seal test port. The closure lid is secured to the cask body during transport with 60 – 2 inch diameter closure bolts. The vent and drain ports are closed by a plug assembly to maintain containment integrity during transportation.

The Transportation Cask's containment boundary consists of: the inner cylindrical shell, the bottom plate forging (which forms the bottom closure of the cask), the top ring forging and sealing surfaces, the closure lid and sealing surfaces, the welds associated with the above components, the closure bolts, the innermost closure lid o-ring seal, the cavity vent port seal gland and o-ring seal, and the cavity drain port seal gland and o-ring seal. The package is designed to be "leaktight" as defined by ANSI N14.5 (leakage rate less than or equal to 1×10^{-7} ref-cm^3/s. The structural components of the Transportation Cask are made of high-strength austenitic stainless steel. The gamma shielding is made of lead and is completely enclosed within the annular region between the inner and outer steel shells. The neutron shielding is solid hydrogenous material that is completely enclosed within the annular region between the cask outer shell and neutron shield jacket with tie-down rings at each end.

The FuelSolutions™ TS125 Transportation Cask has identical energy-absorbing impact limiters at both ends. Each impact limiter assembly consists of crushable aluminum honeycomb energy-absorbing core segments that are encased in a sealed stainless steel shell. In addition to confining the aluminum honeycomb core segments in the event of a free drop, the impact limiter shell protects the aluminum honeycomb material from the weather. Both the top and bottom impact limiters are attached to the transportation cask body tie-down rings with 12, one inch diameter bolts. A tamper-indicating device is provided which connects each impact limiter to the transportation cask to assure that the package has not been opened by unauthorized personnel during transport.

A FuelSolutions™ canister consists of a steel shell assembly and an internal basket assembly. The shell assembly maintains a helium atmosphere for transport conditions. Credit is not taken for containment provided by the canister shell for transport conditions. The shell assembly also provides radiological shielding in both the radial and axial directions. The internal basket assembly provides geometric spacing, structural support, and criticality control for the spent nuclear fuel (SNF) assemblies for transport conditions. here are two classes of W21 canisters (W21T and W21M), differing primarily in materials of construction. Each W21 canister class includes four different canister types, as follows. The W21T canister class includes a long canister with lead shield plugs

(W21T-LL), a long canister with carbon steel shield plugs (W21T-LS), a short canister with lead shield plugs (W21T-SL), and a short canister with carbon steel shield plugs (W21T-SS). The W21M canister class includes a long canister with depleted uranium shield plugs (W21M-LD), a long canister with carbon steel shield plugs (W21M-LS), a short canister with depleted uranium shield plugs (W21M-SD), and a short canister with carbon steel shield plugs (W21M-SS). There are also two classes of W74 canisters (W74T and W74M), differing primarily in materials of construction. Both the W74T and W74M canister classes include only a long canister with carbon steel shield plugs.

A FuelSolutions™ canister shell assembly consists of a steel cylindrical shell, bottom end closure, bottom shield plug, bottom shell extension, bottom outer plate, top shield plug, top inner closure plate, and top outer closure plate. The closure plates at the top and bottom are welded to the cylindrical shell. All structural components of the canister shell assembly are constructed of austenitic stainless steel, with the exception of the shield plugs. The shield plug materials may be composed of lead, depleted uranium or carbon steel, depending upon the specific canister variant. To prevent any corrosion, galvanic, or chemical reactions between the shield plug materials and the cask environment or contents, the shield materials are isolated from the environment and cask interior. The lower shield plugs are encased within stainless steel. The upper shield plugs that are made of lead or depleted uranium are encased in stainless steel. The carbon steel upper shield plug is electroless nickel-plated.

A FuelSolutions™ W21 canister basket assembly consists of 21 guide tubes that are positioned and supported by a series of circular spacer plates, which are in turn positioned and supported by support rod assemblies. The W21 guide tubes include neutron absorber sheets on all four sides.

The W74 canister includes two stackable basket assemblies with a capacity to accommodate up to 64 Big Rock Point fuel assemblies. Each basket includes 37 cell locations, with the center five cell locations mechanically blocked to prevent fuel loading in these locations. The W74 basket assembly consists of a series of circular spacer plates that are positioned and supported by four support tubes that run through the spacer plates and support sleeves between the spacer plates. Each basket cell location, with the exception of the four support tubes and the five blocked-out center cells, contain a guide tube assembly. The W74 guide tube assemblies include borated stainless steel neutron absorber sheets on either one side or two opposite sides. The guide tubes are arranged in the basket to position at least one poison sheet between adjacent fuel assemblies, with the exception of intact fuel assemblies placed in the support tubes.

In the W74 basket, damaged fuel is placed in damaged fuel cans that are accommodated in the support tube cell locations. The W74 damaged fuel cans are similar to the W74 guide tubes, but include a screened bottom end, a screened removal lid, and borated stainless steel neutron absorber sheets on all four sides.

(3) Drawings

The FuelSolutions™ TS125 Transportation Package is constructed and assembled in accordance with the following drawings:

FS-200, Revision 1, Sheets 1 through 3
FS-205, Revision 2, Sheets 1 through 3
FS-210, Revision 2, Sheets 1 through 9
FS-220, Revision 1, Sheets 1 through 7
FS-230, Revision 1, Sheets 1 and 2
W21-110, Revision 4, Sheets 1 through 9
W21-120, Revision 5, Sheets 1 through 10
W21-121, Revision 5, Sheet 1
W21-122, Revision 3, Sheets 1 and 2
W21-130, Revision 4, Sheets 1 through 9
W21-131, Revision 3, Sheets 1 and 2
W21-140, Revision 5, Sheets 1 through 4
W21-150, Revision 4, Sheets 1 and 2
W21-190, Revision 4, Sheet 1
W74-110, Revision 5, Sheets 1 and 2
W74-120, Revision 5, Sheets 1 through 6
W74-121, Revision 7, Sheet 1
W74-122, Revision 6, Sheet 1
W74-130, Revision 6, Sheets 1 and 2
W74-140, Revision 5, Sheets 1 through 4
W74-150, Revision 5, Sheets 1 and 2
3319, Revision 6, Sheets 1 through 5

(b) Contents

(1) Type and Form of Material

Shipment of spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies, is not authorized.

(i) W21 Canister

The contents of the W21 canister are limited to 21 pressurized water reactor (PWR) SNF assemblies meeting the requirements of Table 1 and Table 2. Two different loading configurations, designated as W21-1 and W21-2, are permitted in the W21 canister. The W21-2 loading configuration, which accommodates SNF with higher initial ^{235}U enrichments, consists of up to 20 PWR SNF assemblies meeting the requirements of Table 1 and Table 2. The W21-2 loading configuration requires that the center guide tube be mechanically blocked to prevent inadvertent loading of a SNF assembly. If less than the maximum number of PWR assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the PWR assemblies they are replacing, must be loaded in the empty guide tubes.

The SNF assemblies that are permitted in the W21 canister must meet all of the parameter requirements of at least one criticality class. Table 2 lists the dimensional and initial enrichment limits for each criticality class of PWR fuel assembly. Table 2 provides separate assembly initial ^{235}U enrichment limits for the W21-1 and W21-2 canister loading configurations. The initial enrichment limits presented in Table 2 are bounding for assemblies containing any type of control insert, including assemblies with fuel rods replaced with any type of rod of equal or greater diameter and height.

Table 3 lists minimum required cooling times, as a function of burnup, for PWR assemblies loaded into the W21 canister. For a given fuel burnup level, assembly radiation sources increase with decreasing initial enrichment. Table 3 lists two minimum initial enrichment values for each assembly burnup level. Table 3 also lists two different minimum allowable cooling times, corresponding to the two minimum initial enrichment levels. An assembly must have an initial enrichment level equal to or greater than the value shown in Table 3, to qualify for the corresponding minimum allowable cooling time also shown in Table 3. Assemblies with initial enrichment levels lower than the lowest values shown (for the assembly's burnup level) in Table 3 are not qualified for transportation in the W21 canister.

Table 3 also gives limits on the total amount of initial (pre-irradiation) cobalt that may be present in the assembly active fuel zone (including both assembly and control insert hardware). For assemblies with less than 11 grams of cobalt in the fuel zone, the shorter cooling times shown in Table 3 may be used (provided that the minimum initial enrichment requirement is also met). The longer cooling times shown in Table 3 must be used for assemblies with over 11 grams of cobalt in the fuel zone. Cobalt present in control components that do not extend into the assembly fuel zone (such as thimble plug assemblies) or that do not reside in the core during operation (such as control rod assemblies) do not need to be included in the total fuel zone cobalt content.

All PWR SNF assembly control inserts placed in the W21 canister must be intact, and may contain B_4C, borosilicate glass, silver-indium-cadmium, hafnium, or Gd_2O_3 poison materials. Control insert rod cladding, and other insert hardware may consist of any type of zircaloy, stainless steel, or inconel. Any PWR assembly control insert that meets these material requirements may be loaded into the W21 canister. Control inserts that employ solid inconel rods that reside in the core, such as the B&W Grey APSRA, are not qualified for transportation in the W21 canister. Any insert that contains significant quantities of inconel (such as inconel rod cladding) requires an evaluation of total assembly fuel zone cobalt quantity. Fuel rods may also be replaced with solid steel or Inconel rods, or rods containing any of the above poison materials, provided that the fuel zone cobalt requirements are met. UO_2 fuel rods containing Gd_2O_3 poison material are also permissible, although the poison is not relied upon to increase allowable ^{235}U initial enrichment levels for the fuel rod or assembly in question.

(ii) W74 Canister

The W74 canister contents are limited to 64 Big Rock Point (BRP) SNF assemblies without channels, including intact, partial, and damaged UO_2 and mixed oxide (MOX) fuel assemblies meeting the applicable acceptance criteria specified in Table 4 through Table 9. Specifications W74-1 and W74-2 for intact UO_2 and MOX fuel assemblies are provided in Table 4 and Table 5, respectively. Specifications W74-3 and W74-4 for partial UO_2 and MOX fuel assemblies are provided in Table 6 and Table 7, respectively. Lastly, specifications W74-5 and W74-6 for damaged UO_2 and MOX fuel assemblies are provided in Table 8 and Table 9, respectively. All UO_2 rods may contain any quantity of Gd_2O_3 poison material, provided that the specified ^{235}U initial enrichment limits are satisfied. BRP assemblies containing any amount of plutonium fuel (before irradiation) must meet the requirements of the MOX fuel specifications given in Table 5, Table 7, or Table 9. If less than the maximum number of BRP assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the BRP assemblies they are replacing, must be loaded in the empty guide tubes or support tubes.

The BRP UO_2 fuel assembly types permitted in the W74 canister are identified in Table 10. Any BRP fuel assemblies that do not meet all of the parameter requirements given for any fuel assembly class in Table 10 may only be loaded into the W74 canister damaged fuel can, as long as the requirements given in the applicable damaged fuel loading specification (W74-5 or W74-6) are still met. Any BRP fuel assemblies that meet all of the parameter requirements shown in Table 10, except for the requirement for the number of non-corner water holes, are classified as partial assemblies. The lower initial enrichment limits given in Specification W74-3 apply for those assemblies.

The specific BRP intact MOX fuel assembly types accommodated in the W74 canister are shown in Figure 1 through Figure 4. The specific BRP partial MOX fuel assembly types accommodated in the W74 canister are shown in Figure 5 through Figure 8. These figures show the maximum initial ^{235}U enrichment levels for the uranium present in all UO_2 and MOX fuel rods in each MOX assembly array. The figures also show the maximum overall weight percentage of PuO_2 in the initial MOX fuel rod (metal-oxide) material composition, with one exception. For the two MOX rods shown in Figure 4, the maximum total plutonium (metal) content, rather than the maximum overall weight percent of PuO_2, is specified. The limits on maximum burnup, maximum heavy metal loading, and minimum cooling time for each BRP MOX fuel type are shown in Table 11.

Table 1 - Generic Requirements for All W21 Canister PWR SNF Contents

Fuel Assembly Parameter	Requirement
Fuel Rod Cladding Material	Zircaloy 2, 4
Assembly Condition	Intact (1)
Maximum Assembly Width (inch)	8.54
Maximum Burnup Level (MWd/MTU)	60,000 (2)
Maximum Uranium Loading (MTU/assy)	0.471
Axial Uranium Loading (kg/assy-inch)	3.27
Maximum Fuel Zone Height (inch)	150
Maximum Fuel Pellet Stack Density	96.5% (3)
Minimum Bottom Nozzle Height (inch)	1.97 (4)

Notes:

(1) Intact assemblies have no known or suspected fuel rod cladding defects greater than pinhole leaks and hairline cracks. Intact fuel also has no detectable grid spacer damage, or axial shifting in grid spacer location. Fuel assemblies with missing fuel rods (from the standard rod array configuration) may be loaded if all missing fuel rods are replaced with dummy rods that have a height and diameter at least as great as that of a standard fuel rod (i.e., by rods that displace an equal or greater volume of water).

(2) For assembly burnups exceeding 45,000 MWd/MTU, it is necessary to verify that the cladding oxide layer thickness does not exceed 70 μm, by measurement of a statistical sample of limiting fuel assemblies. The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

(3) Defined as the average material density within the cylindrical envelope volume covered by the fuel pellets, relative to the theoretical UO_2 density of 10.97 g/cc. Thus, "smearing" over fuel pellet dishes and chamfers to determine the "stack" density is acceptable.

(4) The bottom nozzle height is defined as the distance between the assembly bottom and the bottom of the active fuel.

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Table 2 - W21 Canister SNF Assembly Dimensional and Enrichment Limits

Fuel Assembly Class(1)	Criticality Class(1)	Max. Initial Enrichment (w/o ^{235}U)(2)		Number of Fuel Rods	Min. Clad O.D. (in.)	Min. Clad Thickness (in.)	Min. Pellet Diameter (in.)	Fuel Rod Pitch (in.)	No. Guide / Instrument Tube Locations(5)
		W21-1(3)	W21-2(4)						
B&W 15x15	B&W 15x15	4.70	5.00	208	0.4300	0.0265	0.3675	0.568	17
B&W 17x17	B&W 17x17	4.60	4.90	264	0.3770	0.0220	0.3232	0.502	25
CE 14x14	CE 14x14	5.00	5.00	176	0.4400	0.0260	0.3700	0.580	5(6)
	CE 14x14 A	5.00	5.00	176	0.4400	0.0260	0.3795	0.568	5(6)
Palisades	CE 15x15 P	5.00	5.00	208 - 216	0.4135	0.0240	0.3500	0.550	1-9
Yankee Rowe	15x16	5.00	5.00	231	0.3650	0.0240	0.3105	0.472	1
	15x16 A	5.00	5.00	237	0.3650	0.0240	0.3105	0.468	1
CE 16x16 CE System 80 St. Lucie 2	CE 16x16	5.00	5.00	236	0.3820	0.0250	0.3250	0.506	5(6)
WE 14x14	WE 14x14	5.00	5.00	179	0.4000	0.0243	0.3444	0.556	17
WE 15x15	WE 15x15	4.70	5.00	204	0.4200	0.0240	0.3569	0.563	21
	WE 15x15 A	4.90	5.00	204	0.4240	0.0300	0.3565	0.563	21
WE 17x17	WE 17x17	4.70	5.00	264	0.3740	0.0225	0.3195	0.496	25
	WE 17x17 A	4.60	4.90	264	0.3600	0.0225	0.3088	0.496	25
	WE 17x17 B	4.60	4.90	264	0.3600	0.0250	0.3030	0.496	25

Notes:

(1) Assembly class defined per Energy Information Administration, *Spent Nuclear Fuel Discharges from U.S. Reactors 1993*, U. S. Department of Energy, 1995. The fuel assembly criticality classes are arbitrary designations given to each set of assembly parameters that are evaluated for criticality.

(2) The maximum allowable enrichments apply for all assemblies that meet the specified physical parameter requirements for the defined assembly class. The maximum allowable enrichments are defined as the maximum planar average enrichment at any axial assembly location. An exception is the CE 15x15 P assembly class, for which the maximum allowable enrichment applies to each individual fuel pin within the assembly.

(3) This enrichment limit applies for up to 21 SNF assemblies, in any W21 canister guide tube.

(4) This enrichment limit applies for up to 20 SNF assemblies, with the center guide tube empty.

(5) Whereas the number of guide tube locations is a specified parameter, the materials and dimensions of the guide tubes are not specified, since any quantity of steel or zircaloy in the guide tube locations will reduce assembly reactivity. Guide tube locations may contain nothing, hollow zircaloy or stainless rods (or rod clusters), solid zircaloy or stainless rods (or rod clusters), or poison rods (or rod clusters).

(6) The CE 14x14 and CE 16x16 assembly guide tubes occupy four fuel rod locations within the assembly array.

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Table 3 - W21 Canister Minimum PWR Assembly Cooling Time Requirements

Assembly Burnup Level (GWd/MTU)(1)	Assembly Initial Enrichment (w/o ^{235}U)(1)	Assembly Fuel Zone Cobalt Qty (g/assy)(2)	Required Cooling Time (years)
≤35	≥2.8 %	≤ 11	≥ 6
≤40	≥3.0 %	≤11	≥ 8
≤45	≥3.3 %	≤11	≥ 10
≤50	≥3.5 %	≤11	≥ 12
≤55	≥3.8 %	≤11	≥ 15
≤ 60	≥4.0 %	≤11	≥18
≤35	≥1.5 %	≤50	≥15
≤40	≥1.5 %	≤50	≥20
≤45	≥1.5 %	≤50	≥25
≤50	≥2.5 %	≤50	≥25
≤55	≥3.0 %	≤ 50	≥25
≤60	≥3.5 %	≤50	≥25

Notes:

(1) Assembly average values.
(2) Defined as the total initial (pre-irradiation) cobalt mass within the assembly fuel zone, including any cobalt present in inserted control components.

Table 4 - W74 Canister Contents Specification W74-1
Intact UO_2 Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact UO_2 fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-2 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment(4)	≤4.10 w/o ^{235}U.
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U.
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.(5)

<u>W74-1 Notes:</u>

(1) Loaded assemblies must meet all of the assembly geometry requirements specified in Table 10, for any one of the defined assembly classes.

(2) Intact fuel assemblies include those BRP fuel assemblies with 1 to 4 corner rods missing, and BRP 9x9 fuel assemblies with 1 rod missing from a non-corner location. This includes assemblies with partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations. It also includes 9x9 assemblies with 11x11 assembly rods in corner locations.

(3) Intact UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. The empty array or guide tube locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component.

(4) Defined as the maximum array-average enrichment, which is the peak planar average initial enrichment considering all elevations along the assembly axis.

(5) If an intact UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 5 - W74 Canister Contents Specification W74-2
Intact MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact MOX fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-1 and W74-3 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location in the MOX assembly array in Figure 1 through Figure 4.(4,5)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.(6)

W74-2 Notes:

(1) Intact MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. They may also have hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component placed in the empty array or guide tube locations, including all forms of inserts or control components.

(2) J2 (Figure 1) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 9x9 fuel in Table 10. DA and G-Pu (Figure 2 and Figure 3, respectively) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 11x11 fuel in Table 10. One exception is that J2 MOX assemblies with a cladding thickness of 0.05 inches and a fuel pellet diameter of 0.4515 inches are also acceptable. UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) must meet all of the assembly geometry requirements shown for Siemens 9x9 in Table 10.

(3) Intact G-Pu MOX assemblies may have 0 to 4 fuel rods in the array corner locations. G-Pu MOX assemblies may also have partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations.

(4) The maximum ^{235}U enrichment shown in Figure 1 through Figure 4 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have 235U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 1 through Figure 4 for that fuel rod array location.

(5) Figure 4 specifies a maximum total MOX fuel rod plutonium metal mass as opposed to a maximum PuO_2 weight percentage.

(6) If an intact MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 6 - W74 Canister Contents Specification W74-3 Partial UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤64 Big Rock Point BWR partial UO_2 fuel assemblies.[1,2] Partial fuel assemblies are defined as those assemblies having one or more full-length fuel rods missing from the intact fuel assembly array (except as permitted by W74-1 Notes 2 and 3). The affected array locations may contain nothing, partial length rods, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component with a lower length or diameter than a full-length fuel rod. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1, W74-2, and W74-4 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly
Maximum Initial Enrichment[3]	≤3.55 w/o ^{235}U (9x9) ≤3.6 w/o ^{235}U (11x11)
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU
Minimum Cooling Time	≥6.0 years[4]

W74-3 Notes:

(1) Partial UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods.

(2) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(3) Defined as the maximum array average initial enrichment, which is the peak planar average initial enrichment considering all elevations along the fuel assembly axis. The averaging is applied only to those fuel rods that are present in the partial array.

(4) If a partial UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 7 - W74 Canister Contents Specification W74-4
Partial MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR partial MOX fuel assemblies.(1,2,3) Partial MOX assemblies must conform exactly to one of the four partial assembly array configurations shown in Figure 5 through Figure 8, with respect to the number and location of missing fuel rods within the assembly array. The missing fuel rod array locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-3, W74-5, and W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by fuel assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location (in each of the four allowable partial MOX assembly array configurations) in Figure 5 through Figure 8.(4)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.

<u>W74-4 Notes:</u>

(1) Partial MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod.

(2) If a partial MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

(3) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(4) The maximum ^{235}U enrichment shown in Figure 5 through Figure 8 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have ^{235}U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 5 through Figure 8 for that fuel rod array location.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9276	4	71-9276	USA/9276/B(U)F-85	14	OF	27

Table 8 - W74 Canister Contents Specification W74-5
Damaged UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤8 Big Rock Point BWR damaged UO_2 fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where fuel rod structural integrity cannot be assured, or where grid spacers have moved from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a basket support tube in the upper or lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-4 and W74-6, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial UO_2 fuel assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.
Cladding Material/Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment	≤4.61 w/o ^{235}U peak fuel pellet initial enrichment.
Maximum Pellet Density	≤96.5% (as defined in Table 10, Note 1).
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.[(1)]

W74-5 Note:

(1) If a damaged UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 9 - W74 Canister Contents Specification W74-6
Damaged MOX Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤ 8 Big Rock Point BWR damaged MOX fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where the fuel rod structural integrity cannot be assured, or where the grid spacers have shifted vertically from their design position) are also considered to be damaged fuel assemblies.Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a support tube locations in the upper and lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-5, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial MOX assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.(1)
Cladding Material/ Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Pellet Density	96.5% (as defined in Table 10, Note 1)
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	≤4.61 w/o ^{235}U for all UO_2 fuel pellets. All MOX fuel pellets must meet the maximum ^{235}U enrichment and PuO_2 weight percentage requirements for one of the four MOX fuel material compositions described in Figure 1 through Figure 3.
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.(2)

W74-6 Notes:

(1) The UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) may not be loaded into the W74 damaged fuel can.

(2) If a damaged MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 10 - W74 Canister Fuel Geometry Specifications

Fuel Assembly Parameter	Fuel Assembly Class			
	GE 9x9	**Siemens 9x9**	**Siemens 11x11**	**Siemens 11x11A**
Fuel Pellet Stack Density(1)	≤ 96.5%	≤96.5%	≤96.5%	≤96.5%
Number of Fuel Rods	≤ 81	≤ 81	≤121	≤ 121
Clad O.D. (in)	0.5625	0.5625	0.449	0.449
Clad Thickness (in)	0.040	0.040	0.034	0.034
Pellet Diameter (in)	0.471	0.4715(2)	0.3715	0.3735
Fuel Rod Pitch (in)	0.707	0.707	0.577	0.577
Active Fuel Length (in)	≤70	≤70	≤70	≤ 70
Number of Array Corner Rods(3)	0-4	0-4	0-4	0-4
Number of Non-Corner Water Holes(3)	≤ 1	0	0	0
Number of Inert Rods(3)	≥0	≥ 0	≥0	≥0
Bottom Tie Plate Height (in)(4)	≥1.25	≥1.25	≥1.25	≥1.25

<u>Notes:</u>

(1) The fuel pellet stack density is defined as the average density of the fuel pellet material (within the cylindrical envelope volume covered by the pellet stack) divided by the theoretical UO_2 density of 10.97 g/cc. Thus, smearing the fuel material over the dishing and chamfer voids in the pellet stack is acceptable for determining the stack density.

(2) Assemblies E65 and E72 may each contain two MOX fuel rods with either solid pellets or annular pellets with a 0.1 inch or 0.2 inch inside diameter. In any given MOX fuel rod, the entire pellet stack must contain the same pellet type (i.e., solid, 0.1-inch annular, or 0.2-inch annular).

(3) The definitions of corner rods, non-corner rods, and inert rods are given in the W74-1 and W74-3 assembly loading specifications.

(4) Defined as the distance from the bottom of the assembly to the bottom of the active fuel.

Table 11 - W74 Canister Assembly Specific Requirements for Big Rock Point MOX Fuel

BRP MOX Assembly Type	Maximum Heavy Metal Loading (kg)	Maximum Burnup (MWd/MTIHM)(1)	Minimum Cooling Time (years)
J2 (9x9)	124	22,820	22
DA (11x11)	126	21,850	22
G-Pu (11x11)	127	34,220	15
UO_2 9x9 with 2 inserted MOX rods	142.1	32,000	6

Note:

(1) The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

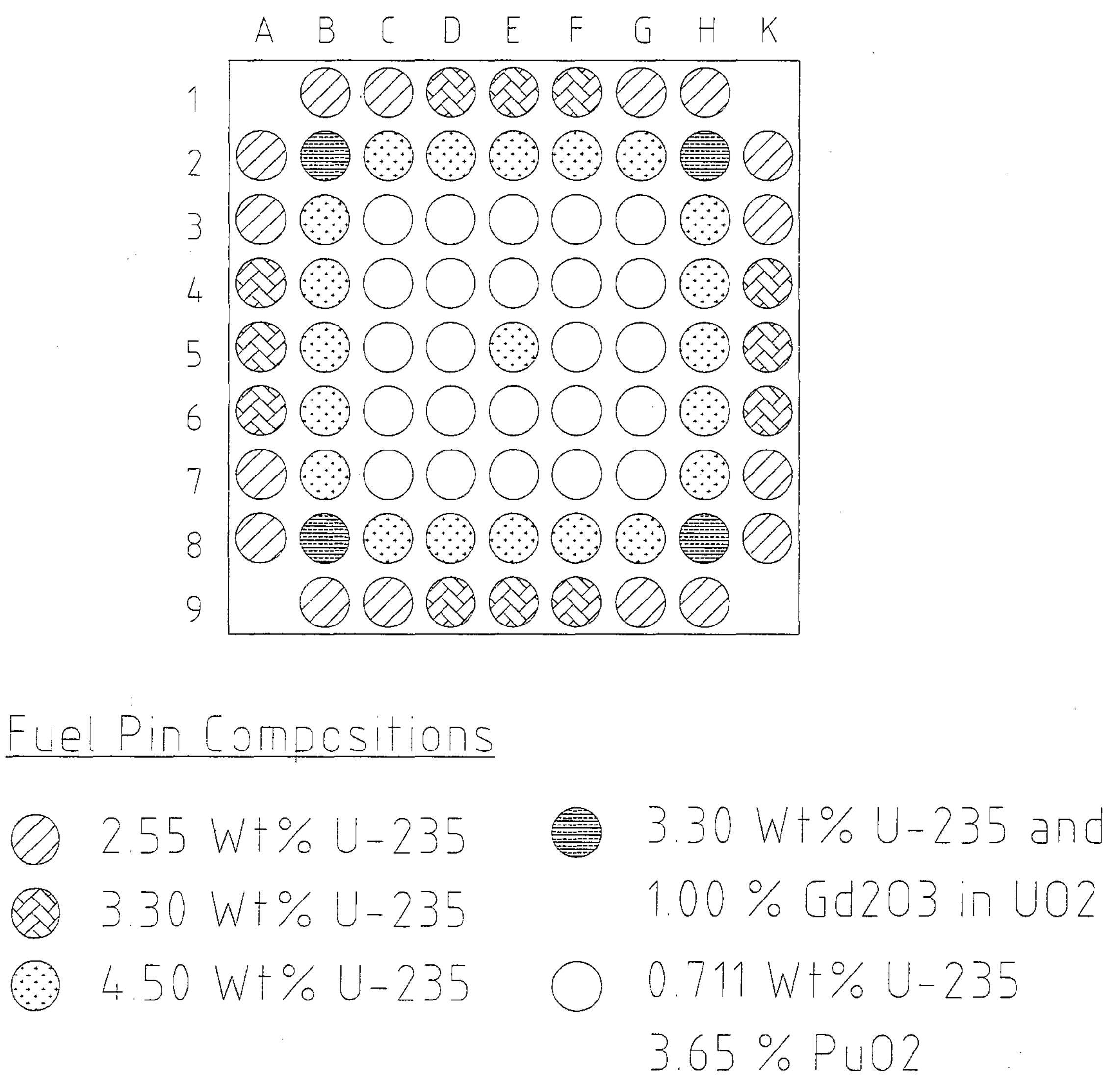
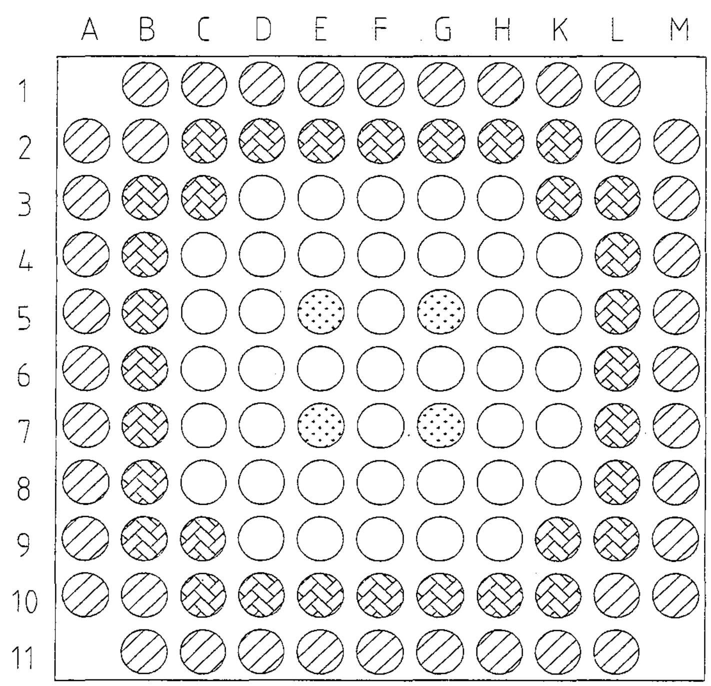


Figure 1 - J2 (9x9) BRP MOX Assembly Array



Fuel Pin Compositions

2.40 Wt% U-235

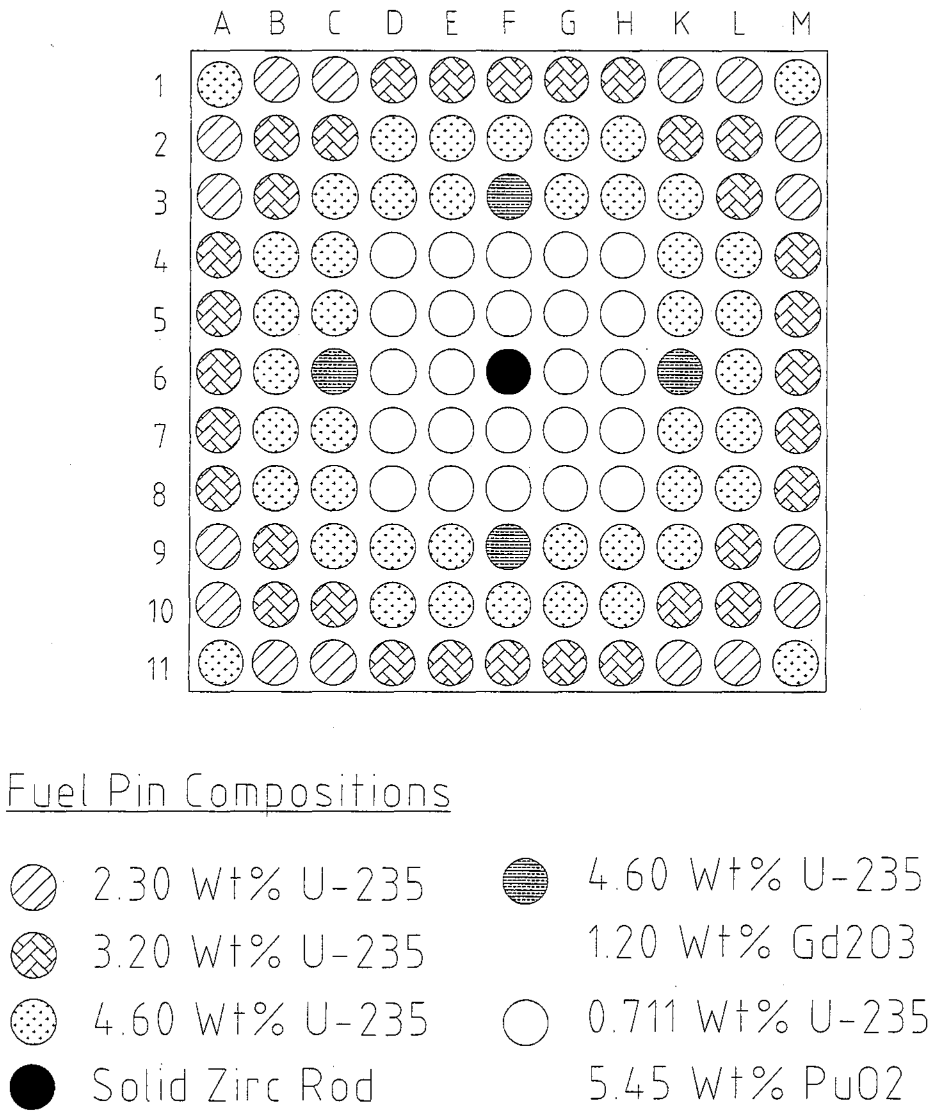
1.56 Wt% U-235
1.03 Wt% PuO2

2.40 Wt% U-235
2.45 Wt% PuO2

Water Rods

Note: Water rods are identical to the fuel rods (same diameter and cladding thickness), except that they contain no fuel pellets.

Figure 2 - DA (11x11) BRP MOX Assembly Array



Note: G-Pu assemblies may have any number of fuel rods missing (or present) in the four array corner locations

Figure 3 - G-Pu (11x11) BRP MOX Assembly Array

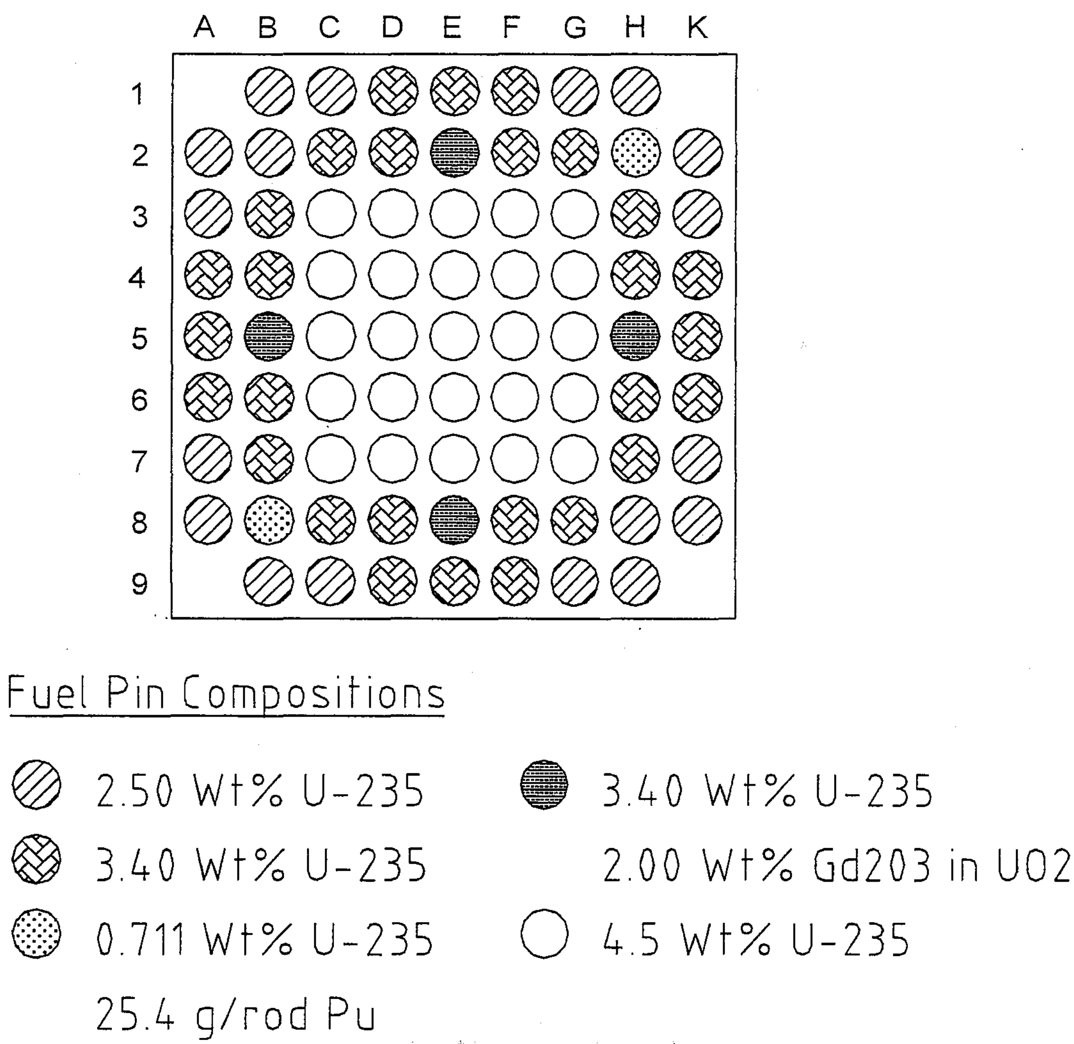


Figure 4 - UO2 9x9 BRP Assembly with Two Inserted MOX Rods

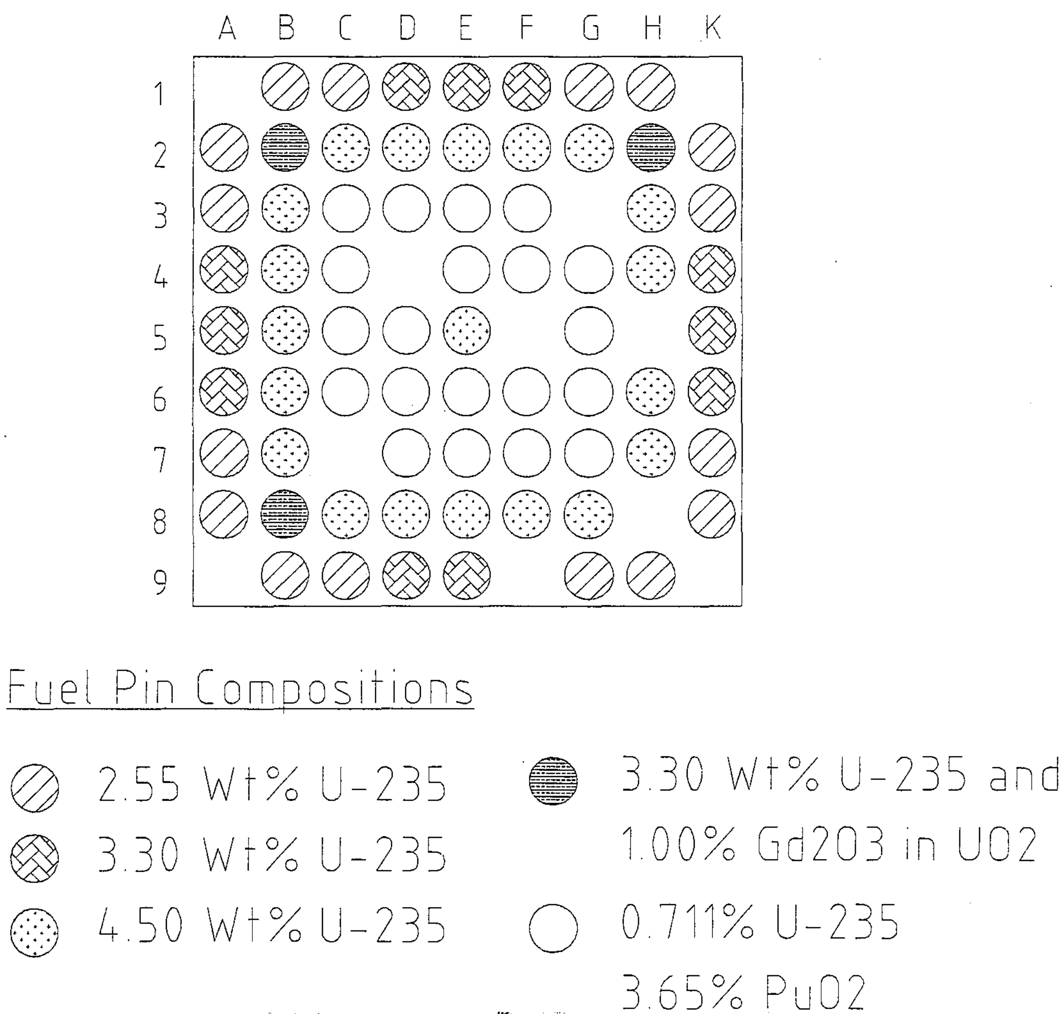


Figure 5 - J2 Partial MOX Assembly Array #1

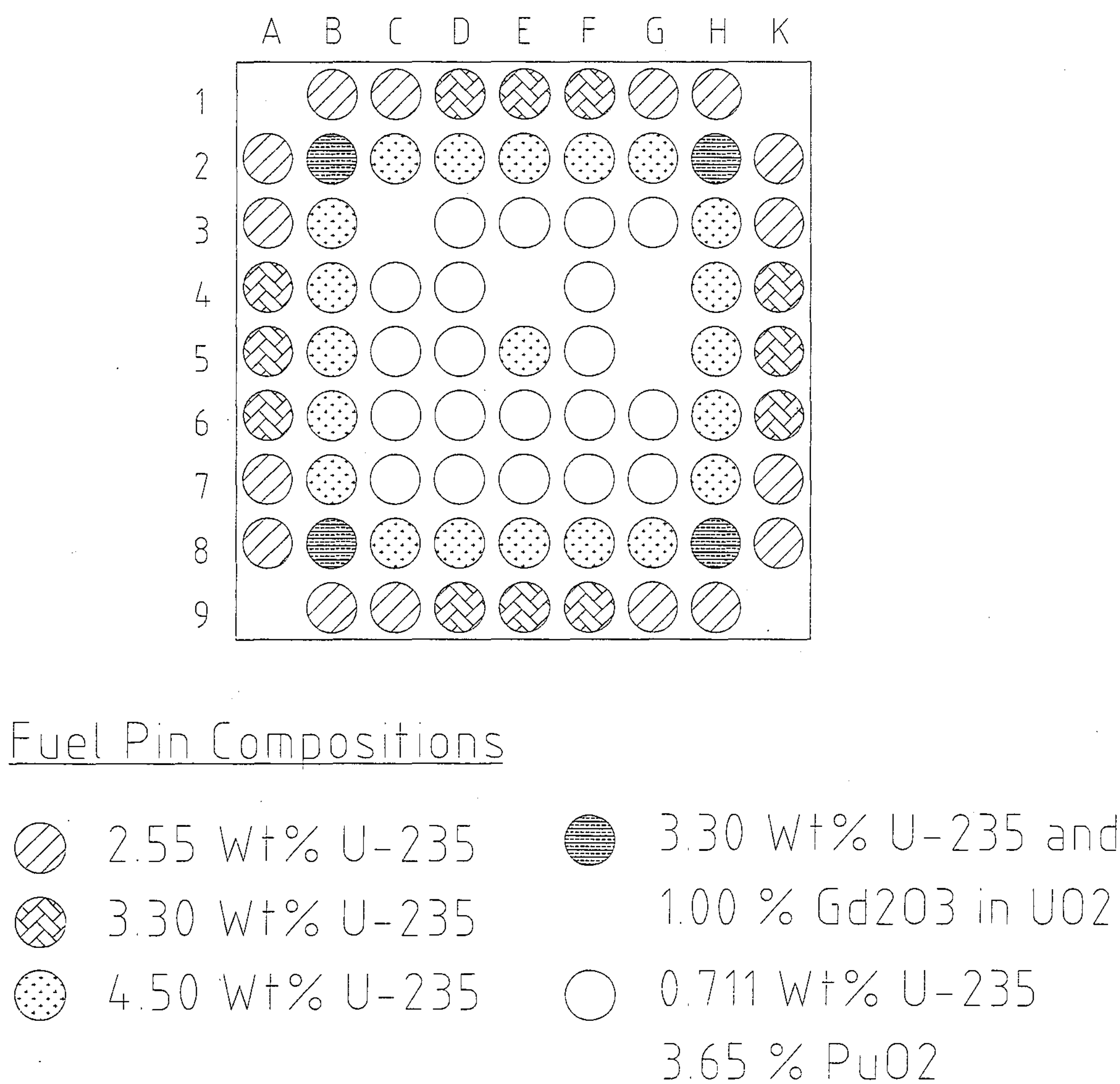


Figure 6 - J2 Partial MOX Assembly Array #2

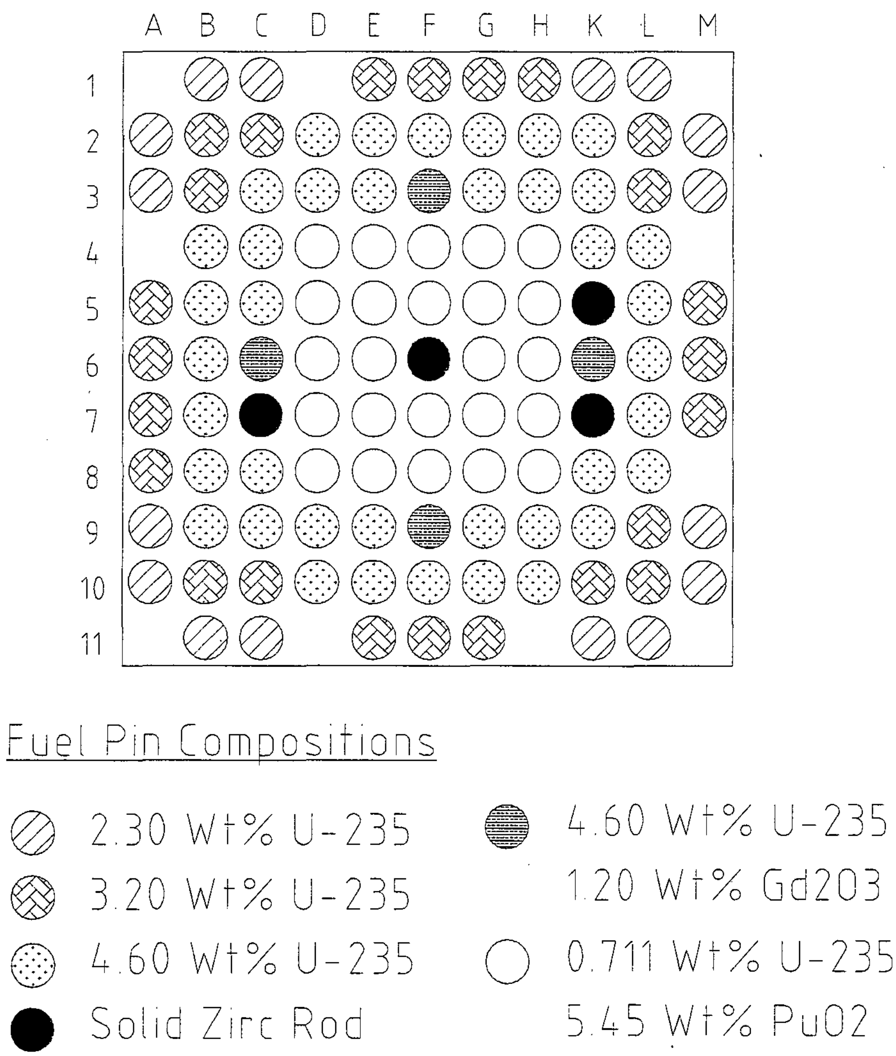


Figure 7 - G-Pu Partial MOX Assembly Array #1

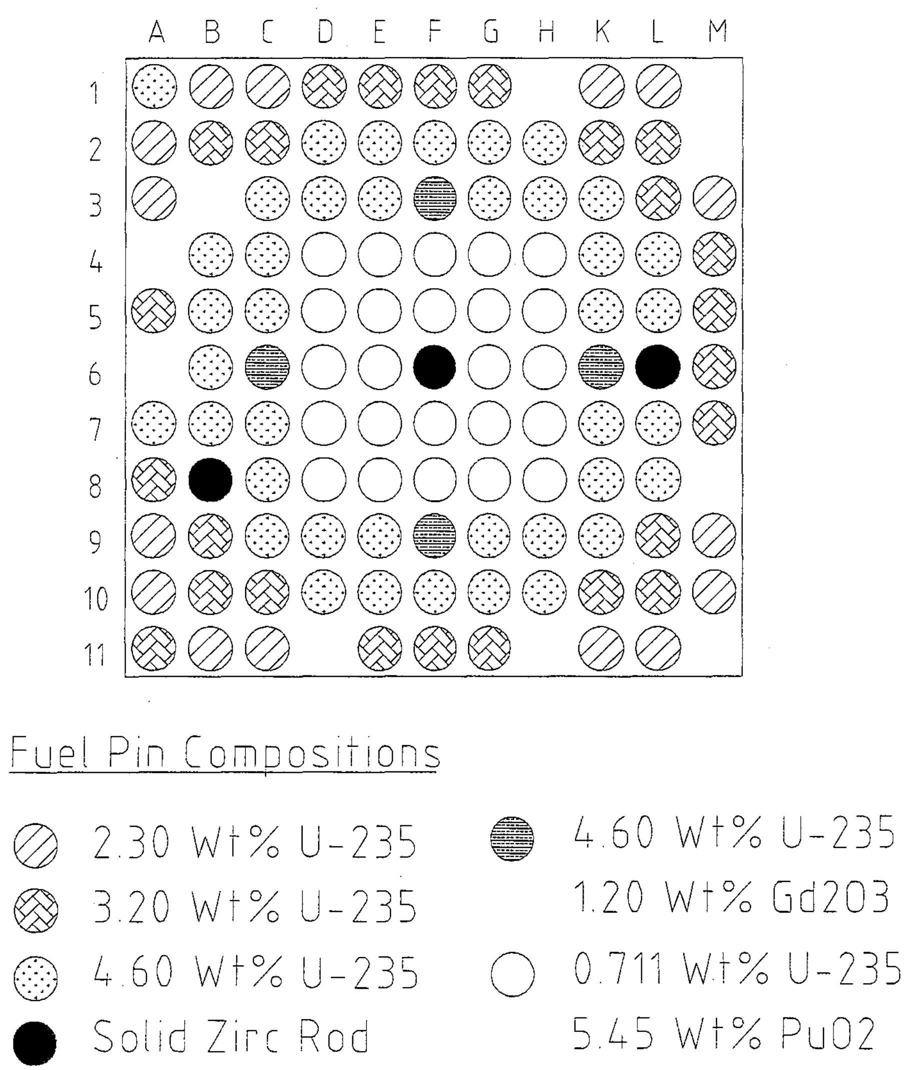


Figure 8 - G-Pu Partial MOX Assembly Array #2

(2) Maximum Quantity of Material Per Package

The maximum payload weight of the TS125 Transportation cask is 85,000 pounds. The payload weight includes the weight of the FuelSolutions™ canister and its SNF payload, plus the weight of the cask cavity spacer for short canisters.

(3) Decay Heat Limit
The W74 canister loading criteria can be described as follows:

A Big Rock Point spent fuel assembly is allowed to be shipped in the canister if Q (heat generation per assembly) ≤ 0.275 kW.

No decay heat limit is specified for the W21 canister. The PWR assembly fuel parameters requirements given in Table 3 ensure that assembly heat generation levels will not exceed the heat generation level that was analyzed in the thermal licensing evaluations (1.05 kW/assembly).

(c) Criticality Safety Index

(Minimum transport index
to be shown on label for
nuclear criticality control): 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(1) The package shall meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

(2) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17, provided the fabrication of the packaging was satisfactorily completed by December 31, 2006.

8. Transport by air of fissile material is not authorized.

9. Expiration date: October 31, 2017

10. Fabrication of new packagings is not authorized.

REFERENCES

BNFL Fuel Solutions Corporation, application dated April 20, 2001.

Supplements dated June 7, 2001; January 22, February 5, February 28, April 11, and April 30,2002; January 17, August 7, and November 26, 2003; and April 20, April 28, April 29, May 7, May 12, 2004, August 27, 2007, and September 10, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Bernard H White IV for

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 10/26/12

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9279	7	71-9279	USA/9279/B(U)F-96	1 OF	5

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Department of Energy
Washington, DC 20586

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Waste Partnership, LLC application dated March 27, 2013.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HalfPACT Waste Shipping Container

(2) Description

A stainless steel and polyurethane foam insulated shipping container designed to provide single containment for shipment of contact-handled transuranic waste. The packaging consists of an unvented, 1/4-inch thick stainless steel inner containment vessel (ICV), positioned within an outer confinement assembly (OCA) consisting of an unvented 1/4-inch thick stainless steel outer confinement vessel (OCV), an approximate 8-inch thick layer of polyurethane foam, a 1/4-inch thick layer of ceramic fiber paper and a 1/4 to 3/8-inch thick outer stainless steel shell. The package is a right circular cylinder with outside dimensions of approximately 94 inches diameter and 92 inches height. The package weighs not more than 18,100 pounds when loaded with the maximum allowable contents of 7,600 pounds.

The OCA has a domed lid which is secured to the OCA body with a locking ring. Although not part of the containment boundary, the OCV confinement seal is provided by an optional butyl rubber O-ring. The OCV is equipped with a seal test port and a vent port.

The ICV is a right circular cylinder with domed ends. The outside dimensions of the ICV are approximately 74 inches diameter and 69 inches height. The ICV lid is secured to the ICV body with a locking ring. The ICV containment seal is provided by a butyl rubber O-ring. The ICV is equipped with a seal test port and vent port. Aluminum spacers are placed in the top

5.(a)(2) Description (continued)

and bottom domed ends of the ICV during shipping. The cavity available for the contents is a cylinder of approximately 73 inches diameter and 45 inches height.

5.(a)(3) Drawings

The package is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing 707-SAR, sheets 1-12, Rev. 9. The standard pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-001, sheets 1-3, Rev. 7. The S100 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-002, sheets 1 and 2, Rev. 5. The S200 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-003, sheets 1 and 2, Rev. 4. The S300 pipe overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-004, sheet 1, Rev. 2. The compacted puck drum spacers needed for the purpose of maintaining subcriticality in 55-, 85-, and 100-gallon drums are constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-006, Rev. 1. The shielded container is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-008, sheets 1–6, Rev. 2. The criticality control overpack is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawing No. 163-009, sheets 1-2, Rev. 0.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, standard waste box (SWB), 85-gallon drum, standard pipe overpack, S100 pipe overpack, S200 pipe overpack, S300 pipe overpack, 100-gallon drum, shielded container, or criticality control overpack (CCO). The payload containers are described in Section 2.9, "Payload Container/Assembly Configuration Specifications," of the CH-TRAMPAC, Rev. 4. Explosives, corrosives (pH less than 2 or greater than 12.5), nonradioactive pyrophorics, and compressed gases are prohibited. Within a payload container radioactive pyrophorics must not exceed 1 weight percent by weight and residual liquids must not exceed 1 percent by volume. Flammable organics and methane are limited along with hydrogen to ensure the absence of flammable gas mixtures in TRU waste payloads as described in Chapter 5.0 of the CH-TRAMPAC, Rev. 4. For payloads of content code LA 154 and SQ 154, the absence of flammable gas mixtures is ensured as described in Appendix 6.12 of the CH-TRU Payload Appendices, Rev. 3. For payload configurations with unvented heat-sealed bag layers, the absence of flammable gas mixtures is ensured as described in Appendix 6.13 of the CH-TRU Payload Appendices, Rev. 3. For Analytical Category payload containers containing puck drums, the absence of flammable gas mixtures is ensured as described in Appendix 6.14 of the CH-TRU Payload Appendices, Rev. 3.

(2) Maximum quantity of material per package

The package contents are limited to 7,600 pounds, including the weight of the payload containers and any other components of the payload assembly. The maximum gross weight for a payload container not to exceed the following:

(i) 328 pounds per 6-inch standard pipe overpack,
(ii) 547 pounds per 12-inch standard pipe overpack,
(iii) 550 pounds per S100 pipe overpack,
(iv) 547 pounds per S200 pipe overpack,
(v) 547 pounds per S300 pipe overpack,
(vi) 1,000 pounds per 100-gallon drum,
(vii) 1,000 pounds per 55-gallon drum,
(viii) 1,000 pounds per 85-gallon drum,
(ix) 4,000 pounds per SWB,
(x) 2,260 pounds per shielded container, or
(xi) 350 pounds per CCO.

Maximum number of payload containers per package and authorized packaging configurations as follows:

(i) 7 55-gallon drums,
(ii) 7 standard pipe overpacks,
(iii) 7 S100 pipe overpacks,
(iv) 7 S200 pipe overpacks,
(v) 7 S300 pipe overpacks,
(vi) 4 85-gallon drums,
(vii) 3 100-gallon drums,
(viii) 1 SWB,
(ix) 3 shielded containers, or
(x) 7 CCOs.

Fissile material not to exceed the limits specified in CH-TRAMPAC, Rev. 4, Section 3.1, "Nuclear Criticality." Fissile material in the CCCs shall not be machine compacted and shall not exceed 380 fissile gram equivalent of Pu-239 containing less than or equal to 1% by weight Be/BeO.

All payloads shall meet the activity limits specified in CH-TRAMPAC, Rev. 4, Section 3.3, "Activity Limits." The payload is limited to 10^5 A_2 quantities.

Maximum decay heat per package not to exceed 30 watts. Decay heat per payload container not to exceed the values in Table 5.2-1 of the CH-TRAMPAC, Rev. 4, "List of Approved Alphanumeric Shipping Categories, Maximum Allowable Hydrogen Gas Generation Rates, and Maximum Allowable Wattages," or calculated for approved shipping categories in accordance with the methodology specified in Section 5.2.3 of the CH-

5.(b)(2) Maximum quantity of material per package (continued)

TRAMPAC, Rev. 4. For content code LA 154 and SQ 154 payloads, decay heat per payload container not to exceed the values determined as specified in Appendix 6.12 of CH-TRU Payload Appendices, Rev. 3.

5.(c) Criticality Safety Index: 0.0

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight and center of gravity; and radiation dose rate must be determined and limited in accordance with CH-TRAMPAC, Rev. 4.

7. Each payload container must be assigned to a shipping category in accordance with Section 5.1, "Payload Shipping Category" of CH-TRAMPAC, Rev. 4. Each payload container and payload assembly must not exceed the allowable wattage in accordance with Section 5.2.3, "Hydrogen Gas Generation Rate and Decay Heat Limits for Analytical Category," or must be tested for gas generation in accordance with Section 5.2.5, "Unified Flammable Gas Test Procedure," of CH-TRAMPAC, Rev. 4. For a payload made up of payload containers with different (nonequivalent) shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with CH-TRAMPAC, Rev. 4, Section 6.2.4, "Mixing of Shipping Categories," and Appendix 2.4 of the CH-TRU Payload Appendices, "Mixing of Shipping Categories and Determination of the Flammability Index." For Analytical Category payload drums containing puck drums, the absence of flammable gas mixtures is ensured as described in Appendix 6.14 of the CH-TRU Payload Appendices, Rev. 3. Each content code LA 154 and SQ 154 payload container must be assigned to a shipping category in accordance with Appendix 6.12 of CH-TRU Payload Appendices. Content code LA 154 and SQ 154 payload containers may only be assembled with other payload containers belonging to content code LA 154 and SQ 154, respectively, or dunnage in accordance with Appendix 6.12 of CH-TRU Payload Appendices. For a payload of content code LA 154 or SQ 154 containers with different shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with Appendix 6.12 of CH-TRU Payload Appendices.

8. Payload containers within a package shall be selected in accordance with Section 6.0, "Payload Assembly Requirements," of CH-TRAMPAC, Rev. 4. Payload containers of content code LA 154 and SQ 154 shall be assembled in accordance with Appendix 6.12 of CH-TRU Payload Appendices, Rev. 3.

9. Each payload container must be vented in accordance with Section 2.5, "Filter Vents," of CH-TRAMPAC, Rev. 4. Payload containers which were not equipped with filtered vents during storage must be aspirated in accordance with Section 5.3, "Venting and Aspiration," of CH-TRAMPAC, Rev. 4.

10. For close-proximity and controlled shipments meeting the conditions specified in Appendices 3.5 and 3.6, respectively, of CH-TRU Payload Appendices, shipping periods of 20 days and 10 days

10. (continued)

may be applicable. The shipping period for any mode of transport is not to exceed 60 days. The content code LA 154 and SQ 154 shipments, the shipping period as defined in Appendix 6.12 of the CH-TRU Payload Appendices is not to exceed 5 and 10 days, respectively.

11. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented. For content code LA 154 and SQ 154 payloads, each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0 of the application, as modified by Appendix 6.12 of CH-TRU Payload Appendices.

(b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

(c) All free standing water must be removed from the inner containment vessel cavity and the outer confinement vessel cavity before shipment.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. Revision No. 6 of this certificate may be used until June 30, 2014.

14. Expiration date: October 31, 2015.

REFERENCES

Nuclear Waste Partnership, LLC, application dated March 27, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William C. Allen for

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 14, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9282	2	71-9282	USA/9282/B(U)-96	1	OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Source Production
and Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc. application dated June 28, 1999, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SPEC-300

(2) Description

The SPEC-300 is a radiographic device that consists of a source assembly, a depleted uranium shield, and a stainless steel enclosure. The radioactive source assembly is housed in a zircaloy or titanium "S" tube that is surrounded by the depleted uranium shield. The depleted uranium shield is secured in the stainless steel enclosure. The void space between the depleted uranium shield and the enclosure is filled with high density polyurethane foam. The package is approximately 26 inches long, 14 inches wide, and 15 inches high. The maximum gross weight of the package is 780 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Co., Inc. General Arrangement drawings: 19B000 sheets 1-8, Rev. 4 and B190700 sheet 1, Rev. 3.

(b) Contents

(1) Type and form of material

Cobalt-60 sources which meet the requirements of special form radioactive material.

5.(b) Contents (continued)

(2) Maximum quantity of material per package

11.1 TBq (300 Ci) (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly must be fabricated of materials capable of resisting a 1475 F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented; and

(b) The package must meet the Acceptance Test and Maintenance Program of Chapter 8.0 of the application, as supplemented.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration date: April 30, 2015.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9282	2	71-9282	USA/9282/B(U)-96	3	OF	3

REFERENCES

Source Production and Equipment Company, Inc., application dated June 28, 1999.

Supplements dated: October 6, November 4, November 22, and December 15, 1999; February 29 and March 27, 2000; March 14, 2005; and October 28, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: November 20, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9284	5	71-9284	USA/9284/B(U)F-85	1	OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Eco-Pak Specialty Packaging application dated June 19, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: ESP-30X Protective Shipping Package for 30-inch UF_6 Cylinders

(2) Description

An overpack for the transport of 30-inch enriched uranium hexafluoride (UF_6) cylinders. The shape of the overpack is a right circular cylinder constructed of two 11 gauge carbon steel shells. The area between the shells is filled with fire retardant, phenolic foam per ESP specification ESP-PF-1. The volume between the 1/2" inch thick end plates of the two shells is also filled with phenolic foam. A stepped horizontal joint permits the top half of the overpack to be removed from the base. The horizontal joint of each half of the overpack is covered with steel and a 5/8" thick silicone gasket seals the joint. The overpack halves are secured with ten 3/4" diameter steel bolts and nuts.

The approximate dimensions and weights of the package are as follows:

Outer shell inside diameter	43"
Outer shell length	96"
Inner shell inside diameter	30 7/8"
Inner shell length	82 5/8"
Overpack weight	2,955 pounds
30B Cylinder weight	1,390 pounds
UF_6 maximum load	5,020 pounds
Maximum package gross weight (including contents)	9,365 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with ESP Drawing Nos.:

30X-1 SAR, Rev. 2, Sheets 1-4

5.(b) Contents

(1) Type and form of material

The UF_6 must be packaged in Model 30B UF_6 cylinders which have been fabricated, inspected, tested and maintained in accordance with the requirements of ANSI N14.1. The UF_6, which may contain either virgin or recycled uranium, must not contain more than the following maximum quantities of radionuclides and impurities:

U^{232}	5.0E-09 g/gU
U^{234}	2.0E-03 g/gU
U^{235}	5.0E-02 g/gU
U^{236}	2.5E-02 g/gU
U^{238}	balance of total uranium content
Pu and Np	Alpha activity not exceed 3.3 Bq/gU
Tc^{99}	5.0E-06 g/gU
Th^{228}	1.17E-09 g/gU

Fission Products 4.4 X 10^5 Mev Bq/d kgU (total contribution from gamma emitting fission products); this results in the following individual maximum activities:

Ru^{106}/Rh^{106}	2095 Bq/gU
Ru^{103}/Rh^{103}	885 Bq/gU
$Ce^{144}/Pr^{144}/Pr^{144}$	8349 Bq/gU
Sb^{125}	1030 Bq/gU
Cs^{134}	283 Bq/gU
Cs^{137}/Ba^{137}	778 Bq/gU
Zr^{95}	598 Bq/gU
Nb^{95}	574 Bq/gU

The total concentration of elements that form non-volatile fluorides (including Al, Ba, Bi, Cd, Co, Cr, Cu, Fe, Pb, Li, Mg, Mn, Ni, K, Ag, Na, Sr, Th, Sn, Zn, and Zr) must not exceed 3.0E-03 g/gU.

The contents of other elements must not exceed the following concentrations in g/gU.

Sb<1	As<3	B<1	Bi<5	Cl<100
Cr<10	Nb<1	P<50	Ru<1	Si<100
Ta<1	Ti<1	Mo<1.4	W<1.4	V<1.4

Additionally, for reprocessed UF_6, the maximum total activity present in the package is limited to 957 mixture A_2 values.

(2) Maximum quantity of material per package

The package contents are limited to a maximum of 5,020 pounds of UF_6 enriched to not more than 5 wt%U^{235}. The maximum H/U atomic ratio for the UF_6 is 0.088.

5. (c) Criticality Safety Index 5.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(2) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(3) The package shall be maintained in accordance with the Maintenance Program of Chapter 8 of the application.

7. The 30-inch diameter UF_6 cylinder must be inspected, tested and maintained in accordance with American National Standard N14.1-1995 or latest revision.

8. The 30-inch diameter UF_6 cylinder valve stem and plug may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

9. The leak tightness of the 30B UF_6 cylinder shall be verified using a test having a sensitivity of at least 1 x 10^{-3} std-cc/sec per ANSI Standard N14.5-1997 prior to loading into the ESP-30X overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Fabrication of new packaging is not authorized.

12. Revision No. 4 of this certificate may be used until May 31, 2011.

The contents of other elements must not exceed the following concentrations in g/gU.

Sb<1	As<3	B<1	Bi<5	Cl<100
Cr<10	Nb<1	P<50	Ru<1	Si<100
Ta<1	Ti<1	Mo<1.4	W<1.4	V<1.4

Additionally, for reprocessed UF_6, the maximum total activity present in the package is limited to 957 mixture A_2 values.

(2) Maximum quantity of material per package

The package contents are limited to a maximum of 5,020 pounds of UF_6 enriched to not more than 5 wt%U^{235}. The maximum H/U atomic ratio for the UF_6 is 0.088.

5. (c) Criticality Safety Index 5.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(2) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(3) The package shall be maintained in accordance with the Maintenance Program of Chapter 8 of the application.

7. The 30-inch diameter UF_6 cylinder must be inspected, tested, and maintained in accordance with American National Standard N14.1-1995.

8. The 30-inch diameter UF_6 cylinder valve stem and plug may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

9. The leak tightness of the 30B UF_6 cylinder shall be verified using a test having a sensitivity of at least 1 x 10^{-3} std-cc/sec per ANSI Standard N14.5-1997 prior to loading into the ESP-30X overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Fabrication of new packaging is not authorized.

12. Revision No. 4 of this certificate may be used until May 31, 2011.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9284	5	71-9284	USA/9284/B(U)F-85	4	OF	4

13. Expiration date: May 31, 2015.

REFERENCES

ESP application dated June 19, 1998.

Supplements dated: August 27, 1999; March 22, May 12, and May 18, 2000; April 11, 2002; January 28, and April 12, 2005; March 24, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date April 6, 2010.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9285	3	71-9285	USA/9285/AF-85	1	OF	2

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Global Nuclear Fuel - Americas, L.L.C.
Mail Code K-84
3901 Castle Hayne Road
Wilmington, NC 28401

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Electric Company application dated August 4, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SRP-1

(2) Description

A steel drum for the transport of solid uranium contaminated residues. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is 825 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with General Electric Company Drawing. No. 0025E98, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium-contaminated solid residues.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGE
9285	3	71-9285	USA/9285/AF-85	2 OF	2

5. (b) Contents (Continued)

(2) Maximum quantity of material per package: 775 pounds.
The maximum uranium enrichment is 5.0 weight percent U-235. The maximum fissile mass is 104 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. In addition, the uranium may not exceed 0.05 weight percent U-234 and 0.025 weight percent U-236.

(c) Criticality Safety Index (CSI): 0.6

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application.

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17.

8. Air transport of fissile material is not authorized.

9. Revision No. 2 of this certificate may be used until October 31, 2009.

10. Expiration date: October 31, 2013.

REFERENCES

General Electric Company application dated August 4, 1998.

Supplements dated: October 2, 1998; October 14, 1999; August 6, 2003; and October 8, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: Oct 24, 2008

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9287	4	71-9287	USA/9287/B(U)-85	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA Federal Services, LLC
505 S. 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated November 18, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SteriGenics Eagle

(2) Description

A stainless steel, lead shielded shipping cask for special form cobalt-60 sealed sources. The package consists of a cylindrical cask body with closure lid, and removable toroidal impact limiters, and a basket that carries and positions the cobolt-60 sealed source capsules. The packaging is constructed primarily of ASTM Type 304 stainless steel. The package is designed to transport up to 330,000 curies of cobalt-60.

The outside diameter of the cask body is approximately 37-11/16 inches. The diameter of the inner cavity is approximately 10-3/4 inches. The stainless steel inner shell has a minimum thickness of 1 inch and the stainless steel outer shell is 1 inch thick. The region between the two shells is filled with lead shielding. The closure lid and cask bottom end each consist of two stainless steel plates with lead between the two plates. The lead shielding thickness is approximately 10-3/8 inches on the side, 14-3/8 inches in the closure lid, and 11-7/8 inches on the cask bottom. The closure lid is secured by 12, 3/4-inch bolts. The closure lid is equipped with a Viton O-ring seal. The lid has a drain port and a vent port, and the cask body has a drain port. Each port is closed by a plug.

A double stainless steel thermal radiation shield is provided on the outside of the cask body in the region between the two impact limiters. The inner thermal shield is about 3/4-inches

5.(a) (2) Description (continued)

thick and is radially separated from the cask outer shell by 12 gauge spacers at each end. The outer shield is a sheet of 10 gauge material separated from the inner shield by a spiral wrap of 12 gauge wire.

The top and bottom impact limiters are toroidal stainless steel shells. They are attached to either end of the cask body using 12, 1-inch diameter ball-lock pins orientated radially around the cask body. One pin on each limiter is installed with a lockwire to provide a tamper-indicating device.

The cask lifting attachments thread into the upper cask body. The cask lid is also equipped with removable lid-lifting attachments. The cask rests on a steel pallet and is held down to the pallet by means of a steel frame placed on the top impact limiter. This steel frame is used to tie the cask to the conveyance. The maximum weight of the package, including contents is 20,000 lbs.

The approximate dimension and weights of the package are as follows:

Cask Body Outer Diameter	37-11/16 inches
Cask Body Height	49-7/8 inches
Cask Cavity Inner Diameter	10-3/4 inches
Cask Cavity Inner Height	19 inches
Lead Shield Sidewall Thickness	10-3/8 inches
Overall Package Dimension	
Diameter at Impact Limiters	60 inches
Diameter at Body	37-11/16 inches
Height with Impact Limiters	76 inches
Maximum Contents Weight	50 pounds
Maximum Package Weight (Including Contents)	20,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Packaging Technology, Incorporated, Drawing No. 98003-SAR, Rev.1, Sheets 1 through 8.

(b) Contents

(1) Type and form of material

Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.

5.(b) (2) Maximum quantity of material per package:

12,210 terabecquerels (330,000 curies).
Not to exceed 680.8 terabecquerels (18,400 curies) per special form source.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented.

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17, provided the fabrication of the package was satisfactorily completed by December 31, 2006.

8. Expiration date: December 31, 2014.

REFERENCES

Packaging Technology, Inc., application dated November 18, 1998.

Supplements dated: August 20, 1999, November 29, 2004, November 26, 2007, November 12, 2009, and October 31, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 18, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9288	9	71-9288	USA/9288/B(U)F-96	1	OF	6

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: CHT-OP-TU

(2) Description

A shipping container for uranium oxide pellets, powder, and uranium-bearing materials. The package is roughly cubical and is approximately 45-inches x 45-inches x 62-inches high. The package has four internal sleeves in which Oxide Vessels are inserted.

The outer shell of the package is constructed of 11-gauge mild or stainless steel and the space between the outer shell and the sleeves are filled with fire retardant, closed cell phenolic or polyurethane foam.

The sleeves are constructed of 11-gauge mild or stainless steel with an inner diameter of 10-1/4 inches. The sleeves are closed with twelve 1/2-inch-diameter bolts using an outer lid assembly on a 1/16-inch-thick neoprene or silicone gasket. The outer lid assembly is filled with fire-retardant, closed cell phenolic or polyurethane foam.

The Oxide Vessel is constructed of series 300 stainless steel, with an inner diameter of either 6, 7.5, or 8 inches. The Oxide Vessel is closed by eight 1/2-inch-diameter bolts on a 5/8-inch-thick stainless steel lid with a double O-ring seal. The O-ring seal material is either

5.(a) (2) Description (Continued)

silicon rubber, fluorosilicon or fluorocarbon (viton). A pellet shipping assembly is used within the Oxide Vessel for certain shipments.

The approximate dimensions and weights of the package are as follows:

Sleeve inside diameter	10 1/4-inches
Oxide Vessel inside diameter	6, 7.5, or 8 inches
Oxide Vessel inside height	40 3/4-inches
Overall package dimensions	
width	45 inches
length	45 inches
height	62 inches
Maximum contents weight per Oxide Vessel	402 pounds
Maximum empty transport weight including four empty Oxide Vessels	2576 pounds
Maximum loaded package weight (with four filled Oxide Vessels)	3757 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos.:

OP-TU-SAR, Rev. 12, Sheets 1 of 2 and 2 of 2;
OP-TU-A2, Rev. 12, Sheet 1 of 1;
OP-TU-A3, Rev. 12, Sheet 1 of 1;
OP-TU-A4, Rev. 12, Sheet 1 of 1; and,
OPTU-V-AB1, Rev. 8, Sheets 1 of 2 and 2 of 2.

The Oxide Vessel Pellet Shipping Assembly is constructed and assembled in accordance with AREVA NP, Inc., Drawing No. 9046816, Rev. 1.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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5.(b) Contents

(1) Type and form of material

Uranium-bearing compounds in solid form, heterogenous or homogenous (i.e., pellets and powder). The contents may include up to 1000 grams of polyethylene or other plastics as packaging, waste or impurities per Oxide Vessel (4000 grams per package), provided that: (1) the total water equivalent of the plastic is less than 1307 grams per Oxide Vessel (5228 grams per package); and, (2) the decay heat is less than 0.068 W/m^3. Materials with a decay heat greater than 0.068 W/m^3 may not be packaged using hydrogen bearing plastics, and may only use non-hydrogen bearing plastics such as Teflon^{TM} (polytetrafluoroethylene or PTFE) or metallic containers. In addition, the contents are limited to:

A. Unirradiated uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope.

B. Unirradiated uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope.

C. Reprocessed uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1.

Table 1: Allowable Content for Shipment of Reprocessed Uranium Oxide

	Maximum Content		
Isotope	Type A	Type B Level I	Type B Level II
U-232 (g/gU)	Mixtures of isotopes shall be evaluated and designated as a Type A quantity per 10 CFR Part 71 Appendix A. The maximum enrichment per package is 5 weight per cent ^{235}U.	2.00E-09	5.00E-09
U-234 (g/gU)		2.00E-03	2.00E-03
U-235 (g/gU)		5.00E-02	5.00E-02
U-236 (g/gU)		2.50E-02	2.50E-01
Np-237 (g/gU)		1.66E-06	5.00E-03
Pu-238 (g/gU)		6.20E-11	4.00E-08
Pu-239 (g/gU)		3.04E-09	3.04E-09
Pu-240 (g/gU)		3.04E-09	6.00E-09
Gamma Emitters (MeV-Bq/kgU)		6.38E+05	1.91E+06

5. (b) (1) Type and Form of Material (Continued)

D. Reprocessed uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope, with the limits specified in Table 1.

E. Homogeneous (powder or crystalline form) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

F. Heterogeneous (pellets or previously pelletized materials) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

5. (b)(2) Maximum quantity of material per package

The maximum allowable contents heat generation rate is 1.0 BTU/hr/ft^3 (10.3 W/m^3). The maximum weight of contents, including the uranium compounds and all packaging materials within the Oxide Vessel, is 402 pounds per 8-inch, 7.5-inch, or 6-inch diameter Oxide Vessel, and a maximum of 1608 pounds per package.

For contents described in Condition Nos. 5(b)(1)(B), 5.(b)(1)(D), and 5.(b)(1)(F), the Oxide Vessel Pellet Shipping Assembly, as described in Condition No. 5(a)(3), must be used within the 8-inch diameter Oxide Vessel. The Oxide Vessel Pellet Shipping Assembly is not required when using the 7.5-inch, or 6-inch diameter Oxide Vessel.

(c) Criticality Safety Index 2.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application, as supplemented.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application, as supplemented.

7. Transport by air of fissile material is not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Revision No. 8 of this certificate may be used until February 28, 2011.

10. Expiration date: March 31, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9288	9	71-9288	USA/9288/B(U)F-96	6	OF	6

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006.

Supplements dated: April 10, 2006; July 17 and August 29, 2007; January 18, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: February 2, 2010

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9289	4	71-9289	USA/9289/B(U)F-85	1	OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA NP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome Cogema Fuels application dated May 1, 2002, as supplemented.

4. CONDITIONS

is certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: WE-1

(2) Description

A fresh fuel assembly shipping container. The package has two shipping configurations: one for shipping a single BW 17x17 fuel assembly composed of uranium dioxide pellets within zircalloy cladding; and the other for shipping up to 48 Pathfinder fuel assemblies within a steel canister which functions as a secondary containment vessel. The package consists of a cylindrical outer container and a rectangular inner container bolted to a strongback. The outer container is constructed of 11 gauge carbon steel and opens into two semi-cylindrical halves. The inner container is comprised of 1-inch thick carbon steel plates that are bolted together. The inner container is secured to the strongback by bolts and clamp arms. Wood blocks surround the region between the inner container and the strongback. The strongback is supported by 14 shock mounts attached to the outer container.

<u>For BW 17x17 Fuel Shipment Configuration:</u>

The BW 17x17 fuel assembly shipment configuration consists of the fuel assembly placed into the inner container. The fuel assembly is surrounded by thermal insulation and secured inside the inner container with nine integral clamp frames.

5. (2) Description (Continued)

For Pathfinder Fuel Shipment Configuration:

Pathfinder Fuel shipment configuration consists of the Pathfinder fuel in the Pathfinder Canister, which is placed into the inner container. The Pathfinder Canister is a sealed cylindrical canister which houses up to 48 Pathfinder fuel assemblies. Wood dunnage or empty sheaths may be used to fill empty spaces in the canister. The canister is made of austenitic stainless steel and has a welded body and a bolted closure lid. The Pathfinder Canister is surrounded by thermal insulation, and secured inside the inner container with five integral clamp frames. The clamp frames, which consist of bolted clamp arms, are bolted to the inner rectangular container. Wood blocks surround both ends of the Pathfinder Canister. A stainless steel spacer tube is used to fill the space between the Pathfinder Canister and the inner container.

The approximate dimensions and weights of the package are as follows:

Inner container length	165 inches
Inner container width (sq)	16 ½ inches
Outer container length	216 inches
Outer container diameter	44 inches
Maximum content weight	1610 pounds
Maximum package weight (including contents)	9090 pounds

(3) Drawings

The packaging is constructed in accordance with the following Framatome Cogema Fuels Drawing Nos.:

1273964, Rev. 0
1273965, Rev. 1
1273966, Rev. 0
1273967, Rev. 0
1273968, Rev. 0

The Pathfinder Canister Configuration is constructed in accordance with the following Framatome ANP Drawing Nos.:

5016270, Rev. 1
5021426, Sheets 1 and 2, Rev. 0.

(b) Contents

(1) Type and form of material

(i) For BW 17x17 Fuel Shipment Configuration:

A fuel assembly composed of uranium dioxide pellets within zircalloy cladding. The fuel assembly has the following specifications:

Assembly type	BW 17x17
No. fuel rods	264
No. non-fuel tubes	25
Nominal fuel rod pitch, in.	0.496
Maximum fuel pellet OD, in.	0.3232
Nominal clad OD, in.	0.374
Nominal clad thickness, in.	0.022
Nominal guide and instrument tube OD, in.	0.48
Nominal guide and instrument tube ID, in.	0.452
Nominal active fuel length, in.	144
Maximum uranium enrichment, weight percent U-235	4.6
Maximum U-235 mass, kg	22.14

(ii) For Pathfinder Fuel Shipment Configuration:

An unirradiated fuel assembly composed of six fuel pins clustered around a center absorber pin in a hexagonal pattern. The fuel pins consist of uranium dioxide pellets inside Incoloy 800 cladding. The absorber pin consists of Incoloy 800 cladding with or without poison material. Fuel pins and absorber pins are separated by spacer wires and enclosed in a cylindrical sheath made of stainless steel, incoloy or incoloy alloy. The fuel assembly has the following specifications:

Assembly type	Pathfinder
No. fuel pins per assembly	6
No. non-fuel pins per assembly	1
Maximum uranium enrichment, weight percent U-235	7.51
Maximum uranium mass per assembly, kg U	2.2281

5. (b) Contents (continued)

Maximum UO_2 density, g/cm^3	10.61
Fuel pellet outer diameter (OD), in.	0.207 ± 0.0005
Nominal active fuel length, in.	72.0
Minimum clad OD, in.	0.246
Maximum clad inner diameter (ID), in.	0.212
Nominal center-to-center pin pitch, in.	0.289
Nominal sheath ID, in.	0.945
Nominal sheath OD, in.	1.00

(2) Maximum quantity of material per package

(i) For the contents described in Item 5(b)(1)(i):

One BW 17x17 fuel assembly contents, not to exceed 1610 pounds. The radioactive material may not exceed any of the following limits:

U-232	0.01 microgram per gram of uranium
U-234	0.001 gram per gram of uranium
U-236	0.013 gram per gram of uranium
Tc-99	5 micrograms per gram of uranium
Fission Products	4.4×10^5 MeV-Becquerel per kilogram of uranium
Np and Pu	35 Becquerels per gram of uranium

(ii) For the contents described in Item 5(b)(1)(ii):

Up to 48 unirradiated Pathfinder fuel assemblies inside a Pathfinder Canister. The weight of the fully loaded canister not to exceed 800 pounds.

(c) Criticality Safety Index (CSI): 100

In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17.

8. Transport by air of fissile material is not authorized.

9. Revision No. 3 of this certificate may be used until February 28, 2010.

10. Expiration date: February 28, 2014.

REFERENCES

Framatome ANP, Inc. application dated: May 1, 2002.

Supplement dated: November 12, 2002, and January 8, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: February 10, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9290	7	71-9290	USA/9290/B(U)-85	1 OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Best Theratronics
413 March Road
Ottawa, Ontario
Canada K2K 0E4

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated February 20, 2003, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No. F-430/GC-40 Transport Package

(2) Description

The Model No. F-430/GC-40 Transport package is designed to transport MDS Nordion's Gammacell-40 (GC-40) irradiator containing cesium-137 sealed sources in special form. The F-430 overpack provides impact and thermal protection for the radioactive contents. Containment is provided by the special form sealed source and shielding is provided by the GC-40 irradiator body.

The F-430 is stainless steel cylindrical package with a 50" diameter and a height of 50" that is placed on a removable mild steel skid. The maximum weight of the package is 7000 pounds. The maximum weight of the GC-40 contents is 3835 pounds.

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

The package contents consist of a Cesium-137 sealed source contained within an MDS Nordion GC-40 irradiator (upper or lower heads). The GC-40 is a research irradiator with lead shielding and a lead filled source drawer.

5.(a)(2) (continued)

The approximate dimensions and weights of the package are as follows:

Package outside diameter	50 inches
Package height	50 inches
Cavity diameter	36 inches
Cavity height	35.25 inches
Removable skid	50 inches x 50 inches x 8 inches (height)
Overpack weight	2640 pounds
Contents weight	3835 pounds
Maximum package weight	7000 pounds

(3) Drawings

The packaging is constructed in accordance with the Best Theratronics drawings F643001-001, Rev. P, sheet 1 of 3, and F643001-001, Rev. H, sheet 2 of 3, and F643001-001, Rev. B, sheet 3 of 3.

(b) Contents

(1) Type and form of material

Cesium-137 as a sealed source which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

2,000 Curies.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Transport by air of fissile material is not authorized.

9. Expiration date: February 28, 2017.

REFERENCES

MDS Nordion application dated February 20, 2003.

Supplements dated: July 21, August 25, and December 18, 2003; January 16, July 16, July 21, and July 23, 2004; April 21, and October 30, 2006; February 27 (Best Theratronics), March 31 (MDS Nordion), 2009, October 7, 2011 (Best Theratronics), October 21, 2011, February 15, and March 9, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 4, 2012

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9291	8	71-9291	USA/9291/B(U)F-96	1 OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Columbiana Hi Tech, LLC, application dated August 23, 2011.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: Liqui-Rad (LR) Transport Unit Package

(2) Description

The LR Package is designed to transport Type B quantities of fissile uranyl nitrate solutions. The package uses thermal and impact limiting systems to protect the containment vessel and prevent the contents from being released. The primary structural components of the LR packaging consist of a stainless steel containment vessel, a carbon steel outer vessel and a carbon steel framing system. The containment vessel is built in accordance with ASME Pressure Vessel Code (Section VIII, Division 1) but does not require an ASME stamp. Double O-ring seals on the containment vessel's primary and secondary lids provide a leak tight seal which is leak testable. A closed-cell phenolic foam or polyurethane foam surrounds the top and bottom head area of the containment vessel and ceramic fiber blanket and board insulation are used in the sidewalls and outer lid for thermal insulation and impact absorption. The maximum volume of the contents is limited to 230 gallons which maintains a minimum ullage of 33 gallons.

The LR is a cylindrical package set in a rectangular angle frame. The dimensions of the package are approximately 56"(l) x 56"(w) x 73"(h). The maximum weight of the package is 5692 pounds. The outer vessel is constructed of 10 gauge carbon steel. The containment vessel is constructed of 1/4 inch stainless steel with 1/4 inch thick flanged and dished heads. The containment vessel is rated at 50 psig pressure. Closed-cell phenolic or polyurethane foam and ceramic fiber insulation are sandwiched between the containment vessel and the package's outer shell.

5.(a)(2) Continued

The package is designed to be leak-tight (maximum allowable leakrate of 1×10^{-7} ref-cm^3/sec). The containment vessel is closed using a double O-ring and is secured by sixteen 5/8 inch stainless steel studs. The outer lid is closed with twelve 5/8 inch stainless steel studs and the Manual Vent Enclosure (MVE) lid is secured with four 5/8 inch stainless steel bolts and nuts. The package is also equipped with plastic plugs to vent any gases that may be generated by the insulation during a fire event. All valves and fittings are provided within sealed enclosures to contain any leakage during valve failure.

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos. LR-SAR, sheets 1 through 4, Rev. 8.

5.(b) Contents

(1) Type and form of material

Low enriched Uranyl Nitrate solutions with the specifications shown in Table 1 below. The uranium concentration must be less than or equal to 125 gU/liter with an enrichment less than or equal to 5.0 wt% U-235. Non-fissile chemical impurities may be present up to the chemical impurity specification in Table 1. Additionally, fissile isotopes are also limited to the quantities in Table 1.

(2) Maximum quantity of material per package

230 gallons of Uranyl Nitrate solution with limits as shown in Table 1.

Table 1

ITEM	SPECIFICATION
Solution Density	≤1.17 g/cc
Chemical Impurities	≤1500 µg/gU
Nitric Acid Normality	0.1 - 0.7
Uranium Concentration	≤125 gU/l
U-232	≤2.0E-03 µg/gU
U-234	≤2.0E+03 µg/gU
U-235	≤0.05 g/gU (12 pounds maximum quantity of U-235 per LR)
U-236	≤2.5E+04 µg/gU
U-238	remainder of uranium

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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Pu/Np Alpha Activity	≤93 Bq/gU
Gamma Emitters	0.515E-01 Ci

5. (c) Criticality Safety Index 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Packagings must be marked with Package Identification Number USA/9291/B(U)F-96.

9. Transport by air of fissile material is not authorized.

10. Revision No. 7 of this certificate may be used until September 30, 2012.

11. Expiration date: October 31, 2016.

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated August 23, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Kimberly J. Hardin

Kimberly J. Hardin, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 20, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9292	6	71-9292	USA/9292/AF-85	1	OF	5

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company, LLC
P. O. Box 355
Pittsburg, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company, LLC consolidated application dated April 15, 2010.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PATRIOT

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

There are two versions of the metal inner container. Both versions measure approximately 11-1/4 inches high by 18-1/8 inches wide by 182 inches long. There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 30-1/4 inches wide by 31-1/4 inches high by 207-3/4 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on a series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,988 pounds with the version #1 inner container and 2,964 pounds with the version #2 (optional) inner container.

5.(a)(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC Drawing Nos.:

10014E27, Rev. 1,
10014E28, Sheets 1 and 2, Rev. 2,
10015E58, Sheets 1 and 2, Rev. 2,

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of UO_2 fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches.

(i) Description of Assembly Type #1

Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(A) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly. Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod. Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36. Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods. Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(B) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly. Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod. Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52. Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods. Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b) Contents (continued)

(C) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly. Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod. Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80. Maximum U-235 enrichment is 4.0 weight percent for all corner rods. Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.4 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Description of Assembly Type #2

Each assembly is made up of 96 fuel rods having a maximum active fuel length of 150 inches. Each assembly contains four one-third length fuel rods and eight two-thirds length fuel rods. The four one-third length fuel rods are located on the outside corners of the assembly. The eight two-thirds length fuel rods, arranged as two rods in each quadrant of the assembly, are located symmetric to the geometric diagonal, toward the center of the assembly. The fuel pellet diameter is 0.848 cm nominal, encapsulated in 0.061 cm nominal zirconium alloy cladding. There is a 0.0075 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. The fuel assembly must be transported in channels. The specifications for each one-third length axial section of the fuel assembly are as follows:

(A) Upper section must contain 84 fuel rods, arranged as 21 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least eight fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly, arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain 12 water holes, arranged as three water holes in each quadrant of the assembly. One of the three water holes within each quadrant must be located on the outside corner location of the assembly, and the other two water holes must be located on the geometric diagonal of the fuel assembly. Other fuel rods containing gadolinia may be present.

(B) Middle section must contain 92 fuel rods, arranged as 23 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least ten fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The ten gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain four water holes, arranged as one water hole in each quadrant of the assembly. Each water hole within each quadrant must be located on the outside corner location of the assembly. Other fuel rods containing gadolinia may be present.

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5. (b) Contents (continued)

(C) Lower section must contain 96 fuel rods, arranged as 24 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least twelve fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b)(2) Maximum quantity of material per package

Two fuel assemblies. The total weight of contents not to exceed 1,320 pounds.

5.(c) Criticality Safety Index: 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into, or out of, the sheathed fuel assembly.

For the contents described in 5.(b)(1)(i), polyethylene inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated. No polyethylene inserts may be used for the contents described in 5.(b)(1)(ii).

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be maintained in accordance with the Maintenance Program in Chapter 8 of the application.

9. For packagings fabricated in accordance with Drawing No. 10015E58, Rev. 1 (referred to as version #2 inner containers), only Serial Nos. 001 through 039, inclusive, are authorized for use.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Revision No. 5 of this certificate may be used until May 31, 2011.

12. Fabrication of new packaging is not authorized.

13. Expiration date: August 31, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Westinghouse Electric Company, LLC consolidated application dated: April 15, 2010.

Supplement dated: April 22, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: April 30, 2010

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Transnuclear, Inc
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Transnuclear, Inc., application dated May 19, 1999, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No. TN-68 Transport Package

(2) Description

The TN-68 is predominantly a steel package that is used to transport up to 68 intact BWR fuel assemblies with or without channels. The overall dimensions of the package are 271 inches long and 144 inches in diameter with the impact limiters installed.

The package generally consists of four components, the fuel basket assembly, a containment vessel within a forged steel cask body, a radial neutron shield, and impact limiters.

The basket assembly locates and supports the fuel assemblies, transfers heat to the cask body wall and provides neutron absorption to satisfy sub-criticality requirements. The basket structure consists of an assembly of stainless steel cells, joined by fusion welding of 1.75 inch wide stainless steel plates. Above and below the plates are slotted borated aluminum (or boron carbide/aluminum) metal matrix composite neutron poison plates which form an egg-crate structure. This construction forms a honey-comb like structure of cell liners which provides compartments for 68 fuel assemblies. The nominal dimensions of each cell is 6.0 inches x 6.0 inches.

A thick-walled (6.0 inch), forged steel cask body for gamma shielding surrounds the containment vessel, by an independent shell and bottom plate of carbon steel. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The thickness of the bottom of the cask body is 8.25 inches. A 4.5 inch thick steel gamma shield is also welded to the inside of the containment lid.

5.(a)(2) continued

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plates, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment vessel length is approximately 189 inches with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 69.5 inches and a length of 178 inches. The containment lid is 5 inches thick and is fastened to the cask body with 48 bolts. Double metallic O-ring seals are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure. There are two penetrations through the containment vessel which are located in the lid. These penetrations are for draining and venting. Double metallic seals are also used on these two lid penetrations. The OP port provides access to the interspace lid seals for leak testing purposes. The OP transport cover is not part of the containment boundary.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

The package has impact limiters at each end of the cask body. The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells that maintain the wood in a dry atmosphere and provide wood confinement when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters are attaching to each other using 13 tie rods and to the cask by eight bolts attaching to brackets welded to the outer shell in eight locations (four bolting locations per impact limiter).

The approximate dimensions and weights of the package are as follows:

Overall length (with impact limiters, in)	271
Overall length (without impact limiters, in)	197
Impact Limiter Outside diameter, (in)	144
Outside diameter (without impact limiters, in)	98
Cavity diameter (in)	69.5
Cavity length (in)	178
Containment shell thickness (in)	1.5
Containment vessel length (in)	184
Body wall thickness (in)	7.5
Containment lid thickness (in)	5
Overall lid thickness (in)	9.5
Bottom thickness (in)	9.75
Resin and aluminum box thickness (in)	6
Outer shell thickness (in)	0.75
Overall basket length (in)	164

5.(a)(2) continued

Maximum weight of package (pounds)	272,000
Maximum weight of BWR fuel contents (pounds)	47,900
Maximum weight of impact limiters and attachments (pounds)	32,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with TN drawings:

972-71-1, Revision 1
972-71-2, Revision 2
972-71-3, Revision 4
972-71-4, Revision 2
972-71-5, Revision 1
972-71-6, Revision 1
972-71-7, Revision 3
972-71-8, Revision 2
972-71-9, Revision 2
972-71-10, Revision 1
972-71-11, Revision 1
972-71-12, Revision 0
972-71-13, Revision 0
972-71-14, Revision 1

5.(b) Contents

(1) Type and form of material

Contents are limited to 68 unconsolidated intact irradiated GE BWR fuel assemblies with zircalloy cladding. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. Partial fuel assemblies (i.e. spent fuel assemblies from which fuel rods are missing), shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water equal to that displaced by the original rod(s).

Spent nuclear fuel may be transported with or without channels. Any fuel channel thickness up to 0.120 is acceptable on any of the fuel designs shown below. The maximum initial rod pressurization is 155 psig. The maximum fuel assembly length is 176.2 inches and the maximum fuel assembly width is 5.44 inches.

Permissible fuel assemblies are limited as stated in table 1 (fuel types may be C, D, or S lattice):

5.(b)(1) continued

Table 1, Fuel characteristics

GE fuel generation	model	array	rod pitch	fuel rods	rod od	clad thick	pellet dia.	water rods	water rod od	water rod id	U content (MTU/ Assembly)	Max active fuel length
2A	2a	7x7	0.738	49	0.570	0.036	0.488	0	x	x	0.1977	144
2, 2B	2	7x7	0.738	49	0.563	0.032	0.487	0	x	x	0.1977	144
3, 3A, 3B	3	7x7	0.738	49	0.563	0.037	0.477	0	x	x	0.1896	144
4, 4A,4B	4	8x8	0.640	63	0.493	0.034	0.416	1	0.493	0.425	0.1880	146
5	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
6, 6B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
7, 7B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
8, 8B -2w	82	8x8	0.640	62	0.483	0.032	0.411	2	0.591	0.531	0.1885	150
8, 8B-4W*	84	8x8	0.640	60	0.483	0.032	0.411	4	0.591	0.531	0.1824	150
8, 8B-4W**	84	8x8	0.640	60	0.483	0.032	0.411	4	0.483	0.431	0.1824	150
9, 9B	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
10	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
11	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
13	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
12	12	10x10	0.510	92	0.404	0.026	0.345	2	0.98	0.92	0.1857	150 full, 93 partial

*2 large water rods
**2 small water rods

Notes on table 1:

1. All dimensions in inches.
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inches thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt% U-235 average enrichment.
4. The fuel pitch is for C and D lattice designs. The S lattice fuels have a smaller pitch, which is less reactive.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

5.(b)(1) continued

Provided all of the requirements of this section are met, the bounding fuel characteristics are: a) maximum initial lattice-average enrichment is 3.7%; b) the minimum initial bundle average enrichment is 3.3%; c) the maximum assembly average burnup is 40,000 MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies. The maximum burnup, minimum initial enrichments and cooling times for each of the three fuel assembly types is contained in the tables below.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment.

In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR
Cooling times (years)
TYPE I BWR Fuel

	Burnup (GWd/MTU)											
Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	10										
1.1	10	10										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	10	10	11	11	11					
1.6	10	10	10	10	10	11	11	11				
1.7	10	10	10	10	10	11	11	11	12			
1.8	10	10	10	10	10	11	11	11	11	12		
1.9	10	10	10	10	10	11	11	11	11	12		
2.0	10	10	10	10	10	10	11	11	11	12	12	
2.1	10	10	10	10	10	10	11	11	11	12	12	12
2.2	10	10	10	10	10	10	11	11	11	12	12	12
2.3	10	10	10	10	10	10	11	11	11	11	12	12
2.4	10	10	10	10	10	10	10	11	11	11	12	12
2.5	10	10	10	10	10	10	10	11	11	11	12	12
2.6	10	10	10	10	10	10	10	11	11	11	12	12
2.7	10	10	10	10	10	10	10	10	11	11	11	12
2.8	10	10	10	10	10	10	10	10	10	11	11	12
2.9	10	10	10	10	10	10	10	10	10	11	11	12
3.0	10	10	10	10	10	10	10	10	10	10	11	12
3.1	10	10	10	10	10	10	10	10	10	10	11	12
3.2	10	10	10	10	10	10	10	10	10	10	10	11
3.3	10	10	10	10	10	10	10	10	10	10	10	10
3.4	10	10	10	10	10	10	10	10	10	10	10	10
3.5	10	10	10	10	10	10	10	10	10	10	10	10
3.6	10	10	10	10	10	10	10	10	10	10	10	10
3.7	10	10	10	10	10	10	10	10	10	10	10	10

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Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR
Cooling times (years)
TYPE II BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	18	21										
1.1	17	20										
1.2	17	20										
1.3	17	20										
1.4	17	20										
1.5	16	19	25	26	26							
1.6	16	19	25	26	26							
1.7	16	19	25	25	26	26	27					
1.8	16	19	24	25	26	26	27	27				
1.9	16	19	24	25	25	26	27	27				
2.0	16	18	24	25	25	26	26	27	28			
2.1	15	18	23	25	25	26	26	27	27			
2.2	15	18	23	25	25	25	26	27	27			
2.3	15	18	23	24	25	25	26	26	27	27		
2.4	15	18	22	24	24	25	26	26	27	27		
2.5	15	17	22	24	24	25	25	26	26	27		
2.6	15	17	22	24	24	24	25	26	26	27		
2.7	15	17	22	24	24	24	25	26	26	26	27	27
2.8	14	17	22	23	24	24	25	25	26	26	27	27
2.9	14	17	22	23	23	24	24	25	26	26	27	27
3.0	14	17	21	23	23	23	24	25	25	26	27	27
3.1	14	17	21	23	23	23	24	25	25	26	27	27
3.2	13	16	21	23	23	23	24	24	25	25	26	27
3.3	13	16	21	23	22	23	23	24	25	25	26	26
3.4	13	16	21	23	22	23	23	24	25	25	26	26
3.5	13	16	21	22	22	23	23	24	25	25	26	26
3.6	13	16	21	21	22	22	23	24	25	25	26	26
3.7	12	15	20	21	22	22	23	24	25	25	25	26

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Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE III BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	11										
1.1	10	11										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	15	16	16	17	17					
1.6	10	10	14	16	16	17	17	17				
1.7	10	10	14	15	16	16	17	17	17			
1.8	10	10	14	15	15	16	16	17	17	18		
1.9	10	10	14	15	15	16	16	17	17	18		
2.0	10	10	14	15	15	16	16	16	17	17	18	
2.1	10	10	14	15	15	15	16	16	16	17	18	18
2.2	10	10	13	14	15	15	16	16	16	17	17	18
2.3	10	10	13	14	15	15	16	16	16	17	17	18
2.4	10	10	13	14	15	15	15	16	16	17	17	18
2.5	10	10	13	14	14	15	15	16	16	16	17	18
2.6	10	10	13	14	14	15	15	16	16	16	17	17
2.7	10	10	13	14	14	15	15	15	16	16	17	17
2.8	10	10	13	13	14	14	15	15	16	16	17	17
2.9	10	10	13	13	14	14	15	15	15	16	16	17
3.0	10	10	12	13	14	14	14	15	15	16	16	17
3.1	10	10	12	13	14	14	14	15	15	15	16	16
3.2	10	10	12	13	14	14	14	15	15	15	16	16
3.3	10	10	12	13	13	14	14	14	15	15	16	16
3.4	10	10	12	13	13	13	14	14	15	15	16	16
3.5	10	10	12	13	13	13	14	14	14	15	15	16
3.6	10	10	12	12	13	13	14	14	14	15	15	15
3.7	10	10	12	12	13	13	14	14	14	15	15	15

5.(b) continued

(2) Maximum quantity of material per package

The maximum contents weight is 75,600 pounds. The maximum weight of the irradiated fuel contents is 47,900 pounds.

(3) Decay Heat Limit

Maximum decay heat per package not to exceed 21.2kW. The maximum heat load per assembly is 0.313 kW/assembly.

(c) Criticality Safety Index 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

7. Known or suspected fuel assemblies with cladding defects greater than pin hole leaks and or hairline cracks are not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Fabrication of new packagings in not authorized.

10. Transport by air of fissile material is not authorized.

11. Expiration date: February 29, 2016.

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U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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REFERENCES

Transnuclear, Inc., application dated May 19, 1999.

Supplements dated March 2, October 18, and November 13, 2000, January 12, 2001, January 20, 2006, January 6, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Section
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 2/9/11

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9294	6	71-9294	USA/9294/AF-96	1 OF 3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Global Nuclear Fuel - Americas, LLC
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated April 16, 2010.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NPC

(2) Description

A cubic stainless steel and foam outer packaging with nine cylindrical containment vessels for the transport of type A quantities of low-enriched uranium oxide powder, pellets, and compounds of uranium as defined in 5(b). The overall package dimensions are approximately 45 inches wide, 45 inches deep, and 44 inches high.

The outer packaging consists of a 10-gage stainless steel outer shell with a ceramic fiber board liner and rigid polyurethane foam filler. The foam filler has a three-by-three array of vertical cylindrical cutouts that accommodate stainless steel sleeves for placement of the containment vessels. The outer packaging is equipped with a top cover that is secured to the outer packaging body by a combination of 16 closure cap screws and four closure strips secured by 24 bolts.

The containment vessel is a maximum 8.515 inches in inner diameter and approximately 32 inches in overall length. The containment vessel is constructed of 18-gage stainless steel, surrounded by a cadmium sheet and polyethylene wrap within a 24-gage stainless steel jacket. The containment vessel is closed by a 16-gage closure lid, a silicone rubber gasket, and a band clamp assembly, which is composed of a 0.063-inch thick strap and retainer, a T-bolt, and a nut.

The gross weight of the package (packaging and contents) is 1,302 kg (2,870 pounds). The maximum weight of the contents is 540 kg (1,190 pounds).

5.(a) (3) Drawings

The packaging is fabricated and assembled in accordance with the following Global Nuclear Fuel - Americas, LLC, Drawing Nos.:

177D4970, Sheet 1, Revision 1

177D4970, Sheet 2, Revision 0
177D4970, Sheet 3, Revision 0
177D4970, Sheet 4, Revision 0
177D4970, Sheet 5, Revision 0
177D4970, Sheet 6, Revision 0
177D4970, Sheet 7, Revision 0
177D4970, Sheet 8, Revision 1
SK105E4037, Sheet 2, Revision 1

(b) Contents

Type, Form, and Maximum Quantity of Material Per Package

Material Forms[1] (≤5.00 wt.% U-235)	Particle Size Restriction: Minimum OD (Inches)	Maximum Loading per ICCA (kgs)		Maximum Loading per NPC (kgs)	
		Net[4]	Uranium	Net[4]	Uranium
Homogenous Uranium Oxide/Compounds[2]	N/A	60.0	52.89	540.0	476.1
Heterogenous UO_2 Pellets (BWR)	0.342	60.0	40.54	540.0	364.8
Heterogenous UO_2 Pellets(PWR)	0.300	60.0	40.54	540.0	364.8
Heterogenous Uranium Compounds[3]	Unrestricted particle size	60.0	40.54	540.0	364.8

[1]No solutions or liquids are authorized and there shall be no free liquid present. The Material Form within any NPC must be the same.
[2]Homogenous compounds limited to UO_2, U_3O_8, $UO_{x,\, x>2}$, dried calcium-containing sludges, $UO_2(NO3)_2 \cdot 6H_2O$, and uranium oxide bearing ash.
[3]Heterogenous compounds limited to UO_2, U_3O_8, and $UO_{x,\, x>2}$.
[4]Maximum content weight of any Inner Containment Canister Assemblies (ICCA) including plastic or secondary packaging (i.e., dunnage). Materials with a hydrogen atom density greater than that of water are limited to a mass of 3.7 kg per ICCA.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9294	6	71-9294	USA/9294/AF-96	3	OF	3

(c) Criticality Safety Index 0.7

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented. Within each ICCA, the contents and secondary packaging (i.e., dunnage) must provide a snug fit. The payload may be enclosed in plastic receptacles (e.g., bags, bottles, etc.). For payloads in plastic bottles, empty bottles may be used to minimize movement of the bottles within the ICCA.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Transport by air of fissile material is not authorized.

Revision No. 5 of this certificate may be used until June 30, 2011.

10. Expiration date: November 30, 2015.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated May 11, 2010.

Supplements dated: May 19, 2010 and June 3, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: June 10, 2010

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9295	4	71-9295	USA/9295/B(U)F-96	1	OF	8

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA Federal Services LLC
505 S. 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated June 25, 2004, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Mixed Oxide Fresh Fuel Package (MFFP)

(2) Description

The MFFP is designed to transport unirradiated mixed oxide (MOX) fuel assemblies and individual MOX fuel rods contained in rod boxes.

The MFFP body is made of a 9/16-inch thick XM-19 austenitic stainless steel cylindrical shell with the flange section and a 1-1/2 inch bottom end plate welded to it. A circumferentially continuous doubler plate, constructed of Type XM-19 austenitic stainless steel, is welded to each end of the shell, near the end of each impact limiter. Welded to the doubler plate are the impact limiter attachment lugs, six per impact limiter. The doubler plate also serves to provide a tiedown interface with the transportation skid.

The seal flange is located at the open end of the body, and consists of a locally thicker wall section to accommodate the closure lid sealing area and the closure bolt threaded holes. The transition between the shell and the seal flange section is a 3:1 taper. Polyurethane foam is used to build the outer diameter of the body out to the full diameter of the sealing flange and closure lid.

5.(a)(2) continued

The closure lid is a weldment constructed of Type XM-19 3/4-inch outer plate and 5/8-inch thick inner plate, stiffened with eight 1/2-inch thick radial ribs that are three inches deep. A 1/2-inch thick, 6 inch inner diameter cylinder forms a hub at the inner end of the radial ribs. The ribs are welded on all four edges to the adjacent structure. Each rib has a projection that passes through a slot in the outer plate, and the ribs and outer plate are welded together.

The closure lid inner plate is welded to the outer ring. The seal flange of the closure lid has a minimum thickness of one inch, and provides location for three O-ring bore seals with the middle seal providing the containment seal. The seals are 3/8-inch diameter butyl rubber O-ring.

Up to three unirradiated fuel assemblies are held in place inside the overpack by a strongback assembly which is constructed from 1/4-inch thick Type 304 stainless steel weldment, a series of clamp arm assemblies, a top, and a bottom plate assemblies. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design. Neutron poison plates are placed inside the weldment. A series of fuel control structure (FCS) limits lateral expansion of fuel rods during vertical and near vertical hypothetical accident condition (HAC) free drops and also hold neutron poison plates.

A pair of conical-shaped impact limiters filled with polyurethane foam provide thermal and impact protections. The closure lid end impact limiter has 1/4-inch thick shells to resist perforation from the HAC puncture drop, and to protect the closure lid and sealing area from puncture and HAC fire damage. Shock indicators are attached to the outside of the MFFP shell.

The approximate dimensions and weights of the package are as follows:

Overall package outside dimensions (inches)	
Without Impact Limiters	
Diameter	30
Length	171
With Impact Limiters	
Diameter	60
Length	201
Maximum content weight	4,740 lbs
Maximum package weight (Including contents)	14,260 lbs

(3) Drawings

The packaging shall be constructed and assembled in accordance with Packaging Technology, Inc., drawing numbers:

(a) Shipping Package	99008-10, Rev. 5, Sheet 1
(b) Body Assembly	99008-20, Rev. 4, Sheets 1 through 6
(c) Strongback Assembly	99008-30, Rev. 6, Sheets 1 through 7
(d) Top Plate Assembly	99008-31, Rev. 2, Sheets 1 through 3
(e) Bottom Plate Assembly	99008-32, Rev. 2, Sheets 1 and 2
(f) Clamp Arm Assembly	99008-33, Rev. 4, Sheets 1 through 4
(g) Fuel Control Structure Assembly	99008-34, Rev. 5, Sheets 1 and 2
(h) Impact Limiter	99008-40, Rev. 3, Sheets 1 through 3
(i) AFS-B Assembly	99008-60, Rev. 1, Sheets 1 and 2
(j) AFS-C Assembly	99008-61, Rev. 1, Sheets 1 and 2

(b) Contents

(1) Type and Form of Material

Unirradiated 17 x 17 fuel assemblies with solid PuO_2+UO_2 pellets in zirconium based alloy (M5) tubes. The fuel assemblies are based on the MK-BW/MOX1 17 x 17 PWR design. The fuel assemblies may contain Burnable Poison Rod Assemblies (BPRA). The physical specifications for the unirradiated fuel assemblies and the burnable poison rod assemblies are provided in Tables 1 and 2. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design.

5.(b)(1) continued

The ARB-17 is a rod container designed to transport up to 17 MOX fuel rods. The rods type is identical to the rods comprising the standard MOX fuel assembly. The rods may be either undamaged, damaged, or a combination of both (e.g., 9 undamaged and 8 damaged). Damaged fuel rods may be bent, scratched, or dented, but under no circumstances may exhibit cladding breaches. A 2-inch Schedule 40 pipe mounted with pipe clamps against one wall of the ARB-17 is used to transport undamaged or slightly damaged fuel rods. Damaged fuel rods may be transported within this pipe only if the bending in the fuel rod is minor. The ARB-17 MOX fuel rod container has been designed with outer dimensions consistent with a standard fuel assembly so that it will interface with the strongback and clamp arms.

The AFS-B Rod Container is designed to contain up to 175 MOX fuel rods. The container has outer cross sectional dimensions of 8.4 inches square, a length from bottom to top of 159.9 inches, and an overall length (to the lift ring bolt head) of 161.2 inches. The primary material of construction of the container is ASTM 6061-T651 aluminum alloy.

The AFS-C Rod Container is designed to contain up to 116 Exxon rods, up to 69 Pacific Northwest Laboratory (PNL) rods, or both quantities together. The container is the same as the AFS-B Rod Container except the AFS-C container has two internal 2-inch thick aluminum plates which form rod cavities to accommodate both types of rods the AFS-C Rod Container may hold.

The EMA is similar to MOX fuel assemblies with the exceptions that the OD of the fuel pellets may be out of tolerance (nominal pellet diameter = 0.323 inch), and the weight percent Pu-238 exceeds the 0.05 wt.% limit specified in Table 1.2-2 of the SAR (EMA fuel rods have Pu-238/Pu as high as 0.19 wt.%).

5.(b)(1) continued

Table 1 - Fuel Assembly Physical Parameters
(nominal values unless stated otherwise)

Parameter	Values
Fuel Rod Cladding Material	M5
Fuel Rod Array	17 x 17
Fuel Rods per Fuel Assembly	264
Guide Tubes per Fuel Assembly	24
Instrument Tubes per Fuel Assembly	1
Guide/Instrument Tube Thickness (inches)	0.016
Fuel Assembly Length (inches)	161.61
Fuel Assembly Maximum Width (inches)	8.565
Fuel Rod Pitch (inches)	0.496
Fuel Rod Length (inches)	152.4
Fuel Rod Outside Diameter (inches)	0.374
Fuel Rod Clad Thickness (inches)	0.023
Active Fuel Length (inches)	144.0
$PuO_2 + UO_2$ Weight (pounds)	1,157
Heavy Metal Weight (pounds)	1,020
Maximum Fuel Assembly Weight including Burnable Poison Rod Assembly (pounds)	1,580
Maximum Initial Pu Loading (weight% of heavy metal)	6.0

Table 2 - Burnable Poison Rod Assembly Parameters

Parameter	Value
Poison Rod Cladding Material	Zircaloy-4
Poison/Thimble Plug Rod Array	Up to 24 rods
Burnable Poison Material	Al_2O_3-B_4C

5. (b) (2) Maximum Quantity of Material per Package

Three unirradiated fuel assemblies with specifications on fuel pellets and enrichment are provided in Table 3. Three Areva Rod Box 17 (ARB-17) containers may contain up to 17 standard MOX fuel rods. One AFS-B rod container may contain up to 175 standard MOX fuel rods and one Excess Material Assembly. Three AFS-C rod containers may contain up to 116 Exxon rods and 69 PNL rods. The permissible configurations of contents are summarized in Table 4.

Table 3 - Nuclear Design Parameters for Fuel Assemblies

Parameter	Value
Nominal Pellet Diameter (inches)	0.323
Maximum Effective Pellet Density (gram/cm^3)	10.85
Maximum Total Plutonium (Pu) Content	0.06 g Pu/g Heavy Metal (Pu+U)
Plutonium Isotopic Contents	Pu-238: Up to 0.0005 g/g Pu Pu-239: 0.90 to 0.95 g/g Pu Pu-240: 0.05 to 0.09 g/g Pu Pu-241: Up to 0.01 g/g Pu Pu-242: Up to 0.001 g/g Pu
Minimum Total Uranium (U) Content	0.94 g U/g Heavy Metal (Pu+U)
Uranium Isotopic Contents	U-234: Up to 0.0005 g/g U U-235: Up to 0.003 g/g U U-238: Remainder of U content

Table 4 – Payload Table

Payload Type	Strongback Positions (3)		
MOX Fuel	MOX FA	MOX FA or dummy FA	MOX FA or dummy FA
MOX Fuel and ARB-17 Rod Container	MOX FA or ARB-17	MOX FA, ARB-17, or dummy FA	MOX FA, ARB-17, or dummy FA
EMA	EMA	dummy FA	dummy FA
AFS-B and EMA	AFS-B	EMA or dummy FA	dummy FA
AFS-C	AFS-C	AFS-C or dummy FA	AFS-C or dummy FA

(c) Criticality Safety Index 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations of Chapters 7, 7A, 7B, and 7C of the application, as applicable, as supplemented.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapters 8, 8A, 8B, and 8C of the application, as applicable, as supplemented.

(c) The boron-10 areal density within each of the internal neutron poison plates shall be verified as described in Section 8.1.5.2 of the application, as supplemented.

(d) Wrapping shall not be used on the unirradiated fuel assemblies.

(e) Non-fuel dummy assemblies with the same nominal size and weight as the MK-BW/MOX1 17 x 17 design shall be used in the case of loading less than three fuel assemblies in a MFFP packaging.

7. Transport by air of fissile material is not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Revision No. 2 of this certificate may be used until June 30, 2011.

10. Expiration date: June 30, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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REFERENCES

Packaging Technology, Inc., application dated June 25, 2004.

Supplement dated: February 4 and 10, April 8, June 3, 2005, and January 19, August 15, and November 26, 2007, and April 4 and July 25, 2008, May 24, 2010, and October 31, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 18, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9296	8	71-9296	USA/9296/B(U)-96	1	OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., consolidated application, Revision No. 8, dated April 11, 2011.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. 880 Series Packages

(2) Description

The Model No. 880 series packages are designed for use as a radiography exposure device and a transport package for Type B quantities of radioactive material in special form. The Model No. 880 series packages have three versions called the 880 Delta, 880 Sigma and the 880 Elite. The 880 Delta has a maximum capacity of 150 Curies of Iridium-192 or 150 Curies of Selenium-75, the 880 Sigma has a maximum capacity of 130 Curies of Iridium-192 or 150 Curies of Selenium-75, and the 880 Elite has a maximum capacity of 50 Curies of Iridium-192 or 150 Curies of Selenium-75. The Delta and Sigma versions are identical and the Elite has a lighter weight depleted uranium shield. An optional jacket can be placed on the package to facilitate its use as an industrial radiography exposure device or a transport package. There are two versions of the jacket.

All versions of the package, without the jacket, are cylindrical in shape with a diameter of 5 inches (127 mm) and a length of 13 5/16 inches (338 mm). With the first version of the jacket, the shape of the package is an extruded triangle 9 inches (229 mm) high, 7 ½ inches (191 mm) wide, and 13 5/16 (343 mm) inches long. With the second version of the jacket, the package measures 13 ½ inches (343 mm) long by 6 inches (152 mm) wide by 11.33 inches (288 mm) tall.

5.(a) (2) Description (continued)

The weight of the Delta and Sigma versions is 46 pounds (21 kg) without the jacket, 52 pounds (24 kg) with version 1 of the jacket and 55 pounds (25 kg) with version 2 of the jacket. The weight of the Elite version is 37 pounds (17 kg) without the jacket, 42 pounds (19 kg) with version 1 of the jacket, and 45 pounds (20 kg) with version 2 of the jacket.

The major components of the packages consist of a welded stainless steel cylindrical body, a depleted uranium shield, a stainless steel rear plate with a locking assembly, a stainless steel front plate with a shielded port, and optional jackets.

The welded cylindrical body consists of a 5 inch (127 mm) diameter, 0.06 inch (1.5 mm) wall tube shell with 0.12 inch (3 mm) end-plates. A U-bracket is welded to each end-plate and is located on the inside cavity of the shell tube. The depleted uranium shield is centrally located within the welded body between the end-plate and is fastened to each U-bracket by a 0.37 inch (9.5 mm) diameter titanium shield pin. A U-shaped copper spacer fills the gap between the shield and the U-bracket. An S-shaped titanium source tube is cast into the center of the shield to provide a cavity for the source wire assembly to travel through during use.

The front and rear plates are attached to the welded body with four tamperproof screws through rivnuts assembled into end-plates. The rear plate assembly consists of a source locking mechanism fastened to the rear plate. The front plate assembly consists of a shielded port mechanism contained within the front plate.

An optional polyurethane jacket covers the package cylinder, provides a handle and a stable base, and is attached to the shell cylinder by screws located outside the shield cavity area. Version 1 of the jacket has a handle section that contains a wire molded in for additional reinforcement. Version 2 of the jacket incorporates wheels on the base to facilitate movement during use as a radiography exposure device.

(3) Drawings

The packaging is constructed in accordance with the QSA Global, Inc., drawings R88000, Rev. R, sheets 1-6, and R88095, Rev. A, sheets 1-2.

(b) Contents

(1) Type and form of material

Iridium-192 as a sealed source which meets the requirements of special form radioactive material.

Selenium-75 as a sealed source which meets the requirements of special form radioactive material.

5. (b) (2) Contents (continued)

(2) Maximum quantity of material per package

150 Curies (5.55 TBq) (output) Ir-192 for the Model No. 880 Delta.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Delta.

130 Curies (4.81 TBq) (output) Ir-192 for the Model No. 880 Sigma.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Sigma.

50 Curies (1.85 TBq) (output) Ir-192 for the Model No. 880 Elite.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Elite.

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr - Ci Iridium-192 at 1 meter and 0.20 R/hr - Ci Selenium-75 at 1 meter. (Ref: Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.)

(3) Maximum weight: 18 grams.

(4) Maximum decay heat: 3 Watts.

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly must be fabricated of materials capable of resisting a 1475° F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application; and,

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9296	8	71-9296	USA/9296/B(U)-96	4	OF	4

10. Revision No. 7 of this certificate may be used until June 30, 2012.

11. Expiration date: June 30, 2016.

REFERENCES

QSA Global, Inc., consolidated application, Revision No. 8, dated April 11, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 28, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9297	5	71-9297	USA/9297/AF-96	1 OF	8

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Westinghouse Electric Company, LLC
P.O. Drawer R
Columbia, SC 29250

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Westinghouse Electric Company, LLC, application dated June 17, 2011.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: Traveller STD and Traveller XL

(2) Description

The Traveller package is designed to transport non-irradiated uranium fuel assemblies or rods with enrichment up to 5.0 weight percent. The package is designed to carry one fuel assembly or one container for loose rods. The package consists of three components: 1) an outerpack, 2) a clamshell, and 3) a fuel assembly or rod container.

The outerpack serves as the primary impact and thermal protection for the fuel assembly and also provides for lifting, stacking, and tie down during transportation. Two independent impact limiters consisting of two sections of foam of different densities sandwiched between three layers of sheet metal are integral parts of the outerpack. Polyethylene foam sheeting may be positioned between the clamshell and the lower outerpack to augment shock absorbing characteristics during routine transportation. A weather gasket between the mating surfaces of the upper and lower outerpack provides a seal to prevent rain from entering the package.

The clamshell protects the contents during routine handling and limits the rearrangement of the contents in the event of an accident. The clamshell consists of an aluminum "v" extrusion, two aluminum door extrusions, and a small access door. Each extruded aluminum door is connected to the "v" extrusion with piano-type hinges (continuous hinges). These

5.(a)(2) Description (Continued)

doors are held closed with a latching mechanism and quarter-turn bolts. Neutron absorber plates are installed in each leg of the "v" extrusion and in each of the doors. The "v" extrusion and the bottom plate are lined with a cork rubber pad to cushion and protect the contents during normal handling and transport conditions. The clamshell is fastened to the lower outerpack using shock absorbing rubber mounts.

The Traveller package is designed to carry loose rods using a container or rod pipe. The rod pipe consists of a 15.2 cm (6 in.) standard 304 stainless steel, Schedule 40 pipe, and standard 304 stainless steel closures at each end. The closure is a 0.635 cm (0.25 in.) thick cover secured with Type 304 stainless steel hardware to a flange fabricated from 0.635 cm (0.25 in.) thick plate.

There are two models of the Traveller packaging, the Traveller STD and the Traveller XL.

Traveller STD:	
Package gross weight	2,041 kilograms (kg) (4,500 pounds (lbs))
Packaging gross weight	1,293 kg (2,850 lbs)
Contents gross weight	748 kg (1,650 lbs)
Outer dimensions	
Length	500.4 cm (197 in.)
Width	68.8 cm (27.0 in.)
Height	99.8 cm (39.3 in.)
Traveller XL:	
Package gross weight	2,313 kg (5,100 lbs)
Packaging gross weight	1,431 kg (3,155 lbs)
Contents gross weight	894 kg (1,971 lbs)
Outer dimensions	
Length	574 cm (226.1 in.)
Width	68.9 cm (27.1 in.)
Height	99.8 cm (39.3 in.)

(3) Drawings

The packagings are fabricated and assembled in accordance with the following Westinghouse Electric Company's Drawing Nos.:

10004E58, Rev. 6 (sheets 1-9)
10006E58, Rev. 5

5. (b) Contents (Type and Form of Material)

(1) Fuel Assembly

(i) Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent. The parameters of the fuel assemblies that are permitted are as follows:

Parameters for 14 x 14 Fuel Assemblies

Fuel Assembly Description	14 x 14	14 x 14	14 x 14
Fuel Assembly Type	W-STD	W-OFA	CE-1/CE-2
No. of Fuel Rods per Assembly	179	179	176
No. of Non-Fuel Rods	17	17	20
Nominal Guide Tube Wall Thickness	0.043 cm (0.017 in.)	0.043 cm (0.017 in.)	0.097 cm (0.038 in.)
Nominal Guide Tube Outer Diameter	1.369 cm (0.539 in.)	1.336 cm (0.526 in.)	2.822 cm (1.111 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.875 cm (0.344 in.)	0.956/0.966 cm (0.376/0.381 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.016 cm (0.400 in.)	1.118 cm (0.440 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)	0.071/0.066 cm (0.028/0.026 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.70 cm (7.76 in.)	19.70 cm (7.76 in.)	20.60 cm (8.11 in.)
Nominal Lattice Pitch	1.412 cm (0.556 in.)	1.412 cm (0.556 in.)	1.473 cm (0.580 in.)

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 15 x 15 Fuel Assemblies

Fuel Assembly Description	15 x 15	15 x 15
Fuel Assembly Type	STD/OFA	B&W
No. of Fuel Rods per Assembly	204	208
No. of Non-Fuel Rods	21	17
Nominal Guide Tube Wall Thickness	0.043 cm (0.017 in.)	0.043 cm (0.017 in.)
Nominal Guide Tube Outer Diameter	1.387/1.354 cm(0.546/0.533 in.)	1.354 cm (0.533 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.929 cm (0.366 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.072 cm (0.422 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)
Clad Material	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.66 cm (8.53 in.)
Nominal Lattice Pitch	1.430 cm (0.563 in.)	1.443 cm (0.568 in.)

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 16 x 16 Fuel Assemblies

Fuel Assembly Description	16 x 16	16 x 16	16 x 16	16 x 16	16 x 16	16 x 16
Fuel Assembly Type	W-STD	NGF	ATOM	CE16NVA	CE16VA	CE16NGF
No. of Fuel Rods per Assembly	235	235	236	236	236	236
No. of Non-Fuel Rods	21	21	20	20	20	20
Nominal Guide Tube Wall Thickness	0.046 cm (0.018 in.)	0.041 cm (0.016 in.)	0.070 cm (0.028 in.)	0.102 cm (0.040 in.)	0.102 cm (0.040 in.)	0.102 cm (0.040 in.)
Nominal Guide Tube Outer Diameter	1.196 cm (0.471 in.)	1.204 cm (0.474 in.)	1.380 cm (0.543 in.)	2.489 cm (0.980 in.)	2.489 cm (0.980 in.)	2.489 cm (0.980 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.784 cm (0.309 in.)	0.911 cm (0.359 in.)	0.826 cm (0.325 in.)	0.827 cm (0.326 in.)	0.819 cm (0.323 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.914 cm (0.360 in.)	1.075 cm (0.423 in.)	0.970 cm (0.382 in.)	0.970 cm (0.382 in.)	0.950 cm (0.374 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.057 cm (0.023 in.)	0.072 cm (0.029 in.)	0.064 cm (0.025 in.)	0.064 cm (0.025 in.)	0.057 cm (0.023 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.72 cm (7.76 in.)	19.72 cm (7.76 in.)	22.95 cm (9.03 in.)	20.63 cm (8.12 in.)	20.63 cm (8.12 in.)	20.63 cm (8.12 in.)
Nominal Lattice Pitch	1.232 cm (0.485 in.)	1.232 cm (0.485 in.)	1.430 cm (0.563 in.)	1.285 cm (0.506 in.)	1.285 cm (0.506 in.)	1.285 cm (0.506 in.)

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 17 x 17 and 18 x 18 Fuel Assemblies

Fuel Assembly Description	17 x 17	17 x 17	18 x 18
Fuel Assembly Type	W-STD/XL	W-OFA	ATOM
No. of Fuel Rods per Assembly	264	264	300
No. of Non-Fuel Rods	25	25	24
Nominal Guide Tube Wall Thickness	0.041/0.051 cm (0.016 /0.020 in.)	0.041 cm (0.016 in.)	0.065 cm (0.026 in.)
Nominal Guide Tube Outer Diameter	1.204/1.224/1.24 cm (0.474/0.482/0.488 in.)	1.204 cm (0.474 in.)	1.240 cm (0.488 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.784 cm (0.309 in.)	0.805 cm (0.317 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.914 cm (0.360 in.)	0.950 cm (0.374 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.057 cm (0.023 in.)	0.064 cm (0.025 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.39 cm (8.42 in.)	22.94 cm (9.03 in.)
Nominal Lattice Pitch	1.260 cm (0.496 in.)	1.260 cm (0.496 in.)	1.270 cm (0.500 in.)

(ii) Non-fissile base-plate mounted core components, and spider-body core components, including secondary source rods and axial spacer assembly, are permitted.

(iii) Primary neutron sources or other radioactive material are not permitted.

(iv) Materials with moderating effectiveness greater than full density water are not permitted, except for polyethylene sleeves used to protect the fuel assemblies.

(v) There is no restriction on the length of top and bottom annular blankets.

(vi) Replacement of fuel rods with any number of solid stainless steel rods is permitted.

5.(b) (2) Loose Fuel Rods

Unirradiated uranium dioxide fuel rods with a maximum uranium-235 enrichment of 5.0 weight percent. Fuel rods shall be transported in the Traveller package inside a rod pipe as specified in Drawing 10006E58. The fuel rods shall meet the parametric requirements given below:

Parameter	Limit
Maximum Enrichment	5.0 weight percent uranium-235
Pellet diameter	0.508 – 1.524 cm (0.20 – 0.60 in.)
Maximum stack length	Up to rod container length
Cladding	Zirconium alloy
Integral absorber	Gadolinia, erbia, and boron
Wrapping or sleeving	Plastic or other material with moderating effectiveness no greater than full density water
Maximum number of rods per container	Up to rod container capacity

Wrapping or sleeving: Materials with moderating effectiveness greater than full density of water are not permitted, except for polyethylene sleeve used to protect the fuel rods.

5.(c) Criticality Safety Index

(1) When transporting fuel assemblies: 0.7

(2) When transporting loose rods in a rod container: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the Traveller License Application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the Traveller License Application, as supplemented.

7. The package authorized by this certificate is hereby authorized for use under the general license provisions of 10 CFR 71.17.

8. The package is not authorized by this certificate for air transport.

9. Revision No. 3 of this certificate may be used until June 30, 2013.

10. Expiration date: March 31, 2015.

REFERENCES

Westinghouse Electric Company, LLC, application dated June 17, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael D. Waters for

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: July 6, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9299	5	71-9299	USA/9299/B(U)-85	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Best Theratronics
413 March Road
Ottawa, Ontario
Canada K2K 0E4

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated November 29, 2006, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: F-423

(2) Description

A double-walled welded stainless steel overpack for shipping sealed sources within the Gammacell 220 (GC220) gamma irradiator. The packaging consists of concentric box-like stainless steel shells separated by an annulus of rigid polyurethane foam. The overall overpack wall thickness is eight inches on the sides, twelve inches on the front and rear, and four inches on the base. The overpack lid is constructed of a sheet of 1/2-inch thick stainless steel on top, a sheet of 1/4-inch thick cold-rolled steel on the bottom, and 4-inches of polyurethane foam in between. The package is closed by bolting the lid to the body with 40 one-inch diameter bolts.

The GC220 irradiator is positioned inside the cavity formed by the inner stainless steel shell, along with an inner steel frame and a rigid polyurethane foam bonnet and lower crush pad. Shielding is provided by the GC220 irradiator, which is a welded steel lead-filled device. The GC220 is a lead-filled shielding head mounted on a steel stand. The GC220 shielding head consists of inner and outer steel shells with lead in between. The nominal lead thickness is 10 inches. The GC220 has an irregular shape, however, the base is 60-inches long by 40-inches wide. In its shipping configuration, the GC220 is 58-inches high. The GC220 shielding plug is welded from 304 stainless steel and lead filled. The GC220 drawer is welded from 304 stainless steel and is lead filled.

5(a) (2) (continued)

The maximum package weight (including contents) is 21,000 lbs (9,524 kgs). The approximate package component dimensions and weights are as follows:

Component	Weight (lbs / kg)	Nominal Dimensions (L x W x H inches)
Overpack Lid	1,036 / 470	67.50 x 55.00 x 4.75
Inner Frame	1,257 / 570	60.50 x 48.00 x 54.13
Bonnet	871 / 395	52.00 x 41.50 x 36.75
GC220	8,576 / 3.890	60.00 x 40.00 x 58.00
Overpack Body	8,708 / 3.950	86.50 x 66.00 x 80.37
Lower Crush Pad	386 / 175	47.00 x 31.00 x 7.00

(3) Drawings

The packaging is constructed in accordance with MDS Nordion Drawing No. F642301-001, Sheet 1, Revision G, and Sheet 2, Revision D.

(b) Contents

(1) Type and form of material

i. Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.

ii. Cobalt-60 as sealed sources described in Condition No. 6 below.

(2) Maximum quantity of material per package

26,000 curies, a maximum of 48 sources per package, and a maximum of 5,000 curies per source.

6. Sealed sources limited to MDS Nordion sealed source capsules manufactured before February 19, 1973: C-166, C-167, and C-185. In addition, these sources must meet the following:

   (a) Sources must conform to the specifications identified in the application in Figure 4.2 for the C-166 source, Figure 4.3 for the C-167 source, and Figure 4.4 for the C-185 source;

   (b) Sources must be shown to not be leaking within six months prior to shipment; and

   (c) Sources must not have been damaged during their service life.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Transport by air of fissile material is not authorized.

10. Expiration date: March 31, 2017

REFERENCES

MDS Nordion application dated November, 20, 2006

Supplement dated: February 8, 2007; February 27 (Best Theratronics), March 31 (MDS Nordian), 2009, and October 7 (Best Theratronics), 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Kimberly J. Hardin

Kimberly Hardin, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: March 23, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9301	5	71-9301	USA/9301/AF-96	1	OF	6

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Transnuclear, Inc.
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Packaging Technology, Inc., application dated July 24, 2002, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: TNF-XI

(2) Description

A shipping container for unirradiated enriched forms of homogenous and heterogeneous uranium oxides. The packaging body is a parallelepiped and is approximately 44 inches x 44 inches x 37 inches. The package contents are enclosed in pails which each have a borated stainless steel ring. Three pails are stacked inside four inner wells of the packaging body. Each inner well is closed by a primary lid and an upper plug.

The packaging body is constructed of an outer stainless steel envelope which is 0.08 inches thick. The space between the outer shell and the inner wells is filled with fire-retardant, open cell phenolic foam.

The four inner wells each have an inside diameter of 14 inches and height of 27 inches. The inner wells are constructed of (1) and outer shell of stainless steel sheet 0.04 inches thick, with a diameter of 17 inches, (2) and inner shell of stainless steel sheet 0.04 inches thick with a diameter of 14 inches, and (3) a flat bottom of 0.04 inch thick stainless steel sheet with a 0.08 inch thick borated stainless steel plate glued to it. A molded annular layer of neutron-poison BORA resin is inserted between the inner and outer steel shells of the inner well.

Each upper plug consists of two thermal insulating disks of phenolic foam, with an internal stiffener disk made of aluminum alloy. The upper plug assembly is encapsulated inside a 0.03 inch thick stainless steel envelope.

5.(a) (2) Description (continued)

The four primary lids closing off the inner wells are stainless steel circular plates 0.2 inches thick on the center part, and 0.4 inches thick on the periphery. Four bayonet teeth are welded to the primary lid to lock in the well flanges. A primary lid locker is located between the well flange and the primary lid to prevent the rotation of the primary lid during transport. The primary lid and the inner well are sealed by an elastomer gasket set in a rectangular groove machined on the inner face of the primary lid.

The approximate dimensions and weights of the package are as follows:

Inner well inside diameter	14 inches
Overall package dimensions	
Width	44 inches
Length	44 inches
Height	41 inches
Maximum weight of contents in any pail	25 kg
Maximum content weight	300 kg
Maximum package weight (including contents)	1050 kg

(3) Drawings

The packaging is constructed in accordance with the Packaging Technology, Inc., Drawing No. 10799-SAR, Rev. 3, Sheets 1 through 7.

(b) Contents

(1) Type and form of material

(i) The uranium oxide pellets, powder, and scrap meets the requirements of Enriched Commercial Grade Uranium, as defined in ASTM C996-10. U_3O_8 or $UO_{x,\ x>2}$ are authorized provided that the equivalent UO_2 mass is less than the limits specified below:

5.(b)(1)(i) Type and Form of Material (continued)

Max ^{235}U Enrichment (weight percent)	Homogenous UO_2 Powder Maximum Loading (kg)	Heterogeneous UO_2 Material (Pellet and Scrap) Maximum Loading (kg)
≤ 4.05	300	300
4.1	300	293
4.15	300	287
4.25	300	271
4.35	300	259
4.45	300	247
4.55	294	238
4.65	281	228
4.75	265	219
4.85	255	208
4.95	244	202
5.0	239	197

(ii) The uranium oxide pellets, powder, and scrap meets the requirements of Enriched Commercial Grade Uranium, as defined in ASTM C996-10. U_3O_8 or $UO_{x,\ x>2}$ are authorized provided that the equivalent UO_2 mass is less than the limits specified below:

Max ^{235}U Enrichment (weight percent)	Homogenous UO_2 Powder Maximum Loading (kg)	Heterogeneous UO_2 Material (Pellet and Scrap) Maximum Loading (kg)
≤ 4.05	300	300
4.15	300	284
4.25	300	271
4.35	300	256
4.45	300	247
4.55	286	236
4.65	271	224
4.75	259	216
4.85	248	208
4.95	238	202
5.0	232	196

(2) Maximum quantity of material per package

(i) For the contents described in 5.(b)(1)(i), no more than 25 kg of contents per pail. No more than 300 kg of contents per package. Presence of hydrogenated materials (with a hydrogen concentration less than hydrogen concentration in water) or water inside cavities and pails is allowed.

The auto-ignition temperature of the hydrogenated materials (with a hydrogen concentration less than hydrogen concentration in water) shall be greater than 140°C (284°F).

The presence of materials containing more hydrogen than water is not allowed in the package.

(ii) For the contents described in 5.(b)(1)(ii), no more than 25 kg of contents per pail. No more than 300 kg of contents per package. In each pail, the contents can be put in a polyethylene bag (CH_2) or in a bag made of a material with a hydrogen concentration less than that of polyethylene. The maximum hydrogen content of the bags within each cavity is a mass of 56 g H, which is equivalent to a maximum mass of 390 g polyethylene, considering all sources of hydrogenous material within each cavity.

The auto-ignition temperature of the bag material shall be greater than 140°C (284°F).

The presence of materials containing more hydrogen than polyethylene is not allowed in the package.

(c) Criticality Safety Index: 0.5

6. Transport by air is not authorized.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the operating procedures in Chapter 7 of the application, as supplemented;

(b) The package must by acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented; and,

(c) Prior to each shipment, the stainless steel components of the packaging must be visually inspected. Packagings in which stainless steel components show pitting corrosion, cracking, or pinholes are not authorized for transport.

8. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.71.

9. Packagings may be marked with Package Identification Number USA/9301/AF-85 until January 31, 2013, and must be marked with Package Identification Number USA/9301/AF-96 after January 31, 2013.

10. Revision No. 4 of this certificate may be used until January 31, 2013.

11. Expiration date: August 31, 2013.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9301	5	71-9301	USA/9301/AF-96	6	OF	6

REFERENCES

Packaging Technology, Inc., application dated July 24, 2002.

Supplements provided by Packaging Technology, Inc., dated: October 29, 2002; March 7, April 3, May 6, June 26, July 21, 2003; November 26, 2007; and August 6, 2008.

Supplements provided by Transnuclear, Inc., dated: September 8, October 28 and December 23, 2011; January 6, 2012

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: January 17, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATION NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9302	5	71-9302	USA/9302/B(U)F-96	1 OF	17

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION.

c. ISSUED TO (Name and Address)

Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, MD 21045

d. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear Inc., application dated August 17, 2011, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos: NUHOMS®-MP197, NUHOMS®-MP197HB

(2) Description: NUHOMS®-MP197

The NUHOMS®-MP197 package consists of an outer packaging, into which a NUHOMS®-61BT transportable dry shielded canister (DSC) is placed. During shipment, energy-absorbing impact limiters are used for additional package protection. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the cask body. Weights and dimensions in the following discussion are approximate values, unless otherwise noted.

Cask

The NUHOMS®-MP197 transport package is fabricated primarily of stainless steel. Non-stainless steel members include the cask lead shielding between the containment boundary inner shell and the structural shell, the o-ring seals, the neutron shield, and carbon steel closure bolts. The body of the packaging consists of a 1.25 inch thick, 68 inch inside diameter, stainless steel inner (containment) shell and a 2.5 inch thick, 82 inch outside diameter stainless steel structural shell, without impact limiters which sandwich the 3.25 inch thick cast lead shielding. The overall external dimensions of the cask are 208 inches long and 91.5 inches in outer diameter. The weight of the package body is 148,840 pounds including about 10,000 pounds of neutron shield and 60,000 pounds of cast lead.

5.(a)(2) Description, NUHOMS®-MP197 (continued)

The containment system of the NUHOMS®-MP197 transportation package consists of the inner shell, a 6.50 inch thick bottom plate, 2.5 inch thick radioactive material (RAM) access closure with a diameter of approximately 24 inches, a top closure flange, a 4.5 inch thick top closure lid with closure bolts, drain port closures and bolts, and double o-ring seals for each penetration. The containment vessel prevents leakage of radioactive material from the package cavity. The package cavity is pressurized to above atmospheric pressure with an inert gas (helium). Helium assists in the heat removal. Shielding is provided by approximately 4 inches of stainless steel, 3.25 inches of lead, and approximately 4.5 inches of neutron shielding. Four removable trunnions are provided for handling and lifting of the package.

Dry Shielded Canister (DSC)

The purpose of the DSC, which is placed within the transport package, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The shell has an outside diameter of about 67 inches and an external length of about 200 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is given to the DSC as a containment boundary. The basket is designed to accommodate 61 intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The poison plates are constructed from borated aluminum, and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell is a laminate of balsa wood and redwood. Each impact limiter is attached to the cask top (front) and bottom (rear) by 12 bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two hoist rings for handling. The hoist rings are threaded into the impact limiter shell. During transportation, the impact limiter hoist rings are removed. An aluminum thermal shield is added to the bottom impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts, is approximately 28,000 lbs.

5.(a)(3) Description, NUHOMS®-MP197HB

The NUHOMS®-MP197HB package consists of an outer packaging, which is used for the off-site transport of any one of the four NUHOMS® DSCs (24PT4, 61BT, 61BTH, and 69BTH). It is also used to transport a secondary container (Radioactive Waste Container (RWC)) with dry irradiated and/or contaminated non-fuel bearing solid materials. During shipment the package uses energy-absorbing impact limiters for additional package protection. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the cask body. Weights and dimensions in the following discussions are nominal values, unless otherwise noted.

Package

The MP197HB package is a modified version of the MP197 package described in 5(a)(2).

The packaging is fabricated primarily of nickel-alloy steel (NAS). Other materials include the cast lead shielding between the containment boundary inner shell and the structural shell, the O-ring seals, the resin neutron shield, and the carbon steel closure bolts. Socket headed cap screws (bolts) are used to secure the lid to the package body and the RAM access closure plate to the bottom of the package. The body of the package consists of a 1.25 inch thick, 70.5 inch inside diameter NAS inner (containment) shell, and a 2.75 inch thick, 84.5 inch outside diameter NAS structural shell which sandwiches the 3 inch thick cast lead shielding material.

The overall dimensions of the NUHOMS®-MP197HB packaging are a length of 271.25 inches and a diameter of 126 inches with both impact limiters installed. The transport package body is 210.25 inches long and 84.5 inches in diameter. The package diameter including the radial neutron shield is 97.75 inches or 104.25 inches with the fins (optional feature). The packaging cavity is 199.25 inches long and 70.5 inches in diameter without the internal sleeve (discussed below) or 68 inches in diameter with the sleeve.

The MP197HB uses an internal aluminum sleeve for smaller diameter DSCs and secondary containers. The inner sleeve is designed with slots to accommodate the existing rails inside the package and to provide rails inside the sleeve on which the smaller diameter DSCs or secondary containers slide during horizontal loading or unloading of the package.

The gross weight of the loaded package is 152 tons including a maximum payload of 56 tons. Four removable trunnions, attached to the package body, are provided for lifting and handling operations, including rotation of the packaging between the horizontal and vertical orientations.

The package containment boundary consists of the inner shell, a 6.5 inch thick bottom plate with a 28.88 inch diameter, 2.5 inch thick RAM access closure plate, a package body flange, a 4.5 inch thick lid with lid bolts, vent and drain port closures and bolts, 5.(a)(3) Description, NUHOMS®-MP197HB (continued)

and O-ring seals for each of the penetrations. The containment vessel prevents leakage of radioactive material from the package cavity. It also maintains an inert atmosphere (helium) in the package cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation. The cask cavity is pressurized with helium to above atmospheric pressure. Shielding is provided by approximately 4 inches of steel, 3 inches of lead and 6.25 inches of neutron shielding assembly.

To accommodate the NUHOMS®-69BTH DSC with heat loads greater than 26 kW, removable external fins are provided as an option for the package.

Dry Shielded Canister (DSC)

The function of the DSC, which is placed within the transport package, is identical to that described for the MP197 cask in paragraph 5(a)(2) above. The DSC consists of a stainless steel shell and a basket assembly. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is taken for the DSC as a containment boundary.

There are four DSC designs and a radioactive waste canister authorized for transport in the NUHOMS®-MP197HB packaging. The MP197HB packaging cavity is designed to accommodate the larger 69.8 inch diameter DSCs (69BTH DSC). To accommodate the smaller 67.3 inch diameter DSCs (24PT4, 61BT, and 61BTH DSC) or secondary container (RWC), an aluminum inner sleeve is provided. To accommodate the varying lengths of the DSCs and secondary containers, stainless steel or aluminum spacers are provided to limit axial movement of the payload. Spacers are to be installed in the DSC cavity, if necessary, to limit the maximum axial gap between any fuel assembly and the DSC to 0.5 inches or less. Similarly, spacers are to be installed in the overpack cavity, if necessary, to limit the maximum axial gap between the DSC and the overpack to 0.5 inches or less.

The maximum weight of the payload (DSC including the fuel) is limited to 56 tons.

The DSC basket poison plates are constructed from Boral®, borated aluminum or aluminum/B_4C metal matrix composite (MMC) and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Radioactive Waste Container (RWC)

The RWC consists of a payload of dry irradiated and/or contaminated non-fuel bearing solid materials. No credit is taken for the containment provided by the RWC.

The RWC assembly together with any appropriate cask cavity spacers shall provide an equivalent of 1.75 inches minimum steel shielding in the radial direction. A minimum of 5.75 inches equivalent steel shielding shall be provided at the bottom of the canister and a minimum of 7.00 inches equivalent steel shielding at the top of the canister. The maximum weight of the payload (RWC, including waste) is limited to 56 tons.

5.(a)(3) Description, NUHOMS®-MP197HB (continued)

Impact Limiters

Impact limiters consisting of balsa wood and redwood encased in stainless steel shells are attached at the front and rear end of the MP197HB package during shipment. Each impact limiter is attached to the package by twelve (12) attachment bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving internal pressure. Each impact limiter has three hoist rings for handling, and two support angles for supporting the impact limiter in a vertical position during storage. The hoist rings are threaded into the impact limiter shell, while the support angles are welded to the shell. Prior to transport, the impact limiter hoist rings are removed and replaced with bolts. An aluminum thermal shield is added to each impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts, is 25,000 lbs.

5.(a)(4) Drawings, NUHOMS®-MP197

The package shall be constructed and assembled in accordance with the following Transnuclear, Inc., Drawing numbers:

1093-71-1, Revision 0,
NUHOMS®-197 Packaging
Transport Configuration

1093-71-2, Revision 1,
NUHOMS®-197 Packaging
General Arrangement

1093-71-3, Revision 1,
NUHOMS®-MP197 Packaging
Parts List

1093-71-4, Revision 1,
NUHOMS®-MP197 Packaging
Cask Body Assembly

1093-71-5, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-6, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-7, Revision 0,
NUHOMS®-MP197 Packaging
Lid Assembly & Details

1093-71-8, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Assembly

1093-71-9, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Details

1093-71-10, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Assembly

1093-71-11, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Details

1093-71-12, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Details

1093-71-13, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

5.(a)(4) Drawings, NUHOMS®-MP197
(continued)

1093-71-14, Revision 1, NUHOMS®-61BT Transportable Canister for BWR Fuel General Assembly

1093-71-15, Revision 2, NUHOMS®-61BT Transportable Canister for BWR Fuel Shell Assembly

1093-71-16, Revision 0, NUHOMS®-61BT Transportable Canister for BWR Fuel Shell Assembly

1093-71-17, Revision 2, NUHOMS®-61BT Transportable Canister for BWR Fuel Canister Details

1093-71-18, Revision 1, NUHOMS®-61BT Transportable Canister for BWR Fuel Canister Details

1093-71-20, Revision 0, NUHOMS®-MP197 Packaging Regulatory Plate

1093-71-21, Revision 0, NUHOMS®-MP197 Packaging on Transport Skids

5.(a)(5) Drawings, NUHOMS®-MP197HB

The NUHOMS®-MP197HB package shall be constructed and assembled in accordance with the following Transnuclear, Inc. drawings:

MP197HB-71-1001 Rev 1	NUHOMS®-MP197HB Packaging Transport Configuration (2 sheets)
MP197HB-71-1002 Rev 3	NUHOMS®-MP197HB Packaging Parts List (2 sheets)
MP197HB-71-1003 Rev 1	NUHOMS®-MP197HB Packaging General Arrangement (1 sheet)
MP197HB-71-1004 Rev 3	NUHOMS®-MP197HB Packaging Cask Body Assembly (1 sheet)
MP197HB-71-1005 Rev 2	NUHOMS®-MP197HB Packaging Cask Body Details (3 sheets)
MP197HB-71-1006 Rev 0	NUHOMS®-MP197HB Packaging Lid Assembly & Details (1 sheet)
MP197HB-71-1007 Rev 0	NUHOMS®-MP197HB Packaging Regulatory Plate (1 sheet)
MP197HB-71-1008 Rev 1	NUHOMS®-MP197HB Packaging Impact Limiter Assembly (1 sheet)
MP197HB-71-1009 Rev 1	NUHOMS®-MP197HB Packaging Impact Limiter Details (1 sheet)
MP197HB-71-1011 Rev 0	NUHOMS®-MP197HB Packaging Transport Configuration Outer Sleeve With Fins Option (1 sheet)
MP197HB-71-1014 Rev 1	NUHOMS®-MP197HB Packaging Internal Sleeve Design (2 sheets)
NUH24PT4-71-1001 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Basket Assembly (5 sheets)
NUH24PT4-71-1002 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Main Assembly (8 sheets)
NUH24PT4-71-1003 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Failed Fuel Can (4 sheets)
NUH61BT-71-1000 Rev 1	NUHOMS® 61BT Transportable Canister For BWR Fuel Parts List (1 sheet)
NUH61BT-71-1001 Rev 1	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Assembly (1 sheet)
NUH61BT-71-1002 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Details (1 sheet)

NUH61BT-71-1003 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1004 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1005 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1006 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1007 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1008 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1009 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Details (1 sheet)
NUH61BT-71-1010 Rev 1	NUHOMS® 61BT Transportable Canister For BWR Fuel Additional Basket Details – Damaged Fuel (4 sheets)
NUH61BTH-71-1000 Rev 1	NUHOMS® 61BTH Type 1 Transportable Canister For BWR Fuel Main Assembly (5 sheets)
NUH61BTH-71-1100 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Main Assembly (7 sheets)
NUH61BTH-71-1101 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Shell Assembly (2 sheets)
NUH61BTH-71-1102 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Basket Assembly (8 sheets)
NUH61BTH-71-1103 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Transition Rails (2 sheets)
NUH61BTH-71-1104 Rev 1	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Damaged Fuel End Caps (1 sheet)
NUH61BTH-71-1105 Rev 1	NUHOMS® 61BTHF Type 2 Transportable Canister For BWR Fuel Failed Fuel Can (2 sheets)
NUH61BTH-71-1106 Rev 2	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Top Grid Assembly Alternate 3 (2 sheets)
NUH69BTH-71-1001 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Main Assembly (4 sheets)

NUH69BTH-71-1002 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket – Shell Assembly (4 sheets)
NUH69BTH-71-1003 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Shell Assembly (4 sheets)
NUH69BTH-71-1004 Rev 3	NUHOMS® 69BTH Transportable Canister For BWR Fuel Alternate Top Closure (6 sheets)
NUH69BTH-71-1011 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket Assembly (5 sheets)
NUH69BTH-71-1012 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Transition Rail Assembly And Details (6 sheets)
NUH69BTH-71-1013 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Holddown Ring Assembly (2 sheets)
NUH69BTH-71-1014 Rev 1	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel Modification (1 sheet)
NUH69BTH-71-1015 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel End Caps (1 sheet)
NUHRWC-71-1001 Rev 1	NUHOMS® System RWC Canister - Welded Top Shield Plug Design Main Assembly (5 sheets)
NUHRWC-71-1002 Rev 1	NUHOMS® System RWC Canister - Welded Top Shield Plug Design Inner Liner (3 sheets)
NUHRWC-71-1003 Rev 0	NUHOMS® System RWC Canister - Bolted Top Shield Plug Design Main Assembly (4 sheets)

5.(b) Contents of Packaging, NUHOMS®-MP197

(1) Type and Form of Material

(a) Intact irradiated BWR fuel assemblies with or without fuel channels, with uranium oxide pellets and zircaloy cladding. Channel thickness is limited to 0.065 to 0.120 inches. Prior to irradiation, the fuel assemblies must meet the dimensions and specifications of Table 1. Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the NUHOMS®-61BT DSC.

(b) The maximum burn-up and minimum cooling times for the individual assemblies shall meet the requirements of Table 2.

(b) In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5(b)(1)(c). The maximum total allowable cask heat load is 15.86 kW.

(c) The maximum assembly decay heat of an individual assembly is 260 watts.

(d) BWR fuel assembly poison material shall meet the design requirements of Table 3.

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TABLE 1[1]

Assembly Type	7x7 49/0	8x8 63/1	8x8 62/2	8x8 60/4	8x8 60/1	9x9 74/2	10x10 92/2
Maximum Initial Enrichment (wt% ^{235}U)	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
Number of Fuel Rods per Assembly	49	63	62	60	60	66-full 8-partial	78-full 14-partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Minimum Cladding Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Pellet Diameter	0.487	0.416	0.410	0.410	0.411	0.376	0.345
Maximum Active Fuel Length (in)	144	146	150	150	150	146-full 90-partial	150-full 93-partial

[1] Maximum Co-59 content in the Top End Fitting region is 4.5 gm per assembly
Maximum Co-59 content in the Plenum region is 0.9 gm per assembly
Maximum Co-59 content in the In-Core region (including the whole fuel channel) is 4.5 gm per assembly
Maximum Co-59 content in the Bottom region is 4.1 gm per assembly

TABLE 2

Intact BWR Fuel Assembly Characteristics	
Physical Parameters:	
Fuel Design	7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel
Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR fuel"
Channels	Fuel may be stored with or without fuel channels
Maximum assembly weight	705 lbs
Radiological Parameters:	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	6 Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.0 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	12 Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.65 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

Intact BWR Fuel Assembly Characteristics	
Radiological Parameters:	
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.38 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	15-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.4 wt.% ^{235}U
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

TABLE 3

Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment

NUHOMS®-61BT DSC Basket Type	Maximum Fuel Assembly Lattice Average Enrichment (wt.% ^{235}U)	Minimum Boron-10 Areal Density for Boral® (g/cm^2)	Minimum Boron-10 Areal Density for Borated Aluminum, Metamic®, and Boralyn® (g/cm^2)	Areal Density Used in the Criticality Evaluation [75% Credit for Boral®] (g/cm^2)
	Intact Fuel Assemblies			
A	3.7	0.025	0.021	0.019
B	4.1	0.038	0.032	0.029
C	4.4	0.048	0.040	0.036

5.(b) Contents of Packaging, NUHOMS®-MP197 (continued)

(2) Maximum quantity of material per package

(a) The quantity of material authorized for transport is 61 intact standard BWR fuel assemblies with or without fuel channels. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For material described in 5(b)(1) the approximate maximum payload is 43,505 lbs.

5.(c) Contents of Packaging, NUHOMS®-MP197HB

The MP197HB packaging is designed to transport fuel assemblies stored inside any one of the four DSCs as described in Appendices A.1 and A.7 through A.9 of this CoC. The MP197HB package is also designed to transport dry irradiated and/or contaminated nonfuel bearing solid materials in an RWC as described in Appendix A.10 of this Certificate of Compliance (CoC). Each Appendix of this CoC provides the following for each DSC:

(1) Type and Form of Material (Fuel Specification and Characteristics).

(2) Maximum quantity of material per package.

(3) The maximum length of a natural or lower than natural enriched uranium blanket shall not exceed 5% of the active fuel length at each end of the fuel assembly.

(4) The users of this packaging system shall confirm that the maximum peaking factor of the fuel assembly average burnup in all fuel contents does not exceed 1.326 and 1.152 for BWR and PWR fuel, respectively.

5.(d) Criticality Safety Index "0"

6. For the NUHOMS®-MP197 and the NUHOMS®-MP197HB packages, fuel assemblies with missing fuel rods shall not be shipped as intact fuel unless the missing fuel rods are replaced with dummy rods that displace an equal or greater amount of water.

7. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71, the NUHOMS®-MP197 package shall meet the requirements listed below in Sections 7(a) and 7(b), while the NUHOMS®-MP197HB package shall meet the requirements listed in Sections 7(c) and 7(d).

(a) Each NUHOMS®-MP197 package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application. In addition this will include:

(1) verification of the basket type A, B, or C, by inspection of the last digit of the serial number on the grapple ring at the bottom of the DSC.

(2) verification that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Tables 2 and 3. The enrichment limit must correspond to the basket type determined in 7(a)(1) above.

(b) All fabrication acceptance tests and maintenance shall be performed for the NUHOMS®-MP197 in accordance with Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented. In addition, the package lid bolts will be replaced after 85, or fewer, round trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during normal conditions of transport.

(c) Each MP197HB package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter A.7 of the application, as supplemented. Detailed site-specific procedures shall be developed to include these steps as applicable to address the particular operational considerations related to the use of the MP197HB cask. Site specific conditions and requirements may require the use of different equipment and ordering of steps to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.

(d) For the MP197HB package, fabrication acceptance tests and maintenance shall be performed in accordance with the Acceptance Test and Maintenance Program in Chapter A.8 of the application.

8. For canisters exposed to a coastal saltwater marine environment prior to transportation under 10 CFR Part 72, the package user must evaluate the condition of the canister to verify 1) that canister degradation has not occurred to the extent that the fuel has incurred gross breaches due to oxidation and 2) that degradation of neutron absorbers and basket materials has not occurred to the extent they would no longer comply with the applicable materials and dimensions specified in section 5(a)(4) and 5(a)(5) for

Drawings. For these canisters, the aging management plan and evaluation for each canister or set of canisters shall be submitted to the NRC at least 120 days prior to shipment.

9. Transport by air is not authorized.

10. NUHOMS®-MP197 and NUHOMS®-MP197HB packages are approved for exclusive use by rail, truck, or marine transport.

11. The NUHOMS®-MP197 and NUHOMS®-MP197HB packages authorized by this certificate are hereby approved for use under the general license provisions of 10 CFR 71.17.

12. Revision No. 4 of this certificate may be used until August 31, 2013.

13. Expiration Date: August 31, 2017.

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REFERENCES

Transnuclear, Inc., Safety Analysis Report for the NUHOMS®-MP197 Transport Packaging, dated August 17, 2011.

Transnuclear Inc., letters dated September 15, 2011, and July 24, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Bernard White IV for

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 18, 2012

Attachments to CoC 9302, Revision 5:

Appendix A.1:	MP197HB Packaging Contents Loaded with NUHOMS®-24PT4 DSC
Appendix A.2:	NOT USED
Appendix A.3:	NOT USED
Appendix A.4:	NOT USED
Appendix A.5:	NOT USED
Appendix A.6:	NOT USED
Appendix A.7:	MP197HB Packaging Contents Loaded with NUHOMS®-61BT DSC
Appendix A.8:	MP197HB Packaging Contents Loaded with NUHOMS®-61BTH DSC
Appendix A.9:	MP197HB Packaging Contents Loaded with NUHOMS®-69BTH DSC
Appendix A.10:	MP197HB Packaging Contents Loaded with Radioactive Waste Canister (RWC)

CoC 9302 Revision 5, Appendix A.1

MP197HB Packaging Contents Loaded with NUHOMS®-24PT4 DSC

(1) Type and Form of Material

(a) Intact or damaged CE 16x16 PWR fuel assemblies which meet specifications listed in Tables A.1-1 and A.1-2, respectively, are authorized for transportation in the NUHOMS®-24PT4 DSC. Fuel debris and damaged fuel rods placed in a rod storage basket are considered as damaged fuel. Damaged fuel assemblies are to be stored in specially designed failed fuel cans [See Drawing NUH24PT4-71-1003, Rev. 0 (4 sheets)].

(b) For maximum assembly average burnup, minimum cooling time and decay heat limits, the fuel assemblies shall meet all the requirements of the cross referenced tables and figures listed in Tables A.1-1 and A.1-2. The fuel to be transported in the 24PT4 DSC is limited to a maximum initial enrichment of 4.85 wt.% ^{235}U. The maximum allowable assembly average burnup is given as a function of initial fuel enrichment but does not exceed 45,000 MWd/MTU. The minimum cooling time is 7 years.

(c) The minimum areal Boron-10 (^{10}B) densities for the standard (Type A basket) and high (Type B basket) loadings are 0.025 and 0.068 g/cm^2, respectively. Fuel to be transported in the Type A basket is limited to an initial ^{235}U enrichment of 4.1 wt.%. Fuel to be transported in the Type B basket is limited to an initial ^{235}U enrichment of 4.85 wt.%. In addition, poison rodlets shall be used per Table A.1-4.

(2) Maximum Quantity of Material per Package

(a) The quantity of material authorized for transport is 24 intact or up to 12 damaged and balance intact standard PWR fuel assemblies as shown in Figure A.1-1. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For materials described in A.1(1) above, the approximate maximum payload is 36,000 lbs. (does not include the weight of the poison rodlets).

Table A.1-1
PWR Fuel Specification of Intact Fuel to be Transported in the 24PT4 DSC

Fuel Design:	Intact CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.1-3 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be transported as "intact PWR Fuel."
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500(2) (3)
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 24 intact assemblies
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1-4 and Figure A.1-1
Fuel Assembly Average Burnup and Minimum Cooling Time(4)	Per Table A.1-5 and decay heat restrictions below
Decay Heat(4)	Per Figures A.1-2, A.1-3 or A.1-4

Notes:

(1) Nominal values shown unless stated otherwise.
(2) Does not include weight of Poison Rodlets (25 lbs each) installed in accordance with Table A.1-4.
(3) Includes the weight of fuel assembly Poison Rods installed for 10 CFR Part 50 criticality control in spent fuel pool racks.
(4) Minimum cooling time is the longer of that given in Table A.1-5 for a given burnup and enrichment of a fuel assembly and that calculated via the decay heat equation based on the restrictions provided in Figures A.1-2, A.1-3 or A.1-4.

Table A.1-2
PWR Fuel Specifications of Damaged Fuel to be Transported in the 24PT4 DSC

Fuel Design	Damaged CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.1-3 and the following requirements:
Fuel Damage	Damaged fuel may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). Damaged fuel assemblies shall be encapsulated in individual Failed Fuel Cans and placed in Zones A and/or B as shown in Figure A.1-1. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met.
Physical Parameters[1]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[2] [3]
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 12 damaged assemblies, balance intact.
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table A.1-4 and Figure A.1-1
Fuel Assembly Average Burnup and Minimum Cooling Time[4] [5]	Per Table A.1-5 and decay heat restrictions below
Decay Heat[4]	Per Figures A.1-2, A.1-3 or A.1-4

Notes:

(1) Nominal values shown unless stated otherwise.
(2) Does not include weight of Poison Rodlets (25 lbs each) installed in accordance with Table A.1-4.
(3) Includes the weight of fuel assembly Poison Rods installed for 10 CFR Part 50 criticality control in spent fuel pool racks.
(4) Minimum cooling time is the longer of that given in Table A.1-5 for a given burnup and enrichment of a fuel assembly and that calculated via the decay heat equation based on the restrictions provided in Figures A.1-2, A.1-3 or A.1-4.
(5) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1-5, when 5 or more damaged fuel assemblies are loaded.

Table A.1-3
PWR Fuel Assembly Design Characteristics

Assembly Class	CE 16x16
Parameters[1]	
Assembly Length	See Table A.1-1or A.1-2
Max. Initial ^{235}U Enrichment (wt. %)	4.85
Fissile Material	UO_2, or $(U, Er)O_2$, or $(U, Gd)O_2$
Number of Fuel Rods	≤ 236
Fuel Rod Pitch (in)	≤ 0.506
Fuel Rod O.D. (in)	≥ 0.380
Clad Thickness (in)	≥ 0.023
Fuel Pellet O.D., (in)	≤ 0.326
Number of Guide/Instrument Tubes	≤ 5

Notes:

(1) The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within the CE 16x16 fuel assembly class.

Table A.1-4
Maximum Fuel Enrichment v/s Neutron Poison Requirements for the 24PT4 DSC

Storage Configuration	Maximum No. of Damaged Fuel Assemblies[(1)]	Maximum ^{235}U Fuel Enrichment (wt. %)	DSC Basket, Minimum BORAL® Areal Density (gm/cm²)	Minimum No. of Poison Rodlets Required[(2)(3)]
All Intact Fuel Assemblies	0	4.1	0.025 (Type A Basket)	0
	0	4.85	0.068 (Type B Basket)	0
Combination of Damaged and Intact Fuel Assemblies	4	4.1	0.025 (Type A Basket)	0
	4	4.85	0.068 (Type B Basket)	0
	12	3.7 (damaged) 4.1 (intact)	0.025 (Type A Basket)	0
	12	4.1 (damaged) 4.85 (intact)	0.068 (Type B Basket)	0
	12	4.1	0.025 (Type A Basket)	1[(2)] (Located in center guide tube of each intact assembly)
	12	4.85	0.068 (Type B Basket)	5[(2)] (Located in all five guide tubes of each intact assembly)

Notes:

(1) See Figure A.1-1 for location of damaged fuel assemblies within the 24PT4 DSC (Zones A and/or B only).

(2) Poison rodlets are only required for a specific DSC configuration with a payload of 5-12 damaged assemblies in combination with maximum fuel enrichment levels as shown. The poison rodlets are to be located within the guide tubes of the Zone C intact assemblies as shown in Figure A.1-1.

(3) The minimum diameter of the poison rodlet is 0.55 inches (1.4 cm) with sufficient length to cover the active fuel length. The minimum diameter of the absorber material in the rodlet is 0.35 inches (0.9 cm) with a minimum linear loading of 0.70 grams B_4C per cm.

Table A.1-5
PWR Fuel Qualification Table for the 24PT4 DSC

(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																														
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
15	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
20	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
25	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
28	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
30	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
32	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
34	9.5	9.5	9.0	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
36	11.5	11.0	10.5	10.0	9.5	9.0	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
38											8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
39											9.0	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
40											9.5	9.5	9.0	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
41											10.5	10.0	9.5	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
42											11.0	10.5	10.0	10.0	9.5	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0
43																9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5
44																10.5	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0
45																					10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 year for fuel assemblies in the 12 peripheral locations of the canister with cooling times less than 11 years. For fuel assemblies with cooling times greater than 11 years or in the center of the basket, no adjustment is required.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% ^{235}U is unacceptable for transport.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 8-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt.% ^{235}U and a burnup of 41.5 GWd/MTU is acceptable for transport after a 8.0-year cooling time as defined by 4.8 wt.% ^{235}U (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- When loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.

Table A.1-6
PWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -44.8 + 41.6*X1 - 37.1*X2 + 0.611*X1^2 - 6.80*X1*X2 + 24.0*X2^2$$
$$DH = F1*Exp(\{[1-(1.8/X3)]* -0.575\}*[(X3-4.5)^{0.169}]*[(X2/X1)^{-0.147}]) + 20$$

where,

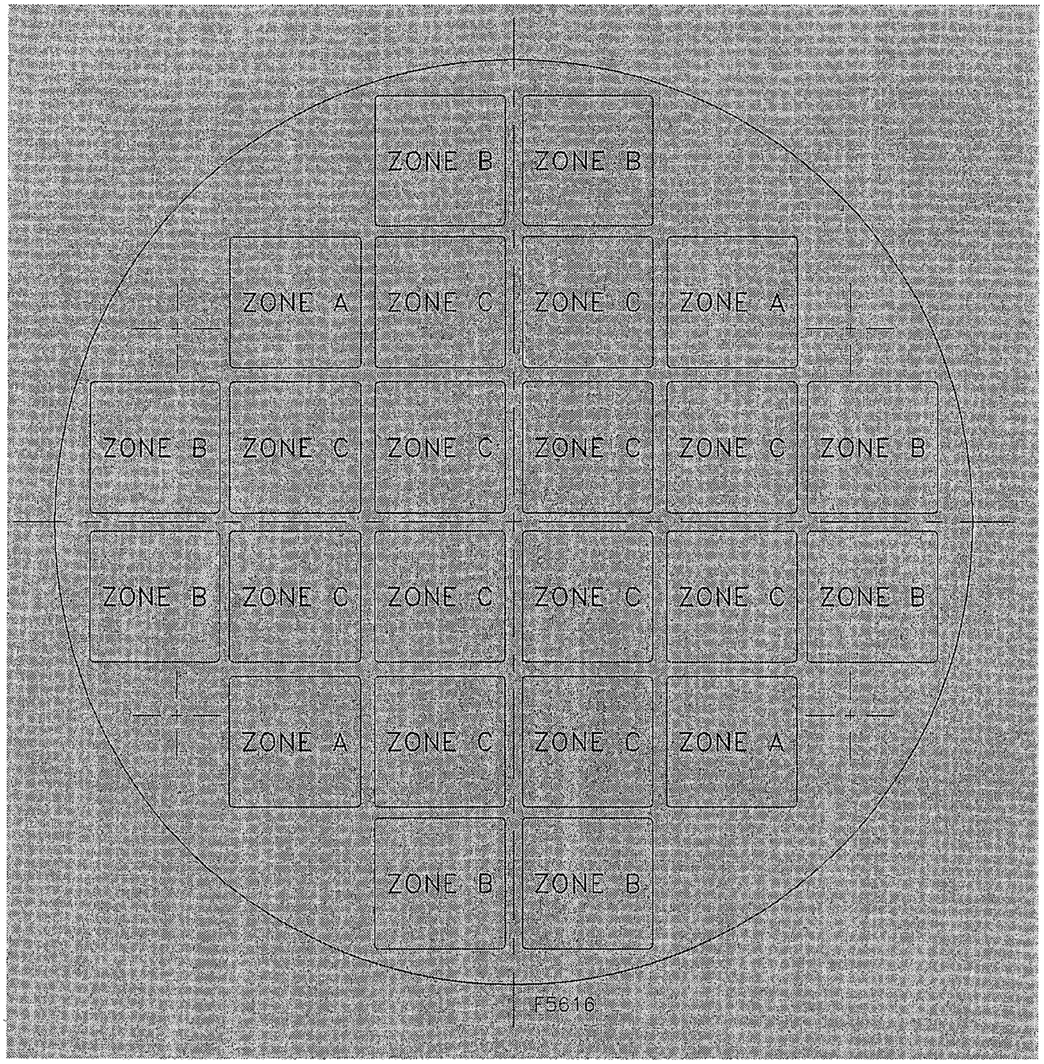
F1 Intermediate Function

X1 Assembly Burnup in GWD/MTU

X2 Initial Enrichment in wt.% ^{235}U

X3 Cooling Time in Years (minimum 7 years)

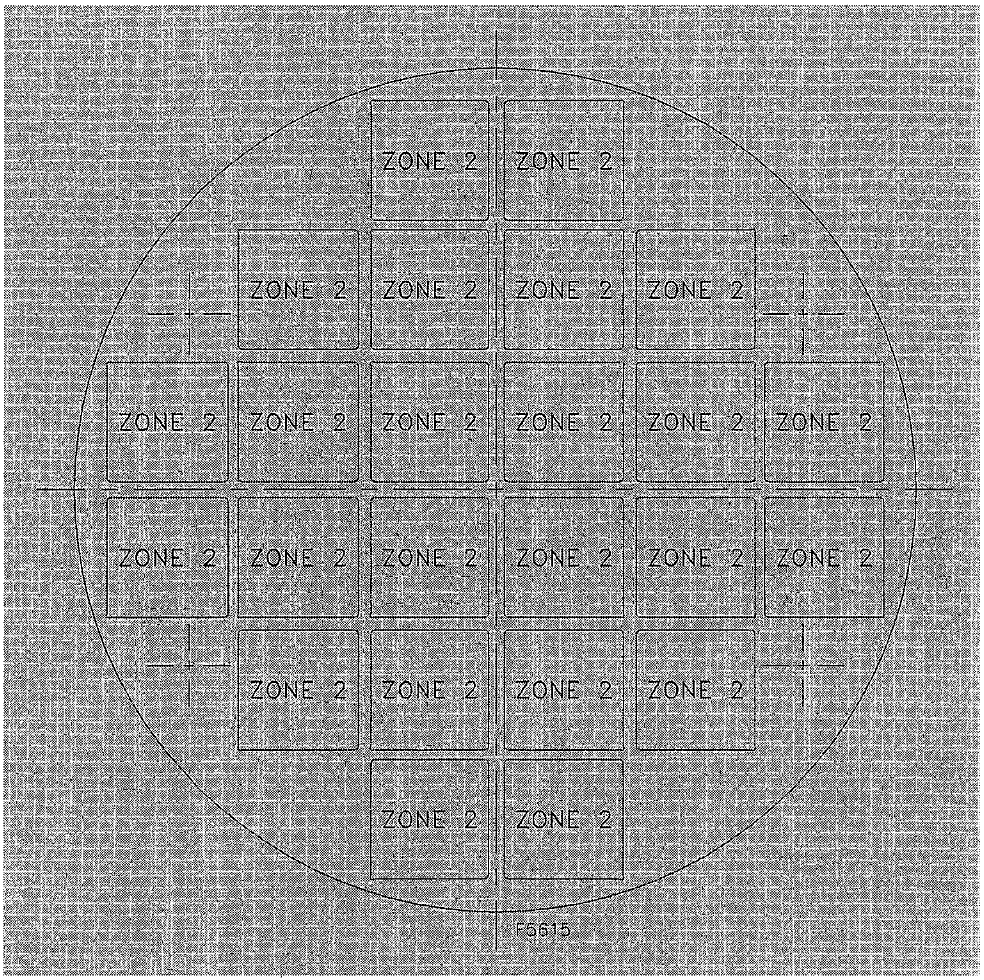
Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.1-5.



Notes:

1. Locations identified as Zone A are for placement of up to 4 damaged fuel assemblies.
2. Locations identified as Zone B are for placement of up to 8 additional damaged fuel assemblies (Maximum of 12 damaged fuel assemblies allowed, Zones A and B combined).
3. Locations identified as Zone C are for placement of up to 12 intact fuel assemblies, including 4 empty slots in the center as shown in Figure A.1-4.
4. Poison Rodlets are to be located in the guide tubes of intact fuel assemblies placed in Zone C only per Table A.1-4.

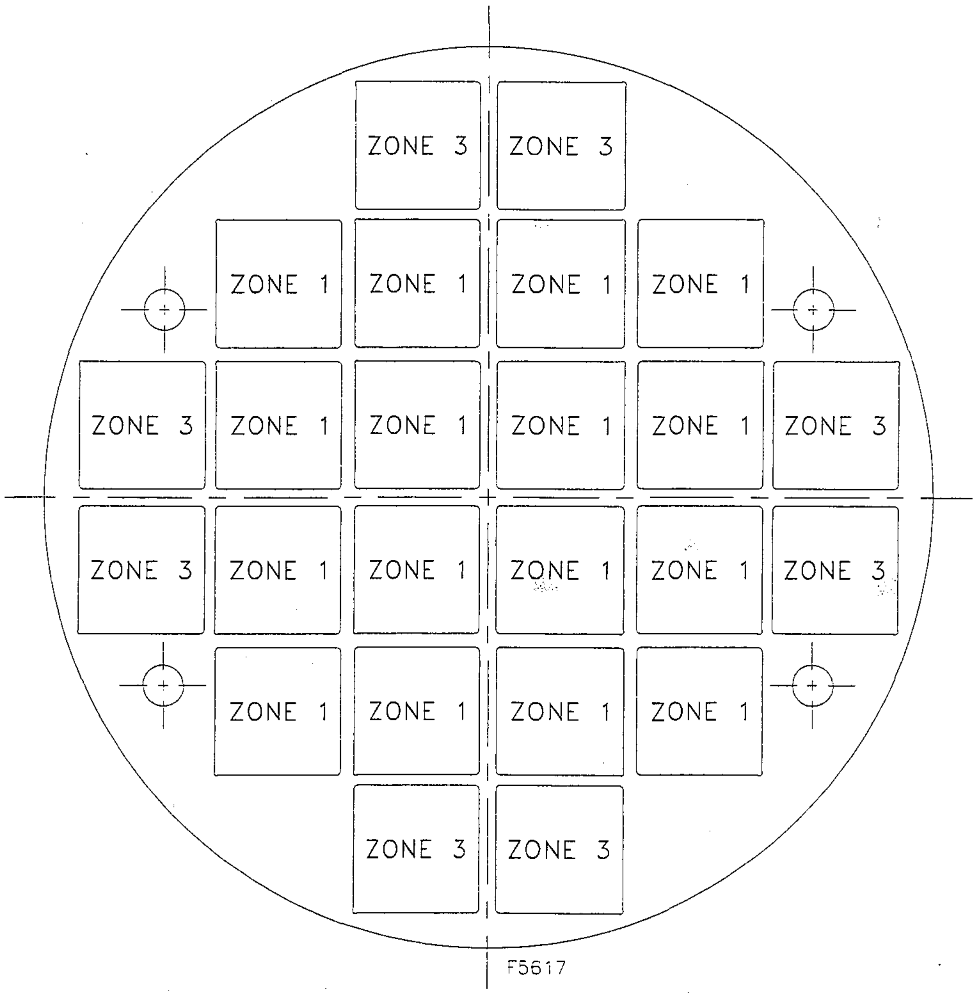
Figure A.1-1
Location of Failed Fuel Cans Inside the 24PT4 DSC



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts/FA)(1)	NA	1.0	NA	NA
Maximum Decay Heat per Zone (kW)	NA	24.0	NA	NA
Maximum Decay Heat per DSC (kW)	24.0			

(1) Decay heat per fuel assembly shall be determined using Table A.1-6.

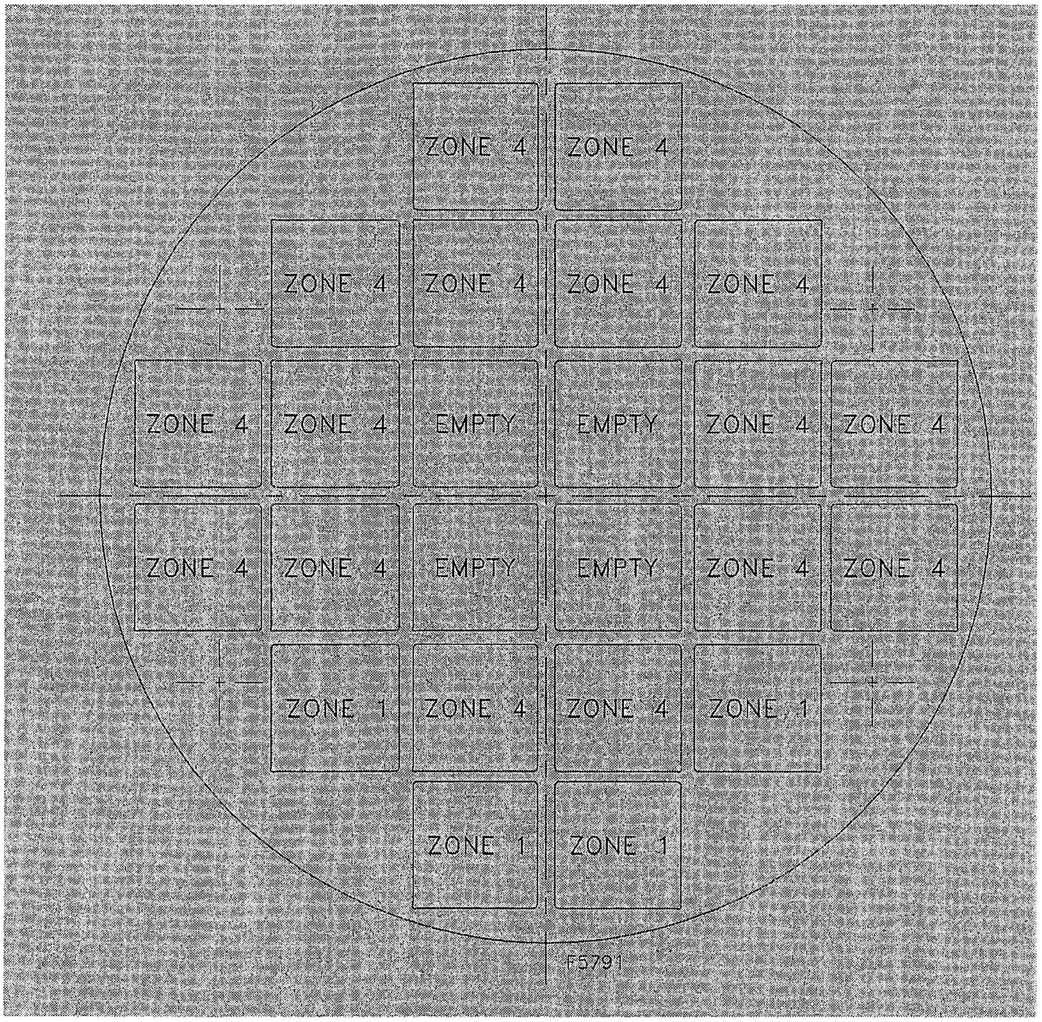
Figure A.1-2
Heat Load Configuration No. 1 for the 24PT4 DSC



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts/FA)(1)	0.9	NA	1.2	NA
Maximum Decay Heat per Zone (kW)	14.4	NA	9.6	NA
Maximum Decay Heat per DSC (kW)	24.0			

(1) Decay heat per fuel assembly shall be determined using Table A.1-6.

Figure A.1-3
Heat Load Configuration No. 2 for the 24PT4 DSC



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts/FA)[(1)]	0.9	NA	NA	1.26
Maximum Decay Heat per Zone (kW)	3.6	NA	NA	20.16
Maximum Decay Heat per DSC (kW)	24.0			

(1) Decay Heat per fuel assembly shall be determined using Table A.1-6.

Figure A.1-4
Heat Load Configuration No. 3 for the 24PT4 DSC

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MP197HB Packaging Contents Loaded with NUHOMS®-61BT DSC

(1) Type and Form of Material

(a) Intact or damaged irradiated BWR fuel assemblies with or without channels which meet specifications listed in Table A.7-1 are authorized for transportation in the NUHOMS®-61BT DSC. Damaged fuel is restricted to the 7x7 and 8x8 designs only.

(b) For maximum assembly average burnup, minimum cooling time and decay heat limits, the fuel assemblies shall meet all the requirements of the cross referenced tables listed in Table A.7-1. The fuel to be transported in the 61BT DSC is limited to a maximum lattice average initial enrichment of 4.4 wt.% ^{235}U for intact fuel (4.0 wt.% ^{235}U for damaged fuel) and a minimum of 1.4 wt.% ^{235}U. The maximum allowable assembly average burnup is given as a function of lattice average initial enrichment but does not exceed 40,000 MWd/MTU. The minimum cooling time is 7 years.

(c) The NUHOMS®-61BT DSC is authorized to transport BWR fuel assemblies arranged in one heat load zoning configuration with a maximum decay heat of 0.3 kW per assembly and a maximum heat load of 18.3 kW per canister. The heat load zoning configuration is shown in Figure A.7-1.

(d) The NUHOMS®-61BT DSC has three basket configurations: A, B and C based on the boron content in the poison plates as shown in Table A.7-3. The poison plates are constructed from borated aluminum, or an aluminum/boron carbide metal matrix composite (MMC), or Boral® and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control. The maximum lattice average initial enrichment authorized for Type A, B and C NUHOMS®-61BT DSCs is shown in Table A.7-3. Damaged BWR fuel assemblies shall only be transported in Type C NUHOMS®-61BT DSCs with end caps installed on each of the four corner 2x2 compartment assemblies. The locations are shown in Figure A.7-2.

(2) Maximum Quantity of Material per Package

(a) The quantity of material authorized for transport is (i) up to 61 intact or (ii) up to 16 damaged and balance intact BWR fuel assemblies with or without channels. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For materials described in A.7(1) above, the approximate maximum payload is 43,005 lbs.

Table A.7-1
BWR Fuel Specification for Fuel to be Transported in the NUHOMS®-61BT DSC

PHYSICAL PARAMETERS:	
Fuel Design	Intact or damaged unconsolidated 7x7, 8x8, 9x9, or 10x10 intact BWR fuel assemblies manufactured by General Electric or Exxon/ANF or reload fuel manufactured by the same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.7-2.
Fuel Damage	Damaged BWR fuel assemblies are 7x7 and 8x8 fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly needs to be handled by normal means. Damaged fuel may only be transported in the "Type C" NUHOMS®-61BT Canister. Damaged fuel is restricted to the 7x7 and 8x8 designs only. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Channels	Fuel may be transported with or without fuel channels, channel fasteners, or finger springs
No. of Intact Assemblies	≤61
No. and Location of Damaged Assemblies	Up to sixteen (16) damaged fuel assemblies with balance intact or dummy assemblies, are authorized for transport in the 61BT DSCs. Damaged fuel assemblies may only be transported in the 2x2 compartments as shown in Figure A.7-2. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus fuel channel weight	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS(1): Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.7-3
Fuel Assembly Average Burnup and minimum Cooling Time(1) (3)	Per Table A.7-4 and decay heat restrictions below
Decay Heat(1)(2)	0.300 kW/Assembly calculated per Table A.7-5

(1) Minimum cooling time is the longer of that given in Table A.7-4; that calculated via the decay heat equation given in Table A.7-5 to meet the 0.300 kW/assembly limit.
(2) For FANP9 9x9-2 fuel assemblies, the maximum decay heat is limited to 0.21 kW/assembly.
(3) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.7-4, when 5 or more damaged fuel assemblies are loaded.

Table A.7-2
BWR Fuel Assembly Design Characteristics(1) for the NUHOMS®-61BT DSC

Transnuclear ID	7x7-49/0	8x8-63/1	8x8-62/2	8x8-60/4	8x8-60/1	9x9-74/2
Initial Design or Reload Fuel Designation	GE1	GE4	GE-5	GE8 Type II	GE9	GE11
	GE2		GE-Pres		GE10	GE13
	GE3		GE-Barrier			
			GE8 Type I			
			FANP 8x8-2			
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 49	≤ 63	≤ 62	≤ 60	≤ 60	≤ 74
Initial Uranium Content (kg)	≤ 198	≤ 192	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.738	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.566
Pellet Diameter (in)	≤ 0.487	≤ 0.416	≤ 0.411	≤ 0.411	≤ 0.411	≤ 0.376
Clad Outer Diameter (in)	≥ 0.563	≥ 0.493	≥ 0.483	≥ 0.483	≥ 0.483	≥ 0.440
Clad Thickness (in)	≥ 0.032	≥ 0.034	≥ 0.032	≥ 0.032	≥ 0.032	≥ 0.028

Transnuclear ID	10x10-92/2	7x7-49/0Z	7x7-48/1Z	8x8-60/4Z	FANP 9x9	Siemens
Initial Design or Reload Fuel Designation	GE12	ENC-IIIA	ENC-III	ENC Va	FANP 9x9-72	QFA 9x9
	GE14		ENC-IIIE	ENC Vb	FANP 9x9-79	
			ENC-IIIF		FANP 9x9-80	
					FANP 9x9-81	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 92	≤ 49	≤ 48	≤ 60	≤ 81	≤ 72
Initial Uranium Content (kg)	≤ 192	≤ 198	≤ 198	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.738	≤ 0.738	≤ 0.642	≤ 0.572	≤ 0.570
Pellet Diameter (in)	≤ 0.345	≤ 0.491	≤ 0.491	≤ 0.420	≤ 0.357	≤ 0.374
Clad Outer Diameter (in)	≥ 0.404	≥ 0.570	≥ 0.570	≥ 0.501	≥ 0.424	≥ 0.433
Clad Thickness (in)	≥ 0.026	≥ 0.035	≥ 0.035	≥ 0.035	≥ 0.030	≥ 0.026

Notes:

(1) The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type). Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.

Table A.7-3
BWR Fuel Assembly Poison Material Design Requirements for 61BT DSC

NUHOMS®-61BT DSC Type	Maximum Lattice Average Enrichment(1) (wt.% ^{235}U)	Borated Aluminum or MMC Minimum B10 Content in Poison Plates (gm/cm^2)	BORAL® Minimum B10 Content in Poison Plates (gm/cm^2)
Intact Fuel Assemblies			
A	3.7	0.021	0.025
B	4.1	0.032	0.038
C	4.4	0.040	0.048
Up to 4 Damaged Assemblies			
C	4.4	0.040	0.048
Five or more Damaged Assemblies			
C	3.2	0.040	0.048

(1) Maximum pin enrichment is 5.0 wt.% ^{235}U in all cases.

Table A.7-4
BWR Fuel Qualification Table for the NUHOMS®-61BT DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU	Initial Enrichment																													
	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4
10	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
15	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
20	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
25	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
28	Not Acceptable or Not Analyzed				8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
30					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
32					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
34					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
36					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
38					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
39					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
40					8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8

Notes:

- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 1.4 and greater than 4.4 wt.% ^{235}U is unacceptable for transportation.
- Fuel with a burnup greater than 40 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 8 years cooling.
- Example: An assembly with an initial enrichment of 4.15 wt.% ^{235}U and a burnup of 31.5 GWd/MTU is acceptable for transport after a 8-year cooling time as defined by 4.1 wt.% ^{235}U (rounding down) and 32 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- When loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.

Table A.7-5
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as: $F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$ $DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$ where, F1 Intermediate Function X1 Assembly Burnup in GWD/MTU X2 Initial Enrichment in wt.% ^{235}U X3 Cooling Time in Years (minimum 7 years)

Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.7-4.

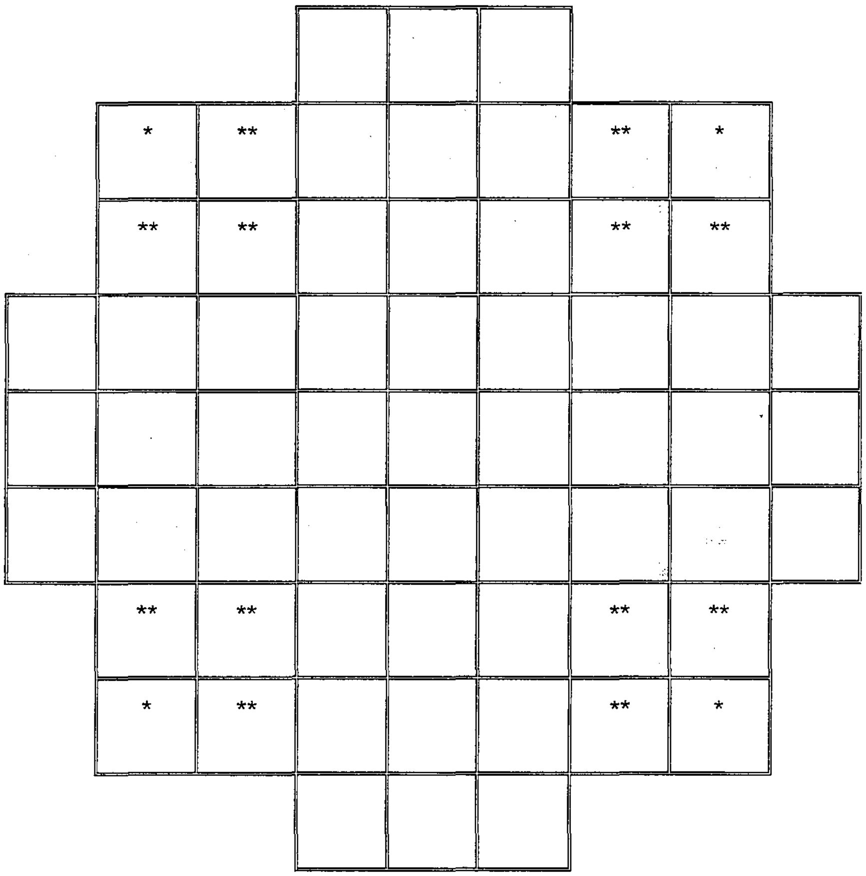
			Z1	Z1	Z1			
	Z1 *	Z1 *	Z1	Z1	Z1	Z1 *	Z1 *	
	Z1 *	Z1 *	Z1	Z1	Z1	Z1 *	Z1 *	
Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1
Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1
Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1	Z1
	Z1 *	Z1 *	Z1	Z1	Z1	Z1 *	Z1 *	
	Z1 *	Z1 *	Z1	Z1	Z1	Z1 *	Z1 *	
			Z1	Z1	Z1			

* Denotes only locations where damaged fuel assembly can be transported

	Zone 1
Maximum Decay Heat (kW/FA)[(1)]	0.30
Maximum Decay Heat per Zone (kW)	18.3
Maximum Decay Heat per DSC (kW)	18.3

(1) Decay heat per fuel assembly shall be determined per Table A.7-5.

Figure A.7-1
Heat Load Zoning Configuration for 61BT DSCs



*	Corner Locations See Note 1	**	Interior Locations See Note 2

Note 1: These corner locations shall only be used to load up to four damaged assemblies with the remaining intact in a 61BT Basket. The maximum lattice average initial enrichment of assemblies (damaged or intact transported in the 2x2 compartment assemblies) is limited to that applicable to "Up to 4 Damaged Assemblies" row of Table A.7-3.

Note 2: If loading more than four damaged assemblies, place first four damaged assemblies in the corner locations per Note 1, and up to 12 additional damaged assemblies in these interior locations, with the remaining intact in a 61BT Basket. The maximum lattice average initial enrichment of assemblies (damaged or intact transported in the 2x2 compartment assemblies) is limited to that applicable to "Five or More Damaged Assemblies" row of Table A.7-3.

Figure A.7-2
Location of Damaged and Failed Fuel Assemblies Inside 61BT DSC

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MP197HB Packaging Contents Loaded with NUHOMS®-61BTH DSC

(1) Type and Form of Material

(a) Intact or damaged or failed irradiated BWR fuel assemblies with or without channels which meet specifications listed in Table A.8-2 are authorized for transportation in the NUHOMS®-61BTH DSC. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps to assure retrievability. Failed fuel assembly/fuel debris is to be encapsulated in an individual failed fuel can [See Drawing NUH61BTH-71-1105, Rev. 0 (2 sheets)] provided with a welded bottom closure and a removable top closure which assures retrievability of a loaded FFC.

(b) For maximum assembly average burnup, minimum cooling time and decay heat limits, the fuel assemblies shall meet the all the requirements of the cross referenced tables and figures listed in Table A.8-2. The fuel to be transported in the 61BTH DSC is limited to a maximum lattice average initial enrichment of 5.0 wt.% ^{235}U for intact fuel (a maximum of 3.6 wt.% ^{235}U for 16 damaged fuel assemblies and 3.5 wt.% ^{235}U for 4 failed fuel assemblies) and a minimum of 0.7 wt.% ^{235}U. The maximum allowable assembly average burnup is given as a function of lattice average initial enrichment but does not exceed 45,000 MWd/MTU. The minimum cooling time is 7 years.

(c) Three separate types of 61BTH DSC designs are provided. Type 1 61BTH DSC baskets have steel rails while the Type 2 61BTH DSC baskets have aluminum rails. 61BTHF DSC is a modified version of the 61BTH DSC designed to accommodate up to 4 FFCs.

(d) The NUHOMS®-61BTH Type 1 DSC is authorized to transport BWR fuel assemblies with a maximum decay heat of 0.54 kW per assembly and a maximum heat load of 22 kW per DSC in four configurations as shown in Figures A.8-1 through A.8-4. The NUHOMS®-61BTH Type 2 and NUHOMS®-61BTHF DSC are authorized to transport BWR fuel assemblies with a maximum decay heat of 0.7 kW per assembly and a maximum heat load of 24 kW per DSC in eight configurations as shown in Figures A.8-1 through A.8-8.

(e) The NUHOMS®-61BTH DSC has six basket configurations: A, B, C, D, E and F based on the boron content in the poison plates. The poison plates are constructed from borated aluminum, or an aluminum/boron carbide metal matrix composite (MMC), or Boral® and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control. The maximum lattice average initial enrichment authorized for Type A, B, C, D, E, and F NUHOMS®-61BTH DSCs is shown in Table A.8-4 for intact fuel and Table A.8-5 for damaged and failed fuel.

(2) Maximum Quantity of Material per Package

(a) The quantity of material authorized for transport is (i) up to 61 intact or (ii) up to 16 damaged and balance intact or (iii) up to 4 failed and up to 12 damaged and balance intact BWR fuel assemblies with or without channels. The location of damaged and failed fuel assemblies within the DSC basket are shown in Figure A.8-9. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For materials described in A.8(1) above, the approximate maximum payload is 43,100 lbs

Table A.8-1

(not used)

Table A.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC
(Part 1 of 2)

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged or failed 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.8-3. Damaged fuel assemblies beyond the definition contained below are not authorized for transport in damaged fuel locations shown in Figure A.8-9.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel rods is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Failed Fuel	Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly can not be handled by normal means. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris, not contained in a rod storage basket may also be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at a position of at least 10" above the top of the bottom shield plug of the DSC. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its content shall be less than 705 lb.
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods	4
• Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly	4
• Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	61
No. of Intact Assemblies	≤61

Table A.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC
(Part 2 of 2)

PHYSICAL PARAMETERS:	
No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 61BTH DSC. Damaged fuel assemblies may only be transported in the 2x2 compartments as shown in Figure A.8-9. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	Up to 4 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Failed fuel assemblies are to be placed as shown in Figure A.8-9. Failed fuel assembly/fuel debris is to be encapsulated in an individual failed fuel can (FFC) provided with a welded bottom closure and a removable top closure.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lb
THERMAL/RADIOLOGICAL PARAMETERS[1]: Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.8-4 or Table A.8-5.
Fuel Assembly Average Burnup and minimum Cooling Time[2]	Type 1 Per Table A.8-6.
	Type 2 Per Table A.8-7.
Decay Heat per DSC	≤22.0 kW for Type 1 DSC, per Figures A.8-1 through A.8-4
	≤24.0 kW for Type 2 DSC, per Figures A.8-1 through A.8-8
Minimum B10 Content in Poison Plates	Per Table A.8-4 or Table A.8-5.

Notes:

(1) Minimum cooling time is the longer of that given in Table A.8-6, Table A.8-7, and that calculated via the decay heat equation given in Table A.8-8 based on the restrictions provided in Figures A.8-1 through A.8-8.

(2) An additional cooling time of 8 years is required for damaged fuel assemblies (and failed fuel assemblies, if applicable) in addition to that obtained from Table A.8-6 or Table A.8-7, when 5 or more damaged fuel assemblies (or a combination of damaged and failed fuel assemblies, if applicable) are loaded.

Table A.8-3
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-61BTH DSC
(part 1 of 2)

Transnuclear ID	7x7-49/0	8x8-63/1	8x8-62/2	8x8-60/4	8x8-60/1	9x9-74/2
Initial Design or Reload Fuel Designation	GE1	GE4	GE-5	GE8 Type II	GE9	GE11
	GE2		GE-Pres		GE10	GE13
	GE3		GE-Barrier			
			GE8 Type I			
			FANP 8x8-2			
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 49	≤ 63	≤ 62	≤ 60	≤ 60	≤ 74
Initial Uranium Content (kg)	≤ 198	≤ 192	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.738	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.566
Pellet Diameter (in)	≤ 0.487	≤ 0.416	≤ 0.411	≤ 0.411	≤ 0.411	≤ 0.376
Clad Outer Diameter (in)	≥ 0.563	≥ 0.493	≥ 0.483	≥ 0.483	≥ 0.483	≥ 0.440
Clad Thickness (in)	≥ 0.032	≥ 0.034	≥ 0.032	≥ 0.032	≥ 0.032	≥ 0.028

Transnuclear ID	10x10-92/2	7x7-49/0Z	7x7-48/1Z	8x8-60/4Z	FANP 9x9	Siemens
Initial Design or Reload Fuel Designation	GE12	ENC-IIIA	ENC-III	ENC Va	FANP 9x9-72	QFA 9x9
	GE14		ENC-IIIE	ENC Vb	FANP 9x9-79	
			ENC-IIIF		FANP 9x9-80	
					FANP 9x9-81	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 92	≤ 49	≤ 48	≤ 60	≤ 81	≤ 72
Initial Uranium Content (kg)	≤ 192	≤ 198	≤ 198	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.738	≤ 0.738	≤ 0.642	≤ 0.572	≤ 0.570
Pellet Diameter (in)	≤ 0.345	≤ 0.491	≤ 0.491	≤ 0.420	≤ 0.357	≤ 0.374
Clad Outer Diameter (in)	≥ 0.404	≥ 0.570	≥ 0.570	≥ 0.501	≥ 0.424	≥ 0.433
Clad Thickness (in)	≥ 0.026	≥ 0.035	≥ 0.035	≥ 0.035	≥ 0.030	≥ 0.026

Table A.8-3
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-61BTH DSC
(part 2 of 2)

Transnuclear ID	10x10-91/1	ABB-8x8	ABB-10x10-1	ABB-10x10-2
Initial Design or Reload Fuel Designation	ATRIUM-10	SVEA-64	SVEA-92	SVEA-100
	ATRIUM-10XM		SVEA-96	
			SVEA-96 +	
			OPTIMA	
			OPTIMA 2	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 91	≤ 64	≤ 96	≤ 100
Initial Uranium Content (kg)	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.622	≤ 0.512	≤ 0.512
Pellet Diameter (in)	≤ 0.350	≤ 0.411	≤ 0.346	≤ 0.375
Clad Outer Diameter (in)	≥ 0.395	≥ 0.462	≥ 0.378	≥ 0.443
Clad Thickness (in)	≥ 0.023	≥ 0.027	≥ 0.024	≥ 0.024

Notes:

(1) The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type). Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.

Table A.8-4
BWR Fuel Assembly Initial Lattice Average Initial enrichment v/s Minimum B10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Intact Fuel)

61BTH DSC Type	Basket Type	Maximum Lattice Average Initial enrichment(1) (wt% ^{235}U)	Minimum B10 Areal Density, gram/cm^2	
			Borated Aluminum/MMC	Boral®
1	A	3.7	0.021	0.025
	B	4.1	0.032	0.038
	C	4.4	0.040	0.048
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075
2	A	3.7	0.022	0.027
	B	4.1	0.032	0.038
	C	4.4	0.042	0.050
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075

(1) For LaCrosse fuel assemblies, the enrichment shall be reduced by 0.1 wt.% ^{235}U.

Table A.8-5
BWR Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Damaged/Failed Fuel)

61BTH DSC Type	Basket Type	Maximum Lattice Average Initial Enrichment (wt% ^{235}U) (1): Up to 4 Damaged Assemblies(2)(3)	Maximum Lattice Average Initial Enrichment (wt% ^{235}U) (1): Five or More Damaged Assemblies (16 Maximum)(2)	Minimum B10 Areal Density, gram/cm^2: Borated Aluminum/MMC	Minimum B10 Areal Density, gram/cm^2: Boral®
1	A	3.7	2.80	0.021	0.025
	B	4.1	3.10	0.032	0.038
	C	4.4	3.20	0.040	0.048
	D	4.6	3.40	0.048	0.058
	E	4.8	3.50	0.055	0.066
	F	5.0	3.60	0.062	0.075
2	A	3.7	2.80	0.022	0.027
	B	4.1	3.10	0.032	0.038
	C	4.4	3.20	0.042	0.050
	D	4.6	3.40	0.048	0.058
	E	4.8	3.50	0.055	0.066
	F	5.0	3.60	0.062	0.075
61BTH DSC Type	**Basket Type**	**Maximum Lattice Average Initial Enrichment (wt% ^{235}U) (1): Up to 4 Failed Assemblies (Corner Locations)(3)(4)**	**Maximum Lattice Average Initial Enrichment (wt% ^{235}U) (1): Up to 4 Failed Assemblies (Corner Locations) and up to 12 Damaged Assemblies(2)(4)**	**Minimum B10 Areal Density, gram/cm^2: Borated Aluminum/MMC**	**Minimum B10 Areal Density, gram/cm^2: Boral®**
2	A	3.7	2.80	0.022	0.027
	B	4.0	3.10	0.032	0.038
	C	4.4	3.20	0.042	0.050
	D	4.6	3.40	0.048	0.058
	E	4.8	3.40	0.055	0.066
	F	5.0	3.50	0.062	0.075

Note
(1) For LaCrosse fuel assemblies, the enrichment shall be reduced by 0.1 wt.% ^{235}U
(2) See Figure A.8-9 for the location of damaged assemblies within the 61BTH DSC.
(3) Maximum Pellet Enrichment 5.0 wt.% ^{235}U
(4) Failed fuel assemblies are allowed only in the 61BTH Type 2 DSC. See Figure A.8-9 for the location of failed assemblies within the 61BTH Type 2 DSC.

Table A.8-6
BWR Fuel Qualification Table for NUHOMS®-61BTH Type 1 DSC

(Minimum required years of cooling time after reactor core discharge)

BU, GWD/	Lattice Average Initial U-235 Enrichment, wt %																																	
MTU	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
15	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
20	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
23	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
25	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
28	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
30	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
32				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
34				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
36				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
38				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
39				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
40										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
41										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
42										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
43										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
44										8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
45										12.0	11.5	8.5	10.0	9.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0

Note: Explanatory notes and limitations regarding the use of this table follow Table A.8-7.

Table A.8-7
BWR Fuel Qualification Table for NUHOMS®-61BTH Type 2 DSC

(Minimum required years of cooling time after reactor core discharge)

BU, GWD/ MTU	Lattice Average Initial U-235 Enrichment, wt %																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
15	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
20	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
23	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
25	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
28	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
30	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
32				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
34				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
36				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
38				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
39				8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
40										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
41										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
42										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
43										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
44										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
45										8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0

Note: Explanatory notes and limitations regarding the use of this table follow Table A.8-7.

Notes: Tables A.8-6 and Table A.8-7:

- Burnup = assembly average burnup
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with a lattice average initial enrichment less than 0.9 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% ^{235}U is unacceptable for transportation.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 8-years cooling.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 year for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- The cooling times for failed, damaged, and intact assemblies are identical. However, when loading five or more damaged fuel assemblies per DSC (or a combination of damaged and failed fuel assemblies, if applicable), an additional cooling time of 8 years is required for only damaged fuel assemblies (and failed fuel assemblies, if applicable).
- Example: An assembly with an initial enrichment of 4.85 wt.% ^{235}U and a burnup of 41.5 GWd/MTU is acceptable for transport after a 8-year cooling time as defined by 4.8 wt.% ^{235}U (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).

Table A.8-8
BWR Assembly Decay Heat for Heat Load Configurations

The decay heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate function

X1 Assembly burnup in GWD/MTU

X2 Initial enrichment in wt.% ^{235}U

X3 Cooling time in years (minimum 7 years)

Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies (or a combination of damaged and failed fuel assemblies, if applicable) from shielding requirements is per Table A.8-6 for Type 1 DSC and A.8-7 for Type 2 DSC.

			Z3	Z3	Z3			
	Z3	Z3	Z3	Z3	Z3	Z3	Z3	
	Z3	Z3	Z3	Z3	Z3	Z3	Z3	
Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3
Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3
Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3	Z3
	Z3	Z3	Z3	Z3	Z3	Z3	Z3	
	Z3	Z3	Z3	Z3	Z3	Z3	Z3	
			Z3	Z3	Z3			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (3)	NA	NA	0.393	NA	NA	NA
Maximum Decay Heat per Zone (kW)	NA	NA	22.0	NA	NA	NA
Maximum Decay Heat per DSC (kW)	22.0(3)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) This configuration is not allowed for a 61BTH Type 1 basket with MMC or Boral® Poison Plates.
(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 15.4 kW per DSC for HLZC No. 1.

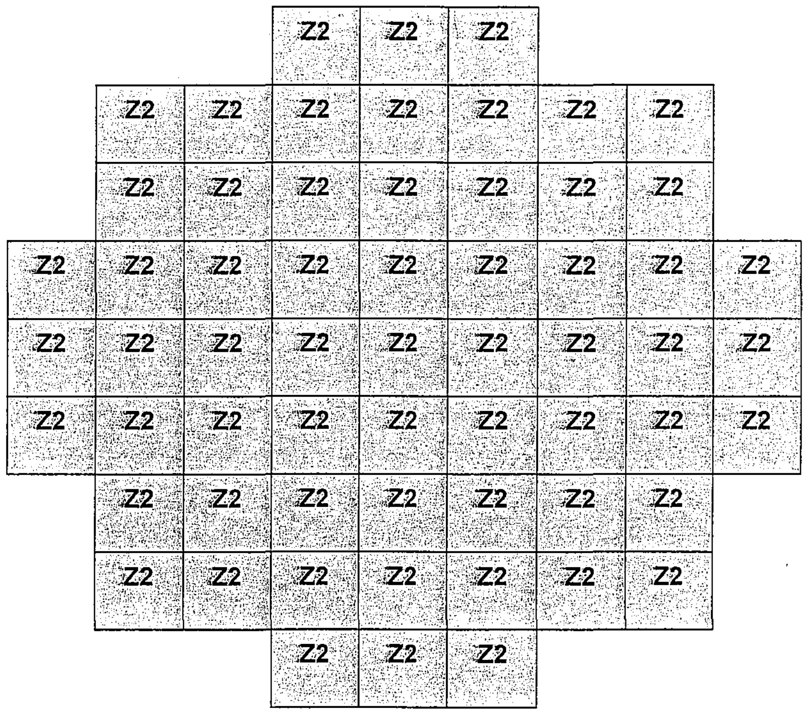
Figure A.8-1
Heat Load Zoning Configuration No. 1 for Type 1 or Type 2 61BTH DSCs(2)

			Z5	Z5	Z5			
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
	Z4	Z2	Z2	Z2	Z2	Z2	Z4	
Z5	Z4	Z2	Z2	Z2	Z2	Z2	Z4	Z5
Z5	Z4	Z2	Z2	Z2	Z2	Z2	Z4	Z5
Z5	Z4	Z2	Z2	Z2	Z2	Z2	Z4	Z5
	Z4	Z2	Z2	Z2	Z2	Z2	Z4	
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
			Z5	Z5	Z5			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (3)	NA	0.35	NA	0.48	0.54	NA
Maximum Decay Heat per Zone (kW)	NA	8.75	NA	11.52	6.48	NA
Maximum Decay Heat per DSC (kW)	22.0 (3)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) This configuration is not allowed for a 61BTH Type 1 basket with MMC or Boral® Poison Plates.
(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 15.4 kW per DSC for HLZC No. 2.

Figure A.8-2
Heat Load Zoning Configuration No. 2 for Type 1 or Type 2 61BTH DSCs(2)



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)[(1) (2)]	NA	0.35	NA	NA	NA	NA
Maximum Decay Heat per Zone (kW)	NA	19.4	NA	NA	NA	NA
Maximum Decay Heat per DSC (kW)	19.4[(2)]					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 13.58 kW per DSC for HLZC No. 3.

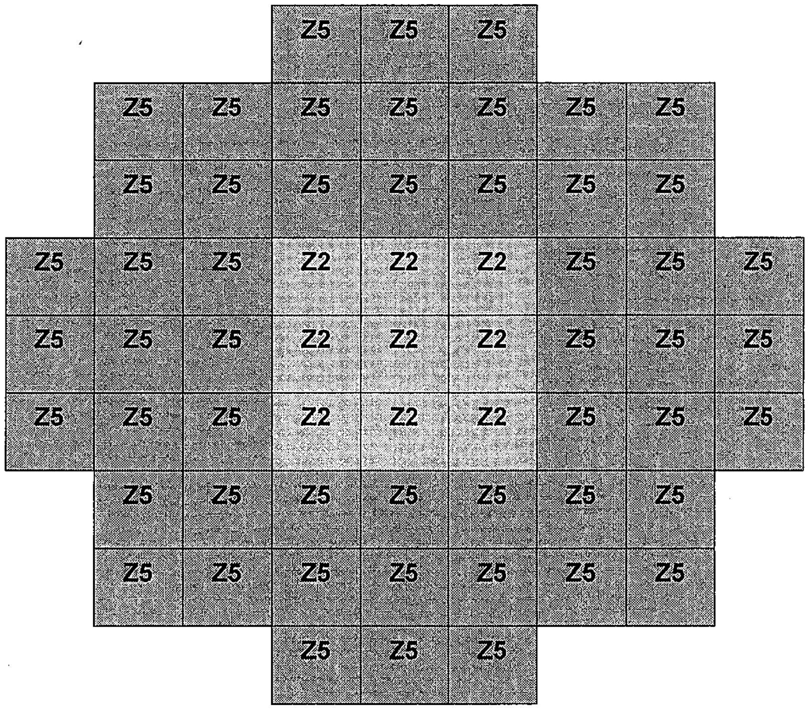
Figure A.8-3
Heat Load Zoning Configuration No. 3 for Type 1 or Type 2 61BTH DSCs

			Z5	Z5	Z5			
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
	Z4	Z2	Z2	Z2	Z2	Z2	Z4	
Z5	Z4	Z2	Z1	Z1	Z1	Z2	Z4	Z5
Z5	Z4	Z2	Z1	Z1	Z1	Z2	Z4	Z5
Z5	Z4	Z2	Z1	Z1	Z1	Z2	Z4	Z5
	Z4	Z2	Z2	Z2	Z2	Z2	Z4	
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
			Z5	Z5	Z5			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (2)	0.22	0.35	NA	0.48	0.54	NA
Maximum Decay Heat per Zone (kW)	1.98	5.60	NA	11.52	6.48	NA
Maximum Decay Heat per DSC (kW)	19.4(2)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 13.58 kW per DSC for HLZC No. 4.

Figure A.8-4
Heat Load Zoning Configuration No. 4 for Type 1 or Type 2 61BTH DSCs



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (3)	NA	0.35	NA	NA	0.54	NA
Maximum Decay Heat per Zone (kW)	NA	3.15	NA	NA	24.0	NA
Maximum Decay Heat per DSC (kW)	24.0(3)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) This configuration is not allowed for a 61BTH Type 2 basket with MMC or Boral® Poison Plates.
(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 16.8 kW per DSC for HLZC No. 5.

Figure A.8-5
Heat Load Zoning Configuration No. 5 for Type 2 61BTH DSC(2)

			Z5	Z5	Z5			
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
	Z4	Z6	Z6	Z6	Z6	Z6	Z4	
Z5	Z4	Z6	Z1	Z1	Z1	Z6	Z4	Z5
Z5	Z4	Z6	Z1	Z1	Z1	Z6	Z4	Z5
Z5	Z4	Z6	Z1	Z1	Z1	Z6	Z4	Z5
	Z4	Z6	Z6	Z6	Z6	Z6	Z4	
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
			Z5	Z5	Z5			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (3)	0.22	NA	NA	0.48	0.54	0.70
Maximum Decay Heat per Zone (kW)	1.98	NA	NA	11.52	6.48	11.20
Maximum Decay Heat per DSC (kW)	24.0(3)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) This configuration is not allowed for a 61BTH Type 1 basket with MMC or Boral® Poison Plates.
(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 16.8 kW per DSC for HLZC No. 6.

Figure A.8-6
Heat Load Zoning Configuration No. 6 for Type 2 61BTH DSC(2)

			Z5	Z5	Z5			
	Z5	Z5	Z5	Z5	Z5	Z5	Z5	
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
Z5	Z5	Z4	Z4	Z4	Z4	Z4	Z5	Z5
Z5	Z5	Z4	Z4	Z4	Z4	Z4	Z5	Z5
Z5	Z5	Z4	Z4	Z4	Z4	Z4	Z5	Z5
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
	Z5	Z5	Z5	Z5	Z5	Z5	Z5	
			Z5	Z5	Z5			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (3)	NA	NA	NA	0.48	0.54	NA
Maximum Decay Heat per Zone (kW)	NA	NA	NA	12.00	19.44	NA
Maximum Decay Heat per DSC (kW)	24.0(3)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) This configuration is not allowed for a 61BTH Type 1 basket with MMC or Boral® Poison Plates.
(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 16.8 kW per DSC for HLZC No. 7.

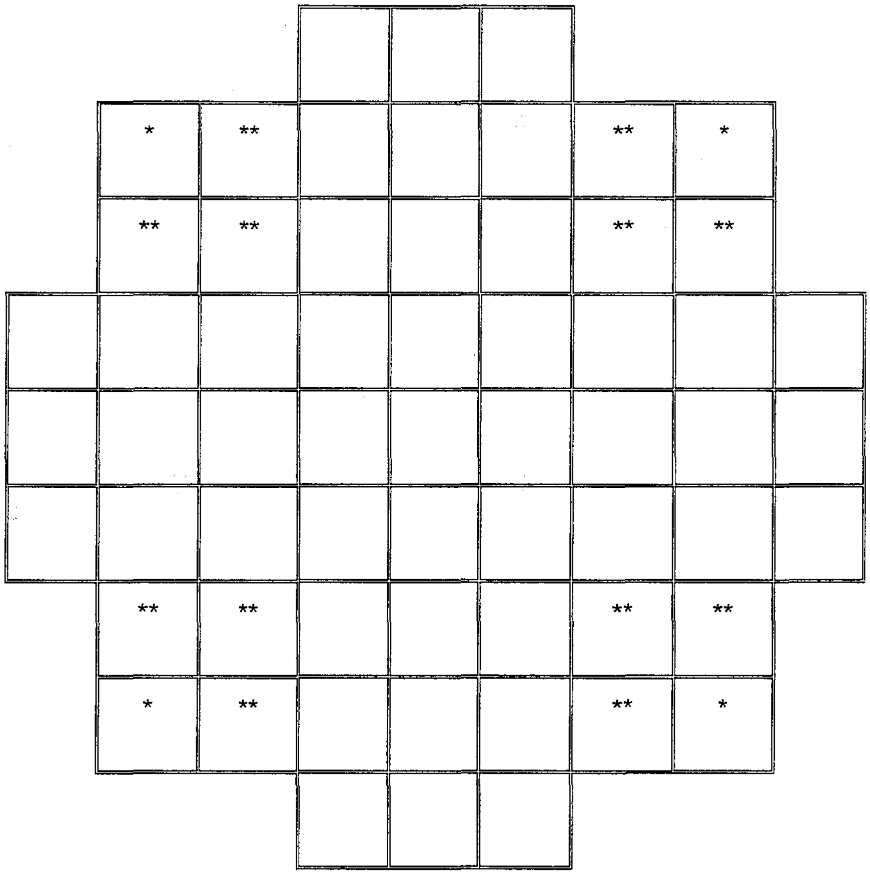
Figure A.8-7
Heat Load Zoning Configuration No. 7 for Type 2 61BTH DSC(2)

			Z5	Z5	Z5			
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
	Z4	Z3	Z3	Z3	Z3	Z3	Z4	
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
	Z4	Z3	Z3	Z3	Z3	Z3	Z4	
	Z4	Z4	Z4	Z4	Z4	Z4	Z4	
			Z5	Z5	Z5			

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Maximum Decay Heat (kW/FA)(1) (2)	NA	0.35	0.393	0.48	0.54	NA
Maximum Decay Heat per Zone (kW)	NA	3.15	6.288	11.52	6.48	NA
Maximum Decay Heat per DSC (kW)	24.0(2)					

(1) Decay heat per fuel assembly shall be determined per Table A.8-8.
(2) Reduce the maximum decay heat to 70% of the listed values for LaCrosse fuel assembly. The total decay heat for LaCrosse fuel assembly is 16.8 kW per DSC for HLZC No. 8.

Figure A.8-8
Heat Load Zoning Configuration No. 8 for Type 2 61BTH DSC



*	Corner Locations See Note 1		**	Interior Locations See Note 2

Note 1: These corner locations shall only be used to load up to four damaged or failed assemblies with the remaining intact in a 61BTH Basket. The maximum lattice average initial enrichment of assemblies (damaged or intact transported in the 2x2 compartment assemblies) is limited to the "Up to 4 Damaged Assemblies" column of Table A.8-5. For the Type 2 DSC containing failed fuel assemblies, this enrichment is limited to the "Up to 4 Failed Assemblies" column of Table A.8-5.

Note 2: If loading more than four damaged assemblies, place first four damaged assemblies in the corner locations per Note 1, and up to 12 additional damaged assemblies in these interior locations, with the remaining intact in a 61BTH Basket. The maximum lattice average initial enrichment of assemblies (damaged or intact transported in the 2x2 compartment assemblies) is limited to the "Five or More Damaged Assemblies" column of Table A.8-5. For the Type 2 DSC containing failed fuel assemblies, this enrichment is limited to the "and up to 12 Damaged Assemblies" column of Table A.8-5.

Figure A.8-9
Location of Damaged and Failed Fuel Assemblies Inside 61BTH DSC

CoC 9302 Revision 5, Appendix A.9

MP197HB Packaging Contents Loaded with NUHOMS®-69BTH DSC

(1) Type and Form of Material

(a) Intact or damaged BWR fuel assemblies with or without channels which meet specifications listed in Table A.9-1 are authorized for transportation in the NUHOMS®-69BTH DSC. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

(b) For maximum assembly average burnup, minimum cooling times and decay heat limits, the fuel assemblies shall meet the all the requirements of the cross referenced tables and figures listed in Table A.9-1. The fuel to be transported in the 69BTH DSC is limited to a maximum lattice average initial enrichment of 5.0 wt.% ^{235}U for intact fuel (a maximum of 3.4 wt.% ^{235}U for 24 damaged fuel assemblies) and a minimum of 0.7 wt.% ^{235}U. The maximum allowable assembly average burnup is given as a function of lattice average initial enrichment but does not exceed 45,000 MWd/MTU. The minimum cooling time is 6 years.

(c) The NUHOMS®-69BTH DSC is authorized to transport BWR fuel assemblies with a maximum decay heat of 0.7 kW per assembly and a maximum heat load of 32 kW per DSC in four configurations as shown in Figures A.9-2 through A.9-5.

(d) The NUHOMS®-69BTH DSC has six basket configurations: A, B, C, D, E and F based on the boron content in the poison plates. The poison plates are constructed from borated aluminum, or boron carbide/aluminum metal matrix composite (MMC), or Boral® and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control. The maximum lattice average initial enrichment authorized for Type A, B, C, D, E and F NUHOMS®-69BTH DSCs is shown in Table A.9-3 for intact fuel and damaged fuel.

(2) Maximum Quantity of Material per Package

(a) The quantity of material authorized for transport is (i) up to 69 intact or (ii) up to 24 damaged and balance intact BWR fuel assemblies with or without channels. The location of damaged fuel assemblies within the DSC basket are shown in Figure A.9-1. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For materials described in A.9(1) above, the approximate maximum payload is 48,700 lbs.

Table A.9-1
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-69BTH DSC

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.9-2. Damaged fuel assemblies beyond the definition contained below are not authorized for transport.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 4 69
No. of Intact Assemblies	≤69
No. and Location of Damaged Assemblies	Up to 24 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 69BTH DSC. Damaged fuel assemblies may only be transported in the four outer "6-compartment" arrays as shown in Figure A.9-1. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.9-3.
Allowable Heat Load Zoning Configurations for each 69BTH DSC	Per Figure A.9-2 or Figure A.9-3 or Figure A.9-4 or Figure A.9-5.
Fuel Assembly Average Burnup and minimum Cooling Time[1]	Per Table A.9-4
Decay Heat per DSC	Per Figure A.9-2 or Figure A.9-3 or Figure A.9-4 or Figure A.9-5.
Minimum B10 Content in Poison Plates	Per Table A.9-3.

[1] An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.9-4, when five or more damaged fuel assemblies are loaded.

Table A.9-2
BWR Fuel Assembly Design Characteristics(1) for the NUHOMS®-69BTH DSC
(part 1 of 2)

Transnuclear ID	7x7-49/0	8x8-63/1	8x8-62/2	8x8-60/4	8x8-60/1	9x9-74/2
Initial Design or Reload Fuel Designation	GE1	GE4	GE-5	GE8 Type II	GE9	GE11
	GE2		GE-Pres		GE10	GE13
	GE3		GE-Barrier			
			GE8 Type I			
			FANP 8x8-2			
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 49	≤ 63	≤ 62	≤ 60	≤ 60	≤ 74
Initial Uranium Content (kg)	≤ 198	≤ 192	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.738	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.566
Pellet Diameter (in)	≤ 0.487	≤ 0.416	≤ 0.411	≤ 0.411	≤ 0.411	≤ 0.376
Clad Outer Diameter (in)	≥ 0.563	≥ 0.493	≥ 0.483	≥ 0.483	≥ 0.483	≥ 0.440
Clad Thickness (in)	≥ 0.032	≥ 0.034	≥ 0.032	≥ 0.032	≥ 0.032	≥ 0.028

Transnuclear ID	10x10-92/2	7x7-49/0Z	7x7-48/1Z	8x8-60/4Z	FANP 9x9	Siemens
Initial Design or Reload Fuel Designation	GE12	ENC-IIIA	ENC-III	ENC Va	FANP 9x9-72	QFA 9x9
	GE14		ENC-IIIE	ENC Vb	FANP 9x9-79	
			ENC-IIIF		FANP 9x9-80	
					FANP 9x9-81	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 92	≤ 49	≤ 48	≤ 60	≤ 81	≤ 72
Initial Uranium Content (kg)	≤ 192	≤ 198	≤ 198	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.738	≤ 0.738	≤ 0.642	≤ 0.572	≤ 0.570
Pellet Diameter (in)	≤ 0.345	≤ 0.491	≤ 0.491	≤ 0.420	≤ 0.357	≤ 0.374
Clad Outer Diameter (in)	≥ 0.404	≥ 0.570	≥ 0.570	≥ 0.501	≥ 0.424	≥ 0.433
Clad Thickness (in)	≥ 0.026	≥ 0.035	≥ 0.035	≥ 0.035	≥ 0.030	≥ 0.026

Table A.9-2
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-69BTH DSC
(part 2 of 2)

Transnuclear ID	**10x10-91/1**	**ABB-8x8**	**ABB-10x10-1**	**ABB-10x10-2**
Initial Design or Reload Fuel Designation	ATRIUM-10	SVEA-64	SVEA-92	SVEA-100
	ATRIUM-10XM		SVEA-96	
			SVEA-96 +	
			OPTIMA	
			OPTIMA 2	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Fissile Material	UO_2	UO_2	UO_2	UO_2
No. of Fuel Rods	≤ 91	≤ 64	≤ 96	≤ 100
Initial Uranium Content (kg)	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.622	≤ 0.512	≤ 0.512
Pellet Diameter (in)	≤ 0.350	≤ 0.411	≤ 0.346	≤ 0.375
Clad Outer Diameter (in)	≥ 0.395	≥ 0.462	≥ 0.378	≥ 0.443
Clad Thickness (in)	≥ 0.023	≥ 0.027	≥ 0.024	≥ 0.024

Notes:

(2) The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type). Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.

Table A.9-3
BWR Fuel Assembly Initial Lattice Average Enrichment v/s Minimum B10 Requirements for the NUHOMS®-69BTH DSC Poison Plates

Basket Type	Maximum Lattice Average Enrichment(1) (wt% ^{235}U)	Minimum B10 Areal Density, gram/cm^2	
		Borated Aluminum/MMC	Boral®
A	3.7	0.021	0.025
B	4.1	0.031	0.037
C	4.4	0.039	0.047
D	4.6	0.046	0.055
E	4.8	0.053	0.064
F	5.0	0.061	0.073

	Maximum Lattice Average Initial Enrichment(1) (wt.% ^{235}U)			
Basket Type	Intact Assemblies	Up to 4 Damaged Assemblies(2)	5 to 8 Damaged Assemblies(2)	9 to 24 Damaged Assemblies(2)
A	3.70	3.70	3.30	2.80
B	4.10	4.10	3.60	3.00
C	4.40	4.20	3.60	3.10
D	4.60	4.40	3.70	3.20
E	4.80	4.40	3.70	3.20
F	5.00	4.80	3.90	3.40

(1) For LaCrosse fuel assemblies, the enrichment shall be reduced by 0.1 wt.% ^{235}U.

(2) Allowable locations in basket per Figure A.9-1.

Table A.9-4
BWR Fuel Qualification Table for the NUHOMS®-69BTH DSC

(Minimum required years of cooling time after reactor core discharge)

BU, GWD/ MTU	Lattice Average Initial U-235 Enrichment, wt %																																
	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
20	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
25	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
30				6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
35				6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
39				6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
40										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
44										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
45										7.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0

Note: Explanatory notes and limitations regarding the use of this table follow.

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Notes, Table A.9-4:

- Burnup = Assembly Average burnup.
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with a lattice average initial enrichment less than 0.7 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% ^{235}U is unacceptable for transportation.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 6-years cooling.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- The cooling times for damaged and intact assemblies are identical. However, when loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.
- Example: An assembly with an initial enrichment of 4.85 wt.% ^{235}U and a burnup of 41.5 GWd/MTU is acceptable for transport after a 6-year cooling time as defined by 4.8 wt.% ^{235}U (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).

Table A.9-5
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

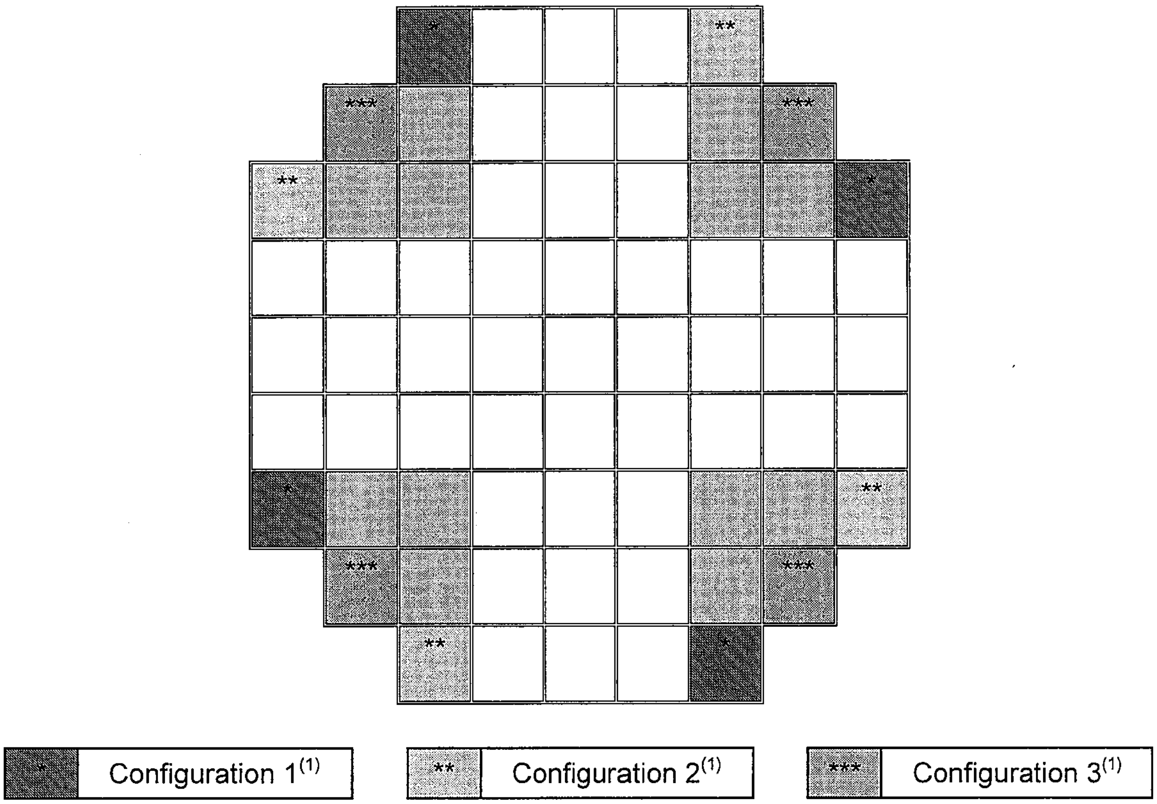
F1 Intermediate Function

X1 Assembly Burnup in GWD/MTU

X2 Initial Enrichment in wt.% ^{235}U

X3 Cooling Time in Years (minimum 6 years)

Note: Even though a minimum cooling time of 6 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.9-4.



	Either one of these three sets of corner locations shall only be used to load up to four damaged assemblies with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment of fuel assemblies (damaged or intact transported in either magenta set of cells for configuration 1, gold set of cells for configuration 2, or blue set of cells for configuration 3) is limited to the "up to 4 damaged assemblies" column of Table A.9-3.
1	Following the placement of damaged fuel assemblies in either configuration 1 or 2, the remaining gold or magenta locations shall be used to load up to 4 additional damaged assemblies, with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment for these fuel assemblies (damaged or intact transported in gold or magenta cells available) is limited to the "5 to 8 damaged assemblies" column of Table A.9-3.
	Following the placement of eight damaged fuel assemblies in the set of corner locations marked with a "*" (shaded in magenta) and a "**" (shaded in gold), the locations shaded in green or blue in Figure shall be used to load up to sixteen additional damaged assemblies, with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment for all 24 fuel assemblies (damaged or intact transported in these 24 locations) is limited to the "9 to 24 Damaged Assemblies" column of Table A.9-3.

Figure A.9-1
Location of Damaged Fuel Assemblies Inside 69BTH DSC

		Z6	Z6	Z6	Z6	Z6		
	Z6	Z5	Z5	Z4	Z5	Z5	Z6	
Z6	Z5	Z4	Z4	Z3	Z4	Z4	Z5	Z6
Z6	Z5	Z4	Z3	Z2	Z3	Z4	Z5	Z6
Z6	Z4	Z3	Z3	Z1	Z3	Z3	Z4	Z6
Z6	Z5	Z4	Z3	Z2	Z3	Z4	Z5	Z6
Z6	Z5	Z4	Z4	Z3	Z4	Z4	Z5	Z6
	Z6	Z5	Z5	Z4	Z5	Z5	Z6	
		Z6	Z6	Z6	Z6	Z6		

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Max. Decay Heat (kW/FA) (3)(4)	0.10	0.27	0.30	0.40	0.55	0.45
No. of Fuel Assemblies (1)	1	2	10	16	16	24
Max. Decay Heat per Zone (kW) (3)	0.10	0.54	3.0	6.4	8.8	10.8
Max. Decay Heat per DSC (kW)	26.0 (2) (3)					

Notes:

(1) Total number of fuel assemblies is 69 for HLZC # 1

(2) Adjust payload to maintain the total DSC heat load within the specified limit

(3) Reduce the maximum decay heat to 70% of the listed values for LaCrosse Fuel assembly. The total decay heat for LaCrosse fuel assembly is 18.2 kW per DSC for HLZC No. 1.

(4) Decay heat per fuel assembly shall be determined per Table A.9-5.

Figure A.9-2
Heat Load Zoning Configuration No. 1 for 69BTH Basket

		Z5	Z5	Z5	Z5	Z5		
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
Z5	Z4	Z4	Z3	Z3	Z3	Z4	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z1	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z4	Z3	Z3	Z3	Z4	Z4	Z5
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
		Z5	Z5	Z5	Z5	Z5		

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.25	0.0 (1)	0.40	0.60	0.50
No. of Fuel Assemblies (2)	1	0	12	24	24
Max. Decay Heat per Zone (kW) (4)	0.25	0	4.8	14.4	12.0
Max. Decay Heat per DSC (kW)	26.0 (3) (4)				

Notes:
(1) Aluminum dummy assemblies replace the fuel assemblies in zone 2
(2) Total number of fuel assemblies is 61 for HLZC # 2
(3) Adjust payload to maintain the total DSC heat load within the specified limit
(4) Reduce the maximum decay heat to 70% of the listed values for LaCrosse Fuel assembly. The total decay heat for LaCrosse fuel assembly is 18.2 kW per DSC for HLZC No. 2.
(5) Decay heat per fuel assembly shall be determined per Table A.9-5.

Figure A.9-3
Heat Load Zoning Configuration No. 2 for 69BTH Basket

		Z5	Z5	Z5	Z5	Z5		
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
Z5	Z4	Z4	Z3	Z3	Z3	Z4	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z1	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z4	Z3	Z3	Z3	Z4	Z4	Z5
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
		Z5	Z5	Z5	Z5	Z5		

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.25	0.0 (1)	0.40	0.60	0.50
No. of Fuel Assemblies (2)	1	0	12	24	24
Max. Decay Heat per Zone (kW) (4)	0.25	0	4.8	14.4	12.0
Max. Decay Heat per DSC (kW)	29.2 (3) (4)				

Notes:
(1) Aluminum dummy assemblies replace the fuel assemblies in zone 2
(2) Total number of fuel assemblies is 61 for HLZC # 3
(3) Adjust payload to maintain the total DSC heat load within the specified limit
(4) Reduce the maximum decay heat to 70% of the listed values for LaCrosse Fuel assembly. The total decay heat for LaCrosse fuel assembly is 20.4 kW per DSC for HLZC No. 3.
(5) Decay heat per fuel assembly shall be determined per Table A.9-5.

Figure A.9-4
Heat Load Zoning Configuration No. 3 for 69BTH Basket

		Z5	Z5	Z5	Z5	Z5		
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
Z5	Z4	Z3	Z3	Z3	Z3	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z1	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z2	Z2	Z2	Z3	Z4	Z5
Z5	Z4	Z3	Z3	Z3	Z3	Z3	Z4	Z5
	Z5	Z4	Z4	Z4	Z4	Z4	Z5	
		Z5	Z5	Z5	Z5	Z5		

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.0 (1)	0.45	0.0 (2)	0.70	0.60
No. of Fuel Assemblies (3)	0	8	0	20	24
Max. Decay Heat per Zone (kW) (4)	0	3.6	0	14.0	14.4
Max. Decay Heat per DSC (kW)	32.0 (4)				

Notes:
(1) The fuel compartment in zone 1 remains empty
(2) Aluminum dummy assemblies replace the fuel assemblies in zone 3
(3) Total number of fuel assemblies is 52 for HLZC # 4
(4) Reduce the maximum decay heat to 70% of the listed values for LaCrosse Fuel assembly. The total decay heat for LaCrosse fuel assembly is 22.4 kW per DSC for HLZC No. 4.
(5) Decay heat per fuel assembly shall be determined per Table A.9-5.
(6) Borated Aluminum is the only poison material allowed for HLZC #4.

Figure A.9-5
Heat Load Zoning Configuration No. 4 for 69BTH Basket

CoC 9302 Revision 5, Appendix A.10

MP197HB Packaging Contents Loaded with Radioactive Waste Canister (RWC)

(3) Type and Form of Material

(a) The NUHOMS®-MP197HB packaging is designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment. Typical composition of the payload consists of the following components either individually or in combinations:

1. BWR Control Rod Blades
2. BWR Local Power Range Monitors (LPRMs)
3. BWR Fuel Channels
4. BWR Poison Curtains
5. PWR Burnable Poison Rod Assemblies (BPRAs)
6. PWR and BWR Reactor Vessel and Internals

(b) The decay heat load of the radioactive material is less than 5 kW.

Components with high specific activity are generally placed near the center of the cask/container. For each shipment, the cask/container is normally filled to capacity, which prevents shifting of the contents during transport. If the cask/container is not full, appropriate component spacers or shoring is used to prevent significant movement of the contents.

(4) Maximum Quantity of Material per Package

(a) The quantity of radioactive material is limited to a maximum of 8,182 A_2. The radioactive material is primarily in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. When a wet load procedure (i.e., in-pool) is followed for cask loading, the cask cavity and secondary container are drained and dried to ensure that there are no free liquids in the package during transport.

(b) The NUHOMS®-MP197HB packaging is designed to transport a payload of up to 56.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in this secondary container.

(c) The maximum quantity of non-fuel bearing radioactive material loaded into a package shall not exceed 90,000 Ci of ^{60}Co. If there are other radionuclides (e.g., contaminants) in addition to ^{60}Co, the total energy release from the total waste shall not exceed 225,000 MeV/sec.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9305	7	71-9305	USA/9305/B(U)F-96	1 OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
AREVA Federal Services LLC
505 South 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
AREVA Federal Services LLC
application dated June 30, 2007, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: TRUPACT-III Package

(2) Description

A package used to transport transuranic waste contained in a Standard Large Box 2 (SLB2) primarily by highway trucks. The packaging body is a rectangular box with an external width of 2,500 mm (98.4 inches), external height of 2,650 mm (104.3 inches), and an external length of 4,288 mm (168.8 inches). The internal cavity dimensions are 1,840 mm (72.4 inches) wide, 2,000 mm (78.7 inches) tall, and 2,790 mm (109.8 inches) long.

The TRUPACT-III packaging is comprised of the containment structural assembly (CSA) made from 8-mm inner and outer stainless steel plates with 4-mm thick V-shaped stiffeners in between. A debris shield receptacle is located all around the open end of the CSA inner cavity. The receptacle is a 26-mm x 38-mm cross section bar with a 15-mm wide by 20-mm deep groove cut along its length. The 109 - 120-mm polyurethane foam, 10-mm thick puncture resistant stainless steel plate, 60-mm balsa wood layer, and the 6-mm stainless steel skin form the integral energy-absorbing overpack structure. A 409-mm deep octagonal recess in the bottom end with 6-mm thick stainless steel plate, a 60-mm thick balsa wood layer, a 15-mm or 16-mm thick puncture-resistant stainless steel plate, and a 120-mm thick foam layer protect the bottom end of the packaging during drops or punctures.

A rectangular closure lid made from 4-mm thick V-shaped stiffeners sandwiched between an inner and an outer 12-mm thick stainless steel plate is attached to the packaging body

5.(a) Packaging (continued)

by 44 socket head cap screws and contains two elastomer O-ring face seals. A sampling/vent port with elastomer O-ring seals is recessed into the closure lid. The inner stainless steel plates of the closure lid and the body along with the inner elastomer O-ring seal, the sampling/vent port insert, and the sampling/vent port inner elastomer O-ring seal form the single containment boundary.

An overpack cover is designed to protect the closure lid. The outer face of the overpack cover contains an octagonal recess 393 mm deep. The cover structure consists of a 6-mm thick stainless steel cover sheet plate encasing a 60-mm thick layer of balsa wood, a 15-mm or 16-mm thick puncture resistant stainless steel plate, a 120-mm thick layer of polyurethane foam, and a 6-mm thick inner stainless steel cover plate. The edges of the overpack cover consist of an inner 6-mm stainless steel plate, a 42-mm thick layer of calcium silicate insulation, a 15-mm or 16-mm thick puncture-resistant stainless steel plate, a 380-mm thickness of 0.48 kg/dm^3 polyurethane foam, a 6-mm thick puncture-resistant stainless steel plate, a 140-mm thick layer of 0.16 kg/dm^3 polyurethane foam, and an 8-mm thick external stainless steel plate.

The approximate dimensions and weights of the package are as follows:

Overall package outside dimensions	
Width	2,500 mm (98.4 inches)
Length	4,288 mm (168.8 inches)
Height	2,650 mm (104.3 inches)
Maximum content weight	5,210 kg (11,486 lbs)
Maximum package weight (Including contents)	25,000 kg (55,116 lbs)

(3) Drawings

The packaging is constructed in accordance with AREVA Federal Services LLC, Drawing No. 51199-SAR, Rev. 14, sheets 1 through 21.

(b) Contents

(1) Type and form of material

Dewatered, solid or solidified transuranic contaminated materials and wastes, any particle size, large objects, and bulky objects are directly loaded into an SLB2 to be placed in a TRUPACT-III packaging, in accordance with TRUPACT-III TRAMPAC, Revision 2.

(2) Maximum quantity of material per package

The TRUPACT-III packaging is designed to transport contact-handled transuranic (CH-TRU) waste and other authorized payloads that do not exceed 10^5 A_2 quantities. No more than

5.(b) Contents (continued)

325 grams of Pu-239 fissile gram equivalent (FGE) is generally allowed per TRUPACT-III package. Per the TRUPACT-III TRAMPAC, Revision 2, the FGE limit per TRUPACT-III package may be increased if the payload is documented to contain Pu-240. A TRUPACT-III payload shall not contain greater than 1 percent by weight beryllium and/or beryllium oxide nor machine compacted waste. Only one SLB2 may be loaded in a TRUPACT-III package at a time.

(3) Maximum decay heat per package not to exceed 80 watts.

5.(c) Criticality Safety Index (CSI): 0

6. The package is for transport of the CH–TRU materials and other authorized payloads that are limited in form to solid or solidified material. Materials must be restricted to prohibit explosives, corrosives, nonradioactive pyrophorics, and pressurized containers. Within a payload container, radioactive pyrophorics must not exceed 1 percent by weight, and residual liquid volumes greater than 1 percent are prohibited.

7. Limits for physical, nuclear, chemical, and gas generation properties shall be as defined in the TRUPACT-III TRAMPAC, Revision 2.

8. Hydrogen must be limited to a molar quantity that would be no more than 5% by the volume of the innermost layer of confinement during transport.

9. Each payload shipping container must be assigned to a shipping category in accordance with TRUPACT-III TRAMPAC, Revision 2, Section 5.0.

10. The gas generated in the payload and released into the cavity shall be controlled to maintain the pressure within the containment vessel below the acceptable Maximum Normal Operating Pressure of 25 psig.

11. Venting and aspiration are required to the TRUPACT-III containers stored in an unvented condition prior to transport, to ensure equilibration of gases that may have accumulated in the closed container in accordance with TRUPACT-III TRAMPAC, Revision 2, Section 5.3.

12. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

(b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application, as supplemented.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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13. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

14. Transport by air of fissile material is not authorized.

15. Expiration date: June 30, 2015.

REFERENCES

AREVA Federal Services LLC application dated June 30, 2007, as amended January 26, 2010, May 28, 2010, April 28, 2011, October 17, 2011, December 23, 2011, January 6 and 24, 2012, February 9 and 17, 2012, March 13, and 20, 2012, April 23 and 30, 2012, June 6, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 18, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Global Nuclear Fuel - Americas, LLC
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated May 5, 2009, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: RAJ-II

(2) Description

The RAJ-II package is a rectangular box that is 742 mm (29.21 in) high by 720 mm (28.35 in) wide by 5,068 mm (199.53 in) long to transport a maximum of two Boiling Water Reactor (BWR) fuel assemblies or individual rods that meet the ASTM C996-96 standard of enriched commercial grade uranium, enriched reprocessed uranium, uranium oxide generic pressurized water reactor (PWR) or uranium carbide loose fuel rods in a 5 inch diameter stainless steel pipe.

It is comprised of one inner container and one outer container both made of stainless steel. The inner container is comprised of a double-wall stainless steel sheet structure with alumina silicate thermal insulator filling the gap between the two walls to reduce the flow of the heat into the contents in the event of a fire. Foam polyethylene cushioning material is placed on the inside of the inner container for protection of the fuel assembly. The outer container is comprised of a stainless steel angular framework covered with stainless steel plates. Inner container clamps are installed inside the outer container with a vibro-isolating device between to alleviate vibration occurring during transportation. Wood and honeycomb resin impregnated kraft paper are placed as shock absorbers to reduce shock in the event of a drop of the package. The fuel rod clad and ceramic nature of the fuel pellets provide primary containment of the radioactive material.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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5.(a)(2) continued

The approximate dimensions and weights of the package are as follows:

Maximum gross shipping weight	1,614 kg (3,558 lbs)
Maximum weight of inner container	308 kg (679 lbs)
Maximum weight of outer container	622 kg (1,371 lbs)
Maximum weight of packaging	930 kg (2,050 lbs)
Dimensions of inner container	
Length	4,686 mm (184.49 in)
Width	459 mm (18.07 in)
Height	286 mm (11.26 in)
Dimensions of outer container	
Length	5,068 mm (199.53 in)
Width	720 mm (28.35 in)
Height	742 mm (29.21 in)

(3) Drawings

This packaging is constructed in accordance with the Global Nuclear Fuel (GNF) Drawing Nos.:

Outer Container Drawings
105E3737, Rev. 6
105E3738, Sheets 1 and 2, Rev. 8
105E3738, Sheet 3, Rev. 7
105E3739, Rev. 4
105E3740, Rev. 4
105E3741, Rev. 1
105E3742, Rev. 3
105E3743, Rev. 5
105E3744, Rev. 6

Inner Container Drawings
105E3745, Rev. 8
105E3746, Rev. 1
105E3747, Rev. 4
105E3748, Rev. 2
105E3749, Rev. 6

Contents Containers
105E3773, Rev. 1
0028B98, Rev. 1

5. continued

(b) Contents

(1) Type and form of material

Enriched commercial grade uranium or enriched reprocessed uranium, as defined in ASTM C996-96, uranium oxide or uranium carbide fuel rods enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1 and Table 2 below.

Table 1: Maximum weight of uranium dioxide pellets per fuel assembly

Type 8x8 fuel assembly	**Type 9x9 fuel assembly**	**Type 10x10 fuel assembly**
235 kg	240 kg	275 kg

Table 2: Maximum Authorized Concentrations

Isotope	**Maximum content**
U-232	2.00×10^{-9} g/gU
U-234	2.00×10^{-3} g/gU
U-235	5.00×10^{-2} g/gU
U-236	2.50×10^{-2} g/gU
Np-237	1.66×10^{-6} g/gU
Pu-238	6.20×10^{-11} g/gU
Pu-239	3.04×10^{-9} g/gU
Pu-240	3.04×10^{-9} g/gU
Gamma Emitters	5.18×10^{5} MeV - Bq/kgU

5.(b)(1) continued

(i) 8 x 8 fuel assemblies comprised of 60 to 64 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(ii) 9 x 9 fuel assemblies comprised of 72 to 81 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(iii) 10 x 10 fuel assemblies comprised of 91 to 100 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(iv) Oxide fuel rods configured loose, in a 5 inch diameter schedule 40 stainless steel pipe/protective case or strapped together. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 4 below.

(v) Uranium carbide or generic PWR uranium oxide fuel rods configured loose, in a 5 inch diameter schedule 40 stainless steel pipe. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 4 below.

5.(b)(1) continued

Table 3: Fuel Assembly Parameters

Parameter	Units	Type	Type	Type	Type
Fuel Assembly Type	Rods	8x8	9x9	FANP 10x10	GNF 10x10
UO_2 Density		≤ 98% Theoretical	≤ 98% Theoretical	≤ 98% Theoretical	≤ 98% Theoretical
Number of water rods (See Condition 8)	#	0, 2x2	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3
Number of fuel rods	#	60 - 64	72 - 81	91 - 100	91 - 100
Fuel Rod OD	cm	≥ 1.176	≥ 1.093	≥ 1.000	≥ 1.010
Fuel Pellet OD	cm	≤ 1.05	≤ 0.96	≤ 0.895	≤ 0.895
Cladding Type		Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Cladding ID	cm	≤ 1.10	≤ 1.02	≤ 0.933	≤ 0.934
Cladding Thickness	cm	≥ 0.038	≥ 0.036	≥ 0.033	≥ 0.038
Active fuel length	cm	≤ 381	≤ 381	≤ 385	≤ 385
Nominal Fuel Rod Pitch	cm	1.63	≤ 1.45	≤ 1.30	1.30
U-235 Pellet Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Maximum Lattice Average Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Channel Thickness[a]	cm	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048
Partial Length Fuel Rods (1/3 through 2/3 normal length)	Max #	None	12	14	14
Gadolinia Requirements Lattice Average Enrichment[b]	# @ wt% Gd_2O_3				
≤ 5.0 wt % U-235		7 @ 2 wt %	10 @ 2 wt %	12 @ 2 wt %	12 @ 2 wt %
≤ 4.7 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	12 @ 2 wt %	12 @ 2 wt %
≤ 4.6 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	10 @ 2 wt %	10 @ 2 wt %
≤ 4.3 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	9 @ 2 wt %	9 @ 2 wt %
≤ 4.2 wt % U-235		6 @ 2 wt %	6 @ 2 wt %	8 @ 2 wt %	8 @ 2 wt %
≤ 4.1 wt % U-235		4 @ 2 wt %	6 @ 2 wt %	8 @ 2 wt %	8 @ 2 wt %
≤ 3.9 wt % U-235		4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.8 wt % U-235		4 @ 2 wt %	4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.7 wt % U-235		2 @ 2 wt %	4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.6 wt % U-235		2 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %
≤ 3.5 wt % U-235		2 @ 2 wt %	2 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %
≤ 3.3 wt % U-235		2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %
≤ 3.1 wt % U-235		None	2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %
≤ 3.0 wt % U-235		None	None	2 @ 2 wt %	2 @ 2 wt %
≤ 2.9 wt % U-235		None	None	None	None
Polyethylene Equivalent Mass (Maximum per Assembly)[c]	kg	11	11	10.2	10.2

a. Transport with or without channels is acceptable

b. Required gadolinia rods must be distributed symmetrically about the major diagonal

c. Polyethylene equivalent mass calculation (refer to 6.3.2.2 of the application)

5.(b)(1) continued

Table 4: Fuel Rod Parameters

Parameter	**Units**	**Type**					
Fuel Assembly Type		8x8 (1) (UO_2)	9x9 (1) (UO_2)	10x10 (1) (UO_2)	CANDU-14 (UC)	CANDU-25 (UC)	Generic PWR (UO_2)
UO_2 or UC Fuel Density		<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical
Fuel rod OD	cm	≥1.10	≥1.02	≥1.00	≥1.340	≥0.996	≥1.118
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.90	≤1.254	≤0.950	≤0.98
Cladding Type		Zirc. Alloy	Zirc. Alloy	Zirc. Alloy	Zirc. Alloy or SS	Zirc. Alloy or SS	Zirc. Alloy or SS
Cladding ID	cm	≤1.10	≤1.02	≤1.00	≤1.267	≤0.951	≤1.004
Cladding Thickness	cm	≥0.038	≥0.036	≥0.038	≥0.033	≥0.033	≥0.033
Active fuel Length	cm	≤381	≤381	≤385	≤47.752	≤40.013	≤450
Maximum U-235 Pellet Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Maximum Average fuel rod Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Loose Rod Configuration							
Freely Loose		≤25	≤25	≤25	N/A	N/A	N/A
Packed in 5" SS Pipe or Protective Case (3)		≤22	≤26	≤30	≤74 (2)	≤130 (2)	≤105 (2)
Strapped Together		≤25	≤25	≤25	N/A	N/A	N/A

(1) Previous analysis (Ref. 1) based on most conservative loose rod configuration (i.e., no credit taken for 5" SS pipe)
(2) Including partial rods (in reality, apply dense packing of congruent rods in the pipe) and only in 5" SS pipes
(3) Protective case consists of SS box with lid

5.(b)(2) Maximum quantity of material per package

Total weight of payload contents (fuel assemblies, or fuel rods and rod shipping containers) not to exceed 684 kg (1508 pounds).

(i) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), and 5(b)(1)(iii): two fuel assemblies.

(ii) For the contents described in 5(b)(1)(iv) and 5(b)(1)(v): allowable number of fuel rods per compartment (2 compartments per package).

(c) Criticality Safety Index, except for contents described in 5(b)(1)(v) and limited in 5(b)(2)(ii) 1.0

Criticality Safety Index for contents described in 5(b)(1)(v) and limited in 5(b)(2)(ii) 2.1

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations of Chapter 7 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

(c) Prior to each shipment, the stainless steel components of the packaging must be visually inspected. Packages in which stainless steel components show pitting corrosion, cracking, or pinholes are not authorized for transport.

(d) If wrapping is used on the unirradiated fuel assemblies, the ends must be assured to be open during the shipment in the package.

7. Cluster separators are optional and may be comprised of polyethylene or other plastics. Polyethylene or plastic mass limits shall be determined in accordance with Section 6.3.2.2 (Material Specifications) of the application.

8. Water rods are limited as shown in Table 3 above.

   For 8 x 8 fuel assembly designs, there can be either 0 or 1 water rod, and the water rod location occupies a space equivalent to 2 x 2 fuel rods. This is designated as 0, 2 x 2 in the table.

   For 9 x 9 and 10 x 10 fuel assembly designs, there can be either 0, 1, or 2 water rods in the assembly, and the water rod location occupies a space equivalent to (a) two 2 x 2 fuel rod equivalent spaces on a diagonal at the center of the assembly, or (b) one 3 x 3 fuel rod equivalent space (9 fuel rods space) in the center of the assembly. These configurations are designated as 0, 2 - 2x2 off-center diagonal, 3x3 in the table.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Transport by air of fissile material is not authorized.

11. Revision No. 7 of this certificate, amended to reference Drawing Nos.: 105E3738, Sheets 1 and 2, Rev. 8; 105E3743, Rev. 5; and 105E3744, Rev. 6; as listed in Condition 5(a)(3), may be used until August 31, 2010.

12. Expiration date: November 30, 2014.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated May 5, 2009.

Supplement dated: August 24, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: August 26, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Best Theratronics
413 March Road
Ottawa, Ontario
Canada K2K 0E4

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated May 27, 2003, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. F-431 Transport Package

(2) Description

The Model No. F-431 Transport Package is designed to transport Cesium-137 in either special form or RAMCO-50 non-special form sealed sources. The F-431 Transport Package consist of: (1) the overpack which provides impact and thermal protection; (2) either the MDS Nordion Gammacell-1000 irradiator (GC-1000), or the MDS Nordion Gammacell-3000 irradiator (GC-3000) which provides shielding protection; and (3) the radioactive contents in either special form or RAMCO-50 non-special form sealed sources which provide containment.

The F-431 Transport Package is a stainless steel cylindrical package with a 1,067-millimeter (mm) (42-inch (in.)) outside diameter and a height of 1,283 mm (50.5 in.) that is placed on a removable mild steel skid. The maximum weight of the package is 2,270 kilograms (kg) (5000 pounds (lb)).

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

5.(a)(2) continued

The GC-1000 and the GC-3000 are lead-shielding casks each with a source cavity. The package contents may consist of up to eight cesium-137 special form sealed sources or RAMCO-50 non-special form sealed sources (provided Condition 5.(b)(1)(ii) is met) inside a source holder, within the source cavity. The maximum total activity of cesium-137 is 113 tera-Becquerels (TBq)(3,050 Curies (Ci)). The following are the features of the GC-1000 and GC-3000:

Irradiator Model	Rated Capacity	Diameter*	Height*	Lead Thickness*	Steel Shell Thickness*	Weight*
GC-1000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	150 mm (6 in.)	9.5 mm (0.375 in.)	1,054 kg (2,324 lb)
GC-3000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	110 mm (4.35 in.)	9.5 mm (0.375 in.)	1,091 kg (2,404 lb)

* Nominal Values

The approximate dimensions and weights of the package are as follows:

Package outside diameter	1,067mm (42 inches)
Package height	1,283 mm (50.5 inches)
Cavity diameter	559 mm (22 inches)
Cavity height	813 mm (32 inches)
Removable skid	1,118 mm (44 inches) x 1,003 mm (39.5 inches) x 203 mm (8 inches)
Overpack weight	1044 kg (2300 lbs)
Contents weight (max.)	1225 kg (2700 lbs)
Maximum package weight	2,270 kg (5000 lbs)

(3) Drawings

The packaging is constructed in accordance with the Best Theratronics drawing F643101-001,Sheet 1, Revision J and Sheet 2, Revision E.

(b) Contents

(1) Type and form of material

(i) Cesium-137 as a sealed source which meets the requirements of special form radioactive material. The sealed sources consist of the following special form sources: C-378, C-1000, C-1001, C-3000, C-3001, or ISO-1000.

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5.(b) continued

(ii) Cesium-137 as the RAMCO-50 non-special form sealed source, provided the following conditions are met:

- Source must conform to the specifications given in Figure 4.8 of the Safety Analysis Report and sealed source registry Certificate No. NR-0880-S-804-S.
- Source must have been shown to not be leaking within six months prior to shipment.
- Source must not have been damaged during its service in the GC-1000.

(2) Maximum quantity of material per package

113 TBq (3,050 Curies).

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Transport by air of fissile material is not authorized.

9. Expiration date: June 30, 2014.

NRC FORM 618
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10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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REFERENCES

MDS Nordion application dated May 27, 2003.

Supplements dated: April 16, July 16, July 21, and July 23, 2004; February 27 (Best Theratronics), March 31 (MDS Nordion), 2009, May 29, 2009 (Best Theratronics), October 21, 2011, February 15, and March 9, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 4, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, Maryland 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc., application dated
August 7, 2006, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: TN-40

(2) Description: For descriptive purposes, all dimensions are approximated nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

The TN-40 is designed to transport up to 40 Pressurized Water Reactor (PWR) spent nuclear fuel assemblies discharged from the Prairie Island Nuclear Generating Plant (PINGP). These assemblies have been stored prior to shipment in the TN-40 package used as a dry storage cask at PINGP under SNM-2506. These 29 loaded packages at the PINGP are authorized for single use. The TN-40 packaging consists of a basket assembly, a containment vessel, a package body which also functions as the gamma shield and neutron shield, and impact limiters. A transport frame, which is not part of the packaging, is used for tie-down purposes.

The containment vessel components consist of the inner shell and bottom inner plate, shell flange, lid outer plate, lid bolts, penetration cover plates and bolts (vent and drain), and the inner metallic seals of the lid seal and the vent and drain seals. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. The overall containment vessel length is approximately 170.5 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a nominal diameter of 72.0 in. and a length of 163 in.

Double metallic seals are used for the lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium above atmospheric pressure. The cask cavity is accessed via draining and venting ports. Double metallic seals are utilized to seal these two lid penetrations. The over-pressure (OP) port provides access to the volumes between the double seals in the lid and cover plates for leak testing purposes. The OP port cover is not part of the containment boundary.

5.(a)(2) Description (Continued)

The carbon steel packaging body, which also functions as the gamma shielding, is around the inner shell and the bottom inner plate of the containment vessel. The 8.0 in. and 8.75 in. gamma shield completely surround the containment vessel shell and bottom plate, respectively. A 6.0 in. thick shield plate is also welded to the inside of the 4.5 in. thick lid outer plate.

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The total radial thickness of the resin and aluminum is 4.50 in. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell. The aluminum container walls also provide a path for heat transfer from the gamma shield shell to the outer shell. A pressure relief valve is mounted on top of the resin enclosure to limit the possible internal pressure increase under hypothetical accident conditions.

The basket structure consists of an assembly of stainless steel cells joined by a fusion welding process and separated by aluminum and poison plates which form a sandwich panel. The panel consists of two aluminum plates separated by a poison plate. The aluminum plates provide the heat conduction paths from the fuel assemblies to the cask inner plate. The poison material provides the necessary criticality control. The opening of the cells is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The overall basket length (160.0 in.) is less than the cask cavity length to allow for thermal expansion and fuel assembly handling.

The impact limiters consist of balsa wood and redwood blocks encased in stainless steel plates. The impact limiters have an outside diameter of 144 in., and an inside diameter of 92 in. to accommodate the cask ends. The bottom limiter is notched to fit over the lower trunnions. The impact limiters are attached to each other using tie rods. The impact limiters are also attached to the outer shell of the cask with bolts. Each impact limiter is provided with fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has lifting lugs for handling, and support angles for holding the impact limiter in a vertical position during storage. An aluminum spacer is placed on the cask lid prior to mounting the top impact limiter to provide a smooth contact surface between the lid and the top impact limiter.

The nominal external dimensions, with impact limiters, are 261 in. long by 144 in. wide. The total weight of the package is 271,500 pounds (lbs.).

5.(a)(3) Drawings

The packagings are fabricated and assembled in accordance with the Transnuclear, Inc., Drawing Nos.:

10421-71-1, Rev. 5.
10421-71-2, Rev. 2, sheets 1 and 2.
10421-71-3, Rev. 2.
10421-71-4, Rev. 0.
10421-71-5, Rev. 0.
10421-71-6, Rev. 0.
10421-71-7, Rev. 2.

5.(a)(3) Drawings (Continued)

10421-71-8, Rev. 0.
10421-71-9, Rev. 0.
10421-71-10, Rev. 0.
10421-71-40, Rev. 1.
10421-71-41, Rev. 1.
10421-71-42, Rev. 0.
10421-71-43, Rev. 0.
10421-71-44, Rev. 0.

5.(b) Contents

(1) Type, form, and quantity of material

The characteristics of the contents of the TN-40 packaging are limited to the following.

I. Fuel shall be unconsolidated.

II. Fuel shall be limited to the following fuel types with specifications depicted in Table 1-1 of this certificate:

i. Exxon 14X14 Standard,
ii. Exxon 14x14 High Burnup,
iii. Exxon 14X14 TOPROD,
iii. Westinghouse (WE) 14X14 Standard, and
iv. Westinghouse 14X14 OFA.

III. Fuel shall only have been irradiated at the PINGP Unit 1, cycles 1 through 16 or Unit 2, cycles 1 through 15.

IV. The fuel assemblies from Unit 1, Region 4, i.e., assemblies identified as D-01 through D-40, are not authorized contents.

V. Fuel may include burnable poison rod assemblies (BPRAs) provided:

i. the BPRAs have cooled for a minimum of 25 years, and
ii. the maximum exposure of the BPRA(s) shall be 30,000 Megawatt-Days per Metric Ton of Uranium (MWd/MTU).

VI. Fuel may include thimble plug assemblies (TPAs) provided:

i. the minimum cooling time of the TPAs is 25 years,
ii. the maximum exposure of the TPA(s) shall not exceed 125,000 MWd/MTU, and

5.(b)(1) Contents – Type, form, and quantity of material (Continued)

iii. only TPAs that do not have water displacement rods extending into the active fuel may be loaded into the cask.

VII. The combined weight of a fuel assembly and any BPRA or TPA shall not exceed 1330 lbs.

VIII. The combined weight of all fuel assemblies, BPRAs, and TPAs in a single cask shall not exceed 52,000 lbs.

IX. The fuel shall not be a Damaged or Oxidized Fuel Assembly; a Damaged or Oxidized Fuel Assembly is:

- a partial fuel assembly from which fuel pins are missing unless dummy fuel pins are used to displace an amount of water equal to or greater than that displaced by the original pins;
- has known or is suspected to have gross cladding failures (other than pinhole leaks) or have structural defects sufficiently severe to adversely affect fuel handling and transfer capability; or
- has been exposed to air oxidation during storage, as indicated by maintenance or operating records

X. The number of assemblies in the container shall not exceed 40.

XI. The assembly average burnup shall be greater than or equal to the burnup calculated according to the following equations:

$B = -1,259.8X^2 + 20,242X - 23,617$; for fuel assemblies with BPRA insertions during depletion
$B = -366.95X^2 + 14,770X - 17,200$; for fuel assemblies without BPRA insertions during depletion

Where:
B = Burnup (MWd/MTU),
X = Initial enrichment (weight percent (wt%) U-235)

XII. The minimum cooling time for the fuel assemblies is 30 years. Content may include BPRAs or TPAs, which have a minimum cooling time of 25 years. Various combinations of minimum assembly average enrichment and maximum assembly average burnup prior to transport shall be in accordance with Table 1-2 in this certificate.

XIII. The maximum decay heat per fuel assembly shall not be more than 0.475 kW and 19 kW per package including the BPRAs and TPAs.

XIV. The boron-10 (B-10) in the Boral neutron poison plates in the basket must be uniformly distributed in the plates with a minimum areal density of 10 mg/cm^2.

5.(b)(1) Contents – Type, form and quantity of material (Continued)

XV. Integral Fuel Burnable Absorber is not an authorized content.

XVI. Fuel assemblies with the following irradiation history shall be authorized for transport:

i. The minimum average specific power shall be 14 MW/Assembly,
ii. The minimum hot leg average moderator density shall be 0.705 g/cm^3,
iii. The maximum hot leg average moderator temperature shall be 584 K (592°F),
iv. The average fuel temperature shall not exceed 901 K (1,162°F), and
v. The maximum average soluble boron concentration shall not exceed 675 parts per million based on an average over the limiting non-linear boron letdown curve.

XVII. The nominal length of the assembly axial blankets shall not exceed 6.2 in.

XVIII. The maximum cooling time of the spent fuel shall not exceed 200 years.

Table 1-1 Fuel Assembly Specifications[1,2]

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Max. Active Fuel Length (in.)	144	144	144	144	144
Max. Number of Fuel Rods per Assembly	179	179	179	179	179
Max. Fuel Rod Pitch (in.)	0.556	0.556	0.556	0.556	0.556
Min. Clad Thickness (in.)	0.0300	0.0310	0.0295	0.0243	0.0243
Min. Clad Outer Diameter (OD) (in.)	0.424	0.417	0.426	0.422	0.400
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Max. Pellet OD (in.)	0.3565	0.3565	0.3505	0.3659	0.3444
Min. Guide/Instrument Tube OD (in.)	16@0.541 1@0.424	16@0.541 1@0.424	16@0.541 1@0.424	16@0.539 1@0.422	16@0.528 1@0.4015
Max. Guide/Instrument Tube Inner Diameter (in.)	16@0.507 1@0.374	16@0.507 1@0.374	16@0.507 1@0.374	16@0.505 1@0.3734	16@0.490 1@0.3499
Max. Assembly and BPRA Length (in.)	161.3	161.3	161.3	161.3	161.3
Max. Assembly Width (in.)	7.763	7.763	7.763	7.763	7.763

Table 1-1 Fuel Specifications (Continued)

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Maximum MTU/Assembly	0.380	0.380	0.380	0.410	0.380
Maximum Initial Assembly Average Enrichment (wt% U-235)	3.85	3.85	3.85	3.85	3.85
Maximum Assembly Average Burnup (MWd/MTU)	45,000 (see Table 1-2[2])	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)
Minimum Cooling Time (years)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)

Notes:

1. Pre-irradiated nominal dimensions used in the design analyses and may be verified against as-built records.
2. Table 1-2 is located in this certificate.

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies[1,2,3,4]

	Minimum Assembly Average Initial Enrichment (wt.% U-235)								
Maximum Assembly Average Burnup (GWd/MTU)	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
17	30	30	30	30	30	30	30	30	30
18	30	30	30	30	30	30	30	30	30
19	30	30	30	30	30	30	30	30	30
20	30	30	30	30	30	30	30	30	30
21	30	30	30	30	30	30	30	30	30
22	30	30	30	30	30	30	30	30	30
23	30	30	30	30	30	30	30	30	30
24	30	30	30	30	30	30	30	30	30
25	30	30	30	30	30	30	30	30	30
26	30	30	30	30	30	30	30	30	30
27	30	30	30	30	30	30	30	30	30
28	30	30	30	30	30	30	30	30	30

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies (Continued)

	Minimum Assembly Average Enrichment (wt.% U-235)								
Maximum Assembly Average Burnup (GWd/MTU)	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
29			30	30	30	30	30	30	30
30			30	30	30	30	30	30	30
31			30	30	30	30	30	30	30
32			30	30	30	30	30	30	30
33			30	30	30	30	30	30	30
34			30	30	30	30	30	30	30
35			30	30	30	30	30	30	30
36			30	30	30	30	30	30	30
37			30	30	30	30	30	30	30
38			30	30	30	30	30	30	30
39			30	30	30	30	30	30	30
40			30	30	30	30	30	30	30
41			30	30	30	30	30	30	30
42			30	30	30	30	30	30	30
43					30	30	30	30	30
44						30	30	30	30
45						30	30	30	30

Notes:

1. For fuel characteristics that fall between the assembly average enrichment values in Table 1-2 of this certificate, use the next lower enrichment, and next higher burnup to determine minimum fuel cooling time.
2. Fuel assemblies that were located in the Rod Cluster Control Assembly control bank D position during Unit 1 cycle 1 and Unit 2 cycle 1 shall have a minimum cooling time of greater than 35 years.
3. The assembly average enrichment and the assembly average burnup are the enrichment and burnup averaged over the fuel assembly, including the axial blankets.
4. Fuel assemblies with a maximum average burnup and a minimum average enrichment for which no cooling time is specified in the table are not authorized contents.

5.(c) Criticality Safety Index: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the "Operating Procedures" in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the "Acceptance Tests and Maintenance Program" in Chapter 8 of the application, as supplemented.

(c) The package contents shall be limited to the contents that were in storage in the package under SNM License No. 2506 (10 CFR Part 72) as of May 2011. Any additional reuse of the packaging after post-shipment unloading of the original content is prohibited.

(d) This certificate applies to only the 29 TN-40 packages already fabricated and in use at the PINGP under SNM License No. 2506 (10 CFR Part 72).

(e) As part of the preparation for transport, the 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts shall be replaced by the SA-540 Grade B23 Class 1 bolts of the same configuration.

(f) As part of the preparation for transport, a 0.75-in. thick by 71.75-in. diameter aluminum spacer shall be installed between the cask lid and the payload.

(g) As part of the preparation for transport, the metallic seals used in the package and the vent and drain ports shall be replaced and tested to a maximum allowable leak rate of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5.

(h) Within 12 months prior to shipment, the user shall perform a leak rate test of the entire containment boundary, with an acceptance criterion of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5. This test is necessary to meet the intent of the containment acceptance tests.

(i) A temperature survey shall be performed on each loaded package and the results compared to calculated outer shell temperatures from SAR thermal model analysis in Section 3.4.7 of the application, as supplemented, with appropriate adjustments for decay heat and ambient temperature. The temperature difference between calculated and measured values shall not exceed ±25°F.

(j) To comply with 10 CFR 71.85(a), a neutron and a gamma dose rate survey must be performed over the entire surface of the overpack. Total dose rates from these surveys must meet the regulatory limits in 10 CFR 71.47.

(k) For casks that are configured for storage, the operating procedures prescribed in Section 7.4 of the application, as supplemented, must be used to convert the storage configuration to transportation configuration of the package.

7. Transport by air is not authorized.

8. Packagings must be marked with Package Identification Number USA/9313/B(U)F-96.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. The personnel barrier shall be installed at all times while transporting a loaded overpack.

14. Expiration date: June 30, 2016.

<u>REFERENCES</u>

Transnuclear, Inc., application dated: August 7, 2006.

As supplemented: June 29 and September 11, 2007; August 29, 2008; December 10, 2009; March 6, 15, and 30, April 23, May 7, June 18, July 30, August 26, September 15, and December 22, 2010; May 24, and 27, and June 9, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 10, 2011.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
QSA Global, Inc., application dated April 20, 2009, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 976 Series

(2) Description

The Model No. 976 Series transport packages include three versions called the 976A, 976C, and 976F, all designed for Type B quantities of radioactive material in special form. All versions of the Model No. 976 package include an inner shield container and a stainless steel drum with cork liner inserts to position and support the individual shield containers within the package. The drum is a 20 gallon capacity drum, with a 19 ¾" (502 mm) diameter and a height of 21 ¼" (540 mm), with 16 gauge, 0.06" (1.5 mm) thick 304 series stainless steel walls per ASTM 240 specifications. The drum lid is secured in place with a lid closure band, and four 3/8" - 16 x ¾" (19 mm) long 304 series stainless steel lid closure bolts. The lid bolts are inserted through four 3/8" (9.5 mm) diameter holes spaced equidistantly around the drum diameter. The drum lid has four 304 series stainless steel blocks measuring 1" (25.4 mm) by 1" (25.4 mm) by ¾ " (19 mm) tall; the steel blocks are welded on all four sides to the underside of the drum lid and the block welds are on the full length of the block on each side. The cork liner inserts provide shield stability during transport and act as a thermal insulator in case of fire.

The Model 855 inner shield container for the Model No. 976A package is comprised of a depleted uranium shield, secured within a steel welded housing, capable of loading up to

5.(a)(2) Description (Continued)

eight individual sources with titanium "J" tubes. Locking assemblies secure the sources at the bottom of the “J” tubes. The Model 855 is approximately 11 ¼" (286 mm) in diameter at the base by 11 ¾" (298 mm) tall, without the eyebolt height. Copper separators are installed around all exposed surfaces of the depleted uranium to prevent any steel-uranium interactions inside the shield container. The shield is further retained in place by polyurethane foam to fill the voids between the shield and the inner surfaces of the Model 855 housing. The cover is bolted to the top of the shield container during shipment. The Model 855 shield weighs a maximum of 225 lbs (102 kg) and contains a maximum of 135 lbs (61 kg) of depleted uranium.

The Model 3056 inner shield container for the Model No. 976C package is a lead shield pot measuring approximately 7.7" (196 mm) in diameter (including the handle bosses) with a height of 10.4" (264 mm). The Model 3056 inner shield container includes a depleted uranium inner core shield to provide additional shielding in close proximity to the source positions during transport. An insert contains the “J” tubes which are closed by tube caps. The Model 3056 container includes a cover to protect the source tubes and caps during shipment, stainless steel strapping, handle bosses, lifting handles and weighs a maximum of 114 lbs (52kg).

The Model 1911 inner shield container for the Model No. 976F package is a lead shield pot with a maximum thickness of 2 1/4 “ (57 mm), encased by a welded steel cylinder, 8” (203 mm) in diameter, 8 ¾” (222 mm) high and a maximum weight of 184 lbs (84 kg). The shield lid is secured to the shield container body by four stainless steel bolts and washers. The Model 1911 container is designed to be lifted by a steel eyebolt which is threaded onto a recess in the shield lid. The eyebolt is removed after loading the Model 1911 into the Model No. 976 F package cork lined drum and during transportation. There are three inner shield insert configurations to allow for different source loading applications within the Model 1911 shield container: a depleted uranium upper and lower shield insert, a tungsten upper and lower shield insert or a lead upper and lower shield insert. Additional handling source stainless steel, aluminum or tungsten capsule holders or cans may be used in the shield insert cavities.

The maximum package weights of the Model No. 976 Series Transport Packages are indicated below:

Model No.	Maximum Package Weight
976A	300 lbs (136 kg)
976C	190 lbs (86 kg)
976F	263 lbs (119 kg)

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(3) Drawings

The Model No. 976 Series transport package is constructed in accordance with the following AEA Technology or QSA Global, Inc. drawings:

R97608, Rev. H, Sheet 1	20 Gallon Drum Model 976
RCLM009, Rev. C, Sheet 1	Clamp, Band
R97637, Rev. A, Sheet 1	Cork Spacer Top Inner
R97623, Rev. B, Sheet 1	Bottom Inner Cork Insert
R97623A, Rev. B, Sheet 1	Bottom Inner Cork Insert
R97615, Rev. C, Sheet 1	Top Outer Cork Insert
R97615-1, Rev. B, Sheet 1	Top Outer Cork Insert
R97615-2, Rev. A, Sheet 1	Bottom Cork Insert
R97616, Rev. B, Sheet 1	Bottom Outer Cork Insert
R976A, Rev. F, Sheet 1	976A Type B Package with 855 Shield Container
R85590, Rev. G, Sheets 1-6	Model 855 Source Changer
R976C, Rev. H, Sheet 1	976C Type B Package with 3056 Shield Container
R3056, Rev. F, Sheets 1-4	Model 3056 Shield Container Top Level Assy
R976F, Rev. E, Sheet 1	976F Type B Package with 1911 Shield Container
R1911, Rev. F, Sheets 1-6	Model 1911 Shield

The Models No. 976A, 976C and 976F drum and cork inserts, and the Model 1911 inner shield container, are authorized for fabrication.

5. (b) Contents

(1) Type and form of material

Iridium-192, Selenium-75, and Ytterbium-169 as special form sealed sources.

(2) Maximum quantity of material per package

Model No.	Inner Shield	Nuclide	Maximum Capacity[1] Ci	Maximum content weight (grams)
976A	855	Ir-192	1,000 (37 TBq)	176
		Se-75	1,000 (37 TBq)	
		Yb-169	865 (32 TBq)	
976C	3056	Ir-192	1,250 (46.25 TBq)	220
		Se-75	1,250 (46.25 TBq)	
		Yb-169	1,000 (37 TBq)	
976F	1911	Ir-192	1,000 (37 TBq)	3.3
		Se-75	1,000 (37 TBq)	
		Yb-169	1,000 (37 TBq)	

[1]For Ir-192, the maximum capacity is based on output curies which are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter.
For Se-75 and Yb-169, the maximum capacity is based on the content curies contained in the radioactive source(s).

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated with the sources secured in the shielded positions of the package, in accordance with Chapter 7 of the application, as supplemented.

(b) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

(b) No new fabrication of the Model No. 855 and 3056 inner shield containers is authorized. Replacement components are provided as part of service and maintenance for existing units. Service operations for the Model No. 3056 shield container are limited to non welded components.

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(c) Minimum values for the tensile and yield strengths of construction materials are indicated in Table 2.2.a of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17.

8. Revision No. 3 of this certificate may be used until June 30, 2010.

9. Expiration date: July 31, 2014.

REFERENCES

QSA Global, Inc., application dated April 20, 2009.

Supplements dated June 30 and July 27, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: July 31, 2009

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10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

 a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

 b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
BWXT Y-12, L.L.C., application dated March 3, 2011.

:NDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

 (1) Model No.: ES-3100

 (2) Description

 The ES-3100 package is a cylindrical container that is approximately 110 cm (43 in) in overall height and 49 cm (19 in) in overall diameter and is composed of an outer drum assembly and an inner containment vessel. The containment vessel is placed inside the drum and surrounded by a cement based borated neutron absorber, Catalog 277-4. The purpose of the ES-3100 is to transport bulk high enriched uranium in various forms.

 The outer drum assembly consists of a reinforced stainless steel, standard mil spec 30-gal drum with an increased length. The volume formed between the drum and the attached inner liner is filled with an inorganic, castable refractory material, Kaolite 1600™, which is comprised of concrete and vermiculite. The Kaolite 1600™ acts as both a thermal insulating and an impact limiting material.

 The containment vessel is approximately 82 cm (32 in) in overall height and 13 cm (5 in) in overall diameter and is constructed of 304L stainless steel. The containment boundary consists of the 0.1 in thick containment vessel body and the lid assembly. The lid assembly consists of a sealing lid, a closure nut, and external retaining ring, which holds both the assembly and closure nut together. The double ethylene-propylene elastomer O-rings in the top flange of the containment vessel permit leak testing of the containment vessel. The maximum gross weight of the package, including contents, is 190.5 kg (420 lb).

5.(a) Packaging (continued)

(3) Drawings

The Model No. ES-3100 package is constructed and assembled in accordance with:

(i) BWXT Y-12, L.L.C., Drawing No. M2E801580A037, sheets 1 through 6, Rev. C, "Consolidated Assembly Drawing."

(ii) BWXT Y-12, L.L.C., Drawing No. M2E801580A026, Rev. C, "Heavy Can Spacer Assembly."

(iii) Equipment Specification JS-YMN3-801580-A001, Rev. E, "ES-3100 Containment Vessel."

(iv) Equipment Specification JS-YMN3-801580-A002, Rev. D, "ES-3100 Drum Assembly."

(v) Equipment Specification JS-YMN3-801580-A003, Rev. C, "Manufacturing Process Specification for Casting Kaolite 1600™ into the ES-3100 Shipping Package."

(vi) Equipment Specification JS-YMN3-801580-A005, Rev. G, "Casting Catalog No. 277-4 Neutron Absorber for the ES-3100 Shipping Package."

(vii) BWXT Y-12, L.L.C., Drawing No. M2E801580-A043, Rev. 0, "Heavy Can Spacer Assembly (SST)."

5.(b) Contents (Type and form of material, maximum quantity of material per package, and Criticality Safety Index (CSI)).

The weight of the radioactive contents, convenience containers, can lift attachments, polyethylene bags, spacers, and other material in the containment vessel shall not exceed 90 lb. The maximum mass of off-gassing packaging materials in the containment vessel (e.g., polyethylene containers or bagging, silicone rubber pads, nylon bags, etc.) shall not exceed 500 grams. The maximum content decay heat load shall not exceed 0.4 watts.

With the use of Teflon bottles as convenience containers, an additional 990 g of off-gassing material is authorized in the containment vessel. Off-gassing materials may be any type of hydrogenous material, except in the case of shipping uranium in the form of broken metal, in which case the hydrogenous material must have a hydrogen atom density less than or equal to that of water.

5.(b) Contents (continued)

The concentration limits of uranium and transuranic constituents shall be the following:

Isotope	Maximum Concentration
U-232	0.040 μg/gU[a]
U-233	0.006 g/gU[b]
U-234	0.02 g/gU
U-235	1.00 g/gU
U-236	0.40 g/gU
Transuranics (except Np)	40.0 μg/gU
Np-237	0.025 g/gU

[a] μg/gU = 10^{-6} grams per gram of total uranium
[b] g/gU = grams per gram of total uranium

(1) Uranium as solid metal or alloy, packaged in stainless-steel or tin-plated carbon steel convenience cans. Alloys of uranium include uranium-aluminum, uranium-molybdenum, and uranium-zirconium. Mass of the non-uranium portion of the alloy shall be assumed to be uranium-235.

The maximum uranium enrichment is 100 weight percent U-235.

For contents that must be shipped with spacers, the spacers must be in accordance with BWXT Y-12, L.L.C., Drawing No. M2E801580A026 or M2E801580A043, and Equipment Specification JS-YMN3-801580-A005, as specified in Condition No. 5.(a)(3). The quantity of fissile material in any convenience can shall not exceed one third of the mass loading limit per package for that content. Spacers must be positioned between every two convenience cans, or in the case of shipping one convenience can only, the spacer must be positioned on top of the single can.

(i) For metal and alloy in the form of solid geometric shapes, meeting the following restrictions, mass limits are listed in Table 1. Contents not meeting the following restrictions must be shipped as broken metal (see Condition No. 5.(b)(1)(ii)).

(A) Cylinders having a diameter no larger than 4.25 in (maximum of one cylinder per convenience can)

(B) Square bars having a cross section no larger than 2.29 in × 2.29 in (maximum of one bar per convenience can)

(C) Slugs having dimensions of 1.5 in diameter × 2 in tall (maximum of 10 slugs per convenience can)

5.(b)(1) Contents (continued)

Table 1: Loading Limits for Metal and Alloy in Solid Geometric Shapes

Solid uranium metal or alloy (specified geometric shapes)	Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)		No Spacers Maximum Mass U-235 Per Package (kg)
			Per Convenience Can	Per Package	
Cylinders (3.24 in.< diameter ≤ 4.25 in.)	≤ 100	0.0	8.333	25.000	15.000
Cylinders (diameter ≤ 3.24 in.)	≤ 100	0.0	10.000	30.000	18.000
Square Bars	≤ 100	0.0	11.733	35.200	30.000
Slugs	≤ 95	0.0	-	-	17.374
Slugs	> 80 and ≤ 95	0.0	8.108	24.324	Spacer req'd
Slugs	> 80 and ≤ 95	0.4	11.583	34.749	Spacer req'd
Slugs	≤ 80	0.0	9.773	29.318	Spacer req'd

(ii) For metal and alloy defined as broken metal, mass limits are specified in Table 2. Uranium metal and alloy pieces must have a surface-area-to-mass ratio of not greater than 1.00 cm^2/g or must not pass freely through a 3/8-inch (0.0095m) mesh sieve. The uranium metal must also have had no more than a limited contact with water and been subsequently dried. Particles and small shapes that do not pass this size restriction, as well as powders, foils, turnings, and wires, are not permitted, unless they are in a sealed container under an inert cover gas. Uranium material or alloy which has been stored in water or is visibly wet at the time of packaging is not authorized to be shipped in this package.

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5.(b)(1) Contents (continued)

Table 2: Loading Limits for Solid Metal or Alloy in the Form Defined as Broken Metal

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a]
		Per Convenience Can	Per Package	
> 95 and ≤ 100	0.0	0.925	2.774	Spacer req'd
	0.4	1.850	5.549	Spacer req'd
	0.8	3.083	9.248	Spacer req'd
	2.0	4.624	13.872	Spacer req'd
	3.2	8.323	24.969	Spacer req'd
> 90 and ≤ 95	0.0	1.172	3.516	Spacer req'd
	0.4	2.051	6.154	Spacer req'd
	0.8	3.516	10.549	Spacer req'd
	2.0	6.154	18.461	Spacer req'd
	3.2	8.791	26.373	Spacer req'd
> 80 and ≤ 90	0.0	1.111	3.333	Spacer req'd
	0.4	2.500	7.500	Spacer req'd
	0.8	4.167	12.500	Spacer req'd
	2.0	6.667	20.000	Spacer req'd
	3.2	9.445	28.334	Spacer req'd
> 70 and ≤ 80	00	1.483	4.450	2.967
	0.4	2.967	8.900	5.192
	0.8	5.439	16.317	8.900
	2.0	8.406	25.218	17.059
	3.2	9.395	28.184	27.692

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a]
		Per Convenience Can	Per Package	
> 60 and ≤ 70	0.0	1.733	5.198	3.249
	0.4	4.332	12.996	5.848
	0.8	6.931	20.793	13.646
	2.0	8.231	24.692	21.444
	3.2	8.231	24.692	24.692
≤ 60	0.0	3.718 kgU	11.154 kgU	5.576 kgU
	0.4	9.604 kgU	28.813 kgU	14.872 kgU
	0.8	11.733 kgU	35.200 kgU	28.814 kgU
	2.0	11.733 kgU	35.200 kgU	35.200 kgU
	3.2	11.733 kgU	35.200 kgU	35.200 kgU

[a] All limits are expressed in kg U-235 unless specified as kgU, which means kilograms of total uranium.

(2) Uranium as oxide, which may include UO_2, UO_3, and U_3O_8, packaged in stainless-steel, tin-plated carbon steel, or nickel-alloy convenience cans, or polyethylene bottles. The physical form of all contents is dense, loose powder which may contain clumps and pellets. Moisture content in oxide is limited to 3 weight percent water. Carbide compounds are not authorized. Two types of loading are authorized:

(i) A mass limit of 15.13 kg of oxide, with a maximum mass of 9.682 kg U-235 and 921 g carbon, with a CSI of 0.0.

(ii) A mass limit of 15.13 kg oxide, with a maximum mass of 12.32 kg U-235 and no carbon, with a CSI of 0.4.

The maximum uranium enrichment is 100 weight percent U-235. No spacers are required in the containment vessel. Shipments of oxide must be complete within 12 months of sealing the containment vessel.

(3) Solid uranyl nitrate in the form of uranyl nitrate crystals, UN_x, and [$UO_2(NO_3)_2 \cdot xH_2O$, where x is ≤ 6]. Uranyl nitrate crystals must be contained in a non-metallic convenience container (such as Teflon bottles). The mass limits are specified in Table 3. The maximum uranium enrichment is 100 weight percent U-235. No spacers are required in the containment vessel.

5.(b)(3) Contents (continued)

Table 3: Loading Limits for Solid Uranyl Nitrate Crystals

UNx (X value)	Seal Time[a] (months)	CSI	UNx loading limit (kg)	U Content (wt %)
> 0 and ≤3	2	0.4	11.90	> 52 and ≤ 61
	4	0.4	6.70	> 52 and ≤ 61
> 3 and ≤ 6	2	0.4	9.17	> 46 and ≤ 52
	4	0.0	4.75	> 46 and ≤ 52

a. Seal time is the length of time after the containment vessel is sealed that the shipment must be complete.

(4) Unirradiated TRIGA fuel elements and pellets (sections). The fuel is composed of uranium zirconium hydride (UZrH). The uranium concentration in the fuel is a nominal 8.5 weight percent, and the maximum H to Zr ratio in the fuel is 2.0. The maximum uranium enrichment is 70 weight percent U-235. The fuel sections may be from any of three types of fuel elements: standard fuel elements, instrumented standard fuel elements, and fuel follower control rods. The U-235 mass for standard and instrumented fuel elements is a nominal 136 grams per element, and the U-235 mass for fuel follower control rods is a nominal 112 grams per element. Each fuel element contains three fuel sections, either stainless steel or aluminum clad or unclad. The fuel elements are approximately 15 inches in length, with sections approximately 5 inches in length; the approximate diameter of the fuel is 1.44 inches for the standard and instrumented fuel elements, and 1.31 inches for the fuel follower control rods. The fuel elements and sections are packaged within stainless steel or tin-plated carbon steel convenience cans. Disassembled fuel elements are to be packaged with a maximum of three fuel sections, or three fuel elements, per convenience can. Fuel sections from different fuel elements may not be mixed within a single convenience can. A maximum of three convenience cans with disassembled fuel elements may be loaded into a single package. Three stainless steel or aluminum clad elements with crimped ends are to be packaged in a single convenience can with a maximum of one can per package. No spacers are required. The maximum quantity of fissile material per package is 408 grams U-235. The CSI is 0.0.

6. The vent holes on the outer steel drum shall be capped closed during transport and storage to preclude entry of rain water into the insulation cavity of the drum.

7. Content forms may not be mixed in a single ES-3100 containment vessel.

8. Any combination of convenience can sizes is allowed in a single package, as long as the total height of the can stack (including silicone rubber pads and spacers, if required) does not exceed the inside working height of the containment vessel (31 in). Any closure on the convenience can is allowed.

Empty convenience cans, spacers, silicone rubber pads, and/or stainless-steel scrubbers (i.e., stainless steel trimmings that act as dunnage) may be used to fill the void space in the containment vessel. Empty convenience cans must have a minimum 0.125 in diameter hole through the lid.

10. The contents and the convenience cans may be bagged or wrapped in polyethylene or nylon for contamination control provided the limits of Condition No. 5.(b) are met.

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CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
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11. The mass of unidentified constituents in the content to be shipped shall be counted against the fissile mass loading limit. Content shall not contain unevaluated moderating materials.

12. Transport by air is not authorized, except for shipment of unirradiated TRIGA fuel pellets, as described and limited in Condition No. 5(b)(4).

13. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7 of the application (with the exception of the uranyl nitrate shipping times in Section 7.1.3.3 of the SAR). The uranyl nitrate shipping times shall be in accordance with Condition 5.(b)(3).

    (b) Each package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application.

The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15. Expiration date: April 30, 2016.

REFERENCES

BWXT Y-12, L.L.C., application dated March 3, 2011.
U.S. Department of Energy letter dated May 14, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 29, 2012

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Alpha-Omega Services, Inc.
9156 Rose Street
P.O. Box 789
Bellflower, CA 90706

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AOS application, Revision H, dated December 30, 2012, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: AOS-025A, AOS-050A, AOS-100A, AOS-100B, and AOS-100A-S

(2) Description

A cylindrical stainless steel packaging, designed to transport Type B quantities of encapsulated solid materials or solid metals meeting Normal or Special Form criteria. The packaging is available in three model sizes – AOS-025, AOS-050, and AOS-100. Tungsten alloy is used as shielding material in model numbers with the suffix A, while carbon steel is the shielding material for model numbers with the suffix B. The Model No. AOS-100A-S has a double-ended opening configuration to be either loaded or unloaded from either end of the package. All models use a double O-ring arrangement seal in the lid joint.

The packaging includes an outer shell, a cavity, a shielding cylinder and shielding plugs, a bottom plate, a lid and lid plug. The outer shell and the cavity cylinder interlock to encase the shielding cylinder, made of either tungsten or carbon steel. A weldment attaches the upper portion of the cavity to its lower portion encasing the shielding. At the cavity's closed end, the shielding plug is encased between the cavity bottom wall and the packaging bottom plate. The shielding plug encased on the lid plug is of the same size and material (tungsten or carbon steel) as the one encased at the bottom of the packaging. The lid consists of a flat disk, with recessed areas concentric with the bolt holes on the top surface, to protect the bolts from impact loads. The packaging may use either elastomeric or metallic lid seals: the Model Nos. AOS-025A and AOS-050A elastomeric seal has two O-rings and one flat metal

5.(a)(2) Description (Continued)

retainer ring, while the Model No. AOS-100 has two O-rings and two SS300 series flat retainer rings. The metallic seal for all models is a double "C" cross section seal.

Additional packaging components include lid bolts and port plugs with their threaded pipe plugs, O-ring seals, port plug covers, and a pair of trunnions with their attachment bolts. The impact limiters consist of a thin-walled stainless steel cylindrical shell, filled with polyurethane foam, with a dish head at one end and a flat disk at the other end. At the dish-head end, another recess is provided to reduce the area available for impact during a head-on drop event. Twelve (12) squared ribs are attached to the inner wall of the cylindrical recess section of the flat disk end. Eight (8) of these ribs extend beyond the flat disk plate and are used as turnbuckle attachment points. The turnbuckles join the impact limiters and partially enclose the packaging. For the Model No. AOS-025 package, the turnbuckles are replaced with "J" hooks. The package is transported in the upright position, using a shipping cage and a pallet. The maximum weights of the package, including contents, impact limiters, all associated hardware, packing and shoring material, shall not exceed the values listed below:

Model	Width in a transport configuration (in.)	Height in a transport configuration (in.)	Packaging OD (in.)	Packaging Height (in.)	Cavity OD (in.)	Cavity Height (in.)	Maximum Package Weight (lbs.)
AOS-025A	18	21.38	7	9	1.62	5	220
AOS-050A	35.75	36.63	14	18	3.25	10	1,500
AOS-100A	60.96	71.65	28	36	6.50	20	12,500
AOS-100B	60.96	71.65	28	36	6.50	20	11,000
AOS-100A-S	60.96	71.65	28	36	6.50	20	12,500

(3) Drawings

The packaging is constructed and assembled in accordance with the following drawings:

Model	Assembly	Rev.	Impact Limiter	Rev.	Packaging	Rev.	Liner/Axial Shielding Plates	Rev.
AOS-025A	166D8142	H	105E9722	G	166D8143	G	183C8485	F
AOS-050A	105E9718	H	166D8138	G	166D8137	G	-	-
AOS-100A	105E9711	H	105E9713	G	105E9712 G001	H	183C8491	G
AOS-100B	105E9711	H	105E9713	G	105E9712 G002	H	183C8491	G
AOS-100A-S	105E9711	H	105E9713	G	105E9719	H	183C8491	G

5.(b) Contents

(1) Type and form of material

Activation product radioactive materials as Normal or Special Form. Special Form materials shall have a current certificate. Normal Form materials shall be enclosed in an inner container. The inner container is considered to be a "shoring device."

Any material with a melting point less than 900°F shall be in Special Form.

(2) Maximum quantity of material per package

(i) Maximum decay heat: 10 watts for Model No. AOS-025A; 100 watts for Model No. AOS-050A; 400 watts for Model Nos. AOS-100A, AOS-100A-S, and AOS-100B.

(ii) Maximum weight of contents: 10 lbs for Model No. AOS-025A; 60 lbs. for Model No. AOS-050A; 500 lbs. for Model Nos. AOS-100A, AOS-100A-S, and AOS-100B. Maximum weight includes any shoring devices and any additional shielding plates.

(iii) Fissile materials and irradiated fissile materials containing fission products are prohibited. Free-standing liquid is not authorized.

Table 1- Activity Limits (TBq)

Isotope	AOS-025	AOS-050	AOS-100A AOS-100A-S	AOS-100B
Co-60	4.55E-03	7.84E-02	123	0.362
Co-60 (1)	-	-	810	4.14
Cs-137	0.392	11.1	2950	19.5
Hf-181	-	81.4	3370	138
Ir-192	2.68	47.7	2410	85.8
Zr/Nb-95	-	1.06	913	2.36
Ho-166	0.44	6.55	-	-
Yb-169	147	1470	-	-
Shipping Configuration	Use of Liner required	No additional shielding required	(1) Axial shielding plates required	(1) Axial shielding plates required

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter No. 7 of the application, and

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter No. 8 of the application.

7. For transport by air, quantities are limited to the lesser of Table 1 of this certificate or 3,000 A_2.

8. For contents meeting Normal Form requirements, the package must be leak-tested to 10^{-7} std cm^3/sec prior to the first use of the package, and prior to each subsequent use.

9. When contents are loaded under water, or if water is introduced in the cavity of the package, the package must be vacuum dried prior to shipment and the cavity of the package filled with helium for such shipments.

10. The sealing surfaces of the package must be inspected. The metallic seal shall be replaced prior to each shipment. The elastomeric seal can be used only for shipment of Special Form material.

11. Appropriate shoring devices, to secure and immobilize inner containers, must be comprised of materials compatible with the radioactive contents and the cask cavity material. All shoring materials within the cavity must have a melting point greater than 900°F.

12. Torque values for the lid bolts and the connectors of the impact limiters must be as follows:

Model	Lid Bolt (ft-lb), lubricated	Impact limiter connector (ft-lb), lubricated
AOS-025A	35	10
AOS-050A	62.5	3
AOS-100A	500	70
AOS-100B	500	70
AOS-100A-S	500	70

13. The weight of the foam in each impact limiter must be measured and its average density calculated based on the known volume of foam fill.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15. Revision No. 1 of this certificate may be used until May 31, 2014.

16. Expiration date: February 28, 2017.

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CERTIFICATE OF COMPLIANCE
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REFERENCES

Radioactive Material Transport Packaging System Safety Analysis Report for Model AOS-025, AOS-050, and AOS-100 Transport Packages, Rev. H, dated December 30, 2012.

Supplements dated: April 4 and May 14, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 29, 2013

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Transnuclear, Inc.
7135 Minstrel Way
Suite 300
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
AREVA NP, Inc., application dated March 13, 2007, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: MAP-12 and MAP-13

(2) Description

The MAP package is designed to transport unirradiated uranium fuel assemblies with enrichment up to 5.0 weight percent. The package is designed to carry two fuel assemblies with core components. The package consists of two components: a base and lid. The containment system of the MAP package is the fuel rod cladding.

The base consists of a fixed stainless steel strong-back which supports the fuel assemblies. A series of inner stiffeners are secured to the underside of the strong-back to support the fuel assemblies. A neutron moderator and absorber are positioned directly beneath the strong-back between each inner stiffener. The base inner stiffeners are retained by a stainless steel cover. Exterior to the cover is a layer of rigid polyurethane foam and an outer shell of 11 gauge stainless steel. A 12-gauge stainless steel sheet is provided between the two middle stiffeners. Four stainless steel outer stiffeners support the package base. The payload rests on the "W" shaped strong-back (referred to as a W-plate) and is held in place with hinged and latched aluminum doors. The lid is very similar to that of the base – a "W" shaped stainless steel inner shell is fitted with a series of inner stiffeners, neutron moderator and absorbers, and a stainless steel cover is fitted over the stiffeners. The lid is fitted with trapezoidal impact limiters at each end. The impact limiters are constructed from rigid polyurethane foam encased by the package outer stainless shell skin. The base and lid include end plates with interlocking, interfacing angles.

5.(a) (2) Description (continued)

There are two models of the MAP package, the MAP-12 and MAP-13. The weights and dimensions of the package are as follows:

MAP-12 (for 144-in Maximum Nominal Active Fuel Length):

Maximum Gross Weight	8,630 lbs
Maximum Payload Weight	3,400 lbs
Outer Dimensions	
Length	208 in
Width	45 in
Height	31 in

MAP-13 (for 150-in Maximum Nominal Active Fuel Length):

Maximum Gross Weight	8,630 lbs
Maximum Payload Weight	3,400 lbs
Outer Dimensions	
Length	221 in
Width	45 in
Height	31 in

(3) Drawings

The MAP-12 and MAP-13 packages are fabricated and assembled in accordance with the following AREVA NP, Inc. Drawing Nos.:

9045393, Rev. 6;	9045402, Rev. 4;
9045397, Rev. 1;	9045403, Rev. 3;
9045399, Rev. 2;	9045404, Rev. 3;
9045401, Rev. 3;	9045405, Rev. 3.

(b) Contents

(1) Type and Form of Material

Enriched commercial grade uranium or enriched reprocessed uranium, as defined in ASTM C996-04, oxide fuel rods enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1 below.

5.(b) Contents (continued)

(2) Maximum Quantity of Material per Package

Table 1: Maximum Authorized Concentrations

Isotope	Maximum Content
U-232	2.00×10^{-9} g/g U
U-234	2.00×10^{-3} g/g U
U-235	5.00×10^{-2} g/g U
U-236	2.50×10^{-2} g/g U
U-238	Balance of Uranium
Np-237	1.66×10^{-6} g/g U
Pu-238	6.20×10^{-11} g/g U
Pu-239	3.04×10^{-9} g/g U
Pu-240	3.04×10^{-9} g/g U
Gamma Emitters	5.18×10^{5} MeV – Bq/kg U

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5.(b) Contents (continued)

(3) Fuel Assembly

(i) The parameters of the fuel assemblies that are permitted are specified in the table below.

Fuel Rod Array	14x14		15x15					16x16	17x17	
Assembly Type	1	2	1			2	3	1	1	2
No. of Fuel Rods	176	179	208			216	204	236	264	264
No. of Non-Fuel Cells	20	17	17			9	21	20	25	25
Nominal Fuel Rod Pitch (in)	0.580	0.556	0.568			0.550	0.563	0.506	0.502	0.496
Maximum Pellet Outer Diameter (in)	0.3812	0.3682	0.3622	0.3707	0.3742	0.3617	0.3682	0.3282	0.3252	0.3232
Minimum Fuel Rod Outer Diameter (in)	0.438	0.422	0.414	0.428	0.428	0.414	0.422	0.380	0.377	0.372
Minimum Clad Wall Thickness (in)	0.0245	0.0230	0.0220	0.0245	0.0230	0.0220	0.0230	0.0220	0.0220	0.0205
Minimum Guide Tube Wall Thickness (in)	N/A	N/A	0.0140	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Minimum Guide Tube Outer Diameter (in)	N/A	N/A	0.528	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Number of Guide Tubes	N/A	N/A	16	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Minimum Instrument Tube Wall Thickness (in)	N/A	N/A	0.0240	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Minimum Instrument Tube Outer Diameter (in)	N/A	N/A	0.491	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Number of Instrument Tubes	N/A	N/A	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Clad/Tube Material Type	Zr Alloy	Zr Alloy	Zr Alloy			Zr Alloy	Zr Alloy	Zr Alloy	Zr Alloy	Zr Alloy
Maximum Active Fuel Length (in)	160	160	160			160	160	160	160	160

5.(b) Contents (continued)

(3) Fuel Assembly (continued)

(ii) Non-fissile base-plate mounted and spider body core components are permitted.

(iii) Fuel rods assembled into the fuel assemblies are those loaded with sintered pellets of uranium oxides and/or with sintered pellets of uranium oxides mixed with various additives (e.g., Chromium, Boron, Gadolinium, and Europium).

(c) Criticality Safety Index: 2.8

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7 of the application, as supplemented.

(b) Each package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application, as supplemented.

(c) Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene or polypropylene sheath, which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assembly.

(d) The fuel rods must be leak tested after fabrication to ensure that the leakage rate of the containment boundary is less than 1E-7 ref cc/sec.

7. Transport by air of fissile material is not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Expiration date: January 31, 2018.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9319	5	71-9319	USA/9319/B(U)F-96	6	OF	6

REFERENCES

AREVA NP, Inc., application dated March 13, 2007.

Supplements dated: October 24, December 6 and 14, 2007; April 11, October 13 and 31, 2008; June 8 and 18, 2009; July 22, 2010; January 14, 2011; December 5 and 12, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: December 20, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9320	2	71-9320	USA/9320/B(U)-96	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Energy*Solutions* Spent Fuel Division
2105 South Bascom Ave., Suite 230
Campbell, CA 95008

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Energy*Solutions* Spent Fuel Division application dated June 20, 2008.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: MIDUS

(2) Description

A depleted-uranium shielded package for the transport of medical isotopes. The package has two primary components: (1) an inner cask assembly that provides containment of the radioactive material and radiation shielding, and (2) an overpack that provides impact and thermal protection.

The cask assembly consists of the cask body, closure lid, shield plug, and shield lid. The cask body is a monolithic, machined 2.5-mm thick stainless steel containment vessel, surrounded by approximately 62 mm of depleted uranium gamma shielding, and a 4-mm thick stainless steel outer shell. The containment system closure lid is a 19-mm thick stainless steel plate which is attached to the cask body by 8, M10 X 1.5 X 30 socket head cap screws. The containment system is sealed by two concentric ethylene propylene O-rings, and the lid is equipped with a leak test port. A stainless steel clad depleted uranium shield plug in the cask cavity and a shield lid that is installed over the closure lid provide shielding at the top end of the package. The overpack base and lid are constructed of thin stainless steel shells filled with rigid polyurethane foam. The overpack lid is attached to the base by eight recessed alloy steel bolts. The overpack lid is equipped with four stainless steel lugs for lifting and tie-down, and the overpack base has a bottom flange with four lugs that may also be used for tie-down.

5.(a) (2) Description (Continued)

The approximate dimensions and weight of the package are:

Overall package outer diameter	520 mm
Overall package height	551 mm
Cask assembly diameter	225 mm
Cask assembly height	347 mm
Cask cavity inner diameter	85 mm
Cask cavity inner height	134 mm
Maximum package weight	330 kg

(3) Drawings

The packaging is constructed and assembled in accordance with EnergySolutions Drawing Nos.:

TYC01-1601, Sheets 1 and 2, Rev. 0	General Arrangement of Packaging and Contents
TYC01-1602, Sheets 1 through 4, Rev. 1	General Arrangement of Cask Assembly
TYC01-1603, Sheets 1 through 3, Rev. 1	General Arrangement of Overpack Assembly
TYC01-1604, Sheets 1 through 3, Rev. 1	Containment System
TYC01-1605, Sheets 1 and 2, Rev. 0	Closure Devices
TYC01-1606, Sheets 1 through 3, Rev. 0	Gamma Shielding
TYC01-1607, Sheets 1 and 2, Rev. 0	Heat Transfer Features
TYC01-1608, Sheet 1, Rev. 0	Energy Absorbing Features
TYC01-1609, Sheets 1 and 2, Rev. 0	Lifting and Tie-Down Devices

(b) Contents

(1) Type and form of material

Molybdenum-99 with its daughter products as natrium molybdate ($NaNO_3$ 1M / NaOH 0.2M) in liquid form.

The liquid may be contained within product bottles, consisting of stainless steel flasks with stainless steel caps, with or without elastomeric seals. Various stainless steel components may be used as dunnage. The total volume of the payload hardware may not exceed 125 ml (as indicated by a maximum mass of 1.0 kg).

(2) Maximum quantity of material per package

4,400 Ci molybdenum-99. The maximum specific activity is 60 Ci/ml Mo-99. The product volume may vary from 0 to 150 ml.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7.0 of the application. Optional polymeric dunnage may be placed in the space between the cask assembly and the overpack.

   (b) The package must meet the Acceptance Tests and Maintenance Program in Section 8.0 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

8. Revision No. 1 of this certificate may be used until February 28, 2013.

9. Expiration date: May 31, 2017.

REFERENCES

Energy*Solutions* Spent Fuel Division application dated June 20, 2008.

Supplement dated February 3, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine Lipa

Christine Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: February 22, 2012

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9321	0	71-9321	USA/9321/B(U)-96	1 OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."
b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Energy*Solutions*
140 Stoneridge Drive
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Energy*Solutions* application, Revision No. 3, dated July 27, 2010.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: 3-60B

(2) Description

A cylindrical austenitic stainless steel and lead shielded packaging for shipment of Type B quantities of radioactive waste materials. The packaging is transported in the horizontal position in a shipping cradle where it is supported and tied down by its four trunnions. Trunnions are structural parts of the packaging.

Approximate dimensions and weights are as follows:

Packaging Height	125-5/8 inches
Packaging Outer Diameter	51-1/2 inches
Packaging Cavity Height / Diameter	109-3/8 inches / 35 inches
Overall Package Height, with impact limiters	165 inches
Overall Package Diameter, with impact limiters	82 inches
Package Total Gross Weight	80,000 lbs
Maximum Total Weight of Contents, Secondary Containers and Cavity Spacers	9,500 lbs

5.(a)(2) Description (Continued)

The packaging body consists of a 1 ¼-inch thick external stainless steel shell (ASTM A-240, Type 304L) and a 3/4-inch thick internal stainless steel shell. The annular space between the inner and outer shells is filled with a 6-inch thick layer of lead (ASTM B-29 commercial grade). A 12-gauge stainless steel (Type 304 L) thermal shield is welded to the exterior of the external shell to provide protection during hypothetical accident fire condition events.

A bolting ring provides sealing and bolting surfaces for the lid at the top end of the packaging. The lid, constructed of several circular stainless steel plates with a total thickness of 10 ½ inches, is sealed with a pair of elastomer O-rings and sixteen equally spaced 1 ½-inches diameter bolts. A test port hole is provided through the seal ring plate between the O-rings for periodic and pre-shipment leak-testing to verify proper seal closure.

The bottom end of the packaging consists of an external circular 3-inch thick stainless steel plate, a 5-inch thick lead shield, and a ¾-inch inner containment baseplate. The containment boundary is defined as the inner shell of the packaging body, the inner baseplate, the lid, the primary lid bolts, the inner O-rings, and the vent and drain port plugs.

The toroidal-shaped impact limiters extend approximately 15 inches beyond the outside wall of the packaging, and are constructed of fully welded stainless steel shells filled with a crushable foamed-in-place closed-cell rigid polyurethane foam with specifications described in the application.

(3) Drawings

The packaging is constructed and assembled in accordance with Energy*Solutions* Drawing No. C-002-165024-001, sheets 1- 5, Rev. 0.

5.(b) Contents

(1) Type and form of material

(a) Byproduct, source, and special nuclear material in the form of inorganic solids, inorganic solidified material, and inorganic resins.

(b) Radioactive material in the form of activated and/or contaminated non-fuel bearing reactor or accelerator components or segments of components.

(2) Maximum quantity of material per package

(a) 1110 TBq (30,000 Ci) of Co-60 or equivalent. Equivalency to other radionuclides is determined by the total energy and its spectrum.

(b) Decay heat of contents must not exceed 500 watts. For contents with residual water, the total decay heat must not exceed 4.46 times the volume fraction divided by the mass fraction of water in the contents.

(c) Contents may include fissile material in compliance with the mass limits of 10 CFR 71.15. Any contents that contain more than 0.74 TBq (20 Ci) of plutonium must be in solid form.

(d) The specific activity of radioactive powdered or dispersible solids shall not exceed 330 Ci/gram of Co-60 or equivalent.

(e) Materials that auto-ignite or change phase below 350°F, not including water, shall not be included into the contents.

(f) Total weight of contents, including shoring and secondary containers, must not exceed 9,500 pounds.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

(b) The package shall meet the acceptance tests and be maintained in accordance with Chapter 8 of the application.

7. Contents shall be packaged in secondary containers.

8. Contents shall be placed such that the center of gravity of the package is at approximately the same location as the geometric center of the package – "approximately the same location" being defined as having a ±10% difference in distance of the cavity inside dimensions from the geometric center of the package in any direction.

9. Shoring must be placed between the secondary container and the packaging cavity to prevent movement during accident conditions of transport.

10. Flammable gas (hydrogen) concentration is limited to less than 5 % in volume. Inerting is not allowed to limit the concentration of flammable gases.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

12. Expiration date: August 31, 2015.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9321	0	71-9321	USA/9321/B(U)-96	4	OF	4

REFERENCES

Energy*Solutions* application "Safety Analysis Report for the Model No. 3-60B Type B Shipping Cask," Revision No. 3, dated July 27, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: August 26, 2010.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9325	0	71-9325	USA/9325/B(U)F-96	1	OF	6

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Holtec International Report No. HI-2073681, *Safety Analysis Report on the HI-STAR 180 Transport Package,* Revision 3, dated September 25, 2009.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: HI-STAR 180

(2) Description

The HI-STAR 180 package is designed for transportation of undamaged irradiated Uranium Oxide (UO_2) and Mixed Oxide (MOX) fuel assemblies. The fuel basket provides criticality control and the packaging body provides the containment boundary, helium retention boundary, moderator exclusion barrier, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the HI-STAR 180 packaging is approximately 2700 mm without impact limiters and approximately 3250 mm with impact limiters. The maximum gross weight of the loaded HI-STAR 180 package is 140 Metric Tons.

Fuel Basket

Metamic-HT, a metal matrix composite of aluminum and boron carbide, is the principal constituent material of the fuel basket, both as structural material and neutron absorber material. Two interchangeable fuel basket models, designated F-32 and F-37, contain either 32 or 37 Pressurized Water Reactor (PWR) fuel assemblies respectively, in regionalized and uniform loading patterns. The fuel basket features a honeycomb structure and flux traps between some but not all cells.

5.(a)(2) Description (continued)

Packaging Body

The cylindrical steel shell containment system is welded to a bottom steel baseplate and a top steel forging machined to receive two independent steel closure lids, with each lid being individually designated as a containment boundary component. The outer surface of the the cask inner shell is buttressed with a monolithic shield cylinder for gamma and neutron shielding. Each closure lid features a dual metallic self-energizing seal system designed to ensure its containment and moderator exclusion functions. For this package, the inner closure lid inner seal and the inner closure lid vent/drain port cover inner seals are the containment boundary components on the inner lid; the outer closure lid inner seal and the outer closure lid access port plug seal are the containment boundary components on the outer lid.

Impact Limiters

The HI-STAR 180 package is fitted with two impact limiters fabricated of aluminum honeycomb crush material completely enclosed by an all-welded austenitic stainless steel skin. Both impact limiters are attached to the cask with 16 bolts.

(3) Drawings

The packaging shall be constructed and assembled in accordance with the following Holtec International Drawings Numbers:

(a) HI-STAR 180 Cask	Drawing No. 4845, Sheets 1-6, Rev. 7
(b) F-37 Fuel Basket	Drawing No. 4847, Sheets 1-4, Rev. 5
(c) F-32 Fuel Basket	Drawing No. 4848, Sheets 1-4, Rev. 5
(d) HI-STAR 180 Impact Limiter	Drawing No. 5062, Sheets 1-5, Rev. 5

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Only undamaged UO_2 and MOX PWR fuel assemblies, with a Zr cladding type, meeting the specifications and requirements provided in Conditions 5.b(1)(b) through 5.b(1)(j), and with the characteristics listed in Table 1 below, are authorized for transportation.

5.(b)(1)(a) continued

Table 1- PWR Fuel Assembly Characteristics

Fuel Assembly Type	14x14
Design Initial Heavy Metal Mass (kg/assembly)	≤341
Maximum Fuel Assembly Mass (kg)	500
No. of Fuel Rod Locations	179
Fuel Rod Clad O.D. (mm)	≥10.72
Fuel Rod Clad I.D. (mm)	≤9.58
Fuel Pellet Diameter (mm)	≤9.31
Fuel Rod Pitch (mm)	≤14.224
Active Fuel Length (mm)	≤3070
Maximum Fuel Assembly Length (mm)	3524
Fuel Assembly Width (mm)	≤199.3
No. of Guide and/or Instrument Tubes	17
Guide/Instrument Tube Thickness (mm)	≥0.325
Minimum Cooling Time for Assemblies with Zr Guide/Instrument Tubes (years)	3
Minimum Cooling Time for Assemblies with Stainless Steel Guide/Instrument Tubes (years)	15
Minimum Cooling Time for Assemblies with NFH insertion more than 38 cm into the active region during full power operation (years)	20

(b) Damaged fuel assemblies, i.e., assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks and which cannot be handled by normal means, as well as fuel debris, non-fuel hardware and neutron sources are not authorized contents.

(c) The maximum initial enrichment of any UO_2 assembly is 5.0 percent by weight of uranium-235.

(d) Each loaded MOX fuel assembly must meet one of the criteria sets (1-4) from Table 2 and one of the criteria sets (1-3) from Table 3. MOX fuel isotopic compositions in Table 2 are bounding for dose and decay heat and used to establish the loading patterns. MOX fuel isotopic characteristics in Table 3 are bounding for criticality purposes.

Table 2

Isotopic Characteristics of MOX Fuel

	Isotopic Composition (gram/assembly)			
Criteria / Isotope	1	2	3	4
Pu238	≤700	≤202	≤202	≤202
Pu239	≥13000	≥11000	≥7524	≥8000
Pu240	≥5800	≥3800	≥1700	≥1700
Pu241	≤2300	≤1600	≤1250	≤1600
Pu242	≤1900	≤751	≤700	≤751
U235	≥730	≥720	≥2100	≥720
U238	≤297000	≤320200	≤326000	≤326000

Table 3

Isotopic Characteristics of MOX Fuel

Criteria / Composition	1	2	3
Pu-239 (g/kg-HM)	≤39.5	≤49	≤26
Pu-238/Pu-239 (g/g)	≥0.0	≥0.015	≥0.0
Pu-240/Pu-239 (g/g)	≥0.27	≥0.38	≥0.21
Pu-241/Pu-239 (g/g)	≤0.15	≤0.20	≤0.16
Pu-242/Pu-239 (g/g)	≥0.012	≥0.06	≥0.012
Am-241(g/kg-HM)	≥0.0	≥0.0	≥0.0
U-235 (g/kg-HM)	≤7.1	≤7.1	≤7.1

5(b)(1) Continued

(e) The post-irradiation minimum cooling time, maximum burnup, maximum decay heat load, and minimum initial enrichment per assembly are listed in Tables 1.2.8 and 1.2.9 of the application. The F-32 and F-37 fuel basket cell numbering and quadrant identification are depicted in Figures 1.2.3 and 1.2.4 respectively.

(f) Regions, cells and quadrants for regionalized loading of the F-32 and F-37 baskets are identified in Tables 1.2.6.a and 1.2.6.b of the application. Table 1.2.7 of the application provides the minimum burnup requirements for the F-37 basket, based on initial enrichment.

(g) In-core operating limits for those assemblies that need to meet the burnup requirements in Table 1.2.7 of the application are as follows:

Parameter	Requirement
Assembly Average Specific Power	≤39.4 MW/MTU
Assembly Average Moderator Temperature	≤597° K
Maximum Assembly Average Fuel Temperature	1127°K
Core Average Soluble Boron Concentration	≤700 ppmb

(h) For those spent fuel assemblies that need to meet the burnup requirements specified in Table 1.2.7 of the application, a burnup verification shall be performed either in accordance with Section 6.F.3.1 or 6.F.3.2 of the application.

(i) Allowable loading patterns and fuel specifications for each basket region are referenced in Tables 1.2.8 and 1.2.9 of the application. Alternative fuel specifications for each regional loading pattern are presented in Table 1.2.10 of the application.

(j) The maximum decay heat for either the F-32 or F-37 basket model is 32 kW per basket, with 8 kW maximum decay heat per basket quadrant.

5.b.(2) Maximum Quantity of Material Per Package

32 or 37 PWR fuel assemblies, as described in 5(b)(1), in the F-32 or F-37 basket respectively.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

   (b) The package shall meet the acceptance tests and be maintained in accordance with Chapter 8 of the application.

7. The personnel barrier shall be installed and remain installed while transporting the package if necessary to meet package surface temperature and/or package dose rates requirements.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Transport by air of fissile material is not authorized.

10. The package may be used in the U.S. for shipment of UO_2 fuel meeting the above specifications.

11. Expiration Date: October 31, 2014

REFERENCES:

Holtec International application dated September 25, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Steven Baggett, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 1 October, 2009

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9328	5	71-9328	USA/9328/AF-96	1	OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA Federal Services LLC
505 S. 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated August 28, 2006, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a. Packaging

(1) Model No.: TN-55

(2) Description

The Model No. TN-55 packaging is designed to transport unirradiated uranium oxide powder. The packaging consists of a 55-gallon drum within an overpack. The unirradiated uranium oxide powder is transported inside the drum. The 55-gallon drum is a DOT specification 7A Type A drum with a reinforced closure system and ceramic gasket. The overpack for the drum has an outer surface of nominal 18-gauge galvanized carbon steel, and a reinforced fiberglass liner. The void space between the liner and outer surface is filled with a polyurethane foam. Forklift pockets, made of 12-gauge galvanized carbon steel, are located on the bottom of the overpack. The nominal outside dimensions of the overpack are 32 inch diameter and 51-5/8 inch height. The maximum gross weight of the package is 1,010 pounds.

(3) Drawing

The packaging shall be constructed and assembled in accordance with AREVA Drawing No. 60699-SAR, Revision No. 1, sheets 1 through 4.

b. Contents (continued)

(1) Type and Form of Material

Unirradiated uranium oxide powder with a maximum moisture content of 2 percent. The maximum enrichment is 1.2 weight percent U-235. The content may be packaged in plastic packaging or with other plastic material within the 55-gallon drum, provided the total plastic per drum does not exceed 1,307 grams water-hydrogen equivalent or (1,000 grams of polyethylene). Materials with a hydrogen density greater than water are not authorized, with the exception of polyethylene.

(2) Maximum Quantity of Material per Package

Up to 650 pounds (295 kilograms) of unirradiated uranium oxide powder per package.

c. Criticality Safety Index: 1.7

6. Transport by air is not authorized.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

a. Each package shall be prepared for shipment and operated in accordance with the "Package Operations," in Chapter 7 of the application.

b. Each packaging shall be tested and maintained in accordance with the "Acceptance Tests and Maintenance Program," in Chapter 8 of the application.

8. The package authorized by this certificate is hereby approved for use under general license provisions of 10 CFR 71.17.

9. Revision No. 4 of this certificate may be used until April 30, 2013.

10. Expiration date: April 30, 2017.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9328	5	71-9328	USA/9328/AF-96	3	OF	3

REFERENCES

Packaging Technology, Inc., application dated August 28, 2006.

Supplement dated: January 31, 2007, November 26, 2007, October 31, 2011, and March 5, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine A. Lipa

Christine A. Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 3, 2012

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

National Nuclear Security Administration
P.O. Box 5400
Albuquerque, NM 87185

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Los Alamos National Laboratory
Application, Revision No. 5, dated June 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a. Packaging

(1) Model No.: S300

(2) Description

The Model No. S300 package is a cylindrical container that is approximately 89 centimeters (35 inches) in overall height and 60 centimeters (23 inches) in overall diameter. The Model No. S300 is comprised of an overpack, pipe component, and a shielding insert. The Model No. S300 is designed to transport a single special form capsule (SFC). The maximum gross weight of the package is 217.7 kilograms (480 lbs).

The overpack design utilizes a standard 55-gallon drum as the outer container. A standard bolted clamping ring secures the drum lid to the drum body. Within the drum body is a rigid polyethylene liner (body and lid). Lid liner and lid are pierced and the drum lid is fitted with a filter vent. Within the liner is cane fiberboard dunnage and a sheet of plywood to provide shock absorption for the pipe component.

The pipe component consists of a stainless steel cylindrical pipe welded to a stainless steel flat cap at the bottom end and a bolted pipe flange at the other end. The pipe component is closed with a stainless steel flat lid attached to the flange with 12 stainless steel bolts. A filter vent is installed in the lid. The flange-to-lid seal is either a butyl or ethylene propylene elastomeric o-ring.

5. a. Packaging (continued)

(2) Description (continued)

The shielding insert is located within the pipe component. The shielding insert is made from solid high density polyethylene plastic. Within the shielding insert is a cylindrical opening sized to accommodate the SFC.

(3) Drawings

The packaging is constructed in accordance with AREVA Drawing No. 60999-SAR, sheets 1 through 3, Revision 1, S300 Packaging SAR Drawing.

b. Contents

(1) Type and form material

Content No. 1: Plutonium-Beryllium (α,n) neutron sources (not to exceed 1.519E+5 neutrons/second per gram of plutonium), or plutonium-based (α,n) neutron sources.

Content No. 2: Plutonium, other than neutron sources with (α,n) target material, in solid form.

Content Nos. 1 and 2 must meet the requirements of special form sources and are limited to:

(a) The Model II source capsule - IAEA Certificate of Competent Authority Special Form Radioactive Materials Certificate Number USA/0696/S–96, issued by the U.S. Department of Transportation (DOT), assembled in accordance with AEA Technology QSA, Inc., Drawing No. R20047, Rev. B, or LANL Drawing No. 90Y-219998, Rev. H.

(b) The Model III source capsule - IAEA Certificate of Competent Authority Special Form Radioactive Materials Certificate Number USA/0695/S–96, issued by the DOT, assembled in accordance with AEA Technology QSA, Inc., Drawing No. R20048, Rev. B, or LANL Drawing No. 90Y-220045, Rev. A.

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5. b. Contents (continued)

(2) Maximum quantity of material per package:

One source capsule, containing a maximum quantity of fissile plutonium (Pu-239 plus Pu-241) as shown below.

Non-Exclusive Use Shipment		Exclusive Use Shipment	
Model II	Model III	Model II	Model III
Content No. 1			
206 grams fissile plutonium	160 grams fissile plutonium	350 grams fissile plutonium	160 grams fissile plutonium
Content No. 2			
300 grams plutonium	160 grams plutonium	300 grams plutonium	160 grams plutonium

Source capsule may contain radionuclides listed below within the ranges shown.

Radionuclide	Percentage of total plutonium mass
Pu-238	0 – 0.5%
Pu-239	73 – 97%
Pu-240	3 – 21%
Pu-241	0 – 3%
Pu-242	0 – 2%
Am-241	0 – 2.5%

Total quantity of radioactive material within a package may not exceed a Type A quantity.

c. Criticality Safety Index

Content No. 1 0.3

Content No. 2 4.0

6. Transport by air is not authorized.

7. In addition to the requirements of Subpart G of 10 CFR Part 71:

   a. Each package shall be prepared for shipment and operated in accordance with the "Package Operations," in Chapter 7 of the application.

   b. Each package shall be tested and maintained in accordance with the "Acceptance Tests and Maintenance Program," in Chapter 8 of the application.

8. Prior to each shipment, the package must be inspected to ensure the packaging is conspicuously and durably marked with its model number, serial number, gross weight, and package identification number, USA/9329/AF-96.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Revision No. 3 of this certificate may be used until January 31, 2012.

11. Expiration date: January 31, 2017.

REFERENCES

Los Alamos National Laboratory Application, Revision No. 5, dated June 2010.

Supplement dated: October 20, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: January 24, 2012

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9330	6	71-9330	USA/9330/AF-96	1 OF	5

2.. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
U.S Department of Energy consolidated application dated June 23, 2011, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: ATR FFSC

(2) Description

An insulated stainless steel package for the transport of unirradiated research reactor fuel, including intact fuel elements or fuel plates. The packaging consists of (1) a body, (2) a closure lid, and (3) inner packaging internals. The approximate dimensions and weights of the package are:

Overall package outer width and height	8 inches
Overall package length	73 inches
Cavity diameter	5-3/4 inches
Cavity length	68 inches
Packaging weight (without internals)	240 pounds
Maximum package weight (including internals and contents)	290 pounds

The body is composed of two thin-walled, stainless steel shells. The outer shell is a square tube with an 8-inch cross section, a 73-inch length, and a 3/16 inch wall thickness. The inner shell is a round tube with a 6-inch diameter and a 0.120-inch wall thickness. The inner tube is wrapped with ceramic fiber thermal insulation, overlaid with a stainless steel sheet. At the bottom end, the shells are welded to a 0.88-inch thick stainless steel base plate. At the top end (closure end), the shells are welded to a 1.5-inch thick stainless steel flange.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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5.(a)(2) Description (Continued)

The closure is composed of circular stainless steel plates with ceramic fiber insulation. The closure engages the top end flange by way of four bayonets that are rotated and secured by two spring pins. The closure is equipped with a handle, which may be removed during transport. The closure does not have a gasket or seal.

The package internals consist of either a Fuel Handling Enclosure (FHE) for intact Advanced Test Reactor (ATR), Massachusetts Institute of Technology (MIT), University of Missouri Research Reactor (MURR), or Rhode Island Nuclear Science Center (RINSC) fuel elements and Small Quantity Payloads, or a Loose Fuel Plate Basket for ATR fuel plates. The RINSC, MIT, MURR, and Small Quantity Payload FHE use ball lock pins and end spacers to lock closed while the ATR FHE uses a spring plunger.

(3) Drawings

The packaging is constructed and assembled in accordance with the following Areva Federal Services LLC. or Packaging Technology, Inc., Drawing Nos.:

60501-10, Sheets 1-5, Rev. 3	ATR Fresh Fuel Shipping Container SAR Drawing
60501-20, Rev. 1	ATR Loose Fuel Plate Basket
60501-30, Rev. 1	ATR Fuel Handling Enclosure
60501-40, Rev. 0	MIT Fuel Handling Enclosure
60501-50, Rev. 0	MURR Fuel Handling Enclosure
60501-60, Rev. 0	RINSC Fuel Handling Enclosure
60501-70, Rev. 0	Small Quantity Payload Fuel Handling Enclosure

(b) Contents

(1) Type and form of material

Unirradiated Mark VII ATR fuel. The ATR fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum 94 weight percent U-235; the maximum U-234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Intact ATR fuel elements contain 19 curved fuel plates fitted within aluminum side plates, and the maximum channel thickness between fuel plates is 0.087 inch. The fuel meat thickness is a nominal 0.02 inch for all 19 plates, and the fuel meat width ranges from approximately 1.5 inches to 3.44 inches. The nominal active fuel length is approximately 48 inches. The maximum mass of U-235 per intact ATR fuel element is 1200 grams. The ATR fuel element must be contained within the ATR Fuel Handling Enclosure, as specified in 5.(a)(3).

Unirradiated MIT fuel element. The MIT fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum of 94 weight percent U-235; the maximum U- 234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Each MIT fuel element contains 15 flat fuel plates fitted within aluminum side plates and the maximum channel thickness between fuel plates is 0.090 inch. The fuel meat thickness is a

5.(b)(1) Type and Form of Material (continued)

nominal 0.03 inch for all 15 plates and the fuel meat width ranges from approximately 1.98 inches to 2.17 inches. The nominal active fuel length is 22.375 inches. The maximum mass of U-235 per intact MIT fuel element is 515 grams. The MIT fuel element must be contained within the MIT Fuel Handling Enclosure, as specified in 5.(a)(3).

Unirradiated MURR fuel element. The MURR fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum of 94 weight percent U-235; the maximum U- 234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Each MURR fuel element contains 24 curved fuel plates fitted within aluminum side plates and the maximum channel thickness between fuel plates is 0.090 inch. The fuel meat thickness is a nominal 0.02 inch for all 24 plates and the fuel meat width ranges from approximately 1.71 inches to 5.72 inches. The nominal active fuel length is 24 inches. The maximum mass of U-235 per intact MURR fuel element is 785 grams. The MURR fuel element must be contained within the MURR Fuel Handling Enclosure, as specified in 5.(a)(3).

Small Quantity Payloads (RINSC fuel elements, ATR Full-size plate In Flux trap Position (AFIP) elements, U-Mo foils, Design Demonstration Elements (DDEs) and similar test elements, MIT loose fuel element plates, or MURR loose fuel element plates) where the maximum mass of U-235 is 400 grams and maximum U-235 enrichment is 94 weight percent. Aluminum plates, shapes, and sheets, and miscellaneous steel or aluminum fasteners may be used as dunnage to fill gaps between the small quantity payloads and the small quantity FHE. 1/8" neoprene strips may be used between the small quantity FHE and small quantity payloads and/or between the optional aluminum dunnage and the small quantity payload. The 1/8" neoprene strips shall not be stacked in more than two layers between the small quantity payload and any interior face of the small quantity FHE.

Unirradiated RINSC fuel element. The RINSC fuel material is composed of uranium silicide (U_3Si_2) dispersed in aluminum powder. The uranium is enriched to a maximum of 20 weight percent U-235; the maximum U- 234 content is 0.5 weight percent; and the maximum U-236 content is 1.0 weight percent. Each RINSC fuel element contains 22 flat fuel plates fitted within aluminum alloy side plates and the maximum channel thickness between fuel plates is 0.096 inch. The fuel meat thickness is a nominal 0.02 inch for all 22 plates. The maximum mass of U-235 per intact RINSC fuel element is 283 grams. The RINSC fuel element must be contained within the RINSC Fuel Handling Enclosure, as specified in 5.(a)(3).

AFIP fuel element. The AFIP fuel element is composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy coated with a thin zirconium interlayer. The uranium is enriched to approximately 20 weight percent U-235. Each AFIP element contains 4 curved fuel plates fitted within 6061 aluminum side plates. The maximum mass of U-235 AFIP element is 365 grams. Loose plates from an AFIP fuel element are also permitted. The AFIP fuel element must be

5.(b)(1) Type and Form of Material (continued)

contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

U-Mo Foils. The U-Mo foils are composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy and may contain a zirconium coating. The uranium is enriched to a maximum of 94 weight percent U-235. The maximum mass of U-235 is 160 grams. More than one U-Mo foil type may be transported at a time. The U-Mo foils must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

DDEs and similar test elements. The DDEs and similar test elements are composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy. The uranium is enriched to a maximum of 94 weight percent U-235. The maximum mass of U-235 is 365 grams. Loose plates from a DDE or similar test element are also permitted. The DDEs or similar test elements must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

MIT and MURR loose fuel element plates. MIT and MURR loose plates may either be flat or curved and may be banded or wire-tied in a bundle. The MIT and MURR loose plate payload is limited to 400 grams of U-235. The approximate mass of U-235 of each MIT fuel plate is 34.3 grams. The approximate mass of U-235 per each MURR fuel plate is 19 to 46 grams. A mixture of MIT and MURR fuel plates may be shipped together. The fuel plates must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

ATR loose fuel plates: ATR loose plates may either be flat or curved and may be banded or wire-tied in a bundle. The ATR loose plate payload is limited to 600 grams of U-235. Additional aluminum plates may be used as dunnage to fill gaps between the fuel plates and the basket payload cavity. The fuel plates must be contained within the ATR Loose Fuel Plate Basket, as specified in 5.(a)(3).

(2) Maximum quantity of material per package

The maximum total weight of contents and internals, including dunnage and other secondary packaging, is 50 lbs. Radioactive contents are not to exceed a Type A quantity.

For intact ATR, MURR, RINSC, and MIT fuel elements: One fuel element.

For ATR loose fuel plates: A maximum of 600 grams U-235.

For Small Quantity Payloads: A maximum of 400 grams U-235.

(c) Criticality Safety Index (CSI):

For ATR, MURR, MIT fuel elements or ATR loose fuel plates:	4.0
For Small Quantity Payloads:	25

6. Fuel elements and fuel plates may be bagged or wrapped in polyethylene. The maximum weight of the polyethylene wrap shall not exceed 100 grams per package.

7. Types of small quantity payloads cannot be mixed in a single Fuel Handling Enclosure.

8. Air transport of fissile material is not authorized.

9. In addition to the requirements of 10 CFR 71 Subpart G:

   (a) The package must be loaded and prepared for shipment in accordance with the Package Operations in Section 7 of the application.

   (b) The package must be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Revision No. 5 of this certificate may be used until September 30, 2014.

12. Expiration date: May 30, 2014.

REFERENCES

U.S. Department of Energy consolidated application dated June 23, 2011, as supplemented: August 18, 2011, January 10, 2012, and July 23, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: Sept. 17, 2013

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

   a. ISSUED TO *(Name and Address)*

   Holtec International
   Holtec Center
   555 Lincoln Drive West
   Marlton, NJ 08053

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Holtec International Report No. HI-2073710, *Safety Analysis Report on the HI-STAR 60 Transport Package,* Revision 2, dated May 15, 2009.

4. CONDITIONS

   This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 60

(2) Description

The HI-STAR 60 packaging is designed for transportation of irradiated nuclear fuel assemblies. The fuel basket provides criticality control and the cask provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the HI-STAR 60 package is approximately 1924 mm without impact limiters and approximately 2864 mm with impact limiters. The maximum gross weight of the loaded HI-STAR 60 package, as presented for transportation, is 74.4 Metric Tons.

Fuel Basket

The fuel basket, designated F-12 for the transport of 12 Pressurized Water Reactor (PWR) fuel assemblies, is a fully welded, stainless steel, honeycomb structure and features flux traps between some but not all cells.

Fuel Impact Attenuators

Fuel Impact Attenuators are spacers designed to limit internal gaps between the fuel assembly end-fittings and the internal surfaces of the package. Fuel Impact Attenuators also mitigate the G loads on the fuel assemblies due to secondary internal impact.

5.(a)(2) Description (continued)

Cask

The HI-STAR 60 cask is a multi-layer steel cylinder with a welded base-plate and bolted lid (closure plate). The inner shell of the cask forms an internal cylindrical cavity for housing the basket. The outer surface of the cask inner shell is buttressed with intermediate steel shells for radiation shielding. The cask closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the cask inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 60 cask is fitted with two impact limiters fabricated of aluminum honeycomb crush material completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the cask with 8 bolts at the top and bottom, respectively.

Fastener Strain Limiters

Fastener strain limiters are collapsible devices designed to limit the axial stress imparted to the impact limiter attachment bolts.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Holtec International Drawings Numbers:

(a) HI-STAR 60 Cask — Drawing No. 5238, sheets 1-7, Rev. 4

(b) HI-STAR 60 Fuel Basket — Drawing No. 5217, sheets 1-3, Rev. 6

(c) HI-STAR 60 Impact Limiter — Drawing No. 5237, sheets 1-3, Rev. 4

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Undamaged fuel assemblies meeting the specifications and requirements provided in Conditions 5.b(1)(b) through 5.b(1)(h) below. Fuel assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks and which cannot be handled by normal means are not authorized for transportation.

5.(b)(1) Type, Form and Quantity of Material (continued)

(b) Fuel assemblies with missing fuel rods in fuel rod locations shall not be transported unless dummy fuel rods that displace an amount of water greater than or equal to that displaced by the original fuel rod(s) have been installed in the fuel assembly.

(c) Fuel assembly authorized for transportation is irradiated 15x15 PWR fuel with uranium oxide pellets and a Zr-4 per ASTM B 811-1997 cladding type. The maximum initial enrichment of any assembly to be transported is 4.1 percent by weight of uranium-235. The fuel assembly weight is not to exceed 471 kg per assembly.

(d) The post-irradiation cooling time, average burnup and minimum initial enrichment of each assembly are listed in Table 1.

Table 1 - Fuel Assembly Cooling Time, Average Burnup and Initial Enrichment

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt% U^{235})
≥ 5	≤ 45,000	≥ 3.6
≥ 5	≤ 40,000	≥ 3.4
≥ 5	≤ 37,000	≥ 3.0
≥ 5	≤ 30,000	≥ 2.67
≥ 5	≤ 27,000	≥ 2.4

(e) The maximum decay heat of an individual assembly is 0.875 kW.

(f) Fuel assemblies shall not contain non-fuel hardware.

(g) The characteristics of the fuel assemblies authorized for transportation are listed in Table 2. All parameters are design nominal values.

5(b)(1)(g) Type, Form and Quantity of Material (Continued)

Table 2 - PWR Fuel Assembly Characteristics

Fuel Assembly Type	15x15
Design Initial U (kg/assembly)	≤ 300
No. of Fuel Rod Locations	204
Fuel Rod Clad O.D. (mm)	≥ 10.0
Fuel Rod Clad Thickness (mm)	≥ 0.7
Fuel Pellet Diameter (mm)	≤ 8.43
Fuel Rod Pitch (mm)	≤ 13.3
Active Fuel Length (mm)	≤ 2900
Fuel Assembly Length (mm)	≤ 3530
Fuel Assembly Width (mm)	≤ 199.3
No. of Guide and/or Instrument Tubes	21
Guide/Instrument Tube Thickness (mm)	≥ 0.5

(h) The major fuel parameters and host reactor operating parameters are listed in Table 3 below

Table 3 – Fuel and Host Reactor Operating Parameters

Fuel Parameters	
Initial Fill Pressure	<3.44 MPa
Maximum End Of Life (EOL) Hoop Stress in the Cladding at 400°C Peak Cladding Temperature	90 MPa
Co^{59} content of Fuel Assembly Hardware	<1200 ppm
Maximum Cladding Oxide Thickness at EOL	0.05 mm
Host Reactor Operating Parameters	
Average - Maximum Rod Power during Normal Operations	<20-60 kW/m
Minimum Reactor Coolant Inlet Temperature	273°C
Maximum Reactor Coolant Outlet Temperature	329°C
Maximum Soluble Boron Content in Core	<1500 ppm
Typical Cycle Length	12 to 24 months
pH Value of Primary Coolant at 25°C	4.2 – 10.5
Hydrogen Control of Primary Coolant System	25-50 cm^3 (STP)/kg-H_2O

5.b.(2) Maximum Quantity of Material Per Package

12 PWR fuel assemblies, as described in 5(b)(1), in the F-12 basket.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

   (b) The package shall meet the acceptance tests and be maintained in accordance with Chapter 8 of the application.

7. The personnel barrier shall be installed and remain installed while transporting the package if necessary to meet package surface temperature and/or package dose rates.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Transport by air of fissile material is not authorized.

10. Expiration Date: May 31, 2014

REFERENCES:

Holtec International Report No. HI-2073710, *Safety Analysis Report on the HI-STAR 60 Transport Package*, Revision 2, dated May 15, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: May 22 2009

C FORM 618 (8-2000) 10 CFR 71		**CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES**		U.S. NUCLEAR REGULATORY COMMISSION
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Croft Associates Limited
Building F4, Culham Science Centre
Culham, Abingdon
Oxfordshire, OX14 3BD, United Kingdom

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Croft Associates Limited application dated July 30, 2009, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 3979A

(2) Description

The Model No. 3979A is a package for the transport of radioisotopes used in a wide range of therapeutic and diagnostic applications and research. The packaging consists of an outer stainless steel keg and an inner containment vessel surrounded by insulating cork packing. There are three specific inserts authorized for use in the Model No. 3979A, designated as Shielding Insert Design Nos. 3983, 3984, and 3986. The outer keg provides impact and thermal protection. Containment is provided by the containment vessel. Shielding is provided by the containment vessel and shielding insert.

The keg has a stainless steel outer shell and a stainless steel liner, between which insulating cork is fitted. The keg lid is attached to the body by 8 stainless steel studs and nuts, with a single O-ring weather seal. An inner cork liner is fitted between the keg liner and the top and sides of the containment vessel, consisting of a cork body and cork top, with no cork between the bottom of the containment vessel and the keg liner.

The containment vessel consists of a body and lid. The body has a stainless steel outer wall, base, and flange/cavity wall. The flange/cavity wall is welded to the outer wall to form a cavity into which

5.(a) (2) Description (Continued)

the lead shielding is cast. The base is then welded to the outer wall. The containment vessel lid top and lid shielding casing are stainless steel, with 22 mm of lead cast inside. The containment vessel lid is secured by eight, M-10x1.5x20, alloy steel recessed hexagon socket head cap screws. The containment system is sealed by two concentric ethylene propylene rubber O-rings, and the lid is equipped with a leak test port.

There are three Shielding Inserts designed for use in the Model No. 3979A packaging. Design No.. 3983, LS-31x73-Tu, is a tungsten insert with inner cavity size of 31 mm diameter by 73 mm height. The approximate mass of the insert is 4.9 kg. Design No. 3984, LS-12x65-Tu, is a tungsten insert with inner cavity size of 12 mm diameter by 65 mm in height. The approximate mass of the insert is 5.8 kg. The third design, Design No. 3986, LS-50x103-SS, is a stainless steel insert with inner cavity size of 50 mm diameter by 103 mm height. The approximate mass of the insert is 1.0 kg.

The radioactive material may be enclosed in any convenient product container such as a quartz vial or aluminum capsule. Irradiated items may be carried in plastic or metal can or wrapping to minimize the contamination of the insert.

The approximate dimensions and mass of the package are:

Overall package outer diameter	424 mm
Overall package height	483 mm
Containment vessel outer diameter	175 mm
Containment vessel height	204 mm
Containment vessel cavity inner diameter	65 mm
Containment vessel cavity inner height	109 mm
Maximum package mass	65 kg

(3) Drawings

The packaging is constructed and assembled in accordance with Croft Associates Limited Drawing Nos:

1C-6040, Rev. F	Cover Sheet for Safkeg LS Design No. 3979A (Licensing Drawing)
0C-6041, Rev. C	SAFKEG LS Design No 3979A (Licensing Drawing)
0C-6042, Rev. D	Keg Design No. 3979 (licensing Drawing)
0C-6043, Rev. C	Cork Set for Safkeg LS (Licensing Drawing)
1C-6044, Rev. F	Containment Vessel Design No. 3980 (Licensing Drawing)
1C-6045, Rev. E	Containment Vessel Lid (Licensing Drawing)
1C-6046, Rev. E	Containment Vessel Body (Licensing Drawing)

5.(a) (3) Drawings (Continued)

The shielding inserts are constructed and assembled in accordance with Croft Associates Limited Drawing Nos:

2C-6171, Rev. C LS-12x65-Tu Insert Design No. 3984 (Licensing Drawing)
2C-6172, Rev. C LS-31x73-Tu Insert Design No. 3983 (Licensing Drawing)
2C-6175, Rev. D LS-50X103-SS Insert Design No. 3986 (Licensing Drawing)

5.(b) Contents

(1) Type and form of material

Solid material must have melting point greater than 250° C.

(i) Solid, normal form material, as limited in Table 1, within insert Design No. 3984.

(ii) Solid, normal form material, as limited in Table 2, within insert Design No. 3983.

(iii) Solid, normal form material, as limited in Table 3, within insert Design No. 3986.

(iv) Krypton-79, and Xenon-133 gas, as limited in Table 4, within insert Design No. 3983.

(v) Solid, normal form material or solid sealed sources that meet the requirements of special form radioactive material, as limited in Table 5, within insert Design No. 3986.

(2) Maximum quantity of material per package

Decay heat not to exceed 10 watts per package. The contents may include fissile materials provided the mass limits of 10 CFR 71.15, 71.22, or 71.23 are not exceeded. Mixtures of nuclides are allowed providing the sum of the proportionate amounts of each nuclide with respect to the quantities shown in the respective table does not exceed unity.

(i) For the contents described in 5(b)(1)(i):

Total mass of contents and insert not to exceed 5.8 kg. Maximum mass of radioactive material, is 30 g.

TABLE 1

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Ac-225	1.22E-01	I-131	1.34E+00	Re-188	5.74E-01
Ac-227	8.38E-01	In-111	1.42E+02	Rh-105	2.71E+02
Ac-228	1.07E-02	Ir-192	9.60E-01	Se-75	1.54E+02
Am-241	3.90E+00	Ir-194	2.58E-01	Sm-153	1.90E+02
As-77	1.95E+02	Lu-177	3.43E+02	Sr-89	1.07E+02

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Au-198	2.33E+00	Mo-99	2.80E-01	Sr-90	1.62E+01
Ba-131	4.52E-01	Na-24	7.80E-04	Tb-161	3.19E+02
C-14	4.80E+00	Np-237	7.80E-04	Th-227	1.79E+00
Co-60	2.28E-03	P-32	1.90E-02	Th-228	2.53E-03
Cs-131	2.24E+03	P-33	8.15E+02	Tl-201	4.84E+02
Cs-134	2.24E-02	Pb-203	1.45E+01	W-187	1.96E-01
Cs-137	1.42E-01	Pb-210	8.40E+01	W-188	6.02E-01
Cu-67	2.30E+02	Pd-109	1.73E+02	Y-90	8.76E-03
Hg-203	1.86E+02	Ra-223	8.46E-01	Yb-169	1.47E+02
Ho-166	2.42E-01	Ra-224	3.33E-03	Yb-175	3.69E+02
I-125	1.06E+03	Ra-226	3.62E-03		
I-129	1.95E-04	Re-186	1.38E+02		

(ii) For the contents described in 5(b)(1)(ii):

Total mass of contents and insert not to exceed 5.3 kg. Maximum mass of radioactive material, is 200 g.

TABLE 2

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Ac-225	8.35E-02	I-131	6.71E-01	Re-188	3.55E-01
Ac-227	4.70E-01	In-111	1.42E+02	Rh-105	2.71E+02
Ac-228	6.90E-03	Ir-192	4.30E-01	Se-75	1.54E+02
Am-241	1.13E+01	Ir-194	1.66E-01	Sm-153	1.90E+02
As-77	7.84E+01	Lu-177	3.43E+02	Sr-89	6.64E+01
Au-198	1.32E+00	Mo-99	1.52E-01	Sr-90	6.89E+00
Ba-131	2.56E-01	Na-24	5.66E-04	Tb-161	2.99E+02
C-14	3.20E+01	Np-237	5.20E-03	Th-227	1.01E+00
Co-60	1.53E-03	P-32	1.35E-02	Th-228	1.86E-03
Cs-131	2.24E+03	P-33	8.15E+02	Tl-201	4.84E+02
Cs-134	1.29E-02	Pb-203	7.34E+00	W-187	1.01E-01
Cs-137	7.09E-02	Pb-210	5.60E+02	W-188	3.72E-01
Cu-67	2.30E+02	Pd-109	1.73E+02	Y-90	6.41E-03
Hg-203	1.86E+02	Ra-223	4.74E-01	Yb-169	1.47E+02
Ho-166	1.66E-01	Ra-224	2.44E-03	Yb-175	3.65E+02
I-125	1.06E+03	Ra-226	2.54E-03		
I-129	1.30E-03	Re-186	7.21E+01		

5.(b)(2) (continued)

(iii) For the contents described in 5(b)(1)(iii):

Total mass of contents and insert not to exceed 2.5 kg. Maximum mass of radioactive material, is 800 g.

TABLE 3

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Ac-225	2.08E-02	I-131	5.03E-02	Re-188	6.02E-02
Ac-227	5.40E-02	In-111	1.42E+02	Rh-105	148E+01
Ac-228	1.41E-03	Ir-192	2.10E-02	Se-75	1.28E+00
Am-241	1.13E+01	Ir-194	3.35E-02	Sm-153	3.15E+01
As-77	2.85E+00	Lu-177	3.43E+02	Sr-89	1.06E+01
Au-198	7.61E-02	Mo-99	1.70E-02	Sr-90	8.94E-01
Ba-131	2.31E-02	Na-24	1.79E-04	Tb-161	1.69E+01
C-14	1.28E+02	Np-237	2.08E-02	Th-227	1.16E-01
Co-60	3.68E-04	P-32	2.20E-02	Th-228	5.96E-04
Cs-131	2.24E+03	P-33	8.15E+02	Tl-201	4.84E+02
Cs-134	1.62E-03	Pb-203	5.70E-01	W-187	8.88E-03
Cs-137	5.85E-03	Pb-210	2.39E+02	W-188	6.31E-02
Cu-67	7.67E+01	Pd-109	1.50E+01	Y-90	6.02E-03
Hg-203	6.03E+01	Ra-223	5.46E-02	Yb-169	5.06E+01
Ho-166	4.46E-02	Ra-224	7.83E-04	Yb-175	2.56E+00
I-125	1.06E+03	Ra-226	6.81E-04		
I-129	5.20E-03	Re-186	6.93E+00		

(iv) For the contents described in 5(b)(1)(iv):

Total mass of contents and insert not to exceed 5.3 kg. Maximum mass of contents; i.e., radioactive material, is 429 g. Maximum volume of contents, including the material of the gas container, is 10cc.

TABLE 4

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Kr-79	2.00E-01	Xe-133	3.45E+02

5.(b)(2) (continued)

(v) For the contents described in 5(b)(1)(v):

Total mass of contents and insert not to exceed 2.5 kg. Maximum mass of radioactive material, is 800 g.

TABLE 5

Radionuclide	Maximum TBq	Radionuclide	Maximum TBq	Radionuclide	Maximum TBq
Pu-238	1.14E+01	Pu-240	6.72E+00	U-235	Note 1
Pu-239	Note 1	Pu-241	Note 1		

Note1: Fissile material must meet the mass limits and conditions of 10 CFR 71.15, "Exemption from classification as fissile material," or of the general license in 10 CFR 71.22 or 71.23. For shipment under the general license for fissile material, 10 CFR 71.22, and plutonium-beryllium special form material, 10 CFR 71.23, package contents are limited to no more than a Type A quantity of radioactive material.

5.(c) Criticality Safety Index

For the contents described in 5(b)(1)(v), as limited in 5(b)(2)(v), the criticality safety index must be calculated in accordance with the provisions of 10 CFR 71.22 or 71.23, as appropriate, and rounded up to the first decimal place. Criticality safety index is not required for material meeting the mass limits and conditions of 10 CFR 71.15.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7.0 of the application.

(b) The package must meet the Acceptance Tests and Maintenance Program in Section 8.0 of the application.

7. Air transport of plutonium is not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

9. Revision No. 0 of this certificate may be used until November 30, 2013.

10. Expiration date: January 31, 2016.

C FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9337	1	71-9337	USA/9337/B(U)-96	7	OF	7

REFERENCES

Croft Associates Limited application dated July 30, 2009.

Supplements dated: October 15, 2009, March 31, 2010, September 30, 2010, April 19 and September 6, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 16, 2012

NRC FORM 618 (8-2000) 10 CFR 71				U.S. NUCLEAR REGULATORY COMMISSION
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9341	2	71-9341	USA/9341/B(U)F–96	1 OF 6

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AREVA FEDERAL SERVICES LLC
505 336th ST Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA Federal Services LLC
application dated March 25, 2009.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: BEA Research Reactor (BRR) Package

(2) Description

A package used to transport fuel elements that have been irradiated in various test and research reactors. The package is comprised of a lead–shielded cask body, payload basket, an upper shield plug, a closure lid, upper and lower impact limiters, and utilizes ASTM Type 304 stainless steel as its primary structural material. The cask is a right circular cylinder 77.1 inches long and 38 inches in diameter, not including the impact limiter attachments and the thermal shield. Lead shielding is located between two circular shells, in the lower end structure, and in the shield plug. The payload cavity has a diameter of 16 inches and a length of 54 inches.

Impact limiters are attached to each end, having essentially identical design. Each limiter is 78 inches in diameter and 34.6 inches long overall, with a conical section 15 inches long towards the outer end. The impact limiter design consists of ASTM Type 304 stainless steel shells and approximately 9 lb/ft^3 polyurethane foam. There are four baskets used with the package, one for each type of fuel transported. The baskets are made from welded construction using ASTM Type 304 stainless steel in plate, bar, pipe, and tubular forms. Each basket has a diameter of 15.63 inches and a length of 53.45 inches, and features a number of cavities that fit the size and shape of the fuel.

The package is designed to be transported as one package per conveyance, with its longitudinal axis vertical, by highway truck or by rail in exclusive use. When loaded and prepared for transport, the package is 119.5 inches long, 78 inches in diameter (over the impact limiters), and weighs 32,000 lb.

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with AREVA Federal Services LLC drawings:

- 1910–01–01–SAR, BRR Package Assembly SAR Drawing, Sheets 1-4, Rev. 4
- 1910–01–02–SAR, BRR Package Impact Limiter SAR Drawing, Sheets 1-2, Rev. 1
- 1910–01–03–SAR, BRR Package Fuel Baskets SAR Drawing, Sheets 1-3, Rev. 4

(b) Contents

(1) Type and form of material

(i) Irradiated MURR fuel element to a maximum burnup of 180 MWD or a U-235 depletion of 30.9%. The minimum cooling time is 180 days after reactor shutdown. Each MURR element contains 24 fuel plates. Each fresh MURR element contains 775.0 ± 7.8 g U–235. The enrichment range is 93 ±1 wt.% U-235. The MURR element overall length, including irradiation growth, is 32.75 inches. The maximum decay heat per fuel element is 158 W. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 15 lb. Pre-irradiated MURR fuel element dimensions are in Table 1.1.

Table 1.1

MURR - Key Fuel Element Parameters	
Maximum active fuel length (inches)	24.8
Overall length (inches)	32.75
Minimum cladding thickness (inch)	0.008
Nominal fuel matrix thickness (inch)	0.02
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	782.8
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	46.0

(ii) Irradiated MITR–II fuel element to a maximum burnup of 165 MWD or a U–235 depletion of 43.9%. The minimum cooling time is 120 days after reactor shutdown. Each MITR-II element contains 15 fuel plates. Each fresh MITR–II element contains 510.0 +3.0/-10.0 g U–235, which is 500 - 513 g U-235. The enrichment range is 93 ±1 wt.% U-235. The MITR–II element overall length, including irradiation growth, is 26.52 inches. The maximum decay heat per element is 150 W. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 10 lb. Pre-irradiated MITR–II fuel element dimensions are in Table 1.2.

5.(b)(1) Type and form of material (continued)

Table 1.2

MITR-II - Key Fuel Element Parameters	
Maximum active fuel length (inches)	22.76
Overall length (inches)	26.52
Minimum cladding thickness (inch)	0.008
Nominal fuel matrix thickness (inch)	0.03
Maximum fuel matrix width (inches)	2.171
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	513
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	34.3

(iii) Irradiated ATR fuel element to a maximum burnup of 480 MWD or a U–235 depletion of 58.6%. The minimum cooling time is 1,670 days (4.6 years) after reactor shutdown. Each ATR fuel element contains 19 plates. The YA fuel element has 19 plates, but only 18 contain fuel. There are two general classes of ATR fuel element, XA and YA. The enrichment range is 93 ±1 wt.% U-235. The XA fuel element has a fresh fuel loading of 1,075 ± 10 g U–235. The YA fuel element has a fresh fuel loading of 1,022.4 ± 10 g U–235. A second YA fuel element design (YA–M) has the side plate width reduced by 15 mils. The ATR element overall length, after removal of the end box structures, 51.0 inches max. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 25 lb. The maximum decay heat per element is 30 W. Pre-irradiated ATR fuel element dimensions are in Table 1.3.

Table 1.3

ATR - Key Fuel Element Parameters	
Maximum active fuel length (inches)	48.77
Overall length (inches)	51
Minimum cladding thickness for Plate 1 (inch)	0.018
Minimum cladding thickness for Plates 2-18 (inch)	0.008
Minimum cladding thickness for Plate 19 (inch)	0.018
Nominal fuel matrix thickness (inch)	0.02
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	1,085
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	85.2

5.(b)(1) Type and form of material (continued)

(iv) Irradiated TRIGA fuel elements. Pre-irradiated TRIGA fuel element dimensions are in Table 1.4. The TRIGA fuel matrix is uranium mixed with zirconium hydride. The BRR package is limited to five specific TRIGA fuel types:

1. 8 wt% uranium in the fuel matrix, U - U/Zr with uranium aluminum clad element (General Atomics catalog number 101).
2. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 103).
3. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element, high enriched uranium (General Atomics catalog number 109). This fuel element is sometimes referred to in the literature as a Fuel Life Improvement Program (FLIP) element.
4. 20 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 117). This fuel element is sometimes referred to in the literature as a FLIP–LEU–I element.
5. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element, instrumented (General Atomics catalog number 203).

Table 1.4
TRIGA - Fresh Fuel Element Characteristics

Parameter	GA Cat. # 101	GA Cat. # 103	GA Cat. # 109	GA Cat. # 117	GA Cat. # 203
Maximum Active Fuel Length (in)	14	15	15	15	15
Fuel Pellet OD (in)	1.41	1.44	1.44	1.44	1.44
Overall Element Length (in)	28.37	28.9	28.9	29.68	45.25
Cladding OD (in)	1.48	1.48	1.48	1.48	1.48
Minimum Cladding Thickness (in)	0.0285	0.0185	0.0185	0.0185	0.0185
Graphite Reflector Length Top/Bottom (in)	4.0 / 4.0	2.6 / 3.7	2.6 / 3.7	2.6 / 3.7	3.1 / 3.4
Maximum Zr Mass in Fuel Matrix (g)	2,070	2,088	2,060	2,060	2,088
Maximum U-235 Mass (g) per element	36	39	137	101	39
Maximum U-235 Enrichment (wt%)	20	20	70	20	20
Maximum H/Zr atom ratio	1.0	1.7	1.6	1.6	1.7

The maximum length of a TRIGA fuel element, including irradiation growth, is 45.50 inches. For all fuel elements, spacers are utilized within the TRIGA baskets. The bounding weight of any TRIGA fuel element is 10 lb. The maximum decay heat per element is 20 W. The number of TRIGA rods per element is 1. TRIGA fuel parameters are in Table 1.5.

5.(b)(1) Type and form of material (continued)

Table 1.5

TRIGA - Fuel Parameters			
Fuel Type	**Maximum U-235 depletion (%)**	**Maximum Burnup (MWD/MTU)**	**Minimum Decay Time**
GA Cat. # 101	22.42	36,953	28 days
GA Cat. # 103/203	20.72	34,111	28 days
GA Cat. # 109	59.74	339,368	1 year
GA Cat. # 117	43.81	75,415	1 year

5.(b)(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i):

8 irradiated MURR fuel elements. Only one fuel element is allowed per basket location.

(ii) For the contents described in 5(b)(1)(ii):

8 irradiated MITR-II fuel elements. Only one fuel element is allowed per basket location.

(iii) For the contents described in 5(b)(1)(iii):

8 irradiated ATR fuel elements. Only one fuel element is allowed per basket location.

(iv) For the contents described in 5(b)(1)(iv):

19 irradiated TRIGA fuel elements. Only one fuel element is allowed per basket location.

(c) Criticality Safety Index (CSI): 0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

(b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

NRC FORM 618 (8-2000) 10 CFR 71		U.S. NUCLEAR REGULATORY COMMISSION		
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9341	2	71-9341	USA/9341/B(U)F–96	6 OF 6

8. Transport by air of fissile material is not authorized.

9. Expiration date: January 22, 2015.

REFERENCES

AREVA Federal Services LLC application dated March 25, 2009.

Supplements dated August 6, 2009, November 5, 2009, June 4, 2010, December 16, 2010 and June 24, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 22, 2011

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9342	8	71-9342	USA/9342/AF-96	1	OF	4

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

   a. ISSUED TO *(Name and Address)*
   Century Industries
   P.O. Box 17084
   Bristol, VA 24209

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
   Century Industries application dated March 19, 2013, as supplemented.

4. CONDITIONS

   This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Versa-Pac in two configurations, i.e., VP-55 and VP-110.

(2) Description

The Model No. Versa-Pac is either a 55-gallon (Model No. VP-55) or a 110-gallon (Model No. VP-110) package for shipment of uranium oxides, uranium metal, uranyl nitrate crystals and other uranium compounds, e.g., uranium carbides, uranyl fluorides and uranyl carbonates, and thorium 232 as TRISO fuel.

The exterior skin of the packaging is a UN1A2/X400/S minimum, 16 gauge carbon steel material for the Model No. VP-55 and a UN1A2/Y409/S minimum, 16 gauge carbon steel for the Model No. VP-110.

Both models use a 12 gauge bolted closure ring, ASTM A 307 bolts and nuts, a closed-cell EPDM gasket, a drum cover reinforced by a 10 gauge thick plate with four or eight bolts depending upon the Model No. VP-55 or VP-110, respectively.

Both models are strengthened with vertical stiffeners, two inner liners insulated by a ceramic fiber blanket and a ¼" carbon steel reinforcing plate on the bottom. The packaging's interior

5.(a) (2) Description (Continued)

is completely insulated with layers of a ceramic fiber blanket around the containment cavity with rigid polyurethane foam disks on the top and bottom of the cavity.

A ½" thick fiberglass ring is used as a thermal break at the payload cavity flange. The cavity blind flange is secured to the flange with twelve bolts.

The primary containment boundary is defined as the payload cavity with its associated welds, the containment end plate, the inner flange ring, the silicone-coated fiberglass gasket, the cavity blind flange, and the bolts.

The approximate dimensions and weights of the packaging are as follows:

Model No.	Packaging OD (in.)	Packaging Height (in.)	Payload Containment Cavity ID (in.)	Payload Containment Cavity Height (in.)	Packaging Weight (lbs.)	Maximum loaded weight (lbs.)
VP-55	23-1/16	34 ¾	15	25-7/8	390	640
VP-110	30-7/16	42 ¾	21	29-3/4	705	965

(3) Drawings

The packaging is constructed and assembled in accordance with Century Industries Drawing Nos.:

VP-55-LD-1 Rev. No. 11, VP-55-LD-2 Rev. No. 12, sheets 1 of 2 and 2 of 2.

VP-110-LD-1 Rev. No. 11, VP-110-LD-2 Rev. No. 10, sheets 1 of 2 and 2 of 2.

5.(b) Contents

(1) Type and form of material

Solid, homogeneous (powder or crystalline), or non-homogeneous, uranium materials with no free-standing liquids. Materials shall be stable and in a non-pyrophoric form. Density is not limited.

5(b)(1) Type and Form of Material (Continued)

Contents are limited to:

(i) A. Uranium oxides (U_xO_y).

B. Uranyl nitrate crystals in the form of uranyl nitrate hexahydrate, trihydrate or dihydrate.

C. Other uranium compounds, e.g., uranyl fluorides and uranyl carbonates. Uranium compounds may also contain carbon or be mixed with carbon or graphite. Uranium carbide is authorized for shipment. However, uranium hydrites are not authorized for shipment.

D. Uranium metal or uranium alloys.

(ii) TRISO fuel as C/SIS/C coated $ThUC_2$ particles pressed with a carbon matrix to form rods.

Contents may be pre-packaged in polyethylene, polytetrafluoroethylene, aluminum, and carbon steel per Table No.1-4 of the application. Aluminum Trihydrate, Sodium Borate (Borax, fused), perlite, paper labels, plastic tape, plastic bags, plastic bottles and desiccant such as "Quik-Solid" are also authorized as packing materials. Materials with a hydrogen density greater than 0.141 g/cm^3 are not authorized.

Radioactive contents shall have an auto-ignition temperature and melting point greater than 600°F.

(2) Maximum quantity of material per package:

Not to exceed 350 grams U-235 enriched up to 100 weight percent.

The net weight of the authorized contents shall not exceed 250 lbs for the Model No. VP-55, and 260 lbs for the Model No. VP-110.

(3) Contents are limited to normal form material. The radionuclide inventory of the loaded contents, including U-234 and U-236, shall be less than the calculated mixture A_2 value.

(4) Decay heat is limited to 11.4 W.

5(c) Criticality Safety Index (CSI): 1.0

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6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section No. 7 of the application.

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Section No. 8 of the application.

7. Transport by air of fissile material is not authorized.

8. Transport of plutonium above minimum detectable quantities is not authorized.

9. Packages must be marked with the appropriate model number, i.e., VP-55 or VP-110, as applicable. The neoprene 1/8 inch bottom pad and 3/8 inch top pad are optional for packages that are not intended to be reused.

10. Content forms may not be mixed in a single package.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

12. Expiration date: June 30, 2015.

REFERENCES

Century Industries application "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container," Revision No. 7, dated March 19, 2013.

Supplements dated: April 15, 23, and 25, and May 7, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 23, 2013

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
QSA Global, Inc., application dated April 20, 2011.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: SENTRY

(2) Description

The Model No. SENTRY package includes the Model Nos. SENTRY 110, SENTRY 330, and SENTRY 867, as three variations of the same design. The external dimensions of all models in their standard transport configurations, i.e., with the handling rib and link plate assemblies, are identical and are approximately 19 inches (48 cm) wide, 19 inches (48 cm) tall, and 19 inches (48 cm) deep.

The primary components of the Model No. SENTRY package include (i) a depleted uranium shield completely encased and supported in a cylindrically shaped, stainless steel, welded body, (ii) the rear plate lock and front plate assemblies, (iii) the handling rib and link plate, and (iv) the source assembly. The inner cavity of the welded body around the shield is filled with polyurethane foam. The Model Nos. SENTRY 110 and 330 packages can contain only one source wire assembly during transport, while two source wire assemblies can be loaded into the Model No. SENTRY 867 package. The radioactive contents are securely positioned by either a lock slide for the Model Nos. SENTRY 110 and 330 packages or locking pins for the Model No. SENTRY 867 package. All lock assemblies include a dust cover with a plunger lock to prevent rotation of the selector ring and further secure the source in the package during transport.

The optional rib/link assemblies provide lifting attachments and are bolted to the body weldment. The maximum weight, including the optional rib/link assemblies, is 780 pounds (354 kg) for the Model Nos. SENTRY 330 and 867 packages, and 605 pounds (274 kg) for the Model No. SENTRY 110 package.

(3) Drawings

The package is constructed in accordance with QSA Global, Inc., Drawing No. R86000, Rev. F, sheets 1-10.

(b) Contents

(1) Type and form of material

Cobalt-60, as a sealed source, which meets the requirements of special form radioactive material.

All source wire assemblies consist of a special form capsule crimped onto the end of a flexible steel wire.

(2) Maximum quantity of material per package:

Co-60: 110 curies (4.07 TBq) (output) for the Model No. SENTRY 110 package.

Co-60: 330 curies (12.2 TBq) (output) for the Model Nos. SENTRY 330 and 867 packages.

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 1.30 R/(h-Ci) (Ref: American National Standards Institute N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography").

(3) Maximum weight of contents:

0.09 pounds (40 grams) for the Model Nos. SENTRY 110 and 330 packages.

0.18 pounds (80 grams) for the Model No. SENTRY 867 package.

The maximum content weight includes the mass of radioactive material and the source capsule handling wire assembly for a shipment containing the maximum number of source wire assemblies that can be transported in a package, i.e., 1 source wire assembly for the Model Nos. SENTRY 110 and 330 packages, and 2 source wire assemblies for the Model No. SENTRY 867 package.

(4) Maximum decay heat: 5.5 watts

6. A cover over the source wire connector prevents access to the source assembly until a keyed lock is actuated and the cover removed. This cover stays in place during transport of the package.

7. The nameplate shall maintain its legibility and be fabricated of materials capable of resisting the fire test of 10 CFR Part 71.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application;

   (b) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

9. Supplemental shielding shall not exceed 5% of the maximum weight of the depleted uranium casting, with a thickness not to exceed 0.5 inch.

10. Revision No. 1 of this certificate may be used until September 30, 2014.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

12. Expiration date: July 30, 2016.

<u>REFERENCES</u>

QSA Global Inc., application dated April 20, 2011.

Supplements dated May 22, 2012 and July 19, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 13, 2013

NRC FORM 618

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, MD 21405

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

TN-LC Transportation Package Safety Analysis Report, Revision No. 6, dated November 2012, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TN-LC

(2) Description

The packaging, designed for transport of irradiated test, research, and commercial reactor fuel in either a closed transport vehicle or an ISO container, consists of a payload basket, a shielded body, a shielded closure lid and top and bottom impact limiters. The packaging body is a right circular cylinder, approximately 197.5 inches long and 30 inches in diameter, composed of top and bottom end flange forgings connected by inner and outer shells. Lead shielding, made of ASTM B29 copper lead, is placed between the two cylindrical shells, in the bottom end assembly, and in the lid. Neutron shielding, composed of a borated resin compound inserted into twenty aluminum shield boxes, is set between the outer shell and a 0.25 inch-thick Type 304 stainless steel outer sheet. Two removable trunnions are bolted to the packaging body using eight 1-8UNC bolts for each trunnion. Two pocket trunnions in the bottom flange, used for rotating the package, may also be used for horizontal package lifting. Impact limiters, with an approximate outside diameter of 66 inches and height of 22.75 inches, consisting of balsa and redwood blocks encased in stainless steel shells, are attached to each end of the packaging during shipment, each with eight 1-8UNC bolts.

Four basket designs are provided for transport of Boiling Water Reactor (BWR), Pressurized Water Reactor (PWR), Mixed Oxide Fuel (MOX), Evolutionary Pressurized Reactor (EPR), National Research Universal Reactor (NRU), National Research Experimental

5.(a)(2) Description (Continued)

Reactor (NRX), Material Test Reactor (MTR), and Training, Research, and Isotope General Atomics Reactor (TRIGA) fuel assemblies, fuel elements or fuel rods.

The packaging may be loaded or unloaded either in a pool or a hot cell environment. The spent fuel payload is shipped dry in a helium atmosphere.

Nominal weights and dimensions are as follows:

- Overall length with impact limiters: 230 inches
- Overall length without impact limiters: 197.50 inches
- Cavity length (minimum): 182.50 inches
- Cavity inner diameter: 18 inches
- Lid thickness: 7.50 inches
- Weight of contents: 7,100 lbs
- Weight of lid: 1,000 lbs
- Weight of impact limiters: 3,000 lbs
- Total loaded weight of the package: 51,000 lbs

(3) Drawings

The packaging is constructed and assembled in accordance with the following drawings:

65200-71-01 Revision 5	TN-LC Cask Assembly (11 sheets)
65200-71-02 Revision 0	TN-LC Transport Cask Regulatory Plate (1 sheet)
65200-71-20 Revision 3	TN-LC Impact Limiter Assembly (3 sheets)
65200-71-21 Revision 0	TN-LC Transport Packaging Transport Configuration (1 sheet)
65200-71-40 Revision 3	TN-LC-NRUX Basket Basket Assembly (5 sheets)

65200-71-50 Revision 3	TN-LC-NRUX Basket Basket Tube Assembly (5 sheets)
65200-71-60 Revision 3	TN-LC-MTR Basket General Assembly (4 sheets)
65200-71-70 Revision 3	TN-LC-MTR Basket Fuel Bucket (2 sheets)
65200-71-80 Revision 3	TN-LC-TRIGA Basket (5 sheets)
65200-71-90 Revision 3	TN-LC-1FA Basket (5 sheets)
65200-71-96 Revision 3	TN-LC-1FA BWR Sleeve and Hold-Down Ring (2 sheets)
65200-71-102 Revision 3	TN-LC-1FA 25 Pin Can Basket (5 sheets)

5.(b) Contents

(1) Type and Form of Material

(i) Intact or damaged NRU and NRX Mk I fuel assemblies which meet the specifications listed in Table 1 below, respectively, are authorized for transportation in the TN-LC-NRUX basket.

Intact fuel assemblies are fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

Damaged fuel assemblies, with cladding damage in excess of pin hole leaks or hairline cracks, are authorized only if the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod.

5.(b)(1) Type and Form of Materials (continued)

Table 1

NRU and NRX Mk I Fuel Specifications for Transport in the TN-LC-NRUX Basket

Parameter	NRU	NRX Mk I
Physical and Material Description		
Number of Assemblies	≤ 26	≤ 26
Number of rods/assembly	≤ 12	7
Assembly length (inch) [(1)]	≤ 116	≤ 116
Nominal Assembly mass (g)	4660	5780
Fuel form	U-Al	U-Al
^{235}U per rod (g)	≤ 45.4	≤ 75.2
Enrichment (wt.% ^{235}U)	≤ 93	≤ 93
Cladding and Spacer Material	Al	Al
Thermal and Radiological Parameters		
Cooling Time (years) [(2)]	≥ 10	≥ 10
Depletion (wt.% ^{235}U) [(3)]	≤ 80	≤ 80
Decay Heat per Assembly (watts) [(4)]	≤ 15	≤ 15

Notes:

1. Maximum length of the fuel assembly (unirradiated) for shipment.
2. The cooling time of the fuel assembly rounded down to 0.5 years.
3. The depletion (or burnup) of the fuel assembly rounded up to 0.5%.
4. The decay heat of the fuel assembly is less than 15 watts at the maximum burnup and minimum cooling time.

(ii) Intact or damaged MTR fuel elements that are enveloped or bounded by the fuel element design characteristics listed in Table 2 below, with an average burnup and minimum cooling time as specified in Table 3 below, and a maximum decay heat of 25 watts per element, are authorized for transportation in the TN-LC-MTR basket.

Intact fuel elements are fuel elements containing fuel plates with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

Damaged fuel elements, with cladding damage in excess of pin hole leaks or hairline cracks, are authorized only if the total surface area of the damaged cladding does not exceed 5% of the total surface area of each element.

The MTR fuel assemblies shall meet all the requirements in Table 3.

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Table 2

MTR Fuel Element Design Characteristics

Fuel Element Class	M-01	M-02	M-03	M-04	M-05	M-06	M-07	M-08[(1)]
Number of Fuel Plates [(2)]	≤23	≤21	≤19	≤17	≤10	≤18	≤17	≤23
^{235}U mass per Plate (g)	≤16	≤16.5	≤17.5	≤19	≤22	≤20.5	≤11.5	≤22
Active Fuel Width (cm)	≤6.7	≤6.7	≤6.7	≤6.7	≤6.7	≤5.9	≤6.7	≤6.7
Active Fuel Length (cm)	≥ 56	≥ 56	≥ 56	≥ 56	≥ 56	≥ 56	≥ 27.5	≥ 56
Enrichment (wt.% ^{235}U)	≤ 94	≤ 94	≤ 94	≤ 94	≤ 94	≤ 94	≤ 94	≤ 94
Fuel Element Depth (cm)	≥7.5	≥7.5	≥7.5	≥7.5	≥7.5	≥7.5	≥7.5	≥7.5

Notes:
1. The M-08 Element class requires that the central stack of fuel elements remain empty. Also, the total ^{235}U mass is limited by the maximum value in Table 3.
2. The plate thickness is greater than 0.12 cm and the clad thickness is greater than 0.02 cm.

Table 3

MTR Fuel Element Qualification

Enrichment Type	Burnup (MWd/MTU)	Cooling Time (days)
Type A ^{235}U Enrichment ≥ 90% ^{235}U Mass ≤ 380 g	66,000	740
	165,000	1120
	330,000	1440
	495,000	1680
	660,000	1950
Type B ^{235}U Enrichment ≥ 90% 380 g < ^{235}U Mass ≤ 460 g	57,750	770
	144,375	1150
	288,750	1470
	433,125	1710
	577,500	1950
Type C 40% ≤ ^{235}U Enrichment < 90% ^{235}U Mass ≤ 380 g	29,330	740
	73,325	1120
	146,650	1440
	219,975	1690
	293,300	1940
Type D 19% ≤ ^{235}U Enrichment < 40% ^{235}U Mass ≤ 470 g	13,930	830
	34,825	1220
	69,650	1560
	104,475	1850
	139,300	2150

5.(b)(1) Type and Form of Materials (continued)

Notes

- Burnup = fuel element average burnup.
- Use burnup (MWd/MTU) and Enrichment Type (A, B, C, or D with limits on ^{235}U enrichment and ^{235}U mass per element) to look up minimum cooling time in days. Licensee is responsible for ensuring that uncertainties in burnup, enrichment, and mass are applied conservatively.
- Fuel with burnups greater than those listed for each Enrichment Type is not authorized for transport.
- Burnups may be either rounded up to the next higher burnup or linear interpolation may be used to determine the minimum cooling time. However, for conservatism, an additional cooling time of 30 days must be added to any linearly interpolated value.
- Example: An M-06 class element with an enrichment of 45 wt.% ^{235}U and a ^{235}U mass of 350 grams is classified as enrichment Type C. A burnup of 100,000 MWd/MTU is acceptable for transport after 1440 days cooling time as defined by 146,650 MWd/MTU from the qualification table (when linear interpolation is not employed). When linear interpolation is employed the minimum required cooling time is 1267 days (1237 days based on interpolation + 30 days additional cooling time).

(iii) Intact TRIGA fuel assemblies/elements that are enveloped by the fuel assemblies/element design characteristics listed in Table 4, intact TRIGA fuel follower control rods that are enveloped by the fuel assembly/element design characteristics listed in Table 5, with an average burnup and minimum cooling time meeting the specifications of Table 6 for fuel assemblies/elements or of Table 7 for follower control rods, and a maximum decay heat of 8 watts per assembly/element, are authorized for shipment with the TN-LC-TRIGA basket.

Intact fuel assemblies/elements are fuel assemblies/elements containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks. The design characteristics of the TRIGA fuel assemblies/elements are described in Tables 4 and 5 below.

The fuel qualification Tables 6 and 7 specify the maximum assembly/element average burnup and minimum cooling time. The fuel elements/assemblies shall meet all the requirements of Tables 6 and 7.

The poison plates in TN-LC-TRIGA basket are constructed from either boron aluminum alloy, or metal matrix composite (MMC), or Boral®. The minimum areal density of Boron-10 (^{10}B) for either the boron enriched aluminum alloy or the metal matrix composite is 5.56 mg/cm². The minimum areal density of ^{10}B for Boral® is 6.67 mg/cm².

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Table 4

TRIGA Fuel Assembly/Element Design Characteristics

Assembly/Element Type	Al Clad	ACPR (1)	Standard	FLIP (2)	FLIP (2) LEU-I (3)	FLIP (2) LEU-II (3)
Element ID	T-01	T-02	T-03	T-04	T-05	T-06
Fuel Material	U-ZrH	U-ZrH	U-ZrH	U-ZrH	U-ZrH	U-ZrH
Enrichment (wt.% ^{235}U)	≤ 20	≤ 20	≤ 20	≤ 70	≤ 20	≤ 20
^{235}U-Mass (g)	≤ 41	≤ 56	≤ 41	≤ 137	≤ 101	≤ 169
Active Fuel Length (inch)	≤ 15	≤ 15	≤ 15	≤ 15	≤ 15	≤ 15
Pellet Diameter (inch)	≤ 1.41	≤ 1.41	≤ 1.44	≤ 1.44	≤ 1.44	≤ 1.44
Clad Material	Al	SS304	SS304	SS304	SS304	SS304

Table 5

TRIGA Fuel Follower Control Rods Design Characteristics

Assembly/Element Type	Standard	FLIP (2) LEU-I (3)	ACPR (1)
Element ID	T-07	T-08	T-09
Fuel Material	U-ZrH	U-ZrH	U-ZrH
Enrichment (wt. % ^{235}U)	≤ 20	≤ 20	≤ 20
^{235}U-Mass (g)	≤ 38	≤ 97	≤ 56
Active Fuel Length (inch)	≤ 15	≤ 15	≤ 15
Pellet Diameter (inch)	≤ 1.32	≤ 1.32	≤ 1.32
Clad Material	SS304	SS304	SS304

Notes:
1. ACPR - Annular Core Pulse Reactor
2. FLIP - Fuel Life Improvement Program
3. LEU - Low Enriched Uranium

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5.(b)(1) Type and Form of Materials (continued)

Table 6

TRIGA Fuel Qualification for Fuel Assembly/Elements

Element ID	Burnup (MWd/MTU)	Cooling Time (days)
T-01	35,750	400
	71,500	560
	107,250	640
	143,000	710
T-02	35,750	650
	71,500	970
	107,250	1310
	143,000	1870
T-03	35,750	520
	71,500	840
	107,250	1170
	143,000	1730
T-04	112,500	1000
	225,000	1380
	337,500	1820
	450,000	2520
T-05	35,750	920
	71,500	1290
	107,250	1710
	143,000	2360
T-06	36,500	1190
	73,000	1690
	109,500	2320
	146,000	3170

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5.(b)(1) Type and Form of Materials (continued)

Table 7

TRIGA Fuel Qualification for Fuel Follower Control Rods

Element ID	Burnup (MWd/MTU)	Cooling Time (days)
T-07	35,750	540
	71,500	890
	107,250	1280
	143,000	1960
T-08	35,750	940
	71,500	1350
	107,250	1840
	143,000	2580
T-09	35,750	670
	71,500	1020
	107,250	1420
	143,000	2100

Notes for Tables 6 and 7:

- Burnup = fuel element / assembly / follower control rod average burnup.
- Use burnup (MWd/MTU) and Element ID to look-up minimum cooling time in days. Licensee is responsible for ensuring that uncertainties in burnup are applied conservatively.
- Fuel with a burnup greater than that listed for each element type in Tables 6 and 7 is unacceptable for transport.
- Burnups may be either rounded up to the next higher burnup or linear interpolation may be used to determine the minimum cooling time. However, for conservatism, an additional cooling time of 30 days must be added to any linearly interpolated value.
- Example: A T-03 element with a burnup of 100,000 MWd/MTU is acceptable for transport after 1170 days cooling time as defined by 107,250 MWd/MTU (Table 6, rounding up) on the qualification table (when linear interpolation is not employed). When linear interpolation is employed the minimum required cooling time is 1133 days (1103 days based on interpolation + 30 days additional cooling time).

(iv) Intact PWR fuel assembly, as specified in Table 8, or intact BWR fuel assembly, as specified in Table 13, or fuel rods in a pin can are authorized for transport with the TN-LC-1FA basket.

Intact fuel assemblies are fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

5.(b)(1) Type and Form of Materials (continued)

The fuel rods include irradiated PWR, BWR, MOX, and EPR fuel rods. PWR intact and BWR intact fuel rods may be from any of the fuel assemblies listed in Table 8 or Table 13, respectively.

MOX rods have the same geometry as PWR or BWR rods, as defined in Table 8 and Table 13. The composition of MOX fuel is specified in Table 12.

The EPR fuel rods are specified in Table 10.

The poison plates in the TN-LC-1FA basket are constructed from boron aluminum alloy, or metal matrix composite (MMC), or Boral®. The minimum ^{10}B aeral density of the poison plate is 16.7 mg/cm^2 for either the boron aluminum alloy or the MMC. The minimum ^{10}B aeral density of the poison plate is 20.0 mg/cm^2 for Boral®.

In addition to the poison plates provided in the basket, Poison Rod Assemblies (PRAs) are required for transportation of PWR fuel assemblies. The minimum required B_4C content of the absorber rods in the PRA is 40% Theoretical Density (TD). A summary of the number of absorber rods required in the PRA for each PWR fuel class is shown in Table 11. PRA loading configurations are also illustrated in Figure 1 through Figure 4.

The PWR fuel assemblies fuel qualification table (FQT) is provided in Table 15.

The BWR fuel assemblies FQT is provided in Table 16.

The PWR rod FQTs are shown in Table 17 and Table 18 for the 25 and 9 rod configurations, respectively.

The BWR rod FQTs are shown in Table 19 and Table 20 for the 25 and 9 rod configurations, respectively.

The MOX rod FQT, provided in Table 21 for both 25 and 9 rods, is applicable to both BWR and PWR MOX rods.

The FQTs for the UO_2 Standard EPR rods are governed by the PWR rod FQTs (Table 17 and Table 18), while the FQT for the MOX EPR rods is governed by the MOX rod FQT (Table 21).

5.(b)(1) Type and Form of Materials (continued)

Table 8
PWR Fuel Specifications for Transport in the TN-LC-1FA Basket

Fuel Class (1) (2)	One intact unconsolidated B&W 17x17, WE 17x17, CE 16x16, B&W 15x15, WE 15x15, CE 15x15, WE 14x14, or CE 14x14 class PWR assembly (without control components) that are enveloped by the fuel assembly design characteristics listed in Table 9. Reload fuel manufactured by the same or other vendors, but enveloped by the design characteristics listed in Table 9, is also acceptable.
Maximum Assembly + PRA Weight	1850 lbs
Fissile Material	UO_2
Maximum Initial Uranium Content(4)	490 kg/assembly
Maximum Unirradiated Assembly Length	178.3 inches
Fuel Assembly Average Burnup, Enrichment and Minimum Cooling Time	Per Table 15
Maximum Planar Initial Enrichment	5.0(3) wt.% ^{235}U
Maximum Decay Heat(5)	3.0 kW per Assembly
Minimum ^{10}B areal density in poison plates	• 16.7 mg/cm^2 (Natural or Enriched Boron Aluminum Alloy / Metal Matrix Composite (MMC)) • 20.0 mg/cm^2 (Boral®)
Minimum number of absorber rods per PRA as a function of assembly class	Per Table 11

Notes:
1. Up to 25 PWR fuel rods from any of the PWR fuel assemblies listed in Table 9 may also be transported in the TN-LC-1FA basket in a 25 pin can. The fuel rods are loaded in a 25 pin can with a cavity length of 168.5 inches (Option 3) which is placed within the TN-LC-1FA basket. The maximum peak burnup for the fuel rods is 90 GWd/MTU. The required cooling time, as a function of a PWR fuel rod burnup and enrichment, is provided in Table 17 for 25 rods and Table 18 for 9 rods, respectively.
2. Up to 25 EPR fuel rods from any of the fuel class listed in Table 9 and meeting EPR rod parameters specified in Table 10 may also be loaded in the TN-LC-1FA basket. The fuel rods are loaded in a 25 pin can with a cavity length of 179.5 inches (Option 1 and Option 2) which is placed within the TN-LC-1FA basket. The maximum peak burnup for the fuel rods is 90 GWd/MTU. The required cooling time, as a function of an EPR fuel rod burnup and enrichment, is provided in Table 17 for 25 rods and Table 18 for 9 rods, respectively.
3. For CE 15x15, the maximum planar average initial enrichment is 3.70 wt.% ^{235}U.
4. The maximum initial uranium content is based on the shielding analysis. The listed value is higher than the actual.
5. The maximum decay heat per rod is 220 watts when loading up to 9 rods. The maximum decay heat per rod is 120 watts when loading 10 or more (up to 25) rods.

5.(b)(1) Type and Form of Materials (continued)

Table 9
PWR Fuel Assembly Design Characteristics for Transportation in the TN-LC-1FA Basket

Assembly Class	B&W 15x15	B&W 17x17	WE 17x17	CE 15x15	WE 15x15	CE 14x14	WE 14x14	CE 16x16
Maximum Number of Fuel Rods	208	264	264	216	204	176	179	236
Maximum Number of Guide/Instrument Tubes	17	25	25	9	21	5	17	5
Rod Pitch[1] (inch)	≤ 0.568	≤ 0.502	≤ 0.496	≤ 0.550	≤ 0.563	≤ 0.580	≤ 0.556	≤ 0.506
Pellet Diameter[1] (inch)	≤ 0.374	≤ 0.323	≤ 0.323	≤ 0.360	≤ 0.367	≤ 0.382	≤ 0.368	≤ 0.326
Clad Outer Diameter[1] (inch)	≥ 0.416	≥ 0.379	≥ 0.360	≥ 0.417	≥ 0.422	≥ 0.440	≥ 0.400	≥ 0.382
Clad Thickness[1] (inch)	≥ 0.024	≥ 0.024	≥ 0.022	≥ 0.026	≥ 0.024	≥ 0.026	≥ 0.022	≥ 0.023

Note 1. The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type).

Table 10
Irradiated EPR Fuel Rod Parameters

Parameter	Value
Maximum Unirradiated Length	179.5 inches
Cladding Thickness	Nominal 0.022 inch
Maximum Initial Uranium Content	2.05 kgU/rod

Table 11
Summary of PRA Requirements for PWR Fuel Assembly Classes

Assembly Class	Number of Absorber Rods in PRAs and Locations	Diameter of B_4C Absorber (cm)	Minimum B_4C Content (g/cm)
WE 17x17	8, Per Figure 4	0.88	0.613
CE 16x16	5, All Guide Tubes	1.02	0.824
BW 15x15	8, Per Figure 3	0.88	0.613
CE 15x15	1, Center Guide Tube	0.76	0.475
WE 15x15	8, Per Figure 2	0.88	0.613
CE 14x14	5, All Guide Tubes	1.02	0.824
WE 14x14	8, Per Figure 1	0.88	0.613
BW 17x17	8, Per Figure 4	0.76	0.475

5.(b)(1) Type and Form of Materials (continued)

Table 12
MOX Fuel Rods Specifications for Transport in the TN-LC-1FA Basket

PHYSICAL PARAMETERS:	• Up to 25 PWR MOX fuel rods with physical parameters as those listed in Table 8. • Up to 25 BWR MOX fuel rods with physical parameters as those listed in Table 13. • Up to 25 EPR MOX fuel rods with physical parameters as those listed in Table 10.
Fissile Material	UO_2, PuO_2 (Mixed Oxide or MOX)
Heavy Metal (HM) Content	≤ 2.5 kgU/rod
CRITICALITY PARAMETERS: Initial MOX composition	• ^{235}U Content in UO_2: $0.5 \le {}^{235}U \le 0.7$ wt.% • Plutonium Content: $Pu / (U + Pu) \le 7.0$ wt.% • Initial ^{239}Pu Content in $PuO_2 \le 60.0$ wt.% • Initial ^{241}Pu Content in $PuO_2 \le 7.5$ wt.%
THERMAL/RADIOLOGICAL PARAMETERS: Initial MOX Composition for Fuel Qualification	• $^{238}Pu / {}^{239}Pu \le 4.0$ wt.% • $^{239}Pu/ PuO_2 \ge 50$ wt.% • $^{241}Am / PuO_2 \le 70$ ppm • $^{235}U/U \le 0.5$ wt.%
Burnup and Minimum cooling time for MOX rods	Per Table 21.
Maximum Decay heat per 25 pin can	• 3.0 kW for the 25 pin can with up to 25 rods • 1.98 kW for the 25 pin can with up to 9 rods
Minimum ^{10}B aeral density in poison plates	• 16.7 mg/cm^2 Boron Aluminum Alloy / Metal Matrix Composite (MMC) • 20.0 mg/cm^2 (Boral®)

5.(b)(1) Type and Form of Materials (continued)

Table 13
BWR Fuel Specification for Transport in the TN-LC-1FA Basket

PHYSICAL PARAMETERS: Fuel Class(1)	One intact 7x7, 8x8, 9x9, or 10x10 BWR assembly manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table 14.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Fissile Material	UO_2
Maximum Assembly Weight with Channels	790 lbs
Maximum Unirradiated Assembly Length	176.6 inches
THERMAL/RADIOLOGICAL PARAMETERS: Maximum Planar Average Initial Enrichment	5.0 wt.% ^{235}U
Fuel Assembly Average Burnup, Enrichment and Minimum Cooling Time	Per Table 16.
Maximum Decay Heat(2)	2.0 kW per Assembly
Minimum ^{10}B aeral density in poison plates	• 16.7 mg/cm^2 Boron Aluminum Alloy / Metal Matrix Composite (MMC) • 20.0 mg/cm^2 (Boral®)

Notes:

1. Up to 25 fuel rods from any of the BWR fuel assemblies listed in Table 14 may also be transported in the TN-LC-1FA basket in the 25 pin can. The fuel rods are loaded in a 25 pin can with a cavity length of 168.5 inches which is placed within the TN-LC-1FA basket. The required cooling time as a function of BWR fuel rod burnup and enrichment are provided in Table 19 for 25 rods and Table 20 for 9 rods, respectively.
2. The maximum decay heat per rod is 220 watts when loading up to 9 rods. The maximum decay heat per rod is 120 watts when loading 10 or more (up to 25) rods.

5.(b)(1) Type and Form of Materials (continued)

Table 14
BWR Fuel Assembly Design Characteristics[1] for Transportation in the TN-LC-1FA Basket
(Part 1 of 3)

Transnuclear ID	7x7-49/0	8x8-63/1	8x8-62/2	8x8-60/4	8x8-60/1	9x9-74/2
Initial Design or Reload Fuel Designation	GE1	GE4	GE-5	GE8 Type II	GE9	GE11
	GE2		GE-Pres		GE10	GE13
	GE3		GE-Barrier			
			GE8 Type I			
			FANP 8x8-2			
Maximum Number of Fuel Rods	49	63	62	60	60	74
Maximum Initial Uranium Content (kg)	198	192	192	192	192	192
Rod Pitch[5] (inch)	≤ 0.738	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.566
Pellet Diameter[5] (inch)	≤ 0.487	≤ 0.416	≤ 0.411	≤ 0.411	≤ 0.411	≤ 0.376
Clad Outer Diameter[5] (inch)	≥ 0.563	≥ 0.493	≥ 0.483	≥ 0.483	≥ 0.483	≥ 0.440
Clad Thickness[5] (inch)	≥ 0.032	≥ 0.034	≥ 0.032	≥ 0.032	≥ 0.032	≥ 0.028

Table 14
BWR Fuel Assembly Design Characteristics[1] for Transportation in the TN-LC-1FA Basket
(Part 2 of 3)

Transnuclear ID	10x10-92/2	7x7-49/0Z	7x7-48/1Z	8x8-60/4Z	FANP 9x9	Siemens QFA
Initial Design or Reload Fuel Designation	GE12	ENC-IIIA	ENC-III[2]	ENC Va	FANP9 9x9[3]	9x9
	GE14			ENC Vb		
Maximum Number of Fuel Rods	92	49	48	60	81	72
Maximum Initial Uranium Content (kg)	192	198	198	192	192	192
Rod Pitch[5] (inch)	≤ 0.510	≤ 0.738	≤ 0.738	≤ 0.642	≤ 0.572	≤ 0.570
Pellet Diameter[5] (inch)	≤ 0.345	≤ 0.488	≤ 0.491	≤ 0.420	≤ 0.357	≤ 0.374
Clad Outer Diameter[5] (inch)	≥ 0.404	≥ 0.570	≥ 0.570	≥ 0.501	≥ 0.424	≥ 0.433
Clad Thickness[5] (inch)	≥ 0.026	≥ 0.035	≥ 0.035	≥ 0.036	≥ 0.030	≥ 0.026

5.(b)(1) Type and Form of Materials (continued)

Table 14
BWR Fuel Assembly Design Characteristics[1] for Transportation in the TN-LC-1FA Basket
(Part 3 of 3)

Transnuclear ID	10x10-91/1	ABB-8x8	ABB-10x10	LaCrosse
Initial Design or Reload Fuel Designation	ATRIUM 10	SVEA-64	SVEA-100[4]	Allis Chalmers - 10x10
	ATRIUM 10XM			Exxon/ANF 10x10
Maximum Number of Fuel Rods	91	64	100	100
Maximum Initial Uranium Content (kg)	192	192	192	125
Rod Pitch[5] (inch)	≤ 0.510	≤ 0.622	≤ 0.512	≤ 0.565
Pellet Diameter[5] (inch)	≤ 0.350	≤ 0.411	≤ 0.346	≤ 0.350
Clad Outer Diameter[5] (inch)	≥ 0.405	≥ 0.378	≥ 0.378	≥ 0.394
Clad Thickness[5] (inch)	≥ 0.023	≥ 0.024	≥ 0.022	≥ 0.020

Notes:

1. Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
2. Includes ENC-IIIE and ENC-IIIF.
3. Includes FANP 9x9-72, 9x9-79, 9x9-80, and 9x9-81.
4. Includes SVEA-92, SVEA-96, SVEA-96+, SVEA-96 OPTIMA, SVEA-96 OPTIMA2, SVEA-96+/L.
5. The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type).

(2) Maximum quantity of material per package

(i) For the contents described in Item 5(b)(1)(i): 26 intact or damaged either NRU or NRX Mk I fuel assemblies, with an approximate maximum payload of 331 lb.

(ii) For the contents described in Item 5(b)(1)(ii): 54 intact or damaged MTR fuel elements, with an approximate maximum payload of 1,620 lb.

(iii) For the contents described in Item 5(b)(1)(iii): 180 intact TRIGA fuel elements/assemblies with an approximate maximum payload of 2,380 lb.

5.(b)(2) Maximum quantity of material per package (continued)

(iv) For the contents described in Item 5(b)(1)(iv): one intact PWR fuel assembly, or one intact BWR fuel assembly, or up to 25 intact PWR (including MOX and EPR) or BWR fuel rods in a pin can. When transporting 9 or fewer fuel rods, the rods shall be placed in the center 3x3 region of the pin can. The approximate maximum payload is 1,650 lb per PWR assembly, 1,850 lb per BWR assembly with PRAs, 710 lb per PWR assembly, 790 lb per BWR assembly with channels, and 16 lb per fuel rod.

(3) The maximum decay heat for any payload is 3.0 kW.

5(c) Criticality Safety Index (CSI): 100

Table 15
Fuel Qualification Table for a PWR Fuel Assembly

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/ MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	2.25	2.25	2.20	2.10	2.05	2.05	2.05	2.00	2.00	2.00	2.00	2.00	2.00	1.95	1.95	1.95	1.95	1.95	1.95	1.95	1.95	1.95	1.95	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90	1.90
20	3.37	3.35	3.30	3.20	3.05	2.90	2.90	2.85	2.85	2.80	2.80	2.80	2.75	2.75	2.75	2.75	2.75	2.75	2.70	2.70	2.70	2.70	2.65	2.65	2.65	2.65	2.65	2.65	2.60	2.60	2.60	2.60	2.60	2.60	2.60	2.55
30			4.70	4.35	4.10	3.80	3.70	3.65	3.60	3.60	3.55	3.50	3.45	3.45	3.40	3.35	3.35	3.35	3.30	3.30	3.25	3.25	3.20	3.20	3.15	3.15	3.15	3.15	3.15	3.15	3.10	3.10	3.10	3.05	3.05	3.05
39						4.95	4.85	4.75	4.65	4.55	4.45	4.40	4.35	4.25	4.20	4.15	4.10	4.00	3.95	3.95	3.90	3.85	3.80	3.75	3.70	3.70	3.70	3.65	3.65	3.60	3.55	3.55	3.50	3.50	3.50	3.50
40												4.55	4.45	4.35	4.30	4.25	4.15	4.15	4.10	4.05	4.00	3.90	3.90	3.90	3.85	3.80	3.75	3.70	3.70	3.65	3.65	3.65	3.60	3.55	3.55	3.50
45												5.40	5.25	5.15	5.05	4.95	4.85	4.80	4.70	4.60	4.55	4.50	4.45	4.35	4.35	4.30	4.20	4.15	4.10	4.10	4.05	4.00	3.95	3.95	3.90	3.85
50												6.80	6.60	6.50	6.25	6.15	6.00	5.85	5.75	5.60	5.50	5.40	5.30	5.20	5.10	5.05	4.95	4.90	4.85	4.75	4.70	4.65	4.55	4.55	4.50	4.40
55												8.85	8.60	8.30	8.05	7.85	7.65	7.35	7.15	7.00	6.80	6.65	6.45	6.30	6.20	6.05	5.90	5.85	5.70	5.65	5.50	5.45	5.35	5.30	5.25	5.15
60												11.55	11.20	10.85	10.50	10.15	9.80	9.55	9.20	8.95	8.70	8.45	8.25	8.00	7.80	7.55	7.40	7.20	7.05	6.85	6.75	6.60	6.45	6.35	6.25	6.10
61												12.15	11.80	11.45	11.10	10.70	10.35	10.10	9.75	9.45	9.20	8.90	8.65	8.35	8.20	7.90	7.75	7.55	7.40	7.20	7.00	6.85	6.75	6.55	6.50	6.40
62												12.80	12.40	12.05	11.65	11.30	10.90	10.65	10.25	9.95	9.70	9.40	9.10	8.85	8.55	8.35	8.15	7.90	7.70	7.50	7.30	7.20	7.05	6.85	6.75	6.65
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

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Table 16
Fuel Qualification Table for a BWR Fuel Assembly

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/ MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.65	0.65	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60
20	0.95	0.95	0.90	0.85	0.80	0.80	0.80	0.80	0.80	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75
30			1.25	1.20	1.15	1.10	1.10	1.10	1.10	1.10	1.10	1.10	1.10	1.05	1.05	1.05	1.05	1.05	1.05	1.05	1.05	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	0.95	0.95	0.95	0.95
39						1.40	1.40	1.40	1.35	1.35	1.35	1.35	1.30	1.30	1.30	1.30	1.30	1.25	1.25	1.25	1.25	1.25	1.20	1.20	1.20	1.20	1.20	1.20	1.15	1.15	1.15	1.15	1.15	1.15	1.15	1.15
40												1.40	1.40	1.35	1.35	1.35	1.35	1.30	1.30	1.30	1.30	1.30	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.20	1.20	1.20	1.20	1.20	1.20	1.20
45												1.60	1.60	1.60	1.55	1.55	1.55	1.55	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.45	1.45	1.45	1.45	1.45	1.45	1.40	1.40	1.40
50												1.85	1.85	1.85	1.80	1.80	1.80	1.75	1.75	1.75	1.75	1.75	1.75	1.75	1.75	1.70	1.70	1.70	1.70	1.65	1.65	1.65	1.65	1.65	1.60	1.60
55												2.10	2.10	2.10	2.05	2.05	2.05	2.00	2.00	2.00	1.95	1.95	1.95	1.95	1.95	1.95	1.95	1.90	1.90	1.90	1.90	1.90	1.85	1.85	1.85	1.85
60												2.35	2.35	2.35	2.30	2.30	2.30	2.25	2.25	2.25	2.20	2.20	2.20	2.20	2.20	2.20	2.15	2.15	2.15	2.15	2.10	2.10	2.10	2.10	2.05	2.05
61												2.40	2.40	2.40	2.35	2.35	2.35	2.30	2.30	2.30	2.25	2.25	2.25	2.20	2.20	2.20	2.20	2.20	2.20	2.20	2.15	2.15	2.15	2.15	2.10	2.10
62												2.45	2.45	2.45	2.40	2.40	2.40	2.35	2.35	2.35	2.30	2.30	2.30	2.25	2.25	2.25	2.25	2.25	2.25	2.25	2.20	2.20	2.20	2.20	2.15	2.15
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

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Table 17
Fuel Qualification Table for 25 PWR/EPR Fuel Rods (UO_2)

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/ MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
20	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
30			0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
39						0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
40												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
45												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
50												0.30	0.30	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
55												0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30
60												0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30
61												0.40	0.40	0.40	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.30	0.30
62												0.40	0.40	0.40	0.40	0.40	0.40	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35
65																							0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.35	0.35	0.35	0.35	0.35	0.35
70																							0.50	0.50	0.50	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45
75																							0.65	0.65	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.60	0.50	0.50	0.50	0.50
80																							0.85	0.85	0.75	0.75	0.75	0.75	0:75	0.70	0.70	0.70	0.70	0.70	0.70	0.70
85																							1.05	1.00	1.00	1.00	0.90	0.90	0.90	0.90	0.85	0.85	0.85	0.85	0.85	0.85
90																							1.25	1.25	1.25	1.15	1.15	1.15	1.10	1.10	1.10	1.00	1.00	1.00	1.00	0.95
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

Table 18
Fuel Qualification Table for 9 PWR/EPR Fuel Rods (UO_2)

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/ MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
20	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
30			0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
39						0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
40												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
45												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
50												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
55												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
60												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
61												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
62												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
65																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
70																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
75																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
80																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
85																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
90																							0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

Table 19
Fuel Qualification Table for 25 BWR Fuel Rods (UO_2)

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
20	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
30			0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30
39						0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35
40												0.40	0.40	0.40	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35
45												0.45	0.45	0.45	0.45	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40
50												0.60	0.60	0.60	0.60	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.50	0.50	0.50	0.50	0.50	0.50	0.50
55												0.75	0.75	0.75	0.75	0.75	0.70	0.70	0.70	0.70	0.70	0.70	0.70	0.70	0.70	0.65	0.65	0.65	0.65	0.65	0.65	0.65	0.65	0.65	0.65	0.65
60												1.00	1.00	1.00	1.00	0.90	0.90	0.90	0.90	0.90	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75
61												1.05	1.05	1.00	1.00	1.00	1.00	1.00	0.90	0.90	0.90	0.90	0.90	0.90	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.75	0.75	0.75	0.75
62												1.10	1.05	1.05	1.05	1.05	1.00	1.00	1.00	1.00	1.00	0.90	0.90	0.90	0.90	0.90	0.90	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.85	0.85
65																							1.05	1.05	1.00	1.00	1.00	1.00	1.00	1.00	1.00	0.90	0.90	0.90	0.90	0.90
70																							1.20	1.20	1.20	1.15	1.15	1.15	1.15	1.15	1.15	1.10	1.10	1.10	1.10	1.10
75																							1.45	1.45	1.45	1.40	1.40	1.40	1.30	1.30	1.30	1.30	1.25	1.25	1.25	1.25
80																							1.70	1.70	1.65	1.65	1.60	1.60	1.60	1.50	1.50	1.50	1.45	1.45	1.45	1.45
85																							2.15	2.05	2.00	2.00	1.95	1.85	1.85	1.80	1.80	1.70	1.70	1.65	1.65	1.65
90																							2.60	2.55	2.50	2.40	2.35	2.30	2.20	2.15	2.15	2.10	2.00	2.00	1.95	1.95
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

Table 20
Fuel Qualification Table for 9 BWR Fuel Rods (UO_2)

(Minimum required years of cooling time after reactor core discharge)

Burnup, GWd/MTU	Enrichment (wt. % ^{235}U)																																		
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
20	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
30			0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
39						0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
40												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
45												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
50												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
55												0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
60												0.30	0.30	0.30	0.30	0.30	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25	0.25
61												0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.25	0.25	0.25
62												0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30
65																							0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35	0.35
70																							0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40
75																							0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40
80																							0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45	0.45
85																							0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50
90																							0.60	0.60	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55	0.55
	0.7	0.8	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

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Table 21

Fuel Qualification Table for MOX PWR/BWR 25 Rods and MOX PWR/BWR/EPR 9 Rods

Burnup, GWd/MTHM	9 Rods		25 Rods	
	0.5 wt.% of ^{235}U	0.7 wt.% of ^{235}U	0.5 wt.% of ^{235}U	0.7 wt.% of ^{235}U
10	0.25	0.25	0.25	0.25
20	0.25	0.25	0.30	0.30
30	0.25	0.25	0.50	0.50
40	0.25	0.25	0.95	0.95
45	0.25	0.25	1.25	1.25
50	0.35	0.35	1.70	1.70
55	0.40	0.40	2.20	2.10
60	0.45	0.45	2.80	2.70
62	0.55	0.55	3.75	3.65

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Notes:

1. Explanatory notes and limitation regarding the use of this table are provided on the following page.

Notes:

General

1. Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
2. For values not explicitly listed in the tables, round burnups **up** to the first value shown, round enrichments **down**, and select the cooling time listed.
3. UO_2 Fuel with an initial enrichment less than 0.7 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% ^{235}U is unacceptable for transportation.
4. Shaded areas in these Tables indicate fuel is not analyzed for loading.

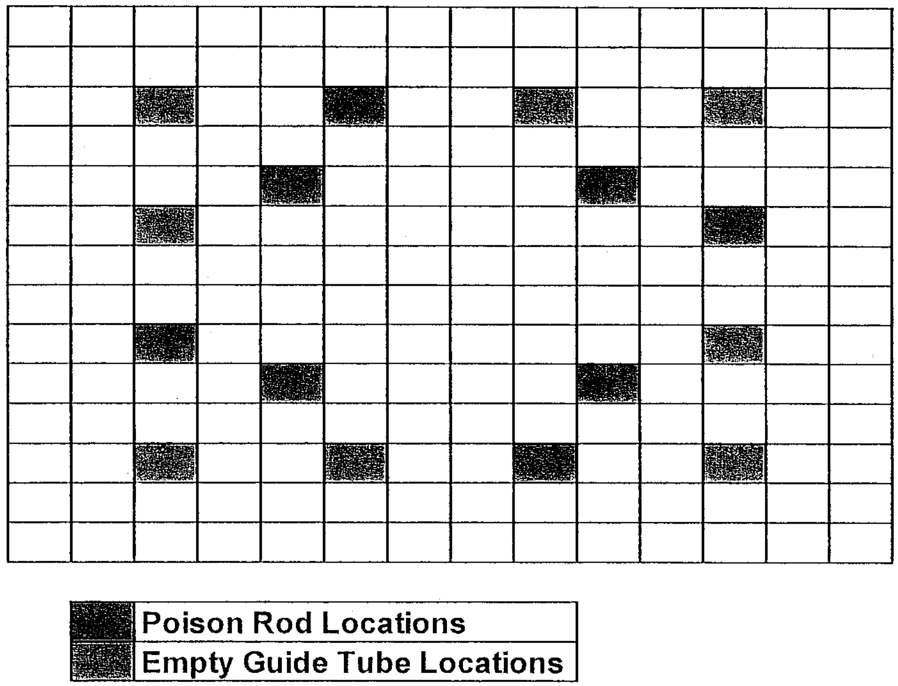
For Fuel Assemblies

1. Burnup = Assembly Average burnup.
2. Enrichment = Assembly Average Enrichment.
3. Fuel assembly with a burnup greater than 62 GWd/MTU is unacceptable for transportation.

588 For Fuel Rods

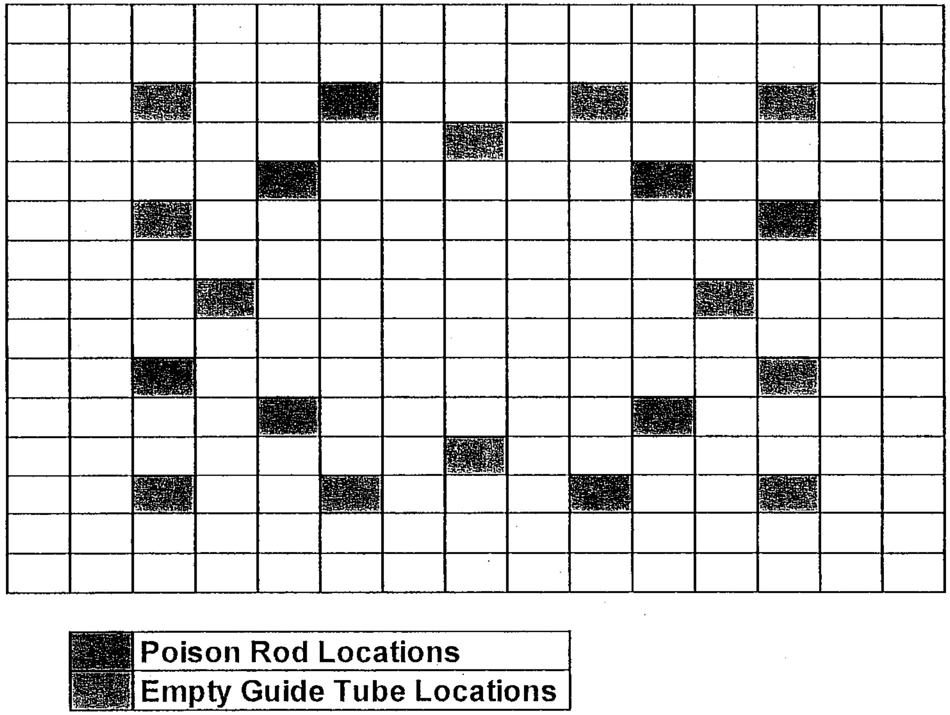
4. Burnup = Maximum burnup.
5. Enrichment = Rod Average Enrichment.
6. When transporting 25 or less fuel rods, the rods shall be placed in a specially designed 25 pin can.
7. When transporting 9 or less fuel rods, the rods shall be placed in the 3x3 region of the 25 pin can.
8. Fuel rods with a burnup greater than 90 GWd/MTU are unacceptable for transportation.

Example: Per Table 15, a PWR assembly with an initial enrichment of 4.85 wt.% ^{235}U and a burnup of 41.5 GWd/MTU is acceptable for transport after a 3.95-year cooling time as defined by 4.8 wt.% ^{235}U (rounding down) and 45 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).



Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

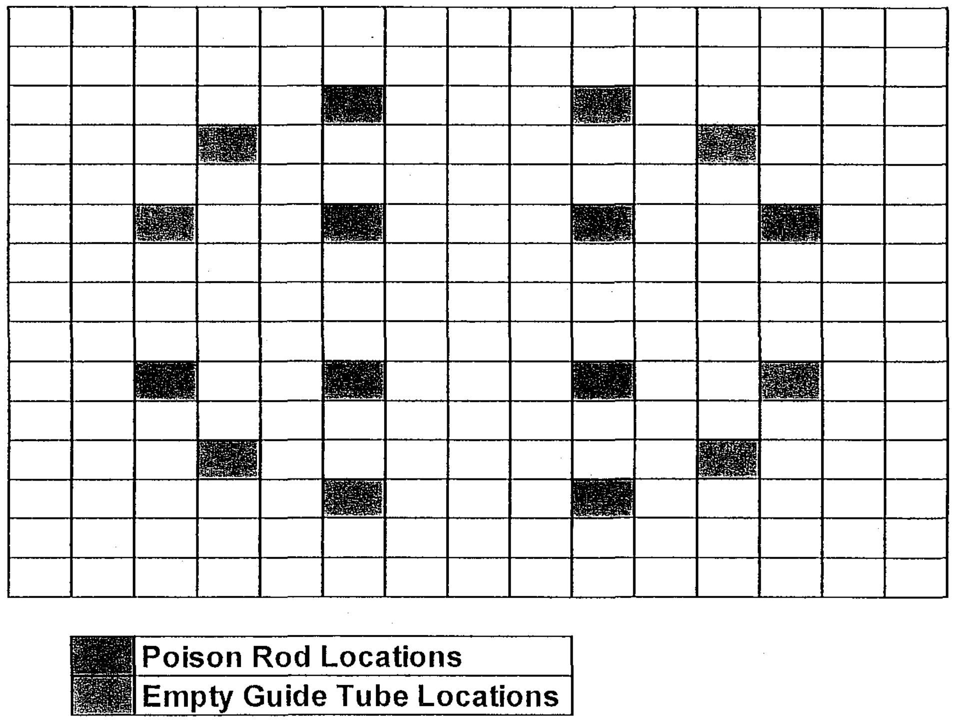
Figure 1
PRA Insertion Locations for WE 14x14 Class Assemblies



Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

Figure 2
PRA Insertion Locations for WE 15x15 Class Assemblies

590



Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

Figure 3
PRA Insertion Locations for BW 15x15 Class Assemblies

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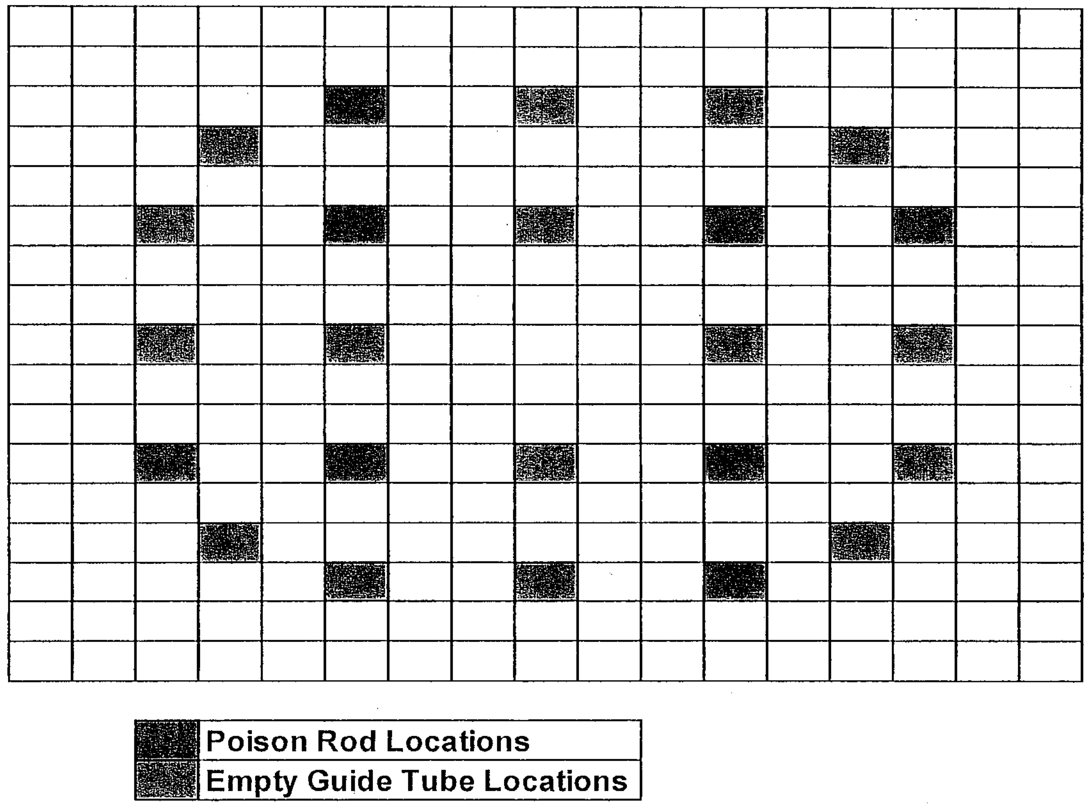

Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

Figure 4
PRA Insertion Locations for BW 17x17 and WE 17x17 Class Assemblies

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter No. 7 of the application, and

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter No. 8 of the application.

7. Transport by air of fissile material is not authorized.

8. Prior to the first shipment, the package shall be tested for the entire containment boundary, e.g., all base metal, all joining containment welds, vent port plug seal, drain port plug seal, lid seal, and bottom plug seal, in accordance with ANSI N14.5, by helium leakage rate testing to meet the leaktight criteria of $1.0x10^{-7}$ ref-cm^3/sec for fabrication leakage tests.

9. Poison Rod Assemblies, required for shipment of PWR assemblies, shall be installed such that the active fuel length is covered by the absorber, and measures shall be taken against their inadvertent removal from the fuel assembly.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Expiration date: December 31, 2017.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE OF PAGES
9358	0	71-9358	USA/9358/B(U)F-96	31 OF 31

REFERENCES

Transnuclear, Inc., TN-LC Transportation Package Safety Analysis Report, Revision No. 6, dated November 2012.

Supplements dated November 27 and December 18, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele M. Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: December 31, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9786	7	71-9786	USA/9786/B(U)	1 OF	4

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
S3G Core Basket Disposal Container
Safety Analysis Report for Packaging
dated June 1980, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: S3G Core Basket Disposal Container Assembly

(2) Description

The package consists of either one irradiated S3G or S7G core basket packaged in an inner, lead-filled container (S5W Core Basket Removal Container (CBRC)) which is placed inside an outer container (S3G Core Basket Disposal Container (CBDC)). The package weighs approximately 172,000 pounds.

The S3G CBDC is a 4-inch thick steel cylinder, 89 inches in outside diameter, 131 inches long, with an 8-inch thick top end plate and a 5-inch thick bottom end plate. Both end plates are welded to the cylinder with full penetration welds.

The S5W CBRC, which will be disposed of along with the outer S3G CBDC and inner core basket, is basically a cylindrical shaped container comprised of lead shielding sandwiched between two 304 stainless steel shells. The 1-inch thick inner shell is 60 inches O.D. and 107.5 inches long. The outer shell is made up of two geometries, a 72.5-inch O.D., 0.5-inch thick cylindrical shell that measures 66 inches long and joins a truncated conical shell which has a 64-inch O.D. at the small end. The two shells are joined by a full thickness penetration weld and a weld backup strap on the inside shell surface. Full penetration welds are also made on both ends of the shells to the top canning and shield ring.

The S5W CBRC will contain either an S3G or S7G core basket. The irradiated S3G core basket is an Inconel 600 cylindrical shell. Three, 3-inch thick 304 stainless steel plates are positioned in the core basket prior to removal to provide overhead radiation shielding. The lower plate is 46.2 inches in diameter. The upper plates have the same diameter but contain six extensions that fit inside recessed cutouts within the core basket. The total core basket weight is approximately 9,650 pounds.

5.(a) (2) Description (continued)

The S7G core basket is an Inconel 600 cylindrical shell. A 304 stainless steel laminated plate (8-inches thick) with lifting attachments is attached to the top of the S7G core basket to provide radiation shielding during handling. The core basket weight is approximately 8,873 pounds.

The package may alternatively consist of S8G irradiated components positioned within an irradiated components discharge rack (ICDR) which is placed in an S3G CBDC. The ICDR is a steel rack approximately 128 inches high and 80 inches in diameter, and is designed to fit inside the S3G CBDC. The ICDR consists of a center cylinder assembly surrounded by 23 storage tubes, a top plate and a cylinder support base. The center cylinder is HY-80 steel, has a 36-inch outer diameter and a 4.5-inch wall thickness, and is 117 inches high. There are 9 storage tubes positioned inside the center cylinder. The total weight of the irradiated components, the ICDR, and the S3G CBDC is approximately 125,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Bettis Drawing No. 1527E40 for the S3G Core Basket Assembly and KAPL Drawing No. 232B4874 for the S7G Core Basket Assembly and KAPL Drawing No. 978E644 for the S8G Irradiated Components.

(b) Contents

(1) Type and form of material

(i) An irradiated core basket either the S3G or S7G and S5W CBRC. The shipment may include surface contamination in the form of activated corrosion products and for the S3G core basket approximately 8 gallons of residual water.

(ii) S8G irradiated components within an ICDR. The shipment may include surface contamination in the form of activated corrosion products.

(2) Quantity of material per package

(i) Item 5(b)(1)(i) above:

One irradiated core basket and S5W CBRC as described in 5(b)(1). Surface contamination not to exceed 20.6 curies for the S3G core basket, or 1.2 curies for the S7G core basket. The activation level of the irradiated S3G core basket is not to exceed 131,000 curies; and the activation level of the irradiated S7G core basket is not to exceed 140,000 curies.

(ii) Item 5(b)(1)(ii) above:

Irradiated components, including 141 instrument lines, 18 lower control drive mechanism assemblies, 4 filled sleeves, and 1 instrumentation stalk. Surface contamination not to exceed 65.5 curies. Activation level of the irradiated components not to exceed 2,440 curies.

6. Shipment of an irradiated S3G core basket must be made no earlier than 75 days after reactor shutdown.

7. Shipment of an irradiated S7G core basket must be made no earlier than 180 days after reactor shutdown.

8. Shipment of S8G irradiated components must be made no earlier than 100 days after reactor shutdown.

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each packaging must meet the following Acceptance Tests and Maintenance Program:

<u>S3G Core Basket</u>

Section 8.0 of application dated June 1980

<u>S7G Core Basket</u>

Section 8.0 of application dated May 1987

<u>S8G Irradiated Components</u>

Section 8.0 of application dated September 1991

(b) The package shall be prepared for shipment and operated in accordance with the following operating procedures:

S3G Core Basket

Section 7.0 of application dated June 1980

S7G Core Basket

Section 7.0 of application dated May 1987

S8G Irradiated Components

Section 7.0 of application dated September 1991

10. Air transport of fissile material is not authorized.

11. Revision No. 6 of this certificate may be used until April 30, 2012.

12. Expiration date: August 31, 2016.

REFERENCES

S3G Core Basket Disposal Container Safety Analysis Report for Packaging, WAPD-REO(C)-122, dated June 1980, as revised (Revision 2, dated May 5, 1986).

S7G Core Basket in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, dated May 1987.

S8G Irradiated Components in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, Revision 2, dated September 1991.

DOE memorandums G#7627 dated November 16, 1983; G#C86-3736 dated May 24, 1986; G#C86-3750 dated July 15, 1986; G#87-5663 dated July 7, 1987; G#91-10937 dated July 31, 1991; G#C91-11007 dated September 18, 1991; G#96-03335 dated February 16, 1996; G#01-03414 dated January 31, 2001; and G#11-00740 dated February 8, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M W Waters

Michael Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 15, 2011

NRC FORM 618 U.S. NUCLEAR REGULATORY COMMISSION
(8-2000)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9788	17	71-9788	USA/9788/B(U)-96	1	OF	6

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Deactivated S5W Reactor Compartment Safety Analysis Report for packaging, dated July 1981, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: S5W Reactor Compartment and SSN 688 Class Reactor Compartment.

(2) Description

The package consists of a deactivated and defueled S5W or SSN 688 Class reactor compartment which has been separated from the remainder of the submarine hull and prepared for shipment by sealing all openings and attaching handling fixtures. For each package model, the reactor compartment itself is between two containment bulkheads which are added to the package before shipping. The ship's hull and the containment bulkheads define the package containment boundaries. The containment bulkheads are either installed at the ends of the package or recessed. The strength of all package boundary closures is at least equivalent to the strength of the bulkheads. The deactivated reactor plant remains in place within the reactor compartment during shipment. The plant is defueled and drained except for small inaccessible pockets of liquid, primarily water. Potentially radioactively contaminated components and piping from other locations in the ship may be placed within the package and secured.

5. (a) Packaging (Continued)

(2) Description (Continued)

The S5W Reactor Compartment package is between 35 and 45 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The forward containment bulkhead may include existing ship structures which have been sealed to form a watertight bulkhead. The hull is constructed of HY-80 steel. The maximum weight of the S5W package is 2,160,000 pounds for the 598 and 585 classes and is 2,262,400 pounds for all other classes.

The SSN 688 Class Reactor Compartment package is approximately 46 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The hull is constructed of HY-80 steel. The maximum weight of the package is 3,360,000 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application, as supplemented (see References, below).

(b) Contents

Activated structural components associated with the S5W and SSN 688 Class reactor vessel complex, plant piping, ion exchanger resin, purification filter media (SSN 688 Class only), residual liquid and other miscellaneous components and materials contaminated with radioactive corrosion products (crud).

6. Residual liquids contained within plant systems must be removed prior to transport to the maximum extent practical, in accordance with established procedures, methods, and controls, as described in submittal dated April 5, 1996, or in the Safety Analysis Report for the individual submarine class reactor compartment packages. Not more than 660 gallons of residual liquids remain in the S5W Reactor Compartment package and not more than 1,200 gallons of residual liquids remain in the SSN 688 Class Reactor Compartment package.

7. For packages with recessed containment bulkheads, the aft containment bulkheads and stiffeners, horizontal divider plate, and any structure between the pressure hull and the outer non-pressure hull must be recessed at least 7 inches from the aft end of the S5W package. The forward containment bulkhead and stiffeners, existing tank stiffeners, deck structure, and horizontal girders must be recessed at least 15 inches from the forward end of the S5W package. For SSN 688 Class packages with recessed containment bulkheads, both the aft

and forward containment bulkheads, stiffeners and horizontal girders must be recessed at least 15 inches from the end of the package.

8. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the normal daily minimum temperature expected during the shipment of the package as determined on the basis of weather forecasts.

9. Ion exchanger resin with up to 3.1 curies (1.1×10^{11} becquerels) of Co-60 may be shipped in the S5W package. Shipment of the S5W packages shall not occur before 180 days after final reactor shutdown.

10. For SSN 688 Class packages, the Co-60 activity of the ion exchanger resin (A_{IX}) and activity of the purification filter media (A_{PF}) must meet the following conditions (Reference Table 5-3 of the application):

    (a) when only the purification filter is solidified,
    $A_{PF} * 27.2 + A_{IX} * 283.8 \leq 995$ mr/hr (Curies)
    $A_{PF} * 7.35 \times 10^{-10} + A_{IX} * 7.67 \times 10^{-9} \leq 995$ mr/hr (becquerels)

    (b) when only the ion exchanger is solidified,
    $A_{IX} * 53.9 + A_{PF} * 299.6 \leq 995$ mr/hr (Curies)
    $A_{IX} * 1.46 \times 10^{-9} + A_{PF} * 8.10 \times 10^{-9} \leq 995$ mr/hr (becquerels)

    (c) when both the purification filter and ion exchanger are solidified,
    $A_{IX} * 53.9 + A_{PF} * 27.2 \leq 995$ mr/hr (Curies)
    $A_{PF} * 1.46 \times 10^{-9} + A_{IX} * 7.35 \times 10^{-10} \leq 995$ mr/hr (becquerels)

    (d) when neither the ion exchanger or purification filter are solidified,
    $A_{PF} \leq 3.3$ curies of Co-60 (1.22×10^{11} becquerels),
    $A_{IX} \leq 3.5$ curies of Co-60 (1.30×10^{11} becquerels), and
    $A_{IX} + A_{PF} \leq 4.5$ curies of Co-60 (1.67×10^{11} becquerels)

    If the activity exceeds any of the above values, supplemental shielding must be added. See Condition 12 and 13 for the supplemental shielding requirements.

    These activity limits are based on limiting the total radiation levels to 995 mr/hr under hypothetical accident conditions. If contaminated filters are shipped with the package, the sum of the radiation levels from each filter at 1 meter, shielded by the hull, must be subtracted from this 995 mr/hr, and the A_{PF} and A_{IX} activity limits must be reduced accordingly for SSN 688 Class packages.

11. Shipment of the SSN 688 Class packages shall not occur before 365 days after final reactor shutdown.

12. Prior to shipment, radiation surveys of the exterior of the unshielded package must be taken. Additional shielding must be provided on the exterior of the package by steel plates securely welded to the package surface so as to remain in place under the hypothetical accident conditions in 10 CFR Part 71 if either of the following conditions exists:

(a) Radiation levels on the exterior of the package obtained via surveys exceed 200 mr/hr on-contact or 10 mr/hr at 2 meters.

(b) When the purification filter and ion exchanger are not solidified, and $A_{PF} > 3.3$ curies of Co-60, $A_{IX} > 3.5$ curies of Co-60, or $A_{IX} + A_{PF} > 4.5$ curies of Co-60. (SSN 688 Class only)

Radiation surveys must be re-performed after adding supplemental shielding. Final radiation levels must not exceed 200 mr/hr on-contact and 10 mr/hr at 2 meters.

13. When condition 12(a) exists, 0.5-inch thick steel plates must extend from one inch above the bottom of the shielded tunnel on the starboard side to 87° beyond the keel on the port side and extend 110 inches forward and aft of the centerline of the Pressure Vessel. Additional 1.25" steel plate must extend 22° beyond the keel on the port and starboard side and 77 inches forward and aft of the centerline of the Pressure Vessel.

When condition 12(b) exists, 1-inch thick steel plates must either:

(a) Extend from one inch above the bottom of the shielded tunnel on the starboard side to 64° beyond the keel on the port side and extend from frame 73 to frame 76. In this case, A_{PF} cannot exceed 7.8 curies of Co-60 (2.89×10^{11} becquerels), A_{IX} cannot exceed 8.2 curies of Co-60 (3.03×10^{11} becquerels), and the total of A_{PF} and A_{IX} cannot exceed 8.5 curies of Co-60 for D1G-2 cores or 9.0 curies of Co-60 for D2W cores (3.14×10^{11} and 3.33×10^{11} becquerels, respectively); or

(b) Extend from one inch above the bottom of the shielded tunnel on the starboard side to 37° beyond the keel on the port side and extend from frame 73 to frame 76. In this case, A_{PF} cannot exceed 7.8 curies of Co-60 (2.89×10^{11} becquerels), A_{IX} cannot exceed 3.5 curies of Co-60 (1.30×10^{11} becquerels), and the total of A_{PF} and A_{IX} cannot exceed 8.5 curies of Co-60 for D1G-2 cores or 9.0 curies of Co-60 for D2W cores (3.14×10^{11} and 3.33×10^{11} becquerels, respectively).

When both conditions 12(a) and 12(b) exist, 1-inch thick steel plates must extend from one inch above the bottom of the shielded tunnel on the starboard side to 87° beyond the keel on the port side and extend 110 inches forward and aft of the centerline of the Pressure Vessel. Additional 0.75 inch steel plates must extend 22° beyond the keel on the port and starboard side and extend 77 inches forward and aft of the centerline of the Pressure Vessel. In this case, A_{PF} cannot exceed 7.8 curies of Co-60 (2.89×10^{11} becquerels), A_{IX} cannot exceed 8.2 curies of Co-60 (3.03×10^{11} becquerels), and the total of A_{PF} and A_{IX} cannot exceed 8.5 curies of Co-60 for D1G-2 cores or 9.0 curies of Co-60 for D2W cores (3.14×10^{11} and 3.33×10^{11} becquerels, respectively).

The activity limits for the ion exchanger and purification filter are based on limiting the total radiation levels to 995 mr/hr under hypothetical accident conditions. If contaminated filters are shipped with the package, the sum of the radiation levels from each filter at 1 meter, shielded by the hull, must be subtracted from this 995 mr/hr, and the A_{PF} and A_{IX} activity limits must be reduced accordingly for SSN 688 Class packages.

14. Ensure the radiation surveys are completed as described in Enclosures 3, 4, and 5 of the supplement dated December 2, 2011.

15. In addition to the requirements of Subpart G of 10 CFR Part 71:

    (a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application.

    (b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application.

16. The hydrogen concentration within the package must be less than 5 percent by volume during the shipment period, as demonstrated by test or analysis.

17. Expiration date: September 30, 2018.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9788	17	71-9788	USA/9788/B(U)-96	6	OF	6

REFERENCES

Deactivated S5W Reactor Compartment Safety Analysis Report for Packaging, WAPD-REO(C)-250, dated July 1981.

Supplements: Naval Reactors Memoranda Nos. Z#C90-14416 dated March 29, 1990, and supplement dated July 6, 1990; Z#C90-14456 dated August 30, 1990; Z#C92-14438 dated August 3, 1992; Z#C93-00069 dated October 14, 1993; Z#C95-00113 dated March 16, 1995; Z#96-14430 dated April 5, 1996; Z#96-14434 dated April 10, 1996; Z#C95-00191 dated December 14, 1995; Z#96-14457 dated June 20, 1996; Z#C96-14520 dated November 22, 1996; Z#C96-14549 dated December 19, 1996; Z#C97-14698 dated October 31, 1997; Z#C98-00021 dated February 27, 1998; Z#C02-03057 dated March 15, 2002; Z#C07-02023 dated September 19, 2007; Z#08-03540 dated September 11, 2008; Z#C07-04862 dated January 3, 2008; Z#C09-02922 dated July 1, 2009; and NR (Angerhofer) email dated July 30, 2008; S#C11-01953 dated April 29, 2011; S#C11-02849 dated June 30, 2011; S#C11-04480 dated October 27, 2011; S#C11-05155 dated December 2, 2011; and Z#13-02768 dated June 21, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 9/27/2013

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9791	8	71-9791	USA/9791/B(U)-85	1	OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging dated January 1982, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PWR-2 Lower Core Barrel Shipping and Disposal Container

(2) Description

The PWR-2 Lower Core Barrel Shipping and Disposal Container package consists of an inner burial container and a reusable outer container. The inner container is loaded with a D1G prototype pressure vessel assembly. The package weighs approximately 400,000 pounds.

The outer container is a 4-inch thick steel cylinder, 127 inches in outside diameter, 212 inches long, with two 6-inch thick end plates. The bottom end plate is welded to the cylinder with a full penetration weld and the top end plate is bolted with 107, 2-inch diameter fasteners.

The package is equipped with two 2.5-inch thick by 10-inch long circumferential impact limiter rings on the side, two concentric impact limiter rings on the ends, and aluminum honeycomb crush blocks in the top and bottom spaces between the inner and outer containers.

The container is supported horizontally on the railroad car by eight gussets attached to two horizontal plates. Each plate is bolted to the top flange of an I-beam. The bottom flange of the I-beam is bolted to a 300-ton railroad car.

5. (a) Packaging (continued)

The inner disposal container (liner) is of the following design:

The D1G prototype pressure vessel assembly has an inner burial container that consists of two cylinders constructed of HY-80 steel connected by a transition ring that is welded to the two cylinders. The maximum outer diameter of the cylinder is approximately 118 inches at the upper flange. The overall length of the inner container is 184.5 inches. The container wall is 3.12 inches in the upper cylinder and 4 inches in the bottom cylinder. The bottom plate varies in thickness from 6 to 2.4 inches and is attached to the container by 12, 4.5-inch thick gussets. The cover plate is approximately 10 inch thick and is attached to the container by a 3.25-inch thick closure weld. The container is axially positioned within the outer container by aluminum honeycomb energy absorbers.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Drawing Nos. 1575E12, 1574E96, and KAPL, Inc., Drawing Nos. 108E6847 and 108E6846.

(b) Contents

(1) Type and form of material

An irradiated D1G prototype pressure vessel assembly, including pressure vessel, core barrel, thermal shields, and two surveillance train assemblies. In addition, the contents may include surface contamination in the form of activated corrosion products and 119 gallons of residual water.

(2) Quantity of material in package

One irradiated D1G prototype pressure vessel assembly. Surface contamination not to exceed 4.61 curies. Displaced material from cutting operations not to exceed 10.6 curies. The irradiated components not to exceed 60,000 curies.

6. The package shall be operated in accordance with the procedures described in Chapter 7 of the application and in accordance with Naval Reactors letter G#C98-10723 dated February 13, 1998. The package shall be tested and maintained in accordance with the procedures in Chapter 8 of the application and in accordance with Naval Reactors letter G#C98-10723 dated February 13, 1998.

7. Revision No. 7 of this certificate may be used until March 31, 2013.

8. Fabrication of packages must have been completed by December 31, 2006, in accordance with 10 CFR 71.19(c).

9. Expiration date: July 31, 2017.

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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REFERENCES

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging, WAPD-LP(CES)CS-670 dated January 1982.

Supplements: Naval Reactors letters G#7241 dated December 2, 1982; G#84-452 dated March 28, 1984; G#C92-03331 dated January 29, 1992; G#92-03546 dated June 5, 1992; G#92-03589 dated July 2, 1992; G#97-053513 dated June 11, 1997; G#C97-03596 dated August 28, 1997; G#C98-10723 dated February 13, 1998; G#98-10801 dated May 5, 1998; G#02-0688 dated January 16, 2002; G#07-00297 dated January 18, 2007; and G#12-00635 dated January 30, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine Lipa

Christine Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: March 7, 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9792	9	71-9792	USA/9792/B(U)	1	OF	3

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Department of Energy application dated April 22, 1991, as supplemented.

4. CONDITIONS
This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: Model 1 D1G Core Basket-Thermal Shield Shipping and Storage Container

(2) Description

The Model 1 D1G Core Basket-Thermal Shield (CB-TS) Shipping and Storage Container is a right circular cylinder approximately 115 inches in diameter and either 209 inches long (D1G design including impact limiter assembly) or 216 inches long (D2W design including impact limiter assembly). Access for loading is provided by a removable closure head. The container, consisting of the cylindrical side walls and the bottom end, has a three layer construction with a steel inner vessel approximately eight inches thick covered with approximately nine inches of reinforced concrete which is encased by a 3/8-inch thick outer shell. The CB-TS is secured in place inside the container with an 8-inch thick steel preload ring which is bolted to the inner vessel with 72 high strength bolts.

Closure of the containment vessel is provided by the 6-inch thick steel closure head which is fastened to the inner vessel with 72 high strength bolts. A steel closure ring is welded over the bolts and provides containment. A carbon steel inner impact limiter is welded to the top end of the closure ring. A wood outer impact limiter is bolted to the top plate of the container outer shell.

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5. (a) (2) Description (continued)

For land transport, the shipping container is transported with its axis horizontal and is supported by a shipping skid. For sea transport, the shipping container is transported with its axis vertical and is supported by a shipping frame assembly. The loaded container weighs up to 185 tons.

(3) Drawings

Packagings for which fabrication was begun before March, 1991, are constructed in accordance with the General Electric Company Drawings contained in Appendix 2.10.4 of the application, and packagings for which fabrication was begun after March, 1991, are constructed in accordance with the KAPL Drawings for the redesign configuration in Appendix 2.10.4 of the application.

(b) Contents

One irradiated D1G core basket-thermal shield assembly, and not more than one core's worth of irradiated D1G support assemblies, D1G lower control rod drive mechanisms, and D1G upper support assemblies; surface contamination in the form of activated corrosion products; and not more than 3.5 gallons of residual water.

6. (a) Preloading of the preload plate and the closure head and sealing the container must be done with a temperature at or above +40 °F.

(b) Shipment shall be made only when the average daily temperature is expected to be above +10 °F.

(c) The D1G CB-TS Shipment shall be made no earlier than 150 days after shutdown of the reactor.

7. The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, and each packaging shall be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8.0 of the application.

8. For sea transport, the supplemental operating procedures and acceptance tests in Sections 11.0 and 12.0 of the submittal dated April 5, 2002, shall be used.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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9. Expiration Date: January 31, 2018.

REFERENCES

Department of Energy, Division of Naval Reactors, application dated April 22, 1991.

Supplements dated: Naval Reactors Letters G#92-03668, dated August 27, 1992; G#C95-10762, dated April 10, 1995; G#C96-03576, dated November 1, 1996; G#C02-0751, dated April 5, 2002; G#07-01492, dated April 17, 2007; and G#12-02134, dated May 4, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 18 2012

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U. S. Department of Energy
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

"Core Independent M-140 Safety Analysis Report for Packaging" transmitted February 27, 1991, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a) Packaging

(1) Model No.: M-140

(2) Description

The M-140 is a stainless steel cask for transporting spent fuel. The cask is a right circular cylinder and is transported in the upright position. The package's approximate dimensions and weights are as follows:

Cavity diameter	70 inches
Cavity height	46 inches
Body outer diameter	98 inches
Body steel wall thickness	14 inches
Package overall outer diameter	126 inches
Package overall height	194 inches
Packaging weight, including standard internals	315,000 pounds
Maximum package weight, including contents	375,000 pounds

The cask body is made from 304 stainless steel forgings. The cask walls are 14-inches thick and the bottom plate is 12-inches thick. The cask body flange provides a seating surface for

5.(a)(2) Description (continued)

the closure head and its protective dome. The flange contains 36 wedge assemblies located radially around the inside diameter. Retention of the closure head is achieved by engaging the wedges in a tapered groove in the circumferential edge of the closure head. The cask body has 180 external cooling fins welded to the exterior wall. A support ring is welded to the external cooling fins at a point above the center of gravity. The support ring is bolted to a specially designed well-type railcar. The cask bottom is equipped with an energy absorber which is composed of five concentric stainless steel rings varying in thickness and height.

The closure head is made from forged 304 stainless steel and is approximately 13-inches thick and 81.7 inches in diameter. The closure head is equipped with an access port, which is approximately 24 inches in diameter, and is offset from the center of the closure head. The access port plug is a stepped design with a maximum diameter of approximately 31 inches and is attached to the closure head by 24 bolts. The closure head and access port are sealed with double ethylene propylene O-ring seals. Seal test ports are provided for the closure head and access port seals. A stainless steel protective dome is positioned over the closure head and is secured to the cask body flange by 12, 1.38-inch diameter, 38.5-inch long studs installed in a vertical direction and 6, 2.5-inch diameter, 9-inch long shear bolts installed in the radial direction.

The containment system is composed of the cask body, the closure head, and the closure head access port plug. There are seven penetrations in the standard containment system: a closure head, a drain port, a vent port, and an access port in the closure head, a thermocouple penetration, a water inlet penetration, and a water outlet penetration in the cask body. Each penetration is sealed with a plug and a double ethylene propylene O-ring seal and is equipped with a leak test port. For some shipping configurations, two additional penetrations may be present in the closure head: a removable fuel assembly (RFA) access port and another vent penetration.

The spent fuel modules are positioned in an internals assembly. The internals assembly is composed of stacked internal spacer plates which have openings for the spent fuel modules. The internals assembly has a top plate or top plate subassembly which is preloaded by springs against a retaining ring fitted in a groove in the cask cavity wall. The internals assembly may be a standard, Type 1, Type 2, or Type 3 internals assembly.

(3) Drawings

The packaging is constructed and assembled in accordance with the Westinghouse Electric Corporation Drawings in Appendix 1.3.2 of the application. Internals assemblies and fuel modules are constructed and assembled in accordance with drawings in Chapter 1 of the applicable Safety Analysis Reports for Packaging.

5.(b) Contents

(1) Type and form of material

Spent fuel, limited to the following types, including associated activated corrosion products:

(i) S3G-3 spent fuel.

(ii) S8G spent fuel.

(iii) D2W spent fuel.

(iv) A1G spent fuel.

(v) S6W spent fuel.

(vi) S9G spent fuel.

(2) Maximum quantity of material per package

Total package weight, including spent fuel and internals assembly, not to exceed 375,000 pounds; and

(i) For contents described in 5(b)(1)(i):

S3G-3 spent fuel modules, not to exceed 62,300 Btu/hr decay heat per package.

(ii) For contents described in 5(b)(1)(ii):

S8G spent fuel, not to exceed 51,609 Btu/hr decay heat per package (prototype spent fuel modules), or 45,713 Btu/hr decay heat per package (shipboard modules).

5.(b)(2) Maximum quantity of material per package (continued)

(iii) For contents described in 5(b)(1)(iii):

For Type 3 core, the total core decay heat level shall not exceed 68,390 BTU/hr at the time of container draining. The total core decay heat level shall not exceed 66,550 BTU/hr when the container is shipped. For the Type 5 core, the total core decay heat level shall not exceed 62,210 BTU/hr at the time of container draining.

(iv) For contents described in 5(b)(1)(iv):

A1G spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging or an administrative hold time of 50 days, whichever hold time is greater.

(v) For contents described in 5(b)(1)(v):

S6W spent fuel modules, not to exceed 46,011 Btu/hr decay heat per package for a shipboard core or 47,160 Btu/hr for a prototype core at the time of container draining.

(vi) For contents described in 5(b)(1)(vi):

S9G spent fuel modules, not to exceed 55,002 BTU/hr decay heat per package at the time of container draining.

(c) Criticality Safety Index

Spent fuel module	Criticality Safety Index
S3G-3	100
S8G	100
D2W	0
A1G	0
S6W	100
S9G	0

6. For S3G-3 spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) Minimum fuel cooling time is 130 days after shutdown.

   (c) Core age must be at least 4,000 logging corrected full power hours.

   (d) Control rod hold-down devices must be installed on cells which have control rods. Module grapple adapters serve as poison shipping rod holddown devices for refueling shipments.

7. For S8G spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) Minimum fuel cooling time is 248 days after shutdown for prototype modules and 157 days after shutdown for shipboard modules.

   (c) Full and partial fuel modules may be shipped in any combination, but all modules must be shipped with control rods.

   (d) Control rod holddown devices must be installed on the cells. Module grapple adapters serve as control rod holddown devices.

8. For D2W spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) Minimum fuel cooling time is 150 days after shutdown.

   (c) Control rod holddown devices must be installed on all rodded modules. The universal grapple adapters serve as the rod holddown devices.

9. For A1G spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) All fuel clusters must be shipped with either control rods or poison shipping rods, with rod holddown devices installed.

(c) Minimum fuel cooling time shall be the greater of 50 days after shutdown or that calculated using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging.

10. For S6W spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) The minimum fuel cooling time before container draining shall be 300 days after shutdown for a shipboard core or 450 days after shutdown for a prototype core.

(c) All fuel modules must be shipped with control rods, control rod restraints, and grapple adapters installed. A lower pedestal must be installed in each module holder port.

11. For S9G spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) The minimum fuel cooling time is 100 days.

(c) All S9G spent fuel modules must have control rods, control rod holddown devices, and grapple adapters installed.

12. The package must contain no more than 6 gallons of residual water, except that shipments of D2W recoverable irradiated fuel may contain up to 11 gallons of residual water.

13. Failed fuel or fuel with defective cladding is not authorized for shipment.

14. Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as amended, except:

All containment seals, including the main closure head seal, must be replaced with new seals within the 12-month period prior to each shipment, or earlier if inspection shows any defect.

15. The package must be prepared for transport and operated in accordance with Chapter 7 of the application, except:

The containment seals, excluding the main closure head seal, must pass a leak test after final closure prior to each shipment. The leak test must have a sensitivity of at least 1×10^{-3} std-cm^3/sec.

16. Prior to first use, and within the 12-month period prior to each shipment, all containment seals, including the main closure head seal, must be leak tested to show a leak rate no greater than 1×10^{-4} std-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-5} std-cm^3/sec.

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17. Fabrication of packages must have been completed by December 31, 2006 in accordance with 10 CFR 71.19(c).

18. Transport by air of fissile material is not authorized.

19. Expiration date: October 31, 2016.

REFERENCES

"Core Independent M-140 Safety Analysis Report For Packaging," transmitted February 27, 1991.

Supplements dated: May 23, June 21, and July 17, 1991; February 4 and 7, August 17, and December 2, 1992; October 14, 1994; September 1, and November 16, 1995; May 13, August 7, September 26, and November 26, 1996; February 10, 1997; June 11, 1998; April 11, 2001; March 5 and November 27, 2002; April 18, 2006; August 5, 2009; July 27, and October 12, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christine Lipa

Christine Lipa, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 11, 2012

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9795	5	71-9795	USA/9795/B(U)-96	1	OF	3

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Irradiated Component Disposal Container
Safety Analysis Report for Packaging
dated July 10, 1997, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Irradiated Component Disposal Container (ICDC)

(2) Description

The Model No. ICDC is stainless steel cask with an impact limiter at the upper end. The cask body is cylindrical in shape with overall dimensions of approximately 134.6 inches long by 122 inches diameter at the container body flange. The cask cavity is approximately 134.6 inches long by 91 inches diameter. The wall of the cask is 304 stainless steel, 10 inches thick at the bottom and 5 inches thick at the top. The bottom of the cask is an 11 inch thick circular steel plate. The cask lid is closed by a full penetration weld. The upper impact limiter is a stainless steel ring attached with 21 studs to the cask body. A centering plate and pedestals, welded to the bottom end plate, are used to position the contents within the package. The maximum weight of the package is 200,000 pounds. The maximum weight of the contents is approximately 36,300 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures and sketches included in the application documents (see References, below).

5. (b) Contents

The contents of the package are cell support housings and other miscellaneous core components from a spent reactor core. The maximum number of these components per package is specified in Section 1.1 of the application. The other contents of the package include potential residual water not greater than 6 gallons, diatomaceous earth desiccant to absorb the residual water and a stainless steel pumpdown lance which may be left in the package. The maximum radioactivity of the contents is 5,600 curies. The total radioactivity is based on transport no earlier than 50 days after core shutdown.

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

(c) The package may contain no more than 6 gallons of residual water.

(d) The ICDC shall be shipped no earlier than 50 days after core shutdown.

(e) The total number of cluster joint stud remnants loaded into each ICDC must not exceed 25.

(f) The gross weight of the package shall not exceed 200,000 pounds.

7. Transport by air of fissile material is not authorized.

8. Expiration date: April 30, 2018.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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REFERENCES

Irradiated Component Disposal Container Safety Analysis Report for Packaging dated July 10, 1997.

Supplements dated: U.S. Department of Energy, Division of Naval Reactors letters G#C98-11009, dated December 2, 1998; G#99-03507, dated May 3, 1999; G#C02-4083, dated October 23, 2002; G#07-04227, dated November 5, 2007; G#C09-01386 dated, March 3, 2009; and G#12-04231 dated September 20, 2012.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michele M. Sampson

Michele M. Sampson, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 28, 2012.

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: BYPROD. NORM. FORM

Package ID #	Model	Expiration Date
USA/9184/B(U)	USA/9184/B(U)	08/31/2014
USA/9320/B(U)-96	MIDUS	05/31/2017
USA/9337/B(U)-96	3979A	01/31/2016
USA/9342/AF-96	VERSA-PAC VP110	06/30/2015
USA/9342/AF-96	VERSA-PAC	06/30/2015

Type of Packaging: BYPROD. SPEC. FORM

Package ID #	Model	Expiration Date
USA/6613/B(U)-96	702	06/30/2018
USA/9027/B(U)-96	741-OP	10/31/2015
USA/9035/B(U)-96	680-OP	10/31/2015
USA/9036/B(U)-96	C-1	10/31/2016
USA/9056/B(U)	SPEC 2-T	04/30/2015
USA/9148/B(U)-85	770	03/31/2018
USA/9157/B(U)-96	IR-100	10/31/2014
USA/9185/B(U)-96	OP-100	02/28/2014
USA/9187/B(U)-96	865	03/31/2019
USA/9215/B(U)	NPI-20WC-6 MKII	05/31/2018
USA/9263/B(U)-96	SPEC-150	06/30/2015
USA/9269/B(U)-96	650L	11/30/2015
USA/9282/B(U)-96	SPEC-300	04/30/2015
USA/9283/B(U)-96	OP-660	06/30/2013
USA/9287/B(U)-85	EAGLE	12/31/2014
USA/9290/B(U)-85	F-430/GC-40	02/28/2017
USA/9296/B(U)-96	880 SERIES PKG	06/30/2016
USA/9299/B(U)-85	F-423	03/31/2017
USA/9310/B(U)-96	F-431	06/30/2014
USA/9314/B(U)-96	976 SERIES	07/31/2014
USA/9316/B(U)-96	AOS-100A-S	02/28/2017
USA/9316/B(U)-96	AOS-100B	02/28/2017
USA/9316/B(U)-96	AOS-100A	02/28/2017
USA/9316/B(U)-96	AOS-050A	02/28/2017

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Package ID #	Model	Expiration Date
USA/9316/B(U)-96	AOS-025A	02/28/2017
USA/9321/B(U)-96	3-60B	08/31/2015
USA/9357/B(U)-96	SENTRY 330	07/30/2016
USA/9357/B(U)-96	USA/9357/B(U)96	07/30/2016
USA/9357/B(U)-96	SENTRY 867	07/30/2016

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: FISSILE URANIUM

Package ID #	Model	Expiration Date
USA/5086/B(U)F	UNC-2600	11/30/2014
USA/5797/B(U)F	INNER HFIR UN	10/31/2017
USA/5797/B(U)F	OUTER HFIR UN	10/31/2017
USA/6386/B(U)F-96	235R001	04/30/2015
USA/9034/AF	TRIGA-I	12/31/2015
USA/9037/AF	TRIGA-II	12/31/2015
USA/9099/B(U)F-85	ATR	01/31/2014
USA/9186/B(U)F-96	MODEL 2 S-6213	03/31/2017
USA/9186/B(U)F-96	MODEL 1 S-6213	03/31/2017
USA/9196/B(U)F-96	UX-30	02/28/2016
USA/9217/AF	ANF-250	06/30/2015
USA/9239/AF	MCC-5	03/31/2017
USA/9239/AF	MCC-4	03/31/2017
USA/9239/AF	MCC-3 -4 & -5	03/31/2017
USA/9246/AF	ST	01/31/2017
USA/9248/AF	SP-3	04/30/2014
USA/9248/AF	SP-2	04/30/2014
USA/9248/AF	SP-1 SP-2 SP-3	04/30/2014
USA/9250/B(U)F-85	5X22	10/31/2014
USA/9251/AF	BW-2901	01/31/2018
USA/9252/AF	51032-2	10/31/2013
USA/9274/AF	ABB-2901	09/30/2017
USA/9281/AF-85	UBE-2	08/31/2013
USA/9284/B(U)F-85	ESP-30X	05/31/2015

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Package ID #	Model	Expiration Date
USA/9285/AF-85	SRP-1	10/31/2013
USA/9288/B(U)F-96	CHT-OP-TU	03/31/2015
USA/9289/B(U)F-85	WE-1	02/28/2014
USA/9291/B(U)F-96	LIQUI-RAD	10/31/2016
USA/9292/AF-85	PATRIOT	08/31/2015
USA/9294/AF-96	NPC	11/30/2015
USA/9295/B(U)F-96	MFFP	06/30/2015
USA/9297/AF-96	TRAVELLER XL	03/31/2015
USA/9297/AF-96	TRAVELLER STD	03/31/2015
USA/9301/AF-85	TNF-XI	08/31/2013
USA/9309/B(U)F-96	RAJ-II	11/30/2014
USA/9315/B(U)F-96	ES-3100	04/30/2016
USA/9319/B(U)F-96	MAP-12, MAP-13	01/31/2018
USA/9328/AF-96	TN-55	04/30/2017
USA/9330/AF-96	ATR FFSC	05/30/2014

Type of Packaging: IRRADIATED FUEL

Package ID #	Model	Expiration Date
USA/9225/B(U)F-96	NAC-LWT	02/28/2015
USA/9226/B(U)F-85	GA-4	10/31/2013
USA/9228/B(U)F-96	2000	05/31/2016
USA/9235/B(U)F-96	NAC-STC	05/31/2014
USA/9253/B(U)F-96	TN-FSV	06/30/2014
USA/9255/B(U)F-85	NUHOMS MP187	11/30/2018
USA/9261/B(U)F-96	HI-STAR 100	03/31/2014
USA/9270/B(U)F-96	UMS UNIVERSAL	10/31/2017
USA/9276/B(U)F-85	TS125	10/31/2017
USA/9293/B(U)F-85	TN-68	02/29/2016
USA/9302/B(U)F-85	NUHOMS-MP197	08/31/2017
USA/9313/B(U)F-96	07109313	06/30/2016
USA/9325/B(U)F-96	HI-STAR 180	10/31/2014
USA/9336/B(U)F-96	HI-STAR 60	05/31/2014
USA/9341/B(U)F-96	BRR	01/22/2015
USA/9358/B(U)F-96	TN-LC	12/31/2017
USA/9793/B(U)F-85	M-140	10/31/2016

Type of Packaging: PU AIR

Package ID #	Model	Expiration Date
USA/0361/B(U)F-96	PAT-1	12/31/2015

Type of Packaging: PU NORM. FORM

Package ID #	Model	Expiration Date
USA/9212/B(M)F-96	RH-TRU 72-B	02/28/2015
USA/9218/B(U)F-96	TRUPACT-II	08/31/2014
USA/9221/B(M)F-96	NRBK-41	04/30/2018
USA/9279/B(U)F-96	HALFPACT	10/31/2015
USA/9305/B(U)F-96	TRUPACT-III	06/30/2015

Type of Packaging: PU SPEC. FORM

Package ID #	Model	Expiration Date
USA/9329/AF-96	S300	01/31/2017

Type of Packaging: WASTE, B

Package ID #	Model	Expiration Date
USA/9168/B(U)	CNS 8-120B	08/31/2017
USA/9204/B(U)-85	CNS 10-160B	10/31/2015
USA/9233/B(U)-96	TN-RAM	04/30/2015
USA/9786/B(U)	S3G CBDCA	08/31/2016
USA/9788/B(U)-96	S5W REC. COMPT.	09/30/2013
USA/9788/B(U)-96	SSN 688	09/30/2013
USA/9791/B(U)-85	PWR-2 CORE BAR.	07/31/2017
USA/9792/B(U)	D1G CB-TS	01/31/2018
USA/9794/B(U)-96	CGN RCDP	02/28/2011
USA/9795/B(U)-96	ICDC	04/30/2018

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11. ABSTRACT (200 words or less) The purpose of this directory is to make available a convenient source of information on package designs approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packages, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volume 2. The report includes all package designs approved prior to the publication date of the directory as of September 2013.			
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