December 21, 2000

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

SUBJECT: CERTIFICATE OF COMPLIANCE FOR AMENDMENT 1 TO THE HOLTEC INTERNATIONAL, HI-STAR 100 CASK STORAGE SYSTEM TO ACCOMMODATE REVISED CASK CONTENTS

Dear Mr. Gutherman:

As requested by your application dated November 24, 1999, as supplemented, enclosed is Amendment 1 to Certificate of Compliance No. 1008 for the Holtec International (Holtec) HI-STAR 100 Cask System to incorporate revised contents. This certificate is issued pursuant to 10 CFR Part 72. As stated in the Federal Register (65 FR 60339; October 11, 2000), the effective date of this certificate is December 26, 2000. The staff's Safety Evaluation Report is also enclosed. We request that you update and submit the final safety analysis report to conform to the certificate in accordance with 10 CFR 72.248.

Changes made to the enclosed certificate are indicated by vertical lines in the margin.

This certificate constitutes the approval and conditions for use of the Holtec HI-STAR 100 Cask System for storage of spent nuclear fuel under the general licensing provisions of 10 CFR 72.210. A general license has been granted to all holders of licenses for nuclear power reactors issued under 10 CFR Part 50.

If you have any questions regarding this certificate, please contact me or Mr. Christopher Jackson of my staff at 301-415-8500.

Sincerely,
/RA/ original signed by /s/
E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1008
TAC No. L23116

Enclosures: 1. Certificate of Compliance No. 1008
2. Safety Evaluation Report

Distribution: w/o encls* (closes TAC L23116)

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DATE	12/21/00		12/21/00		12/21/00		12/21/00		12/21/00	

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ENCLOSURE 1
CERTIFICATE OF COMPLIANCE NO. 1008

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1008	10/04/99	10/04/19	72-1008	1	12/26/00	USA/72-1008

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International Inc., Final Safety Analysis Report for the HI-STAR 100 Cask System
Docket No. 72-1008

CONDITIONS

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions specified below:

1. CASK

   The HI-STAR 100 Cask System is certified as described in the Final Safety Analysis Report (SAR) and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance. It is designed for both storage and transfer of irradiated nuclear fuel.

   a. Model No.: HI-STAR 100 (MPC-24, MPC-68, MPC-68F)

      The HI-STAR 100 Cask System is comprised of the multi-purpose canister (MPC), which contains the fuel, and the overpack, which contains the 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. The MPC-68 is designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 containing fuel assemblies with known or suspected defects, such as ruptured fuel rods, severed rods, loose fuel pellets, or which cannot be handled by normal means due to fuel cladding damage, is designated as MPC-68F. The MPC-24 and the MPC-68 (including the MPC-68F) are identical in external dimensions and will fit into the same overpack design.

1. b. Description

The complete HI-STAR 100 Cask System for storage of spent nuclear fuel is comprised of two discrete components: the MPC and the storage/transport overpack. The HI-STAR 100 CASK System consists of interchangeable MPCs which constitute the confinement boundary for BWR or PWR spent nuclear fuel and an overpack which provides the helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. All MPCs have identical exterior dimensions which render them interchangeable. A single overpack design is provided which is capable of storing each type of MPC.

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring. The outer diameter and cylindrical height of each MPC is fixed. However, the number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. The MPC provides the confinement boundary for the stored fuel. The confinement boundary is a seal-welded enclosure constructed entirely of a stainless steel alloy. The inner surfaces of the HI-STAR 100 overpack form an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate shells of gamma shielding.

The fuel transfer and auxiliary equipment necessary for Independent Spent Fuel Storage Installation operation are not included as part of the HI-STAR 100 Cask System reviewed for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Such equipment may include, but is not limited to, special lifting devices, transfer trailers or equipment, and vacuum drying/helium leak test equipment.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the SAR.

3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the SAR.

4. QUALITY ASSURANCE

Activities in the areas of design, procurement, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

Certificate No. 1008
Amendment No. 1

5. HEAVY LOADS REQUIREMENTS

   Each lift of a HI-STAR 100 spent fuel cask must be made in accordance with the existing heavy loads requirements and procedures of the licensed facility in which the lift is made. A plant-specific safety review (in accordance with 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant-specific heavy loads requirements.

6. APPROVED CONTENTS

   Contents of the HI-STAR 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

7. APPROVED DESIGN FEATURES

   Features or characteristics for the site or cask must be in accordance with Appendix B to this certificate.

8. CHANGES TO THE CERTIFICATE OF COMPLIANCE

   The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

9. AUTHORIZATION

   The HI-STAR 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION
/RA/ original signed by /s/
E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Attachments:
1. Appendix A
2. Appendix B

CERTIFICATE OF COMPLIANCE NO. 1008

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STAR 100 CASK SYSTEM

AMENDMENT 1

TABLE OF CONTENTS

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REVISION HISTORY

Amendment	Section	Change Description
1	Throughout	Editorial changes and typographical corrections.
	1.1	Revised definitions of DAMAGED FUEL ASSEMBLY and DAMAGED FUEL CONTAINER.
	2.1.1	Replaced the MPC helium backfill density limit with a helium backfill pressure limit.
		Revised the leak rate units in Table 2-1 from std cc/sec to atm-cc/sec.

1.0 USE AND APPLICATION

1.1 Definitions

--Note --
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a 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 CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STAR 100 design are shown in Figures 2.1.1 and 2.1.2 of the Final Safety Analysis Report (SAR) for the HI-STAR 100 Cask System.
FUEL BUILDING	The FUEL BUILDING is the site-specific power plant facility, licensed pursuant to 10 CFR Part 50, where the loaded OVERPACK is transferred to or from the transporter.
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

(continued)

1.1 Definitions (continued)

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within the perimeter fence licensed for storage of spent fuel within SFSCs. (See also 10 CFR 72.3)
INTACT FUEL ASSEMBLY	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).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an SFSC while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the SFSC and end when the SFSC is suspended from or secured on the transporter.
MULTI-PURPOSE CANISTER (MPC)	MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
OVERPACK	OVERPACKs are the casks which receive and contain the sealed MPCs. They provide the helium retention boundary, gamma and neutron shielding, and a set each of lifting and pocket trunnions for handling.
PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR-AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

(continued)

1.1 Definitions (continued)

SPENT FUEL STORAGE CASKS (SFSCs)	SFSCs are storage containers approved for casks of spent fuel assemblies at the ISFSI. The HI-STAR 100 SFSC System consists of the OVERPACK and its integral MPC.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while an SFSC containing spent fuel is sitting on a storage pad within the ISFSI perimeter.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an SFSC loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the SFSC is first suspended from or secured on the transporter and end when the SFSC is at its destination and no longer secured on or suspended from the transporter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	1. Stop . . . <u>OR</u> A.2.1 Verify . . . <u>AND</u> A.2.2.1 Reduce . . . <u>OR</u> A.2.2.2 Perform . . . <u>OR</u> A.3 Remove. . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector <u>OR</u> and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector <u>AND</u>. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector <u>OR</u> indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, provided that the SFSC is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the SFSC is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

--Note --

When “Immediately” is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	AND	
	B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit.	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1. AND B.2 Complete action B.2.	12 hours 36 hours

When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS
--------------------------------------NOTE--
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1 AND B.2 Complete action B.2	6 hours 12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 2.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 2.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 2.0.4 imposes no restriction.

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 2.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 2.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 2.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 2.0.1.

If the interval as specified by SR 2.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 2.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 2.0.4

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify vacuum drying pressure is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 2.0.2.

"Thereafter" indicates future performances must be established per SR 2.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 2.0.1 LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 2.0.2.

LCO 2.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 2.0.5.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 2.0.3 Not applicable.

LCO 2.0.4 When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.

LCO 2.0.5 Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 2.0.2 for the system returned to service under administrative control to perform the testing.

2.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 2.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 2.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.

SR 2.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply. If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 2.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

2.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY (continued)

SR 2.0.4 Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

2.1 SFSC INTEGRITY

2.1.1 Multipurpose Canister (MPC)

LCO 2.1.1 The MPC shall be dry and helium filled.

APPLICABILITY: During TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

--- NOTE ---

Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u>	
	A.2 Determine and complete corrective actions necessary to return the MPC to an analyzed condition.	30 days
B. MPC helium backfill pressure limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential.	72 hours
	<u>AND</u>	
	B.2 Determine and complete corrective actions necessary to return the MPC to an analyzed condition.	14 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. MPC helium leak rate limit not met.	C.1 Perform an engineering evaluation to determine impact of increased helium leak rate on heat removal capability and offsite dose release effects.	24 hours
	AND	
	C.2 Determine and complete corrective actions necessary to return MPC to an analyzed condition.	7 days
D. Required Actions and associated Completion Times not met.	D.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 2.1.1.1	Verify MPC cavity vacuum drying pressure is within the limit specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS
SR 2.1.1.2	Verify MPC helium backfill pressure is within the limit specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS
SR 2.1.1.3	Verify that the total helium leak rate through the MPC lid confinement weld and the drain and vent port confinement welds is within the limit specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS

Table 2-1
MPC Model-Dependent Limits

MPC MODEL	LIMITS
1. MPC-24	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. OVERPACK Annulus Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
c. MPC Helium Backfill Pressure[1]	≤ 22.2 psig
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 atm cc/sec (He)
2. MPC-68	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. OVERPACK Annulus Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
c. MPC Helium Backfill Pressure[1]	≤ 28.5 psig
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 atm cc/sec (He)
3. MPC-68F	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. OVERPACK Annulus Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
c. MPC Helium Backfill Pressure[1]	≤ 28.5 psig
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 atm cc/sec (He)

[1] Helium used for backfill of MPC shall have a purity of ≥99.995%

2.1 SFSC INTEGRITY

2.1.2 OVERPACK

LCO 2.1.2 The OVERPACK shall be dry and helium filled.

APPLICABILITY: TRANSPORT OPERATIONS and STORAGE OPERATIONS

ACTIONS

-- NOTE --

Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. OVERPACK annulus vacuum drying pressure limit not met.	A.1 Perform an engineering evaluation to determine quantity of moisture left in OVERPACK.	7 days
	<u>AND</u>	
	A.2 Determine and complete corrective actions necessary to return OVERPACK to analyzed condition.	30 days
B. OVERPACK annulus helium backfill pressure limit not met	B.1 Perform an engineering evaluation to determine impact of helium pressure differential.	72 hours
	<u>AND</u>	
	B.2 Determine and complete corrective actions necessary to return the OVERPACK to analyzed condition.	30 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. OVERPACK helium leak rate limit not met.	C.1 Perform an engineering evaluation to determine impact of increased helium leak rate on heat removal capability and off-site dose release effects.	7 days
	AND	
	C.2 Determine and complete corrective actions necessary to return OVERPACK to analyzed condition.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 2.1.2.1	Verify OVERPACK annulus vacuum drying pressure is within limit specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS
SR 2.1.2.2	Verify OVERPACK annulus helium backfill pressure is within the limit specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS
SR 2.1.2.3	Verify that the total helium leak rate through the OVERPACK closure plate inner mechanical seal, the OVERPACK vent port plug seal, and the OVERPACK drain port plug seal is within the limits specified in Table 2-1 for the applicable MPC model.	During LOADING OPERATIONS

2.1 SFSC INTEGRITY

2.1.3 SFSC Lifting Requirements

LCO 2.1.3 An OVERPACK loaded with spent fuel shall be lifted in accordance with either of the following requirements

a. i. A lift height $\leq$ 21 inches when oriented vertically.

AND

ii. A lift height $\leq$ 72 inches when oriented horizontally.

OR

b. The OVERPACK is lifted with lifting devices designed in accordance with ANSI N14.6 and having redundant drop prevention design features.

OR

c. Site-specific analysis to evaluate site-specific conditions to ensure that the drop accidents impact loads remain within HI-STAR 100 SAR limits of 60g.

APPLICABILITY: During TRANSPORT OPERATIONS.

--- NOTE ---
This LCO is not applicable when the SFSC is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.)

ACTIONS

--- NOTE ---
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC lifting requirements not met.	A.1 Initiate actions to meet SFSC lifting requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 2.1.3.1 Verify SFSC lifting requirements are met.	After the SFSC is suspended from, or secured in the transporter and prior to the transporter beginning to move the SFSC within ISFSI

2.1 SFSC INTEGRITY

2.1.4 Fuel Cool-Down

LCO 2.1.4 The MPC exit gas temperature shall be $\leq 200^{o}$ F.

--- NOTE ---
The LCO is only applicable to wet UNLOADING OPERATIONS.

APPLICABILITY: UNLOADING OPERATIONS prior to flooding.

ACTIONS
--- NOTE ---
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC exit temperature not within limit.	A.1 Establish MPC helium gas exit temperature within limit.	Prior to initiating MPC re-flooding operations
	AND	
	A.2 Ensure adequate heat transfer from MPC to the environment.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 2.1.4.1 Verify MPC helium gas exit temperature within limit.	Prior to initiation of MPC re-flooding operations.

2.2 SFSC RADIATION PROTECTION

2.2.1 OVERPACK Average Surface Dose Rates

LCO 2.2.1 The average surface dose rates of each OVERPACK shall not exceed:

a. 125 mrem/hour (neutron + gamma) on the side;

b. 80 mrem/hour (neutron + gamma) on the top;

APPLICABILITY: TRANSPORT OPERATIONS and STORAGE OPERATIONS

ACTIONS

-- NOTE --
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. OVERPACK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading.	24 hours
	<u>AND</u>	
	A.2 Perform written evaluations to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.	48 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 2.2.1.1	Verify average surface dose rates of OVERPACK containing fuel assemblies are within limits. OVERPACK dose rates shall be measured at locations shown in Figure 2.2.1-1.	During LOADING OPERATIONS
NOTE:	SR 2.2.1.1 shall be performed after the MPC has been vacuum dried.	
NOTE:	If a loaded OVERPACK is placed into storage after transport from an on off-site location, SR 2.2.1.1 shall be performed after receipt of the OVERPACK and prior to STORAGE OPERATIONS.	

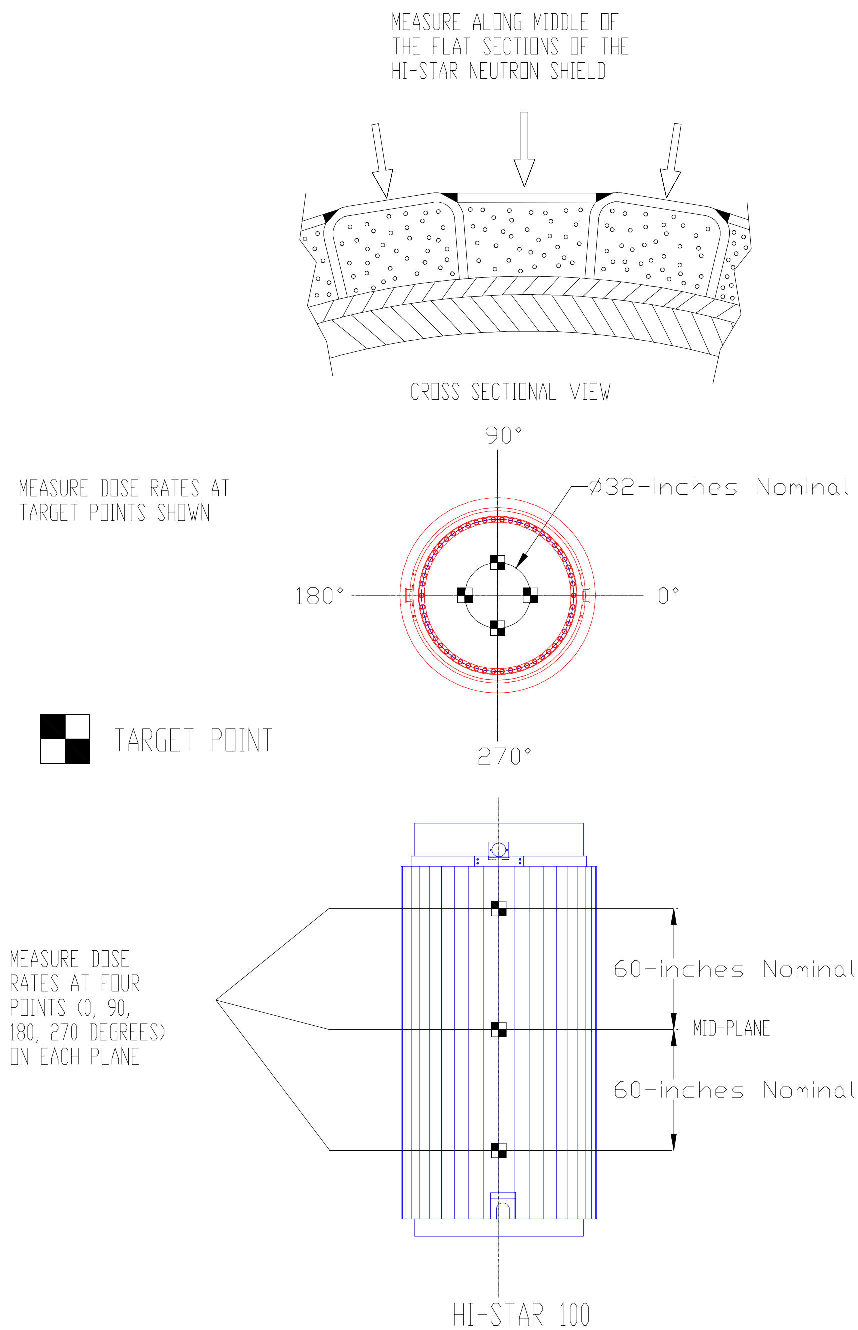

Figure 2.2.1-1
OVERPACK Surface Rate Measurement Locations

2.2 SFSC RADIATION PROTECTION

2.2.2 SFSC Surface Contamination

LCO 2.2.2 — Removable contamination on the exterior surfaces of the OVERPACK and accessible portions of the MPC shall each not exceed:

a. 1000 dpm/100 cm^2 from beta and gamma sources; and

b. 20 dpm/100 cm^2 from alpha sources.

APPLICABILITY: TRANSPORT OPERATIONS and STORAGE OPERATIONS

ACTIONS

----- NOTE -----
Separate Condition entry is allowed for each SSSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC removable surface contamination limits not met.	A.1 Restore SFSC removable surface contamination to within limits.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 2.2.2.1	Verify that the removable contamination on the exterior surfaces of OVERPACKs and accessible portions of the MPC containing fuel is within limits.	During LOADING OPERATIONS
NOTE:	If a loaded OVERPACK is placed into storage after transport from an off-site location, SR 2.2.2.1 shall be performed after receipt of the OVERPACK and prior to STORAGE OPERATIONS.	

3.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

The following programs shall be established, implemented, and maintained.

3.1 Training Program

A training program for the HI-STAR 100 cask system shall be developed under the general licensee's systematic approach to training (SAT). Training modules shall include comprehensive instructions for the operation and maintenance of the HI-STAR 100 spent fuel storage cask system and the independent spent fuel storage installation (ISFSI).

3.2 Pre-Operational Testing and Training Exercise

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STAR 100 system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC/OVERPACK. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the HI-STAR 100 MPC/OVERPACK into the spent fuel pool.

b. Preparation of the HI-STAR 100 Cask System for fuel loading.

c. Selection and verification of specific fuel assemblies to ensure type conformance.

Locating specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

e. Remote installation of the MPC lid and removal of HI-STAR 100 MPC/OVERPACK from the spent fuel pool.

f. MPC welding, NDE inspections, hydrostatic testing, draining, vacuum drying, helium backfilling, and leakage testing.

g. HI-STAR 100 OVERPACK closure, draining, vacuum drying, helium backfilling and leakage testing.

h. HI-STAR 100 OVERPACK upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site’s cask handling arrangement.

i. Placement of the HI-STAR 100 Cask System at the ISFSI.

j. HI-STAR 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds.

3.3 Special Requirements For First Systems In Place

The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STAR 100 SFSC systems (MPC-24 and MPC-68) placed into service with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the SAR.

Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW. (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.

Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.

3.4. Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

a. The HI-STAR 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 2.1.1, Multi-Purpose Canister (MPC), provides assurance that there are no radioactive effluents from the SFSC.

b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring program for 10 CFR Part 50 operations.

c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

CERTIFICATE OF COMPLIANCE NO. 1008

APPENDIX B

APPROVED CONTENTS AND DESIGN FEATURES

FOR THE HI-STAR 100 CASK SYSTEM

AMENDMENT 1

TABLE OF CONTENTS

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REVISION HISTORY

Amendment	Section	Change Description
1	Throughout	Editorial changes and typographical corrections.
	1.0	Revised definitions of DAMAGED FUEL ASSEMBLY, DAMAGED FUEL CONTAINER, and PLANAR-AVERAGE INITIAL ENRICHMENT.
	1.1	Revised Section 1.1.1 to permit storage of certain non-fuel hardware.
	1.4	Revised Item 6 to clarify the requirements for cask storage pad.
	1.5	Revised Section 1.5.2 to replace the term "painted surface of" with "paint used on."
	Table 1.1-1	Added limits for non-fuel hardware (BPRAs and TPDs), array class 8x8F, and Thoria Rods.
	Table 1.1-2	Revised certain fuel assembly parameters, added array/class 15x15H fuel assembly, and added clarifying notes.
	Table 1.1-3	Revised certain fuel assembly parameters, added array/class 8x8F fuel assembly, and added clarifying notes.
	Tables 1.1-4 and 1.1-5	Revised to clarify limits, reflect addition of BPRAs and TPDs, and permit linear interpolation between points.
	Table 1.1-6	Added table specifying cooling and average burnup limits for non-fuel hardware.
	Table 1.3-1	Added exception to ASME Code NB-5230 for the MPC lid-to-shell weld and added clarifying text.

APPENDIX B DESIGN FEATURES

1.0 Definitions

--Note --
The defined terms of this section appear in capitalized type and are applicable throughout this Appendix.

Term	Definition
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a 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 CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STAR 100 design are shown in Figures 2.1.1 and 2.1.2 of the Final Safety Analysis Report (SAR) for the HI-STAR 100 Cask System.
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.
INTACT FUEL ASSEMBLY	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).
PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR-AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

1.1 Fuel Specifications

1.1.1 Fuel To Be Stored In The HI-STAR 100 SFSC

1. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and certain non-fuel hardware meeting the limits specified in Table 1.1-1 (which refers to Tables 1.1-2 through 1.1-6) may be stored in the HI-STAR 100 SFSC System.

2. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the maximum decay heat generation limit for the stainless steel clad fuel assemblies.

3. For MPCs partially loaded with DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, all remaining Zircaloy clad INTACT FUEL ASSEMBLIES in the MPC shall meet the maximum decay heat generation limits for the DAMAGED FUEL ASSEMBLIES.

4. For MPC-68's 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 maximum decay heat generation limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies.

1.1.2 Preferential Fuel Loading

Preferential fuel loading shall be used whenever fuel assemblies with significantly different post-irradiation cooling times (equal to or greater than one year) are to be loaded in the same MPC. That is, fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket.

1.2 Functional and Operating Limits Violations

If any Fuel Specifications defined in Section 1.1 are violated, the following actions shall be completed:

1. The affected fuel assemblies shall be placed in a safe condition without delay and in a controlled manner.

2. Within 24 hours, notify the NRC Operations Center.

3. Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

The above actions are not a substitute for the reporting requirements ontained in 10 CFR 72.75.

1.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing Code for the HI-STAR 100 Cask System, as clarified in Specification 1.3.1 below.

1.3.1 Exceptions to Codes, Standards, and Criteria

Table1.3-1 lists approved exceptions to the ASME Code for the design of the HI-STAR 100 Cask System.

1.3.2 Construction/Fabrication Exceptions to Codes, Standards, and Criteria

Proposed alternatives to the ASME Code, Section III, 1995 Edition with Addenda through 1997 including exceptions allowed by Specification 1.3.1 may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2. Compliance with the specified requirements of the ASME Code, Section III, 1995 Edition with Addenda through 1997, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

   Requests for exceptions shall be submitted in accordance with 10 CFR 72.4

1.4 Site Specific Parameters and Analyses

Site-specific parameters and analyses that need verification by the system user are, as a minimum, as follows:

1. The temperature of 80^{o} F is the maximum allowed average yearly temperature.

2. The allowed temperature extremes, averaged over a three day period, shall be greater than -40^{o} F and less than 125^{o} F.

3. The horizontal and vertical seismic acceleration levels are bounded by the values listed below in Table 1-4.

Table 1-4

Design-Basis Earthquake Input on the Top Surface of an ISFSI Pad

Horizontal g-Level in Each of Two Orthogonal Directions	Horizontal g-Level Vector Sum	Corresponding Vertical g-Level (Upward)
0.222 g	0.314 g	1.00 x 0.222 g = 0.222 g
0.235 g	0.332 g	0.75 x 0.235 g = 0.176 g
0.24 g	0.339 g	0.667 x 0.24 g = 0.160 g
0.25 g	0.354 g	0.500 x 0.25 g = 0.125 g

4. The analyzed flood condition of 13 fps water velocity and a height of 656 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of combustible transporter fuel.

6. In addition to the requirements of 10CFR72.212(b)(2)(ii), the cask storage pads and foundation shall include the following characteristics as applicable to the drop and tipover analyses:

   a. Concrete Thickness: $\leq$ 36 inches

   b. Concrete Compressive Strength: $\leq$ 4,200 psi at 28 days

   c. Reinforcement top and bottom (both directions):
      Reinforcement area and spacing determined by analysis
      Reinforcement shall be 60 ksi yield strength ASTM material

   d. Soil Effective Modulus of Elasticity: $\leq$ 28,000 psi
      (measured prior to installation of ISFSI)

   An acceptable method of defining the soil effective modulus of elasticity applicable to the drop and tipover analyses is provided in Table 13 of NUREG/CR-6608 with soil classification in accordance with ASTM D2487-93, Standard Classification of Soils for Engineering Purposes (Unified Soil Classification System, USCS), and density determination in accordance with ASTM D1586-84, Standard Test Method for Penetration Test and Split/Barrel Sampling of Soils.

7. In cases where engineered features (i.e., berms, shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

1.5 Design Specifications

1.5.1 Specifications Important for Criticality Control

1.5.1.1 MPC-24

1. Minimum flux trap size: 1.09 in.

2. Minimum ^{10}B loading in the Boral neutron absorbers: 0.0267 g/cm^2

1.5.1.2 MPC-68 and MPC-68F

1. Minimum fuel cell pitch: 6.43 in.

2. Minimum ^{10}B loading in the Boral neutron absorbers: 0.0372 g/cm^2 in the MPC-68, and 0.01 g/cm^2 in the MPC-68F.

1.5.2 Specifications Important for Thermal Performance

1.5.2.1 OVERPACK

The paint used on the HI-STAR 100 OVERPACK must have an emissivity no less than 0.85.

Table 1.1-1 (Page 1 of 16)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES, with or without Burnable Poison Rods (BPRAs) or Thimble Plug Devices (TPDs) listed in Table 1.1-2, and meeting the following specifications:

a. Cladding type:	Zircaloy (Zr) or stainless steel (SS) as specified in Table 1.1-2 for the applicable fuel assembly array/class
b. Initial enrichment:	As specified in Table 1.1-2 for the applicable fuel assembly array/class.
c. Decay heat per assembly	
i. Zr Clad:	An assembly decay heat as specified in Table 1.1-4 for the applicable post-irradiation cooling time.
ii. SS Clad	$\leq$ 575 watts
d. Post-irradiation cooling time and average burnup per assembly	
i. Zr clad:	An assembly post-irradiation cooling time and average burnup as specified in Table 1.1-5. BPRA and TPD post-irradiation cooling time and average burnup as specified in Table 1.1.6.
ii. SS clad:	An assembly post-irradiation cooling time $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU. <u>OR</u> An assembly post-irradiation cooling time $\geq$ 15 years and an average burnup $\leq$ 40,000 MWD/MTU.
e. Nominal fuel assembly length:	$\leq$ 176.8 inches
f. Nominal fuel assembly width:	$\leq$ 8.54 inches
g. Fuel assembly weight:	$\leq$ 1,680 lbs (including non-fuel hardware)

Table 1.1-1 (Page 2 of 16)
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

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

C. Fuel assemblies shall not contain control components except as specifically authorized by this certificate of compliance. BPRAs and TPDs are authorized for loading in the HI-STAR 100 System with their associated fuel assemblies provided the burnup and cooling time limits specified in Table 1.1-6 are met.

D. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading into the MPC-24.

Table 1.1-1 (Page 3 of 16)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES listed in Table 1.1-3, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	Zircaloy (Zr) or stainless steel (SS) as specified in Table 1.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 1.1-3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for the applicable fuel assembly array/class.
d. Decay heat per assembly	
i. Zr clad	An assembly decay heat as specified in Table 1.1-4 for the applicable post-irradiation cooling time, except for (1) array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a decay heat $\leq$ 115 watts and (2) array/class 8x8F fuel assemblies, which shall have a decay heat $\leq$ 183.5 watts.
ii. SS clad	$\leq$ 95 watts
e. Post-irradiation cooling time, average burnup per assembly:	
i. Zr clad:	An assembly post-irradiation cooling time and average burnup as specified in Table 1.1-5, except for (1) array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU.
ii. SS clad:	An assembly cooling time after discharge $\geq$ 10 years, an average burnup $\leq$ 22,500 MWD/MTU.
e. Nominal fuel assembly length:	$\leq$ 176.2 inches
f. Nominal fuel assembly width:	$\leq$ 5.85 inches
g. Fuel assembly weight	$\leq$ 700 lbs, including channels

Table 1.1-1 (Page 4 of 16)
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. BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 1.1-3 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 1.1-3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for the applicable fuel assembly array/class.
d. Decay heat per assembly	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
f. Nominal fuel assembly length:	$\leq$ 135.0 inches
g. Nominal fuel assembly width:	$\leq$ 4.70 inches
h. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 5 of 16)
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 1.1-3 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 1.1-3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for fuel assembly array/class 6x6B.
d. Decay heat per assembly	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
f. Nominal fuel assembly length:	$\leq$ 135.0 inches
g. Nominal fuel assembly width:	$\leq$ 4.70 inches
h. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 6 of 16)
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 1.1-3 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 1.1-3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for array/class 6x6B.
d. Decay heat per assembly	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
f. Nominal fuel assembly length:	$\leq$ 135.0 inches
g. Nominal fuel assembly width:	$\leq$ 4.70 inches
h. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 7 of 16)
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 2.1.2A of the SAR) 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. Nominal fuel cladding O.D.:	$\geq$ 0.412 inches
h. Nominal fuel cladding I.D.:	$\leq$ 0.362 inches
i. Nominal fuel pellet O.D.:	$\leq$ 0.358 inches
j. Nominal active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table 1.1-1 (Page 8 of 16)
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 1.1-1 (Page 9 of 16)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without Zircaloy channels. BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 1.1-3 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 1.1-3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for the applicable fuel assembly array/class.
d. Decay heat per assembly	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
f. Nominal fuel assembly length:	$\leq$ 176.2 inches
g. Nominal fuel assembly width:	$\leq$ 5.85 inches
h. Fuel assembly weight	$\leq$ 700 lbs, including channels

Table 1.1-1 (Page 10 of 16)
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. BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 1.1-3 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 1.1-3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for the applicable fuel assembly array/class.
d. Decay heat per assembly	≤ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	A post-irradiation cooling time after discharge ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU.
f. Nominal fuel assembly length:	≤ 135.0 inches
g. Nominal fuel assembly width:	≤ 4.70 inches
h. Fuel assembly weight	≤ 400 lbs, including channels

Table 1.1-1 (Page 11 of 16)
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 1.1-3 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 1.1-3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for the applicable original fuel assembly array/class.
d. Decay heat per DFC:	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	A post-irradiation cooling time after discharge $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU for the original fuel assembly.
f. Nominal original fuel assembly length:	$\leq$ 135.0 inches
g. Nominal original fuel assembly width:	$\leq$ 4.70 inches
h. Fuel debris weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 12 of 16)
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 1.1-3 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 1.1-3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for fuel assembly array/class 6x6B.
d. Decay heat per assembly	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	An assembly post-irradiation cooling time after discharge $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
f. Nominal fuel assembly length:	$\leq$ 135.0 inches
g. Nominal fuel assembly width:	$\leq$ 4.70 inches
h. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 13 of 16)
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 1.1-3 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 1.1-3 for array/class 6x6B.
c.	Initial maximum rod enrichment:	As specified in Table 1.1-3 for array/class 6x6B.
d.	Decay heat per assembly	$\leq$ 115 watts
e.	Post-irradiation cooling time and average burnup per assembly:	A post-irradiation cooling time after discharge $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
f.	Nominal fuel assembly length:	$\leq$ 135.0 inches
g.	Nominal fuel assembly width:	$\leq$ 4.70 inches
h.	Fuel assembly weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 14 of 16)
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 1.1-3 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 1.1-3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table 1.1-3 for original fuel assembly array/class 6x6B.
d. Decay heat per DFC	$\leq$ 115 watts
e. Post-irradiation cooling time and average burnup per assembly:	A post-irradiation cooling time after discharge $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM for the original fuel assembly.
f. Nominal original fuel assembly length:	$\leq$ 135.0 inches
g. Nominal original fuel assembly width:	$\leq$ 4.70 inches
h. Fuel debris weight	$\leq$ 400 lbs, including channels

Table 1.1-1 (Page 15 of 16)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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 2.1.2A of the SAR) 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.	Nominal fuel cladding O.D.:	$\geq$ 0.412 inches
h.	Nominal fuel cladding I.D.:	$\leq$ 0.362 inches
i.	Nominal fuel pellet O.D.:	$\leq$ 0.358 inches
j.	Nominal active fuel length:	$\leq$ 111 inches
k.	Canister weight:	$\leq$ 550 lbs, including fuel

Table 1.1-1 (Page 16 of 16)
Fuel Assembly Limits

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

B. Quantity per MPC:

Up to four (4) DFCs 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 DFCs;
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 1.1-2 (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)	$\leq$ 407	$\leq$ 407	$\leq$ 425	$\leq$ 400	$\leq$ 464
Initial Enrichment (wt % ^{235}U)	$\leq$ 4.6	$\leq$ 4.6	$\leq$ 4.6	$\leq$ 4.0	$\leq$ 4.1
No. of Fuel Rods (Note 5)	179	179	176	180	204
Clad O.D. (in.)	$\geq$ 0.400	$\geq$ 0.417	$\geq$ 0.440	$\geq$ 0.422	$\geq$ 0.418
Clad I.D. (in.)	$\leq$ 0.3514	$\leq$ 0.3734	$\leq$ 0.3880	$\leq$ 0.3890	$\leq$ 0.3660
Pellet Dia. (in.)	$\leq$ 0.3444	$\leq$ 0.3659	$\leq$ 0.3805	$\leq$ 0.3835	$\leq$ 0.3580
Fuel Rod Pitch (in.)	$\leq$ 0.556	$\leq$ 0.556	$\leq$ 0.580	$\leq$ 0.556	$\leq$ 0.550
Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 144	$\leq$ 150
No. of Guide Tubes	17	17	5 (Note 4)	16	21
Guide Tube Thickness (in.)	$\geq$ 0.017	$\geq$ 0.017	$\geq$ 0.038	$\geq$ 0.0145	$\geq$ 0.0165

Table 1.1-2 (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 (Note 5)	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 1.1-2 (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 (Note 5)	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 1.1-2 (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.

5. Missing fuel rods must be replaced with dummy fuel rods that displace an equal or greater amount of water as the original fuel rods.

Table 1.1-3 (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 (Note 14)	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 1.1-3 (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 (Note 14)	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 1.1-3 (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)	$\leq$ 177	$\leq$ 177	$\leq$ 177	$\leq$ 177	$\leq$ 186
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.1	$\leq$ 4.1	$\leq$ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rods (Note 14)	80	79	76	76	92/78 (Note 8)
Clad O.D. (in.)	$\geq$ 0.4230	$\geq$ 0.4240	$\geq$ 0.4170	$\geq$ 0.4430	$\geq$ 0.4040
Clad I.D. (in.)	$\leq$ 0.3640	$\leq$ 0.3640	$\leq$ 0.3640	$\leq$ 0.3860	$\leq$ 0.3520
Pellet Dia. (in.)	$\leq$ 0.3565	$\leq$ 0.3565	$\leq$ 0.3530	$\leq$ 0.3745	$\leq$ 0.3455
Fuel Rod Pitch (in.)	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.510
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 11)	1	2	5	5	2
Water Rod Thickness (in.)	$\geq$ 0.020	$\geq$ 0.0300	$\geq$ 0.0120	$\geq$ 0.0120	$\geq$ 0.0300
Channel Thickness (in.)	$\leq$ 0.100	$\leq$ 0.100	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120

Table 1.1-3 (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 (Note 14)	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 1.1-3 (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 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

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

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

14. Missing fuel rods must be replaced with dummy fuel rods that displace an equal or greater amount of water as the original fuel rods. Storage of 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies with missing fuel rods are permitted provided the assemblies are stored as DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS.

Table 1.1-4

FUEL ASSEMBLY COOLING AND DECAY HEAT GENERATION (Note 1)

Post-Irradiation Cooling Time (years)	MPC-24 PWR Assembly With or Without BPRAs or TPDs Decay Heat (Watts)	MPC-68 BWR Assembly Decay Heat (Watts)
≥ 5	≤792	≤272
≥ 6	≤773	≤261
≥ 7	≤703	≤238
≥ 8	≤698	≤236
≥ 9	≤692	≤234
≥10	≤687	≤232
≥11	≤683	≤231
≥12	≤678	≤229
≥13	≤674	≤228
≥14	≤669	≤227
≥15	≤665	≤226

Note: 1. Linear interpolation between points permitted.

Table 1.1-5

FUEL ASSEMBLY COOLING AND AVERAGE BURNUP (Note 1)

Post-Irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup (Without BPRAs and With or Without TPDs) (MWD/MTU)	MPC-24 PWR Assembly Burnup (With BPRAs) (MWD/MTU)	MPC-68 BWR Assembly Burnup (MWD/MTU)
≥ 5	≤28,700	≤28,300	≤26,000
≥ 6	≤32,700	≤32,300	≤29,100
≥ 7	≤33,300	≤32,700	≤29,600
≥ 8	≤35,500	≤35,000	≤31,400
≥ 9	≤37,000	≤36,500	≤32,800
≥10	≤38,200	≤37,600	≤33,800
≥11	≤39,300	≤38,700	≤34,800
≥12	≤40,100	≤39,500	≤35,500
≥13	≤40,800	≤40,200	≤36,200
≥14	≤41,500	≤40,800	≤36,900
≥15	≤42,100	≤41,400	≤37,600

Note: 1. Linear interpolation between points permitted.

Table 1.1-6

NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP (Note 1)

Post-Irradiation Cooling Time (years)	MPC-24 BPRA Burnup (MWD/MTU)	MPC-24 TPD Burnup (MWD/MTU)
≥ 3	≤20,000	NC (Note 2)
≥ 4	NC	≤20,000
≥ 5	≤30,000	NC
≥ 6	≤40,000	≤30,000
≥ 7	NC	≤40,000
≥ 8	≤50,000	NC
≥ 9	≤60,000	≤50,000
≥10	NC	≤60,000
≥11	NC	NC
≥12	NC	≤90,000
≥13	NC	≤180,000
≥14	NC	≤630,000

Notes:

1. Linear interpolation between points is permitted, except that TPD burnups >180,000 MWD/MTU and ≤630,000 MWD/MTU must be cooled ≥14 years.
2. Not Calculated

LIST OF ASME CODE EXCEPTIONS FOR THE HI-STAR 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3).	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.

LIST OF ASME CODE EXCEPTIONS FOR THE HI-STAR 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be hydrostatically tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant hydrostatic test. All MPC enclosure vessel welds (except the closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The vent/drain cover plate weld is confirmed by liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant examination. The inspection process, including findings, (indications) shall be made a permanent part of the certificate holder's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.

Table 1.3-1 (Page 3 of 5)
LIST OF ASME CODE EXCEPTIONS FOR THE HI-STAR 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. The function of the MPC enclosure vessel is to contain the radioactive contents under normal, off-normal, and accident conditions. The MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STAR 100 Cask System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
Overpack Helium Retention Boundary	NB-1100	Statement of requirements for Code stamping of components	Overpack helium retention boundary is designed, and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
Overpack Helium Retention Boundary	NB-2000	Requires materials to be supplied by ASME approved Material Supplier	Material will be supplied by Holtec approved suppliers with CMTRs per NB-2000.
Overpack Helium Retention Boundary	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. Function of overpack vessel is to contain helium contents under normal, off-normal, and accident conditions. Overpack vessel is designed to withstand maximum internal pressure and maximum accident temperatures.

Table 1.3-1 (Page 4 of 5)
LIST OF ASME CODE EXCEPTIONS FOR THE HI-STAR 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
Overpack Helium Retention Boundary	NB-8000	Statement of requirements for nameplates, stamping and reports per NCA-8000	The HI-STAR 100 Cask System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs per NG-2000 requirements.
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STAR 100 Cask System will be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No code stamping is required. The MPC basket data package will be in conformance with Holtec's QA program.
Overpack Intermediate Shells	NF-4622	All welds, including repair welds, shall be post-weld heat treated (PWHT).	Intermediate shell-to-top flange welds and intermediate shell-to-bottom plate welds do not require PWHT. These welds attach non-pressure retaining parts to pressure retaining parts. The pressure retaining parts are >7 inches thick. Localized PWHT will cause material away from the weld to experience elevated temperatures which will have an adverse effect on the material properties.

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
Overpack Helium Retention Boundary	NG-2000	Perform radiographic examination after post-weld heat treatment (PWHT)	Radiography of helium retention boundary welds after PWHT is not required. All welds (including repairs) will have passed radiographic examination prior to PWHT of the entire containment boundary. Confirmatory radiographic examination after PWHT is not necessary because PWHT is not known to introduce new weld defects in nickel steels.
Overpack Intermediate Shells	NF-2000	Requires materials to be supplied by ASME approved Material Supplier	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.
Overpack Helium Retention Boundary	NB-2330	Defines the methods for determining the T_{NDT} for impact testing of materials	T_{NDT} shall be defined in accordance with Regulatory Guides 7.11 and 7.12 for the helium retention boundary components.

ENCLOSURE 2
SAFETY EVALUATION REPORT

SAFETY EVALUATION REPORT

Docket No. 72-1008
Model No. HI-STAR 100 Cask System
Certificate of Compliance No. 1008
Amendment No. 1

SUMMARY

This Safety Evaluation Report (SER) documents the review and evaluation of an amendment application for the HI-STAR 100 Cask System. The application was submitted by Holtec International (the applicant) on November 24, 1999, as supplemented. The application requests changes to Certificate of Compliance 1008 (the certificate), including its appendices, the design drawings, and Revision 10 of the Safety Analysis Report (SAR) for the HI-STAR 100 Cask System (the cask).

The requested changes include:

(1) revisions to limits for existing fuel array/classes,
(2) addition of pressurized water reactor (PWR) Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs),
(3) addition of two new fuel assembly array/classes,
(4) addition of a new damaged fuel container (DFC),
(5) addition of thoria rods in canisters,
(6) addition of antimony-beryllium neutron sources, and
(7) clarifications, editorial corrections, and other minor changes.

The application, as supplemented, included the necessary engineering analyses and proposed SAR page changes. The proposed SAR revisions are required to be incorporated into the Final Safety Analysis Report that must be submitted to the U. S. Nuclear Regulatory Commission (NRC) after the amendment has been approved (in accordance with 10 CFR 72.248).

The NRC staff has reviewed the application, as supplemented, including the engineering analyses, proposed SAR revisions, and other supporting documents submitted with the application. Based on the statements and representations in the application, as supplemented, the staff concludes that the HI-STAR 100 Cask System, as amended, meets the requirements of 10 CFR Part 72.

References

Holtec International application dated November 24, 1999.

Supplements dated February 4, 18 and 28, March 2, 16 and 31, and May 23, 2000.

1.0 GENERAL

The applicant proposed several minor changes to Chapter 1 of the SAR and corresponding changes to the certificate, as applicable. These changes are described in detail in the application, as supplemented, and include minor revisions to the design drawings, definitions, and the Holtite neutron shield specifications. The purpose of the changes are primarily to provide clarification, enhance consistency between the certificate and SAR, remove unnecessary information, and provide flexibility where appropriate.

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 cask and will not affect the ability of the cask to meet the requirements of 10 CFR Part 72.

2.0 PRINCIPAL DESIGN CRITERIA

The applicant proposed several changes to Chapter 2 of the SAR and corresponding changes to the certificate, as applicable. These changes are described in detail in the application, as supplemented. The changes include minor SAR revisions to provide clarification, enhance consistency between the certificate and SAR, remove unnecessary information, and provide flexibility where appropriate. 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 cask and will not affect the ability of the cask to meet the requirements of 10 CFR Part 72.

Chapter 2 of the SAR was also revised to address the revisions to the limits for existing fuel array/classes and the addition of BPRAs, TPDs, the 8x8F and 15x15H fuel assembly array/classes, a new DFC, thoria rods in canisters, and antimony-beryllium neutron sources. These changes primarily affect shielding and criticality and are evaluated in Chapters 5 and 6 of this SER.

The NRC staff reviewed the proposed changes to Chapter 2 of the SAR and found that changes to the design criteria complied with 10 CFR Part 72 requirements and provided adequate assurance that the specified spent fuel can be stored safely.

3.0 STRUCTURAL EVALUATION

The applicant proposed several changes to Chapter 3 of the SAR to address (1) the revisions to the design drawings; (2) the revisions to the limits for existing fuel array/classes; (3) the addition of BPRAs, TPDs, the 8x8F and 15x15 H fuel assembly array/classes, a new DFC, thoria rods in canisters, and antimony-beryllium neutron sources; and (4) other clarification or editorial changes. The requested changes do not have an impact on the structural performance or integrity of the cask 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 multi-purpose canister (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 meets 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 cask.

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 approved contents. The TN/D-1 DFC and Thoria Rod Canister were structurally evaluated and found to meet all required design requirements for storage in the HI-STAR 100 Cask System. The structural analysis is provided in Appendix 3.AI of the proposed 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), BPRAs, TPDs, 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 the weight of the contents or cask and, therefore, do not affect the existing structural evaluation.

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

4.0 THERMAL EVALUATION

The applicant proposed several changes to Chapter 4 of the SAR to address (1) the revisions to limits for existing fuel array/classes; (2) the addition of BPRAs, TPDs, the 8x8F and 15x15H fuel assembly array/classes, a new DFC, thoria rods in canisters, and antimony-beryllium neutron sources; and (3) other clarification or editorial changes. These changes do not involve an increase to the decay heat load or a change to the heat transfer characteristics of the cask. The new contents and content limits are bounded by the thermal analysis for previously approved contents. Therefore, the staff finds the proposed changes acceptable.

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) addition of BPRAs and TPDs to the approved contents;

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

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

TN/D-1 Damaged Fuel Container

The HI-STAR 100 is currently approved to store damaged fuel or fuel debris when the fuel is contained in a Holtec DFC. The applicant requested the addition of the TN/D-1 DFC to the HI-STAR 100 approved contents. Figure 2.1.2 in the proposed SAR shows 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 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 proposed SAR Tables 5.2.32 and 5.2.33, were bounded by the source terms for the design basis boiling water reactor (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 for the design basis fuel.

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 100 approved contents. The Dresden Unit 1 fuel assembly was previously approved for storage in the HI-STAR 100 cask. 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 5.4.15 of the proposed SAR 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 are 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.

BPRAs and TPDs

The applicant requested revision of the approved contents to allow storage of PWR fuel assemblies with BPRAs and TPDs in the HI-STAR 100 cask. The only significant radiation source is from irradiation of the impurities in the stainless steel and Inconel in the BPRAs and TPDs, which creates Co-60. The applicant determined the bounding BPRA and TPD, described in proposed SAR Table 5.2.29, by analyzing different BPRAs and TPDs to determine which produced the highest source term and decay heat for a specific burnup and cooling time. The applicant used the SAS2H and ORIGEN-S modules of the SCALE code to calculate the radiation source term and decay heat load for the BPRAs and TPDs which is the same code used for the fuel source term calculations. The total curies of cobalt and the decay heat were then calculated as a function of burnup and cooling time. The BPRA and TPD limits for burnup and cooling time are given in Table 1.1-6 of Appendix B of the proposed certificate. The allowable BPRA and TPD decay heat load was subtracted from the assembly decay heat load to determine the assembly burnup and cooling times for fuel with TPDs or BPRAs. The decay heat load from the TPDs is negligible. To account for the heat load from the BPRAs, the fuel assembly burnup and cooling times, given in Appendix B Table 1.1-5 of the proposed certificate, were revised.

The applicant then calculated the dose rates for the HI-STAR 100 cask assuming all fuel assemblies contained either BPRAs or TPDs. The results are discussed in proposed SAR Section 5.4.6. The dose rates were bounded by the current dose rates at all locations except dose point number 6. The differences in dose rates here are negligible and would not affect the cask's ability to meet the requirements of 10 CFR 72.104 and 72.106.

The staff performed confirmatory calculations to determine the additional source term and activity from the BPRAs and TPDs. The cooling time and average burnups given in Appendix B Table 1.1-6 of the proposed certificate were used. The staff's results are in close agreement with the applicant's results. For the confirmatory analysis, the staff used the SAS2H and ORIGENS modules of the SCALE version 4.4 computer code and the accompanying 44-group cross-section library. These codes are standards in the industry for performing shielding analyses and are appropriate for this particular application and fuel system.

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 72 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 cask to meet the dose rate requirements of 10 CFR Part 72.

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 mixed-oxide (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, and 10x10C; and

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

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

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

TN/D-1 Damaged Fuel Containers

The applicant requested that the TN/D-1 DFC be approved for storage in the HI-STAR MPC-68 and MPC-68F. A sketch of the TN/D-1 DFC is provided in Figure 2.1.2 in Chapter 2 of the proposed 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 a 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 6.4.5 of the proposed 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 approval to store 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 2.1.2A in the proposed SAR. The thoria rod contents are described in Table 6.2.42 of the proposed SAR. 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 Appendix B to the proposed certificate.

Antimony-Beryllium Neutron Source in Dresden Unit 1 Fuel Assemblies

The applicant requested approval to store 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 Appendix B to the proposed certificate. 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 k_{eff} from 0.7611 to 0.7824.

The staff has performed a 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 B to the proposed certificate. The staff concludes that, given the bounding fuel assembly dimensions defined in the current and proposed 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 proposed SAR Table 6.2.1 for BWR assemblies and proposed 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 proposed 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 Cask System. Characteristics of the 8x8F and 15x15H assemblies are presented in proposed 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 a k_{eff} 0.9411 for the 15x15H.

Staff verified that all fuel assembly parameters important to criticality safety have been included in Appendix B to the proposed certificate. 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 cask and the contents of the HI-STAR 100 Cask System do not affect the ability of the cask to meet the criticality safety requirements of 10 CFR Part 72.

7.0 CONFINEMENT EVALUATION

The applicant proposed several changes to Chapter 7 of the SAR to address (1) the revisions to limits for existing fuel array/classes; (2) the addition of BPRAs, TPDs, the 8x8F and 15x15H fuel assembly array/classes, a new DFC, thoria rods in canisters, and antimony-beryllium neutron sources; and (3) other clarification or editorial changes. The requested changes do not involve an increase to the confinement source terms or a significant change to the design and operation of the confinement system. The new contents and content limits are bounded by the confinement analysis for previously approved contents. Therefore, the staff finds the proposed changes acceptable.

8.0 OPERATING PROCEDURES

The applicant proposed several editorial and clarification changes to Chapter 8 of the SAR. The requested changes do not result in a significant change to the operation of the cask. Therefore, the staff finds the proposed changes acceptable.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant proposed several editorial and clarification changes to Chapter 9 of the SAR. The requested changes do not result in a significant change to the cask's acceptance testing and maintenance program. Therefore, the staff finds the proposed changes acceptable.

10.0 RADIATION PROTECTION

The radiological impact of the proposed changes are discussed in Chapter 5 of this SER. The proposed changes do not impact the ability of the cask to meet the requirements of 10 CFR 72.104 and 72.106.

11.0 ACCIDENT ANALYSES

The applicant proposed several changes to Chapter 4 of the SAR to address (1) the revisions to limits for existing fuel array/classes; (2) the addition of BPRAs, TPDs, the 8x8F and 15x15H fuel assembly array/classes, a new DFC, thoria rods in canisters, and antimony-beryllium neutron sources; and (3) other clarification or editorial changes. The proposed changes do not have a significant effect on the performance of the cask under off-normal and accident conditions. Therefore, the staff finds the proposed changes acceptable.

12.0 CONDITIONS FOR CASK USE - OPERATING CONTROLS AND LIMITS OR TECHNICAL SPECIFICATIONS

Tables 12-1 and 12-2 below list the proposed changes to the certificate. The staff has reviewed these changes and, as discussed in the preceding chapters of this SER, have found them acceptable.

Table 12-1

Proposed Changes to Certificate of Compliance 1008
Appendix A - Technical Specifications

Appendix A Section	Change Description
Throughout	Editorial changes and typographical corrections.
1.1	Revised definitions of damaged fuel assembly and damaged fuel container for clarification.
2.1.1	Replaced the MPC helium backfill density limit with a helium backfill pressure limit to simplify requirement.
	Revised the leak rate units in Table 2-1 from std cc/sec to atm-cc/sec for clarification.

Table 12-2

Changes to Certificate of Compliance 1008
Appendix B - Approved Contents and Design Features

Appendix B Section	Change Description
Throughout	Editorial changes and typographical corrections.
1.0	Revised definitions of damaged fuel assembly, damaged fuel container, and planar-average initial enrichment for clarification and consistency with Appendix A.
1.1	Revised Section 1.1.1 to permit storage of certain non-fuel hardware.
1.4	Revised Item 6 to clarify the requirements for cask storage pad.
1.5	Revised Section 1.5.2 to clarify surface emissivity requirement.
Table 1.1-1	Added limits for BPRAs and TPDs, array class 8x8F, and thoria rods.
Table 1.1-2	Revised certain fuel assembly parameters, added array/class 15x15H fuel assembly, and added clarifying notes.
Table 1.1-3	Revised certain fuel assembly parameters, added array/class 8x8F fuel assembly, and added clarifying notes.
Tables 1.1-4 and 1.1-5	Revised to clarify limits, reflect addition of BPRAs and TPDs, and permit linear interpolation between points.
Table 1.1-6	Added table specifying cooling and average burnup limits for non-fuel hardware.
Table 1.3-1	Added exception to ASME Code NB-5230 for the MPC lid-to-shell weld and added clarifying text.

In addition to the changes above, the staff revised the certificate to include a condition that requires users to prepare written acceptance tests and a maintenance program consistent with the technical basis described in Chapter 9 of the SAR.

13.0 QUALITY ASSURANCE

The applicant proposed several minor changes to Chapter 13 of the SAR. The purpose of the changes is to provide clarification and correct typographical errors. The proposed changes make Chapter 13 of the HI-STAR SAR consistent with Chapter 13 of the HI-STORM SAR, which the NRC has previously accepted. Therefore, the staff finds the proposed changes acceptable.

14.0 DECOMMISSIONING

The proposed changes do not have a significant impact on the decommissioning considerations for the cask.

CONCLUSION - EVALUATION FINDINGS

The staff has reviewed the HI-STAR 100 Cask System amendment application, as supplemented, including the engineering analyses, proposed SAR revisions, and other supporting documents submitted with the application. Based on the information provided in the application, as supplemented, the staff concludes that the HI-STAR 100 Cask System, as amended, meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1008, Amendment No. 1,
on December 21, 2000.