July 23, 1999

See Ext.

MEMORANDUM TO: Donald A. Cool, Director
Division of Industrial Medical
and Nuclear Safety, NMSS

FROM: E. William Brach, Director (Original Signed by:)
Spent Fuel Project Office, NMSS

SUBJECT: TRANSMITTAL OF FINAL SAFETY EVALUATION REPORT AND CERTIFICATE OF COMPLIANCE FOR THE HOLTEC INTERNATIONAL HI-STAR 100 STORAGE CASK (TAC NO. L22906)

The preliminary Safety Evaluation Report (SER) and Certificate of Compliance for the Holtec International HI-STAR 100 storage cask were transmitted to the Division of Industrial Medical and Nuclear Safety (IMNS) on September 30, 1998. Subsequent to that, this material was made available for public comment. The Spent Fuel Project Office (SFPO) staff completed its review and disposition of the public comments and provided them to your staff in electronic form in June 1999. SFPO has now completed the preparation of the final SER and Certificate of Compliance for the HI-STAR 100. These documents are included as Attachments 1 and 2.

Please keep me informed of the status of the completion of the rulemaking process regarding the HI-STAR 100. SFPO staff will be available to assist IMNS, as necessary, to ensure a timely conclusion to this process.

The Senior Project Manager for the HI-STAR 100 cask is Mark S. Delligatti. He can be reached at (301) 415-8518. During Mr. Delligatti's absence from the office through July 30, 1999, please contact Ross Chappell, Chief of the Spent Fuel Licensing Section, if you have any questions. Mr. Chappell may be contacted at (301) 415-8510.

Attachments: As stated (2)

Docket: 72-1008

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

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NAME:	MFields:jh		VLTharpe		SShankman		MWHodges		EWBrach			
DATE:	7/9/99		7/ /99		7/ /99		7/2/99		7/ /99		7/ /99	

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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 Number	Amendment No.	Amendment Date	Package Identification No.
1008			72-1008			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., Safety Analysis Report for the HI-STAR 100 Cask System, Revision 10
Holtec Report HI-941184

Docket No. 72-1008

CONDITIONS

1. CASK

The HI-STAR 100 Cask System is certified as described in the 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.

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. 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.

4. HEAVY LOADS REQUIREMENTS

Each lift of a HI-STAR 100 spent fuel storage 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.

5. APPROVED CONTENTS

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

6. APPROVED DESIGN FEATURES

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

7. 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.

8. 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 NUCLEAR REGULATORY COMMISSION

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

TABLE OF CONTENTS

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 and Bases.

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 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 to be 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.
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 fuel with known or suspected defects, such as ruptured fuel rods, severed rods, or loose fuel pellets. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered to be FUEL DEBRIS.

1.1 Definitions

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.

1.1 Definitions

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 or 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 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 SFSC is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC.

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 AND and OR. 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 indentations 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.

1.2 Logical Connectors

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.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR 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 OR 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 AND. 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 OR 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 SFSC. 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.

1.3 Completion Times

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.

1.3 Completion Times

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.

1.3 Completion Times

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.

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

1.4 Frequency

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

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

1.4 Frequency

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.

2.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 2.0.3 (continued)	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.
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: 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 quantity of moisture left in MPC. AND A.2 Determine and complete corrective actions necessary to return MPC to analyzed condition.	7 days 30 days
B. MPC helium backfill density limit not met	B.1 Perform an engineering evaluation to determine impact of helium differential. AND B.2 Determine and complete corrective actions necessary to return MPC to analyzed condition.	72 hours 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 off-site dose release effects.	24 hours
	AND	
	C.2 Determine and complete corrective actions necessary to return MPC to analyzed condition.	7 days
D. Required Actions and Associated Completion Time not met.	B.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 density 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

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 the OVERPACK. AND A.2 Determine and complete corrective actions necessary to return the OVERPACK to analyzed condition.	7 days 30 days
B. OVERPACK annulus helium backfill pressure limit not met.	B.1 Perform an engineering evaluation to determine impact of helium pressure differential. AND B.2 Determine and complete corrective actions necessary to return the OVERPACK to analyzed condition.	72 hours 30 days

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

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 limit specified 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 ≤ 21 inches when oriented vertically.

AND

ii A lift height ≤ 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 with HI-STAR 100 TSAR 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, ect.).

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 the ISFSI

2.1 SFSC INTEGRITY

2.1.4 Fuel Cool-Down

LCO 2.1.4 The MPC exit gas temperature shall be ≤ 200° 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. AND A.2 Ensure adequate heat transfer from MPC to the environment.	Prior to initiating MPC re-flooding operations 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 2.1.4.1	Verify MPC helium gas exit temperature within the 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. AND 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.	24 hours 48 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the 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 the 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 off-site location, SR 2.2.1.1 shall be performed after receipt of the OVERPACK and prior to STORAGE OPERATIONS.	

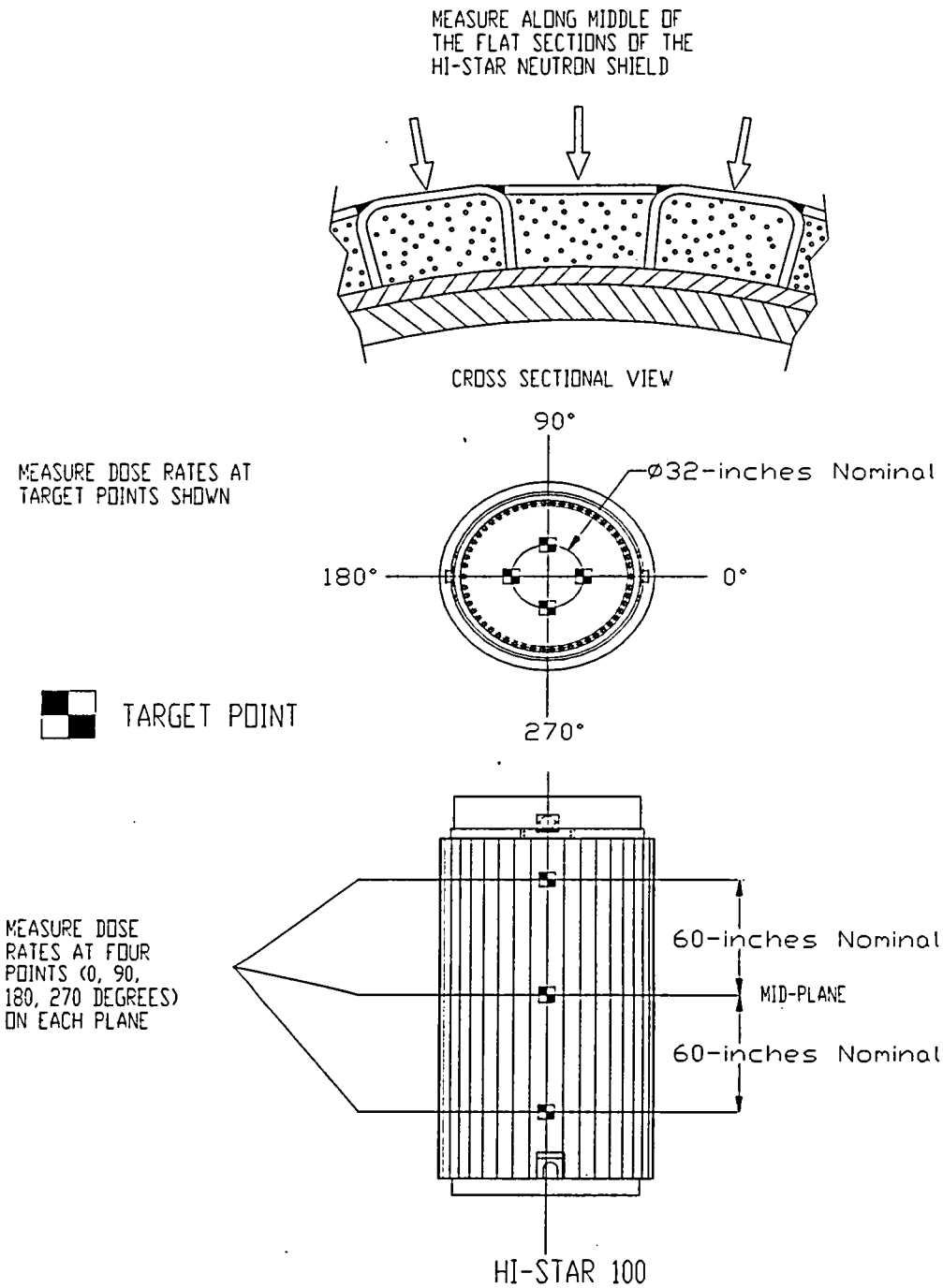

Figure 2.2.1-1
OVERPACK Surface Dose Rate Measurement Locations

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 SFSC.

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 the OVERPACK 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.	

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 Density[1]	0.1212 g-moles/liter +0% or -10%
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 std cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 std 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 Density[1]	0.1218 g-moles/liter +0% or -10%
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 std cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 std 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 Density[1]	0.1218 g-moles/liter +0% or -10%
d. OVERPACK Annulus Helium Backfill Pressure	≥ 10 psig and ≤ 14 psig
e. MPC Helium Leak Rate	≤ 5.0E-6 std cc/sec (He)
f. OVERPACK Helium Leak Rate	≤ 4.3E-6 std cc/sec (He)

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

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.

d. 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 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 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

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 greater than pinhole leaks or hairline cracks, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. A DAMAGED FUEL ASSEMBLY's inability to be handled by normal means must be due to mechanical damage and must not be due to fuel rod cladding damage.
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.
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 simple average of the distributed fuel rod 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 System

a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS meeting the limits specified in Table 1.1-1 (which refers to Tables 1.1-2 through 1.1-5) may be stored in the HI-STAR 100 SFSC System.

b. 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.

c. 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.

c. 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:

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

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

c. 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 contained 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

Table 1.3-1 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 will need verification by the system user, as a minimum, are as follows:

1. The temperature of 80°F is the maximum allowed average yearly temperature.

2. The allowed temperature extremes, averaged over a three day period, shall be greater than -40°F, and less than 125 °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 requirement of 10 CFR 72.212(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

   c. Reinforcement top and bottom (Both Directions):
      Reinforcement area and spacing determined by analysis
      Reinforcement yield strength: $\leq$ 60,000 psi

   d. Soil effective modulus of elasticity: $\leq$ 28,000 psi

      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 10 CFR 72.104(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 0.01g/cm^2 in the MPC-68F.

1.5.2. Specifications Important for Thermal Performance

1.5.2.1 OVERPACK

The painted surface of the HI-STAR 100 OVERPACK must have an emissivity no less than 0.85.

Table 1.1-1
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES 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:	≤ 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.
ii. SS Clad:	An assembly post-irradiation cooling time ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU. OR An assembly post-irradiation cooling time ≥ 15 years and an average burnup ≤ 40,000 MWD/MTU.

e. Nominal Fuel Assembly Length:	≤ 176.8 inches
f. Nominal Fuel Assembly Width:	≤ 8.54 inches
g. Fuel Assembly Weight:	≤ 1,680 lbs

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

C. Fuel assemblies shall not contain control components.

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

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 array/class 6x6A, 6x6C,and 8x8A fuel assemblies, which shall have a decay heat $\leq$ 115 Watts.
ii. SS Clad:	$\leq$ 95 Watts
e. 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, except for array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
ii. SS Clad:	An assembly cooling time after discharge $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.

f.	Nominal Fuel Assembly Length:	≤ 176.2 inches
g.	Nominal Fuel Assembly Width:	≤ 5.85 inches
h.	Fuel Assembly Weight:	≤ 700 lbs, including channels

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 2.1-3 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

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

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

B. Quantity per MPC: Any combination of DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS and INTACT FUEL ASSEMBLIES UP TO A TOTAL OF 68.

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

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 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

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:	A post-irradiation cooling time after discharge $\geq$ 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

3. Uranium oxide, BWR FUEL DEBRIS, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the 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:	< 135.0 inches
g.	Nominal Original Fuel Assembly Width:	$\leq$ 4.70 inches
h.	Fuel Debris Weight:	$\leq$ 400 lbs, including channels

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:	≤ 115 Watts
e.	Post-irradiation Cooling Time and Average Burnup Per Assembly:	An assembly post-irradiation cooling time after discharge ≥ 18 years and an average burnup ≤ 30,000 MWD/MTIHM.
f.	Nominal Fuel Assembly Length:	≤ 135.0 inches
g.	Nominal Fuel Assembly Width:	≤ 4.70 inches
h.	Fuel Assembly Weight:	≤ 400 lbs, including channels

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 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:	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

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

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:

a. Uranium oxide BWR INTACT FUEL ASSEMBLIES;

b. MOX BWR INTACT FUEL ASSEMBLIES;

c. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; or

d. MOX BWR DAMAGED FUEL ASSEMBLIES placed in DFCs.

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

Table 1.1-2
PWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**15x15A**
Clad Material (note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.)	≤ 402	≤ 402	≤ 410	≤ 400	≤ 420
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3840	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3770	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5(note 3)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.040	≥ 0.0145	≥ 0.0165

Table 1.1-2 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (wt % ^{235}U)	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1
No. of Fuel Rods	204	204	208	208	208
Clad O.D. (in.)	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Clad I.D. (in.)	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Pellet Dia. (in.)	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	21	17	17	17
Guide Tube Thickness (in.)	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 1.1-2 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

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

Notes:
1. Initial Uranium weights and all dimensions are design nominal values. Actual uranium weights may be up to 2.0% higher, within the manufacturers tolerance. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. Each guide tube replaces four fuel rods.

Table 1.1-3
BWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ 108	≤ 108	≤ 108	≤ 100	≤ 195	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 3 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.0	≤ 4.0
No. of Fuel Rods	36	36 (up to 9 MOX rods)	36	49	49	64
Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Clad I.D. (in.)	≤ 0.4945	≤ 0.4945	≤ 0.4990	≤ 0.4200	≤ 0.4990	≤ 0.3620
Pellet Dia. (in.)	≤ 0.4940	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4880	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.694	≤ 0.694	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 110	≤ 110	≤ 77.5	≤ 79	≤ 150	≤ 110
No. of Water Rods	0	0	0	0	0	0
Water Rod Thickness (in.)	N/A	N/A	N/A	N/A	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 1.1-3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

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

Table 1.1-3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	**9x9C**	**9x9D**	**9x9E**	**9x9F**	**10x10A**
Clad Material	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ 173	≤ 170	≤ 170	≤ 170	≤ 182
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note 7)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3590	≤ 0.3810	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3525	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods	1	2	5	5	2
Water Rod Thickness (in.)	> 0.020	≥ 0.0305	≥ 0.0305	≥ 0.0305	≥ 0.0300
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.120

Table 1.1-3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	10x10B	10x10C	10x10D	10x10E
Clad Material (note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.)	≤ 182	≤ 180	≤ 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	≤ 5.0
No. of Fuel Rods	91/83 (note 8)	96	100	96
Clad O.D. (in.)	≥ 0.3957	≥ 0.3790	≥ 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	1 (Note 5)	5 (Note 9)	0	4
Water Rod Thickness (in.)	> 0.00	≥ 0.034	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be up to 1.5% higher, within the manufacturer's tolerance. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. ≤ 0.612 wt.% ^{235}U and ≤ 1.578 wt,% total fuel fissile plutonium (^{239}Pu and ^{241}Pu).
4. This assembly class contains 74 rods; 66 full length rods and 8 partial length rods.
5. Square, replacing nine fuel rods.
6. Variable
7. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
8. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
9. One diamond shaped water rod replacing the four center fuel rods and four rectangualr water rods dividing the assembly into four quadrants.

Table 1.1-4
FUEL ASSEMBLY COOLING AND DECAY HEAT GENERATION

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly 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
> 14	≤ 665	≤ 226

Table 1.1-5
FUEL ASSEMBLY COOLING AND AVERAGE BURNUP

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup (MWD/MTU)	MPC-68 BWR Assembly Burnup (MWD/MTU)
≥ 5	≤ 28,700	≤ 26,000
≥ 6	≤ 32,800	≤ 29,100
≥ 7	≤ 33,300	≤ 29,600
≥ 8	≤ 35,600	≤ 31,400
≥ 9	≤ 37,000	≤ 32,800
≥ 10	≤ 38,300	≤ 33,800
≥ 11	≤ 39,300	≤ 34,800
≥ 12	≤ 40,200	≤ 35,500
≥ 13	≤ 40,900	≤ 36,200
≥ 14	≤ 41,500	≤ 36,900
≥ 15	≤ 42,100	≤ 37,600

LIST OF ASME CODE EXCEPTIONS FOR HI-STAR 100 SYSTEM

Table 1.3-1

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 Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root 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.

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 lid-to-shell and closure ring and vent/drain cover plate) are inspected by RT or UT. The MPC lid-to-shell root and final weld layers are PT examined and the entire weld is either UT examined or multilayer PT examined. The vent/drain cover plate weld is confirmed by leakage testing and 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.

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. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. 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.	HI-STAR 100 System 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.	Materials 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.

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.	HI-STAR 100 System 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's 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 in accordance with NG-2000 requirements.
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STAR 100 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.
Overpack Helium Retention Boundary	NG-2000	Perform radiographic examination after post-weld heat treatment (PWHT)	Radiography of the 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.

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
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.

SAFETY EVALUATION REPORT

HOLTEC HI-STAR 100 CASK SYSTEM

TABLE OF CONTENTS

INTRODUCTION

This Safety Evaluation Report (SER) documents the review and evaluation of Revision 10 to the Topical Safety Analysis Report (hereafter referenced in this document as the SAR) for the Holtec International (Holtec) HI-STAR 100 Cask System.[1] The SAR was submitted by Holtec following the format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.[2] This SER uses essentially the same Section-level format, with some differences implemented for clarity and consistency.

The review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the HI-STAR 100. The cask will be used at Independent Spent Fuel Storage Installations (ISFSIs) licensed under 10 CFR Part 72[3] at a reactor site operating with a 10 CFR Part 50 license.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. Decommissioning, to the extent that it is treated in the SAR, presumes that, as a bounding case, the HI-STAR 100 cask is unloaded and subsequently decontaminated before disposition or disposal.

While the HI-STAR 100 Cask System is designed to be used as a dual-purpose storage and transportation cask, the use or certification of the HI-STAR 100 under 10 CFR Part 71 for off-site transport of spent fuel is not a subject of this SER. Certification for transportation could occur only after the completion of a separate staff review of the HI-STAR 100 Safety Analysis Report (SAR) for transportation.

References

1. Topical Safety Analysis Report for the HI-STAR 100 Cask System, Rev. 10.

2. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

3. U.S. Code of Federal Regulations. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the HI-STAR 100 is to ensure that Holtec has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 General Description

As indicated in the certificate of compliance (CoC) and technical specifications (TS), Holtec International's (Holtec's) 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. A variation of the MPC-68, designated as MPC-68F, may contain BWR fuel debris, as defined in the TS. Both MPCs are identical in external dimensions and will fit into the same overpack design.

The general arrangement drawings for the HI-STAR 100 Cask System are contained in Section 1.4 of the Safety Analysis Report (SAR). SAR Figure 1 is a basic representation of the HI-STAR 100.

The approved contents for the HI-STAR 100 include: uranium oxide (UO_2) 14x14, 15x15, 16x16, and 17x17 PWR fuel assemblies without control components; UO_2 6x6, 7x7, 8x8, 9x9, and 10x10 BWR fuel assemblies with or without channels; and mixed-oxide (MOX) 6x6 BWR fuel assemblies with or without channels. All PWR fuel assembly types must be stored as intact fuel. Certain BWR fuel assembly types may be stored as damaged fuel or fuel debris placed in damaged fuel containers (DFCs). The enrichment and physical, thermal, and radiological characteristics of the approved contents are given in the TS. The TS also provide definitions for intact fuel assemblies, damaged fuel assemblies, and fuel debris.

1.2 Evaluation Findings

F1.1 A general description and discussion of the HI-STAR 100 is presented in Section 1 of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for structures, systems, and components (SSCs) important to safety are presented in Section 1.5 of the SAR. Specific SSC's are evaluated in Sections 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the dry cask storage system (DCSS) are provided in SAR Section 2.1. Additional details concerning these

specifications are presented in Section 2 of the SAR and in the TS which are appended to the CoC.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the SAR and are acceptable to the NRC staff.

F1.5 The quality assurance (QA) program and implementing procedures are described in Section 13 of the SAR and are evaluated in Section 13 of the SER.

F1.6 The HI-STAR 100 was not reviewed in this SER for use as a transportation cask.

F1.7 The staff concludes that the information presented in this section of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry cask storage practices detailed in NUREG-1536.

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the principal design criteria related to SSC's important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72.

2.1 Review Summary

Holtec has presented general details of the principal design criteria in Section 2 of the SAR and provided appropriate details in the associated sections of the SAR.

2.2 Evaluation Findings

F2.1 The staff concludes that the principal design criteria for the HI-STAR 100 are acceptable with regard to demonstrating compliance with the regulatory requirements of 10 CFR Part 72. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. More detailed evaluations of design criteria and assessments of compliance with those criteria are presented in Sections 3 through 14 of the SER.

3.0 STRUCTURAL EVALUATION

This section evaluates the structural design of the HI-STAR 100 Cask System. Structural design features and design criteria are reviewed, and analyses related to structural performance under normal, off-normal, accident, and natural phenomena events are evaluated.

3.1 Structural Design

3.1.1 Structural Design Features

Section 1 of this SER provides a general description of the HI-STAR 100 Cask System. The HI-STAR 100 Cask System consists of two principal components: the MPC and the overpack. The MPC is a welded cylindrical structure with flat ends and a honeycomb fuel basket. The structural functions of the MPC in the storage mode are (1) to position the spent fuel in a subcritical configuration, and (2) to provide a confinement boundary. The HI-STAR 100 overpack is a heavy, multilayered steel cylindrical vessel. The overpack serves as a missile barrier, radiation shield, and helium retention boundary in the storage mode.

Some major design features of the MPC are summarized below:

- The closure system for the MPC consists of two components, namely, the MPC lid and closure ring. The MPC lid is a thick circular plate continuously welded to the MPC shell along its circumference. The closure ring is a circular annular ring plate edge-welded to the MPC shell and the lid. The MPC lid is equipped with vent and drain ports which are utilized for evacuating moisture and air from the MPC following fuel loading and the subsequent backfilling with helium. The vent and drain ports are covered by a cover plate and welded shut before the closure ring is installed.

- The MPC fuel baskets consist of an array of interconnecting plates. The plates are welded along their edges to form a honeycomb structure. There are three different baskets to house fuel assemblies with different characteristics. All baskets are designed to fit into the same MPC shell.

- The MPC basket is separated from the support structure within the MPC shell by a small gap. The purpose of the small gap between the basket and the basket support structure is to avoid high thermal stresses induced by thermal growth of the basket.

Some major design features of the overpack are summarized below:

- The overpack features an inner shell and multilayered intermediate shells. The multi-layer approach eliminates the potential for a crack in any layer to travel uninterrupted through the vessel wall and, thus, lessens the concerns over brittle fracture at low temperature.

- To facilitate handling of a loaded cask, the HI-STAR 100 overpack is equipped with lifting trunnions and pocket trunnions. The pocket trunnions are designed for stable rotation for

changing cask orientation. Lifting trunnions are conservatively designed to meet the design safety factor requirements of NUREG-0612 and ANSI N14.6-1993.

3.1.2 Structural Design Criteria

The HI-STAR 100 system structural design criteria for normal, off-normal, accident, and natural phenomena events are based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III. The MPC fuel basket structural components are designed to Subsection NG, and the enclosure vessel or shell, which forms the confinement boundary, is designed to Subsection NB. The threaded holes in the MPC lid, used for critical lifts, are designed to the criteria of ANSI N14.6. The overpack top flange, closure plate, inner shell, and bottom plate are designed to Subsection NB, while the remainder of the overpack steel structure is designed to Subsection NF. While the ASME B&PV Code, Section III, was intended for the design and fabrication of reactor vessels, the U.S. Nuclear Regulatory Commission (NRC) accepts its use, to the extent practical, for spent fuel storage systems. SAR Section 2.2.4 discusses the exceptions to the ASME B&PV Code for the HI-STAR 100 Cask System. These exceptions to the Code design and fabrication criteria were reviewed and included in the CoC.

3.1.2.1 Individual Loads

Individual loads which address each design criterion applicable to the structural design of the HI-STAR 100 are listed in Table 2.2.13 (Notation for Design Loading for Normal, Off-Normal, and Accident Conditions) of the SAR. However, some individual loads such as those induced by the postulated accident and natural phenomena events are defined by the applicant and provided below.

3.1.2.1.1 Tipover

The HI-STAR 100 Cask System will not tipover as a result of a postulated natural phenomenon event, including tornado wind and a tornado-generated missile, seismic, or flood. However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tipover scenario per NUREG-1536 has been analyzed.

3.1.2.1.2 Handling Accident

For handling accidents, a loaded HI-STAR 100 overpack (i.e., cask) is assumed to drop either 21 inches vertically or 72 inches horizontally onto an ISFSI concrete pad.

3.1.2.1.3 Explosion

As indicated by 10 CFR 73.55(d), unauthorized explosive materials are not permitted within the ISFSI. Therefore, the accident condition overpack external pressure (300 psig) specified in Table 2.2.1 of the SAR should bound all credible external explosion events.

3.1.2.1.4 Flood

The maximum design flood water depth of 656 ft. corresponds to a maximum external pressure of 285 psig that is bounded by the design external pressure of 300 psig specified in Table 2.2.1 of the SAR. Stability of the HI-STAR 100 system due to the flood water velocity (13 ft./sec.) and hydrostatic pressure is evaluated.

3.1.2.1.5 Tornado and Tornado Missile

Tornado wind speed and tornado-induced pressure drop are specified in Table 2.2.4 of the SAR. Tornado missiles are listed in Table 2.2.5 of the SAR. Stability of the HI-STAR 100 system due to tornado missile impact plus either steady wind and pressure drop is evaluated.

3.1.2.1.6 Earthquake

The HI-STAR 100 system is subjected to the design-basis seismic event consisting of three orthogonal statistically independent acceleration time-histories. The HI-STAR 100 system is considered as a rigid body subject to a net horizontal inertia force and a vertical inertia force for the purpose of performing a conservative static analysis to show it will not tipover. The design-basis earthquake accelerations are assumed to be applied at the top of the ISFSI concrete pad with the resulting inertia forces applied at the HI-STAR 100 mass center.

Vertical acceleration is conservatively assumed to be 2/3 of the horizontal acceleration. The acceptable horizontal design-basis earthquake inertia force in each of two orthogonal directions is 0.24g (Section 3.4.7.1, page 3.4-49 of the SAR).

3.1.2.1.7 Lightning

The overpack structure contains thousands of pounds of highly conductive carbon steel and has a large external surface area. It is concluded that the overpack should adequately dissipate any lightning which may strike the HI-STAR 100 Cask System.

3.1.2.1.8 Fire

The HI-STAR 100 Cask System is evaluated for a fire accident event. The fire accident for storage is conservatively specified to be the result of the spillage and ignition of the technical specification limit of 50 gallons of combustible transporter fuel. Table 2.2.8 of the SAR provides the fire duration based on the amount of flammable materials available. The temperature of the fire is assumed to be 1475° F which is consistent with 10 CFR 71.73.

3.1.2.2 Loading Combinations

Loading combinations are selected for the normal, off-normal, and accident conditions for the various HI-STAR 100 system components (i.e., MPC shell, basket, and overpack). In the SAR, Table 3.1.1 lists the load combinations that address overpack stability, and Tables 3.1.3 through 3.1.5 list the load combinations of the individual loads for the basket, the MPC shell, and the overpack, respectively.

3.1.2.3 Allowable Stresses

Allowable stresses are developed for the various HI-STAR 100 system components based on the appropriate ASME subsections and service levels. Tables 3.1.6 through 3.1.16 of the SAR contain the numerical values of the allowable stress/stress intensities for all MPC and overpack structural materials according to temperatures.

3.1.3 Weights and Center of Gravity

Table 3.2.1 of the SAR provides the weights of the individual HI-STAR 100 components as well as the total system weights. The locations of the calculated centers of gravity (CGs) are presented in Table 3.2.2 of the SAR. Since the non-axisymmetric effects of the contents are neglected, all CGs are located on the cask centerline. Table 3.2.3 of the SAR provides the lift weight when the HI-STAR 100 system, with the heaviest fully loaded MPC, is being lifted from the fuel pool and Table 3.2.4 provides a set of bounding weights used in the analysis.

3.1.4 Materials

The mechanical properties of materials used in analysis include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion.

3.1.4.1 Alloy X Structural Materials

Holtec defined a hypothetical material termed Alloy X for all MPC structural components. SAR Appendix 1.A describes Alloy X as any one of four stainless steel alloys: Types 304, 304LN, 316, or 316LN. The material properties of Alloy X used in the SAR analyses are the least favorable values from the set of candidate stainless alloys (Types 304, 304LN, 316, or 316LN) to ensure that all structural analyses are conservative, regardless of the actual MPC material selected. Appendix 1.A also lists temperature-specific ASME Code values for design stress intensity (S_m) [Table 1.A.1], tensile strength (S_u) [Table 1.A.2], yield stress (S_y) [Table 1.A.3], coefficient of thermal expansion (α) [Table 1.A.4], and thermal conductivity (k) [Table 1.A.5]. Each table lists the minimum value among the four alloys for use in structural calculations (Table 1.A.4 also lists the maximum coefficient of thermal expansion for use in calculations). Table 3.3.1 of the SAR has listed the appropriate minimum (and maximum for thermal expansion) numerical values for the material properties of Alloy X stainless steel versus temperature. These values, taken from the ASME Code, Section II, Part D, are used in all analytical calculations for the MPC.

The staff finds that the four alloys selected for Alloy X are very similar austenitic stainless steels with small variations in physical properties over the applicable temperature ranges due to slight variations in chemistry. Type 304 stainless steel may be considered the base alloy for Alloy X. Compared to Type 304 stainless steel, Types 316 and 316LN add 2% molybdenum to increase pitting corrosion resistance; Types 304LN and 316LN reduce carbon content from 0.08% to 0.03% for increased welded condition corrosion resistance; and Types 304LN and 316LN add approximately 0.13% nitrogen for strength to account for reduced carbon levels. In addition, as austenitic stainless steel alloys, there is no transition temperature for brittle behavior as is found in ferritic carbon steels. The staff concludes any of the above alloys are suitable for MPC use. The staff also independently verified the tabulated design values and found them acceptable.

3.1.4.2 Other Structural Materials

Tables 3.3.2 through 3.3.4 of the SAR provide the material properties of carbon-manganese steel and low or nickel alloy steel. These values were also taken from the ASME Code, Section II, Part D. For all cask structural materials, the stress limits have been defined at or below the maximum temperature allowed by the ASME Code, Section II, Part D, for each material. The information provided on structural materials is consistent with the application of the accepted design Code, ASME Section III, selected for the storage cask system. The materials selected for the cask are consistent with those allowed by the ASME Code, Subsection NB. Acceptable requirements include the ASME-adopted specifications given in Section II, Part A, "Ferrous Metals," Part B, "Nonferrous Metals," and Part D, "Properties." The HI-STAR 100 SAR contains detailed tables with temperature specific material properties and allowable stresses in accordance with the ASME Code, Section II for all structural materials. The staff concluded that the material properties used are appropriate for the load conditions of interest (e.g., static or dynamic, impact loading, hot or cold temperature, wet or dry condition). The staff verified that the SAR clearly references acceptable sources, primarily the ASME Code, for all material properties.

The staff concluded material properties and characteristics needed to satisfy these functional safety requirements will be maintained over the approval period. The life cycle may include conditions experienced during cask fabrication, loading, transport, emplacement, storage, transfer, retrieval, and decommissioning. Service conditions include normal, off-normal operations, accidents, and natural phenomena events. The staff concluded the materials of construction used for the MPC and overpack, primarily stainless steels for the MPC and carbon-manganese and low or nickel alloy steels for the overpack, are compatible with the environment during loading, storage, and unloading of the MPC. The stainless steels used for the MPC have a long, proven history in nuclear service.

3.1.4.3 Material Compatibility

The staff reviewed the cask structural materials that are in direct contact with each other and verified that they will not produce a significant chemical or galvanic reaction and the attendant corrosion or combustible gas generation[1]. Table 3.4.2 lists material compatibility with operating environments. No zinc, zinc compounds, or zinc-based coatings are used in the HI-STAR 100 system. No appreciable galvanic reactions are expected with the materials of construction. In addition, SAR Section 8, step 24, directs that a vacuum source be connected to the vent port to keep moist air from condensing on the MPC lid weld area. This will reduce the concentration of any combustible gases if any were present during welding operations.

The SAR describes two coatings to be used on the ferritic steels in the MPC overpack: Thermaline 450 and Carboline 890. As shown in the product data sheets included in Appendix 1.C, Thermaline 450 is a polymer with outstanding barrier protection against chemical exposures, and Carboline 890 exhibits excellent chemical resistance. Although unlikely in the short-term exposure during loading, if either were to react to the acidic spent fuel pool environment, there would be no release of hydrogen as could be expected with a zinc-based coating. In addition, only the coating on the outside of the HI-STAR 100 overpack will be in contact with the spent fuel pool. The interior of the overpack will be filled with clean, non-borated water and kept in place with the annulus seal before insertion.

3.1.4.4 Welds

The applicant stated that all materials utilized in the welding of the cask components comply with the provisions of the appropriate subsections of Section III and Section IX of the ASME Code. The staff reviewed confinement boundary weld designs for compliance with the design code used and found them acceptable. The MPC closure weld is a partial penetration weld but will perform its intended structural and confinement functions. As stated in Table 2.2.15, a factor of 0.45 has been applied to the stress analysis of the MPC lid and closure ring welds, reducing the stress on the weld material. A redundant closure of the MPC is provided by the MPC closure ring in accordance with 10 CFR 72.236(e). The cask welds were well-characterized on engineering drawings and diagrams using standard welding symbols and/or notations in accordance with American Welding Society (AWS) Standard A2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination." The HI-STAR 100 materials (stainless, carbon, and low alloy steels) are readily weldable with commonly available welding techniques. The use of an experienced fabricator will ensure that the processes chosen for fabrication will yield a durable component. Although the current HI-STAR 100 is not an ASME Code stamped component, the HI-STAR 100 system is designed in accordance with the ASME Code, as clarified by the exceptions to the Code listed in Table 1-3 of Appendix B.

Additional materials requirements apply for structural designs governed by the ASME B&PV Code, Section III, Subsection NB. SAR Table 2.2.15 states an ASME Code exception for material suppliers for the MPC, MPC basket, and overpack. Specifically, Holtec will use approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements. The staff concludes this practice is acceptable.

3.1.4.5 Bolting Materials

The material properties of the bolting materials used in the HI-STAR 100 system are given in Table 3.3.5. Bolting materials used in the HI-STAR 100 system are specified in accordance with appropriate ASME specifications: SB-637-N07718. Procurement in accordance with these specifications will help ensure mechanical properties and proper heat treatment, as required by the applicable specification. The staff finds these bolting materials acceptable. The staff also independently verified the tabulated design values and found them acceptable.

3.1.4.6 Nonstructural Materials

Nonstructural materials such as the Boral panels and Holtite-A (a Holtec trade name for neutron shield material similar to NS-4-FR) are included in the structural analyses by weight only. Materials that function as neutron absorbers and gamma shields should be fabricated from materials that can perform well under conditions of service that are appropriate for these components over the design period. Boral has a long, proven history in worldwide nuclear service and use in other spent fuel storage and transportation casks. In accordance with NUREG-1536, only 75% credit is taken for the B^{10} in Boral. The SAR includes technical information and scientific studies on the NS-4-FR which show it will provide acceptable properties over the service life of the HI-STAR 100 system. The staff concludes Boral and NS-4-FR should not creep or slump to an extent that impairs the capability to perform its safety function during storage and accident conditions.

3.1.4.7 Brittle Fracture of Materials

The applicant considered the potential for brittle fracture, especially for cask system components that may be subject to impact during exterior handling and transfer operations. Alloy X, used exclusively for the MPC, is an austenitic stainless steel alloy. Thus, there is no transition temperature for brittle behavior as is found in ferritic carbon steels. For the ferritic steels used in the HI-STAR 100 overpack, SAR Section 9.1.2.3 specifies that each plate or forging for the helium retention boundary will be drop weight tested in accordance with Regulatory Guides 7.11 and 7.12 and Charpy V-notch tested in accordance with ASME Section III, Subarticle NB-2300. The staff concludes that with the alloys selected and testing performed, no restrictions regarding cask handling at low temperatures are necessary. The HI-STAR cask does not use any impact limiters during spent fuel storage.

3.1.4.8 Materials Conclusion

The staff concludes that the materials for construction of the HI-STAR 100 system are acceptable for the described structural, thermal, shielding, criticality, and confinement functions.

3.1.5 General Standards for Cask

The structural analyses for the cask must ensure positive closure, adequate safety factors for lifting devices, and that there is no adverse effect to the safe storage of the spent fuel due to chemical or galvanic reactions. However, the most important function of structural analysis is to show sufficient structural capability of the HI-STAR system to withstand the postulated worst-case loads under normal, off-normal, accident, and natural phenomena events with adequate margins of safety to preclude the following negative consequences:

1. unacceptable risk of criticality,
2. unacceptable release of radioactive materials,
3. unacceptable level of radiations, and
4. impairment of ready retrievability.

3.1.6 Supplemental Data

Additional codes and standards referenced in the HI-STAR 100 system design and fabrication are listed in SAR Section 3.6.1. Three computer codes have been utilized to perform structural and mechanical analysis for the HI-STAR 100 systems. These codes are ANSYS, DYNA3D, and WORKING MODEL. The ANSYS and DYNA3D Codes are public domain codes with a well-established history of usage in the nuclear industry. WORKING MODEL V3.0 is a computer aided engineering tool and is commercially available. Holtec has performed an independent QA validation of the WORKING MODEL V3.0 Code by comparing the solution of several classical problems with the numerical results predicted by WORKING MODEL V3.0.

3.2 Normal Operating and Off-Normal Conditions

3.2.1 Chemical and Galvanic Reactions

The MPC is filled with helium. The inert gas environment in the MPC will effectively eliminate corrosion during storage. Experience has shown that Boral and stainless steel or aluminum and stainless steel will have no significant chemical or galvanic reactions in dry or wet service conditions. Also, all aluminum surfaces will be pre-passivated or anodized before installation to preclude aluminum-water reaction inside the MPC during fuel loading operations.

The overpack is constructed from low alloy and nickel alloy steels, carbon steels, bolting materials, Holtite-A, and thermal expansion foams. None of these materials have known chemical or galvanic reactions in close proximity of each other. Pursuant to NRC Bulletin 96-04, a review of the HI-STAR 100 system, its contents, and operating environments has been performed to confirm that no operation (e.g., short-term loading/unloading or long-term storage) will produce adverse chemical or galvanic reactions (Table 3.4.2 of the SAR).

3.2.2 Positive Closure

The MPC is an all welded structure. Thus, the stored spent nuclear fuel is not accessible when the MPC confinement boundary is not breached. In addition, the only access to the MPC is through the overpack closure plate. The closure plate weighs more than 7,000 lb and is fastened to the top forging with 54, 1-5/8-inch diameter closure bolts. Inadvertent opening of the overpack closure system is not possible. Therefore, the HI-STAR 100 Cask System design has provided positive closure.

3.2.3 Lifting Devices Analysis

The loaded overpack is lifted by two trunnions located on the overpack top forging. Threaded holes are provided on the MPC top plate for four, 1-3/4-inch, SB637-N07718, eyebolts capable of lifting a loaded MPC. The overpack top plate is lifted by four, 5/8-inch eyebolts threaded into the top plate and the unloaded MPC enclosure vessel is lifted by four, 3/4-inch thick lifting lugs welded to the vessel shell.

3.2.3.1 Overpack Lifting Devices

The overpack lifting trunnions are designed to meet the requirements of NUREG-0612 with a safety factor of 6 against material yield strength and 10 against material ultimate strength. The design has conservatively assumed an additional dynamic factor of 0.15 on the design lift weight for slow lifting operations. The analyses considered bending and shearing stresses of the trunnion and the adequacy of embedment length of the trunnion into the top forging material. The overpack top plate lifting bolts are conservatively designed to meet the requirements of NUREG-0612 for a non-redundant lifting system. The analyses assumed a minimum lifting angle of 50 degrees from the horizontal and considered bolt lifting capacity, shearing capacity, and embedment length based on the allowable design stresses.

3.2.3.2 MPC Lifting Devices

The loaded MPC can be inserted or removed from an overpack by the lifting bolts installed on the top lid. The strength of the lifting bolts and the base metal are evaluated to meet the requirements of NUREG-0612 for non-redundant lifting systems. The evaluation included an additional dynamic load factor of 0.15 for the loaded MPC and checked the adequacy of the thread engagement length in the top lid. The evaluation results showed that the lifting bolts provided more than 6 times the material yield strength and 10 times the material ultimate strength when lifting a fully loaded MPC. An unloaded MPC vessel is lifted by 4 lifting lugs welded to the vessel shell. The application provided analysis showing that the stresses in the lifting lugs and the connection welds are within the allowable stresses with adequate safety margins.

3.2.4 Hot and Cold Temperature Effects

The thermal evaluation of the HI-STAR 100 Cask System is reported in Section 4 of this SER. The hot and cold temperature effects include thermal gradients and pressure, differential thermal expansions among cask components, and brittle fracture considerations under cold temperature conditions.

3.2.4.1 Internal Pressures and Temperatures

Design pressures and design temperatures for the HI-STAR Cask System are listed in Tables 2.2.1 and 2.2.3 of the SAR. The design pressures and design temperatures are obtained by assuming bounding conditions. Bounding temperature distributions are used for the MPC and overpack finite element thermal stress calculations to maximize the stresses developed due to thermal gradients. The combined thermal and pressure stresses are well below the allowable stresses of the MPC and overpack structure.

3.2.4.2 Differential Thermal Expansion

Simple conservative calculations were performed by Holtec to demonstrate that the initial gaps between the HI-STAR 100 overpack and the MPC canister, and between the MPC canister and the fuel basket, will not be closed due to thermal expansion of the system components under normal, off-normal, and accident conditions. Therefore, it is concluded that the clearances provided between the MPC basket and canister shell, as well as that between the MPC shell and storage overpack, are sufficient to allow free expansion, and temperature-induced interference between components will not occur.

The fire accident has little effect on the MPC temperature because of the short duration of the fire and the large thermal inertia of the overpack. However, the growth of the fuel basket during and after the fire accident was evaluated to show that the fuel basket will not contact either the canister shell or the MPC lid due to free thermal growth. In addition, the applicant performed an analysis to show that the closure bolts of the overpack structure will not lose their pretension during the storage fire accident. Thus, the helium retention capability of the overpack structure will not be adversely affected by the fire accident event.

3.2.4.3 Cold Temperature

The temperature gradients in the cask system under steady-state conditions will remain the same regardless of the value of the ambient temperature. This is because the steady-state temperature of all components in the cask system will go up or down by the same amount of change in the ambient temperature. However, the internal pressure will decline with the lowering of the ambient temperature. Since stresses in the cask components are calculated from pressure and thermal gradients, it follows that the stresses under cold temperature conditions would be smaller than the hot condition and would not control the design.

The structural material used in the MPC canister and basket (Alloy X) is austenitic stainless steel and is not susceptible to brittle fracture. The material for overpack baseplate, top flange, and closure plate is SA350-LF3. The material for overpack inner shell is SA203-E. These materials are selected due to their capabilities to perform at low temperatures with excellent ductility properties. The applicant performed an analysis to derive the fracture toughness criteria for closure bolts, intermediate shells, and the pocket trunnions to meet the requirements of Regulatory Guides 7.11 and 7.12. The material fracture toughness test criteria for overpack components are listed in Table 3.1.18 of the SAR.

3.2.5 Cask and MPC Structural Analysis

The thermal and pressure load stress analysis for the MPC and the overpack was discussed in the above section. The handling loads must be considered for normal conditions during the lifting operation of a fully loaded MPC and the overpack. The handling loads for off-normal conditions are identical to normal operating conditions, and thus, a separate analysis is not required.

3.2.5.1 MPC Analysis for Lifting Operation

The MPC is evaluated for the handling operation under the normal operating condition. A normal handling operation is a vertical lift of a fully loaded MPC by the lifting bolts installed in threaded holes (with all threads engaged) in the MPC top lid. As discussed above, under lifting devices, the strength requirements of the lifting bolts and the embedment length were evaluated to meet the requirements of NUREG 0612 with a safety factor of 6 for material yield strength and 10 for the ultimate strength. The applicant performed hand calculations and finite element analysis to show that the stresses in the MPC (i.e., top lid, top lid peripheral welds, and the baseplate) meet the ASME Code, Subsection NB, Level A requirements for the lifting operation. In addition, the MPC was also evaluated for 3 times the design lifting load, and the resulting stresses in the MPC were shown to be less than the material yield stress.

3.2.5.2 Overpack Analysis for Lifting Operation

The overpack structure is evaluated for the normal lifting operation. The two lifting trunnions are circumferentially spaced at 180 degrees on the top flange. The trunnions were designed for a two-point lift and in accordance with the requirements of NUREG-0612 as described in Section 3.2.3.1 above.

The applicant performed finite element analysis for the top flange to show that the maximum membrane stress and the membrane plus bending stress in the top flange are below the material yield strength when subject to 3 times the lifting load. The overpack cask baseplate is conservatively modeled as a simply supported circular plate, and the MPC dead load is applied as a ring load. The calculated maximum membrane plus bending stress in the overpack baseplate when subject to 3 times the MPC load is lower than the allowable stresses. Thus, the overpack top flange and baseplate will not exceed their yield strength when subject to 3 times the lifting dead load.

3.3 Accident Conditions

3.3.1 Cask Tipover and Side Drop

As discussed in Section 3.3.4 below, the HI-STAR 100 Cask System does not tipover as a result of the design-basis extreme environment events (e.g., tornado and tornado-generated missiles, floods, and earthquake). The tipover accident is a postulated, non-mechanistic event for the purpose of demonstrating the defense-in-depth of the HI-STAR 100 Cask Systems. The cask, however, may be transported to the ISFSI in a horizontal configuration and thus, may be subject to a side drop accident event. Although it is highly unlikely, cask side drop from the handling height limit is considered as a credible accident event.

The HI-STAR 100 Cask System design-basis deceleration for the side drop and tipover accident events is 60 g (Table 3.1.2 of the SAR). The design-basis deceleration g-load is derived from the time-history results of the cask-pad-soil interaction finite element analysis using the LS-DYNA3D computer code. To validate the finite element analysis model and the LS-DYNA3D Code, the applicant performed a benchmarking LS-DYNA3D analysis in a separate benchmark report. The applicant's benchmark report utilized the same billet material and concrete pad as those provided in the Lawrence Livermore National Laboratory (LLNL) billet test report. The analysis assumed the billet as a rigid body and evaluated the side and tipover drops against the test data provided in the LLNL report. The benchmark analysis results have shown to be enveloping the LLNL published billet test results. Based on the comparisons of the analysis and test results, the staff concurs that the applicant has adequately validated the finite element modeling technique and the LS-DYNA3D computer code.

The tipover and side drop impact analysis is provided in Appendix 3.A of the SAR. The HI-STAR 100 cask is conservatively assumed to be a rigid body. The ISFSI pad and the soil foundation are assumed to duplicate the concrete pad and soil in the LLNL billet test report. Essential parameters of the referenced ISFSI pad and soil are listed in Table 3.A.1 of the SAR. The handling height limit for the side drop is determined to be 72 inches and the maximum deceleration of the cask as a result of a 72-inch side drop is shown to be 48.3 g. The deceleration for tipover is proportional to the distance from the center of rotation. Thus, it will vary along the axial length of the cask. The maximum tipover deceleration is determined to be 61.84 g at the top end of the cask. However, at the top of the MPC, the maximum tipover deceleration is 56 g which is below the design base limit of 60 g.

A three-dimensional finite element model of the HI-STAR 100 overpack cask is constructed using the ANSYS computer code. The finite element stress analysis is based on the quasi-static

methods of structural analysis. The effects of dynamic loads are included by showing that the safety factors provided are larger than the dynamic load factors for the drop impact loading conditions. The side drop accident event with the bounding design deceleration of 60 g is evaluated for the overpack cask stress analysis. In a side drop, with the cask axis horizontal, the entire cask will be subject to the 60 g deceleration. In a tipover, the deceleration varies according to the distance from the center of rotation, and the maximum deceleration at the far end of the cask is 61.84 g which is only slightly higher than the 60 g design-basis deceleration. Therefore, the side drop event stresses will bound the tipover accident event. The deceleration load is imposed by multiplying the dead load (gravity) by the design-basis deceleration (60 g) value. The MPC dead load is represented by applying a pressure load on the inner surface of the overpack. The pressure is applied as a sinusoid with a total distribution of 18 degrees in the circumferential direction, a maximum value at the line of symmetry, and uniform along the axial length of the inner surface of the model. The overpack analysis procedures may be summarized as follows:

- The stress and deformation due to each individual load are determined.
- The results for each individual load are combined for loading combinations. Load cases for HI-STAR 100 overpack is listed in Table 3.1.5 of the SAR.
- The results for each loading combination are compared to the allowable values and the margin of safety is determined (Table 3.4.10 of the SAR).

The side drop impact stresses are combined with the stresses resulting from the overpack design pressure load and the closure bolt pre-load. The combined stresses in the overpack for a 72-inch side drop is shown to be less than the design allowable stresses with a minimum safety factor of 1.70 for the helium retention boundary.

3.3.2 Cask Bottom-End Vertical Drop

The HI-STAR 100 overpack cask may be transported to the ISFSI in a vertical configuration. Thus, a cask bottom-end vertical drop is considered as a credible handling accident event. The cask is evaluated for a 21-inch bottom-end vertical drop onto the ISFSI pad. Similar to the side drop event, the end drop analysis also assumed the overpack as a rigid body and that the ISFSI pad and soil foundation duplicate the pad and soil in the LLNL test report. The impact analysis is performed by using the cask-pad-soil interaction model and the LS-DYNA3D computer code. The analysis showed that, for a bottom-end vertical drop at the handling height limit of 21 inches, the corresponding maximum deceleration of the cask is 51.9 g.

The end drop stress analysis is conservatively based on the design-basis deceleration of 60 g. A three-dimensional finite element model of the cask is used, and the weight of the cask is amplified by the design-basis deceleration. The pressure load on the inside face of the overpack bottom plate is the weight of the heaviest fully loaded MPC amplified by the design-basis deceleration. The stress analysis is performed by the ANSYS computer code. The analysis has shown that the stresses in the overpack are within the allowable stress, and the minimum safety factor is equal to 1.27 at the center of the overpack baseplate.

In addition to the finite element analysis, the overpack cask is evaluated for buckling according to ASME Section III, Code Case N-284. The stability analysis of the HI-STAR 100 overpack is carried out in Appendix 3.H of the SAR. The loading cases of normal handling, the accident end drop, and the accident external pressure plus dead weight are evaluated for elastic and plastic stability. It is shown that the cask will not buckle under the normal and accident loadings even when fabrication stresses are included in the calculations. The values calculated by the interaction equations are all less than 1.0 indicating that it is the yield strength limit rather than instability governing the design.

3.3.3 Cask Lid Bolt Analysis

Stresses are induced in the closure bolts due to pre-load, pressure loads, differential thermal expansion, and accident loads. The analysis of the overpack closure bolts under normal and accident conditions appropriate for storage is carried out in Appendix 3.F of the SAR. The analysis method follows the procedures delineated in NUREG/CR-6007. The following load cases have been analyzed:

(a) Normal Condition: Pressure, temperature, and bolt pre-load loads are included.

(b) Puncture: All the normal condition loads plus a 6-inch diameter puncture load.

(c) Drop: All the normal condition loads plus the 60 g top end drop impact loads.

The cask lid bolt analysis is conservatively based on a nearly vertical (i.e., 80 degrees) top end drop and the lid bolts must support the weight of both the lid and a fully loaded MPC amplified by the design-basis deceleration. The top end drop is not a credible event for storage casks but it provides the bounding case for the HI-STAR 100 overpack closure bolts. Based on stress limits specified in NUREG/CR-6007, the closure bolt analysis performed in Appendix 3.F of the SAR showed that the minimum safety factor is 1.08 on bolt tension for the normal condition and 1.30 for the accident top end drop condition.

3.3.4 MPC and Fuel Basket Analysis

The MPC fuel basket and enclosure vessel are evaluated for the load combinations in Tables 3.1.3 and 3.1.4 of the SAR, respectively. The evaluation includes three areas: (1) finite element analysis of the MPC fuel basket and enclosure shell under normal and accident loads, (2) elastic stability analysis of the MPC fuel basket and enclosure shell under axial and lateral loads, and (3) analysis of the fuel support spacers under compression loads.

3.3.4.1 Finite Element Analysis

The finite element model of each MPC type (i.e., MPC-68 and MPC-24) is used to assess the effects of the normal, off-normal, and accident loads. The finite element model is a 1-inch thick cross section of the MPC. It is a two-dimensional structural model that includes the fuel basket, the basket support structures, and the enclosure shell. The analysis considered two impact orientations of the fuel basket, one 0-degree and one 45-degree, for the accident side drop. The design-basis deceleration of 60 g is used to perform a quasi-static structural analysis using the ANSYS computer code. The weights of the fuel assemblies are applied to the basket fuel cell as

a uniformly distributed pressure load. The weight of fuel assemblies, the basket, and the enclosure shell are all amplified by the design-basis deceleration of 60 g for the side drop accident. The resulting stresses are compared with the allowable stresses and the safety factors are computed in Appendix 3.AA of the SAR. Table 3.4.3 of the SAR listed the minimum safety factors for the fuel basket based on loading combinations and the calculated stresses. Similarly, Table 3.4.4 of the SAR listed the minimum safety factors for the enclosure vessel and basket support structure according to the loading combinations and calculated stresses. Since the safety factors are larger than the dynamic load factors calculated in Appendix 3.X, the dynamic load effects have been met.

3.3.4.2 Stability Analysis

The MPC basket and enclosure shell are evaluated for buckling under compressive loads. The governing loading conditions are the side and end drops. The applicant performed buckling analyses using the ANSYS code's large deformation capability for the side drop condition. The finite element model used for the large deformation analysis is identical to the model used for stress analysis. The large deformation analysis results have shown that the collapse load of the MPC basket is greater than 1.5 times the inertia load corresponding to the design-basis deceleration (60 g) for the three types of baskets. In addition, the most critically stressed basket panel is evaluated according to NUREG/CR-6332 to show that local panel buckling will not occur. The MPC-24 basket panel has the smallest ratio of thickness to length and is also subject to the highest compressive stress as determined by the finite element stress analysis. Therefore, the local buckling analysis is based on the MPC-24 basket. The critical buckling stress calculated using the geometry of the MPC-24 is 49.22 ksi. Since the finite element panel stress is 13.717 ksi, the safety factor against basket panel buckling is 49.22/13.717 = 3.59, which is higher than the recommended safety factor of 2.2 for stainless steel members under accident condition loads.

For the end drop condition, stability of the basket panels under axial deceleration loading is evaluated. The weight of each basket (including sheathing and Boral) is multiplied by the design-basis deceleration (60 g) and then divided by the basket bearing area on the MPC baseplate to obtain the average axial stress in the basket panel. The critical axial stress is calculated as 64.0 ksi which is more than 15 times larger than the average axial panel stress. Therefore, it is concluded that buckling of the fuel basket will not occur for the end drop condition.

The stability of the MPC enclosure vessel is also evaluated for combined external pressure and compressive load introduced by the accident vertical drop. Stability evaluations are performed in accordance with ASME Code Case N-284 and the analyses are provided in Appendix 3.H of the SAR. The interaction equations for the ASME Code Case N-284 are shown to give results less than 1.0 for all loading conditions. Therefore, stability of the MPC enclosure vessel is assured.

3.3.4.3 Analysis of Fuel Support Spacers

Upper and lower fuel support spacers are utilized to position the active fuel region of the spent fuel within the poisoned region of the fuel basket. It is necessary to ensure that the spacers will not buckle or yield under the maximum compressive load. Detailed calculations are provided in

Appendix 3.J of the SAR. The analysis has shown large structural margins for stability and strength for the entire range of spacer lengths that may be used in HI-STAR 100 Cask Systems.

3.4 Extreme Natural Phenomena Events

3.4.1 Flood Condition

The flood condition subjects the HI-STAR 100 system to external pressure, together with a horizontal drag force due to water velocity. The HI-STAR 100 Cask System design external pressure bounds the pressure during a flood. However, the horizontal drag force may cause sliding or overturning of the cask. The applicant provided an analysis that was based on a drag coefficient of 0.5, a friction coefficient of 0.25, and design-basis flood water velocity of 13 ft/sec. It was shown that the safety factor is 3.0 against sliding and 5.0 against overturning. Consequently, it can be concluded that the HI-STAR 100 Cask System will not be adversely affected by the flood condition.

3.4.2 Seismic Events

The HI-STAR 100 system is subject to the design-basis seismic event consisting of three orthogonal statistically independent acceleration time-histories. The HI-STAR 100 system is conservatively considered as a rigid body subject to a set of horizontal inertia and vertical inertia forces to determine the design-basis earthquake that will not cause cask tipover. The analysis assumed the vertical seismic load is only a fraction of the horizontal seismic load and it is always acting in the upward direction. Based on the conservative static analysis, the acceptable design-basis earthquake input on the top surface of the ISFSI pad is listed below:

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	1.0 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 = 0.16 g
0.25 g	0.354 g	0.5 x 0.25 g = 0.125 g

3.4.3 Tornado Winds and Large Missile Impact

The analysis to determine the cask response to the combined load of the design-basis tornado wind and the tornado-generated large missile impact is provided in Appendix 3.C of the SAR. Per, Regulatory Guide 1.76, the design-basis tornado wind is specified as a maximum wind speed of 360 mph in Table 2.2.4 of the SAR. The tornado-generated large missile is defined as an automobile with a mass of 1800 kg and a speed of 126 mph. The analysis showed that the

cask will not slide or tip over as a result of the combined effects of tornado wind and missile strike. Therefore, the HI-STAR 100 system is acceptable for the design-basis tornado and large missile impact loads.

3.4.4 Tornado Missile Penetration

The penetration potential of the missile strikes is examined in Appendix 3.G of the SAR. Two missile strikes are investigated. The two missiles considered are a 1-inch diameter steel sphere and a 8-inch diameter rigid cylinder, both traveling at 126 miles per hour.

The 1-inch diameter sphere will dent any surface it impacts, but no significant puncture force is generated because of its small mass. The intermediate missile of an 8-inch diameter cylinder (125 kg) will cause considerable damage of the overpack but it will not penetrate the helium retention boundary of the overpack. Consequently, the MPC confinement function is not affected, and there will be no radiological releases associated with any missile strike during a tornado.

3.5 Evaluation Findings

F3.1 SSCs important to safety are described in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events.

F3.2 Storage systems are designed to allow ready retrieval of spent nuclear fuel for further processing or disposal. No accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The cask is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The staff concludes that the structural design of the HI-STAR 100 Cask System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the HI-STAR 100 Cask System will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted practices, and confirmatory analysis.

3.6 References

1. NRC Bulletin 96-04 "Chemical, Galvanic, or other Reactions in Spent Fuel Storage and Transportation Casks," July 1996.

2. U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 inches (0.1m)," Regulatory Guide 7.11, June 1991.

3. U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 inches (0.1m)," Regulatory Guide 7.12, June 1991.

4. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

5. NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks."

4.0 THERMAL EVALUATION

The thermal review ensures that the cask and fuel material temperatures of the HI-STAR 100 system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. This portion of the review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

4.1 Spent Fuel Cladding

The staff verified that the cladding temperatures for each fuel type proposed for storage are below the acceptable temperatures for which there would not be expected cladding damage that would lead to gross rupture. The applicant relied on the NRC-accepted PNL-6189 research report to establish cladding temperature limits based on internal fuel rod pressure. For the PWR fuels analyzed, the maximum allowable temperature was 720°F. For the BWR fuels analyzed, the maximum allowable temperature was 749°F.

For the short-term accident, the applicant used the limit of 1058°F from PNL-4835. This report is an NRC-accepted document and the basis for the maximum allowable accident temperature for spent fuel cladding in dry storage. The applicant also showed that the maximum temperature rise above the long-term limit was 37°F, well below the limit of 1058°F during any credible accident.

For cask unloading operations, the applicant considered the effect of cladding integrity during a quenching of the hot spent fuel in Section 4.4.1.1.15 of the SAR. The applicant designed the HI-STAR 100 MPCs to be cooled in a gradual manner, first by a closed loop with forced helium convection, and then by flooding the MPC with water. The helium gas circulation cools the MPC internals to less than 200°F and then the water is injected into the MPC without the risk of boiling and with minimal thermal stress effects to the cladding. The procedures for the unloading and quench process are explained in Section 8.3 of the SAR and provide reasonable assurance that the spent fuel cladding will be adequately protected during unloading operations.

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the HI-STAR 100 storage cask have been formulated by the applicant to assure that public health and safety will be protected during the period that spent fuel is stored in the cask. These design criteria cover both the normal storage conditions for the 20-year approval period and postulate accidents that last a short time, such as a fire.

Section 2.0.1 of the SAR defines several primary design criteria for the HI-STAR 100 components:

1) the allowable fuel cladding temperatures to prevent cladding degradation during long-term dry storage conditions for the MPC are based on the NRC-accepted research report, PNL-6189;

2) the short-term allowable cladding temperatures that are applicable to off-normal and accident conditions of storage are based on PNL-4835;

3) the design temperatures for the structural steel components of the MPC are based on the temperature limits provided in ASME Section II, Part D;

4) the different MPCs are bounded by a thermal source term based on fuel assembly type as noted in Appendix B to the CoC and Section 12 of the SAR; and

5) the overpack must perform its intended safety functions in the extreme cold conditions proposed during the life of the storage cask.

4.2.2 Design Features

To provide adequate heat removal capability, the applicant designed the HI-STAR 100 system with the following features:

1) helium backfill gas for heat conduction which also provides an inert atmosphere for the fuel to prevent cladding oxidation and degradation;

2) minimal heat transfer resistance through the basket by fashioning the basket like a honeycomb structure that is welded completely from the basket base to the top;

3) top and bottom plenums for transverse flow of the helium gas aiding in convective heat transfer;

4) continuous metal heat conduction axially provided by the basket structure;

5) flexible aluminum heat conduction elements for heat transfer from the basket periphery to the MPC shell; and

6) high emissivity paint on the overpack exterior surface to maximize radiative heat transfer to the environment.

The staff verified that all methods of heat transfer internal and external to the MPC and overpack are passive. Drawings in Section 1.5 of the SAR along with the material properties in Table 4.2.1-9 of Section 4 provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package as required by 10 CFR 72.24(c)(3).

4.3 Thermal Load Specifications

The design-basis fuel to be stored in the HI-STAR 100 cask is described in Table 2.1.6 of the SAR for both the BWR and PWR fuels. Based on ORIGEN-2 computer analyses, for the PWR fuels (loaded in the MPC-24 basket), the initial maximum average decay heat per assembly is 791.6 W for 5-year cooled fuel. In the MPC-24, this allows for a total package decay heat of 19.0 kW. For the BWR fuels (loaded in the MPC-68 basket), the initial maximum average decay heat per assembly is 272 W for 5-year cooled fuel. In the MPC-68, this allows for a total package decay heat of 18.5 kW. Both fuels vary in maximum burnup and initial enrichments due to the optimization of the allowable total heat load per basket. The burnup and cooling time limits are also specified in Table 2.1.6. The axial profiles for the design-basis fuels are listed in Table 2.1.8 of the SAR. The peak power in the BWR assemblies is a factor of 1.2 times the average power and, for the PWR assemblies, is a factor of 1.1 times the average power. The decay heat load will decrease with time over the storage period.

The staff has reviewed and confirmed the design-basis decay heat based on assembly type. The staff also verified that the design-basis decay heats are the bounding decay heats through independent analysis providing reasonable assurance that the decay heats were determined properly.

The thermal loads are different for the normal storage conditions than for the accident conditions, such as fire. The difference with the thermal loads occurs at the surface of the cask. The application of the surface thermal loads will be for a short time during an accident, while the surface thermal loads are applied continuously during normal storage conditions. The decay heat load during an accident will be the same as for the normal storage condition at the time of the accident.

4.3.1 Normal Storage Conditions

The external environment for normal storage conditions is described in Section 2.2.1.4 of the SAR. The applicant evaluated the cask for conditions with an annual average ambient temperature of 80°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.3.2 Off-Normal Conditions

The external environment for off-normal storage conditions are described in Section 2.2.2.2 of the SAR. The applicant evaluated the cask for conditions with an ambient temperature of 100°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The applicant also evaluated the cask for conditions with ambient temperatures of -40°F, no solar insolation, and no package decay heat.

4.3.3 Design-Basis Natural Phenomena

The extreme environment for the design-basis natural phenomena conditions are described in Section 2.2.3.13 of the SAR. The applicant evaluates the cask for conditions with an ambient

temperature of 125°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.3.4 Accident Conditions

A thermal accident postulated for the HI-STAR 100 storage cask is described in Section 2.2.3.3 of the SAR. A fire with an average flame temperature of 1475°F and an emissivity of 0.9 is hypothesized by the spillage and ignition of 50 gallons of combustible transporter fuel. Based on this amount of fuel, the fire was determined to have a duration of 5 minutes. The initial temperature distribution of the transient is based on the normal storage conditions. Following the fire, the cask is cooled by radiation and natural convection to an ambient temperature of 80°F.

4.3.5 Wet Transfer Conditions

The applicant considered the minimum time to boil during wet transfer. This practice avoids the potential for uncontrolled pressures due to the water boiling in the cask during transfer. By determining a bounding heat-up rate based on maximum heat load and minimum MPC volume, the applicant established the time before water in the cask would boil as a function of the initial temperature of the HI-STAR 100 contents when removed from the pool. The results of the analysis are presented in Table 4.4.21 of the SAR.

4.4 Model Specification

4.4.1 Configuration

A detailed analytical model for thermal design of the HI-STAR 100 system was developed using the FLUENT finite volume Computational Fluid Dynamics code and the industry standard ANSYS modeling package. Transport of heat from the fuel assemblies to the outside environment is analyzed in terms of three interdependent thermal models. The first model considers transport of heat from the fuel assembly to the basket cell walls. The second model considers heat transport within an MPC cross section by conduction and radiation. The third model deals with the transmission of heat from the MPC exterior surface to the external environment through the overpack. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the vertical and top cover surfaces.

4.4.1.1 Fuel Assembly Model

Heat energy transport within the cross section of the fuel assembly was modeled as a combination of radiative energy exchange and conduction through the helium gas that filled the interstices between the fuel rods in the array. First, the applicant used an analytic procedure to calculate the effective thermal conductivities determining the most resistive PWR and BWR fuel types in a conservative manner. The W-17x17 OFA PWR and GE11-9x9 BWR fuel assemblies are determined to be the bounding configurations for analysis of zircaloy-clad fuel at design-basis maximum heat loads. Then, conduction-radiation finite volume models of the bounding PWR and BWR fuel assemblies were developed in the FLUENT Code. The combined

fuel rod-helium matrix was replaced by an equivalent homogeneous material which fills the basket opening by the following two-step procedure. In the first step, the FLUENT-based fuel assembly model is solved by applying an equal heat generation per unit length to the individual fuel rods and a uniform boundary temperature along the basket cell opening inside the periphery. Then the temperature difference across the assembly is used to calculate the effective conductivity of the entire assembly region based on the smeared uniform heat source. The results of the FLUENT Code for the PWR and BWR design-basis fuel assemblies are displayed in Figure 4.4.14 of the SAR.

4.4.1.2 MPC Cross Section Model

The heat rejection capability of each MPC basket design is evaluated by developing a thermal model of the combined fuel assemblies and composite basket wall geometry with the ANSYS finite element code. The ANSYS model includes a geometric layout of the basket structure in which the basket composite wall is replaced by a homogeneous wall with an equivalent thermal conductivity. The fuel assemblies and the surrounding basket cell openings are modeled as homogeneous heat generating regions with effective temperature dependent in-plane conductivity. The two-dimensional ANSYS finite element model of each MPC basket is solved by applying a uniform heat generation per unit length in each basket cell region and a constant basket periphery boundary temperature. Table 4.4.7 of the SAR summarizes the effective thermal conductivity results of each basket design obtained from the ANSYS models.

4.4.1.3 Overpack Model

From the MPC shell to the overpack exterior surface, heat is conducted though an array of concentric shells representing the MPC-to-overpack helium gap, the overpack intermediate shells, the Holtite-A neutron shield, and the overpack outer shell. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the vertical and top cover surfaces. The bottom overpack face, in contact with the ISFSI pad, rejects a small quantity of heat by conduction through the pad to the ground. The analysis also accounts for the reduction in radiative heat exchange between the cask exterior vertical surfaces because of blockage from the neighboring casks arranged for normal storage in a regular square array on the ISFSI pad. The overpack closure plate is modeled as a heated surface in convective and radiative heat exchange with air and as a recipient of heat input through insolation. Insolation on the cask surfaces is based on 12-hour levels prescribed in 10 CFR Part 71, averaged over a 24-hour period, after accounting for partial blockage conditions.

4.4.1.4 Radiation from Overpack Exterior Surfaces

The applicant considered the thermal interaction among casks in an array using the ANSYS finite element code. The ANSYS solution determined view factors between the most adversely located system in the middle with all other neighboring casks. This determined the most conservative view factor for the cask surface.

4.4.2 Material Properties

The material properties used in the thermal analysis of the storage cask system are listed in Section 4.2 of the SAR. The applicant provided material compositions and thermal properties for all components used in the calculational model. The material properties given reflect the accepted values of the thermal properties of the materials specified for the construction of the cask. For homogenized materials such as the basket walls of the MPC, the applicant described the manner in which the effective thermal properties were calculated.

4.4.3 Boundary Conditions

The boundary conditions include the total decay heat for each MPC and the external conditions on the cask surface. The axial peak power for the MPC-24 basket, for PWR fuels, is a factor of 1.1 times the average power of 19.0 kW. The peak power for the MPC-68 basket, for BWR fuels, is a factor of 1.2 times the average power of 18.5 kW.

The boundary conditions depend on the environment surrounding the cask. Four conditions are considered for each MPC basket. The first includes the conditions set forth for normal storage. The second case considers off-normal operation. The third case considers temperatures expected during natural phenomena. The fourth case considers the effect of a fire accident on the thermal performance of the cask.

4.4.3.1 Normal Storage Conditions

The external environment for normal storage conditions is described in Section 2.2.1.4 of the SAR. The applicant evaluated the cask for conditions with an ambient temperature of 80°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.4.3.2 Off-Normal Storage Conditions

The external environment for off-normal storage conditions is described in Section 2.2.2.2 of the SAR. The applicant evaluated the cask for conditions with an ambient temperature of 100°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The applicant also evaluated the cask for conditions with ambient temperatures of -40°F, no solar insolation, and no package decay heat.

4.4.3.3 Design-Basis Natural Phenomena

The extreme environment for the design-basis natural phenomena conditions is described in Section 2.2.3.13 of the SAR. The applicant evaluates the cask for conditions with an ambient temperature of 125°F and a 24-hour insolation period of 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively.

4.4.3.4 Accident Conditions

A thermal accident postulated for the HI-STAR 100 storage cask is described in Section 2.2.3.3 of the SAR. A fire with an average flame temperature of 1475°F and an emissivity of 0.9 is hypothesized by the spillage and ignition of 50 gallons of combustible transporter fuel. Based on this amount of fuel, the fire was determined to have a duration of 5 minutes. The initial temperature distribution of the transient is based on the normal storage conditions. Following the fire, the cask is cooled by radiation and natural convection to an ambient temperature of 80°F.

The heat input from the fire to HI-STAR 100 system is from a combination of radiation and convection heat transfer to all overpack exposed surfaces. The forced heat transfer coefficient is conservatively calculated using a flame and upward flow velocity of 15 m/s. Table 11.2.1 of the SAR provides a summary of parameter selections with justifications which provide the basis for application of the correlation to determine the forced convection heating to the cask.

After the fire event, the outside environment temperature is restored to initial ambient conditions and the HI-STAR 100 system transient analysis is continued, to evaluate temperature peaking in the interior during the post-fire cooldown phase. Heat loss from the outside exposed surfaces of the overpack is modeled through a combination of radiation and convection to the environment. The surface emissivity of the exposed carbon steel surface is used after the fire due to the paint being burned during the fire. Peak cooldown temperatures are listed in Table 11.2.2. along with the short-term temperature limits for cask components. The model for the post-fire is the same overpack model used for the fire analysis.

4.5 Thermal Analysis

4.5.1 Computer Programs

The complete thermal analysis is performed using the industry standard ANSYS finite element modeling package and the finite volume computational fluid dynamics code, FLUENT. Both codes are capable of general 3-D steady-state and transient calculations. The input and output file listings for the thermal evaluations are listed in Holtec's calculation packages: HI-971784, HI-971788, and HI-971789. The code inputs listed in the calculation packages were compared with the staff's confirmatory analysis to verify the HI-STAR 100 system results. The output from the code work was plotted in both the calculation packages and the figures in Section 4.4 of the SAR. Based on the SAR drawings, the staff performed confirmatory calculations using the ANSYS finite element code. Models were constructed which focused on the canister basket structure and MPC internals. Fuel assembly cells were homogenized and assembly heat generation rates were smeared over the individual cell regions. In addition, for highly detailed portions of the basket structure, homogenized thermal properties were developed for the model making it more realistic for a numerical solution. The temperature distributions generated by the staff's model displayed good agreement with the results presented by the applicant. Thus, the staff has reasonable assurance that the spent fuel storage system provided adequate heat removal capacity without an active cooling system as required by 10 CFR 72.236(f).

4.5.2 Temperature Calculations

4.5.2.1 Normal Storage Conditions

The two MPC basket designs developed for the HI-STAR 100 system have been analyzed to determine the temperature distribution under long-term normal storage conditions. The MPC baskets are considered to be loaded at design-basis maximum heat loads with PWR and BWR fuel assemblies, as appropriate. The systems are considered to be arranged in an ISFSI array and subjected to design-basis normal ambient conditions with insolation. The maximum allowable temperatures of the components important to safety are listed in Table 2.2.3 of the SAR. Low temperature conditions were also considered.

The maximum fuel clad temperatures for zircaloy-clad fuel assemblies are listed in Table 4.4.10 for the MPC-24 and in Table 4.4.11 for the MPC-68. The table below summarizes the applicant's temperatures of key components in the cask for various environmental conditions:

HI-STAR 100 Cask Component	Normal/Off-Normal Conditions [°F]			Accident Conditions [°F]		
	MPC-24 Normal Conditions	MPC-68 Normal Conditions	Off-Normal Conditions	Extreme Environmental Conditions	During 5-Minute Fire	Post Fire Cooldown
Fuel Cladding	709	741	761	786	741	771
MPC Basket	675	725	745	770	725	755
MPC Outer Shell Surface	332	331	352	377	331	364
Overpack Inner Surface	292	292	312	337	292	328
Neutron Shield Inner Surface	274	273	294	319	273	314
Overpack Outer Surface	229	228	249	274	854	854

Temperature criteria for the spent fuel cladding are discussed in Section 4.2 of the SER. The fuel cladding for both MPCs remain below their acceptable temperatures.

4.5.2.2 Off-Normal Conditions

The off-normal event considering an environmental temperature of 100°F for a duration sufficient to reach thermal equilibrium was evaluated by the applicant. The evaluation was performed with design-basis fuel with maximum decay heat and the most restrictive thermal resistance. The 100°F environmental temperature was applied with full solar insolation. All of the off-normal temperatures were below the short-term design-basis temperatures.

The off-normal event considering an environmental temperature of -40°F, no decay heat, and no solar insolation for a duration sufficient to reach thermal equilibrium was evaluated by the

applicant. The HI-STAR 100 system was conservatively assumed to reach -40°F throughout the structure. The structural analysis performed in Section 3.1.2.3 of the SAR by the applicant demonstrated that there was no reduction in the performance of the HI-STAR 100 system with respect to brittle fracture. Based on these analyses and review, the staff has reasonable assurance that the off-normal temperatures do not affect the safe operation of the HI-STAR 100 system.

4.5.2.3 Design-Basis Natural Phenomena

The extreme environmental conditions considering an environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium was evaluated by the applicant. The evaluation was performed with design-basis fuel with maximum decay heat and the most restrictive thermal resistance. The 125°F environmental temperature was applied with full solar insolation. All of the extreme environmental temperatures were below the short-term design-basis temperatures with the exception of the neutron shield material. However, the accident condition dose rate limits are shown to remain below the regulatory limit. Based on these analyses and review, the staff has reasonable assurance that the extreme environmental conditions do not affect the safe operation of the HI-STAR 100 system.

4.5.2.4 Accident Conditions

The applicant analyzed a fire accident on the HI-STAR 100 system using the conditions specified earlier in Section 4.4.3.4 of the SER. The peak temperatures of the key cask components due to 5-minute fire with a maximum decay heat based on the MPC-68 are shown in the table in Section 4.5.2.1 of the SER. The initial temperatures are based on the normal storage conditions and an incident solar heat flux based on the specified insolation averaged over 24 hours. During the fire, an upper bound material thermal conductivity for the neutron shield is assumed to maximize heat input to the cask. During the post-fire cooldown phase, no credit is taken for conduction through the neutron shield. As a result, all of the fire accident temperatures were below the short-term design-basis temperatures with the exception of the neutron shield material. However, the accident condition dose rate limits are shown to remain below the regulatory limit of a total dose of 5 rem accepted by the staff. Based on these analyses and review, the staff has reasonable assurance that the cladding integrity will not be compromised during the fire or post-fire cooldown.

4.5.3 Pressure Analysis

4.5.3.1 Normal Conditions of Storage

The applicant considered the pressure in the MPC based on the average cavity gas temperature of 439°F and the normal storage conditions with an ambient air temperature of 80°F. The cask is sealed and backfilled with 28.3 psig of helium at 70°F. During normal conditions, a conservative estimation of the maximum pressure was based on the rupture of 1% of the fuel rods. Under these conditions and with the average cavity gas temperature, the applicant determined a maximum pressure of 58.7 psig. This pressure is bounded by the design pressure for normal/off-normal conditions of 100 psig.

4.5.3.2 Off-Normal Conditions

The applicant considered the pressure in the MPC based on the average cavity gas temperature of 459°F and the off-normal storage conditions with an ambient air temperature of 100°F. The cask is sealed and backfilled with 28.3 psig of helium at 70°F. During off-normal conditions, a conservative estimation of the maximum pressure is the rupture of 10% of the fuel rods. Under these conditions and with the average cavity gas temperature, the applicant determined a maximum pressure of 63.8 psig. This pressure is bounded by the design pressure for normal/off-normal conditions of 100 psig.

4.5.3.3 Design-Basis Natural Phenomena and Accident Conditions

The applicant considered the pressure in the MPC based on the average cavity gas temperature of 655°F and the normal storage conditions after the fire accident with an ambient temperature of 80°F. The cask is sealed and backfilled with 28.3 psig of helium at 70°F. During accident conditions, a conservative estimation of the maximum pressure was based on the rupture of 100% of the fuel rods. Under these conditions and with the average cavity gas temperature, the applicant determined a maximum pressure of 122.5 psig. This pressure is bounded by the design pressure for accident conditions of 125 psig.

4.5.4 Confirmatory Analysis

The confirmatory analysis of the HI-STAR 100 storage cask SAR can be divided into six categories: (1) review the models used in the analyses, (2) review the material properties used in the analyses, (3) review the boundary conditions and assumptions, (4) perform independent analyses to confirm the applicant's analyses, (5) compare the results of the analyses with the applicant's design criteria, and (6) assure that the applicant's design criteria will satisfy the regulatory acceptance criteria and regulatory requirements.

The staff reviewed the models used by the applicant in the thermal analyses. The code inputs in the calculation packages were checked for consistency to confirm that the applicant used the appropriate material properties and boundary conditions where required. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the code model. The material properties presented in the SAR were reviewed to verify that they were appropriately referenced and used conservatively. In addition, the staff performed a confirmatory analysis of the thermal performance of the cask SSCs identified as important to safety. A detailed model of the fuel regions and basket geometry was developed using the ANSYS finite element code to ensure that the SAR results were realistic and conservative. Independent homogenized thermal resistances were determined for the confirmatory calculation and employed in the model. The temperature distributions generated by the staff's model displayed agreement with those values determined by the applicant.

The staff has determined that the thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the HI-STAR 100 system is designed with a heat-removal capability having test ability and reliability consistent with its importance to safety.

Based on the applicant's analyses, there is reasonable assurance that the HI-STAR 100 system provides adequate heat removal capacity without active cooling systems. The staff also has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below the allowable criteria and by providing an inert environment in the cask cavity, thus assuring that the fuel can be readily retrieved for future processing or disposal without significant safety problems.

The staff has further concluded that the design of the heat removal system of the HI-STAR 100 cask is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal system design provides reasonable assurance that the HI-STAR 100 will enable safe storage of spent fuel. This finding is based on a review which considered the requirements of 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted practices.

4.6 Evaluation Findings

F4.1 Thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness [10 CFR 72.24(c)(3)].

F4.2 The staff has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity [10 CFR 72.122(h)(1)].

F4.3 Through the analysis, staff developed reasonable assurance that the HI-STAR 100 system is designed with a heat-removal capability having testability and reliability consistent with its importance to safety [10 CFR 72.128(a)(4)].

F4.4 By analysis, the staff has reasonable assurance that the decay heat loads were determined appropriately and accurately reflect the burnup, cooling times, and initial enrichments specified [10 CFR 72.122].

F4.5 By analysis, the staff has reasonable assurance that the HI-STAR 100 system provides adequate heat removal capacity without active cooling systems [10 CFR 72.236(f)].

F4.6 By analysis, the staff has reasonable assurance that the temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly [10 CFR 72.122].

5.0 SHIELDING EVALUATION

The shielding review is to evaluate the capability of the HI-STAR 100 shielding features to provide adequate protection against direct radiation from its contents. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d) [1,2]. Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is evaluated in Section 10 (Radiation Protection) of the SER.

The shielding review focuses on the calculation of the dose rates from both gamma and neutron radiation at locations near the cask and at assumed distances away from the cask. Off-site dose estimates include the dose contribution from the calculated direct radiation dose rates in Section 5 of the SAR. In addition, Section 10 of the SAR presents estimated occupational exposures that are based on the dose rates calculated in Section 5 of the SAR.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The HI-STAR 100 is designed to provide both gamma and neutron shielding. As discussed in Sections 1.2, 1.5, and 5.1 of the SAR, the principal components of the radial gamma shielding are the 0.5-inch thick steel MPC shell, the 2.5-inch thick steel inner shell of the overpack, and the five intermediate steel shells welded to the inner shell with an equivalent thickness of 6 inches. Gamma shielding at the bottom of the cask is provided by the 2.5-inch thick MPC baseplate and the 6-inch thick overpack bottom plate. The gamma shielding at the top of the cask consists of a 9.5-inch thick MPC lid for the MPC-24, a 10-inch lid for the MPC-68 (and 68F), and a 6-inch thick overpack closure plate.

In addition to the steel components discussed above, neutron shielding is provided by a solid borated polyester resin poured into radial channels that surround the cask body. The thickness of the neutron shield material is approximately 4.3 inches. The neutron shield material is Holtite-A with a nominal hydrogen content of 6.0 weight percent and a minimum B_4C content of 1.0 weight percent.

5.1.2 Shielding Design Criteria

Sections 2.0.2 and 5.1.1 specify design criteria for the surface dose rate on the side of the cask to be less than 125 mrem/hr and the surface dose rate on the cask flange to be less than 375 mrem/hr. In addition, a proposed TS in Appendix 12A of the SAR specifies the top of the cask to have a dose rate less than 80 mrem/hr. Overall design criteria for the HI-STAR 100 are the regulatory dose limits and requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

The staff evaluated the HI-STAR 100 shielding design features and design criteria and found them acceptable. The SAR analysis provides reasonable assurance that the shielding design features and design criteria can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Cask surface dose limits are included in TS 2.2.1 of the CoC.

The staff evaluates the overall radiation protection design features and design criteria of the HI-STAR 100 in Section 10 of the SER.

5.2 Source Specification

5.2.1 Source Specification

The source specification is presented in Section 5.2 of the SAR. Gamma and neutron source terms were generated with the SAS2H and the ORIGEN-S modules of SCALE 4.3, using the 44-group cross-section library[3]. The designs specified in Tables 2.1.3 and 2.1.4 were examined, and design-basis source terms were specified for zircaloy and stainless-steel-clad fuel for both the MPC-24 and MPC-68. The examination included the gamma and neutron sources from the fuel and the gamma sources from activated assembly hardware. In order to obtain conservative source terms, the design-basis source terms were calculated using the initial enrichments listed in SAR Table 5.2.23 for each burnup range examined in the design-basis fuel analysis. These initial enrichment values for each burnup range are lower than the average initial enrichment values specified in EIA Service Report SR/CNEAF/96-01[4].

The characteristics of the Babcox & Wilcox 15x15 assembly are described in SAR Table 5.2.1. Its burnup and cooling times are 40,000 MWD/MTU for 5 years and 45,700 MWD/MTU for 8 years. This assembly was chosen as the design-basis fuel for zircaloy-clad PWR fuel (MPC-24). The source terms for this fuel are listed in SAR Tables 5.2.4, 5.2.9, and 5.2.12. The source terms do not explicitly include gamma radiation from activated fuel-region hardware (e.g., space grids). In order to compensate for activated fuel-region hardware, the SAR analysis assumed an initial uranium loading higher than that specified in SAR Table 2.1.3.

The characteristics of the Westinghouse 15x15 stainless steel clad fuel assembly are described in SAR Table 5.2.18. Its burnup and cooling times are 30,000 MWD/MTU for 9 years and 45,000 MWD/MTU for 15 years. This assembly was chosen as the design-basis fuel for the stainless-steel-clad PWR fuel (MPC-24). The source terms for this fuel are listed in SAR Tables 5.2.20 and 5.2.22.

The characteristics of the General Electric 7x7 assembly are described in SAR Table 5.2.1. Its burnup and cooling times are 35,000 MWD/MTU burnup for 5 years and 45,000 MWD/MTU for 9 years. This assembly was chosen as the design-basis fuel for the zircaloy-clad BWR fuel (MPC-68). The source terms for this fuel are listed in SAR Tables 5.2.5, 5.2.10, and 5.2.13. The fuel assembly was also determined to have bounding source characteristics for the zircaloy-clad 6x6 MOX fuel, the zircaloy-clad damaged BWR fuel, and the zircaloy-clad BWR fuel debris (MPC-68F) for normal and accident conditions, with a maximum burnup and cooling time of 30,000 MWD/MTU for 18 years.

The characteristics of the Allis Chalmers 10x10 assembly are described in SAR Table 5.2.18. Its burnup and cooling time is 22,500 MWD/MTU for 10-years. This assembly was chosen as the design-basis fuel for the stainless steel-clad BWR fuel (MPC-68). The source terms for this fuel are listed in SAR Tables 5.2.19 and 5.2.21.

The staff performed a confirmatory analysis of the design-basis gamma and neutron source terms for the zircaloy-clad and stainless-steel-clad PWR and BWR fuels. The staff used SAS2H and ORIGEN-S of the SCALE 4.3 and the DOE Characteristics Data Base[3,5]. The staff examined the fuels listed in SAR Tables 2.1.3 and 2.1.4 of the SAR, and has reasonable assurance that the design-basis gamma source terms for the PWR and BWR fuels are adequate for the shielding analysis. The staff notes the SAR analysis assumed significantly higher burnups for its design-basis fuel than the maximum burnups (based on thermal limits) that will be allowed in the cask. In addition, the SAR analysis assumed a significantly higher initial uranium loading for the design-basis PWR fuel than the maximum uranium loadings specified in SAR Table 2.1.3.

The staff also notes neutron source terms may increase as initial ^{235}U enrichment decrease at a constant burnup. The staff accepts the lower-than-average initial enrichment values used to determine neutron source terms are adequate for each burnup level examined. The values bound a significant portion of discharged fuel and the source terms are calculated for burnups significantly higher than those allowed in the CoC. In addition, the source term and dose rates are adequately controlled by limits in the TS such as maximum burnup, minimum cooling time, maximum initial uranium loading, and maximum dose rates. Each general licensee will operate the cask under a 10 CFR Part 20 radiological program and perform site-specific dose evaluations to demonstrate compliance with 10 CFR Part 72 radiological requirements. Therefore, license conditions for minimum enrichment are not required for the HI-STAR 100 system for shielding purposes. Each general licensee should consider any potential fuel enrichments that are below those analyzed in the SAR when performing 10 CFR 72.212 written evaluations.

5.3 Model Specifications

5.3.1 Model Specifications

The model specifications for shielding are presented in Section 5.3 of the SAR. The shielding model for normal and accident conditions consisted of a three-dimensional representation of the HI-STAR 100 using the Design Drawings in Section 1.5. A description of the shielding configuration is presented in Section 5.3.1 of the SAR. Radial views of the shielding model are depicted in Figures 5.3.2, 5.3.3, and 5.3.9. An axial view of the shielding model is depicted in Figure 5.3.10.

5.3.1.1 Source Configuration

The shielding source is divided into five axial regions: lower-end piece, fuel, gas plenum springs, gas plenum spacer, and upper-end piece. Axial views of the relative position of the source term within the shielding configurations are depicted in Figures 5.3.7 and 5.3.8 for the

MPC-24 and MPC-68, respectively. The fuel region is modeled as a homogenous zone, and the end fittings and plenum regions are modeled as homogenous regions of steel. The SAR demonstrated that homogenization of the fuel assembly versus explicit modeling does not affect the accuracy of the shielding results. The axial distribution of the gamma source was assumed to follow the relative burnup profiles listed in SAR Table 2.1.8 for PWR and BWR fuel assemblies. The axial distribution of the neutron source was assumed to follow the relative burnup profiles raised to the power of 4.2. This adjustment was performed to account for the non-linear buildup of neutron source terms (primarily Cm-244) as a function of burnup. The total neutron source increased by 15.6% and 36.9% above the total neutron source values calculated for the average burnup of the PWR and BWR fuel, respectively.

5.3.1.2 Streaming Paths and Regional Densities

The shielding models included streaming paths for the cask radial fins and the pocket trunnions. As discussed in Section 5.3.1.2 of the SAR, the cask design eliminates other potential streaming paths. For example, the MPC lid is designed with a block of steel directly beneath the MPC vent port to prevent streaming.

The composition and densities of the materials used in the shielding analysis are presented in SAR Tables 5.3.2 and 5.3.3. Sections 4.4 and 4.5 of the SAR demonstrate that all materials used in the HI-STAR 100 remain below the design temperatures specified in SAR Table 2.2.3 during normal conditions. Therefore, the shielding analysis does not address changes in material density or composition from temperature variations, except for the neutron shield material. The neutron shield material is expected to experience water evaporation during normal conditions. Therefore, the neutron shield density was reduced in the shielding model to account for potential reduction in hydrogen density. The bounding accident condition for shielding assumes complete loss of the neutron shield. Therefore, the neutron shield material is replaced with a void for the accident shielding analysis.

The staff evaluated the SAR shielding model and found it acceptable. The model dimensions and material specifications provide reasonable assurance that the HI-STAR 100 was adequately modeled in the shielding analysis. The staff evaluation of material integrity of the shielding materials is in Section 3 of the SER.

The staff notes that it evaluated a neutron shield material with a thickness of approximately 4.3 inches, a density of 1.61 g/cm^3, a nominal hydrogen content of 6.0 weight percent, and a minimum B_4C content of 1.0 weight percent. Section 9.1.3 of the SER evaluates fabrication control and tests of the neutron shield.

5.4 Shielding Analyses

5.4.1 Shielding Analyses

The shielding analyses are presented in Section 5.4 of the SAR. The shielding analysis of the HI-STAR 100 was performed with MCNP-4A [6], a three-dimensional Monte Carlo transport code. The individual cross-section libraries used for each nuclide are based on ENDF/B-V

cross-section data. The SAR provides references for MCNP photon and neutron benchmarking problems against experimental data. Appendix 5C of the SAR contains a sample MCNP input file used in the shielding analysis. The SAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose-rate conversion factors to calculate dose rates in the shielding analysis.

5.4.1.1 Normal Conditions

The SAR presents calculations for normal condition dose rates of the design-basis zircaloy- clad fuels for the MPC-24 and MPC-68 at the cask locations shown in Figure 5.1.1 of the SAR. As shown in SAR Tables 5.4.8 and 5.4.9, the maximum dose rate calculated for the side and top of the MPC-24 configuration is approximately 120 mrem/hr and 90 mrem/hr, respectively. For the MPC-68 configuration, the side and top dose rates are 120 mrem/hr and 50 mrem/hr, respectively. The SAR also presents calculations for dose rates of the design-basis stainless steel-clad fuels at the cask surface midplane (Dose location no. 2) and demonstrates that the dose rates are lower than those for the zircaloy-clad fuel.

The SAR also compares the streaming dose rates through the overpack radial fins and pocket trunnion to the dose rates on the side of the cask.

5.4.1.2 Accident Conditions

Section 5.1.2 of the SAR presents calculations for accident condition dose rates of the design-basis zircaloy-clad fuels on the cask side and 1 meter from the cask side. SAR Tables 5.1.8 and 5.1.9 show maximum dose rates on the surface and at 1 meter are approximately 1370 mrem/hr and 490 mrem/hr, respectively. The 1 meter dose rate is extrapolated to 100 meters and is determined to be approximately 5 mrem/hr.

5.4.1.3 Occupational Exposures

The SAR analysis used the MPC-24 configuration with the design-basis fuel at 40,000 MWD/MTU burnup and 5-year cooling time to estimate occupational exposures during cask operations. Section 10 of the SAR presents estimated occupational exposures using the calculated dose rates for the locations shown in Figure 5.1.1.

5.4.1.4 Off-site Dose Calculations

The SAR analysis used the MPC-24 configuration with the design-basis PWR fuel at 40,000 MWD/MTU burnup, 5-year cooling time, to estimate dose rates beyond the controlled boundary. Cask array calculations assumed a design-basis fuel loading, a cask-to-cask pitch of 12 feet, level topography, and 100% occupation time. SAR Figure 5.1.2 presents estimated dose rate versus distance curves for various array configurations. SAR Table 5.1.7 specifies distances at which the 25 mrem/hr can be met for the various cask array configurations. For accident conditions, the SAR calculated the accident dose rate at 100 meters to be approximately 5 mrem/hr.

The staff performed confirmatory analysis of the calculated dose rates with the SAS4 module of SCALE 4.4 [7] and MCBEND9D [8]. The staff based its confirmatory calculations on the design

features and model specifications discussed above in Sections 5.1 and 5.3. The staff found that the calculated surface dose rates and estimated dose rates beyond the controlled area boundary are acceptable. The fuel characteristics in SAR Tables 2.1.3 and 2.1.4 and the burnup and cooling times listed in the TS in Appendix 12A of the SAR are included as a condition in the CoC, and the cask surface dose limits are included as a TS.

Section 10 of the SER evaluates the overall off-site dose rates from the HI-STAR 100. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by any site licensee. The general licensee using the HI-STAR 100 system must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask-array configuration, topography, demographics, and use of engineered features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 72.104(a) is the responsibility of each applicant for a site license.

The general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

A requirement has been included to Appendix B of the CoC regarding engineered features used for radiological protection. The requirement states that engineering features (e.g., berms and shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

5.5 Evaluation Findings

F5.1 The SAR sufficiently describes shielding design features and design criteria for the SSCs important to safety.

F5.2 Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The HI-STAR 100 shielding features are designed to assist in meeting these requirements.

F5.4 The staff concludes that the design of the shielding system for the HI-STAR 100 is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the HI-STAR 100 will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.6 References

1. U.S. Code of Federal Regulations. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

2. U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Title 10, Part 20.

3. L.M. Petrie, et al. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-4, Revision 4, 1995.

4. Energy Information Administration, U.S. Department of Energy, "Spent Nuclear Fuel Discharges from U.S. Reactors 1994," February 1996.

5. TRW Environmental Safety Systems, Inc., "DOE Characteristics Data Base, User Manual for the CDB_R, " November 16, 1992.

6. J.F. Briesmeister, Ed. "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A," Los Alamos National Laboratory, LA-12625-M (1993).

7. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-3, Revision 5, 1997.

8. AEA Technology, "MCBEND - A Monte Carlo Program for General Radiation Transport Solutions, User Guide for Version 9," 1997.

6.0 CRITICALITY EVALUATION

6.1 Review Objective

The purpose of the review of Holtec's criticality evaluation of the HI-STAR 100 is to ensure that the spent fuel in the cask remains subcritical under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage.

The staff reviewed the information provided in the SAR to determine whether the HI-STAR 100 system fulfills the following acceptance criteria in NUREG-1536 (as stated):

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.

4. Criticality safety of the cask system should not rely on use of the following credits:

   a. burnup of the fuel
   b. fuel-related burnable neutron absorbers
   c. more than 75% for fixed neutron absorbers when subject to standard acceptance tests.

The staff's evaluation is summarized below.

6.2 Criticality Design Criteria and Features

Criticality safety of the HI-STAR 100 system depends on the geometry of the fuel basket and the use of permanent neutron-absorbing panels (Boral). In all MPC designs, the fuel assemblies are placed in baskets with square fuel cells and Boral panels fixed to the fuel cell walls. In the MPC-24 basket, the primary design features that ensure subcriticality are the minimum size of the flux trap (1.09 inches) and the minimum ^{10}B content of the Boral panels (0.0267 g/cm^2). In the MPC-68 and MPC-68F baskets, the primary design features that ensure subcriticality are the minimum pitch of the fuel cells (6.43 inches) and the minimum ^{10}B content of the Boral panels (0.0372 g/cm^2 in MPC-68 and 0.01 g/cm^2 in the MPC-68F).

The MPC-24, MPC-68, and MPC-68F do not rely on borated water as a means of criticality control. Therefore, the MPCs would remain subcritical when flooded with pure water. In addition, the fuel cells have semi-circular cutouts at the bottom to allow the volume inside and outside the fuel cells to flood and drain at the same rate.

The off-normal and accident condition events will not adversely affect the design features important to criticality safety. Therefore, in terms of reactivity and criticality control, the configuration of the cask after an off-normal or accident event will be identical to or bounded by the normal condition configuration.

The staff reviewed Sections 1, 2, and 6 of the SAR and verified that the design features important to criticality safety are clearly identified and adequately described. The staff verified the consistency of the information between Sections 1, 2, and 6. The staff verified that the SAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the HI-STAR 100 system design meets the "double contingency" requirements of 10 CFR 72.124(a).

6.3 Fuel Specification

The fuel assemblies that can be stored in the HI-STAR 100 system must fit into one of 36 PWR or BWR fuel assembly classes defined by the applicant. The classes of fuel assemblies that are approved for storage in the HI-STAR 100 system are listed in Tables 1.1-2 and 1.1-3 of Appendix B to the CoC. In those tables, the fuel specifications that are important to criticality safety are:

- maximum initial enrichment (PWR) or maximum planar average initial enrichment (BWR)
- number of fuel rods, including number of partial length rods (BWR)
- minimum clad outer diameter
- maximum clad inner diameter
- maximum pellet diameter
- fuel rod pitch
- maximum active fuel length
- number of guide tubes (PWR) or number of water rods (BWR)
- minimum guide tube thickness (PWR) or minimum water rod thickness (BWR)
- maximum channel thickness (BWR)

The parameters listed above represent the limiting or bounding parameters for the fuel assemblies. A fuel assembly having these actual specifications would be the most reactive assembly or bounding assembly in that class. In some assembly classes, the bounding assembly corresponds to an actual assembly design; while in most assembly classes, the bounding assembly is an artificial assembly which is more reactive than any real assembly design that falls into that class.

In terms of criticality safety, the most important fuel specification is the fuel's enrichment. The MPC-24 may contain PWR fuel assemblies with maximum initial enrichments varying from 4.0 to 4.6 wt% ^{235}U depending on the fuel assembly class. The MPC-68 may contain BWR fuel assemblies with maximum planar average initial enrichments varying from 2.7 to 4.2 wt% ^{235}U depending on the fuel assembly class. The MPC-68F may contain BWR fuel assemblies from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes with a maximum planar average initial enrichment of 2.7 wt% ^{235}U.

The maximum planar average initial enrichment is specified for the BWR fuel assemblies because these assemblies typically have fuel rods with varying enrichments. The maximum planar average enrichment is the simple average of the distributed fuel rod enrichments within a given axial plane of the assembly lattice. The applicant's criticality analyses for BWR fuel assumed that the entire fuel assembly was at the maximum planar average initial enrichment. The applicant performed calculations to show that this is more conservative than explicitly modeling the assembly's pin-by-pin enrichments. The calculations considered real assembly designs and hypothetical assembly designs. These calculations are presented in Appendix B of the SAR.

Specifications on the condition of the fuel are also included in the SAR and Appendix B to the CoC. The HI-STAR 100 system is designed to accommodate intact fuel assemblies, damaged fuel assemblies, and fuel debris as defined in the Appendix B to the CoC. The MPC-24 must contain intact PWR fuel assemblies only. The MPC-68 may contain intact BWR fuel assemblies or damaged fuel assemblies from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes. The MPC-68F may contain intact fuel assemblies, damaged fuel assemblies, or fuel debris from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes. The damaged fuel and fuel debris must be placed in DFCs which are designed to confine gross fuel particulates to a known, subcritical geometry.

The staff reviewed the fuel specifications considered in the criticality analyses and verified that they are consistent with or bound the specifications given in Sections 1, 2, and 12 of the SAR and Appendix B to the CoC. Staff verified that all fuel assembly parameters important to criticality safety have been included in Appendix B to the CoC.

The staff agrees that fuel assembly parameters listed above represent the limiting or bounding parameters for fuel assemblies in a cask that does not rely on borated water for criticality control.

The staff reviewed the applicant's calculations that compared the use of planar-averaged enrichment versus explicit pin-by-pin enrichments in BWR fuel assemblies. Based on the results of these calculations and the information in Appendix 6.B of the SAR, the staff agrees that using the maximum planar average initial enrichment in the criticality analyses of BWR fuel assemblies is appropriate.

6.4 Model Specification

6.4.1 Configuration

The applicant used three-dimensional calculational models in its criticality analyses. Sketches of the models are given in Section 6.3 of the SAR. The models are based on the engineering drawings in Section 1.5 of the SAR and take into consideration the dimensional worst-case tolerance values. As previously stated, the design-basis off-normal and accident events do not affect the design of the cask from a criticality standpoint. Therefore, the calculational models for the normal, off-normal, and accident conditions are the same.

To determine the most reactive basket dimensions, considering manufacturing tolerances, the applicant performed two-dimensional CASMO-3 and three-dimensional MCNP4a calculations. These calculations were used to determine the reactivity effect of manufacturing tolerances and the worst-case combination of basket dimensions. Based on the results of these calculations, the MPC-24 was modeled using the nominal fuel cell pitch (10.777 inches), the minimum box inner dimension (8.81 inches), the nominal box wall thickness (5/16 inch), and the maximum flux trap size (1.09 inches). The MPC-68 was modeled using the minimum fuel cell pitch (6.43 inches), the minimum box inner dimension (5.993 inches), and the nominal box wall thickness (1/4 inch).

The calculational models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit)
- 75% credit for the ^{10}B loading in the Boral panels (only 67% credit for the MPC-68F)
- absence of the Holtite neutron shield
- that the Boral panels are only as long as the fuel assembly active length, 150 inches maximum (the engineering drawings specify them to be 156 inches long)
- the Boral panels located on the periphery of the MPC-24 are only 5 inches wide (the engineering drawings specify 12 of the peripheral panels to be 6.25 inches and all other panels to be 7.5 inches wide)
- flooding of the fuel rod gap regions with pure water, even the intact fuel assemblies
- the maximum planar average enrichment of the fuel assemblies in the MPC-68F is 3.0 wt% ^{235}U (Appendix B to the CoC permit a maximum planar average enrichment of 2.7 wt% ^{235}U)

The fuel assemblies were modeled explicitly. For BWR fuel assemblies, the water channels were appropriately included in the model. The models for damaged fuel assemblies and fuel debris considered lost or missing fuel rods, collapsed fuel assemblies, and powdered fuel.

Various moderating conditions (i.e., flooding with full-density and reduced density water and partial flooding) were also considered in the calculation models. Normally, preferential or uneven flooding within the MPC is not a concern because the MPC baskets are designed to allow the volume inside and outside the fuel cells to flood and drain at the same rate. For damaged fuel in DFCs, however, uneven draining may be possible. The drainage holes on the DFCs are covered with 250 mesh debris screens. The staff has learned that the water surface tension in the screen may be capable of supporting water. Thus, the DFCs may hold water or

may not drain at the same rate as the rest of the MPC cavity. The applicant did not consider a case in which the DFCs retained water while the rest of the MPC cavity was drained. However, the staff performed independent analysis considering this scenario. In this analysis, the staff assumed that the entire internal volumes of the DFCs were filled with water while the rest of the MPC cavity was dry. This analysis resulted in a k_{eff} of approximately 0.9 for the most reactive damage fuel assembly class (the 6x6 C). In comparison, the applicant's analysis show a k_{eff} of approximately 0.8 for the 6x6 C assembly when the entire MPC cavity is fully and evenly flooded. Although there is a significant increase in k_{eff}, it still remains well below 0.95. Thus, the staff concludes that even if preferential or uneven flooding is possible with the DFCs, it does not present a criticality concern because the fuel assemblies that may be placed in the DFCs are limited to low reactivity fuel (6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A with a maximum planar average initial enrichment of 2.7 wt% ^{235}U).

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Sections 1 and 2, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances were incorporated into the calculational models.

For its confirmatory analyses, the staff independently modeled the cask using the engineering drawings and bills of material presented in Section 1.5 of the SAR. Specifically, the staff used Drawing Nos. 5014-1395, 5014-1396, 5014-1397, 5014-1401, 5014-1402, and 5014-1784 and Bill of Material Nos. BM-1478, BM-1479, and BM-1819. The staff's fuel assembly models were based on the fuel assembly parameters given in Section 6 of the SAR and in Appendix B to the CoC. The staff found its models of the cask and contents to be compatible with those of the applicant.

6.4.2 Material Properties

The composition and densities of the materials considered in the calculational models are provided in Table 6.3.4 of the SAR.

One of the most important materials in the HI-STAR 100 is the Boral neutron absorber. In Section 1.2.1.3.1 of the SAR, the applicant provided a detailed description of the characteristics, historical applications, service experience, and manufacturing quality assurance of the Boral material. The minimum required ^{10}B content is verified through the acceptance testing program described in Section 9.1.5.3. As previously stated, only 75% credit is taken for the ^{10}B content in the Boral panels.

The continued efficacy of the Boral, over a 20-year storage period, is assured by the design of the HI-STAR 100 system. The applicant demonstrated that the neutron flux from the irradiated fuel results in a negligible depletion of the ^{10}B content in the Boral. In addition, a structural analysis was performed, and presented in Appendix 3.M.1, to demonstrate that the Boral panel will remain in place during accident conditions.

The staff reviewed the composition and number densities presented in Table 6.3.4 of the SAR and found them to be reasonable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Based on the information provided on the Boral material, the staff agrees that the continued efficacy of the Boral poison can be assured by the design of the HI-STAR 100 system, and a surveillance or monitoring program is not necessary. An exemption to 10 CFR 72.124(b) was granted by the NRC to Holtec International in a letter dated December 1, 1998.

The staff reviewed the neutron absorber acceptance test described in Section 9.1.5.3 of the SAR. The staff's acceptance of the neutron absorber test described in this section is based, in part, on the fact that the criticality analyses assumed only 75% of the minimum required ^{10}B content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

6.5 Criticality Analysis

6.5.1 Computer Programs

The applicant's principal criticality analysis code was 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 applicant also performed independent verification of its MCNP4a calculations using the KENO-Va code in the SCALE 4.3 system. The KENO-Va calculations used the 238-group cross-section library.

CASMO-3, a two-dimensional transport theory code, was used to assess the incremental reactivity effects of manufacturing tolerances. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances. Based on the results of the CASMO-3 calculations, the worst-case combination of manufacturing tolerances was determined and incorporated into the three-dimensional MCNP4a and KENO-Va models.

The staff agrees that the codes and cross-section sets used are appropriate for this particular 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.

6.5.2 Multiplication Factor

Results of the applicant's criticality analyses show that the k_{eff} in the HI-STAR 100 system will remain below 0.95 for all fuel loadings. The results of the applicant's MCNP4a criticality calculations for the bounding assemblies are given in Tables 6.1.1, 6.1.2, and 6.1.3 of the SAR. The maximum k_{eff} calculated for each MPC design are summarized in the table below. These results have been adjusted to include all biases and uncertainties at a 95% confidence level.

MPC Design	Most Reactive Fuel Assembly Class	Maximum k_{eff}
MPC-24	15x15 F	0.9478
MPC-68	10x10 A	0.9457
MPC-68F	6x6 C	0.8021

The staff reviewed the applicant's calculated k_{eff} values and agrees that they have been appropriately adjusted to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent criticality calculations for the most reactive fuel assembly class for each fuel assembly array size. The results of the staff's confirmatory calculations were in close agreement with the applicant's results for the corresponding fuel assembly class.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the HI-STAR 100 system will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.5.3 Benchmark Comparisons

The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of variables in the HI-STAR 100 design. The three most important parameters are the fuel enrichment, the ^{10}B loading of the neutron absorbers, and the fuel cell spacing (MPC-68) or flux trap size (MPC-24). Parameters such as reflector material and spacing, fuel pellet diameter and fuel rod pitch, soluble boron concentration, and MOX fuel, have a smaller effect but were also considered in selecting the critical experiments.

Results of the benchmark calculations show that there are no trends in the bias. The benchmark analysis yielded the following calculational biases: 0.0021 ± 0.0006 for MCNP4a, and 0.0036 ± 0.0009 for KENO-Va. These biases were determined by truncating to 1.000 any calculated k_{eff} that exceed unity. The uncertainty associated with each bias has been multiplied by the one-sided K-factor for 95% probability at the 95% confidence level (~2.05 for the number of cases analyzed).

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the calculational bias acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.6 Supplemental Information

All supportive information has been provided in the SAR, primarily in Sections 1, 2, and 6.

6.7 Evaluation Findings

Based on the information provided in the SAR and the staff's own confirmatory analyses, the staff concludes that the HI-STAR 100 system meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

F6.1 SSCs important to criticality safety are described in sufficient detail in Sections 1, 2, and 6 of the SAR to enable an evaluation of their effectiveness.

F6.2 The HI-STAR 100 system is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the 20-year storage period. In addition, there is no credible way to lose the fixed neutron poisons; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for 20 years with an adequate margin of safety.

F6.5 The staff concludes that the criticality design features for the HI-STAR 100 system are in compliance with 10 CFR Part 72, as exempted for 10 CFR 72.124(b), and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the HI-STAR 100 system will allow safe storage of spent fuel. This finding considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CONFINEMENT EVALUATION

The confinement features and capabilities review ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

7.1 Confinement Design Characteristics

The applicant has clearly identified the confinement boundary. The confinement boundary includes the MPC shell, the bottom baseplate, the MPC lid (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code, Section III, Subsection NB to the maximum extent practicable. Exceptions to the ASME Code are listed in the SAR Table 2.2.15. The MPC lid (with the vent and drain port cover plates welded to the lid) and closure ring are welded to the upper part of the MPC shell at the loading site. This provides redundant sealing of the confinement boundary. The welds forming the confinement boundary are described in detail in Subsection 7.1.3 of the SAR. The redundant closures of the MPC satisfy the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The applicant provided procedures for drying and evacuating the cask interior during loading operations. The staff reviewed these procedures and finds that this design, if fabricated properly according to the SAR, will maintain the confinement boundary. Maintaining a stable pressure of 3 torr for greater than 30 minutes with the vacuum pump disconnected, assures that an acceptably low amount of water and, thus, potentially oxidizing material remain in the MPC.

The applicant's testing is performed to an "as tested to leakage rate" of 5×10^{-6} std cm^3/s helium and confirms that the amount of helium lost from the MPC over the approved period due to the hypothetical accident conditions leakage rate is limited to less than 2.5% of the backfilled amount. This ensures an adequate amount of helium remains in the MPC to maintain an inert atmosphere and to support the heat transfer analysis over the lifetime of the cask.

For normal conditions of storage, the MPC uses multiple confinement barriers provided by the fuel cladding and the MPC enclosure vessel to assure that there is no release of radioactive material to the environment. The MPC is backfilled with an inert gas (helium) to protect against degradation of the cladding. Section 3 of the SER shows that all confinement boundary components are maintained within their code-allowable stress limits during normal storage conditions. Section 4 of the SER shows that the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal conditions of storage. Weld examinations, including multiple surface and volumetric examinations, hydrostatic testing, and leakage rate testing on the MPC lid weld; multiple surface examinations; and leakage rate testing of the vent and drain port cover plate welds assure the integrity of the MPC closure. Holtec described the MPC inspection and test acceptance criteria in SAR Table 9.1.1. MPC closure weld examination and acceptance criteria are included in the technical specifications. The all-welded construction of the MPC with redundant closure provided by the fully welded MPC closure ring, and extensive inspection and testing, ensures that no release of radioactive material for normal storage and transfer conditions will occur.

7.2 Confinement Monitoring Capability

For cask systems using canisters with seal weld closures, continuous monitoring of the weld closures is not necessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. However, there will be other forms of continuous monitoring of the cask such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. These monitoring activities include periodic surveillance, inspection, and survey requirements of the casks, as well as the preexisting radiological and environmental monitoring programs applicable to the entire licensee site.

7.3 Nuclides with Potential for Release

The quantity of radioactive nuclides postulated to be released to the environment and the applicable bounding calculational method have been assessed as discussed in NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Ship Various Contents," and ANSI N14.5-1997, "Leakage Tests on Packages for Shipment." The applicant used all isotopes which were greater than or equal to 0.1% of the total activity inventory based on the bounding shielding fuel source terms. This activity cutoff point is acceptable because the analysis captures over 97% of the dose at the site boundary. The release fractions used for the calculation are from NUREG/CR-6487. Design-basis leakage rates are specified in SAR Table 7.1.1. Further details on leakage testing are found in SER Sections 8 and 9. The analysis for normal, off-normal, and hypothetical accident conditions are based on a confinement boundary measured helium leakage rate of less than or equal to 7.5×10^{-6} std cm^3/s measured after fuel loading and welding, which includes the design-basis leakage rate and an adjustment for the sensitivity of the leakage testing device.

7.4 Confinement Analysis

Since the confinement boundary is welded and the temperature and pressure of the MPC are within the design-basis limits, no discernable leakage is credible. However, to demonstrate that the HI-STAR-100 meets the requirements in 10 CFR 72.104(a), the applicant performed an analysis described above with 1% rod failures and an exposure time of 1 year. The staff confirmed this analysis and agrees with the assumptions and calculational methods used. The applicant did not perform a specific analysis for off-normal conditions. However, based on the analysis for normal conditions, an analysis for off-normal conditions, assuming 10% fuel rod failure in one cask, and an exposure time of 1 year, would also meet the requirements of 10 CFR 72.104(a). For normal conditions of storage, the calculations result in a bounding whole body dose for the MPC-68 of < 0.1 mrem and a thyroid dose of $<3 \times 10^{-3}$ mrem.

For hypothetical accident conditions, the staff independently confirmed the applicant's maximum whole body dose to a person located at the site boundary at 100 meters for 30 days of 54 mrem and the maximum thyroid dose of $< 2 \times 10^{-2}$ mrem. The staff agrees with the use of 30 days exposure time which is consistent with what is used for accident analysis in 10 CFR Part 100. The applicant did not include a Kr-85 dose calculation to the skin. However, the staff did perform such an analysis and the results show that the package meets the requirements of 10 CFR 72.106(b) with a bounding accident dose of $< 6 \times 10^{-2}$ mrem to the skin.

7.5 Supportive Information

Supportive information or documentation includes justification of assumptions and analytical procedures, computer spreadsheet inputs, drawings of the MPC confinement boundary, and applicable pages from referenced documents.

7.6 Evaluation Findings

F7.1 Section 7 of the SAR describes confinement SSCs important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the HI-STAR 100 adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the staff's relevant temperature considerations.

F7.3 The design of the HI-STAR 100 provides redundant sealing of the confinement system closure joints using dual welds on the MPC lid and the MPC closure ring.

F7.4 The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.

F7.5 The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases is added to the direct dose to show that the HI-STAR 100 satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The cask confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.7 The staff concludes that the design of the confinement system of the HI-STAR 100 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STAR 100 will allow safe storage of spent fuel. This finding considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis, the staff's confirmatory analysis, and accepted engineering practices.

8.0 OPERATING PROCEDURES

The review of the operating procedures is to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for three key operations.

8.1 Cask Loading

Detailed loading procedures must be developed by each cask user.

The cask loading procedures described in the SAR include appropriate key prerequisite, preparation, and receipt inspection provisions to be accomplished before cask loading. These include verification of lift yoke load test certifications, visual inspection of key components, and reference to the site's heavy load handling procedures. The procedures also verify that tests, inspections and verifications, and cleaning procedures required in preparation for cask loading are specified. The procedure descriptions include actions necessary to ensure that appropriate levels of fluid (e.g., demineralized water in the annulus region between the MPC and the HI-STAR 100 overpack and the spent fuel pool water within the MPC) are properly controlled. The procedures describe the activities sequentially in the anticipated order of performance.

8.1.1 Fuel Specifications

The procedures described in the SAR provide for fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading have been pre-selected for loading into the MPC. After pre-selected fuel assemblies have been loaded in the MPC, a confirmatory post-loading visual verification is performed. Exact fuel specifications for fuel that is permitted to be loaded into an MPC is specifically designated in Appendix B to the CoC Tables 2.1-1 through 2.1-5. Detailed site-specific procedures are necessary to ensure all fuel loaded in the cask meets the fuel specifications as delineated in the CoC and the TS. These procedures are subject to evaluation on a site-specific basis through the inspection process rather than during the licensing review.

8.1.2 ALARA

The HI-STAR 100 generic cask loading procedures incorporate general as low as reasonably achievable (ALARA) principles and practices. ALARA practices include periodic monitoring of dose rates, the use of annulus sealing equipment to reduce occupational exposure and limit MPC contamination, the use of temporary shielding, and the use of special tools to reduce occupational exposure. ALARA principles include warnings and notes that precede steps and identify potential radiological hazards. The procedures incorporate TS 2.2.1 and 2.2.2, which specify limits for surface dose rates and radionuclide contamination, respectively. Each cask user will need to develop detailed loading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.1.3 Draining and Drying

SAR Section 8 contains operating procedures for use in draining and drying the MPC and MPC overpack. These procedures clearly describe the procedures for removing water vapor and oxidizing material to an acceptable level.

After refilling the MPC with demineralized or spent fuel pool water, helium or nitrogen at a pressure less than 20 psig is used to drain as much water as practicable from the cask. The volume of water removed is recorded for later use. The vacuum drying system (VDS) is used with a stepped approach to reduce the potential for blockage of the evacuation system as a result of icing during evacuation. After the VDS reduces the MPC pressure to less than 3 torr, the pump is isolated, and a 30-minute holding period begins. In accordance with the TS, the MPC remains at ≤ 3 torr for ≥ 30 minutes. This dryness verification test is included as TS 2.1.1.

The 3 torr value described above is consistent with methodology in NUREG-1536, which references PNL-6365.[1]. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 are expected if the vacuum drying is performed as described in the SAR. This will serve to reduce the amount of oxidants to below the levels where significant cladding degradation is expected.

The MPC is backfilled with helium on top of the spent fuel pool water for applicable leak testing and then filled with water for the hydrostatic test. After the hydrostatic test, nitrogen or helium is used to force the spent fuel pool water from the cask, effectively removing contaminants. The cask is then re-evacuated using the VDS, tested to ensure ability to maintain a vacuum as described above, and backfilled with helium before final closure. A suitable inert cover gas (≥99.995% pure helium) is specified to minimize this source of contaminants in accordance with the recommendations of PNL-6365. The HI-STAR 100 operating procedures provide for repetition of the evacuation and repressurization cycles using the VDS if the vacuum is lost during the vacuum drying process, as could occur during loading. Similar draining, vacuum drying, and helium backfilling procedures are outlined in Section 8 for the bolted closure of the overpack helium retention boundary.

8.1.4 Welding and Sealing

The HI-STAR 100 SAR, Section 8, describes the use of the Automated Welding System Robot for the MPC lid closure weld. Remote welding helps ensure the dose to welders will be ALARA. Prior to welding, approximately 120 gallons of water are removed from the MPC to keep moisture away from the weld region. In addition, a vacuum pump is connected to keep moist air from condensing on the MPC lid weld area. Section 8 also describes the nondestructive examination (NDE) to be done on closure welds, including visual examination (VT), root and final pass dye penetrant examination (PT), ultrasonic examination (UT) (or optional multi-layer PT examination), leak testing, and hydrostatic tests. The applicant states all NDE will be performed in accordance with the applicable sections of the ASME Code, Sections III and V, with the NDE requirements, applicable code, and acceptance criteria described in detail in SAR Table 9.1.3 and the TS. The applicant states that leak testing will be performed with a mass spectrometer leak detector (MSLD) in accordance with the manufacturer's instructions and ANSI N14.5[2]. The applicant states allowable leakage is 5.0×10^{-6} std cm^3/s helium, in accordance with SAR Section 9 and the TS. The SAR also includes acceptable provisions for correction of weld defects (repair in accordance with the site's approved weld repair procedure, additional NDE, and any additional drying and purging that may be necessary). Similar welding procedures are described for the MPC closure ring and vent and drain port cover plates. The staff recognizes that the MPC lid-to-shell weld may be performed manually. The staff concludes these procedures provide for acceptable welding and NDE of the closure welds.

SAR Section 8 also describes steps for proper sealing of the overpack bolted joint, including helium backfill, leak testing (with acceptance standards per the TS), and necessary bolt torque. Section 8 includes acceptable provisions for placing and tightening any closure bolts are consistent with information presented in SAR Sections 2, 3, and 9, which address applicable design criteria, structural evaluation, and the acceptance tests and maintenance program, respectively. The staff concludes these procedures provide for acceptable sealing of the overpack.

8.2 Cask Handling and Storage Operations

All accident events applicable to the transfer of the cask to the storage location are bounded by the design events described in Sections 2 and 11 of the SAR. All conditions for lifting and handling methods are bounded by the evaluations in Sections 3 and 4 of the SAR. There are TS associated with cask transfer operations, such as restricting lift heights and environmental conditions.

Inspection, surveillance, and maintenance requirements that are applicable during ISFSI storage are discussed. Surveillance and monitoring requirements are included in Section 12 of the SAR. The TSs which comprise Section 12 of the SAR have been appended to the CoC. Maintenance requirements are discussed in Section 9 of the SAR. The staff determined that these discussions were acceptable.

Occupational and public exposure estimates are evaluated in Section 10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.3 Cask Unloading

Detailed unloading procedures must be developed by each cask user.

The HI-STAR 100 cask unloading procedures describe the general actions necessary to prepare the MPC for unloading in a reactor spent fuel pool, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-STAR 100 overpack and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations.

8.3.1 Cooling Venting & Reflooding

The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding. The cool-down process utilizes helium to gradually reduce the cladding temperature to less than 200° F such that the MPC may be flooded with water without thermally shocking the fuel assembles or over-pressurizing the MPC from the formation of steam.

Procedures for obtaining a gas sample are included to provide for assessment of the condition of the fuel assembly cladding. This allows for detection of potentially damaged or oxidized fuel. The procedures include ALARA caution steps to prevent the possible spread of contamination.

8.3.2 ALARA

The HI-STAR 100 generic cask unloading procedures incorporate general ALARA principles and practices. ALARA practices include periodic monitoring of dose rates, the use of annulus sealing equipment to reduce occupational exposure and limit MPC contamination, gas sampling of the MPC volume to identify potential clad damage, the use of temporary shielding, and the use of special tools to reduce occupational exposure. ALARA principles include warnings and notes that precede steps and identify potential radiological hazards. Each cask user will need to develop detailed unloading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

8.3.3 Fuel Crud

The HI-STAR 100 ALARA practices and procedures provide for the mitigation of the possibility of dispersal of fuel crud particulate material. However, experience with wet unloading of BWR fuel after transportation has involved handling significant amounts of crud. This fine crud includes ^{60}Co and ^{55}Fe and will remain suspended in water or air for extended periods. The dry cask reflood process during unloading of BWR fuel has the potential to disperse crud into the fuel transfer pool and the pool area atmosphere, thereby, creating airborne exposure and personnel contamination hazards. By contrast, no significant crud dispersal problems have been observed in handling PWR fuel because of differences in the characteristics of crud on this type of fuel. Therefore, detailed procedures incorporating provisions to mitigate the possibility of fuel crud particulate dispersal must be developed by each cask user.

8.4 Evaluation Findings

F8.1 The HI-STAR 100 can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The bolted closure plate and welded MPC of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The general operating procedures are designed to prevent contamination of the MPC and facilitate decontamination of the overpack. Routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.4 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F8.5 The general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.6 Section 10 of the SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F8.7 The staff concludes that the generic procedures and guidance for the operation of the HI-STAR 100 are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

8.5 References

1. R.W., Knoll, et al., Pacific Northwest Laboratory, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, DE88 003983, November 1987.

2. American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-1987, January 1987.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The objective of the review of the acceptance tests and maintenance program is to ensure that Holtec's SAR includes the appropriate acceptance tests and maintenance programs for the HI-STAR 100. A clear, specific listing of these commitments will help avoid ambiguities concerning design, fabrication, and operational testing requirements when the NRC staff conducts subsequent inspections.

9.1 Acceptance Tests

All materials and components will be receipt inspected for visual and dimensional acceptability, conformance to specification requirements, and material traceability. In addition, materials for the confinement boundary (MPC) and HI-STAR 100 system helium retention boundary will also be inspected per the requirements of ASME Section III, Article NB-2500.

Acceptance criteria for all NDE will be in accordance with the applicable ASME Section III Code by which the component is fabricated, except as modified by the design drawings.

9.1.1 Visual and Nondestructive Examination Inspections

During the resolution of public comments on the rulemaking to include the HI-STAR 100 cask in the list of generally licensed casks in 10 CFR 72, Subpart K, the staff determined that it would be necessary for Holtec to revise its discussion of visual and nondestructive testing in the SAR to conform with the discussion in Revision 8 of the HI-STAR 100 Transportation Safety Analysis Report. The Hi-STAR 100 was certified for the transportation of spent nuclear fuel in March of 1999. The staff has reviewed the proposed revision to the SAR and its determination in this section are based on this review. The MPC confinement boundary and HI-STAR 100 system's helium retention boundary are fabricated and inspected in accordance with ASME Code, Section III, Subsection NB. The MPC lid-to-shell and closure welds are not full penetration welds. The vent and drain port welds and closure ring welds will be examined by the PT method. ASME code exceptions are contained in Table 4-1 of the TS. However, the lid and closure rings are welded independently to provide a redundant seal. Also, the MPC lid to the MPC closure ring weld is examined by the PT method to ensure acceptable weld integrity.

Other parts of the HI-STAR 100 overpack are fabricated and inspected in accordance with ASME Code, Section III, Subsection NF. The MPC basket, basket supports, and fuel spacers are fabricated and inspected in accordance with ASME Code, Section III, Subsection NG.

Table 9.1.1 of the SAR lists the inspection and test criteria for the MPC and Table 9.1.2 of the SAR lists the inspection and test acceptance criteria for the HI-STAR Overpack.

Holtec committed to fabricate and examine cask components in accordance with the ASME B&PV Code, Section III, as summarized in Tables 9.1.1 and 9.1.2. These tables contain inspection and test acceptance criteria for the MPC and overpack, respectively. NDE

commitments, with location, examination type, applicable Code, and acceptance criteria for the HI-STAR 100 are detailed in Table 9.1.3.

The NDE of weldments is well-characterized on drawings, using standard NDE symbols and/or notations in accordance with AWS 2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination." Fabrication inspections include VT, PT, magnetic particle, UT, and radiographic examinations (RT), as applicable. Structural and confinement boundary weld examinations and acceptance criteria meet the applicable requirements of ASME Code, Section III.

The staff finds that this combination of examinations meets the staff's position that the MPC lid-to-shell weld may be inspected using either volumetric or multiple pass dye penetrant techniques subject to the following conditions:

1) Dye PT may only be used in lieu of volumetric examination and only on austenitic stainless steels. PT should be done in accordance with ASME Section V, Article 6, "Liquid Penetrant Examination."

2) If PT alone is used, at a minimum, it must include the root and final layers and sufficient intermediate layers to detect critical flaws.

3) For multiple layer PT, the maximum undetectable flaw size is less than the critical flaw size. The critical flaw size shall 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. Flaws in austentic stainless steel are not expected to exceed the thickness of one weld bead.

4) The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Code, Section III, NB-5350, for PT and NB-5332 for UT.

5) If PT alone is used, a design stress-reduction factor of 0.8 must be applied to the weld design.

6) The results of the PT, including all relevant indications, shall be made a permanent part of the licensee's records by video, photographic, or other means providing a retrievable record of weld integrity. Video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676.

9.1.1.1 MPC Lid-To-Shell Weld Inspection and Critical Flaw Size Determination

In accordance with Interim Staff Guidance (ISG) - 4, Cask Closure Weld Inspections, Holtec proposed to examine the austenitic stainless steel MPC closure welds with a multiple-layer dye PT in lieu of the ASME Code required volumetric examination. The PT will be done in accordance with ASME Section V, Article 6, Liquid Penetrant Examination, with ASME Section III, NB-5350 acceptance standards. Holtec used a design stress-reduction factor of 0.45 applied

to the weld design, exceeding the 0.8 factor required by ISG-4. Holtec calculated the critical flaw size for the MPC closure welds in Holtec Position Paper DS-213, Acceptable Flaw Size in MPC Lid-to-Shell Welds, Revision 2, dated February 23, 1999. The Holtec analysis applied a bounding stress (52.662 thousand pounds per square inch (ksi)) to a hypothetical 360° 50%-through-wall crack (0.375 inches deep for the MPC-24 and MPC-68, 0.625 inches deep for the MPC-68F). Position Paper DS-213 concludes these large hypothetical cracks would not grow under normal or accident conditions, thus preserving structural and containment integrity. This paper concluded that PT on the root, final, and every 3/8 (0.375) inches of the weld would be acceptable.

The staff performed an independent calculation to determine critical flaw size for the MPC weld using the pc-CRACK software package. The stress levels used in the calculation were based on Section 6.0 of Position Paper DS-213, which was based upon the bounding 10 CFR Part 71 top end drop described in Section 2.L.8.1.1 of the transportation SAR. Based upon a 624,000 lb-force load on the MPC lid (SAR Section 2.L.8.3), the shear stress in the MPC lid weld is 4.717 ksi (this calculation assumes a 5/8-inch weld as they neglect the root pass). Actual weld size is 0.75 inch for a stress level 3.931 ksi for the MPC-24 & MPC-68 and 1.25 inches for a stress level 2.358 ksi for the MPC-68F (the only MPC that is required for containment purposes). The staff used the 4.717 ksi stress value and factors of $\sqrt{2}$ (from ASME Section XI, IWB-3600 for emergency and faulted conditions) and 2 (to account for any uncertainties in stress calculations) to determine stress inputs for the critical flaw calculations. The staff used bounding material toughness properties for the Type 316 stainless steel submerged arc weld J-R curve taken from EPRI-TR106092. The results of the calculations are shown below:

Table 9.1
NRC Critical Flaw Size Calculation Results
Critical Flaw Depth [inches]

Weld Size [inches]	Applied Stress (ksi)		
	4.717 ksi (base)	6.671 ksi (base x $\sqrt{2}$)	9.434 ksi (base x 2)
0.75 in (MPC-24, MPC-68)	0.534 in	0.495 in	0.451 in
1.25 in (MPC-68F)*	0.870 in	0.802 in	0.727 in

* NOTE: The HI-STAR 100 MPC is the confinement boundary for storage conditions, and provides the redundant sealing required by 10 CFR 72.236(e). The MPC-68F is the only MPC which is required for secondary containment purposes during transportation in accordance with 10 CFR 71.63(b) due to the presence of failed fuel. The HI-STAR 100 overpack is the primary containment for all MPC designs during transportation.

The values above represent the depth of a 360° circumferential surface crack which is the bounding critical flaw size for this design. The results of the staff's analysis show that the critical flaw size in the MPC design is greater than the depth interval for PT (0.375 inches), thus ensuring that any flaw in the MPC closure weld would be detected before it became large enough to affect the structural design of the MPC. It should be noted that only the MPC-68F is relied upon for containment during transportation. The PT will include the root and final layers

and sufficient intermediate layers (every 3/8 inch) to detect critical flaws in accordance with ISG-4.

9.1.1.2 Conclusion

Welds which do not meet the acceptance criteria will be repaired and reexamined in accordance with the original examination method and associated acceptance criteria. The applicant committed to ASME Code sections for material procurement, design, fabrication, and inspection for the MPC confinement boundary, fuel baskets, fuel basket supports, damaged fuel container, overpack helium retention boundary, overpack components, and trunnions in SAR Table 2.2.7. The staff concludes the applicant's choice of ASME Section III, Subsections NB, NF, and NG as described in Table 2.2.7 is appropriate.

9.1.2 Structural/Pressure Tests

Structural and pressure tests are subdivided into four areas: lifting trunnions, hydrostatic testing, fracture toughness, and neutron shield enclosure vessel. All testing must be performed in accordance with written and approved procedures. The test results must be documented and the documentation will become part of the final quality documentation package.

9.1.2.1 Lifting Trunnions

To ensure that the lifting trunnions do not have any hidden material flaws and that the trunnions are properly installed, the trunnions are tested at 300% of the maximum design lifting load (i.e., 3x250,000 lbs = 750,000 lbs) in accordance with ANSI N14.6. Any evidence of deformation, distortion or cracking of the trunnion or the adjacent HI-STAR 100 cask areas will require replacement of the trunnion and/or repair of the HI-STAR cask. Following any repairs and/or replacements, the load testing will be repeated and components reexamined.

9.1.2.2 Hydrostatic Testing

a. HI-STAR 100 Helium Retention Boundary

The helium retention boundary of the HI-STAR overpack (e.g., the containment boundary during transportation) will be hydrostatically tested to 150 psig +10,-0 psig, in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000. The test pressure of 150 psig is 150% of the design pressure (100 psig). This bounds the ASME Code, Subarticle NB-6221, requirement for hydrostatic testing to 125% of the design pressure.

After completion of the hydrostatic testing, the closure plate will be removed and the cask internal surfaces will be visually examined for cracking or deformation. Any evidence of cracking or deformation will be cause for rejection or repair and retest, as applicable. Liquid PT of all accessible welds will be performed in accordance with ASME Section V, Article 6, with acceptance criteria per ASME Section III, Subsection NB, Article NB-5350. Any unacceptable welds will require repair in accordance with ASME Code, Subarticle NB-4450.

b. MPC Confinement Boundary

Hydrostatic testing of the MPC confinement boundary will be performed in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000, when field welding of the MPC lid-to-shell is completed. The hydrostatic pressure for the test is 125 psig, +5,-0 psig, which is 125% of the design pressure of 100 psig. Following completion of the 10-minute hold period at the hydrostatic test pressure, and while maintaining a minimum test pressure of 125 psig, the surface of the MPC lid-to-shell weld will be visually examined for leakage and then reexamined by PT. If a leak is discovered, the test pressure will be reduced, the MPC cavity water level lowered, the MPC cavity vented (either to the fuel pool or the site licensee's off-gas system), and the weld will be examined to determine the cause of the leakage and/or cracking. Repairs to the lid-to-shell weld will be performed in accordance with an approved written procedure prepared in accordance with the ASME Section III, Subsection NB, NB-4450.

9.1.2.3 Fracture Toughness Testing

Ferritic steels used in the HI-STAR overpack are tested to assure that these materials are not subjected to brittle fracture failures.

Each plate or forging for the helium retention boundary, which includes overpack inner shell, bottom plate, top flange, and the closure plate, will be required to be drop weight tested in accordance with the requirements of Regulatory Guides 7.11 and 7.12. Additionally, per ASME Code Section III, Subsection NB, Article NB-2300, Charpy V-notch testing is performed on these materials as well. Weld material used in welding the helium retention boundary is charpy V-notch tested in accordance with the ASME Section III Code, Subsection NB, Articles NB-2300 and NB-2430. Table 3.1.18 in Section 3.1 of the SAR provided the fracture toughness test criteria for Charpy V-Notch and drop weight testing.

9.1.2.4 Pneumatic Testing

A pneumatic bubble test of the neutron shield enclosure vessel is performed in accordance with Section V, Article 10, Appendix I, of the ASME Code. The test should be performed following the final closure welding of the enclosure vessel. The pneumatic test pressure is 37.5 psig, +2.5, -0 psig, which is 125% of the rupture disk relief set pressure. The acceptance criteria for the bubble test will be no air leakage from any tested weld, as indicated by continuous bubbling of the solution. If air leakage is indicated, the weld will be repaired based on procedures in accordance with the ASME Code Section III, Subsection NF, Subarticle NF-4450.

A leakage testing with a helium mass spectrometer leak detector for the HI-STAR overpack and MPC confinement boundary is performed in accordance with ANSI N14.5. Design leakage rates are specified in SAR Table 7.1.1, which specifies a design helium leakage rate of 5×10^{-6} cm^3 per second helium for the MPC confinement boundary. Additionally, the applicant committed to using an MSLD with a sensitivity of 2.5×10^{-6} standard cm^3 per second helium. The staff finds this practice acceptable. The staff concludes the leakage criteria meets or exceeds those specified in the principal design criteria in SAR Section 2.

9.1.3 Shielding Tests

Each lot of neutron shield material will be tested to verify that the material composition, boron concentration, and neutron shield density meet the requirements in the Bill of Materials and requirements in the CoC. The installation of the neutron shielding material will be performed according to written and qualified procedures. The procedures will ensure that mix ratios and mixing-methods are controlled to achieve proper material composition, boron concentration and distribution, and that pours are controlled to prevent gaps from occurring in the material.

Neutron and gamma effectiveness tests will be performed using approved written procedures. Calibrated neutron and gamma dose meters will be used to measure the actual neutron and gamma dose rates at the surface of the HI-STAR overpack to assure that there are no unacceptable voids in the shielding.

The staff reviewed the shielding fabrication testing and controls and effectiveness tests. Because the poured neutron shield material is in a fluid state before solidification, the staff accessed the potential for voids and neutron streaming, and the potential impact on 10 CFR Parts 20 and 72 radiological requirements. The staff found the fabrication testing and controls to be acceptable and has reasonable assurance that each general licensee's radiological protection and ALARA program will detect and mitigate exposures from any significant or unexpected radiation fields. In addition, Condition Three of the CoC requires fabrication and testing of components important to safety, such as the neutron shield, to be conducted in accordance with a Commission-approved QA program that satisfies the requirements of 10 CFR Part 72, Subpart G.

9.1.4 Neutron Absorber Tests

After manufacturing, a statistical sample of each lot of Boral is tested using wet chemistry and/or neutron attenuation techniques to verify the minimum ^{10}B content at the ends of the Boral panel. The minimum allowable ^{10}B content is 0.0267 g/cm^2 for the MPC-24 Boral panels, 0.0372 g/cm^2 for the MPC-68 Boral panels, and 0.01 g/cm^2 for the MPC-68F Boral panels. Any panel with a ^{10}B loading less than the minimum allowed will be rejected. Tests will be performed using written and approved procedures. Results will be documented and become part of the HI-STAR 100 system quality records documentation package.

The staff's acceptance of the neutron absorber test described above is based, in part, on the fact that the criticality analyses assumed only 75% of the minimum required ^{10}B content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

Installation of the Boral panels into the fuel basket shall be performed in accordance with written and approved procedures. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a Boral panel in accordance with Holtec International Drawing Nos. 1395 and 1401.

9.1.5 Thermal Tests

The applicant will perform a thermal acceptance test of the first fabricated HI-STAR overpack to confirm the analytic assumptions used to calculate the heat transfer capabilities of the inner shell, intermediate shells, and radial channels. The test will be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed, and all inside and outside surfaces are painted with a paint having an emissivity of no less than 0.85. The overpack will be sealed with a test cover plate made of carbon steel and the testing will be performed in accordance with written and approved procedures.

The cask will be injected with steam to produce a uniform temperature on the internal surface. Calibrated thermocouples will be installed on the internal cavity and external walls of the overpack circumferentially at fixed elevations. Temperatures of all the thermocouples will be recorded hourly until thermal equilibrium is reached.

9.1.6 Cask Identification

The cask shall be marked with a model number, unique identification number, and empty weight. This information will appear on a data plate, which is detailed in drawings in SAR Section 1. In addition, the exterior of shielding casks or other structures that may hold the confinement cask while it is in storage shall be marked. This marking provides a unique, permanent, and visible number to permit identification of the cask stored therein.

9.2 Maintenance Program

9.2.1 Inspection

The HI-STAR 100 system maintenance program schedule is described in SAR Table 9.2.1. This table includes a schedule for initial and periodic visual inspections, shielding effectiveness tests, and seal and rupture disk replacements. The staff concludes these inspections are acceptable for initial and continued operation of the HI-STAR 100 system. As discussed in Section 6 of the SER, the SAR does not include procedures for periodic testing of neutron poison (Boral) effectiveness. However, the material has a proven history of nuclear service in various reactors and in other transportation and storage casks. The staff concludes inspections and tests are acceptable.

9.3 Evaluation Findings

F9.1. Section 9.1 of the SAR describes the applicant's proposed program for preoperational testing and initial operations of the HI-STAR 100. Section 9.2 discusses the proposed maintenance program.

F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.2.6 and 8.1.4 of the SAR identify the safety importance of

SSCs. Tables 2.2.6 and 2.2.7 present the applicable standards for their design, fabrication, and testing.

F9.3 The applicant/certificate holder/licensee will examine and/or test the HI-STAR 100 to ensure that it does not exhibit any defects that could significantly reduce its confinement and shielding effectiveness. Section 9.1 of the SAR describes this inspection and testing.

F9.4 Holtec will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. Drawing 1397, Sheet 4 of 7, in SAR Section 1.5 illustrates and describes this data plate.

F9.5 The staff concludes that the acceptance tests and maintenance program for the HI-STAR 100 are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.4 References

1. Structural Integrity Associates, Version 2.0.

10.0 RADIATION PROTECTION EVALUATION

This section evaluates the capability of the radiation protection design-features, design-criteria, and the operating procedures of the HI-STAR 100 to meet regulatory dose requirements. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d) [1,2].

Occupational exposures from the HI-STAR 100 are based on the direct radiation dose rates calculated in Section 5 of the SAR and the operating procedures discussed in Section 8 of the SAR. Doses to individuals beyond the controlled area boundary (members of the public) are determined from the direct radiation (including skyshine) dose rates calculated in Section 5 of the SAR and the dose rates from non-mechanistic, atmospheric releases calculated in Section 7 of the SAR.

10.1 Radiation Protection Design Criteria and Design Features

10.1.1 Design Criteria

Section 10.1.2 of the SAR defines the radiological protection design criteria as the limits and requirements in 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106, and the guidance in Regulatory Guide 8.8 [3]. As required by 10 CFR 72.212, the general licensee is responsible for demonstrating site-specific compliance with these requirements.

10.1.2 Design Features

Section 10.1.2 of the SAR presents radiological protection design features which provide radiation protection to personnel and individuals beyond the controlled area (members of the public). Design features discussed include the following:

- The thick-walled overpack provides shielding of gamma and neutron radiation.
- The confinement system consists of multiple welded barriers to prevent atmospheric release of radionuclides.
- The implementation of ALARA principles in the operating procedures reduces occupational exposures.
- The cask body consists of smooth surfaces to minimize decontamination time and the overpack annulus is filled with demineralized water during loading to prevent contamination of the MPC exterior.
- The low-maintenance design reduces occupational doses during storage.
- The confinement system is designed to maintain confinement of fuel during accident conditions.

Section 10.1.2 of the SAR also discusses design features that address process instrumentation and controls, control of airborne contaminants, decontamination, radiation monitoring, and other ALARA considerations. In addition, Sections 10.1.3 and 10.1.4 of the SAR address operational

considerations and describe optional auxiliary shielding devices to minimize occupational and public doses.

The staff evaluated the radiation protection design features and design criteria for the HI-STAR 100 and found them acceptable. The SAR analysis provides reasonable assurance that use of the HI-STAR 100 storage cask can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Sections 5, 7, and 8 of the SER discuss staff evaluations of the shielding features, confinement features, and operating procedures, respectively. Section 11 of the SER discusses staff evaluations of the capability of shielding and confinement features during off-normal and accident conditions.

10.2 ALARA

Section 10.1 of the SAR presents evidence that the HI-STAR 100 radiation protection design features and design criteria address ALARA requirements, consistent with 10 CFR Part 20 and guidance provided in Regulatory Guides 8.8 and 8.10[4]. The HI-STAR 100 design features are designed to maintain radiation exposures ALARA and Appendix 12A of the SAR proposes surface dose rates and surface contamination limits for the HI-STAR 100. The SAR states each site licensee will apply its existing site-specific ALARA policies, procedures, and practices for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met.

The staff evaluated the ALARA assessment of the HI-STAR and found it acceptable. Section 8 of the SER discusses the staff evaluation of the operating procedures with respect to ALARA principles and practices. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, TS 2.2.1 and 2.2.2 establish limits for the surface dose rates and surface contamination on the HI-STAR 100.

10.3 Occupational Exposures

Section 8 of the SAR discusses general operating procedures that all licensees will use for fuel loading, cask operation, and fuel unloading. Section 10.3 of the SAR discusses the estimated number of personnel, the estimated dose rates, and the estimated time for each task. The estimated doses received by personnel, without credit for temporary shielding, are presented in Tables 10.3.1, 10.3.2, and 10.3.3 of the SAR. The dose estimates indicate the total occupational dose in loading a single cask with design-basis fuel (shielding) in the MPC-24 is approximately 1.4 person-rem. The estimated occupational dose for unloading the cask is≤ 0.934 person-rem. The yearly estimated dose for surveillance and cask maintenance is approximately 0.120 person-rem and 1.5 person-rem, respectively.

The staff reviewed the occupational dose estimates and found them acceptable. Evaluation of the operating procedures is presented in Section 8 of the SER. The occupational exposure dose estimates provide reasonable assurance that occupational limits in 10 CFR Part 20 Subpart C can be achieved. Actual occupational doses will depend on site-specific parameters, including special measures taken to maintain exposures ALARA. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20 Subpart B. In addition,

each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20 Subpart C and other site-specific 10 CFR Part 50 license requirements with evaluations and measurements.

10.4 Public Exposures From Normal and Off-Normal Conditions

Section 10.4.1 of the SAR summarizes the calculated dose rates to individuals beyond the controlled area (members of the public), as presented in Sections 5 and 7 of the SAR. The SAR evaluates and concludes that the confinement functions of the MPC are not affected by normal and off-normal conditions. While no effluents are expected, Section 7.2 of the SAR presents dose rate calculations for a continuous, non-mechanistic atmospheric release of radionuclides during normal conditions. Section 5.4.3 of the SAR presents calculated direct radiation dose rates at distances beyond 100 meters from sample cask-array configurations. Figure 5.1.2 depicts estimated dose rate versus distance curves for various array configurations. Table 10.4.1 specifies distances at which the regulatory design limit of 25 mrem/yr can be achieved for various array configurations. As shown in Table 10.4.1, a single cask loaded with design-basis fuel is below regulatory limits at approximately 250 meters. A two-by-five cask array is below regulatory limits at approximately 400 meters.

The staff evaluated the public dose estimates from direct radiation and non-mechanistic atmospheric release from normal and off-normal (anticipated occurrences) conditions and found them acceptable. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER. The staff concludes that the calculated dose rates from non-mechanistic releases are insignificant compared to the dose rates from direct radiation. Therefore, direct radiation (including skyshine) is the primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each site licensee. The general license holder must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask array configuration, topography, demographics, and use of engineered features (e.g., berms). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each site licensee.

The licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

A TS has been included regarding engineered features used for radiological protection. The TS states that engineering features (e.g., berms, shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered important to safety and must be evaluated to determine the applicable QA Category.

10.5 Public Exposures From Design-Basis Accidents and Natural Phenomena Events

10.5.1 Design-Basis Public Exposures

Section 10.4.2 of the SAR summarizes the calculated dose rates (from Sections 5 and 7) for accident conditions and natural phenomena events to individuals beyond the controlled area (members of the public). The SAR evaluates and concludes that the confinement function of the MPC is not affected by design-basis accidents or natural phenomena events. Section 7.3 of the SAR presents dose rate calculations for a continuous, non-mechanistic atmospheric release of radionuclides during accident conditions. The estimated dose at 100 meters is 54.4 mrem for a 30-day exposure to the non-mechanistic atmospheric release. Section 5.1.2 of the SAR presents the direct radiation dose rate at 100 meters from a single cask, assuming complete loss of the neutron shield. The estimated dose rate at 100 meters is approximately 5 mrem/hr and the SAR states it would take more than 41 days for the dose at 100 meters to reach the 5 rem regulatory limit specified in 10 CFR 72.106(b).

The staff evaluated the public dose estimates from direct radiation and non-mechanistic atmospheric release from accident conditions and natural phenomena events and found them acceptable. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER, respectively. The staff has reasonable assurance that the combined effects of direct radiation and non-mechanistic atmospheric releases from bounding design-basis accidents and natural phenomena will be below the regulatory limit of 5 rem specified in 10 CFR 72.106(b).

10.6 Evaluation Findings

F10.1 The SAR sufficiently describes radiation protection design bases and design criteria for the SSCs important to safety for the HI-STAR 100.

F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F10.3 The HI-STAR 100 is designed to provide redundant sealing of confinement systems.

F10.4 The HI-STAR 100 is designed to facilitate decontamination to the extent practicable.

F10.5 The SAR adequately evaluates the HI-STAR 100 and its systems important to safety, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.6 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.7 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.06 are the responsibility of the site licensee. The HI-STAR 100 is designed to assist in meeting these requirements.

F10.8 The staff concludes that the design of the radiation protection system for the HI-STAR 100 is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STAR 100 will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

10.7 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear and High-level Radioactive Waste," Title 10, Part 72.

2. U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Title 10, Part 20.

3. U.S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.

4. U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1997.

11.0 ACCIDENT ANALYSES

The purpose of the review of the accident analyses is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design-basis events. This ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

1. identified all credible accidents
2. provided complete information in the SAR
3. analyzed the safety performance of the cask system in each review area
4. fulfilled all applicable regulatory requirements

This review was performed using the acceptance criteria in Section 11 of NUREG-1536.

The applicant has stated that there are no instruments and control systems that must remain operational under accident conditions.

11.1 Dose Limits for Off-Normal Events

Section 11.1 of the SAR examines the dose consequences for the identified off-normal events. The SAR determined that the confinement function of the MPC is not affected by off-normal conditions. However, Section 7 of the SAR analyzes, a non-mechanistic continuous atmospheric release during normal conditions. The SAR states that off-normal events do not result in dose consequences and that radiological conditions are the same as normal conditions analyzed in Section 5 of the SAR. A summary of the estimated occupational and public doses from off-normal events is presented in Section 10 of the SAR.

The staff reviewed the off-normal event analyses with respect to 10 CFR 72.104(a) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond the controlled area will not exceed the limits in 10 CFR 72.104(a) during off-normal conditions (anticipated occurrences). Sections 5, 7, and 10 of the SER further evaluate the radiological doses during off-normal events.

11.2 Dose Limits for Design-Basis Accidents and Natural Phenomena Events

Section 11.2 of the SAR examines the dose consequences for the identified design-basis accidents and natural phenomena events. The SAR determined that the confinement function of the MPC is not affected during design-basis accident conditions. However, Section 7 of the SAR analyzes, a non-mechanistic continuous atmospheric release during accident conditions. The SAR specifies the bounding degradation of shielding features from the design-basis accidents and presents bounding dose estimates in Section 5 of the SAR. The applicant determined that the radiological exposure at 100 meters will not exceed the 10 CFR 72.106(b) regulatory limit of 5 rem to the whole body or any organ. A summary of the estimated doses for design-basis accidents is presented in Section 10 of the SAR.

The staff reviewed the design-basis accident analyses with respect to 10 CFR 72.106(b) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 72.106(b). Sections 5, 7, and 10 of the SER further evaluate the radiological doses during accident conditions.

11.3 Criticality

As discussed in SER Section 6, the applicant has shown, and the staff has verified, that the spent fuel remains subcritical ($k_{eff} < 0.95$) under all credible normal, off-normal, and postulated accident events. The design-basis off-normal and accident events do not adversely affect the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the HI-STAR 100 system design meets the "double contingency" requirements of 10 CFR 72.124(a).

11.4 Confinement

11.4.1 Off-Normal Conditions

11.4.1.1 Off-Normal Pressures

The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and temperature. The MPC internal pressure is further increased by the assumption that 10% of the fuel rods will rupture, 100% of the fill gas is released, and 30% of fission gases per Table 4.3.4 of the SAR are also released to the cavity.

The design normal condition MPC internal pressure of 100 psig has been established to bound the off-normal internal pressure. Therefore, the structural evaluation of the MPC vessel for off-normal internal pressure condition is the same as the evaluation for normal internal pressure condition. Consequently, the resulting stresses from the off-normal internal pressure condition will be well within the allowable stress limits as shown in Section 3.4 of the SAR.

There is no effect on the confinement function of the MPC as a result of this off-normal condition. Based on the structural evaluation, all stresses of the MPC are within allowable values and the confinement boundary integrity is not affected.

11.4.1.2 Off-Normal Environment Temperature

The HI-STAR 100 system is designed for off-normal environment temperature extremes of -40°F and 100° F. The off-normal environment temperature is postulated as a constant ambient temperature caused by extreme weather conditions.

The effects on the MPC for the maximum off-normal temperature (heat) condition is an increase of the MPC internal pressure. As discussed above, the resulting pressure will be bound by the normal/off-normal design pressure of 100 psig used in the structural analysis. The effect of

minimum off-normal environment temperature is the potential of brittle fracture. The MPC is constructed from austenitic stainless steel materials (Alloy X) and is not susceptible to brittle fracture. The overpack materials will meet the fracture toughness criteria of Regulatory Guides 7.11 and 7.12. The fracture toughness test criteria for overpack components are listed in Table 3.1.18 of the SAR. Thus, there are no adverse effects on the confinement function of the HI-STAR 100 system as a result of this off-normal condition.

11.4.1.3 Leakage of One MPC Seal

The MPC lid-to-shell weld is postulated to fail to confirm the safety of the HI-STAR 100 Cask Systems confinement boundary. The MPC lid-to-shell weld is chosen because it is the main closure weld for the MPC. It is, however, extremely unlikely that the weld inspections and helium leak test would fail to detect a defective weld and leakage.

11.4.1.3.1 Cause of Leakage of MPC Lid Weld

The weld between the MPC lid and shell is dye penetrant inspected on the root and final pass, examined using UT or multi-layer PT techniques, hydrostatically tested, and helium leak tested. The integrity of the weld will be maintained throughout the design life because the MPC is placed inside the overpack within an inert atmosphere. Failure of the MPC lid weld would require failure of all of the following:

- Substandard weld by a qualified welding machine or welder using approved welding procedures.
- Failure to detect unacceptable flaws during the liquid penetrant inspections and the volumetric examination of the weld by qualified inspectors with approved procedures.
- Failure to detect unacceptable leakage during the hydrostatic test performed by qualified personnel in accordance with approved procedures.
- Failure of the qualified leakage test equipment to detect helium leakage in accordance with approved procedures.

The failure of the MPC lid weld is a postulated event to demonstrate safety of the HI-STAR 100 Cask Systems because of the multiple confinement boundaries to contain radioactive materials.

11.4.1.3.2 Consequences of Leakage of MPC Lid Weld

If the MPC lid seal weld were to fail, the MPC closure ring would retain the design pressure and the MPC confinement function would be maintained. If the MPC closure ring weld joint were also to fail, the overpack helium retention boundary would prevent the release of radioactive materials. Thus, the HI-STAR 100 confinement systems will not be breached in this postulated off-normal event.

11.4.2 Accident and Extreme Environmental Events

11.4.2.1 Handling Accident

11.4.2.1.1 Cause of Handling Accident

During the operation of the HI-STAR 100 system, the loaded overpack is transported to the ISFSI in the vertical or horizontal configuration. Although a handling accident event is remote, a cask drop from the handling height limit has been regarded as a credible accident event.

11.4.2.1.2 Consequences of Handling Accident

Structural analysis of the HI-STAR 100 system as a result of a free drop from the vertical and horizontal handling height limit is performed in Section 3 of the SAR. The analysis has shown that the overpack deceleration remains below the design value of 60 g and that stress in the overpack are below the allowable stress limits. There is no effect on the confinement function of the MPC as a result of this event. All stresses in the MPC remain within allowable values assuring the confinement boundary integrity.

11.4.2.2 Tipover Accident

11.4.2.2.1 Cause of Tipover

The analysis of the HI-STAR 100 system has shown that the cask does not tip over as a result of the postulated accident and extreme environmental events (i.e., tornado and tornado missiles, flood water, and earthquake). Thus, the tipover accident is stipulated as a non-mechanistic accident to demonstrate the defense-in-depth features of the design.

11.4.2.2.2 Consequences of Tipover Accident

The maximum deceleration of the HI-STAR 100 system as a result of a non-mechanistic tipover is calculated in Section 3 of the SAR. The structural analysis of the MPC and overpack for a tipover accident are presented in Section 3.4 of the SAR. The stresses within the MPC and overpack are below the allowable stress limits. Consequently, there will be no effect on the confinement function of the MPC.

11.4.2.3 Fire

There are no structural consequences and the confinement function of the MPC is not affected as a result of fire.

11.4.2.4 Partial Blockage of MPC Basket Vent Holes

There are no structural consequences or effects on the confinement function of the MPC as a result of this event.

11.4.2.5 Tornado

11.4.2.5.1 Cause of Tornado

The HI-STAR 100 system will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. It is possible that the HI-STAR 100 system may experience the extreme environmental conditions of a tornado.

11.4.2.5.2 Consequences of Tornado

Tornado winds, with it's high velocity, not only will exert pressure loading but also will have the potential of generating a large tornado missile striking the overpack. This could cause a loaded overpack to tipover. A tornado missile propelled by high velocity winds may also penetrate the overpack helium retention boundary and damage the shielding.

Section 3 of the SAR provides the analysis of the pressure loading which attempts to tipover the overpack and the analysis of the effects of the different types of tornado missiles. These analyses showed that the loaded overpack does not tipover as a result of the tornado winds or tornado missiles. The analyses also showed that the overpack helium retention boundary is not compromised and only minor shielding damage may incur as a result of a tornado missile. Thus, a tornado accident will not have adverse consequences on the structural or the confinement capabilities of the HI-STAR 100 system.

11.4.2.6 Flood

11.4.2.6.1 Cause of Flood

It is possible for the ISFSI site to be flooded. The potential sources for the flood water could be unusually high water from a river, a dam break, a seismic event, or a hurricane.

11.4.2.6.2 Consequences of Flood

The flood accident affects the HI-STAR 100 system structural analysis in two ways. First, the flood water velocity applies force and an overturing moment which may cause sliding or tipover of the loaded overpack. Secondly, the flood water depth applies an external pressure to the overpack. Section 3 of the SAR provided the analyses of both conditions. The results of the analysis showed that the overpack helium retention boundary is not affected and that the loaded overpack will not slide and/or tipover if the flood water velocity does not exceed the design value of 13 ft/sec. Therefore, the overpack helium retention boundary and MPC confinement boundary will not be adversely affected by the flood accident.

11.4.2.7 Earthquakes

11.4.2.7.1 Cause of Earthquake

It is possible that during the use of the HI-STAR 100 system, the ISFSI site may experience an earthquake.

11.4.2.7.2 Consequences of Earthquake

The earthquake accident analysis evaluates the effects of a design-basis seismic event on the loaded HI-STAR 100 system. The analysis objective is to determine the stability of the HI-STAR 100 system. Conservatively based on static stability criteria, Section 3 of the SAR qualified the HI-STAR 100 system for the maximum horizontal zero period acceleration (ZPA) at the top of the storage pad. The HI-STAR 100 system will not tipover or slide under the maximum horizontal ZPA specified in Table 2.2.8 of the SAR. The seismic activity has no adverse effects on confinement function.

11.4.2.8 100% Fuel Rod Rupture

11.4.2.8.1 Cause of 100% Fuel Rod Rupture

Through all accident and extreme events, the HI-STAR system maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the short-term temperature limits, and there is reasonable assurance that the cladding will maintain confinement integrity during a design-basis drop. Thus, there is no credible cause for 100% fuel rod rupture. This accident condition is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure.

11.4.2.8.2 Consequences of 100% Fuel Rod Rupture

The MPC design pressure bounds the pressure developed by assuming 100% fuel rod rupture. Therefore, the structural evaluation of the MPC for the accident condition internal pressure presented in Section 3 of the SAR has demonstrated that the MPC stresses are within the allowable values and the confinement function will not be adversely affected by this postulated accident condition.

11.4.2.9 Confinement Boundary Leakage

There are no credible structural events that would have consequences that would cause a breach of the confinement boundary.

11.4.2.10 Explosion

11.4.2.10.1 Cause of Explosion

An explosion as a result of combustion of the fuel contained in the cask transport vehicle is possible.

11.4.2.10.2 Consequences of Explosion

The explosion accident is bounded by the design-basis accident external pressure of 300 psig. The structural evaluation for the overpack accident condition external pressure demonstrated that all overpack stresses are within the allowable values. There will be no effect on the confinement function of the MPC as a result of this accident event.

11.4.2.11 Lightning

The HI-STAR 100 system is designed to be stored on an unsheltered ISFSI concrete pad. There is a possibility that lightening may strike the overpack. However, the overpack structure contains thousands of pounds of highly conductive carbon steel and has a large external surface area. Thus, it will dissipate the effects of any lightning striking the overpack.

There are no confinement consequences as a result of this event.

11.4.2.12 Burial Under Debris

There are no adverse structural, confinement, or thermal consequences as a result of this event.

11.4.2.13 Extreme Environmental Temperature

Since the structural evaluation for accident internal pressure bounds the pressure resulting from this event, there are no structural or confinement consequences as a result of this event.

11.5 Post-Accident Recovery

Section 11.2 of the SAR discusses corrective actions for each accident identified in Section 11.2. The SAR did not identify a design-basis accident that would affect the canister confinement boundary or significantly damage the cask system at a level that could result in undue risk to public health and safety.

The staff reviewed the design-basis accident analyses with respect to post-accident recovery and found them to be acceptable. The staff has reasonable assurance that the site licensee can recover the HI-STAR 100 from the analyzed design-basis accidents and that the generic corrective actions outlined in the SAR are appropriate to protect public health and safety.

11.6 Instrumentation

Because of the passive nature of the HI-STAR 100 system, no instrumentation and control systems are needed to monitor SSCs important to safety. Therefore, no instrumentation and control systems must remain operational under accident conditions. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. Since the MPC uses an entirely welded redundant closure system and, under normal and off-normal conditions, there are no anticipated mechanisms that would cause weld failure, no direct monitoring of the closure is required.

11.7 Evaluation Findings

F11.1 SSCs of the HI-STAR 100 are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

F11.2 The spacing of casks, discussed in Section 1.4 of the SER and included as an operating limit in Section 12.3.14 of the SAR will ensure accessibility of the equipment and services required for emergency response.

F11.3 The applicant has evaluated the HI-STAR 100 to demonstrate that it will reasonably maintain confinement of radioactive material under off-normal and credible accident conditions.

F11.4 A design-basis accident or a natural phenomena event will not prevent the ready retrieval of spent fuel for further processing or disposal.

F11.5 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.6 Because instrumentation and control systems are not required, no instruments or control systems are required to remain operational under accident conditions.

F11.7 The applicant has evaluated off-normal and design-basis accident conditions to demonstrate with reasonable assurance that the HI-STAR 100 radiation shielding and confinement features are sufficient to meet the requirements in 10 CFR 72.104(a) and 10 CFR 72.106(b).

F11.8 Table 12.1 of the SER lists the TS for the HI-STAR 100. These TS are further discussed in Section 12 of the SER and are appended to the CoC.

F11.9 The staff concludes that the accident design criteria for the HI-STAR 100 are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

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12.0 CONDITIONS FOR CASK USE —TECHNICAL SPECIFICATIONS

The purpose of the review of the conditions for cask use is to determine whether the applicant has fully evaluated the TS and to ensure that the SER incorporates any additional operating controls and limits that the staff deems necessary.

12.1 Conditions for Use

The conditions for use of the HI-STAR 100 Cask System are fully defined in the CoC and the TS and fuel specifications which are appended to it.

12.2 Technical Specifications

Table 12-1 lists the TS for the HI-STAR 100 Cask System. The staff has appended these TS to the CoC for the HI-STAR 100.

TABLE 12-1
HI-STAR 100 TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
2.1	SFSC Integrity
2.1.1	Multipurpose Canister (MPC)
2.1.2	Overpack
2.1.3	SFSC Lifting Requirements
2.1.4	Fuel Cool-Down
2.2	SFSC Radiation Protection
2.2.1	Overpack Average Surface Dose Rates
2.2.2	SFSC Surface Contamination
3.0	Administrative Controls and Programs
3.1	Training Module
3.2	Preoperational Testing and Training Program
3.3	Special Requirements for First System In Place
3.4	Radioactive Effluent Control Program

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the TS for the HI-STAR 100. These TS are further discussed in Section 12 of the SAR and are part of the CoC.

F.12.2 The staff concludes that the conditions for use of the HI-STAR 100 identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether Holtec has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G.

13.1 Areas Reviewed

QA Organization
QA Program
Design Control
Procurement Document Control
Instructions, Procedures, and Drawings
Document Control
Control of Purchased Material, Equipment, and Services
Identification and Control of Materials, Parts, and Components
Control of Special Processes
Licensee Inspection
Test Control
Control of Measuring and Test Equipment
Handling, Storage, and Shipping Control
Inspection, Test, and Operating Status
Nonconforming Materials, Parts, or Components
Corrective Action
QA Records
Audits

NUREG-1536 provides the criteria for evaluating the above 18 areas. In a number of cases, the description of, or actions to be taken by, personnel involved in quality activities were incorporated by reference to the applicable sections of the Holtec's Quality Assurance Manual (HQAM). It was, therefore, necessary to review such referenced sections in the HQAM to determine whether the QA program, as submitted, met the requirements of 10 CFR Part 72, Subpart G. While this evaluation determined that the QA program is acceptable, proper implementation of the QA program will be assessed during future NRC inspections.

13.2 Evaluation Findings

F13.1 The QA program describes the requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G, and 10 CFR Part 21, "Reporting of Defects and Noncompliance."

F13.2 The structure of the organization and assignment of responsibility for each activity ensure that designated parties will perform the work to achieve and maintain specified quality requirements.

F13.3 Conformance to established requirements will be verified by qualified personnel and groups not directly responsible for the activity being performed. These personnel and

groups report through a management hierarchy which grants the necessary authority and organizational freedom and provides sufficient independence from economic and scheduling influences.

F13.4 The QA program is well-documented and provides adequate control over activities affecting quality, as well as SSCs important to safety, consistent with their relative importance to safety (graded approach).

F13.5 Holtec's QA program complies with the applicable NRC regulations and can be implemented for the design, fabrication, testing, modification, and use of the HI-STAR 100 Cask System.

F13.6 This SAR can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided that the applicant applies its NRC-approved QA program meeting the requirements of 10 CFR Part 50, Appendix B, to the design, construction, and use of SSCs of a spent fuel storage installation that are important to safety.

14.0 DECOMMISSIONING

The purpose of the review of the conceptual decommissioning plan for the HI-STAR 100 is to ensure that it provides reasonable assurance that the owner of the cask can conduct decontamination and decommissioning in a manner that adequately protects the health and safety of the public. Nothing in this review considers, or involves the review of, ultimate disposal of spent nuclear fuel.

14.1 Decommissioning Considerations

The conceptual decommissioning plan for the HI-STAR 100 is provided in Section 2.4 of the SAR. While Holtec anticipates that the HI-STAR 100 could be used as part of a final geologic disposal system, the ability to decommission the HI-STAR 100 is also considered. For example, Table 2.4.1 of the SAR provides the quantities of the major nuclides which Holtec has determined would exist after 40 years of irradiation of a HI-STAR 100 cask. The material activation results presented in Table 2.4.1 confirm that total system activation is low. Holtec states that both the overpack and the MPC can be decontaminated using existing mechanical or chemical methods.

14.2 Evaluation Findings

F14.1 Holtec's proposed cask design includes adequate provisions for decontamination and decommissioning. As discussed in Section 2.4 of the SAR, these provisions include facilitating decontamination of the HI-STAR 100, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.

F14.2 Section 2.4.1 of the SAR also presents information concerning the proposed practices and procedures for decontaminating the cask and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects public health and safety.

F14.3 The staff concludes that the decommissioning considerations for the HI-STAR 100 are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the HI-STAR 100 will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

Conclusions

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. HI-STAR 100 Cask System meets the requirements of Part 72.

Principal Contributors

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Conclusions

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. HI-STAR 100 Cask System meets the requirements of Part 72.

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CONCURRENCE PAGE FOR HI-STAR 100 SAFETY EVALUATION REPORT

Each reviewer is concurring with regard to the sections of the Safety Evaluation that she/he was responsible for.

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