NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9261	0	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 application dated October 23, 1995, as supplemented

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 96 inches at the cask body and approximately 128 inches at the impact limiters. Its height 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 280,000 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 storing 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. An 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.

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.

Drawings

The package shall be constructed and assembled in accordance with the drawings listed below which are found in Section 1.4 of Revision 8 of the Holtec HI-STAR 100 Safety Analysis Report (SAR, Rev. 8).

(a) Drawing 1395, Sheet 1, Revision 10
Sheets 2-3, Revision 9
Sheet 4, Revision 8

(b) Drawing 1396, Sheet 1, Revision 12
Sheets 2-3, Revision 9
Sheets 4-5, Revision 8
Sheet 6, Revision 7

(c) Drawing 1397, Sheet 1, Revision 14
Sheets 2-3, Revision 10
Sheets 4, Revision 11
Sheets 5-7, Revision 8

(d) Drawing 1398, Sheet 1, Revision 12
Sheet 2, Revision 9
Sheet 3, Revision 8

(e) Drawing 1399, Sheet 1, Revision 10
Sheet 2, Revision 8
Sheet 3, Revision 9

(f) Drawing 1401, Sheet 1, Revision 11
Sheets 2 and 4, Revision 8
Sheet 3, Revision 9

(g) Drawing 1402, Sheet 1, Revision 13
Sheets 2-3, Revision 11
Sheets 4-5, Revision 9
Sheet 6, Revision 7

(h) Drawing 1765, Sheet 1, Revision 12
Sheet 2, Revision 9
Sheet 3, Revision 5
Sheet 4, Revision 10
Sheets 5 and 7, Revision 4
Sheet 6, Revision 1

(i) Drawing 1782, Revision 1

(j) Drawing 1783, Revision 1

(k) Drawing 1784, Revision 0

(l) Drawing BM-1476, Sheet 1, Revision 12
Sheet 2, Revision 13

(m) Drawing BM-1478, Sheet 1, Revision 9
Sheet 2, Revision 11

(n) Drawing BM-1479, Sheet 1, Revision 10
Sheet 2, Revision 13

(o) Drawing BM-1819, Revision 1

5.b. Contents of Packaging

(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 items 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. A damaged fuel assembly's inability to be handled by normal means may be due to mechanical damage and must not be due to fuel rod cladding damage.

Damaged Fuel Containers 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.

Fuel Debris refers to ruptured fuel rods, severed rods, and loose fuel pellets 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 simple average of the distributed fuel rod 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.

(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.

5.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. of the CoC.

(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 4.3×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} std 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 1×10^{-3} std 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.

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than 5×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} std 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: 0.1212 g-moles/l (+0,-10%) for the MPC-24 and 0.1218 g-moles/l (+0,-10%) 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 overpack shall be evacuated to 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 ± 90**
Overpack Vent and Drain Port Plugs	22 +2/-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

**Tighten closure plate bolts in 5 passes and in a crisscross pattern.

(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 in accordance with ANSI N14.6.

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

(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 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 depth 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 shield 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) Each 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 8. 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, 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 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 280,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at 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:

The drawings specified in this certificate 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 8.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: March 31, 1999

ENCLOSURE 1

Appendix A - Certificate of Compliance 9261

INDEX TO APPENDIX A

INDEX TO APPENDIX A (Cont'd)

Table A.1
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. Initial Maximum Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum enrichment per assembly	
i. Zr Clad:	An assembly post-irradiation cooling time, average burnup, and minimum 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 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 (continued)
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, and minimum enrichment per assembly:	
i. Zr clad:	An assembly post-irradiation cooling time, average burnup, and minimum enrichment as specified in Table A.7, except for 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.
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 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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.
c. 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

B. Quantity per MPC: 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.

Table A.1 (continued)
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 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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	A post-irradiation cooling time after discharge $\geq$ 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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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 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	A post-irradiation cooling time after discharge $\geq$ 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. Original Fuel assembly length	$\leq$ 135.0 inches (nominal design)
f. Original Fuel assembly width	$\leq$ 4.70 inches (nominal design)
g. Fuel Debris Weight	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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 after discharge $\geq$ 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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

5. 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 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:	A post-irradiation cooling time after discharge $\geq$ 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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	A post-irradiation cooling time after discharge $\geq$ 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. Original fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Original fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel debris weight:	$\leq$ 400 lbs, including channels

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; or
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

Table A.2
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 Uranium (kg/assy.) (Note 4)	≤ 402	≤ 402	≤ 410	≤ 400	≤ 420
Initial Maximum 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	≤ 0.3840	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3770	≤ 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 3)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.040	≥ 0.0145	≥ 0.0165

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. Each guide tube replaces 4 fuel rods.
4. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total initial uranium weight may be up to 2.0 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.2 (continued)
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 Maximum 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

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 weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total initial uranium weight may be up to 2.0 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.2 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 4)	≤ 420	≤ 430	≤ 450	≤ 464	≤ 460
Initial Maximum Enrichment (wt % ^{235}U)	≤ 4.0	≤ 4.6	≤ 4.0	≤ 4.0	≤ 4.0
No. of Fuel Rods	204	236	264	264	264
Clad O.D. (in.)	≥ 0.422	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Clad I.D. (in.)	≤ 0.3890	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Pellet Dia. (in.)	≤ 0.3825	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	0.563	0.506	0.496	0.496	0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	5 (Note 3)	25	25	25
Guide Tube Thickness (in.)	≥ 0.0145	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

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. Each guide tube replaces four fuel rods.
4. Design Initial Uranium Weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total initial uranium weight may be up to 2.0 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.3
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 Uranium (kg/assy.) (Note 4)	$\leq$ 108	$\leq$ 108	$\leq$ 108	$\leq$ 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 3 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	$\leq$ 4.0	$\leq$ 5.0	$\leq$ 4.0
No. of Fuel Rods	36	36 (up to 9 MOX rods)	36	49	49	64
Clad O.D. (in.)	$\geq$ 0.5550	$\geq$ 0.5625	$\geq$ 0.5630	$\geq$ 0.4860	$\geq$ 0.5630	$\geq$ 0.4120
Clad I.D. (in.)	$\leq$ 0.4945	$\leq$ 0.4945	$\leq$ 0.4990	$\leq$ 0.4200	$\leq$ 0.4990	$\leq$ 0.3620
Pellet Dia. (in.)	$\leq$ 0.4940	$\leq$ 0.4820	$\leq$ 0.4880	$\leq$ 0.4110	$\leq$ 0.4880	$\leq$ 0.3580
Fuel Rod Pitch (in.)	0.694	0.694	0.740	0.631	0.738	0.523
Active Fuel Length (in.)	$\leq$ 110	$\leq$ 110	$\leq$ 77.5	$\leq$ 79	$\leq$ 150	$\leq$ 110
No. of Water Rods	0	0	0	0	0	0
Water Rod Thickness (in.)	N/A	N/A	N/A	N/A	N/A	N/A
Channel Thickness (in.)	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.120	$\leq$ 0.100

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. $\leq$ 0.612 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu).
4. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**8x8B**	**8x8C**	**8x8D**	**8x8E**	**9x9A**	**9x9B**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)(Note 6)	$\leq$ 185	$\leq$ 185	$\leq$ 185	$\leq$ 180	$\leq$ 173	$\leq$ 173
Maximum planar-average initial enrichment (wt.% ^{235}U)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rods	63	62	60	59	74/66(Note 3)	72
Clad O.D. (in.)	$\geq$ 0.4840	$\geq$ 0.4830	$\geq$ 0.4830	$\geq$ 0.4930	$\geq$ 0.4400	$\geq$ 0.4330
Clad I.D. (in.)	$\leq$ 0.4250	$\leq$ 0.4250	$\leq$ 0.4190	$\leq$ 0.4250	$\leq$ 0.3840	$\leq$ 0.3810
Pellet Dia. (in.)	$\leq$ 0.4160	$\leq$ 0.4160	$\leq$ 0.4110	$\leq$ 0.4160	$\leq$ 0.3760	$\leq$ 0.3740
Fuel Rod Pitch (in.)	0.636 - 0.641	0.636 - 0.641	0.640	0.640	0.566	0.569
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods	1	2	1 - 4 (Note 5)	5	2	1 (Note 4)
Water Rod Thickness (in.)	$\geq$ 0.034	> 0.00	> 0.00	$\geq$ 0.034	> 0.00	> 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.100	$\leq$ 0.120	$\leq$ 0.120

Notes: 1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 74 total rods, 66 full length rods, and 8 partial length rods.

4. Square, replacing nine fuel rods.

5. Variable

6. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9C**	**9x9D**	**9x9E**	**9x9F**	**10x10A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 4)	≤ 173	≤ 170	≤ 170	≤ 170	≤ 182
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 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 3)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3590	≤ 0.3810	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3525	≤ 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	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	≥ 0.0305	≥ 0.0305	≥ 0.0305	≥ 0.0300
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.120

Notes: I nitial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 92 total fuel rods, 78 full length rods, and 14 partial length rods.

4. Design Initial Uranium Weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.

Table A.3 (continued)
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 6)	$\leq$ 182	$\leq$ 180	$\leq$ 125	$\leq$ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.0	$\leq$ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rods	91/83 (Note 3)	96	100	96
Clad O.D. (in.)	$\geq$ 0.3957	$\geq$ 0.3790	$\geq$ 0.3960	$\geq$ 0.3940
Clad I.D. (in.)	$\leq$ 0.3480	$\leq$ 0.3294	$\leq$ 0.3560	$\leq$ 0.3500
Pellet Dia. (in.)	$\leq$ 0.3420	$\leq$ 0.3224	$\leq$ 0.3500	$\leq$ 0.3430
Fuel Rod Pitch (in.)	0.510	0.488	0.565	0.557
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 83	$\leq$ 83
No. of Water Rods	1 (Note 4)	5 (Note 5)	0	4
Water Rod Thickness (in.)	> 0.00	$\geq$ 0.034	N/A	$\geq$ 0.022
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.055	$\leq$ 0.080	$\leq$ 0.080

Notes: 1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 91 total fuel rod, 83 full length rods, and 8 partial length rods.

4. Square, replacing nine fuel rods.

5. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

6. Design Initial Uranium Weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.

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

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

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY AND
WITH ZIRCALOY IN-CORE GRID SPACERS

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

Table A.6

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

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

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68

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