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

May 11, 2000

Mr. Brian Gutherman, Licensing Manager
Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9261 FOR THE HI-STAR 100 SYSTEM

Dear Mr. Gutherman:

As requested by your application dated November 24, 1999, as supplemented, enclosed is Certificate of Compliance No. 9261, Revision No. 1, for the Model No. HI-STAR 100 System. This certificate supersedes, in its entirety, Certificate of Compliance No. 9261, Revision No. 0, dated March 31, 1999. Changes made to the enclosed certificate are indicated by vertical lines in the margin.

Holtec International has been registered as a user of the package under the general license provisions of 10 CFR 71.12. The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me or Marissa Bailey of my staff at (301) 415-8500.

Sincerely,

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9261
TAC No. L23012

Enclosures: 1. Certificate of Compliance No. 9261, Rev. No. 1
2. Safety Evaluation Report

cc w/encl: R. Boyle, Department of Transportation
M. Wangler, Department of Energy

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9261 | 1 | USA/9261/B(U)F-85 | 1 | 7 |

2. PREAMBLE

a. This certificate is issued to certify that the 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-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

c. DOCKET NUMBER 71-9261

4. CONDITIONS

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

5.

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 which house the spent nuclear fuel and an overpack which 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 203 1/8 inches without impact limiters and approximately 305 7/8 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is approximately 282,000 pounds. Specific tolerances are called out in drawings listed below.

**Multi-Purpose Canister**

There are three Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions. A single overpack design is provided which is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 loaded with material classified as fuel debris is designated as MPC-68F.

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 MPC is fixed. However, the number of spent fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. For the HI-STAR 100 System transporting fuel debris in a MPC-68F, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a strength-welded enclosure constructed entirely of a stainless steel alloy.

5. a. (2) Description (continued)

**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 for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9:

| | |
|---|---|
| (a) HI-STAR 100 MPC-24 | Drawing C1395, Sheets 1-4, Rev. 1<br>Drawing C1396, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0<br>Drawing BM-C1478, Sheets 1& 2, Rev. 1 |
| (b) HI-STAR 100 MPC-68 and MPC-68F | Drawing C1401, Sheets 1-4, Rev. 1<br>Drawing C1402, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0<br>Drawing BM-C1479, Sheets 1& 2, Rev. 1 |
| (c) HI-STAR 100 Overpack | Drawing C1397, Sheet 1, Rev. 2; and Sheets 2-7, Rev. 1<br>Drawing C1398, Sheets 1-3, Rev. 1<br>Drawing C1399, Sheets 1-2, Rev. 1; and Sheet 3, Rev. 2<br>Drawing BM-C1476, Sheet 1, Rev. 1; and Sheet 2, Rev. 2 |
| (d) HI-STAR 100 Impact Limiters | Drawing C1765, Sheets 1-6, Rev. 1; and Sheet 7, Rev. 0 |
| (e) HI-STAR 100 Assembly for Transport | Drawing C1782, Rev. 1 |

5. b. Contents

(1) Type and 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)(g) below are authorized for transportation.

(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, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

**Damaged Fuel Containers** (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris which 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 and 1.2.11 of Holtec International Report No. HI-951251, Rev. 9.

**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.

**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. Partial fuel assemblies, that is 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 greater than or equal to that displaced by the original fuel rod(s).

**Minimum Enrichment** is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

**Planar-Average Initial Enrichment** is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

(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 two limits for the stainless steel clad fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the damaged fuel assemblies or the intact fuel assemblies.

5.. b. (1) Type and Form, and Quantity of Material (continued)

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR control rods, burnable poison rod assemblies, thimble plugs, and other non-fuel hardware are not authorized for transportation.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

c. Transport Index for Criticality Control

The minimum transport index to be shown on the label for nuclear criticality control: 0

6. 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 detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the following provisions:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b above.

(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.

(3) The package must satisfy the following leak testing requirements:

(a) All overpack containment boundary seals shall be leak tested to show a leak rate of not greater than $4.3 \times 10^{-6}$ atm $cm^3$/sec (helium). The leak test shall have a minimum sensitivity of $2.15 \times 10^{-6}$ atm $cm^3$/sec (helium) and shall be performed:

(i) before the first shipment;
(ii) within the 12-month period prior to each successive shipment;
(iii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iv) after each seal replacement.

(b) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of $1 \times 10^{-3}$ atm $cm^3$/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a(3)(a) above.

(c) Each containment boundary seal must be replaced after each use of the seal.

(4) The rupture discs on the neutron shield vessel shall be replaced every 5 years.

6. a. (continued)

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than $5 \times 10^{-6}$ atm $cm^3$/sec (helium).

(6) Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.
(b) The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(7) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium: $\geq$ 1 atm and $\leq$ 28.3 psig for the MPC-24, and $\geq$ 1 atm and $\leq$ 28.5 psig for the MPC-68 and MPC-68F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(9) Following vacuum drying, the overpack shall be backfilled with helium to $\geq$ 10 psig and $\leq$ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

| Fastener | Torque (ft-lbs) |
|---|---|
| Overpack Closure Plate Bolts | 2895 $\pm$ 90 |
| Overpack Vent and Drain Port Plugs | 45 +5/-0 |
| Top Impact Limiter Attachment Bolts | 256 +10/-0 |
| Bottom Impact Limiter Attachment Bolts | 1500 +45/-0 |
| Tie-down Bolts | 250 +20/-0 |
| Transport Frame Bolts | 250 +20/-0 |

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.

b. All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed and shall include the following provisions:

(1) The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.

(2) The MPC shall be pressure tested to 125% of the design pressure. The minimum test pressure shall be 125 psig.

6. b. (continued)

(3) The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV 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.

(5) The radial neutron shield shall have a minimum thickness of 4.3 inches and the impact limiter neutron shields shall have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity shall be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Measurements shall be performed over the entire exterior surface of the radial neutron shield and each impact limiter using, at a maximum, a 6 x 6 inch test grid.

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years of each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be performed at a minimum of 12 locations on the radial neutron shield and at a minimum of 4 locations on each impact limiter

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 1.4 of the SAR, Rev. 9. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

(8) For each package, a periodic thermal performance test shall be performed every 5 years or prior to next use, if the package has not been used for transport for greater than 5 years, to demonstrate that the thermal capabilities of the cask remain within its design basis.

(9) The neutron absorber's minimum acceptable $^{10}B$ loading is 0.0267 g/cm$^2$ for the MPC-24 and 0.0372 g/cm$^2$ for the MPC-68, and 0.01 g/cm$^2$ for the MPC-68F. The $^{10}B$ loading shall be verified by chemistry or neutron attenuation techniques.

(10) The minimum flux trap size for the MPC-24 is 1.09 inches.

(11) The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

(12) The package containment verification leak test shall be per ANSI 14.5.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 6 feet (along the axis of the overpack) from the edge of the vehicle.

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 71.12.

11. Expiration Date: March 31, 2004

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 11, 2000

# APPENDIX A

## CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 1

## MODEL NO. HI-STAR 100 SYSTEM

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| Page A-1 to A15 | Table A1 | 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-8 | | 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-9 | | 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. |

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Table A.1 (Page 1 of 15)
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: | Zircaloy (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, decay heat and minimum initial enrichment per assembly | |
| i. Zr clad: | An assembly post-irradiation cooling time, average burnup, decay heat, 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, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable. |
| d. Fuel assembly length: | $\leq$ 176.8 inches (nominal design) |
| e. Fuel assembly width: | $\leq$ 8.54 inches (nominal design) |
| f. Fuel assembly weight: | $\leq$ 1,680 lbs |

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain control components.

D. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

Table A.1 (Page 2 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (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, decay heat and *minimum* initial enrichment per assembly: | |
| i. Zr clad: | An assembly post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% $^{235}U$, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, a decay heat ≤ 183.5 Watts, and a minimum initial enrichment ≥ 2.4 wt% $^{235}U$. |
| ii. SS clad: | An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% $^{235}U$. |
| e. Fuel assembly length: | ≤ 176.2 inches (nominal design) |
| f. Fuel assembly width: | ≤ 5.85 inches (nominal design) |
| g. Fuel assembly weight | ≤ 700 lbs, including channels |

Table A.1 (Page 3 of 15)
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: | Zircaloy (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 $\geq$ 1.8 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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 4 of 15)
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: | Zircaloy (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 15)
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: | Zircaloy (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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 6 of 15)
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 Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (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 7 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

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 8 of 15)
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: | Zircaloy (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 $\geq$ 1.8 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 9 of 15)
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: | Zircaloy (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 $\geq$ 1.8 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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 10 of 15)
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: | Zircaloy (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 $\geq$ 1.8 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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 11 of 15)
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: | Zircaloy (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 12 of 15)
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: | Zircaloy (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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 13 of 15)
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: | Zircaloy (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 | $\leq$ 400 lbs, including channels |

Table A.1 (Page 14 of 15)
Fuel Assembly Limits

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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 Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

| | |
|---|---|
| a. Cladding Type: | Zircaloy (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 15 of 15)
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.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| **Fuel Assembly Array/Class** | **14x14A** | **14x14B** | **14x14C** | **14x14D** | **15x15A** |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | SS | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 407 | ≤ 407 | ≤ 425 | ≤ 400 | ≤ 464 |
| Initial Enrichment (wt % $^{235}U$) | ≤ 4.6 | ≤ 4.6 | ≤ 4.6 | ≤ 4.0 | ≤ 4.1 |
| No. of Fuel Rods | 179 | 179 | 176 | 180 | 204 |
| Clad O.D. (in.) | ≥ 0.400 | ≥ 0.417 | ≥ 0.440 | ≥ 0.422 | ≥ 0.418 |
| Clad I.D. (in.) | ≤ 0.3514 | ≤ 0.3734 | ≤ 0.3880 | ≤ 0.3890 | ≤ 0.3660 |
| Pellet Dia. (in.) | ≤ 0.3444 | ≤ 0.3659 | ≤ 0.3805 | ≤ 0.3835 | ≤ 0.3580 |
| Fuel Rod Pitch (in.) | ≤ 0.556 | ≤ 0.556 | ≤ 0.580 | ≤ 0.556 | ≤ 0.550 |
| Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 144 | ≤ 150 |
| No. of Guide Tubes | 17 | 17 | 5 (Note 4) | 16 | 21 |
| Guide Tube Thickness (in.) | ≥ 0.017 | ≥ 0.017 | ≥ 0.038 | ≥ 0.0145 | ≥ 0.0165 |

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 15x15B | 15x15C | 15x15D | 15x15E | 15x15F |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 464 | ≤ 464 | ≤ 475 | ≤ 475 | ≤ 475 |
| Initial Enrichment (wt % $^{235}U$) | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 |
| No. of Fuel Rods | 204 | 204 | 208 | 208 | 208 |
| Clad O.D. (in.) | ≥ 0.420 | ≥ 0.417 | ≥ 0.430 | ≥ 0.428 | ≥ 0.428 |
| Clad I.D. (in.) | ≤ 0.3736 | ≤ 0.3640 | ≤ 0.3800 | ≤ 0.3790 | ≤ 0.3820 |
| Pellet Dia. (in.) | ≤ 0.3671 | ≤ 0.3570 | ≤ 0.3735 | ≤ 0.3707 | ≤ 0.3742 |
| Fuel Rod Pitch (in.) | ≤ 0.563 | ≤ 0.563 | ≤ 0.568 | ≤ 0.568 | ≤ 0.568 |
| Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Guide Tubes | 21 | 21 | 17 | 17 | 17 |
| Guide Tube Thickness (in.) | ≥ 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 (wt % $^{235}U$) | ≤ 4.0 | ≤ 3.8 | ≤ 4.6 | ≤ 4.0 | ≤ 4.0 | ≤ 4.0 |
| No. of Fuel Rods | 204 | 208 | 236 | 264 | 264 | 264 |
| Clad O.D. (in.) | ≥ 0.422 | ≥ 0.414 | ≥ 0.382 | ≥ 0.360 | ≥ 0.372 | ≥ 0.377 |
| Clad I.D. (in.) | ≤ 0.3890 | ≤ 0.3700 | ≤ 0.3320 | ≤ 0.3150 | ≤ 0.3310 | ≤ 0.3330 |
| 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 Tubes | 21 | 17 | 5 (Note 4) | 25 | 25 | 25 |
| Guide 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 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.

Table A.3 (Page 1 of 5)
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) | ≤ 110 | ≤ 110 | ≤ 110 | ≤ 100 | ≤ 195 | ≤ 120 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 2.7 | ≤ 2.7 for the $UO_2$ rods. See Note 4 for MOX rods | ≤ 2.7 | ≤ 2.7 | ≤ 4.2 | ≤ 2.7 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 4.0 | ≤ 4.0 | ≤ 4.0 | ≤ 5.5 | ≤ 5.0 | ≤ 4.0 |
| No. of Fuel Rods | 35 or 36 | 35 or 36 (up to 9 MOX rods) | 36 | 49 | 49 | 63 or 64 |
| Clad O.D. (in.) | ≥ 0.5550 | ≥ 0.5625 | ≥ 0.5630 | ≥ 0.4860 | ≥ 0.5630 | ≥ 0.4120 |
| Clad I.D. (in.) | ≤ 0.5105 | ≤ 0.4945 | ≤ 0.4990 | ≤ 0.4204 | ≤ 0.4990 | ≤ 0.3620 |
| Pellet Dia. (in.) | ≤ 0.4980 | ≤ 0.4820 | ≤ 0.4880 | ≤ 0.4110 | ≤ 0.4910 | ≤ 0.3580 |
| Fuel Rod Pitch (in.) | ≤ 0.710 | ≤ 0.710 | ≤ 0.740 | ≤ 0.631 | ≤ 0.738 | ≤ 0.523 |
| Active Fuel Length (in.) | ≤ 120 | ≤ 120 | ≤ 77.5 | ≤ 80 | ≤ 150 | ≤ 120 |
| No. of Water Rods (Note 11) | 1 or 0 | 1 or 0 | 0 | 0 | 0 | 1 or 0 |
| Water Rod Thickness (in.) | ≥ 0 | ≥ 0 | N/A | N/A | N/A | ≥ 0 |
| Channel Thickness (in.) | ≤ 0.060 | ≤ 0.060 | ≤ 0.060 | ≤ 0.060 | ≤ 0.120 | ≤ 0.100 |

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 8x8B | 8x8C | 8x8D | 8x8E | 8x8F | 9x9A | 9x9B |
|---|---|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 177 | ≤ 177 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.2 | ≤ 4.2 | < 3.6 | ≤ 4.2 | ≤ 4.2 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 |
| No. of Fuel Rods | 63 or 64 | 62 | 60 or 61 | 59 | 64 | 74/66 (Note 5) | 72 |
| Clad O.D. (in.) | ≥ 0.4840 | ≥ 0.4830 | ≥ 0.4830 | ≥ 0.4930 | ≥ 0.4576 | ≥ 0.4400 | ≥ 0.4330 |
| Clad I.D. (in.) | ≤ 0.4295 | ≤ 0.4250 | 0.4230 | ≤ 0.4250 | ≤ 0.3996 | ≤ 0.3840 | ≤ 0.3810 |
| Pellet Dia. (in.) | ≤ 0.4195 | ≤ 0.4160 | ≤ 0.4140 | ≤ 0.4160 | ≤ 0.3913 | ≤ 0.3760 | ≤ 0.3740 |
| Fuel Rod Pitch (in.) | ≤ 0.642 | ≤ 0.641 | ≤ 0.640 | ≤ 0.640 | ≤ 0.609 | ≤ 0.566 | ≤ 0.572 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 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 | 1 (Note 6) |
| Water Rod Thickness (in.) | ≥ 0.034 | > 0.00 | > 0.00 | ≥ 0.034 | ≥ 0.0315 | > 0.00 | > 0.00 |
| Channel Thickness (in.) | ≤ 0.120 | ≤ 0.120 | ≤ 0.120 | ≤ 0.100 | ≤ 0.055 | ≤ 0.120 | ≤ 0.120 |

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| **Fuel Assembly Array/Class** | **9x9C** | **9x9D** | **9x9E (Note 13)** | **9x9F (Note 13)** | **10x10A** |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 177 | ≤ 177 | ≤ 177 | ≤ 177 | ≤ 186 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.1 | ≤ 4.1 | ≤ 4.2 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 |
| No. of Fuel Rods | 80 | 79 | 76 | 76 | 92/78 (Note 8) |
| Clad O.D. (in.) | ≥ 0.4230 | ≥ 0.4240 | ≥ 0.4170 | ≥ 0.4430 | ≥ 0.4040 |
| Clad I.D. (in.) | ≤ 0.3640 | ≤ 0.3640 | ≤ 0.3640 | ≤ 0.3860 | ≤ 0.3520 |
| Pellet Dia. (in.) | ≤ 0.3565 | ≤ 0.3565 | ≤ 0.3530 | ≤ 0.3745 | ≤ 0.3455 |
| Fuel Rod Pitch (in.) | ≤ 0.572 | ≤ 0.572 | ≤ 0.572 | ≤ 0.572 | ≤ 0.510 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Water Rods (Note 11) | 1 | 2 | 5 | 5 | 2 |
| Water Rod Thickness (in.) | ≥ 0.020 | ≥ 0.0300 | ≥ 0.0120 | ≥ 0.0120 | ≥ 0.0300 |
| Channel Thickness (in.) | ≤ 0.100 | ≤ 0.100 | ≤ 0.120 | ≤ 0.120 | ≤ 0.120 |

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| **Fuel Assembly Array/Class** | **10x10B** | **10x10C** | **10x10D** | **10x10E** |
|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | SS | SS |
| Design Initial U (kg/assy.) (Note 3) | ≤ 186 | ≤ 186 | ≤ 125 | ≤ 125 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.0 | ≤ 4.0 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5 |
| No. of Fuel Rods | 91/83 (Note 9) | 96 | 100 | 96 |
| Clad O.D. (in.) | ≥ 0.3957 | ≥ 0.3780 | ≥ 0.3960 | ≥ 0.3940 |
| Clad I.D. (in.) | ≤ 0.3480 | ≤ 0.3294 | ≤ 0.3560 | ≤ 0.3500 |
| Pellet Dia. (in.) | ≤ 0.3420 | ≤ 0.3224 | ≤ 0.3500 | ≤ 0.3430 |
| Fuel Rod Pitch (in.) | ≤ 0.510 | ≤ 0.488 | ≤ 0.565 | ≤ 0.557 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 83 | ≤ 83 |
| No. of Water Rods (Note 11) | 1 (Note 6) | 5 (Note 10) | 0 | 4 |
| Water Rod Thickness (in.) | > 0.00 | ≥ 0.031 | N/A | ≥ 0.022 |
| Channel Thickness (in.) | ≤ 0.120 | ≤ 0.055 | ≤ 0.080 | ≤ 0.080 |

Table A.3 (Page 5 of 5)
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 75 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.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 10 | ≤ 24,500 | ≥ 2.3 | ≤ 411 |
| ≥ 12 | ≤ 29,500 | ≥ 2.6 | ≤ 473 |
| ≥ 14 | ≤ 34,500 | ≥ 2.9 | ≤ 540 |
| ≥ 15 | ≤ 37,500 | ≥ 3.2 | ≤ 579 |

Note 1: Linear interpolation between points is permitted.

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 7 | ≤ 24,500 | ≥ 2.3 | ≤ 496 |
| ≥ 8 | ≤ 29,500 | ≥ 2.6 | ≤ 562 |
| ≥ 10 | ≤ 34,500 | ≥ 2.9 | ≤ 610 |
| ≥ 12 | ≤ 39,500 | ≥ 3.2 | ≤ 667 |
| ≥ 15 | ≤ 44,100 | ≥ 3.4 | ≤ 704 |

Note 1: Linear interpolation between points is permitted.

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH STAINLESS STEEL CLAD (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 19 | ≤ 30,000 | ≥ 3.1 | ≤ 377 |
| ≥ 24 | ≤ 40,000 | ≥ 3.1 | ≤ 475 |

Note 1: Linear interpolation between points is permitted.

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68 (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 8 | ≤ 24,500 | ≥ 2.1 | ≤ 179 |
| ≥ 9 | ≤ 29,500 | ≥ 2.4 | ≤ 208 |
| ≥ 12 | ≤ 34,500 | ≥ 2.6 | ≤ 222 |
| ≥ 15 | ≤ 39,100 | ≥ 2.9 | ≤ 238 |

Note 1: Linear interpolation between points is permitted.

**REFERENCE:**

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

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

# SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR 100 System
Certificate of Compliance No. 9261
Revision No. 1

## SUMMARY

By application dated November 24, 1999, as supplemented*, Holtec International (the applicant) requested an amendment to Certificate of Compliance No. 9261 for the Model No. HI-STAR 100 System.

The applicant requested that the certificate be amended to:

(1) incorporate revised drawings,
(2) revise the parameter limits to previously approved fuel assembly array/classes,
(3) allow transportation of two new fuel assembly array/classes,
(4) allow use of a new damaged fuel container (DFC),
(5) allow transportation of thoria rods in canisters,
(6) allow transportation of antimony-beryllium neutron sources,
(7) incorporate minor changes to the Holtite neutron shield characteristics, and
(8) make other minor changes, editorial corrections and clarifications.

In support of its application, Holtec provided the necessary engineering analyses and drawing changes. Holtec also submitted Revision 9 to the Safety Analysis Report (SAR) for the HI-STAR 100 System on April 20, 2000. Revision 9 incorporates the changes proposed by the application, as supplemented, and supersedes the previous revisions to the SAR.

Based on the statements and representations in the application, as supplemented, and Revision 9 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

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*Supplements dated February 4, 18 and 28, March 2, 16 and 31, and April 20, 2000.

## 1.0 GENERAL

### Drawings

The applicant requested that the engineering drawings referenced in the certificate be replaced by a revised set of certificate drawings. The revised drawings do not involve any significant changes to the previously approved package design. Mainly, the drawings were revised to provide flexibility and remove unnecessary information, such as fabrication details and extraneous dimensions and notes that are not considered in the safety evaluation. All safety significant information and all information pertinent to the safety evaluation of the package have been retained, including:

(1) general arrangement of the packaging and contents, including dimensions;

(2) information on design features that affect the evaluation, such as:
- identification of the design feature and its components;
- materials of construction, including appropriate material specifications;
- codes, standards, or other specifications for fabrication, assembly, and testing;
- location with respect to other package features;
- dimensions with appropriate tolerances;
- operational specifications;
- and weld design and inspection method;

(3) package markings; and

(4) maximum allowable weight of the package.

The staff reviewed the revised certificate drawings and finds that the information on the drawings is sufficiently detailed, identifies the package accurately, and is consistent with the package as described and evaluated in the SAR. The staff agrees that the revised drawings do not entail any significant changes to the design and operation of the package. The staff finds the revised certificate drawings acceptable.

### Minor Changes

The applicant requested several minor changes to the certificate. These include:

(1) increasing the specified package weight from 280,000 pounds to 282,000 pounds to match the package weight considered in the existing structural evaluation;

(2) revising the definitions in Condition 5.b(1)(b) of the certificate to be consistent with the definitions in the HI-STAR 100 storage certificate of compliance (Docket No. 72-1008, Certificate of Compliance No. 1008);

(3) revising the MPC helium backfill density limit to a maximum helium backfill pressure for simplicity;

(4) removing the note in Condition 6.a(1) of the certificate that specifies the number of passes and pattern for torquing the closure bolts to provide flexibility to users;

(5) increasing the torque value for the drain port plugs to 45 +5/-0 based on the seal manufacturer's recommendation to ensure sufficient compression of the seal;

(6) revising the trunnion testing requirement to be consistent with the HI-STAR 100 storage certificate of compliance;

(7) revising the thermal test requirement so that only the first fabricated overpack is tested, consistent with the HI-STAR 100 storage certificate of compliance;

(8) permitting linear interpolation between points (cooling time, burnup, minimum enrichment, and decay heat) in Tables A.4, A.5, A.6, and A.7 of Appendix A to the certificate to provide flexibility to users; and

(9) editorial changes and clarification.

The staff has reviewed these changes and finds them acceptable. These changes are consistent with or supported by the analyses that have been previously reviewed and approved by the staff. These changes have no adverse impact on the design and operation of the package and will not affect the ability of the package to meet the requirements of 10 CFR Part 71.

## 2.0 STRUCTURAL EVALUATION

The requested changes do not have an impact on the structural performance or integrity of the packaging and its contents.

The drawings were revised mainly to eliminate inconsistencies, replace non-essential dimensions and tolerances, and remove ambiguities in the verbiage of the drawing notes. A variety of enhancements have also been incorporated into the revised drawings, including: (1) eliminating MPC basket shims to allow flexibility to the manufacturer; (2) adding options to change the sheathing weld length and pitch to the extent that waviness is minimized while the total amount of weld remains the same; (3) adding an optional weld detail for the overpack neutron shield enclosure panel to a radial channel weld (the reduction in the amount of weld material allows for a more efficient fabrication process, yet still meet all structural design requirements); and (4) reducing the closure ring welds to 1/8 inch and deleting the liquid penetrant test required for the root pass of the closure ring welds (the 1/8-inch welds will not have separate root and final passes; the final pass is appropriate in addition to the visual inspection). These enhancements do not affect the structural performance of the package.

A new DFC, the Transnuclear Dresden Unit 1 (TN/D-1) DFC, and a Thoria Rod Canister with 18 thoria rods were added to the list of contents. The TN/D-1 DFC and Thoria Rod Canister were structurally evaluated and found to meet all required design requirements for transportation in the HI-STAR 100 package. The structural analysis is provided in Appendix 2.AO of the SAR. Results of the analysis show all factors of safety to be greater than 1.0.

Two new fuel assembly array/classes (8x8F and 15x15H) and antimony-beryllium sources were added to the list of contents. Also, changes were made to the parameter limits of some previously approved fuel assemblies, including increases to their initial uranium masses. These changes do not increase to the weight of the contents or package and, therefore, do not affect the existing structural evaluation. The other proposed changes also have no structural impact since they generally reflect changes to be consistent with HI-STAR 100 storage certificate of compliance and to eliminate the need for Part 71 certificate amendments for non-safety-significant changes to the design.

The proposed changes to the certificate and drawings have been reviewed and found acceptable. The changes will have no impact on the structural performance of the HI-STAR 100 System under normal conditions of transport and hypothetical accident conditions.

## 3.0 THERMAL EVALUATION

A thermal evaluation is not necessary. The requested changes do not involve an increase to the decay heat load or a change to the heat transfer characteristics of the package. The new contents and content limits are bounded by the thermal analysis for previously approved contents.

## 4.0 CONTAINMENT EVALUATION

A containment evaluation is not necessary. The requested changes do not involve an increase to the containment source terms or a significant change to the design and operation of the containment system. The new contents and content limits are bounded by the containment analysis for previously approved contents.

## 5.0 SHIELDING EVALUATION

The following proposed changes were considered for their impact on the shielding evaluation:

(1) addition of the TN/D-1 DFC to the approved contents;

(2) addition of the Dresden Unit 1 Thoria Rod Canister, with up to 18 thoria rods, to the approved contents;

(3) addition of the Dresden Unit 1 assemblies with one antimony-beryllium neutron source to the approved contents;

(4) revision of the uranium masses for some fuel assemblies;

(5) revision of fuel assembly parameter limits for some fuel assemblies;

(6) addition of two new fuel assembly array/classes to the approved contents;

(7) revision of the SAR and drawings to specify nominal values for the Holtite $B_4C$ content and Holtite specific gravity;

(8) changes to the material composition testing requirements of Holtite.

**TN/D-1 Damaged Fuel Container**

The HI-STAR 100 is currently approved to transport damaged fuel or fuel debris when the fuel is contained in a Holtec DFC described in the SAR. The applicant requested the addition of the TN/D-1 DFC to the HI-STAR approved contents. Figure 1.2.11 in the SAR show the dimensions of the TN/D-1 DFC. The source term for both containers will be the same since the allowed fuel types are identical.

For damaged fuel and fuel debris, the applicant assumed that the fuel collapsed to a height of 80 inches. This height was determined by using the inner dimensions of the Holtec DFC. The source per inch was then calculated. Since the inner diameter of the TN/D-1 DFC is smaller than the inner diameter of the Holtec DFC and the fuel is identical, the height of the collapsed fuel in a TN/D-1 DFC will be greater (i.e., for two cylinders with the same volume but different diameters, the cylinder with a smaller diameter will have a greater height). Therefore, the source per inch will be less in the TN/D-1 DFC and the shielding evaluation for the Holtec DFC in Sections 5.4.2 and 5.4.3 of the SAR bounds the TN/D-1 DFC.

Based on the review of the applicant's analysis, the staff agrees that the TN/D-1 DFC is bounded by the current analysis and further evaluation is not required.

**Dresden Unit 1 Thoria Rod Canister**

The applicant requested the addition of the Dresden Unit 1 Thoria Rod Canister to the HI-STAR 100 approved contents. The canister contains up to 18 thoria rods with a maximum burnup of 16,000 MWD/MTU and a minimum cooling time of 18 years. The applicant used SAS2H and ORIGEN-S to calculate the source terms. The thoria rod source terms, listed in SAR tables 5.2.30 and 5.2.31, were bounded by the source terms for the design basis BWR fuel in all neutron groups and in all gamma groups except in the 2.5-3.0 MeV group. To demonstrate that the gamma dose rate from the thoria rods was bounded by the design basis fuel, the gamma dose rate from a cask completely filled with the thoria rods was compared to the gamma dose rate of a cask filled with the design basis fuel. The cask with the design basis fuel had a higher gamma dose rate; thus, the Thoria Rod Canister is bounded by the shielding analysis for the design basis fuel.

The staff has reviewed the applicant's analysis and agrees that the Thoria Rod Canister is bounded by the current shielding analysis.

## Antimony-Beryllium Source in Dresden Unit 1 Fuel Assemblies

The applicant requested the addition of Dresden Unit 1 fuel assemblies containing an antimony-beryllium source to the HI-STAR approved contents. The Dresden Unit 1 fuel assembly was previously approved for storage in the HI-STAR 100 System. The beryllium produces neutrons through gamma irradiation, with the antimony (Sb-124) used as the gamma source. Since all of the initial Sb-124 has decayed away, the only gamma source available is from decay gammas from the fuel assemblies and Sb-124 activation. The applicant used MCNP to calculate the additional gamma source term from the antimony-beryllium source. The applicant conservatively neglected the reduction of antimony and beryllium while these sources were in the core. The neutron source was then calculated. Table 5.4.25 compares the calculated neutron source for the Dresden Unit 1 fuel with and without antimony-beryllium sources to the design basis fuel. As shown in the table, the Dresden Unit 1 fuel with the antimony-beryllium neutron source is bounded by the design basis fuel. The applicant also considered the gamma source due to activation of the source's stainless steel cladding, which was shown to be bounded by the design basis fuel.

The staff reviewed the applicant's analysis and agrees that Dresden Unit 1 fuel assemblies containing an antimony-beryllium source is bounded by the current shielding analysis for the design basis fuel.

## Revision of Uranium Masses

The applicant requested an increase in the maximum allowed uranium masses for some fuel assemblies. The applicant proposed to increase the masses up to the values used in the shielding analysis.

The staff agrees that the masses may be increased as requested and further evaluation is not required since these values have already been analyzed.

## Revision of Fuel Assembly Parameter Limits

The applicant requested minor changes to certain fuel assembly parameter limits such as cladding thickness and guide tube/water rod thickness. The source term is dependent upon the uranium mass. The allowable mass loadings for the specified burnup and cooling times are not being changed. Therefore, these changes do not affect the shielding analysis.

The staff agrees that the dimensional changes have a negligible impact on the shielding analysis and further evaluation is not required.

## New Fuel Assembly Array Classes

The applicant requested that two new fuel assembly array/classes, the PWR 15x15H and the BWR 8x8F, be added to the HI-STAR 100 approved contents. These assemblies are very similar to currently approved fuel assemblies and the uranium masses are bounded by the design basis fuel assemblies. The burnup and cooling times are also the same as previously analyzed; therefore, additional analysis is not necessary. These assemblies are bounded by the current shielding analysis for the design basis fuel assemblies.

The staff has reviewed the information presented in the application and agrees that these assembly array classes are bounded by the design basis fuel.

**Holtite Specific Gravity and $B_4C$ Content**

The applicant proposed changes to the SAR and drawings to specify the Holtite specific gravity and $B_4C$ content as nominal values instead of maximum and minimum values, respectively. The applicant requested these changes to allow flexibility during fabrication.

A slight increase in the specific gravity will not adversely affect the shielding capabilities of the cask. Instead, an increase in the specific gravity would increase the effectiveness of the shielding, thus reducing the surface dose rates.

The applicant performed a sensitivity study to demonstrate the effects of a slight decrease in the $B_4C$ content. The applicant showed that a reduction from 1 weight percent to 0.75 weight percent in the Holtite will have a minor impact on the dose rates. For the most bounding case, the calculated dose rates increased by 3 percent.

The staff has reviewed the applicant's analysis and agrees that these changes have a negligible impact on the shielding analysis.

**Holtite Composition Testing**

The applicant requested a change in the composition testing frequency of the Holtite shielding material. The applicant requested the frequency be changed to every manufactured lot rather than every mixed batch.

The staff agrees that changing the testing frequency to each manufactured lot will provide an appropriate level of control given that the casks are manufactured and tested under a Part 71 quality assurance program.

**Shielding Evaluation Conclusion**

Based on the review of the application, the staff concludes that the proposed changes will not affect the ability of the package to meet the dose rate requirements of 10 CFR Part 71.

## 6.0 CRITICALITY EVALUATION

The following requested changes required an update of the criticality evaluation:

(1) inclusion of the TN/D-1 DFC already loaded with Dresden Unit 1 fuel assemblies into the MPC-68 and MPC-68F;

(2) inclusion of one Dresden Unit 1 Thoria Rod Canister loaded with 18 thoria pins into the MPC-68 and MPC-68F;

(3) inclusion of Dresden Unit 1 fuel assemblies containing one antimony-beryllium neutron source in the assembly lattice;

(4) revision of allowable U-235 enrichment in the MOX rods of fuel assembly array/class 6x6B;

(5) increases in the maximum allowed design initial uranium masses for the following fuel assembly array/classes: 14x14A, 14x14B, 14x14C, 15x15A, 16x16A, 17x17A, 17x17B, 17x17C, 6x6A, 6x6B, 8x8E, 9x9A, 9x9B, 9x9D, 9x9E, 9x9F, 10x10A, 10x10B, and 10x10C;

(6) revisions to the fuel assembly parameter limits for the following fuel assembly array/classes: 14x14C, 6x6A, 6x6B, 7x7A, 7x7B, 8x8A, 8x8B, 8x8D, 9x9B, 9x9E, 9x9F, 10x10C;

(7) addition of fuel assembly array/classes 15x15H and 8x8F.

The other requested changes do not affect the package criticality evaluation.

The applicant's evaluation and the staff's confirmatory review on the requested changes are described below. The applicant provided supporting analyses similar to analyses previously reviewed by the staff for the HI-STAR 100 transportation package.

### TN/D-1 Damaged Fuel Containers

The applicant requested that the TN/D-1 DFC be approved for transport in the HI-STAR MPC-68 and MPC-68F. A sketch of the TN/D-1 DFC is provided in Figure 1.2.11 in Chapter 1 of the SAR. The model of the packaging is similar to the previous DFC models except that the TN/D-1 DFC is slightly smaller than the original Holtec DFC design. The applicant performed analyses showing that the TN/D-1 DFC may store the 6x6 and 7x7 fuel assemblies with various number of rods missing, a collapsed fuel assembly and dispersed fuel powder. These are the same contents as the original Holtec DFC design. The results of the applicant's analyses are provided in Table 6.4.8 of the SAR. The $k_{eff}$ for the TN/D-1 DFC is bounded by the Holtec DFC design in all cases with one exception. The reactivity of the system is slightly increased for a collapsed fuel array.

The staff performed independent confirmatory analyses that discreetly modeled the TN/D-1 DFC. The staff compared the results of the TN/D-1 DFC and the Holtec DFC and found comparable values for $k_{eff}$.

**Dresden Unit-1 Thoria Rods**

The applicant requested transport of one thoria rod canister within the MPC-68 or MPC-68F canisters. A sketch of the Dresden Unit 1 Thoria Rod Canister is provided in Figure 1.2.11A in the SAR. The thoria rod contents are described in Table 6.2.42. The applicant modeled the Thoria Rod Canister explicitly and performed an analysis for a cask filled with 68 of these canisters. The applicant calculated a $k_{eff}$ of 0.18. The applicant concluded that the MPC-68 or MPC-68F filled with fuel assemblies or DFCs would remain subcritical with the inclusion of a single thoria rod canister.

The staff performed a confirmatory analysis that discreetly modeled the MPC-68 filled with 68 Thoria Rod Canisters. The staff's results were comparable to those of the applicant. In addition, the staff further analyzed an MPC-68 containing 67 bounding BWR fuel assemblies and one Thoria Rod Canister. The $k_{eff}$ for this case was bounded by the MPC-68 containing 68 bounding BWR assemblies. Staff verified that all fuel assembly parameters important to criticality safety have been included in Table A.1 of Appendix A to the certificate of compliance.

**Antimony-Beryllium Neutron Source in Dresden Unit 1 Fuel Assemblies**

The applicant requested transport of several Dresden Unit 1 fuel assemblies containing one antimony-beryllium neutron source in the assembly lattice. The antimony-beryllium source is located within the water rod of the assembly. The applicant stated that the presence of an antimony-beryllium neutron source will not affect the reactivity of the system except for the moderator it displaces.

Staff verified that the antimony-beryllium sources have been included in Table A.1 of Appendix A to the certificate of compliance. The staff has reviewed the applicant's justification and agrees that the presence of a single antimony-beryllium neutron source within a water rod will not increase the overall reactivity of the system.

**Revision of Allowable U-235 Enrichment in MOX Rods**

The applicant requested an increase from 0.612 to 0.635 in the permissible U-235 weight percent for the MOX rods of the 6x6B fuel assembly array/class. The analysis and model of the packaging are similar to those used previously by the applicant. The fuel assemblies were modeled explicitly. The applicant reported that increasing the permissible U-235 weight percent from 0.612 to 0.635 resulted in an increase in keff from 0.7611 to 0.7824.

The staff has performed confirmatory analysis and agrees that the increase in the permissible U-235 weight percent increases the reactivity of the system by only a small amount. The overall $k_{eff}$ of the system remains well below 0.95.

## Increased Maximum Allowed Design Initial Uranium Masses

The applicant requested an increase in the maximum allowed design initial uranium masses for the following fuel array/classes: 14x14A, 14x14B, 14x14C, 15x15A, 16x16A, 17x17A, 17x17B, 17x17C, 6x6A, 6x6B, 6x6C, 8x8E, 9x9A, 9x9B, 9x9C, 9x9D, 9x9E, 9x9F, 10x10A, 10x10B and 10x10C. The applicant increased the design initial uranium masses for consistency between the certificate of compliance and the values used in the shielding analyses.

The fuel assembly dimensions important to criticality safety are included in Appendix A to the certificate of compliance. The staff concludes that, given the bounding fuel assembly dimensions defined the certificate, increases to the initial uranium mass will not affect the overall reactivity of the system.

## Revisions to Fuel Assembly Parameter Limits

The applicant requested a revision to the fuel assembly parameter limits for the following fuel assembly array/ classes: 14x14C, 6x6A, 6x6B, 7x7A, 7x7B, 8x8A, 8x8B, 8x8D, 9x9B, 9x9E, 9x9F, and 10x10C. The revised fuel parameters are provided in SAR Table 6.2.1 for BWR assemblies and SAR Table 6.2.2 for PWR assemblies. The analysis and model of the packaging are similar to those used previously by the applicant. The changes to each assembly type were modeled explicitly. Revised results are documented in Chapter 6 of the SAR. The applicant showed that these revised fuel assembly parameter limits do not change the bounding fuel assembly array/class for the BWR and PWR assemblies.

The staff reviewed the revised fuel specifications considered in the criticality analyses and performed independent confirmatory analyses using explicit models. The staff calculated $k_{eff}$ values comparable to the applicant's results.

## Addition of two new fuel assembly classes, 15x15H and 8x8F

The applicant requested the addition of two new fuel assemblies to the list of permissible contents in the HI-STAR 100 transportation package. Characteristics of the 8x8F and 15x15H assemblies are presented in SAR Tables 6.2.1 and 6.2.2, respectively. The 8x8F array/class includes a cruciform shaped water rod that separates the 64 fuel pins into quadrants. The applicant modeled each of these fuel assemblies explicitly. For the 8x8F, water channels were appropriately included in the model. The applicant calculated a $k_{eff}$ of 0.9153 for the 8x8F assembly and $k_{eff}$ of 0.9411 for the 15x15H.

Staff verified that all fuel assembly parameters important to criticality safety have been included in Appendix A to the certificate of compliance. For its confirmatory analyses, the staff explicitly modeled the two fuel assemblies within the packaging. The staff calculated $k_{eff}$ values comparable to the applicant's results.

**Criticality Evaluation Summary**

The applicant performed all criticality analyses using MCNP4a, a three-dimensional, continuous-energy, Monte Carlo N-Particle code. The MCNP4a calculations used the continuous-energy cross section data distributed with the code. This cross-section data is based on ENDF/B-V cross-section library.

The staff agrees that the codes and cross-section sets used in the analysis are appropriate for this application and fuel system. The staff performed its independent criticality analyses using the CSAS/KENO-Va codes and the 44-group cross-section library in the SCALE 4.3 system.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the changes to the packaging and the contents of the HI-STAR 100 System do not affect the ability of the package to meet the criticality safety requirements of 10 CFR Part 71.

## 7.0 OPERATING PROCEDURES

An evaluation of the operating procedures is not necessary. The requested changes do not result in a significant change to the operation of the package.

## 8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

An evaluation of the acceptance tests and maintenance program is not necessary. The requested changes do not result in a significant change to the package's acceptance testing and maintenance program.

## CONCLUSION

The staff has reviewed the requested amendment. Based on the statements and representations in the application, as supplemented, and Revision 9 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. Certificate of Compliance No. 9261 for the HI-STAR 100 System has been amended as requested by Holtec International.

Issued with Certificate of Compliance No. 9261, Revision 1, on May 11, 2000 , 2000.