

CHAPTER 1[†]: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) for Holtec International's HI-STORM 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under requirements specified in 10CFR72 [1.0.1]. This FSAR describes the basis for NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L, for the HI-STORM 100 System to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI). This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM 100 System, drawings of the structures, systems, and components important to safety, and the qualifications of the certificate holder. This report is also suitable for incorporation into a site-specific Safety Analysis Report, which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility similar in objective and scope. Table 1.0.1 contains a listing of the terminology and notation used in this FSAR.

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable FSAR section that addresses each topic.

The HI-STORM 100 FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all guidance. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.3 herein. Table 1.0.3 also contains the justification for the alternative method for compliance adopted in this FSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

[†] This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

The generic design basis and the corresponding safety analysis of the HI-STORM 100 System contained in this FSAR are intended to bound the SNF characteristics, design, conditions, and interfaces that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This FSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design basis and safety analysis documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM 100 System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM 100 System FSAR identifies a limited number of conditions that are necessarily site-specific and are to be addressed in the licensee's 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage pad (including the embedment for anchored cask users) and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.
- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.
- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.
- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.
- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.
- Performance of pre-operational testing.
- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.
- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

The generic safety analyses contained in the HI-STORM 100 FSAR may be used as input and for guidance by the licensee in performing a 10CFR72.212 evaluation.

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.2.3 is the third figure in Section 1.2 of Chapter 1.

Revisions to this document are made on a section level basis. Complete sections have been replaced if any material in the section changed. The specific changes are noted with revision bars in the right margin. Figures are revised individually. Drawings are controlled separately within the Holtec QA program and have individual revision numbers. Bills-of-Material (BOMs) are considered separate drawings and are not necessarily at the same revision level as the drawing(s) to which they apply. If a drawing or BOM was revised in support of the current FSAR revision, that drawing/BOM is included in Section 1.5 at its latest revision level. Drawings and BOMs appearing in this FSAR may be revised between formal updates to the FSAR. Therefore, the revisions of drawings/BOMs in Section 1.5 may not be current.

1.0.1 Engineering Change Orders

The changes authorized by the following Holtec Engineering Change Orders (ECOs) are reflected in Revision 1 of this FSAR:

MPC-68/68F/68FF: ECOs 1021-1 through 4, 7, 8, 12 through 16, 18 through 23, 27 through 30, 33, 34, 36, 38, 39, 41, 43, and 44; and 71188-43.

MPC-24/24E/24EF: ECOs 1022- 1 through 7, 9, 10, 12 through 26, 28, 31, and 34 through 38.

MPC-32: ECOs 1023-1 and 3 through 10.

HI-STORM overpack: ECOs 1024-1 through 4, 6 through 16, 18 through 21, 24, 25, 27 through 38, 42 through 47, 50, 51, 52, 54 through 58, and 60.

HI-TRAC 125 transfer cask: ECOs 1025-1 through 32, 35, and 36.

HI-TRAC 100 transfer cask: ECOs 1026- 1 through 29.

Ancillary Equipment: ECOs 1027-27, 31, 33, 46, and 53.

General FSAR changes: ECOs 5014-36, 47, 49, 50, 51, 53, 54, 56, 58 through 64, 66, 67, and 68.

Table 1.0.2

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
1. General Description			
1.1 Introduction	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.1
1.2 General Description	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2
1.2.1 Cask Character- istics	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.1
1.2.2 Operational Features	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.2
1.2.3 Cask Contents	1.III.3 DCSS Contents	10CFR72.2(a)(1) 10CFR72.236(a)	1.2.3
1.3 Identification of Agents & Contractors	1.III.4 Qualification of the Applicant	10CFR72.24(j) 10CFR72.28(a)	1.3
1.4 Generic Cask Arrays	1.III.1 General Description & Operational Features	10CFR72.24(c)(3)	1.4
1.5 Supplemental Data	1.III.2 Drawings	10CFR72.24(c)(3)	1.5
NA	1.III.6 Consideration of Transport Requirements	10CFR72.230(b) 10CFR72.236(m)	1.1
NA	1.III.5 Quality Assurance	10CFR72.24(n)	1.3
2. Principal Design Criteria			
2.1 Spent Fuel To Be Stored	2.III.2.a Spent Fuel Specifications	10CFR72.2(a)(1) 10CFR72.236(a)	2.1
2.2 Design Criteria for Environmental Conditions and Natural Phenomena	2.III.2 b External Conditions,	10CFR72.122(b)	2.2
	2.III.3 b Structural, 2.III.3.c Thermal	10CFR72.122(c)	2.2.3.3, 2.2.3.10
		10CFR72.122(b) (1)	2.2
		10CFR72 122(b) (2)	2.2.3.11
		10CFR72 122(h) (1)	2.0

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

BoralTM means Boral manufactured by AAR Advanced Structures.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Commercial Spent Fuel or CSF refers to nuclear fuel used to produce energy in a commercial nuclear power plant.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

Cooling Time for a spent fuel assembly is the time between its discharge from the reactor (reactor shutdown) and the time the spent fuel assembly is loaded into the MPC.

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Table 1.0.1

TERMINOLOGY AND NOTATION

Damaged Fuel Container (or Canister) means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The Damaged Fuel Container/Canister (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC.

Design Heat Load is the computed heat rejection capacity of the HI-STORM system with a certified MPC loaded with CSF stored in uniform storage with the ambient at the normal temperature and the peak cladding temperature (PCT) at 400°C. The Design Heat Load is less than the thermal capacity of the system by a suitable margin that reflects the conservatism in the system thermal analysis.

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this FSAR, if operated and maintained in accordance with this FSAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM 100 System.

Enclosure Vessel means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, and closure ring which provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Fracture Toughness is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

Fuel Basket means a honeycombed structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Debris refers to 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.

High Burnup Fuel, or HBF is a *commercial* spent fuel assembly with an average burnup greater than 45,000 MWD/MTU.

Table 1.0.1

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HI-TRAC transfer cask or HI-TRAC means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. HI-TRAC is an acronym for Holtec International Transfer Cask. In this FSAR there are three HI-TRAC transfer casks, the 125 ton standard design HI-TRAC (HI-TRAC-125), the 125-ton dual-purpose lid design (HI-TRAC 125D), and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to all three HI-TRAC transfer cask design, unless the discussion. The HI-TRAC is equipped with a pair of lifting trunnions and the HI-TRAC 100 and HI-TRAC 125 designs also include pocket trunnions. The trunnions are used to lift and downend/upend the HI-TRAC with a loaded MPC.

HI-STORM overpack or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. The term "overpack" as used in this FSAR refers to both the standard design overpack (HI-STORM 100), the alternate design overpack (HI-STORM 100S), and either of these as an overpack designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA), unless otherwise clarified.

HI-STORM 100 System consists of a loaded MPC placed within the HI-STORM 100 overpack.

Holtite™ is the trade name for all present and future neutron shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A™ is the first and only shielding material qualified under the Holtite R&D program. As such, the terms Holtite and Holtite-A may be used interchangeably throughout this FSAR.

Holtite™ -A is a trademarked Holtec International neutron shield material.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Fuel assemblies without

Table 1.0.1

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fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

License Life means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

Long-term Storage means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad and lasting up to the end of the licensed life of the HI-STORM 100 System (20 years).

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

METAMIC[®] is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs.

METCON[™] is a trade name for the HI-STORM 100 overpack. The trademark is derived from the metal-concrete composition of the HI-STORM 100 overpack.

MGDS is an acronym for Mined Geological Disposal System.

Moderate Burnup Fuel, or MBF is a *commercial* spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions.

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Neutron Absorber Material is a generic term used in this FSAR to indicate any neutron absorber material qualified for use in the HI-STORM 100 System MPCs.

Table 1.0.1

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Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Nominal Helium Fill Pressure is defined as the gas pressure at which a certain quantity of helium is added to an evacuated envelope of a given free volume at a reference temperature of 70 °F.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), water displacement guide tube plugs, and orifice rod assemblies, and vibration suppressor inserts.

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

Plain Concrete is concrete that is unreinforced and is of density specified in this FSAR .

Preferential Fuel Loading is a requirement in the CoC applicable to uniform fuel loading whenever fuel assemblies with significantly different post-irradiation cooling times (≥ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling time are loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times are placed toward the center of the basket. Regionalized fuel loading meets the intent of preferential fuel loading. Preferential fuel loading is a requirement in addition to other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers.

Post-Core Decay Time (PCDT) is synonymous with cooling time.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high heat emitting fuel assemblies to be stored in fuel storage locations in the center of the fuel basket provided lower heat emitting fuel assemblies are stored in the peripheral fuel storage locations. Users choosing regionalized fuel loading must also consider other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers. Regionalized fuel loading meets the intent of preferential fuel loading.

SAR is an acronym for Safety Analysis Report (10CFR71).

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be

Table 1.0.1

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much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Short-term Operating Conditions means those normal operational evolutions necessary to support fuel loading or fuel unloading operations. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and onsite handling of a loaded HI-TRAC transfer cask.

Single Failure Proof means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

Thermal Capacity of the HI-STORM system is defined as the amount of heat in the system containing an MPC loaded with CSF stored in uniform storage will actually reject with the ambient at the normal temperature and the peak fuel cladding temperature (PCT) at 400°C.

Thermosiphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.

Threshold Heat Load is any of the heat loads during short-term operating conditions up to which no time limit or other restriction is imposed on the operating condition.

FSAR is an acronym for Final Safety Analysis Report (10CFR72).

Uniform Fuel Loading is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as preferential fuel loading, and those applicable to non-fuel hardware, and damaged fuel containers.

ZPA is an acronym for zero period acceleration.

ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor. Any reference to Zircaloy fuel cladding in this FSAR applies to any zirconium-based fuel cladding material.

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b) (2)	2.2.3.5
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b) (2)	2.2.3.6
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b) (2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b) (2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a) (4)	2.3.2.2, 4.0
NA	2.III.3f Operating Procedures	10CFR72.24(f) 10CFR72.128(a) (5)	10.0, 8.0
		10CFR72.236(h)	8.0
		10CFR72.24(1)(2)	1.2.1, 1.2.2
		10CFR72.236(1)	2.3.2.1
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a) (1)	9.0

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a) (2)	2.3.5.2
		10CFR72.128(a) (3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
10CFR72.236(e)	2.3.2.1		
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h) (4) 10CFR72.122(i) 10CFR72.128(a) (1)	2.3.5
2.3.4 Nuclear Criticality Safety	2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236(h)	2.4
	14.III.1 Design	10CFR72.130	2.4

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(i)
	14.III.4 License Termination	10CFR72.54	(i)
3. Structural Evaluation			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.24(c)	3.1
3.2 Weights and Centers of Gravity	3.V.1.b 2 Structural Design Features	--	3.2
3.3 Mechanical Properties of Materials	3.V.1.c Structural Materials	10CFR72.24(c)(3)	3.3
	3.V.2.c Structural Materials		
NA	3.III.2 Radiation Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(l)	3.4.4.3 3.4.7.3 3.4.10
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4.3
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.102(f)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.236(g)	3.4.11 3.4.12
3.4 General Standards for Casks	--	--	3.4
3.4.1 Chemical and Galvanic Reactions	3.V.1 b.2 Structural Design Features	--	3.4.1
3.4.2 Positive Closure	--	--	3.4.2

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions --	--	3.4.3, Appendices 3-E, 3-A-C, 3-D
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4, Appendices 3-I, 3-U, 3-V, 3-W
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h) (1)	3.5
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a) (4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria	10CFR72.122(h) (1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
5. Shielding Evaluation			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configura- tions	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a) (2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
6. Criticality Evaluation			
6.1 Discussion and Results	--	--	6.1
6.2 Spent Fuel Loading	6.V.2 Fuel Specification	--	6.1, 6.2
6.3 Model Specifications	6.V.3 Model Specification	--	6.3
6.3.1 Description of Calcula- tional Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6 Supplemental Data	6.V.5 Supplemental Info.	--	Appendices 6.B, 6 C, and 6 D
7. Confinement			
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety	10CFR72.24(c)(3) 10CFR72.24(i)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h) (l)	7.1, 7.1.1, 7.2.2
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.236(1)	7.2
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7.2.1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h) (4) 10CFR72.128(a) (l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose	10CFR72.104(a)	7.3.5
7.2.2 Pressurization of Confinement Vessel	--	--	7.2.2
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.3
7.3.1 Fission Gas Products	--	--	7.3.1
7.3.2 Release of Contents	--	--	7.3.3
NA	--	10CFR72.106(b)	7.3
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.1.5, 8.5.2
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0 Introduction
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0 Introduction
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	8.1, 8.3
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.3
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.3
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.3.3
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.8 Ready Retrieval	10CFR72.122(1)	8.3
8.3 Preparation of the Cask	--	--	8.3.2
8.4 Supplemental Data	--	--	Tables 8.1.1 to 8.1.10
NA	8.III.9 Design to Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a)(5)	8.1, 8.3
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 8.1.6
9. Acceptance Criteria and Maintenance Program			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	8.1, 9.1

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	9.1
	9.III.1.d Test Program	10CFR72.162	9.1
	9.III.1.e Appropriate Tests	10CFR72.236(1)	9.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1 ⁽²⁾
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a) (1)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b) (8)	9.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	⁽³⁾
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	⁽⁴⁾
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	9.1.7, 9.1.1.(12)
	9.III.3 Cask Identification		
10. Radiation Protection			

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1 b Design Features	10CFR72.126(a)(6)	10.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
11. Accident Analyses			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	11.2

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
12. Operating Controls and Limits			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart E 10CFR72 Subpart F	12.0
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 12.A
12.2.2 Limiting	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
Conditions for Operation	12.III.2 a Type of Spent Fuel	10CFR72.236(a)	Appendix 12 A
	12.III.2.b Enrichment		
	12.III.2 c Burnup		
	12.III.2 d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SCC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	8.0
NA	12.III.2 Confinement Effectiveness	10CFR72.236(j)	9.0
NA	12.III.2 Evaluation for Confinement	10CFR72.236(l)	7.1, 7.2, 9.0
13. Quality Assurance			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) 10CFR72.140(d)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Notes.

- (1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application
- (2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission's acceptance of the tests defined in Chapter 9.
- (3) Not applicable to HI-STORM 100 System. The functional adequacy of all important to safety components is demonstrated by analyses.
- (4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.
- (5) The stated requirement is not applicable to the HI-STORM 100 System. No monitoring is required for accident conditions.
- “—” There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.
- “NA” There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>2.V.2.(b)(1) "The NRC accepts as the maximum and minimum "normal" temperatures the highest and lowest ambient temperatures recorded in each year, averaged over the years of record."</p>	<p><u>Exception:</u> Section 2.2.1.4 for environmental temperatures utilizes an upper bounding value of 80°F on the annual average ambient temperatures for the United States.</p>	<p>The 80°F temperature set forth in Table 2.2.2 is greater than the annual average ambient temperature at any location in the continental United States. Inasmuch as the primary effect of the environmental temperature is on the computed fuel cladding temperature to establish long-term fuel cladding integrity, the annual average ambient temperature for each ISFSI site should be below 80°F. The large thermal inertia of the HI-STORM 100 System ensures that the daily fluctuations in temperatures do not affect the temperatures of the system. Additionally, the 80°F ambient temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours.</p>
<p>2.V.2.(b)(3)(f) "10CFR Part 72 identifies several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for spent fuel storage."</p>	<p><u>Clarification:</u> A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM 100 System must be performed prior to use if these events are applicable to the site.</p>	<p>In accordance with NUREG-1536, 2.V.(b)(3)(f), if seiche, tsunami, and hurricane are not addressed in the SAR and they prove to be applicable to the site, a safety analysis is required prior to approval for use of the DCSS under either a site specific, or general license.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."</p>	<p><u>Clarification:</u> As stated in NUREG-1536, 3.V.(d), page 3-11, "Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted outside the spent fuel building, or the maximum deceleration that the cask could experience in a drop." The maximum deceleration for a corner drop is specified as 45g's for the HI-STORM overpack. No carry height limit is specified for the corner drop.</p>	<p>In Chapter 3, the MPC and HI-STORM overpack are evaluated under a 45g radial loading. A 45g axial loading on the MPC is bounded by the analysis presented in the HI-STAR FSAR, Docket 72-1008, under a 60g loading, and is not repeated in this FSAR. In Chapter 3, the HI-STORM overpack is evaluated under a 45g axial loading. Therefore, the HI-STORM overpack and MPC are qualified for a 45g loading as a result of a corner drop. Depending on the design of the lifting device, the type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses may be required to demonstrate that the deceleration limit of 45g's is not exceeded.</p>
<p>3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced.. "</p> <p>3.V.2.b.i (2)(b), Page 3-20, Para 1, "The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359".</p> <p>3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and construction of reinforced concrete structures important to safety but not within the scope of ACI 359 should comply with the requirements of ACI 349".</p>	<p><u>Exception:</u> The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used for the material selection and specification, and construction of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack. ACI 318-95 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.</p>	<p>Concrete is provided in the HI-STORM overpack solely for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose.</p> <p>To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. In addition, the temperature limits for normal and off-normal condition from ACI 349 will be imposed.</p> <p>Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
		utilized plain concrete for shielding purposes, which is important to safety.
<p>3.V.3.b.i.(2), Page 3-29, Para. 1, "The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with Section III of the ASME B&PV Code."</p>	<p><u>Clarification:</u> The HI-STORM overpack steel structure is designed in accordance with the ASME B&PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.</p>	<p>The overpack structure is a steel weldment consisting of "plate and shell type" members. As such, it is appropriate to design the structure to Section III, Class 3 of Subsection NF. The very same approach has been used in the structural evaluation of the "intermediate shells" in the HI-STAR 100 overpack (Docket Number 72-1008) previously reviewed and approved by the USNRC.</p>
<p>4.V.5, Page 4-2 "for each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage."</p> <p>4.V.1, Page 4-3, Para. 1 "the staff should verify that cladding temperatures for each fuel type proposed for storage will be below the expected damage thresholds for normal conditions of storage."</p> <p>4.V.1, Page 4-3, Para. 2 "fuel cladding limits for each fuel type should be defined in the SAR with thermal restrictions in the DCSS technical specifications."</p> <p>4.V.1, Page 4-3, Para. 4 "the applicant should verify that these cladding temperature limits are appropriate for all fuel types proposed for storage, and that the fuel cladding temperatures will remain below the limit for facility</p>	<p><u>Clarification:</u> As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak fuel cladding temperature limit.</p>	<p>As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak <i>normal</i> fuel cladding temperature limit of 400°C. All major variations in fuel parameters are considered in the determination of the peak fuel cladding temperature limits. Minor variations in fuel parameters within an array type are bounded by the conservative determination of the peak fuel cladding temperature limit.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
operations (e.g , fuel transfer) and the worst-case credible accident."		
4.V.4.a, Page 4-6, Para. 3 "applicants seeking NRC approval of specific internal convection models should propose, in the SAR, a comprehensive test program to demonstrate the adequacy of the cask design and validation of the convection models."	<u>Exception:</u> The natural convection model described in Subsection 4.4.1. is based on classical correlations for natural convection in differentially heated cavities which have been validated by many experimental studies. Therefore, no additional test program is proposed.	Many experimental studies of this mechanism have been performed by others and reported in open literature sources. As discussed in Subsection 4.4.1, natural convection has been limited to the relatively large MPC basket to shell peripheral gaps. Subsection 4 4.1 provides sufficient references to experiments which document the validity of the classical correlation used in the analysis.
4.V.4 a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."	<u>Clarification:</u> As discussed in Subsection 4.4.2, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based update of Wooten-Epstein (described in Subsection 4.4.1.1.2)	The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature difference conservatively maximizes the basket wall temperature.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>4.V.4.b, Page 4-7, Para. 2 "if the thermal model is axisymmetric or three-dimensional, the longitudinal thermal conductivity should generally be limited to the conductivity of the cladding (weighted fractional area) within the fuel assembly."</p>	<p><u>Clarification:</u> As described in Subsection 4.4.1.1.4, the axial thermal conductivity of the fuel basket is set equal to the cross-sectional thermal conductivity. An equivalent isotropic thermal conductivity approach is adopted.</p>	<p>Due to the large number of gaps in the cross-sectional heat transfer paths, use of the fuel basket cross-sectional thermal conductivity for the axial thermal conductivity severely underpredicts the axial thermal conductivity of the fuel basket region. This imposed axial thermal conductivity restriction is even more limiting than that imposed by this requirement of NUREG-1536. The computer code FLUENT used in the thermal analyses is not capable of modeling anisotropy in the thermal conductivity of a porous medium, such as the fuel basket region. An equivalent isotropic conductivity is used to conservatively represent the heat dissipation characteristics of the fuel basket.</p>
<p>4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."</p>	<p><u>Exception:</u> All calculations of fuel assembly effective thermal conductivities, described in Subsection 4.4.1.1.2, use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.</p>	<p>Within Subsection 4.4.1.1.2, the calculated effective thermal conductivities based on nominal design fuel dimensions are compared with available literature values and are demonstrated to be conservative by a substantial margin.</p>
<p>4.V.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented."</p>	<p><u>Clarification:</u> No additional heat balance is performed or provided.</p>	<p>The FLUENT computational fluid dynamics program used to perform evaluations of the HI-STORM Overpack and HI-TRAC transfer cask, which uses a discretized numerical solution algorithm, enforces an energy balance on all discretized volumes throughout the computational domain. This solution method, therefore, ensures a heat balance at the surface of the cask.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."</p>	<p><u>Exception:</u> No input or output file listings are provided in Chapter 4.</p>	<p>A complete set of computer program input and output files would be in excess of three hundred pages. All computer files are considered proprietary because they provide details of the design and analysis methods. In order to minimize the amount of proprietary information in the FSAR, computer files are provided in the proprietary calculation packages.</p>
<p>4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures."</p>	<p><u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) are calculated using maximum weights and minimum densities.</p>	<p>Calculating the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) using maximum weights and minimum densities conservatively overpredicts the volume occupied by the internal components and correspondingly underpredicts the remaining free volume.</p>
<p>7.V.4.c, Page 7-7, Para. 2 and 3 "Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145 is not typically applied." and "Note that for an instantaneous release (and instantaneous exposure), the time that an individual remains at the controlled area boundary is not a factor in the dose calculation."</p>	<p><u>Exception:</u> As described in Section 7.3, in lieu of an instantaneous release, the assumed leakage rate is set equal to the leakage rate acceptance criteria (5×10^{-6} atm-cm³/s) plus 50% for conservatism, which yields 7.5×10^{-6} atm-cm³/s. Because the release is assumed to be a leakage rate, the individual is assumed to be at the controlled area boundary for 720 hours. Additionally, the atmospheric dispersion factors of Regulatory Guide 1.145 are applied.</p>	<p>The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g., helium leakage, hydrostatic, and volumetric weld inspection).</p> <p>The NRC letter to Holtec International dated 9/15/97, Subject: Supplemental Request for Additional Information - HI-STAR 100 Dual Purpose Cask System (TAC No. L22019), RAI 7.3 states "use the verified confinement boundary leakage rate in lieu of the assumption that the confinement boundary fails."</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."</p>	<p><u>Clarification:</u> Section 9.1.1.1 and the Design Drawings specify that the ASME Code, Section III, Subsection NB, Article NB-5332 will be used for the acceptance criteria for the volumetric examination of the MPC lid-to-shell weld.</p>	<p>In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria as defined by either the ASME code, or an alternative approach..." The ASME Code, Section III, Subsection NB, Paragraph NB-5332 is appropriate acceptance criteria for pre-service examination.</p>
<p>9.V.1.d, Para. 1 "Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron absorbing material."</p>	<p><u>Exception:</u> Subsection 9.1.5 describes the control of special processes, such as neutron shield material installation, to be performed in lieu of scanning or probing with neutron sources.</p>	<p>The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation.</p> <p>The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures.</p> <p>The composition of the Holtite-A is confirmed by inspection and tests prior to first use.</p> <p>Following the first loading for the HI-TRAC transfer cask and each HI-STORM overpack, a shield effectiveness test is performed in accordance with written approved procedures, as specified in Section 9.1.</p>
<p>13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance' . "</p>	<p><u>Exception:</u> Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.</p>	<p>The NRC has approved Revision 13 of the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec intends to apply this QA program to dry storage cask activities Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>ISG-15, Section X 5.4.2, "No more than 1% of the rods in an assembly have peak cladding oxide thicknesses greater than 80 micrometers and no more than 3% of the rods in an assembly have peak cladding oxide thicknesses greater than 70 micrometers. A high burnup fuel assembly should be treated as potentially damaged fuel if the assembly does not meet both of the above criteria or if the fuel assembly contains fuel rods with oxide that has become detached or spalled from the cladding.</p>	<p>The Fuel Cladding Oxide Thickness Evaluation Program in Section 5.0 of Appendix A to the CoC provides an equation to calculate the maximum allowable high burnup fuel cladding oxide thickness, based on fuel assembly type.</p>	<p>FSAR Appendix 4.A, Section 4.A.9 provides the justification for this deviation from NUREG-1536 (ISG-15).</p>
<p><i>ISG-5, Revision 1 provides review guidance for dry storage cask confinement analyses. Sections 3 and 4 discuss nuclides with potential for release and confinement analysis, respectively. The ISG is silent with regard to credit for gravitational settling of certain isotopes inside the MPC cavity.</i></p>	<p><i>The HI-STORM confinement analysis described in Chapter 7 takes a conservative amount of credit for gravitational settling of certain isotopes inside the MPC in calculating the amount of radioactive material that is available for airborne release.</i></p>	<p><i>NRC Report SMSAB-00-03, "Best-Estimate Offsite Dose from Dry Storage Cask Leakage" provides a methodology for taking credit for gravitational settling of isotopes in the Holtec MPC. Holtec is using this methodology to more accurately estimate the confinement doses, notwithstanding the fact that the MPC confinement boundary is designed to maintain integrity under all normal, off-normal, and accident conditions.</i></p>

1.1 INTRODUCTION

HI-STORM 100 (acronym for Holtec International Storage and Transfer Operation Reinforced Module) is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10CFR72. The annex "100" is a model number designation which denotes a system weighing over 100 tons. The HI-STORM 100 System consists of a sealed metallic canister, herein abbreviated as the "MPC", contained within an overpack. Its design features are intended to simplify and reduce on-site SNF loading, handling, and monitoring operations, and to provide for radiological protection and maintenance of structural and thermal safety margins.

The HI-STORM 100S overpack is a variant of the HI-STORM 100 overpack and has its own set of drawings in Section 1.5. The "S" suffix indicates an enhanced overpack design, as described later in this section. The HI-STORM 100S accepts the same MPCs and fuel types as the HI-STORM 100 and the basic structural, shielding, and thermal-hydraulic characteristics remain unchanged. Hereafter in this FSAR reference to HI-STORM 100 System or the HI-STORM overpack is construed to apply to both the HI-STORM 100 and the HI-STORM 100S. Where necessary, the text distinguishes between the two overpack designs. See Figures 1.1.1A and 1.1.3A for a pictorial view of the HI-STORM 100S overpack design.

The HI-STORM 100A overpack is a third variant of the HI-STORM 100 family and is specially outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the ISFSI pad in high seismic applications. In the following, the modified structure of the HI-STORM 100A, in each of four quadrants, is denoted as a "sector lug." The HI-STORM 100A design is also applicable to the HI-STORM 100S overpack, in which case the assembly would be named HI-STORM 100SA. Hereafter in the text, discussion of HI-STORM 100A applies to both the standard (HI-STORM 100A) and short (HI-STORM 100SA) overpacks, unless otherwise clarified.

The HI-STORM 100 System is designed to accommodate a wide variety of spent nuclear fuel assemblies in a single overpack design by utilizing different MPCs. The external dimensions of all MPCs are identical to allow the use of a single overpack. Each of the MPCs has different internals (baskets) to accommodate distinct fuel characteristics. Each MPC is identified by the maximum quantity of fuel assemblies it is capable of receiving. The MPC-24, MPC-24E, and MPC-24EF contain a maximum of 24 PWR fuel assemblies; the MPC-32 and MPC-32F contains a maximum of 32 PWR fuel assemblies; and the MPC-68, MPC-68F, and MPC-68FF contain a maximum of 68 BWR fuel assemblies.

The HI-STORM overpack is constructed from a combination of steel and concrete, both of which are materials with long, proven histories of usage in nuclear applications. The HI-STORM overpack incorporates and combines many desirable features of previously-approved concrete and metal module designs. In essence, the HI-STORM overpack is a hybrid of metal and concrete systems, with the design objective of emulating the best features and dispensing with the drawbacks of both. The HI-STORM overpack is best referred to as a METCON™ (metal/concrete composite) system.

Figures 1.1.1 and 1.1.1A show the HI-STORM 100 with two of its major constituents, the MPC and the storage overpack, in a cut-away view. The MPC, shown partially withdrawn from the storage overpack, is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the confinement boundary for the stored spent nuclear fuel assemblies with respect to 10CFR72 requirements and attendant review considerations. The HI-STORM 100 storage overpack provides mechanical protection, cooling, and radiological shielding for the contained MPC.

In essence, the HI-STORM 100 System is the storage-only counterpart of the HI-STAR 100 System (Docket Numbers 72-1008 (Ref. [1.1.2]) and 71-9261 (Ref. [1.1.3])). Both HI-STORM and HI-STAR are engineered to house identical MPCs. Since the MPC is designed to meet the requirements of both 10CFR71 and 10CFR72 for transportation and storage, respectively, the HI-STORM 100 System allows rapid decommissioning of the ISFSI by simply transferring the loaded MPC's directly into HI-STAR 100 overpacks for off-site transport. This alleviates the additional fuel handling steps required by storage-only casks to unload the cask and repackage the fuel into a suitable transportation cask.

In contrast to the HI-STAR 100 overpack, which provides a containment boundary for the SNF during transport, the HI-STORM storage overpack does not constitute a containment or confinement enclosure. The HI-STORM-100 overpack is equipped with large penetrations near its lower and upper extremities to permit natural circulation of air to provide for the passive cooling of the MPC and the contained radioactive material. The HI-STORM is engineered to be an effective barrier against the radiation emitted by the stored materials, and an efficiently configured metal/concrete composite to attenuate the loads transmitted to the MPC during a natural phenomena or hypothetical accident event. Other auxiliary functions of the HI-STORM 100 overpack include isolation of the SNF from abnormal environmental or man-made events, such as impact of a tornado borne missile. As the subsequent chapters of this FSAR demonstrate, the HI-STORM overpack is engineered with large margins of safety with respect to cooling, shielding, and mechanical/structural functions.

The HI-STORM 100 System is autonomous inasmuch as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM 100 System since it is completely passive and is composed of materials with long proven histories in the nuclear industry. The HI-STORM 100 System can be used either singly or as the basic storage module in an ISFSI. The site for an ISFSI can be located either at a reactor or away from a reactor.

The information presented in this report is intended to demonstrate the acceptability of the HI-STORM 100 System for use under the general license provisions of Subpart K by meeting the criteria set forth in 10CFR72.236.

The modularity of the HI-STORM 100 System accrues several advantages. Different MPCs, identical in exterior dimensions, manufacturing requirements, and handling features, but different in their SNF arrangement details, are designed to fit a common overpack. Even though the different MPCs have

fundamentally identical design and manufacturing attributes, qualification of HI-STORM 100 requires consideration of the variations in the characteristics of the MPCs. In most cases, however, it is possible to identify the most limiting MPC geometry and the specific loading condition for the safety evaluation, and the detailed analyses are then carried out for that bounding condition. In those cases where this is not possible, multiple parallel analyses are performed.

The HI-STORM overpack is not engineered for transport and, therefore, will not be submitted for 10CFR Part 71 certification. HI-STORM 100, however, is designed to possess certain key elements of flexibility.

For example:

- The HI-STORM overpack is stored at the ISFSI pad in a vertical orientation which helps minimize the size of the ISFSI and leads to an effective natural convection cooling flow around the MPC.
- The HI-STORM overpack can be loaded with a loaded MPC using the HI-TRAC transfer cask inside the 10CFR50 [1.1.4] facility, prepared for storage, transferred to the ISFSI, and stored in a vertical configuration, or directly loaded using the HI-TRAC transfer cask at or nearby the ISFSI storage pad.

The version of the HI-STORM overpack equipped with sector lugs to anchor it to the ISFSI pad is labeled HI-STORM 100A, shown in Figure 1.1.4. Figure 1.1.5 shows the sector lugs and anchors used to fasten the overpack to the pad in closer view. Details on HI-STORM 100A are presented in the drawing and BOM contained in Section 1.5. Users may employ a double nut arrangement as an option. The HI-STORM 100A overpack will be deployed at those ISFSI sites where the postulated seismic event (defined by the three orthogonal ZPAs) exceeds the maximum limit permitted for free-standing installation. The design of the ISFSI pad and the embedment are necessarily site-specific and the responsibility of the ISFSI owner. These designs shall be in accordance with the requirements specified in Appendix 2.A. The jurisdictional boundary between the anchored cask design and the embedment design is defined in Table 2.0.5. Additional description on the HI-STORM 100A configuration is provided in Subsection 1.2.1.2.1.

The MPC is a multi-purpose SNF storage device both with respect to the type of fuel assemblies and its versatility of use. The MPC is engineered as a cylindrical prismatic structure with square cross section storage cavities. The number of storage locations depends on the type of fuel. Regardless of the storage cell count, the construction of the MPC is fundamentally the same; it is built as a honeycomb of cellular elements positioned within a circumscribing cylindrical canister shell. The manner of cell-to-cell weld-up and cell-to-canister shell interface employed in the MPC imparts extremely high structural stiffness to the assemblage, which is an important attribute for mechanical accident events. Figure 1.1.2 shows an elevation cross section of an MPC.

The MPC is identical to those presented in References [1.1.2] and [1.1.3], except for MPC-24E, 24EF, 32, 32F and 68FF, until such time as those CoCs are amended to include these additional MPC models. Referencing these documents, as applicable, avoids repetition of information on the MPCs which is comprehensively set forth in the above-mentioned Holtec International documents docketed with the NRC.

However, sufficient information and drawings are presented in this report to maintain clarity of exposition of technical data.

The HI-STORM storage overpack is designed to provide the necessary neutron and gamma shielding to comply with the provisions of 10CFR72 for dry storage of SNF at an ISFSI. Cross sectional views of the HI-STORM storage overpacks are presented in Figures 1.1.3 1.1.3A. A HI-TRAC transfer cask is required for loading of the MPC and movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. The HI-TRAC is engineered to be emplaced with an empty MPC into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC/MPC assembly is designed to preclude intrusion of pool water into the narrow annular space between the HI-TRAC and the MPC while the assembly is submerged in the pool water. The HI-TRAC transfer cask also allows dry loading (or unloading) of SNF into the MPC.

To summarize, the HI-STORM 100 System has been engineered to:

- minimize handling of the SNF;
- provide shielding and physical protection for the MPC;
- permit rapid and unencumbered decommissioning of the ISFSI;
- require minimal ongoing surveillance and maintenance by plant staff;
- minimize dose to operators during loading and handling;
- allow transfer of the loaded MPC to a HI-STAR overpack for transportation.

1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

1.2.1 System Characteristics

The basic HI-STORM 100 System consists of interchangeable MPCs providing a confinement boundary for BWR or PWR spent nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the storage overpack. Figure 1.2.1 provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a storage overpack. Figure 1.2.1A provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a HI-STORM 100S storage overpack. Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. All structures, systems, and components of the HI-STORM 100 System which are identified as Important to Safety are specified in Table 2.2.6. This discussion is supplemented with a full set of detailed design drawings in Section 1.5.

The HI-STORM 100 System is comprised of three discrete components:

- i. multi-purpose canister (MPC)
- ii. storage overpack (HI-STORM)
- iii. transfer cask (HI-TRAC)

Necessary auxiliaries required to deploy the HI-STORM 100 System for storage are:

- i. vacuum drying (or other moisture removal) system
- ii. helium (He) backfill system with leakage detector
- iii. lifting and handling systems
- iv. welding equipment
- v. transfer vehicles/trailer

All MPCs have identical exterior dimensions that render them interchangeable. The outer diameter of the MPC is 68-3/8 inches[†] and the overall length is 190-1/2 inches. See Section 1.5 for the ~~detailed design~~ *MPC* drawings. Due to the differing storage contents of each MPC, the maximum loaded weight differs among MPCs. See Table 3.2.1 for each MPC weight. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons. Tables 1.2.1 and 1.2.2 contain the key parameters for the MPCs.

A single, base HI-STORM overpack design is provided which is capable of storing each type of MPC. The overpack inner cavity is sized to accommodate the MPCs. The inner diameter of the overpack inner shell is

[†] Dimensions discussed in this section are considered nominal values

73-1/2 inches and the height of the cavity is 191-1/2 inches. The overpack inner shell is provided with channels distributed around the inner cavity to present an inside diameter of 69- 1/2 inches. The channels are intended to offer a flexible medium to absorb some of the impact during a non-mechanistic tip-over, while still allowing the cooling air flow through the ventilated overpack. The outer diameter of the overpack is 132-1/2 inches. The overall height of the HI-STORM 100 and the HI-STORM 100S is 239- 1/2 inches. There are two versions of the HI-STORM 100S overpack, differing only in height and weight. The HI-STORM 100S(232) is 232 inches high, and the HI-STORM 100S(243) is 243 inches high. The HI-STORM 100s(243) is approximately 10,100 lbs heavier, including concrete. Hereafter in the text, these two versions of the HI-STORM 100S overpack will only be referred to as HI-STORM 100S and will be discussed separately only if the design feature being discussed is different between the two overpacks. See Section 1.5 for drawings. The weight of the overpack without an MPC is approximately 135 tons. See Table 3.2.1 for the detailed weights.

Before proceeding to present detailed physical data on the HI-STORM 100 System, it is of contextual importance to summarize the design attributes which enhance the performance and safety of the system. Some of the principal features of the HI-STORM 100 System which enhance its effectiveness as an SNF storage device and a safe SNF confinement structure are:

- the honeycomb design of the MPC fuel basket;
- the effective distribution of neutron and gamma shielding materials within the system;
- the high heat dissipation capability;
- engineered features to promote convective heat transfer;
- the structural robustness of the steel-concrete-steel overpack construction.

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flange plate weldment where all structural elements (i.e., box walls) are arrayed in two orthogonal sets of plates. Consequently, the walls of the cells are either completely co-planar (i.e., no offset) or orthogonal with each other. There is complete edge-to-edge continuity between the contiguous cells.

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass of the basket over the entire length of the basket. Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform basket. In other words, the honeycomb basket is a most effective radiation attenuation device. The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the MPC an effective heat rejection device.

The composite shell construction in the overpack, steel-concrete-steel, allows ease of fabrication and eliminates the need for the sole reliance on the strength of concrete.

A description of each of the components is provided in the following sections, along with information with respect to its fabrication and safety features. This discussion is supplemented with the full set of drawings in Section 1.5.

1.2.1.1 Multi-Purpose Canisters

The MPCs are welded cylindrical structures as shown in cross sectional views of Figures 1.2.2 through 1.2.4. The outer diameter and cylindrical height of each MPC are fixed. Each spent fuel MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring, as depicted in the MPC cross section elevation view, Figure 1.2.5. The number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics.

There are ~~seven~~ eight MPC models, distinguished by the type and number of fuel assemblies authorized for loading. *Section 1.2.3 and Table 1.2.1 describe the allowable contents for each MPC model. The MPC-24 is designed to store up to 24 intact PWR fuel assemblies. The MPC-24E is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies. The MPC-24EF is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies or fuel classified as fuel debris. The MPC-68 is designed to store up to 68 total BWR fuel assemblies including up to 68 damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies. Damaged BWR fuel assemblies other than Dresden Unit 1 and Humboldt Bay are limited to 16 fuel storage locations in the MPC-68 with the remainder being intact BWR fuel assemblies, up to a total of 68. The MPC-68F is designed to store up to 68 intact or damaged Dresden Unit 1 and Humboldt Bay BWR fuel assemblies. Up to four of the 68 fuel storage locations in the MPC-68F may be Dresden Unit 1 and Humboldt Bay BWR fuel assemblies classified as fuel debris. The MPC-68FF is designed to store up to 68 total BWR fuel assemblies including up to 16 damaged BWR fuel assemblies. Up to eight (8) of the 16 BWR damaged fuel assembly storage locations may be filled with BWR fuel classified as fuel debris. In addition, all fuel loading combinations permitted in the MPC-68F are also permitted in the MPC-68FF. Design Drawings for all of the MPCs are provided in Section 1.5.*

The MPC provides the confinement boundary for the stored fuel. Figure 1.2.6 provides an elevation view of the MPC confinement boundary. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring. The confinement boundary is a strength-welded enclosure of all stainless steel construction.

The PWR MPC-24, MPC-24E and MPC-24EF differ in construction from the MPC-32 (*including the MPC-32F*) and the MPC-68 (*including the MPC-68F and MPC-68FF*) in one important aspect: the fuel storage cells *in the MPC-24 series* are physically separated from one another by a "flux trap", for criticality control. The PWR MPC-32 *and -32F are* is designed similar to the MPC-68 (without flux traps) and its design includes credit for soluble boron in the MPC water during wet fuel loading and unloading operations for criticality control.

The MPC fuel baskets of non-flux trap construction (namely, MPC-68, MPC-68F, MPC-68FF, and MPC-32, and MPC-32F) are formed from an array of plates welded to each other at their intersections. In the flux-trap type fuel baskets (MPC-24, MPC-24E, and MPC-24EF), formed angles are interposed onto the orthogonally configured plate assemblage to create the required flux-trap channels (see MPC-24 and MPC-24E design drawings in Section 1.5). In both configurations, two key attributes of the basket are preserved:

- i. The cross section of the fuel basket simulates a multi-flanged closed section beam, resulting in extremely high bending rigidity.
- ii. The principal structural frame of the basket consists of co-planar plate-type members (i.e., no offset).

This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls that must transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (e.g., non-mechanistic tipover, uncontrolled lowering of a cask during on-site transfer, or off-site transport events, etc.).

The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. Between the periphery of the basket, the MPC shell, and the basket supports, optional *aluminum* heat conduction elements (AHCEs) may have been installed in the early vintage MPCs fabricated, certified, and loaded under the original version or Amendment 1 of the HI-STORM 100 System CoC. The presence of these aluminum heat conduction elements is acceptable for MPCs loaded under the original CoC or Amendment 1, since the governing thermal analysis for Amendment 1 conservatively modeled the AHCEs as restrictions to convective flow in the basket, but took no credit for heat transfer through them. The heat loads authorized under Amendment 1 bound those for the original CoC, with the same MPC design. For MPCs loaded under Amendment 2 or a later version of the HI-STORM 100 CoC, the aluminum heat conduction elements shall not be installed since they were removed from the thermal model in Amendment 2. MPCs both with and without aluminum heat conduction elements installed are compatible with all HI-STORM overpacks. If used, these heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design that allows a snug fit in the confined spaces and ease of installation. If used, the heat conduction elements are installed along the full length of the MPC basket except at the drain pipe location to create a nonstructural thermal connection that facilitates heat transfer from the basket to shell. In their operating condition, the heat conduction elements contact the MPC shell and basket walls.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit placement of the empty MPC into the HI-TRAC transfer cask. The lifting lugs also serve to axially locate the MPC lid prior to welding. These internal lifting lugs are not used to handle a loaded MPC. Since the MPC lid is installed prior to any handling of a loaded MPC, there is no access to the lifting lugs once the MPC is loaded.

The top end of the MPC incorporates a redundant closure system. Figure 1.2.6 shows the MPC closure details. The MPC lid is a circular plate (fabricated from one piece, or two pieces - split top and bottom) edge-welded to the MPC outer shell. If the two-piece lid design is employed, only the top piece is analyzed as part of the enclosure vessel pressure boundary. The bottom piece acts as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. The lid is equipped with vent and drain ports that are utilized to remove moisture and air from the MPC, and backfill the MPC with a specified amount of inert gas (helium). The vent and drain ports are covered and seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by threaded holes in the MPC lid.

To maintain a constant exterior axial length between the PWR MPCs and the BWR MPCs the thickness of the PWR MPCs' lid is 1/2 inch thinner than the MPC-68s' lid to accommodate the longest PWR fuel assembly which is approximately a 1/2 inch longer than the longest BWR fuel assembly. For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested values for the upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type. The actual length of fuel spacers will be determined on a site-specific or fuel assembly-specific basis.

The MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and optional aluminum heat conduction elements). No carbon steel parts are permitted in the MPC. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the MPC. All structural components in a MPC shall be made of Alloy X, a designation which warrants further explanation.

Alloy X is a material that is expected to be acceptable as a Mined Geological Disposal System (MGDS) waste package and which meets the thermophysical properties set forth in this document.

At this time, there is considerable uncertainty with respect to the material of construction for an MPC that would be acceptable as a waste package for the MGDS. Candidate materials being considered for acceptability by the DOE include:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The DOE material selection process is primarily driven by corrosion resistance in the potential environment of the MGDS. As the decision regarding a suitable material to meet disposal requirements is not imminent, the MPC design allows the use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this FSAR) may be one of the following materials. Only a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC basket or shell - the basket and shell may be of different alloys in the same MPC.

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable for the entire group for the analysis in question. For example, when calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, we have defined a material, which is referred to as Alloy X, whose thermophysical properties, from the MPC design perspective, are the least favorable of the candidate materials.

The evaluation of the Alloy X constituents to determine the least favorable properties is provided in Appendix 1.A.

The Alloy X approach is conservative because no matter which material is ultimately utilized in the MPC construction, the Alloy X approach guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

1.2.1.2 Overpacks

1.2.1.2.1 HI-STORM 100 Overpack (Storage)

The HI-STORM 100 and 100S overpacks are rugged, heavy-walled cylindrical vessels. Figures 1.2.7, 1.2.8, and 1.2.8A provide cross sectional views of the HI-STORM 100 System, showing both of the overpack designs, respectively. The HI-STORM 100A is an anchored variant of the same structure and hereinafter is identified by name only when the discussion specifically applies to the anchored overpack. The HI-STORM 100A differs only in the diameter of the overpack baseplate and the presence of bolt holes and associated anchorage hardware (see Figures 1.1.4 and 1.1.5). The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The overpack plain concrete is enclosed by cylindrical steel shells, a thick steel baseplate, and a top plate. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation in the vertical direction.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC. The inner shell of the overpack has channels attached to its inner diameter. The channels provide guidance for MPC insertion and removal and a flexible medium to absorb impact loads during the non-mechanistic tip-over, while still allowing the cooling air flow to circulate through the overpack. Shims may be attached to channels to allow the proper inner diameter dimension to be obtained.

The storage system has air ducts to allow for passive natural convection cooling of the contained MPC. Four air inlets and four air outlets are located at the lower and upper extremities of the storage system, respectively. The location of the air outlets in the HI-STORM 100 and the HI-STORM 100S design differ in that the outlet ducts for the HI-STORM 100 overpack are located in the overpack body and are aligned vertically with the inlet ducts at the bottom of the overpack body. The air outlet ducts in the HI-STORM 100S are integral to the lid assembly and are not in vertical alignment with the inlet ducts. The location of the air inlet ducts is same for both the HI-STORM 100 and the HI-STORM 100S. The air inlets and outlets are covered by a fine mesh screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, described in Chapter 11 of this FSAR, evaluates the effects of partial and complete blockage of the air ducts.

The four air inlets and four air outlets are penetrations through the thick concrete shielding provided by the HI-STORM 100 overpack. The outlet air ducts for the HI-STORM 100S overpack, integral to the lid, present a similar break in radial shielding. Within the air inlets and outlets, an array of gamma shield cross plates are installed (see Figure 5.3.19 for a pictorial representation of the gamma shield cross plate designs). These gamma shield cross plates are designed to scatter any particles traveling through the ducts. The result of scattering the particles in the ducts is a significant decrease in the local dose rates around the four air inlets and four air outlets. The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized. The shielding analysis conservatively credits only the mandatory version of the gamma shield cross plate design because they provide less shielding than the optional design. Conversely, the thermal analysis conservatively evaluates the optional gamma shield cross plate design because it conservatively provides greater resistance to flow than the mandatory design.

Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM 100) or the inlet air duct horizontal plates (HI-STORM 100S) (see Figure 1.2.7). The four anchor blocks are located on 90° arcs around the circumference of the overpack. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, Hillman rollers, or other design based on site-specific needs and capabilities. Slings or other suitable devices mate with lifting lugs that are inserted into threaded holes in the top surface of the overpack lid to allow lifting of the overpack lid. After the lid is bolted to the storage overpack main body, these lifting bolts shall be removed and replaced with flush plugs.

The plain concrete between the overpack inner and outer steel shells is specified to provide the necessary shielding properties (dry density) and compressive strength. The concrete shall be in accordance with the requirements specified in Appendix 1.D.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, in an implicit manner it helps enhance the performance of the HI-STORM overpack in other respects as well. For example, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. The case of a postulated fire accident at the ISFSI is another example where the high thermal inertia characteristics of the HI-STORM concrete control the temperature of the MPC. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the inter-shell space, such that, while its cracking and crushing under a tip-over accident is not of significant consequence, its deformation characteristics are germane to the analysis of the structural members.

Density and compressive strength are the key parameters which delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal, and HI-STORM lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.0.5] are used.

To ensure the stability of the concrete at temperature, the concrete composition has been specified in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" [1.0.3]. Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM overpack concrete.

There are two base HI-STORM overpack designs - HI-STORM 100 and HI-STORM 100S. The significant differences between the two are overpack height, MPC pedestal height, location of the air outlet ducts, and the vertical alignment of the inlet and outlet air ducts. The HI-STORM 100 overpack is approximately 240 inches high from the bottom of the baseplate to the top of the lid bolts and 227 inches high without the lid installed. There are two versions of the HI-STORM 100S overpack design, differing only in height and weight. The HI-STORM 100S(232) is approximately 232 inches from the bottom of the baseplate to the top of the lid bolts in its final storage configuration and 211 inches high without the lid installed. The HI-STORM 100S(243) is approximately 243 inches from the bottom of the baseplate to the top of the lid bolts in its final storage configuration and 222 inches high without the lid installed.

The anchored embodiment of the HI-STORM overpack is referred to as HI-STORM 100A. As explained in the foregoing, the HI-STORM overpack is a steel weldment, which makes it a relatively simple matter to extend the overpack baseplate, form lugs, and then anchor the cask to the reinforced concrete structure of the ISFSI. In HI-STORM terminology, these lugs are referred to as "sector lugs." The sector lugs, as shown in Figure 1.1.5 and the drawing in Section 1.5, are formed by extending the HI-STORM overpack baseplate, welding vertical gussets to the baseplate extension and to the overpack outer shell and, finally, welding a horizontal lug support ring in the form of an annular sector to the vertical gussets and to the outer

shell. The baseplate is equipped with regularly spaced clearance holes (round or slotted) through which the anchor studs can pass. The sector lugs are bolted to the ISFSI pad using anchor studs that are made of a creep-resistant, high-ductility, environmentally compatible material. The bolts are pre-loaded to a precise axial stress using a "stud tensioner" rather than a torque wrench. Pre-tensioning the anchors using a stud tensioner eliminates any shear stress in the bolt, which is unavoidable if a torquing device is employed (Chapter 3 of the text "Mechanical Design of Heat Exchangers and Pressure Vessel Components", by Arcturus Publishers, 1984, K.P. Singh and A.I. Soler, provides additional information on stud tensioners). The axial stress in the anchors induced by pre-tensioning is kept below 75% of the material yield stress, such that during the seismic event the maximum bolt axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF (for Level D conditions). Figures 1.1.4 and 1.1.5 provide visual depictions of the anchored HI-STORM 100A configuration. This configuration also applies to the HI-STORM 100SA.

The anchor studs pass through liberal clearance holes (circular or slotted) in the sector lugs (0.75" minimum clearance) such that the fastening of the studs to the ISFSI pad can be carried out without mechanical interference from the body of the sector lug. The two clearance hole configurations give the ISFSI pad designer flexibility in the design of the anchor embedment in the ISFSI concrete. The axial force in the anchors produces a compressive load at the overpack/pad interface. This compressive force, F , imparts a lateral load bearing capacity to the cask/pad interface that is equal to μF ($\mu \leq 0.53$ per Table 2.2.8). As is shown in Chapter 3 of this FSAR, the lateral load-bearing capacity of the HI-STORM/pad interface (μF) is many times greater than the horizontal (sliding) force exerted on the cask under the postulated DBE seismic event. Thus, the potential for lateral sliding of the HI-STORM 100A System during a seismic event is precluded, as is the potential for any bending action on the anchor studs.

The seismic loads, however, will produce an overturning moment on the overpack that would cause a redistribution of the compressive contact pressure between the pad and the overpack. To determine the pulsation in the tensile load in the anchor studs and in the interface contact pressure, bounding static analysis of the preloaded configuration has been performed. The results of the static analysis demonstrate that the initial preloading minimizes pulsations in the stud load. A confirmatory non-linear dynamic analysis has also been performed using the time-history methodology described in Chapter 3, wherein the principal nonlinearities in the cask system are incorporated and addressed. The calculated results from the dynamic analysis confirm the static analysis results and that the presence of pre-stress helps minimize the pulsation in the anchor stud stress levels during the seismic event, thus eliminating any concern with regard to fatigue failure under extended and repetitive seismic excitations.

The sector lugs in HI-STORM 100A are made of the same steel material as the baseplate and the shell (SA516- Gr. 70) which helps ensure high quality fillet welds used to join the lugs to the body of the overpack. The material for the anchor studs can be selected from a family of allowable stud materials listed in the ASME Code (Section II). A representative sampling of permitted materials is listed in Table 1.2.7. The menu of materials will enable the ISFSI owner to select a fastener material that is resistant to corrosion in the local ISFSI environment. For example, for ISFSIs located in marine environments (e.g., coastal reactor sites), carbon steel studs would not be recommended without concomitant periodic inspection and

coating maintenance programs. Table 1.2.7 provides the chemical composition of several acceptable fastener materials to help the ISFSI owner select the most appropriate material for his site. The two mechanical properties, ultimate strength σ_u and yield strength σ_y , are also listed. For purposes of structural evaluations, the lower bound values of σ_u and σ_y from the menu of materials listed in Table 1.2.7 are used (see Table 3.4.10).

As shown in the drawing, the anchor studs are spaced sufficiently far apart such that a practical reinforced concrete pad with embedded receptacles can be designed to carry the axial pull from the anchor studs without overstressing the enveloping concrete monolith. The design specification and supporting analyses in this FSAR are focused on qualifying the overpack structures, including the sector lugs and the anchor studs. The design of the ISFSI pad, and its anchor receptacle will vary from site to site, depending on the geology and seismological characteristics of the sub-terrain underlying the ISFSI pad region. The data provided in this FSAR, however, provide the complete set of factored loads to which the ISFSI pad, its sub-grade, and the anchor receptacles must be designed within the purview of ACI-349-97 [1.0.4]. Detailed requirements on the ISFSI pads for anchored casks are provided in Section 2.0.4.

1.2.1.2.2 HI-TRAC (Transfer Cask) - Standard Design

Like the storage overpack, the HI-TRAC transfer cask is a rugged, heavy-walled cylindrical vessel. The main structural function of the transfer cask is provided by carbon steel, and the main neutron and gamma shielding functions are provided by water and lead, respectively. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. Figure 1.2.9 provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. See Section 1.2.1.2.3 for discussion of the optional HI-TRAC 125D design.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC. The top lid of the HI-TRAC 125 has additional neutron shielding to provide neutron attenuation in the vertical direction (from SNF in the MPC below). The MPC access hole through the HI-TRAC top lid is provided to allow the lowering/raising of the MPC between the HI-TRAC transfer cask, and the HI-STORM or HI-STAR overpacks. The standard design HI-TRAC (comprised of HI-TRAC 100 and HI-TRAC 125) is provided with two bottom lids, each used separately. The pool lid is bolted to the bottom flange of the HI-TRAC and is utilized during MPC fuel loading and sealing operations. In addition to providing shielding in the axial direction, the pool lid incorporates a seal that is designed to hold clean demineralized water in the HI-TRAC inner cavity, thereby preventing contamination of the exterior of the MPC by the contaminated fuel pool water. After the MPC has been drained, dried, and sealed, the pool lid is removed and the HI-TRAC transfer lid is attached (standard design only). The transfer lid incorporates two sliding doors that allow the opening of the HI-TRAC bottom for the MPC to be raised/lowered. Figure 1.2.10 provides a cross section of the HI-TRAC with the transfer lid installed.

In the standard design, trunnions are provided for lifting and rotating the transfer cask body between vertical and horizontal positions. The lifting trunnions are located just below the top flange and the pocket trunnions are located above the bottom flange. The two lifting trunnions are provided to lift and vertically handle the HI-TRAC, and the pocket trunnions provide a pivot point for the rotation of the HI-TRAC for downending or upending.

Two standard design HI-TRAC transfer casks of different weights are provided to house the MPCs. The 125 ton HI-TRAC weight does not exceed 125 tons during any loading or transfer operation. The 100 ton HI-TRAC weight does not exceed 100 tons during any loading or transfer operation. The internal cylindrical cavities of the two standard design HI-TRACs are identical. However, the external dimensions are different. The 100ton HI-TRAC has a reduced thickness of lead and water shielding and consequently, the external dimensions are different. The structural steel thickness is identical in the two HI-TRACs. This allows most structural analyses of the 125 ton HI-TRAC to bound the 100 ton HI-TRAC design. Additionally, as the two HI-TRACs are identical except for a reduced thickness of lead and water, the 125 ton HI-TRAC has a larger thermal resistance than the smaller and lighter 100 ton HI-TRAC. Therefore, for normal conditions the 125 ton HI-TRAC thermal analysis bounds that of the 100 ton HI-TRAC. Separate shielding analyses are performed for each HI-TRAC since the shielding thicknesses are different between the two.

1.2.1.2.3 HI-TRAC 125D Transfer Cask

As an option to using either of the standard HI-TRAC transfer cask design, users may choose to use the optional HI-TRAC 125D design. Figure 1.2.9A provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. Like the standard design, the HI-TRAC 125D is designed and constructed in accordance with ASME III, Subsection NF, with certain NRC-approved alternatives, as discussed in Section 2.2.4. Functionally equivalent, the major differences between the HI-TRAC 125D design and the standard design are as follows:

- No pocket trunnions are provided for downending/upending
- The transfer lid is not required
- A new ancillary, the HI-STORM mating device (Figure 1.2.18) is required during MPC transfer operations
- A wider baseplate with attachment points for the mating device is provided
- The baseplate incorporates gussets for added structural strength
- The number of pool lid bolts is reduce

The interface between the MPC and the transfer cask is the same between the standard design and the HI-TRAC 125D design. The optional design is capable of withstanding all loads defined in the design basis for the transfer cask during normal, off-normal, and accident modes of operation with adequate safety margins. In lieu of swapping the pool lid for the transfer lid to facilitate MPC transfer, the pool lid remains on the HI-TRAC 125D until MPC transfer is required. The HI-STORM mating device is located between, and secured with bolting to, the top of the HI-STORM overpack and the HI-TRAC 125D transfer cask. The mating device is used to remove the pool lid to provide a pathway for MPC transfer between the overpack

and the transfer cask. Section 1.2.2.2 provides additional detail on the differences between the standard transfer cask design and the HI-TRAC 125D design during operations.

1.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure the radiation and exposure requirements in 10CFR72.104 and 10CFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the SNF in the MPC for ALARA considerations during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and ~~Boral~~ neutron poison *absorber* panels attached to the fuel storage cell vertical surfaces provide the initial attenuation of gamma and neutron radiation emitted by the radioactive spent fuel. The MPC shell, baseplate, lid and closure ring provide additional thicknesses of steel to further reduce the gamma flux at the outer canister surfaces.

In the HI-STORM storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the lid, and a thick circular concrete pedestal upon which the MPC rests. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shielding integral to the lid provide additional gamma attenuation in the upward direction, reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness. Gamma shield cross plates, as depicted in Figure 5.3.19, provide attenuation of scattered gamma radiation as it exits the inlet and outlet air ducts.

In the HI-TRAC transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the axial direction, shielding is provided by the top lid, and the pool or transfer lid, as applicable. In the HI-TRAC pool lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC. In the transfer lid, layers of steel-lead-steel provide gamma attenuation. For the HI-TRAC 125 transfer lid, the neutron shield material, Holtite-A, is also provided. The HI-TRAC 125 and HI-TRAC 125D top lids are composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The HI-TRAC 100 top lid is composed of steel only providing gamma attenuation.

1.2.1.3.1 ~~Boral~~ Fixed Neutron Absorbers

(i) *Boral*TM

Boral is a thermal neutron poison material composed of boron carbide and aluminum (aluminum powder and plate). Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The Boral cladding is made of alloy aluminum, a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation,

thermal, and chemical environment of a nuclear reactor, spent fuel pool, or dry cask. See Section 3.4.1 for discussion of the reaction of Boral with spent fuel pool water during fuel loading and unloading operations.

The documented historical applications of Boral, in environments comparable to those in spent fuel pools and fuel storage casks, dates to the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor [1.2.2]). Technical data on the material was first printed in 1949, when the report "Boral: A New Thermal Neutron Shield" was published [1.2.3]. In 1956, the first edition of the Reactor Shielding Design Manual [1.2.4] was published and it contained a section on Boral and its properties.

In the research and test reactors built during the 1950s and 1960s, Boral was frequently the material of choice for control blades, thermal-column shutters, and other items requiring very good thermal-neutron absorption properties. It is in these reactors that Boral has seen its longest service in environments comparable to today's applications.

Boral found other uses in the 1960s, one of which was a neutron poison material in baskets used in the shipment of irradiated, enriched fuel rods from Canada's Chalk River laboratories to Savannah River. Use of Boral in shipping containers continues, with Boral serving as the poison in current British Nuclear Fuels Limited casks and the Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

Boral has been exclusively used in fuel storage applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and several unique characteristics, such as:

- The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.
- Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- Boral is stable, strong, durable, and corrosion resistant.

Boral absorbs thermal neutrons without physical change or degradation of any sort from the anticipated exposure to gamma radiation and heat. The material does not suffer loss of neutron attenuation capability when exposed to high levels of radiation dose.

Holtec International's QA Program ensures that Boral is manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR72, Subpart G. Holtec International has procured over 200,000 panels of Boral from AAR Advanced Structures in over 30 projects. Boral has always been purchased with a minimum ^{10}B loading requirement. Coupons extracted from production runs were tested using the wet chemistry procedure. The actual ^{10}B loading, out of thousands of coupons tested, has never been found to fall below the design specification. The size of this coupon database is sufficient to provide reasonable assurance that all future Boral procurements will continue to yield Boral with full compliance with the stipulated minimum loading. Furthermore, the surveillance, coupon testing, and material tracking processes which have so effectively controlled the quality of Boral are expected to continue to yield Boral of similar quality in the future. Nevertheless, to add another layer of insurance, only 75% ^{10}B credit of the fixed neutron absorber is assumed in the criticality analysis consistent with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

(ii) *METAMIC*[®]

METAMIC[®] is a neutron absorber material developed by the Reynolds Aluminum Company in the mid-1990s for spent fuel reactivity control in dry and wet storage applications. Metallurgically, *METAMIC*[®] is a metal matrix composite (MMC) consisting of a high purity 6061 Aluminum matrix reinforced with Type 1 ASTM C-750, isotopically graded boron carbide. *METAMIC*[®] is characterized by an extremely fine aluminum spherical powder (325 mesh or better) and boron carbide powder (average particle size under 10 microns). As described in the U.S. patents held by *METAMIC, Inc.*[†], the high performance reliability of *METAMIC*[®] derives from the particle size distribution of its constituents, namely, high purity Aluminum 6061 alloy powder and isotopically graded B_4C particulate, rendered into an isotropic metal matrix composite state by the powder metallurgy process which yields excellent homogeneity, and which prevents B_4C from clustering in the final product.

The powders are carefully blended together without binders, chelating agents, or other additives that could potentially become retained in the final product and deleteriously influence performance. The approximate maximum percentage of B_4C that can be dispersed in the Aluminum alloy 6061 matrix is 40%. The pure blend of powders is cold isostatically compacted into a green billet and vacuum sintered to high theoretical density. According to the manufacturer, billets of any size can be produced using this technology; however, a typical production billet is 8 to 9 inches in diameter by approximately 32 inches long, weighing approximately 210 pounds. This billet is subsequently

[†] U.S. Patent No. 6,042,779, "Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super, Hypereutectic Al/Si."

extruded into one of a number of product forms, ranging from sheet and plate to angle, channel, round and square tube, and other profiles. A typical lot of METAMIC[®] (defined as the quantity processed in one load of the vacuum furnace) is in the range of 1,200 to 2,400 lbs.

METAMIC[®] has been subjected to an extensive array of tests sponsored by the Electric Power Research Institute (EPRI) that evaluated the functional performance of the material at elevated temperatures (up to 900°F) and radiation levels (1E+11 rads gamma). The results of the tests documented in an EPRI report* indicate that METAMIC[®] maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state. The main conclusions provided in the above-referenced EPRI report are summarized below:

- The isotropic metal matrix configuration produced by the powder metallurgy process with a complete absence of interconnected internal porosity in METAMIC[®] ensures that its density is essentially equal to the maximum theoretical density.
- Measurements of boron carbide particle distribution show extremely small particle-to-particle distance[†] and near-perfect homogeneity.
- The physical and neutronic properties of METAMIC[®] are essentially unaltered under exposure to elevated temperatures (750° F - 900° F).
- No detectable change in the neutron attenuation characteristics under accelerated test conditions has been observed.

An evaluation of the manufacturing technology underlying METAMIC[®] as disclosed in the above-referenced patents and of the extensive third-party tests carried out under the auspices of EPRI had led Holtec International to designate METAMIC[®] an acceptable neutron absorber material for use in the company's MPCs. Holtec's technical position on METAMIC[®] is also supported by the evaluation carried out by other organizations (see, for example, USNRC's SER on NUHOMS-61BT, Docket No. 72-1004).

Consistent with its role in reactivity control, all METAMIC[®] material procured for use in the Holtec MPCs will be qualified as important-to-safety (ITS) Category A item. ITS category A manufactured items, as required by Holtec's NRC-approved Quality Assurance program, must be produced to essentially preclude the potential of an error in the procurement of constituent materials and the manufacturing processes. Accordingly, material and manufacturing control processes must be

* "Qualification of METAMIC[®] for Spent Fuel Storage Application", EPRI, 1003137, Final Report, October 2001.

† Medium measured neighbor-to-neighbor distance is 10.08 microns according to the article, "METAMIC Neutron Shielding", by K. Anderson, T. Haynes, and R. Kazmier, EPRI Boraflex Conference, November 19-20, 1998.

established to eliminate the incidence of errors, and inspection steps must be implemented to serve as an independent set of barriers to ensure that all critical characteristics defined for the material by the cask designer are met in the manufactured product.

All manufacturing and in-process steps in the production of METAMIC[®] shall be carried out using written procedures that have been reviewed and found to be acceptable by Holtec's QA organization. As required by the company's quality program, the material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances as set forth in the applicable Holtec QA procedures to ensure that all METAMIC panels procured meet with the requirements appropriate for the quality genre of the MPCs. Confirmatory tests, prior to the use of the METAMIC[®] in Holtec's MPCs, are summarized in Subsection 9.1.5.3 of this FSAR.

Because of the absence of interconnected porosities, the time required to dehydrate a METAMIC[®]-equipped MPC is expected to be less compared to an MPC containing the rolled cermet class of neutron absorbers such as Boral.

Although METAMIC[®] exhibits near-theoretical neutron attenuation characteristics (due to the very small B₄C particle size distributed in the homogeneous metal matrix), only 75% of the minimum B-10 areal density (as in the case of Boral) is recognized in the criticality analysis.

(iii) *Structural Integrity of Fixed Neutron Absorbers*

Both Boral and METAMIC[®] neutron absorber panels are completely enclosed in Alloy X (stainless steel) sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld (see the drawings in Section 1.5 for details of this design configuration). Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient evolutions. Finally, the pocket joint detail, borrowed from Holtec's spent fuel rack design (with tens of thousands of successful deployments) ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.

1.2.1.3.2 Neutron Shielding

The specification of the HI-STORM overpack and HI-TRAC transfer cask neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation to appropriate levels;
- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;
- Stability of the homogeneous nature of the shielding material matrix;
- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and
- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

Other aspects of a shielding material, such as ease of handling and prior nuclear industry use, are also considered, within the limitations of the main criteria. Final specification of a shield material is a result of optimizing the material properties with respect to the main criteria, along with the design of the shield system, to achieve the desired shielding results.

Neutron attenuation in the HI-STORM overpack is provided by the thick walls of concrete contained in the steel vessel, lid, and pedestal. Concrete is a shielding material with a long proven history in the nuclear industry. The concrete composition has been specified to ensure its continued integrity at the long term temperatures required for SNF storage.

The HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) will be added to reduce the freezing point for low temperature operations (e.g., below 32 F) [1.2.7].

Neutron shielding in the HI-TRAC 125 and 125D transfer casks in the axial direction is provided by Holtite-A within the top lid. HI-TRAC 125 also contains Holtite-A in the transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal B₄C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

In the following, a brief summary of the performance characteristics and properties of Holtite-A is provided.

Density

The specific gravity of Holtite-A is 1.68 g/cm³ as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm³. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm³ to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The weight concentration of hydrogen is 6.0%. However, all shielding analyses conservatively assume 5.9% hydrogen by weight in the calculations.

Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (nominal) weight concentration. Holtite-A may be specified with a B₄C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal B₄C weight percent of 1%.

Design Temperature

The design temperature of Holtite-A is set at 300 F. The maximum spatial temperature of Holtite -A under all normal operating conditions must be demonstrated to be below this design temperature.

Thermal Conductivity

The Holtite-A neutron shielding material is stable below the design temperature for the long term and provides excellent shielding properties for neutrons. A conservative, lower bound conductivity is stipulated for use in the thermal analyses of Chapter 4 (Section 4.2) based on information in the technical literature.

1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7)[†]. Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the baseplate and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are

[†] The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated. Those overpacks without the shield shell are required to have a higher concrete density in the overpack body to provide compensatory shielding. See Table 1 D.1.

installed in the ducts (Figures 1.2.8 and 1.2.8A) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets.

In the HI-TRAC transfer cask, the primary gamma shielding is provided by lead. As in the storage overpack, carbon steel supplements the lead gamma shielding of the HI-TRAC transfer cask.

1.2.1.4 Lifting Devices

Lifting of the HI-STORM 100 System may be accomplished either by attachment at the top of the storage overpack ("top lift"), as would typically be done with a crane, or by attachment at the bottom ("bottom lift"), as would be effected by a number of lifting/handling devices.

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The anchor blocks are integrally welded to the overpack radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM100) or inlet air duct horizontal plates (HI-STORM 100S). The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment.

A bottom lift of the HI-STORM 100 storage overpack is effected by the insertion of four hydraulic jacks underneath the inlet vent horizontal plates (Figure 1.2.1) A slot in the overpack baseplate allows the hydraulic jacks to be placed underneath the inlet vent horizontal plate. The hydraulic jacks lift the loaded overpack to provide clearance for inserting or removing a device for transportation.

The standard design HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket trunnions. The HI-TRAC 125D is equipped with only lifting trunnions. The lifting trunnions are positioned just below the top forging. The two pocket trunnions are located above the bottom forging and attached to the outer shell. The pocket trunnions are designed to allow rotation of the HI-TRAC. All trunnions are built from a high strength alloy with proven corrosion and non-galling characteristics. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The lifting trunnions are installed by threading into tapped holes just below the top forging.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered through the HI-TRAC transfer cask using lifting cleats. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

1.2.1.5 Design Life

The design life of the HI-STORM 100 System is 40 years. This is accomplished by using material of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 9, is also implemented to ensure the HI-STORM 100 System will exceed its design life of 40 years. The

design considerations that assure the HI-STORM 100 System performs as designed throughout the service life include the following:

HI-STORM Overpack and HI-TRAC Transfer Cask

- Exposure to Environmental Effects
- Material Degradation
- Maintenance and Inspection Provisions

MPC

- Corrosion
- Structural Fatigue Effects
- Maintenance of Helium Atmosphere
- Allowable Fuel Cladding Temperatures
- Neutron Absorber Boron Depletion

The adequacy of the HI-STORM 100 System for its design life is discussed in Sections 3.4.11 and 3.4.12.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The HI-STORM 100 System incorporates some unique design improvements. These design innovations have been developed to facilitate the safe long term storage of SNF. Some of the design originality is discussed in Subsection 1.2.1 and below.

The free volume of the MPCs is inerted with 99.995% pure helium gas during the spent nuclear fuel loading operations. Table 1.2.2 specifies the helium fill requirements for the MPC internal cavity.

The HI-STORM overpack has been designed to synergistically combine the benefits of steel and concrete. The steel-concrete-steel construction of the HI-STORM overpack provides ease of fabrication, increased strength, and an optimal radiation shielding arrangement. The concrete is primarily provided for radiation shielding and the steel is primarily provided for structural functions.

The strength of concrete in tension and shear is conservatively neglected. Only the compressive strength of the concrete is accounted for in the analyses.

The criticality control features of the HI-STORM 100 are designed to maintain the neutron multiplication factor k -effective (including uncertainties and calculational bias) at less than 0.95 under all normal, off-normal, and accident conditions of storage as analyzed in Chapter 6. This level of conservatism and safety margins is maintained, while providing the highest storage capacity.

1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STORM 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the Design Drawings in Section 1.5. A summary of the general actions needed for the loading and unloading operations is provided below. Figures 1.2.16 and 1.2.17 provide a pictorial view of typical loading and unloading operations, respectively.

Loading Operations

At the start of loading operations, the HI-TRAC transfer cask is configured with the pool lid installed. The HI-TRAC water jacket is filled with demineralized water or a 25% ethylene glycol solution depending on the ambient temperature conditions. The lift yoke is used to position HI-TRAC in the designated preparation area or setdown area for HI-TRAC inspection and MPC insertion. The annulus is filled with plant demineralized water (borated if necessary), and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with water. Based on the MPC model and fuel enrichment (as required by the CoC), this may be borated water or plant demineralized water. HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielding lid (the MPC lid) is installed. The lift yoke is remotely engaged to the HI-TRAC lifting trunnions and is used to lift the HI-TRAC close to the spent fuel pool surface. As an ALARA measure, dose rates are measured on the top of the HI-TRAC and MPC prior to removal from the pool to check for activated debris on the top surface. The MPC lift bolts (securing the MPC lid to the lift yoke) are removed. As HI-TRAC is removed from the spent fuel pool, the lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination.

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. ~~Dose rates are measured at the MPC lid and around the mid-height circumference of HI-TRAC to ensure that the dose rates are within expected values.~~ The Automated Welding System baseplate shield (if used) is installed to reduce dose rates around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS) or other approved welding process. Liquid penetrant examinations are performed on the root and final passes. A multi-layer liquid penetrant or volumetric examination is also performed on the MPC lid-to-shell weld. The water level is raised to the top of the MPC and the weld is hydrostatically tested. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid-to-shell) to verify weld integrity and to ensure that leakage rates are within acceptance criteria. The MPC water is

displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line.

For ~~storage of moderate burnup fuel~~ *lower heat load MPCs*, a Vacuum Drying System (VDS) may be used to remove moisture from the MPC cavity. The VDS is connected to the MPC and is used to remove liquid water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated. This process is continued until the pressure in the MPC meets the technical specification limit and can be held there for the required amount of time.

For ~~storage of high burnup fuel~~ *higher heat load MPCs* and as an option for ~~storage of moderate burnup fuel~~ *lower heat load MPCs*, the reduction of residual moisture in the MPC to trace amounts is accomplished using a Forced Helium Dehydration (FHD) system, as described in Appendix 2.B. Relatively warm and dry helium is recirculated through the MPC cavity, which helps maintain the SNF in a cooled condition while moisture is being removed. The warm, dry gas is supplied to the MPC drain port and circulated through the MPC cavity where it absorbs moisture. The humidified gas travels out of the MPC and through appropriate equipment to cool and remove the absorbed water from the gas. The dry gas may be heated prior to its return to the MPC in a closed loop system to accelerate the rate of moisture removal in the MPC. This process is continued until the temperature of the gas exiting the demohumidizing module described in Appendix 2.B meets the limit specified in the technical specifications. *FSAR Section 4.5 and the technical specifications provide the specific limits applicable to the two types of moisture removal.*

Following moisture removal, the VDS or FHD system is disconnected and the Helium Backfill System (HBS) is attached and the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides the means of future leakage rate testing of the MPC confinement boundary welds. Cover plates are installed and seal-welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes. The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.

The MPC closure ring is then placed on the MPC, aligned, tacked in place, and seal welded, providing redundant closure of the MPC lid and cover plates confinement closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in the annulus is drained. The AWS Baseplate shield is removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and HI-TRAC dose rates are measured. The HI-TRAC top lid is installed and the bolts are torqued. The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point of the MPC.

Rigging is installed between the MPC lift cleats and the lift yoke. The rigging supports the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. For the standard design transfer cask, the HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and rigging support the MPC during the transfer operations.

MPC transfer from the HI-TRAC transfer cask into the overpack may be performed inside or outside the fuel building. Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways. The loaded HI-TRAC may be handled in the vertical or horizontal orientation. The loaded HI-STORM can only be handled vertically.

For MPC transfers inside the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100 overpack, the vent duct shield inserts installed. If using HI-TRAC 125D, the HI-STORM mating device is secured to the top of the empty overpack (Figure 1.2.18). The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM, or the mating device, as applicable. After the HI-TRAC is positioned atop the HI-STORM or secured to the mating device, as applicable, the MPC is raised slightly. With the standard HI-TRAC design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and the HI-TRAC is prepared for removal from on top of HI-STORM (with HI-TRAC 125D, the transfer cask must first be disconnected from the mating device). For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and/or mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs are installed and torqued.

For MPC transfers outside of the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts installed. For HI-TRAC 125D, the mating device is secured to the top of the overpack. The loaded HI-TRAC is transported to the cask transfer facility in the vertical or horizontal orientation. A number of methods may be utilized as long as the handling limitations prescribed in the technical specifications are not exceeded.

To place the loaded HI-TRAC in a horizontal orientation, a transport frame or "cradle" is utilized. If the cradle is equipped with rotation trunnions they are used to engage the HI-TRAC 100 or 125 pocket trunnions. While the loaded HI-TRAC is lifted by the lifting trunnions, the HI-TRAC is lowered onto the cradle rotation trunnions. Then, the crane lowers and the HI-TRAC pivots around the pocket trunnions and is placed in the horizontal position in the cradle.

The HI-TRAC 125D does not include pocket trunnions in its design. Therefore, the user must downend the transfer cask onto the transport frame using appropriately designed rigging in accordance with the site's heavy load control program.

If the loaded HI-TRAC is transferred to the cask transfer facility in the horizontal orientation, the HI-TRAC transport frame and/or cradle are placed on a transport vehicle. The transport vehicle may be an air pad, railcar, heavy-haul trailer, dolly, etc. If the loaded HI-TRAC is transferred to the cask transfer facility in the vertical orientation, the HI-TRAC may be lifted by the lifting trunnions or seated on the transport vehicle. During the transport of the loaded HI-TRAC, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms.

After the loaded HI-TRAC arrives at the cask transfer facility, the HI-TRAC is upended by a crane if the HI-TRAC is in a horizontal orientation. The loaded HI-TRAC is then placed, using the crane located in the transfer area, on top of HI-STORM, which has been inspected and staged with the lid removed, vent duct shield inserts installed, the alignment device positioned, and the mating device installed, as applicable.

After the HI-TRAC is positioned atop the HI-STORM or the mating device, the MPC is raised slightly. In the standard design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and HI-TRAC is removed from on top of HI-STORM or disconnected from the mating device, as applicable. For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs and nuts are installed.

After the HI-STORM has been loaded either within the fuel building or at a dedicated cask transfer facility, the HI-STORM is then moved to its designated position on the ISFSI pad. The HI-STORM overpack may be moved using a number of methods as long as the handling limitations listed in the technical specifications are not exceeded. The loaded HI-STORM must be handled in the vertical orientation, and may be lifted from the top by the anchor blocks or from the bottom by the inlet vents. After the loaded HI-STORM is lifted, it may be placed on a transport mechanism or continue to be lifted by the lid studs and transported to the storage location. The transport mechanism may be an air pad, crawler, railcar, heavy-haul trailer, dolly, etc. During the transport of the loaded HI-STORM, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms. Once in position at the storage pad, vent operability testing is performed to ensure that the system is functioning within its design parameters.

In the case of HI-STORM 100A, the anchor studs are installed and fastened into the anchor receptacles in the ISFSI pad in accordance with the design requirements.

Unloading Operations

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover HI-TRAC and empty the MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC overpressurization and thermal shock to the stored spent fuel assemblies.

The MPC is recovered from HI-STORM either at the cask transfer facility or the fuel building using any of the methodologies described in Section 8.1. The HI-STORM lid is removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts are installed, and the MPC lift cleats are attached to the MPC. For HI-TRAC 125D, the mating device is installed. Rigging is attached to the MPC lift cleats. For the HI-STORM 100S and the standard HI-TRAC design, the transfer doors may need to be opened to avoid interfering with the MPC lift cleats. For HI-TRAC 125D, the mating device (possibly containing the pool lid) is secured to the top of the overpack. HI-TRAC is raised and positioned on top of HI-STORM or secured to the mating device, as applicable. For HI-TRAC 125D, the pool lid is ensured to be out of the transfer path for the MPC. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the standard design HI-TRAC transfer lid doors are closed and the locking pins are installed. For HI-TRAC 125D, the pool lid is installed and the transfer cask is unsecured from the mating device. HI-TRAC is removed from on top of HI-STORM.

The HI-TRAC is brought into the fuel building and, for the standard design, manipulated for bottom lid replacement. The transfer lid is replaced with the pool lid. The MPC lift cleats and rigging support the MPC during lid transfer operations.

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the rigging, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water (borated, if necessary). The annulus and HI-TRAC top surfaces are protected from debris that will be produced when removing the MPC lid.

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established around the MPC ports. The RVOAs are attached to the vent and drain port. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is cooled using ~~a closed-loop heat exchanger~~ *to appropriate means, if necessary, to reduce the MPC internal temperature to allow water flooding.* Following the fuel cool-down, the MPC is flooded with borated or unborated water in accordance with the CoC. The MPC lid-to-MPC shell weld is removed. Then, all weld removal equipment is removed with the MPC lid left in place.

The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris. HI-TRAC and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and HI-TRAC are decontaminated in preparation for re-utilization.

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorbing materials in the fuel basket. The MPC-24, MPC-24E, and 24EF (all with lower enriched fuel) and the MPC-68 do not rely on soluble boron credit during loading or the assurance that water cannot enter the MPC during storage to meet the stipulated criticality limits.

Each MPC model is equipped with ~~Boral~~-fixed neutron absorber plates affixed to the fuel cell walls as shown on the design drawings. The minimum ^{10}B areal density specified for the ~~Boral~~ neutron absorber in each MPC model is shown in Table 1.2.2. These values are chosen to be consistent with the assumptions made in the criticality analyses.

The MPC-24, MPC-24E and 24EF(all with higher enriched fuel) and the MPC-32 and MPC-32F take credit for soluble boron in the MPC water for criticality prevention during wet loading and unloading operations. Boron credit is only necessary for these PWR MPCs during loading and unloading operations that take place under water. During storage, with the MPC cavity dry and sealed from the environment, criticality control measures beyond the fixed neutron poisons affixed to the storage cell walls are not necessary because of the low reactivity of the fuel in the dry, helium filled canister and the design features that prevent water from intruding into the canister during storage.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM 100 dry storage system. A detailed evaluation is provided in Section 3.4.

1.2.2.3.3 Operation Shutdown Modes

The HI-STORM 100 System is totally passive and consequently, operation shutdown modes are unnecessary. Guidance is provided in Chapter 8, which outlines the HI-STORM 100 unloading procedures, and Chapter 11, which outlines the corrective course of action in the wake of postulated accidents.

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is seal welded and leak tested. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM overpack exit vents in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 and the Technical Specifications in Appendix A to the CoC for additional details.

1.2.2.3.5 Maintenance Technique

Because of their passive nature, the HI-STORM 100 System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 9 describes the acceptance criteria and maintenance program set forth for the HI-STORM 100.

1.2.3 Cask Contents

The HI-STORM 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key design parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1 and the Approved Contents section of Appendix B to the CoC. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the CoC. A summary of the types of fuel authorized for storage in each MPC model is provided below. All fuel assemblies must meet the fuel specifications provided in Appendix B to the CoC. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers.

MPC-24

The MPC-24 is designed to accommodate up to twenty-four (24) PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

MPC-24E

The MPC-24E is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

MPC-24EF

The MPC-24EF is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

MPC-32

The MPC-32 is designed to accommodate up to thirty-two (32) PWR fuel assemblies ~~classified as intact fuel assemblies~~, with or without non-fuel hardware. *Up to eight (8) of these assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).*

MPC-32F

The MPC-32F is designed to store up to thirty two (32) PWR fuel assemblies with or without non-fuel hardware. Up to eight (8) of these assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).

MPC-68

The MPC-68 is designed to accommodate up to sixty-eight (68) BWR intact and/or damaged fuel assemblies, with or without channels. For the Dresden Unit 1 or Humboldt Bay plants, the number of damaged fuel assemblies may be up to a total of 68. For damaged fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the number of damaged fuel assemblies is limited to sixteen (16) and must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2).

MPC-68F

The MPC-68F is designed to accommodate up to sixty-eight (68) Dresden Unit 1 or Humboldt Bay BWR fuel assemblies (with or without channels) made up of any combination of fuel assemblies classified as intact fuel assemblies, damaged fuel assemblies, and up to four (4) fuel assemblies classified as fuel debris.

MPC-68FF

The MPC-68FF is designed to accommodate up to sixty-eight (68) BWR fuel assemblies with or without channels. Any number of these fuel assemblies may be Dresden Unit 1 or Humboldt Bay BWR fuel assemblies classified as intact fuel or damaged fuel. Dresden Unit 1 and Humboldt Bay fuel debris is limited

to eight (8) DFCs. DFCs containing Dresden Unit 1 or Humboldt Bay fuel debris may be stored in any fuel storage location. For BWR fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the total number of fuel assemblies classified as damaged fuel assemblies or fuel debris is limited to sixteen (16), with up to eight (8) of the 16 fuel assemblies classified as fuel debris. These fuel assemblies must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2). The balance of the fuel storage locations may be filled with intact BWR fuel assemblies, up to a total of 68.

Table 1.2.1

KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	7 8	4 5 for PWR 3 for BWR
MPC storage capacity [†] :	MPC-24 MPC-24E MPC-24EF	Up to 24 intact Zircaloy or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to four damaged fuel assemblies may be stored in the MPC-24E and up to four (4) damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24EF OR
	MPC-32 MPC-32F	Up to 32 intact Zircaloy or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to 8 damaged fuel assemblies may be stored in the MPC-32 and up to 8 damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-32F.
	MPC-68	Any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68. For damaged fuel other than Dresden Unit 1 and Humboldt Bay, the number of fuel assemblies is limited to 16, with the balance being intact fuel assemblies. OR

[†] See Section 1.2.3 and Appendix B to the CoC for a complete description of cask contents and fuel specifications, respectively.

Table 1 2 2

KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725 [†] /-40 ^{††}	725 [†] /-40 ^{††}
Design internal pressure (psig)		
Normal conditions	100	100
Off-normal conditions	100	100
Accident Conditions	200	200
Total heat load, max. (kW)	27.77 (MPC-24) 28.17 (MPC-24E & MPC-24EF) 28.74 (MPC-32) 40 kW	28.19 (MPC-68, MPC-68F, & MPC-68FF) 40 kW
Maximum permissible peak fuel cladding temperature:		
Normal (—°F)	See Table 2.2.3	See Table 2.2.3
Short Term & Accident (—°F)	1058°	1058°
MPC internal environment	29.3—33.3 psig	29.3—33.3 psig
Helium fill	OR 0.1212 gm moles/l of free space <i>Varies with heat load. See CoC Appendix A, Table 3-1</i>	OR 0.1218 gm moles/l of free space <i>Varies with heat load. See CoC Appendix A, Table 3-1</i>
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
Boral-Fixed Neutron Absorber ¹⁰ B Areal Density (g/cm ²)	0.0267 (MPC-24) 0.0372 (MPC-24E, MPC-24EF & MPC-32, & MPC-32F)	0.0372 (MPC-68 & MPC-68FF) 0.01 (MPC-68F)
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

† Maximum normal condition design temperatures for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.2.3.

†† Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2.2 and no fuel decay heat load.

Table 1.2.3

INTENTIONALLY DELETED

Table 1.2.4

INTENTIONALLY DELETED

Table 1.2.5

INTENTIONALLY DELETED

Table 1.2.6

HI-STORM 100 OPERATIONS SEQUENCE

Site-specific handling and operations procedures will be prepared, reviewed, and approved by each owner/user.	
1	HI-TRAC and MPC lowered into the fuel pool without lids
2	Fuel assemblies transferred into the MPC fuel basket
3	MPC lid lowered onto the MPC
4	HI-TRAC/MPC assembly moved to the decon pit and MPC lid welded in place, volumetrically or multi-layer PT examined, hydrostatically tested, and leak tested
5	MPC dewatered, moisture removed, backfilled with helium, and the closure ring welded
6	HI-TRAC annulus drained and external surfaces decontaminated
7	MPC lifting cleats installed and MPC weight supported by rigging
8	HI-TRAC pool lid removed and transfer lid attached (not applicable to HI-TRAC 125D)
9	MPC lowered and seated on HI-TRAC transfer lid (not applicable to HI-TRAC 125D)
9a	HI-STORM mating device secured to top of empty HI-STORM overpack (HI-TRAC 125D only)
10	HI-TRAC/MPC assembly transferred to atop HI-STORM overpack or mating device, as applicable
11	MPC weight supported by rigging and transfer lid doors opened (standard design HI-TRAC) or pool lid removed (HI-TRAC 125D)
12	MPC lowered into HI-STORM overpack, and HI-TRAC removed from atop HI-STORM overpack/mating device
12a	HI-STORM mating device removed (HI-TRAC 125D only)
13	HI-STORM overpack lid installed and bolted in place
14	HI-STORM overpack placed in storage at the ISFSI pad
15	For HI-STORM 100A (or 100SA) users, the overpack is anchored to the ISFSI pad by installation of nuts onto studs and torquing to the minimum required torque.

Table 1.2.7

REPRESENTATIVE ASME BOLTING AND THREADED ROD MATERIALS ACCEPTABLE
FOR THE HI-STORM 100A ANCHORAGE SYSTEM

ASME MATERIALS FOR BOLTING

Composition	I.D.	Type Grade or UNC No	Ultimate Strength (ksi)	Yield Strength (ksi)	Code Permitted Size Range [†]
C	SA-354	BC K04100	125	109	t ≤ 2.5"
3/4Cr	SA-574	51B37M	170	135	t ≥ 5/8"
1 Cr - 1/5 Mo	SA-574	4142	170	135	t ≥ 5/8"
1 Cr-1/2 Mo-V	SA-540	B21 (K 14073)	165	150	t ≤ 4"
5 Cr - 1/2 Mo	SA-193	B7	125	105	t ≤ 2.5"
2Ni - 3/4Cr - 1/4 Mo	SA-540	B23 (H-43400)	135	120	
2Ni - 3/4Cr - 1/3 Mo	SA-540	B-24 (K-24064)	135	120	
17Cr-4Ni-4Cu	SA-564	630(H-1100)	140	115	
17Cr-4Ni-4Cu	SA-564	630(H-1075)	145	125	
25Ni-15Cr-2Ti	SA-638	660	130	85	
22CR-13Ni-5Mn	SA-479	XM-19(S20910)	135	105	

Note: The materials listed in this table are representative of acceptable materials and have been abstracted from the ASME Code, Section II, Part D, Table 3. Other materials listed in the Code are also acceptable as long as they meet the size requirements, the minimum requirements on yield and ultimate strength (see Table 2.0.4), and are suitable for the environment.

[†] Nominal diameter of the bolt (or rod) as listed in the Code tables. Two-inch diameter studs/rods are specified for the HI-STORM 100A

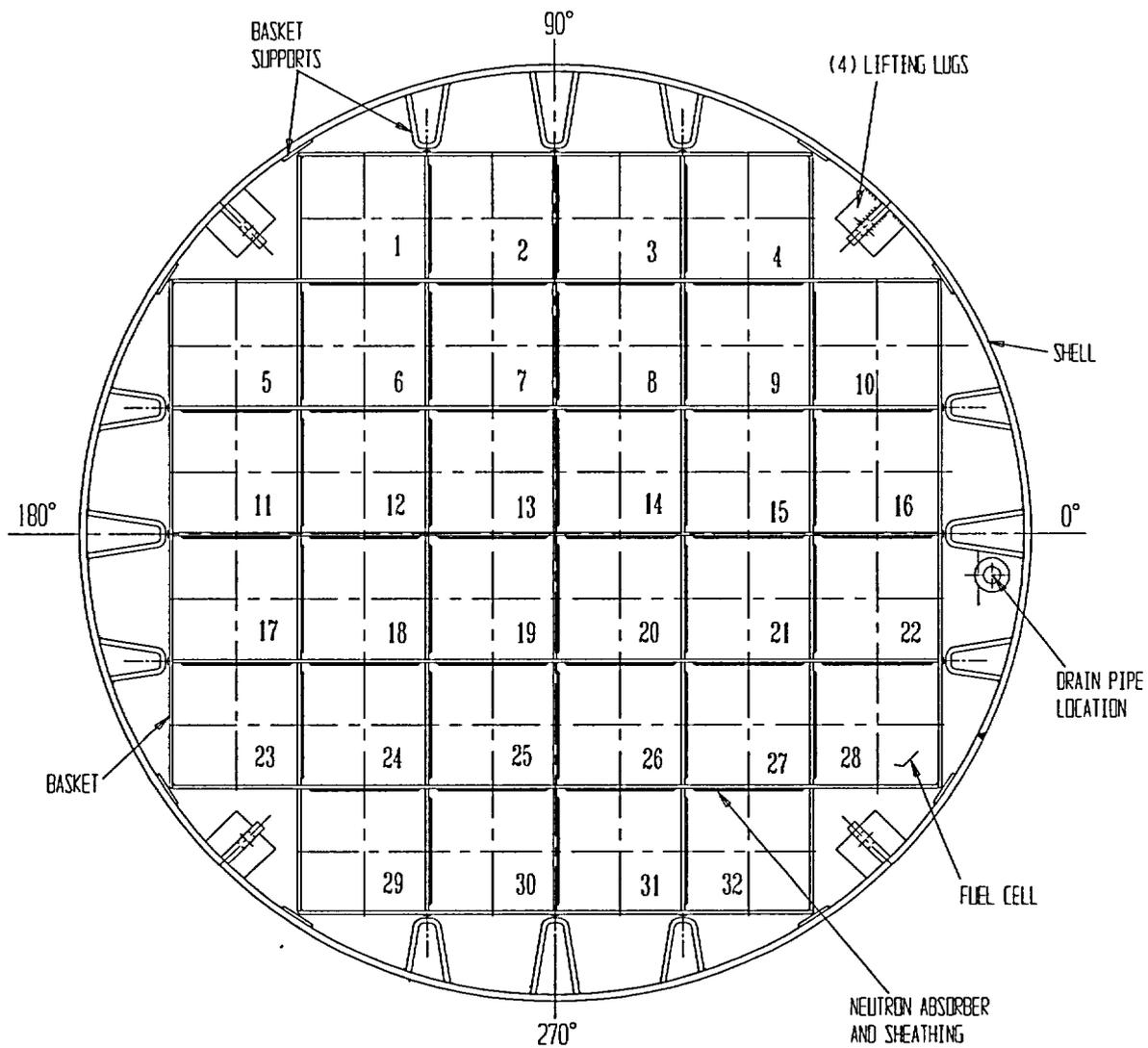


FIGURE 1.2.3; MPC-32/32F CROSS SECTION

CHAPTER 2[†]: PRINCIPAL DESIGN CRITERIA

This chapter contains a compilation of design criteria applicable to the HI-STORM 100 System. The loadings and conditions prescribed herein for the MPC, particularly those pertaining to mechanical accidents, are far more severe in most cases than those required for 10CFR72 compliance. The MPC is designed to be in compliance with both 10CFR72 and 10CFR71 and therefore certain design criteria are overly conservative for storage. This chapter sets forth the loading conditions and relevant acceptance criteria; it does not provide results of any analyses. The analyses and results carried out to demonstrate compliance with the design criteria are presented in the subsequent chapters of this report.

This chapter is in full compliance with NUREG-1536, except for the exceptions and clarifications provided in Table 1.0.3. Table 1.0.3 provides the NUREG-1536 review guidance, the justification for the exception or clarification, and the Holtec approach to meet the intent of the NUREG-1536 guidance.

2.0 PRINCIPAL DESIGN CRITERIA

The design criteria for the MPC, HI-STORM overpack, and HI-TRAC transfer cask are summarized in Tables 2.0.1, 2.0.2, and 2.0.3, respectively, and described in the sections that follow.

2.0.1 MPC Design Criteria

General

The MPC is designed for 40 years of service, while satisfying the requirements of 10CFR72. The adequacy of the MPC design for the design life is discussed in Section 3.4.12.

Structural

The MPC is classified as important to safety. The MPC structural components include the internal fuel basket and the enclosure vessel. The fuel basket is designed and fabricated as a core support structure, in accordance with the applicable requirements of Section III, Subsection NG of the ASME Code, with certain NRC-approved alternatives, as discussed in Section 2.2.4. The enclosure vessel is designed and fabricated as a Class 1 component pressure vessel in accordance with Section III, Subsection NB of the ASME Code, with certain NRC-approved alternatives, as discussed in Section 2.2.4. The principal exception is the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell,

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5)

as discussed in Section 2.2.4. In addition, the threaded holes in the MPC lid are designed in accordance with the requirements of ANSI N14.6 for critical lifts to facilitate vertical MPC transfer.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis, as presented in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination of the root pass and/or final weld surface (if more than one weld pass was required), in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid weld is further verified by performing a volumetric (or multi-layer liquid penetrant) examination, a hydrostatic pressure test and a helium leak test, in accordance with the drawings and the CoC.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, hydrostatic pressure testing, and helium leak testing performed during MPC fabrication and MPC closure, provides assurance of canister closure integrity in lieu of the specific weld joint requirements of Section III, Subsection NB.

Compliance with the ASME Code as it is applied to the design and fabrication of the MPC and the associated justification are discussed in Section 2.2.4. The MPC is designed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. These design loadings include postulated drop accidents while in the cavity of the HI-STORM overpack or the HI-TRAC transfer cask. The load combinations for which the MPC is designed are defined in Section 2.2.7. The maximum allowable weight and dimensions of a fuel assembly to be stored in the MPC are limited in accordance with Section 2.1.5.

The structural analysis to evaluate the margin against fuel rod damage from buckling under the drop accident scenario remains unchanged considering ISG-11, Revision 2 because no credit for the tensile stresses in the fuel rods due to internal pressure is taken. Because recognition of the state of tensile axial stress in the fuel cladding permitted by ISG-11 Revision 2, increases the resistance under axial buckling, neglecting the internal pressure buckling analysis is conservative. Therefore, compliance with ISG-11 Revision 2 does not have material effect on the structural analyses summarized in Chapter 3 of this FSAR.

Thermal

~~The allowable Zircaloy fuel cladding temperature limits to prevent cladding failure during long term dry storage conditions for moderate burnup fuel in the MPC are based on LLNL Report UCID 21181 [2.2.14]. To provide additional conservatism, the permissible fuel cladding temperature limits, which are lower than those calculated with the LLNL methodology, have been calculated based on PNL Report 6189 [2.0.3]. Stainless steel cladding is demonstrated to withstand higher temperatures than that of Zircaloy cladding in EPRI Report TR-106440 [2.2.13]. However, the Zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel fuel cladding. Allowable fuel cladding temperatures for high burnup fuel assemblies are determined using a creep strain model, developed by Holtec, and described in further detail in Appendix 4.A. The allowable fuel cladding temperatures which correspond to varying cooling times for the SNF to be stored in the MPCs are provided in Table 2.2.3.~~

The design and operation of the HI-STORM 100 System meets the intent of the review guidance contained in ISG-11, Revision 2 [2.0.8]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

- i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.*
- ii. The maximum value of the calculated temperature for all CSF (including ZR and stainless steel fuel cladding materials) under long-term normal conditions of storage and for short-term operations, including canister drying, helium backfill, and on-site cask transport operations must not exceed 400°C (752°F). For moderate burnup fuel, some flexibility to the system user is allowed provided certain additional criterion on the maximum cladding stress is satisfied, as discussed in Section 4.5.)*
- iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570°C (1058°F).*
- iv. The confinement analysis criterion that required the assumption of an additional percentage of high burnup fuel (HBF) rods to be leaking under normal and accident conditions of storage is eliminated.*
- v. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65°C (117°F).*

To achieve compliance with the above criteria, certain design and operational changes are necessary, as summarized below.

- i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term normal operations (e.g., MPC cavity drying) is set at 400°C (752°F), except that, with additional analysis, the short term PCT limit for moderate burnup fuel (MBF) may remain at the existing limit of 570°C (1058°F). Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions added to the CoC to ensure these limits are met.*
- ii. For MPCs containing at least one high burnup fuel (HBF) assembly and for relatively high heat load MPCs containing all MBF, the forced helium dehydration (FHD) method of MPC cavity drying must be used to meet the PCT limit and satisfy the 65°C temperature excursion criterion for HBF.*
- iii. The off-normal and accident condition PCT limit remains unchanged (1058°F).*
- iv. The confinement analysis is amended to accord with criterion (iv) above.*

- v. *Limits on the length of time a loaded MPC may reside in a HI-TRAC transfer cask have been established in the CoC to ensure the temperature limits are met. These limits are based on the heat load of the contained MPC and the orientation in which the HI-TRAC is handled (see Section 4.5).*

~~The short-term allowable fuel cladding temperature that is applicable to off-normal and accident conditions, as well as the fuel loading, canister closure, and canister transfer operations in the HI-TRAC transfer cask, is 570°C (1058°F) based on PNL 4835 [2.2.15].~~ *The MPC cavity is dried using either a vacuum drying system, or a forced helium dehydration system (see Appendix 2.B).* The MPC is backfilled with 99.995% pure helium in accordance with the technical specifications during canister sealing operations to promote heat transfer and prevent cladding degradation.

The design temperatures for the structural steel components of the MPC are based on the temperature limits provided in ASME Section II, Part D, tables referenced in ASME Section III, Subsection NB and NG, for those load conditions under which material properties are relied on for a structural load combination. The specific design temperatures for the components of the MPC are provided in Table 2.2.3.

The MPCs are designed for a bounding thermal source term, as described in Section 2.1.6. The maximum allowable fuel assembly heat load for each MPC is limited in accordance with the Approved Contents section of Appendix B to the CoC.

Each MPC model allows for two fuel loading strategies. The first is uniform fuel loading, wherein any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as ~~preferential fuel loading and location requirements for damaged fuel containers (DFCs) and fuel with integral non-fuel hardware (e.g., control rod assemblies).~~ The second is regionalized fuel loading, wherein the basket is segregated into two regions as defined in Appendix B to the CoC. Region 1 is the inner region where fuel assemblies with higher heat emission rates may be stored and Region 2 is the outer region where fuel assemblies with lower heat emission rates are stored. Regionalized loading allows for storage of fuel assemblies with higher heat emission rates (in Region 1) than would otherwise be authorized for loading under a uniform loading strategy. Regionalized loading strategies must also comply with other requirements of the CoC, such as those for DFCs and non-fuel hardware. Specific fuel assembly cooling time, burnup, and decay heat limits for regionalized loading are provided in the Approved Contents section of Appendix B to the CoC. The two fuel loading regions are defined by fuel storage location number in Table 2.1.13 (refer to Figures 1.2.2 through 1.2.4). Regionalized fuel loading meets the intent of preferential fuel loading.

Shielding

The allowable doses for an ISFSI using the HI-STORM 100 System are delineated in 10CFR72.104 and 72.106. Compliance with these regulations for any particular array of casks at an ISFSI is necessarily site-specific and is to be demonstrated by the licensee, as discussed in Chapters 5 and 12. Compliance with these regulations for a single cask and several representative cask arrays is demonstrated in Chapters 5 and 7.

The MPC provides axial shielding at the top and bottom ends to maintain occupational exposures ALARA during canister closure and handling operations. The occupational doses are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

The MPCs are designed for design basis fuel as described in Sections 2.1.7 and 5.2. The radiological source term for the MPCs is limited based on the burnup and cooling times specified in Appendix B to the CoC. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation in accordance with 10CFR20, as discussed in Chapter 10.

Criticality

The MPCs provide criticality control for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 6.1. The effective neutron multiplication factor is limited to $k_{eff} < 0.95$ for fresh unirradiated fuel with optimum water moderation and close reflection, including all biases, uncertainties, and MPC manufacturing tolerances.

Criticality control is maintained by the geometric spacing of the fuel assemblies, fixed borated neutron absorbing materials (Boral) incorporated into the fuel basket assembly, and, for certain MPC models, soluble boron in the MPC water. The minimum specified boron concentration verified during Boral neutron absorber manufacture is further reduced by 25% for criticality analysis. No credit is taken for burnup. The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC is limited in accordance with the Approved Contents section of Appendix B to the CoC. Soluble boron concentration requirements are delineated in the Technical Specifications in Appendix A to the CoC.

Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 7.1. A non-mechanistic breach-leak of the canister and subsequent release of available fission products in accordance with *the calculated leak rate and specified source term* release fractions is considered, as discussed in Section 7.3. The confinement function of the MPC is verified through hydrostatic testing, helium leak testing and weld examinations performed in accordance with the acceptance test program in Chapter 9.

Operations

There are no radioactive effluents that result from storage or transfer operations. Effluents generated during MPC loading are handled by the plant's radwaste system and procedures.

Generic operating procedures for the HI-STORM 100 System are provided in Chapter 8. Detailed operating procedures will be developed by the licensee based on Chapter 8, site-specific requirements that comply with the 10CFR50 Technical Specifications for the plant, and the HI-STORM 100 System CoC.

Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the MPCs are described in Chapter 9. The operational controls and limits to be applied to the MPCs are discussed in Chapter 12. Application of these requirements will assure that the MPC is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

Decommissioning

The MPCs are designed to be transportable in the HI-STAR overpack and are not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM 100 System is addressed in Section 2.4.

2.0.2 HI-STORM Overpack Design Criteria

General

The HI-STORM overpack is designed for 40 years of service, while satisfying the requirements of 10CFR72. The adequacy of the overpack design for the design life is discussed in Section 3.4.11.

Structural

The HI-STORM overpack includes both concrete and structural steel components that are classified as important to safety.

The concrete material is defined as important to safety because of its importance to the shielding analysis. The primary function of the HI-STORM overpack concrete is shielding of the gamma and neutron radiation emitted by the spent nuclear fuel.

Unlike other concrete storage casks, the HI-STORM overpack concrete is enclosed in steel inner and outer shells connected to each other by four radial ribs, and top and bottom plates. Where typical concrete storage casks are reinforced by rebar, the HI-STORM overpack is supported by the inner and outer shells connected by four ribs. As the HI-STORM overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete. Providing further conservatism, the structural analyses for normal conditions demonstrate that the allowable stress limits of the structural steel are met even with no credit for the strength of the concrete. During accident conditions (e.g., tornado missile, tip-over, end drop, and earthquake), only the compressive strength of the concrete is accounted for in the analysis to provide an appropriate simulation of the accident condition. Where applicable, the compressive strength of the concrete is calculated in accordance with ACI-318-95 [2.0.1].

In recognition of the conservative assessment of the HI-STORM overpack concrete strength and the primary function of the concrete being shielding, the applicable requirements of ACI-349 [2.0.2] are invoked in the design and construction of the HI-STORM overpack concrete as specified in Appendix 1.D.

Steel components of the storage overpack are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for Class 3 plate and shell components with certain NRC-approved alternatives.

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the overpack must protect the MPC from deformation, provide continued adequate performance, and allow the retrieval of the MPC under all conditions. These design loadings include a postulated drop accident from the maximum allowable handling height, consistent with the Cask Transport Evaluation program described in Technical Specification Section 5.0 contained in Appendix A to the CoC. The load combinations for which the overpack is designed are defined in Section 2.2.7. The physical characteristics of the MPCs for which the overpack is designed are defined in Chapter 1.

Thermal

The allowable long-term temperature limit for the overpack concrete is less than the limit in NUREG-1536, which limits the local concrete temperature to 300°F, if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range. Appendix 1.D specifies the cement and aggregate requirements to allow the utilization of the 300°F temperature limit of NUREG-1536; however, a conservative long-term temperature limit of 200 °F is applied to the concrete. For short term conditions the concrete temperature limit of 350°F is specified in accordance with Appendix A of ACI 349. The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in Table 2.2.3.

The overpack is designed for extreme cold conditions, as discussed in Section 2.2.2.2. The structural steel materials used for the storage cask that are susceptible to brittle fracture are discussed in Section 3.1.2.3.

The overpack is designed for the maximum allowable heat load for steady-state normal conditions, in accordance with Section 2.1.6. The thermal characteristics of the MPCs for which the overpack is designed are defined in Chapter 4.

Shielding

The off-site dose for normal operating conditions at the controlled area boundary is limited by 10CFR72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. Since these limits are dependent on plant operations as well as site-specific conditions (e.g., the ISFSI design and proximity to the controlled area boundary, and the number and arrangement of loaded storage casks on the ISFSI pad), the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a single cask and a range of typical ISFSIs using the HI-STORM 100 System are provided in Chapters 5 and 10. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10CFR72.212.

The overpack is designed to limit the calculated surface dose rates on the cask for all MPCs as defined in Section 2.3.5. The overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The calculated overpack dose rates are determined in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC transfer operations and a dose assessment for a typical ISFSI, as described in Chapter 10. ~~In addition, overpack dose rates are limited in accordance with the Technical Specifications provided in Appendix A to the CoC.~~

Confinement

The overpack does not perform any confinement function. Confinement during storage is provided by the MPC and is addressed in Chapter 7. The overpack provides physical protection and biological shielding for the MPC confinement boundary during MPC dry storage operations.

Operations

There are no radioactive effluents that result from MPC transfer or storage operations using the overpack. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures under the licensee's 10CFR50 license.

Generic operating procedures for the HI-STORM 100 System are provided in Chapter 8. The licensee is required to develop detailed operating procedures based on Chapter 8, site-specific conditions and requirements that also comply with the applicable 10CFR50 technical specification requirements for the site, and the HI-STORM 100 System CoC.

Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the overpack are described in Chapter 9. The operational controls and limits to be applied to the overpack are contained in Chapter 12. Application of these requirements will assure that the overpack is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

Decommissioning

Decommissioning considerations for the HI-STORM 100 System, including the overpack, are addressed in Section 2.4.

2.0.3 HI-TRAC Transfer Cask Design Criteria

General

The HI-TRAC transfer cask is designed for 40 years of service, while satisfying the requirements of 10CFR72. The adequacy of the HI-TRAC design for the design life is discussed in Section 3.4.11.

Structural

The HI-TRAC transfer cask includes both structural and non-structural biological shielding components that are classified as important to safety. The structural steel components of the HI-TRAC, with the exception of the lifting trunnions, are designed and fabricated in accordance with the applicable requirements of Section III, Subsection NF, of the ASME Code with certain NRC-approved alternatives, as discussed in Section 2.2.4. The lifting trunnions and associated attachments are designed in accordance with the requirements of NUREG-0612 and ANSI N14.6 for non-redundant lifting devices.

The HI-TRAC transfer cask is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the HI-TRAC transfer cask must protect the MPC from deformation, provide continued adequate performance, and allow the retrieval of the MPC under all conditions. These design loadings include a side drop from the maximum allowable handling height, consistent with the technical specifications. The load combinations for which the HI-TRAC is designed are defined in Section 2.2.7. The physical characteristics of each MPC for which the HI-TRAC is designed are defined in Chapter 1.

Thermal

The allowable temperatures for the HI-TRAC transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The top lid of the HI-TRAC 100 and HI-TRAC 125 incorporate Holtite-A shielding material. This material has a maximum allowable temperature in accordance with the manufacturer's test data. The specific allowable temperatures for the structural steel and shielding components of the HI-TRAC are provided in Table 2.2.3. The HI-TRAC is designed for off-normal environmental cold conditions, as discussed in Section 2.2.2.2. The structural steel materials susceptible to brittle fracture are discussed in Section 3.1.2.3.

The HI-TRAC is designed for the maximum allowable heat load *analyzed for storage operations*. ~~provided in the technical specifications~~ *Based on the heat load of the contained MPC and the orientation in which the transfer cask is handled, certain restrictions may apply to the length of time a loaded MPC may reside in the HI-TRAC transfer cask.* The HI-TRAC water jacket maximum allowable temperature is a function of the internal pressure. To preclude over pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is limited to less than the saturation temperature at the shell design pressure. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the minimum allowable temperature and adding

ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1.6. The working area ambient temperature limit for loading operations is delineated in the Design Features section of Appendix B to the CoC.

Shielding

The HI-TRAC transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10CFR20, while also maintaining the maximum load on the plant's crane hook to below either 125 tons or 100 tons, or less, depending on whether the 125-ton or 100-ton HI-TRAC transfer cask is utilized. The HI-TRAC calculated dose rates are reported in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations, as described in Chapter 10. A postulated HI-TRAC accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Section 5.1.2. In addition, HI-TRAC dose rates are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

The HI-TRAC 125 and 125D provide better shielding than the 100 ton HI-TRAC. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would normally dictate that the 125 ton HI-TRAC should be used. However, sites may not be capable of utilizing the 125 ton HI-TRAC due to crane capacity limitations, floor loading limits, or other site-specific considerations. As with other dose reduction-based plant activities, individual users who cannot accommodate the 125 ton HI-TRAC should perform a cost-benefit analysis of the actions (e.g., modifications) which would be necessary to use the 125 ton HI-TRAC. The cost of the action(s) would be weighed against the value of the projected reduction in radiation exposure and a decision made based on each plant's particular ALARA implementation philosophy.

The HI-TRAC provides a means to isolate the annular area between the MPC outer surface and the HI-TRAC inner surface to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC surfaces expected to require decontamination are coated. The maximum permissible surface contamination for the HI-TRAC is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

Confinement

The HI-TRAC transfer cask does not perform any confinement function. Confinement during MPC transfer operations is provided by the MPC, and is addressed in Chapter 7. The HI-TRAC provides physical protection and biological shielding for the MPC confinement boundary during MPC closure and transfer operations.

Operations

There are no radioactive effluents that result from MPC transfer operations using HI-TRAC. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures.

Generic operating procedures for the HI-STORM 100 System are provided in Chapter 8. The licensee will develop detailed operating procedures based on Chapter 8, plant-specific requirements including the Part 50 Technical Specifications, and the HI-STORM 100 System CoC.

Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the HI-TRAC Transfer Cask are described in Chapter 9. The operational controls and limits to be applied to the HI-TRAC are contained in Chapter 12. Application of these requirements will assure that the HI-TRAC is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

Decommissioning

Decommissioning considerations for the HI-STORM 100 Systems, including the HI-TRAC transfer cask, are addressed in Section 2.4.

2.0.4 Principal Design Criteria for the ISFSI Pad

2.0.4.1 Design and Construction Criteria

In compliance with 10CFR72, Subpart F, "General Design Criteria" the HI-STORM 100 cask system is classified as "important-to-safety" (ITS). This final safety analysis report (FSAR) explicitly recognizes the HI-STORM 100 System as an assemblage of equipment containing numerous ITS components. The reinforced concrete pad on which the cask is situated, however, is designated as a non-ITS structure. This is principally because, in most cases, cask systems for storing spent nuclear fuel on reinforced concrete pads are installed as free-standing structures. The lack of a physical connection between the cask and the pad permits the latter to be designated as not important-to-safety.

However, if the ZPAs at the surface of an ISFSI pad exceed the threshold limit for free-standing HI-STORM installation set forth in the CoC, then the cask must be installed in an anchored configuration (HI-STORM 100A)

In contrast to an ISFSI containing free-standing casks, a constrained-cask installation relies on the structural capacity of the pad to ensure structural safety. The Part 72 regulation, require consideration of natural phenomenon in the design. Since an ISFSI pad, in an anchored cask installation, participates in maintaining the stability of the cask during "natural phenomena" on the cask and pad, it is an ITS structure. The procedure suggested in Regulatory Guide 7.10 [2.0.4] and the associated NUREG [2.0.5] indicates that an ISFSI pad used to secure anchored casks should be classified as a Category C ITS structure.

Because tipover of a cask installed in an anchored configuration is not feasible, the pad does not need to be engineered to accommodate this non-mechanistic event. However, the permissible carry height for a loaded HI-STORM 100A overpack must be established for the specific ISFSI pad using the methodology

described in this FSAR, if the load handling device is not designed in accordance with ANSI N 14.6 and does not have redundant drop protection design features. These requirements are specified in the CoC. However, to serve as an effective and reliable anchor, the pad must be made appropriately stiff and suitably secured to preclude pad uplift during a seismic event.

Because the geological conditions vary widely across the United States, it is not possible to, a priori, define the detailed design of the pad. Accordingly, in this FSAR, the limiting requirements on the design and installation of the pad are provided. The user of the HI-STORM 100A System bears the responsibility to ensure that all requirements on the pad set forth in this FSAR are fulfilled by the pad design. Specifically, the ISFSI owner must ensure that:

- The pad design complies with the structural provisions of this report. In particular, the requirements of ACI-349-97 [2.0.2] with respect to embedments must be assured.
- The material of construction of the pad (viz., the additives used in the pad concrete), and the attachment system are compatible with the ambient environment at the ISFSI site.
- The pad is designed and constructed in accordance with a Part 72, Subpart G-compliant QA program.
- The design and manufacturing of the cask attachment system are consistent with the provisions of this report.
- Evaluations are performed (e.g., per 72.212) to demonstrate that the seismic and other inertial loadings at the site are enveloped by the respective bounding loadings defined in this report.

A complete listing of design and construction requirements for an ISFSI pad on which an anchored HI-STORM 100A will be deployed is provided in Appendix 2.A. A sample embedment design is depicted in Figure 2.A.1.

2.0.4.2 Applicable Codes

Factored load combinations for ISFSI pad design are provided in NUREG-1536 [2.1.5], which is consistent with ACI-349-85. The factored loads applicable to the pad design consist of dead weight of the cask, thermal gradient loads, impact loads arising from handling and accident events, external missiles, and bounding environmental phenomena (such as earthquakes, wind, tornado, and flood). Codes ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R; "Guide for Concrete Floor and Slab Construction"; ACI 224R-90, "Control of Cracking in Concrete Structures" should be used in the design and construction of the concrete pad, as applicable. The embedment design for the HI-STORM 100A (and 100SA) are the responsibility of the ISFSI owner and shall comply with Appendix B to ACI-349-97 as described in Appendix 2.A. A later Code edition may be used provided a written reconciliation is performed.

The factored load combinations presented in Table 3-1 of NUREG 1536 are reduced in the following to a bounding set of load combinations that are applied to demonstrate adherence to its acceptance criteria.

a. Definitions

- D = dead load including the loading due to pre-stress in the anchor studs
- L = live load
- W = wind load
- W_t = tornado load
- T = thermal load
- F = hydrological load
- E = DBE seismic load
- A = accident load
- H = lateral soil pressure
- T_a = accident thermal load
- U_c = reinforced concrete available strength

Note that in the context of a complete ISFSI design, the DBE seismic load includes both the inertia load on the pad due to its self mass plus the interface loads transmitted to the pad to resist the inertia loads on the cask due to the loaded cask self mass. It is only these interface loads that are provided herein for possible use in the ISFSI structural analyses. The inertia load associated with the seismic excitation of the self mass of the slab needs to be considered in the ISFSI owner's assessment of overall ISFSI system stability in the presence of large uplift, overturning, and sliding forces at the base of the ISFSI pad. Such considerations are site specific and thus beyond the purview of this document.

b. Load Combinations for the Concrete Pad

The notation and acceptance criteria of NUREG-1536 apply.

Normal Events

$$U_c > 1.4D + 1.7L$$

$$U_c > 1.4D + 1.7(L+H)$$

Off-Normal Events

$$U_c > 1.05D + 1.275(L+H+T)$$

$$U_c > 1.05D + 1.275(L+H+T+W)$$

Accident-Level Events

$$U_c > D+L+H+T+F$$

$$U_c > D+L+H+T_a$$

$$U_c > D+L+H+T+E$$

$$U_c > D+L+H+T+W_t$$

$$U_c > D+L+H+T+A$$

In all of the above load combinations, the loaded cask weight is considered as a live load L on the pad. The structural analyses presented in Chapter 3 provide the interface loads contributing to "E", "F" and "W_t", which, for high-seismic sites, are the most significant loadings. The above set of load combinations can be reduced to a more limited set by recognizing that the thermal loads acting on the ISFSI slab are small because of the low decay heat loads from the cask. In addition, standard construction practices for slabs serve to ensure that extreme fluctuations in environmental temperatures are accommodated without extraordinary design measures. Therefore, all thermal loads are eliminated in the above combinations. Likewise, lateral soil pressure load "H" will also be bounded by "F" (hydrological) and "E" (earthquake) loads. Accident loads "A", resulting from a tipover, have no significance for an anchored cask. The following three load combinations are therefore deemed sufficient for structural qualification of the ISFSI slab supporting an anchored cask system.

Normal Events

$$U_c > 1.4D + 1.7(L)$$

Off-Normal Events

$$U_c > 1.05D + 1.275(L+F)$$

Accident-Level Events

$$U_c > D+L+E \text{ (or } W_t)$$

c. Load Combination for the Anchor Studs

The attachment bolts are considered to be governed by the ASME Code, Section III, Subsection NF and Appendix F [2.0.7]. Therefore, applicable load combinations and allowable stress limits for the attachment bolts are as follows:

Event Class and Load Combination	Governing ASME Code Section III Article for Stress Limits
<u>Normal Events</u>	
D	NF-3322.1, 3324.6
<u>Off-Normal Events</u>	
D+F	NF-3322.1, 3324.6 with all stress limits increased by 1.33
<u>Accident-Level Events</u>	
D+E and D+W _t	Appendix F, Section F-1334, 1335

2.0.4.3 Limiting Design Parameters

Since the loaded HI-STORM overpack will be carried over the pad, the permissible lift height for the cask must be determined site-specifically to ensure the integrity of the storage system in the event of a handling accident (uncontrolled lowering of the load). To determine the acceptable lift height, it is necessary to set down the limiting ISFSI design parameters. The limiting design parameters for an anchored cask ISFSI pad and the anchor studs, as applicable, are tabulated in Table 2.0.4. The design of steel embedments in reinforced concrete structures is governed by Appendix B of ACI-349-97. Section B.5 in that appendix states that "anchorage design shall be controlled by the strength of embedment steel...". Therefore, limits on the strength of embedment steel and on the anchor studs must be set down not only for the purposes of quantifying structural margins for the design basis load combinations, but also for the use of the ISFSI pad designer to establish the appropriate embedment anchorage in the ISFSI pad. The anchored cask pad design parameters presented in Table 2.0.4 allow for a much stiffer pad than the pad for free-standing HI-STORMs (Table 2.2.9). This increased stiffness has the effect of reducing the allowable lift height. However, a lift height for a loaded HI-STORM 100 cask (free-standing or anchored) is not required to be established if the cask is being lifted with a lift device designed in accordance with ANSI N14.6 having redundant drop protection design features.

In summary, the requirements for the ISFSI pad for free-standing and anchored HI-STORM deployment are similar with a few differences. Table 2.0.5 summarizes their commonality and differences in a succinct manner with the basis for the difference fully explained. The CoC provides the specific requirements for ISFSI pad design and establishing lift height limits.

2.0.4.4 Anchored Cask/ISFSI Interface

The contact surface between the baseplate of overpack and the top surface of the ISFSI pad defines the structural interface between the HI-STORM overpack and the ISFSI pad. When HI-STORM is deployed in an anchored configuration, the structural interface also includes the surface where the nuts on the anchor studs bear upon the sector lugs on the overpack baseplate. The anchor studs and their fastening arrangements into the ISFSI pad are outside of the structural boundary of the storage cask. While the details of the ISFSI pad design for the anchored configuration, like that for the free-standing geometry, must be custom engineered for each site, certain design and acceptance criteria are specified herein (Appendix 2.A) to ensure that the design and construction of the pad fully comports with the structural requirements of the HI-STORM System.

Table 2.0.1
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Table 1.2.2
License	20 yrs.	10CFR72.42(a) and 10CFR72.236(g)	-
Structural:			
Design Codes:			
Enclosure Vessel	ASME Code, Section III, Subsection NB	10CFR72.24(c)(4)	Section 2.0.1
Fuel Basket	ASME Code, Section III, Subsection NG	10CFR72.24(c)(4)	Section 2.0.1
MPC Lifting Points	ANSI N14.6/NUREG-0612	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights [†] :			
Max. Loaded Canister (dry)	90,000 lb.	R.G. 3.61	Table 3.2.1
Empty Canister (dry)	42,000 lb. (MPC-24) 45,000 lb. (MPC-24E/EF) 39,000 lb. (MPC-68/68F/68FF) 36,000 lb. (MPC-32)	R.G. 3.61	Table 3.2.1
Design Cavity Pressures:			
Normal:	100 psig	ANSI/ANS 57.9	Section 2.2.1.3
Off-Normal:	100 110 psig	ANSI/ANS 57.9	Section 2.2.2.1
Accident (Internal)	200 psig	ANSI/ANS 57.9	Section 2.2.3.8
Accident (External)	60 psig	ANSI/ANS 57.9	Sections 2.2.3.6 and 2.2.3.10
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(1)	Section 2.0.1

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware.

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Thermal:			
Maximum Design Temperatures:			
Structural Materials:			
Stainless Steel (Normal)	725° F	ASME Code Section II, Part D	Table 2.2.3
Stainless Steel (Accident)	950° F	ASME Code Section II, Part D	Table 2.2.3
Neutron Poison:			
Boron-Neutron Absorber (normal)	800° F	See Section Table 4.3.1 and Section 1.2.1.3.1	Table 2.2.3
Boron-Neutron Absorber (accident)	950° F	See Section Table 4.3.1 and Section 1.2.1.3.1 -	Table 2.2.3
PWR Fuel Cladding (Moderate/High Burnup Fuel):			
5-year cooled	691° / 679° F	PNL 6189/Appendix 4.A	Section 4.3/Appendix 4.A
6-year cooled	676° / 660° F	PNL 6189/Appendix 4.A	Section 4.3/Appendix 4.A
7-year cooled	635° / 635° F	PNL 6189/Appendix 4.A	Section 4.3/Appendix 4.A
10-year cooled	625° / 621° F	PNL 6189/Appendix 4.A	Section 4.3/Appendix 4.A
15-year cooled	614° / 611° F	PNL 6189/Appendix 4.A	Section 4.3/Appendix 4.A
BWR Fuel Cladding (Moderate/High Burnup Fuel):			
5-year cooled	740° F	PNL 6189	Section 4.3/Appendix 4.A
6-year cooled	712° F	PNL 6189	Section 4.3/Appendix 4.A
7-year cooled	669° F	PNL 6189	Section 4.3/Appendix 4.A
10-year cooled	658° F	PNL 6189	Section 4.3/Appendix 4.A
15-year cooled	646° F	PNL 6189	Section 4.3/Appendix 4.A
Canister Drying	See CoC Appendix A, Table 3-1	ISG-11	Section 4.5
Canister Backfill Gas	Helium		Section 12.3.3
Canister Backfill	See CoC Appendix A, Table 3-12	Thermal Analysis	Section 4.3
CSF Cladding Temperature Limit for Long-Term Storage Operations	752° F (400° C)	ISG-11	Section 4.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
<i>CSF Cladding Temperature Limit for Short Term Normal Operating Operations (e.g., MPC drying)</i>	<i>752 °F (400 °C), except certain MPCs containing all MBF may use 1058 °F (570 °C) with additional cladding stress analysis</i>	<i>ISG-11</i>	<i>Section 4.5</i>
Short-Term Allowable Fuel Cladding Temperature for Off-Normal and Accident Events	1058° F (570 °C)	PNL-4835/ISG-11	Sections 2.0.1 and 4.3
Insolation	Protected by overpack or HI-TRAC	-	Section 4.3
Confinement:		10CFR72.128(a)(3) and 10CFR72.236(d) and (e)	
Closure Welds:			
Shell Seams and Shell-to-Baseplate	Full Penetration	-	Section 1.5 and Table 9.1.4
MPC Lid	Multi-pass Partial Penetration	10CFR72.236(e)	Section 1.5 and Table 9.1.4
MPC Closure Ring	Partial Penetration		
Port Covers	Partial Penetration		
NDE			
Shell Seams and Shell-to-Baseplate	100% RT or UT	-	Table 9.1.4
MPC Lid	Root Pass and Final Surface 100% PT; Volumetric Inspection or 100% Surface PT each 3/8" of weld depth	-	Chapter 8 and Table 9.1.4
Closure Ring	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
Port Covers	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Leak Testing:			
Welds Tested	Shell seams, shell-to-baseplate, MPC lid-to-shell, and port covers-to-MPC lid	-	Section 7.1 and Chapters 8, 9, and 12
Medium	Helium	-	Sections 7.2 and Chapter 12
Max. Leak Rate	5×10^{-6} atm-cm ³ /sec (helium)	-	Chapter 12 (TS)
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
Hydrostatic Testing:			
Test Pressure	125 psig (+3, -0 psig)	-	Chapters 8 and 9
Welds Tested	MPC Lid-to-Shell, MPC Shell seams, MPC Shell-to-Baseplate	-	Section 8.1 and 9.1
Medium	Water	-	Section 8.1 and Chapter 9
Retrievability:			
Normal and Off-normal: Post (design basis) Accident	No Encroachment on Fuel Assemblies or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.4, 3.5, and 3.1.2
Criticality:			
Method of Control	Fixed Borated Neutron Absorber, Geometry, and Soluble Boron	10CFR72.124 & 10CFR72.236(c)	Section 2.3.4
Min. ¹⁰ Boron Loading	0.0267 g/cm ² (MPC-24) 0.0372 g/cm ² (MPC-68, MPC-68FF, MPC-24E, MPC-24EF, and MPC-32 and MPC-32F) 0.01 g/cm ² (MPC-68F)	-	Section 2.1.8
Minimum Soluble Boron	See CoC, Appendix A, LCO 3.3.1	Criticality Analysis	Section 6.1; CoC, Appendix B Tables 2.1.14 and 2.1.16
Max. k_{eff}	0.95	-	Sections 6.1 and 2.3.4
Min. Burnup	0.0 GWd/MTU (fresh fuel)	-	Section 6.1

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Radiation Protection/Shielding:		10CFR72.126, & 10CFR72.128(a)(2)	
MPC: (normal/off-normal/accident)			
MPC Closure	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
MPC Transfer	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
Exterior of Shielding: (normal/off-normal/accident)			
Transfer Mode Position	See Table 2.0.3	10CFR20	Section 5.1.1
ISFSI Controlled Area Boundary	See Table 2.0.2	10CFR72.104 & 10CFR72.106	Section 5.1.1 and Chapter 10
Design Bases:		10CFR72.236(a)	
Spent Fuel Specification:			
Assemblies/Canister	Up to 24 (MPC-24, MPC-24E & MPC-24EF) Up to 32 (MPC-32 and MPC-32F) Up to 68 (MPC-68, MPC-68F, & MPC-68FF)	-	Table 1.2.1
Type of Cladding	Zircaloy-ZR and Stainless Steel*	-	Table 2.1.6
Fuel Condition	Intact, Damaged, and Debris*	-	Section 2.1.2 & Table 2.1.6
* See Appendix B to the CoC for specific fuel condition requirements.			
PWR Fuel Assemblies:			
Type/Configuration	Various	-	Table 2.1.3
Max. Burnup	68,200 75,000 MWD/MTU	-	CoC, Appendix B
Max. Enrichment	Varies by fuel design	-	Table 2.1.3
Max. Decay Heat/Assembly MPC†:	40 kW	-	Section 4.0

† The Approved Contents Section of Appendix B to the CoC provides the decay heat limits per assembly

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
(Regionalized fuel loading)			
5-year cooled	1470 W (MPC 24) 1540 W (MPC 24E) 1132 W (MPC 32)		Table 4.4.31
6-year cooled	1470 W (MPC 24) 1540 W (MPC 24E) 1072 W (MPC 32)		Table 4.4.31
7-year cooled	1335 W (MPC 24) 1395 W (MPC 24E) 993 W (MPC 32)		Table 4.4.31
10-year cooled	1235 W (MPC 24) 1290 W (MPC 24E) 950 W (MPC 32)		Table 4.4.31
15-year cooled	1165 W (MPC 24) 1215 W (MPC 24E) 918 W (MPC 32)		Table 4.4.31
Minimum Cooling Time:	5 3 years (Intact Zr-ZR Clad Fuel) 8 years (Intact SS Clad Fuel)		CoC, Appendix B
Max. Fuel Assembly Weight: (including non-fuel hardware and DFC, as applicable)	1,680 lb.		Table 2.1.6
Max. Fuel Assembly Length: (Unirradiated Nominal)	176.8 in.		Table 2.1.6
Max. Fuel Assembly Width (Unirradiated Nominal)	8.54 in.		Table 2.1.6
Fuel Rod Fill Gas:			
Pressure (max.)	500 psig		Section 4.3 & Table 4.3.2
BWR Fuel Assemblies:			
Type	Various		Table 2.1.4
Max. Burnup	59,900 70,000 MWD/MTU		CoC, Appendix B
Max. Enrichment	Varies by fuel design		Section 6.1, Table 2.1.4

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Max. Decay Heat/Assy* [†] MPC [†] . (Regionalized Fuel Loading):	40 kW	-	Section 4.0
5 year cooled	501 W (MPC 68)	-----	Table 4.4.31
6 year cooled	468 W (MPC 68)	-----	Table 4.4.31
7 year cooled	419 W (MPC 68)	-----	Table 4.4.31
10 year cooled	406 W (MPC 68)	-----	Table 4.4.31
15 year cooled	392 W (MPC 68)	-----	Table 4.4.31
Minimum Cooling Time:	5 yrs 3 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)		CoC, Appendix B
Max. Fuel Assembly Weight: w/channels and DFC, as applicable	700 lb.	-	Table 2.1.6
Max. Fuel Assembly Length (Unirradiated Nominal)	176.5in.	-	Table 2.1.6
Max. Fuel Assembly Width (Unirradiated Nominal)	5.85 in.	-	Table 2.1.6
Fuel Rod Fill Gas:			
End-of-Life Hot Standby Pressure (max.)	147 psig	-	Table 4.3.5
Normal Design Event Conditions:		10CFR72.122(b)(1) -	
Ambient Temperatures	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.1.4
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/6 Yield	NUREG-0612 ANSI N14.6	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3

[†] The Approved Contents Section of Appendix B to the CoC provides the decay heat limits per assembly.

[†] The Approved Contents Section of Appendix B to the CoC provides the decay heat limits per assembly.

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
Fuel Rod Rupture Releases:			
Source Term Release Fraction	1% (2.5% for high burnup fuel)	NUREG-1536	Sections 2.2.1.3 and 7.2
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3 and 7.2
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3 and 7.2
Snow and Ice	Protected by Overpack	ASCE 7-88	Section 2.2.1.6
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.2.2
Leakage of One Seal	No Loss of Confinement	ANSI/ANS 57.9	Section 2.2.2.4
Partial Blockage of Overpack Air Inlets	Two Air Inlets Blocked		Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10% (11.5% for high burnup fuel)	NUREG-1536	Sections 2.2.2.1 and 7.2
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1 and 7.2
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1 and 7.2
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Tip Over	See Table 2.0.2		Section 2.2.3.2
End Drop	See Table 2.0.2		Section 2.2.3.1
Side Drop	See Table 2.0.3		Section 2.2.3.1
Fire	See Tables 2.0.2 and 2.0.3	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture Releases:			
Fuel Rod Failures (including non-fuel hardware)	100%	NUREG-1536	Sections 2.2.3.8 and 7.3
Fill Gases	100%	NUREG-1536	Sections 2.2.3.8 and 7.3
Fission Gases	30%	NUREG-1536	Sections 2.2.3.8 and 7.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Particulates & Volatiles	See Table 7.3.1	-	Sections 2.2.3.9 and 7.3
Confinement Boundary Leakage	7.5×10^{-6} atm-cm ³ /sec (helium)	TS leak rate plus test sensitivity	Sections 2.2.3.9 and 7.3
Explosive Overpressure	60 psig (external)	10CFR72.122(c)	Section 2.2.3.10
Airflow Blockage:			
Vent Blockage	100% of Overpack Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Partial Blockage of MPC Basket Vent Holes	Crud Depth (Table 2.2.8)	ESEERCO Project EP91-29	Section 2.2.3.4
Design Basis Natural Phenomenon Design Events and Conditions:		10CFR72.92 & 10CFR72.122(b)(2)	
Flood Water Depth	125 ft.	ANSI/ANS 57.9	Section 2.2.3.6
Seismic	See Table 2.0.2	10CFR72.102(f)	Section 2.2.3.7
Wind	Protected by Overpack	ASCE-7-88	Section 2.2.3.5
Tornado & Missiles	Protected by Overpack	RG 1.76 & NUREG-0800	Section 2.2.3.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	See Table 2.0.2	NFPA 78	Section 2.2.3.11
Extreme Environmental Temperature	See Table 2.0.2	-	Section 2.2.3.14

Table 2.0.2
HI-STORM OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.		Section 2.0.2
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design & Fabrication Codes:			
Concrete			
Design	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Fabrication	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Compressive Strength	ACI 318-95 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Structural Steel			
Design	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Fabrication	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Dead Weights[†]:			
Max. Loaded MPC (Dry)	88,135 lb. (MPC- 32)	R.G. 3.61	Table 3.2.1
Max. Empty Overpack Assembled with Top Lid	270,000 lb.	R.G. 3.61	Table 3.2.1
Max. MPC/Overpack	360,000 lb.	R.G. 3.61	Table 3.2.1
Design Cavity Pressures	N/A		Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

Table 2.0.2 (continued)
HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
Thermal:			
Maximum Design Temperatures:			
Concrete			
Local Maximum (Normal)	200° F	ACI 349 Appendix A	Table 2.2.3
Local Maximum (Accident)	350° F	ACI 349 Appendix A	Table 2.2.3
Steel Structure	350° F	ASME Code Section II, Part D	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.4.1.1.8
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
Retrievability:			
Normal and Off-normal	No damage that precludes Retrieval of MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 and 3.4
Accident			Sections 3.5 and 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:			
Overpack (Normal/Off-normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10
Beyond Controlled Area During Normal Operation and Anticipated Occurrences	25 mrem/yr. to whole body 75 mrem/yr. to thyroid 25 mrem/yr. to any critical organ	10CFR72.104	Sections 5.1.1, 7.2, and 10.1
At Controlled Area Boundary from Design Basis Accident	5 rem TEDE or sum of DDE and CDE to any individual organ or tissue (other than lens of eye) <	10CFR72.106	Sections 5.1.2, 7.3, and 10.1

Table 2.0.2 (continued)
 HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	50 rem. 15 rem lens dose. 50 rem shallow dose to skin or extremity.		
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:			
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load [†] :			
Loaded Transfer Cask (max.)	245,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.2 Section 2.2.1.2
Dry Loaded MPC (max.)	90,000 lb.	R.G. 3.61	Table 3.2.1 and Section 2.2.1.2
Handling:			
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/6 Yield ANSI N14.6	NUREG-0612 ANSI N14.6	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature During Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Snow and Ice Load	100 lb./ft ²	ASCE 7-88	Section 2.2.1.6
Wet/Dry Loading	Dry	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

Table 2.0.2 (continued)
 HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	-40° F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100° F	ANSI/ANS 57.9	Section 2.2.2.2
Partial Blockage of Air Inlets	Two Air Inlet Ducts Blocked	-	Section 2.2.2.5
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.94	
Drop Cases:			
End	11 in.	-	Section 2.2.3.1
Tip-Over (Not applicable for HI-STORM 100A)	Assumed (Non-mechanistic)	-	Section 2.2.3.2
Fire:			
Duration	217 seconds	10CFR72.122(c)	Section 2.2.3.3
Temperature	1,475° F	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture	See Table 2.0.1	-	Section 2.2.3.8
Air Flow Blockage:			
Vent Blockage	100% of Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Ambient Temperature	80° F	10CFR72.128(a)(4)	Section 2.2.3.13
Explosive Overpressure External Differential Pressure	10 psid instantaneous, 5 psid steady state	10 CFR 72.128(a)(4)	Table 2.2.1
Design-Basis Natural Phenomenon Design Events and Conditions:		10CFR72.92 & 10CFR72.122(b)(2)	
Flood			
Height	125 ft.	RG 1.59	Section 2.2.3.6
Velocity	15 ft/sec.	RG 1.59	Section 2.2.3.6
Seismic			
Max. ZPA at top of ISFSI pad	Free Standing: $G_H + 0.53G_V \leq 0.53$ Anchored: $G_H \leq 2.12, G_V \leq 1.5$	10CFR72.102(f)	Section 3.4.7.1 Section 3.4.7.3
Tornado			
Wind			
Max. Wind Speed	360 mph	RG 1.76	Section 2.2.3.5

Table 2.0.2 (continued)
 HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Pressure Drop	3.0 psi	RG 1.76	Section 2.2.3.5
Missiles			Section 2.2.3.5
Automobile			
Weight	1,800 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder			
Weight	125 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	8 in.	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	0.22 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	1 in.	NUREG-0800	Table 2.2.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	Resistance Heat-Up	NFPA 70 & 78	Section 2.2.3.11
Extreme Environmental Temperature	125° F	-	Section 2.2.3.14
Load Combinations:	See Table 2.2.14 and Table 3.1.5	ANSI/ANS 57.9 and NUREG-1536	Section 2.2.7

TABLE 2.0.3
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Section 2.0.3
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design Codes:			
Structural Steel	ASME Code, Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.3
Lifting Trunnions	NUREG-0612 & ANSI N14.6	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights [†] :			
Max. Empty Cask:			
w/top lid and pool lid installed and water jacket filled	143,500 lb. (HI-TRAC 125) 102,000 lb. (HI-TRAC 100) 143,000 lb. (HI-TRAC 125D)	R.G. 3.61	Table 3.2.2
w/top lid and transfer lid installed and water jacket filled (N/A for HI-TRAC 125D)	155,000 lb. (HI-TRAC 125) 111,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.2
Max. MPC/HI-TRAC with Yoke (in-pool lift):			
Water Jacket Empty	245,000 lb. (HI-TRAC 125 and 125D) 202,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3
Water Jacket Cavity	60 psig (internal)	ANSI/ANS 57.9	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Section 2.0.3

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

TABLE 2.0.3 (continued)
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	Continued adequate performance of HI-TRAC transfer cask	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
Thermal:			
Maximum Design Temperature			
Structural Materials	400° F	ASME Code Section II, Part D	Table 2.2.3
Shielding Materials			
Lead	350° F (max.)		Table 2.2.3
Liquid Neutron Shield	307° F (max.)		Table 2.2.3
Solid Neutron Shield	300° F (max.)	Manufacturer Data	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.5.1.1.3
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
Retrievability:			
Normal and Off-normal	No encroachment on MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 & 3.4
After Design-basis (Postulated) Accident			Section 3.5 & 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:			
Transfer Cask (Normal/Off-normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:			
Ambient Temperatures:		10CFR72.122(b)(1)	

TABLE 2.0.3 (continued)
 HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Lifetime Average	100° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load [†]			
Max. Loaded Canister			
Dry	90,000 lb.	R.G. 3.61	Table 3.2.1
Wet (including water in HI-TRAC annulus)	106,570 lb.	R.G. 3.61	Table 3.2.4
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/6 Yield	NUREG-0612 ANSI N14.6	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature for Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Test Loads:			
Trunnions	300% of vertical design load	NUREG-0612 & ANSI N14.6	Section 9.1.2.1
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	0° F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100° F	ANSI/ANS 57.9	Section 2.2.2.2
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1
Fire			
Duration	4.8 minutes	10CFR72.122(c)	Section 2.2.3.3

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

TABLE 2.0.3 (continued)
 HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Temperature	1,475° F	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture	See Table 2.0.1		Section 2.2.3.8
Design-Basis Natural Phenomenon Design Events and Conditions:		10CFR72.92 & 10CFR72.122(b)(2)	
Missiles			Section 2.2.3.5
Automobile			
Weight	1800 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder			
Weight	125 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	8 in.	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	0.22 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	1 in.	NUREG-0800	Table 2.2.5
Load Combinations:	See Table 2.2.14 and Table 3.1.5	ANSI/ANS-57.9 & NUREG-1536	Section 2.2.7

TABLE 2.0.4
LIMITING DESIGN PARAMETERS FOR ISFSI PADS AND ANCHOR STUDS FOR HI-STORM 100A

Item	Maximum Permitted Value†	Minimum Permitted Value
ISFSI PAD		
Pad Thickness	---	48 inches
Subgrade Young's Modulus from Static Tests (needed if pad is not founded on rock)	---	10,000 psi
Concrete compressive strength at 28 days	---	4,000 psi
ANCHOR STUDS		
Yield Strength at Ambient Temperature	None	80,000 psi
Ultimate Strength at Ambient Temperature	None	125,000 psi
Initial Stud Tension	65 ksi	55 ksi

† Pad and anchor stud parameters to be determined site-specifically, except where noted.

TABLE 2.0.5
ISFSI PAD REQUIREMENTS FOR FREE-STANDING AND ANCHORED HI-STORM INSTALLATION

Item	Free-Standing	Anchored	Comments
1. Interface between cask and ISFSI	Contact surface between cask and top surface of ISFSI pad	Same as free-standing with the addition of the bearing surface between the anchor stud nut and the overpack baseplate. (The interface between the anchor stud and the anchor receptacle is at the applicable threaded or bearing surface).	All components below the top surface of the ISFSI pad and in contact with the pad concrete are part of the pad design. A non-integral component such as the anchor stud is not part of the embedment even though it may be put in place when the ISFSI pad is formed. The embedment for the load transfer from the anchor studs to the concrete ISFSI pad shall be exclusively cast-in-place.
2. Applicable ACI Code	At the discretion of the ISFSI owner. ACI-318 and ACI-349 are available candidate codes.	ACI-349-97. A later edition of this Code may be used if a written reconciliation is performed.	ACI-349-97 recognizes increased structural role of the ISFSI pad in an anchored cask storage configuration and imposes requirements on embedment design.
3. Limitations on the pad design parameters	Per Table 2.2.9	Per Table 2.0.4	In free-standing cask storage, the non-mechanistic tipover requirement limits the stiffness of the pad. In the anchored storage configuration, increased pad stiffness is permitted; however, the permissible HI-STORM carry height is reduced.
4. HI-STORM Carry Height	11 inches (for ISFSI pad parameter Set A or Set B) or, otherwise, site-specific. Not applicable if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.	Determined site-specifically. Not applicable if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.	Appendix 3.A provides the technical basis for free-standing installation. Depending on the final ISFSI pad configuration (thickness, concrete strength, subgrade, etc.), and the method of transport, an allowable carry height may need to be established.

TABLE 2.0.5 (continued)
ISFSI PAD REQUIREMENTS FOR FREE-STANDING AND ANCHORED HI-STORM INSTALLATION

<i>Item</i>	<i>Free-Standing</i>	<i>Anchored</i>	<i>Comments</i>
5. Maximum seismic input on the pad/cask contact surface. G_H is the vectorial sum of the two horizontal ZPAs and G_V is the vertical ZPA	$G_H + \mu G_V \leq \mu$	$G_H \leq 2.12$ AND $G_V \leq 1.5$	
6. Required minimum value of cask to pad static coefficient of friction (μ , must be confirmed by testing).	Greater than or equal to 0.53 (per Table 2.2.9).	Same as that for free-standing condition	
7. Applicable Wind and Large Missile Loads	Per Table 2.2.4, missile and wind loading different from the tabulated values, require 10CFR 72.48 evaluation	The maximum overturning moment at the base of the cask due to lateral missile and/or wind action must be less than 1×10^7 ft-lb.	The bases are provided in Section 3.4.8 for free-standing casks; the limit for anchored casks ensures that the anchorage system will have the same structural margins established for seismic loading.
8. Small and medium missiles (penetrant missile)	Per Table 2.2.5, missiles and wind loading different from the tabulated value, require 10CFR 72.48 evaluation.	Same as for free-standing cask construction.	
9. Design Loadings for the ISFSI Pad	Per load combinations in Section 2.0.4 using site-specific load.	Same as for free-standing cask.	

2.1 SPENT FUEL TO BE STORED

2.1.1 Determination of The Design Basis Fuel

The HI-STORM 100 System is designed to store most types of fuel assemblies generated in the commercial U.S. nuclear industry. Boiling-water reactor (BWR) fuel assemblies have been supplied by The General Electric Company (GE), Siemens, Exxon Nuclear, ANF, UNC, ABB Combustion Engineering, and Gulf Atomic. Pressurized-water reactor (PWR) fuel assemblies are generally supplied by Westinghouse, Babcock & Wilcox, ANF, and ABB Combustion Engineering. ANF, Exxon, and Siemens are historically the same manufacturing company under different ownership. Within this report, SPC is used to designate fuel manufactured by ANF, Exxon, or Siemens. Publications such as Refs. [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors. A central object in the design of the HI-STORM 100 System is to ensure that a majority of SNF discharged from the U.S. reactors can be stored in one of the MPCs.

The cell openings and lengths in the fuel basket have been sized to accommodate the BWR and PWR assemblies listed in Refs. [2.1.1] and [2.1.2] except as noted below. Similarly, the cavity length of the multi-purpose canisters has been set at a dimension which permits storing most types of PWR fuel assemblies and BWR fuel assemblies with or without fuel channels. The one exception is as follows:

- i. The South Texas Units 1 & 2 SNF, and CE 16x16 System 80 SNF are too long to be accommodated in the available MPC cavity length.

In addition to satisfying the cross sectional and length compatibility, the active fuel region of the SNF must be enveloped in the axial direction by the neutron absorber located in the MPC fuel basket. Alignment of the neutron absorber with the active fuel region is ensured by the use of upper and lower fuel spacers suitably designed to support the bottom and restrain the top of the fuel assembly. The spacers axially position the SNF assembly such that its active fuel region is properly aligned with the neutron absorber in the fuel basket. Figure 2.1.5 provides a pictorial representation of the fuel spacers positioning the fuel assembly active fuel region. Both the upper and lower fuel spacers are designed to perform their function under normal, off-normal, and accident conditions of storage.

In summary, the geometric compatibility of the SNF with the MPC designs does not require the definition of a design basis fuel assembly. This, however, is not the case for structural, confinement, shielding, thermal-hydraulic, and criticality criteria. In fact, a particular fuel type in a category (PWR or BWR) may not control the cask design in all of the above-mentioned criteria. To ensure that no SNF listed in Refs. [2.1.1] and [2.1.2] which is geometrically admissible in the MPC is precluded, it is necessary to determine the governing fuel specification for each analysis criterion. To make the necessary determinations, potential candidate fuel assemblies for each qualification criterion were considered. Table 2.1.1 lists the PWR fuel assemblies that were evaluated. These fuel assemblies were evaluated to define the governing design criteria for PWR fuel. The BWR fuel assembly designs evaluated are listed in Table 2.1.2. Tables 2.1.3 and 2.1.4 provide the fuel characteristics determined to be acceptable for storage in the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 is acceptable for

storage in the HI-STORM 100 System. These tables present the groups of fuel assembly types defined as "array/classes" as described in further detail in Chapter 6. Table 2.1.5 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and decay heat generation. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STORM 100 System is intact Zircaloy-ZR clad fuel assemblies with the characteristics listed in Table 2.1.6 or intact stainless steel clad fuel assemblies with the characteristics listed in Table 2.1.8. The placement of a single stainless steel clad fuel assembly in a MPC necessitates that all fuel assemblies (stainless steel clad or Zircaloy-ZR clad) stored in that MPC meet the maximum heat generation requirements for stainless steel clad fuel specified in Table 2.1.8. Intact BWR MOX fuel assemblies shall meet the requirements of Table 2.1.7.

Intact fuel assemblies without fuel rods in fuel rod locations cannot be loaded into the HI-STORM 100 unless dummy fuel rods, which occupy a volume greater than or equal to the original fuel rods, replace the missing rods prior to loading. Any intact fuel assembly that falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly, as defined in the Approved Contents section of Appendix B to the CoC can be safely stored in the HI-STORM 100 System.

The range of fuel characteristics specified in Tables 2.1.3 and 2.1.4 have been evaluated in this FSAR and are acceptable for storage in the HI-STORM 100 System.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in Table 1.0.1.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with 250 x 250 fine mesh screens, for storage in the HI-STORM 100 System (*see Figures 2.1.1 and 2.1.2B, C, and D*). The MPC-24E and MPC 32 are designed to accommodate PWR damaged fuel. The MPC-24EF and MPC-32F are designed to accommodate PWR damaged fuel and fuel debris. The MPC-68 is designed to accommodate BWR damaged fuel. The MPC-68F and MPC-68FF are designed to accommodate BWR damaged fuel and fuel debris. The appropriate structural, thermal, shielding, criticality, and confinement analyses have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for damaged fuel assemblies authorized for loading in the HI-STORM 100 System are provided in Table 2.1.7. Restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in the Approved Contents section of Appendix B to the CoC. Dresden Unit 1 fuel assemblies contained in Transnuclear-designed damaged fuel canisters and one Dresden

Unit 1 thoria rod canister have been approved for storage directly in the HI-STORM 100 System without re-packaging (see Figures 2.1.2 and 2.1.2A).

MPC contents classified as fuel debris are required to be stored in DFCs and in the applicable "F" model MPC as specified in Appendix B to the CoC. The "F" (or "FF") indicates the MPC is qualified for storage of intact fuel, damaged fuel, and fuel debris, in quantities and locations specified in Appendix B to the CoC. The basket designs for the standard and "F" model MPCs are identical. The lid and shell designs of the "F" models are unique in that the upper shell portion of the canister is thickened for additional strength needed under hypothetical accident conditions of transportation under 10 CFR 71. This design feature is not required for dry storage, but must be considered in fuel loading for dry storage to ensure the dual purpose function of the MPC by eliminating the need to re-package the fuel for transportation. Figure 2.1.9 shows the details of the differences between the standard and "F" model MPC shells. These details are common for both the PWR and BWR series MPC models.

2.1.4 Deleted

2.1.5 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are listed in Tables 2.1.6, 2.1.7, and 2.1.8. The centers of gravity reported in Section 3.2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10. In order to qualify for storage in the MPC, the SNF must satisfy the physical parameters listed in Tables 2.1.6, 2.1.7, or 2.1.8.

2.1.6 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly, the allowable fuel cladding temperature based on cooling time, and the decay heat removal capabilities of the HI-STORM 100 System. The maximum heat generation rate per assembly for the design basis fuel assembly is based on the fuel assembly type with the highest decay heat for a given enrichment, burnup, and cooling time. This decay heat design basis fuel assembly is listed in Table 2.1.5. Section 5.2 describes the method used to determine the design basis fuel assembly type and calculate the decay heat load.

To ensure the permissible fuel cladding temperature limits are not exceeded, the Approved Contents section of Appendix B to the CoC specifies the allowable decay heat per assembly for each MPC model. For both uniform and regionalized loading of moderate and high burnup Zircaloy-ZR clad fuel assemblies, the allowable decay heat per assembly is a function of cooling time and is presented in Appendix B to the CoC in Tables 2.1.5 and 2.1.7. For stainless steel clad fuel assemblies, the

allowable decay heat per assembly is ~~not dependent upon cooling time and is~~ specified in Table 2.1-1 of Appendix B to the CoC. Due to the large conservatisms in the thermal evaluations and the relatively long cooling times and corresponding low decay heats for stainless steel clad fuel, an age-dependent allowable decay heat limit is not necessary.

The specified decay heat load can be attained by varying burnups and cooling times. The Approved Contents section of Appendix B to the CoC provides the burnup and cooling time limits for intact ~~Zircaloy~~ZR clad fuel to meet the thermal requirements for the various MPC's.

The Approved Contents section of Appendix B to the CoC also includes separate cooling time, burnup, and decay heat limits for uniform fuel loading and regionalized fuel loading. Regionalized loading allows higher heat emitting fuel assemblies to be stored in the center fuel storage locations than would otherwise be authorized for storage under uniform loading conditions.

The fuel cladding temperature is also affected by other factors. A governing geometry that maximizes the impedance to the transmission of heat out of the fuel rods has been defined. The governing thermal parameters to ensure that the range of SNF discussed previously are bounded by the thermal analysis are discussed in detail and specified in Chapter 4. By utilizing these bounding thermal parameters, the calculated peak fuel rod cladding temperatures are conservative for actual spent fuel assemblies which have greater thermal conductivities.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in Refs. [2.1.7] and [2.1.8] are utilized and summarized in Table 2.1.11 and Figures 2.1.3 and 2.1.4 for reference. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Fuel may be stored in the MPC using one of two storage strategies, namely, uniform loading and regionalized loading. Uniform loading allows storage of any fuel assembly in any fuel storage location, subject to additional restrictions specified in the CoC for ~~preferential fuel loading and~~ loading of fuel assemblies containing non-fuel hardware as defined in Table 1.0.1. Regionalized fuel loading allows for higher heat emitting fuel assemblies to be stored in the central core basket storage locations with lower heat emitting fuel assemblies in the peripheral fuel storage locations. Regionalized loading allows storage of higher heat emitting fuel assemblies than would otherwise be permitted using the uniform loading strategy. The definition of the regions for each MPC model and the associated burnup, cooling time, and decay heat limits are found in Appendix B to the CoC. Regionalized fuel loading meets the intent of preferential fuel loading.

2.1.7 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM 100 System are the 10CFR72.104 site boundary dose rate limits and maintaining operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the SNF assembly.

The gamma and neutron sources are separate and are affected differently by enrichment, burnup, and cooling time. It is recognized that, at a given burnup, the radiological source terms increase monotonically as the initial enrichment is reduced. The shielding design basis fuel assembly, therefore, is evaluated at conservatively high burnups, low cooling times, and low enrichments, as discussed in Chapter 5. The shielding design basis fuel assembly thus bounds all other fuel assemblies.

The design basis dose rates can be met by a variety of burnup levels and cooling times. The Approved Contents section of Appendix B to the CoC provides the burnup and cooling time limits for all of the authorized fuel assembly array/classes for both uniform fuel loading and regionalized loading.

Table 2.1.11 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Thoria Rod Canister is authorized for storage in combination with other intact and damaged fuel, and fuel debris as specified in Appendix B to the CoC.

Non-fuel hardware, as defined in Table 1.0.1 and I the CoC has been evaluated and is authorized for storage in the PWR MPCs as specified in Appendix B to the CoC.

2.1.8 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68, MPC-68F, MPC-68FF, and MPC-32 and MPC-32F feature a basket without flux traps. In the aforementioned baskets, there is one panel of neutron absorber between two adjacent fuel assemblies. The MPC-24, MPC-24E, and MPC-24EF employ a construction wherein two neighboring fuel assemblies are separated by two panels of neutron absorber with a water gap between them (flux trap construction).

The minimum ^{10}B areal density in the neutron absorber panels for each MPC model is shown in Table 2.1.15.

For all MPCs, the ^{10}B areal density used for analysis is conservatively established at 75% of the minimum ^{10}B areal density to demonstrate that the reactivity under the most adverse accumulation of tolerances and biases is less than 0.95. This is consistent with NUREG-1536 [2.1.5] which requires a 25% reduction in ^{10}B areal density credit. A large body of sampling data accumulated by Holtec from thousands of manufactured Boral panels indicates the average ^{10}B areal densities to be approximately 15% greater than the specified minimum.

The criticality analyses for the MPC-24, MPC-24E and MPC-24EF (all with higher enriched fuel) and for the MPC-32 and MPC-32F were performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.14 and 2.1.16 provides the required soluble boron concentrations for these MPCs. Minimum soluble boron concentration is also included as Limiting Condition for Operation (LCO)3.3.1 in the Technical Specifications found in Appendix A to the CoC.

2.1.9 Summary of SNF Design Criteria

Tables 2.1.1 through 2.1.8 and Table 2.1.12 provide the design characteristics for spent fuel and non-fuel hardware authorized for storage in the HI-STORM 100 System. Much of this information is repeated in the Approved Contents section of Appendix B to the CoC. Only fuel meeting the specifications in the CoC is authorized for storage. Fuel classified as damaged fuel assemblies or fuel debris must be stored in damaged fuel containers for storage in the HI-STORM 100 System.

Table 2.1.1

PWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	Array Type
B&W 15x15	All
B&W 17x17	All
CE 14x14	All
CE 16x16	All except System 80 TM
WE 14x14	All
WE 15x15	All
WE 17x17	All
St. Lucie	All
Ft. Calhoun	All
Haddam Neck (Stainless Steel Clad)	All
San Onofre 1 (Stainless Steel Clad)	All
Indian Point 1	All

Table 2.1.2

BWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	Array Type			
GE BWR/2-3	All 7x7	All 8x8	All 9x9	All 10x10
GE BWR/4-6	All 7x7	All 8x8	All 9x9	All 10x10
Humboldt Bay	All 6x6	All 7x7 (Zircaloy ZR Clad)		
Dresden-1	All 6x6	All 8x8		
LaCrosse (Stainless Steel Clad)	All			

Table 2.1.3
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	14x14 A	14x14 B	14x14 C	14x14 D	14x14E
Clad Material (Note 2)	Zr/ZR	ZR/Zr	ZR/Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407 361	≤ 407 408	≤ 425 433	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % ²³⁵ U) (Note 7)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 5.0 (24E/24EF)	≤ 5.0 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15 A	15x15 B	15x15 C	15x15 D	15x15 E	15x15 F
Clad Material (Note 2)	ZRZ z					
Design Initial U (kg/assy.) (Note 3)	≤ 464 468	≤ 464 468	≤ 464 468	≤ 475 495	≤ 475 495	≤ 475 495
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % ²³⁵ U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)					
Initial Enrichment (MPC-24, 24E, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	15x15 G	15x15H	16x16 A	17x17A	17x17 B	17x17 C
Clad Material (Note 2)	SS	ZRZ ±				
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475 495	≤ 443	≤ 467 428	≤ 467 469	≤ 474 475
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % ²³⁵ U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≥ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in)	≥ 0.0145	≥ 0.140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. ~~Zr designates cladding material made of zirconium or zirconium alloys~~ See Table 1.0.1 for the definition of "ZR."
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records
4. Each guide tube replaces four fuel rods.
5. Soluble boron concentration per Technical Specification LCO 3.3.1.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum initial enrichment of the intact fuel assemblies, damaged fuel assemblies and fuel debris is 4.0 wt.% ²³⁵U.

Table 2.1.4
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	6x6 A	6x6 B	6x6 C	7x7 A	7x7 B	8x8 A
Clad Material (Note 2)	ZRZ#	ZRZ#	ZRZ#	ZRZ#	ZRZ#	ZRZ#
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 2.7	≤ 2.7 for UO ₂ rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ²³⁵ U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	8x8 B	8x8 C	8x8 D	8x8 E	8x8F	9x9 A
Clad Material (Note 2)	ZRZ ±	ZRZ ±	ZRZ ±	ZRZ ±	ZRZ ±	ZRZ ±
Design Initial U (kg/assy.) (Note 3)	≤ 191 192	≤ 191 183	≤ 191 183	≤ 191 183	≤ 191	≤ 179 180
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	9x9 B	9x9 C	9x9 D	9x9 E (Note 13)	9x9 F (Note 13)	9x9 G
Clad Material (Note 2)	ZRZ#	ZRZ#	ZRZ#	ZRZ#	ZRZ#	ZRZ#
Design Initial U (kg/assy.) (Note 3)	≤ 179 173	≤ 179 175	≤ 179 175	≤ 179 183	≤ 179 183	≤ 179 157
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	10x10 A	10x10 B	10x10 C	10x10 D	10x10 E
Clad Material (Note 2)	ZRZ z	ZRZ z	ZRZ z	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 188 172	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ²³⁵ U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. See Table 1.0.1 for the definition of "ZR." ~~Zr designates cladding material made of zirconium or zirconium alloys.~~
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users' fuel records to account for manufacturer tolerances.
4. ≤ 0.635 wt. % ^{235}U and ≤ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2)
5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
6. Square, replacing nine fuel rods.
7. Variable.
8. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
11. These rods may also be sealed at both ends and contain Zr material in lieu of water.
12. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.
13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.
14. For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum planar average initial enrichment for the intact fuel assemblies is limited to 3.7 wt.% ^{235}U , as applicable.

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	MPC-68/68F/68FF	MPC-24	MPC-24E/24EF	MPC-32
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Class 10x10A)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)
Source Term (Shielding)	GE 7x7 (Class 7x7B)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)
Decay Heat (Thermal- Hydraulic)	GE 7x7 (Class 7x7B)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)	B&W 15x15 (Class 15x15F)

Table 2.1.6

DESIGN CHARACTERISTICS FOR INTACT ZIRCALOY ZR CLAD
FUEL ASSEMBLIES¹

	MPC-68/68FF	MPC-68F	MPC-24	MPC-24E/24EF	MPC-32/32F
PHYSICAL PARAMETERS:					
Max assembly width (in.)	5.85	4.70	8.54	8.54	8.54
Max assembly length (in.)	176.5	135.0	176.8	176.8	176.8
Max assembly weight ² (lb.)	700	400	1680	1680	1680
Max active fuel length (in.)	150	120	150	150	150
RADIOLOGICAL AND THERMAL CHARACTERISTICS:					
	MPC-68/68FF	MPC-68F	MPC-24	MPC-24E/24EF	MPC-32/32F
Max. initial enrichment (wt% ²³⁵ U)	See Table 2.1.4	See Table 2.1.4	See Table 2.1.3	See Table 2.1.3	See Table 2.1.3
Max. heat generation (W)	Table 2.0.1				
Max. average burnup (MWD/MTU) ²	59,000 70,000	30,000	66,200 75,000	68,200 75,000	54,700 75,000
Min. cooling time (years)	5.3	18	5.3	5.3	5.3

- ¹ These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.
- ² Fuel assembly weight including non-fuel hardware, and channels, as applicable, based on DOE MPC DPS [2.1.6]
- ² The maximum burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is 45,000 MWD/MTU

Table 2.1.7
DESIGN CHARACTERISTICS FOR DAMAGED
FUEL ASSEMBLIES¹

	MPC-68/68FF (Damaged Fuel and Fuel Debris)	MPC-68F (Damaged Fuel and Fuel Debris)	MPC-24E/24EF MPC-32/MPC-32F (Damaged Fuel and Fuel Debris)
PHYSICAL PARAMETERS:			
Max. assembly width (in.)	5.5	4.7	8.54
Max. assembly length (in.)	176.5	135.0	176.8
Max. assembly weight ² (lb.)	700	550	1680
Max. active fuel length (in.)	150	110	150
Fuel rod clad material	Zircaloy ZR/SS	Zircaloy ZR	Zircaloy ZR/SS
RADIOLOGICAL AND THERMAL CHARACTERISTICS:			
Max. heat generation (W)	356 See Table 2.0.1	115	927 See Table 2.0.1
Min. cooling time (yr)	5.3	18	5.3
Max. initial enrichment (wt.% ²³⁵ U) for UO ₂ rods	4.0	2.7	4.0, 5.0
Max. initial enrichment for MOX rods	0.635 wt.% ²³⁵ U 1.578 wt. % Total Fissile Plutonium	0.635 wt.% ²³⁵ U 1.578 wt. % Total Fissile Plutonium	N/A
Max. average burnup (MWD/MTU) ³	59,000 70,000 (ZR) 22,500 (SS)	30,000	68,200 75,000 (ZR) 40,000 (SS)

Note: Refer to the Approved Contents section of Appendix B to the CoC for restrictions on the number and location of damaged fuel assemblies and fuel debris authorized for loading in the HI-STORM 100 System.

- ¹ These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.
- ² Fuel assembly weight including non-fuel hardware, channels, and DFC, as applicable, based on DOE MPC DPS [2.1.6].
- ³ ~~The maximum burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is 45,000 MWD/MTU~~

Table 2.1.8

DESIGN CHARACTERISTICS FOR INTACT STAINLESS STEEL CLAD FUEL ASSEMBLIES¹

	BWR MPC-68/68FF	PWR MPC-24/24E/24EF	PWR MPC-32/32F
PHYSICAL PARAMETERS:			
Max. assembly width ² (in.)	5.62	8.54	8.54
Max. assembly length ² (in.)	102.5	176.8	176.8
Max. assembly weight ³ (lb.)	700	1680	1680
Max. active fuel length ² (in.)	83	144	144
RADIOLOGICAL AND THERMAL CHARACTERISTICS:			
Max. heat generation (W)	95	710	500
Min. cooling time (yr)	10	8	9/20
Max. initial enrichment without soluble boron credit (wt.% ²³⁵ U)	4.0	See Table 2.1.3	N/A
Max. initial enrichment with soluble boron credit (wt.% ²³⁵ U)	N/A	5.0	5.0
Max. average burnup (MWD/MTU)	22,500	40,000	30,000/40,000

¹ These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.

² Unirradiated nominal dimensions are shown.

³ Fuel assembly weight including non-fuel hardware and channels, as applicable, based on DOE MPC DPS [2.1.6].

Table 2.1.9

SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length w/o NFH ¹ (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
CE 14x14	157	4.1	137	9.5	10.0
CE 16x16	176.8	4.7	150	0	0
BW 15x15	165.7	8.4	141.8	6.7	4.1
W 17x17 OFA	159.8	3.7	144	8.2	8.5
W 17x17 Std	159.8	3.7	144	8.2	8.5
W 17x17 VSH	160.1	3.7	144	7.9	8.5
W 15x15	159.8	3.7	144	8.2	8.5
W 14x14 Std	159.8	3.7	145.2	9.2	7.5
W 14x14 OFA	159.8	3.7	144	8.2	8.5
Ft. Calhoun	146	6.6	128	10.25	20.25
St. Lucie 2	158.2	5.2	136.7	10.25	8.05
B&W 15x15 SS	137.1	3.873	120.5	19.25	19.25
W 15x15 SS	137.1	3.7	122	19.25	19.25
W 14x14 SS	137.1	3.7	120	19.25	19.25
Indian Point 1	137.2	17.705	101.5	18.75	20.0

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the Boral neutron poison region of the MPC basket with water in the MPC.

¹ NFH is an abbreviation for non-fuel hardware, including control components. Fuel assemblies with control components may require shorter fuel spacers.

Table 2.1.10

SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
GE/2-3	171.2	7.3	150	4.8	0
GE/4-6	176.2	7.3	150	0	0
Dresden 1	134.4	11.2	110	18.0	28.0
Humboldt Bay	95.0	8.0	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	142.1 [†]	11.2	110	17.0	16.9
Humboldt Bay Damaged Fuel or Fuel Debris	105.5 [†]	8.0	79	35.25	35.25
LaCrosse	102.5	10.5	83	37.0	37.5

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the Boral neutron poison region of the MPC basket with water in the MPC.

[†] Fuel assembly length includes the damaged fuel container.

Table 2.1.11
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION ¹		
Interval	Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)	Normalized Distribution
1	0% to 4-1/6%	0.5485
2	4-1/6% to 8-1/3%	0.8477
3	8-1/3% to 16-2/3%	1.0770
4	16-2/3% to 33-1/3%	1.1050
5	33-1/3% to 50%	1.0980
6	50% to 66-2/3%	1.0790
7	66-2/3% to 83-1/3%	1.0501
8	83-1/3% to 91-2/3%	0.9604
9	91-2/3% to 95-5/6%	0.7338
10	95-5/6% to 100%	0.4670
BWR DISTRIBUTION ²		
Interval	Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)	Normalized Distribution
1	0% to 4-1/6%	0.2200
2	4-1/6% to 8-1/3%	0.7600
3	8-1/3% to 16-2/3%	1.0350
4	16-2/3% to 33-1/3%	1.1675
5	33-1/3% to 50%	1.1950
6	50% to 66-2/3%	1.1625
7	66-2/3% to 83-1/3%	1.0725
8	83-1/3% to 91-2/3%	0.8650
9	91-2/3% to 95-5/6%	0.6200
10	95-5/6% to 100%	0.2200

¹ Reference 2.1.7
² Reference 2.1.8

Table 2.1.12

DESIGN CHARACTERISTICS FOR THORIA RODS IN D-1 THORIA ROD CANISTERS

PARAMETER	MPC-68 or MPC-68F
Cladding Type	Zircaloy-(Zr)
Composition	98.2 wt.% ThO ₂ , 1.8 wt.% UO ₂ with an enrichment of 93.5 wt. % ²³⁵ U
Number of Rods Per Thoria Canister	≤ 18
Decay Heat Per Thoria Canister	≤ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time ≥ 18 years and average burnup ≤ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	≤ 27 kg/canister
Fuel Cladding O.D.	≥ 0.412 inches
Fuel Cladding I.D.	≤ 0.362 inches
Fuel Pellet O.D.	≤ 0.358 inches
Active Fuel Length	≤ 111 inches
Canister Weight	≤ 550 lbs., including Thoria Rods

Table 2.1.13
MPC Fuel Loading Regions

MPC MODEL	REGION 1 FUEL STORAGE LOCATIONS*	REGION 2 FUEL STORAGE LOCATIONS
MPC-24, 24E and 24EF	9, 10, 15, and 16	All Other Locations
MPC-32/32F	7, 8, 12 through 15, 18 through 21, 25, and 26	All Other Locations
MPC-68/68F/68FF	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All Other Locations

*Note: Refer to Figures 1.2.2 through 1.2.4

Table 2.1.14

Soluble Boron Requirements for PWR MPC-24/24E/24EF Fuel Wet Loading and Unloading Operations

MPC MODEL	FUEL ASSEMBLY MAXIMUM AVERAGE ENRICHMENT (wt % ²³⁵ U)	MINIMUM SOLUBLE BORON CONCENTRATION (ppmb)
MPC-24	All fuel assemblies with initial enrichment ¹ less than the prescribed value for soluble boron credit	0
MPC-24	One or more fuel assemblies with an initial enrichment ¹ greater than or equal to the prescribed value for no soluble boron credit AND ≤ 5.0 wt. %	≥ 400
MPC-24E/24EF	All fuel assemblies with initial enrichment ¹ less than the prescribed value for soluble boron credit	0
MPC-24E/24EF	<i>All fuel assemblies classified as intact fuel assemblies and Θ one or more fuel assemblies with an initial enrichment¹ greater than or equal to the prescribed value for no soluble boron credit AND ≤ 5.0 wt. %</i>	≥ 300
MPC-24E/24EF	<i>One or more fuel assemblies classified as damaged fuel or fuel debris and one or more fuel assemblies with initial enrichment > 4.0 wt.% AND ≤ 5.0 wt.%</i>	≥ 600
MPC 32	All fuel assemblies with initial enrichment ≤ 4.1 wt. %	≥ 1900
MPC 32	One or more fuel assemblies with an initial enrichment > 4.1 and ≤ 5.0 wt. %	≥ 2600

¹Refer to Table 2.1.3 for these enrichments.

Table 2.1.15

MINIMUM BORAL ^{10}B LOADING IN NEUTRON ABSORBER PANELS

MPC MODEL	MINIMUM B-10 LOADING (g/cm ²)
MPC-24	0.0267
MPC-24E and MPC-24EF	0.0372
MPC-32/32F	0.0372
MPC-68 and MPC-68FF	0.0372
MPC-68F	0.01

Table 2.1.16

Soluble Boron Requirements for MPC-32 and MPC-32F Wet Loading and Unloading Operations

<i>Fuel Assembly Array/Class</i>	<i>All Intact Fuel Assemblies</i>		<i>One or More Damaged Fuel Assemblies or Fuel Debris</i>	
	<i>Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)</i>	<i>Initial Enrichment ≤ 5.0 wt.% ^{235}U (ppmb)</i>	<i>Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)</i>	<i>Initial Enrichment ≤ 5.0 wt.% ^{235}U (ppmb)</i>
<i>14x14A/B/C/D/E</i>	<i>1,300</i>	<i>1,900</i>	<i>1,500</i>	<i>2,300</i>
<i>15x15A/B/C/G</i>	<i>1,800</i>	<i>2,400</i>	<i>1,900</i>	<i>2,700</i>
<i>15x15D/E/F/H</i>	<i>1,900</i>	<i>2,600</i>	<i>2,100</i>	<i>2,900</i>
<i>16x16A</i>	<i>1,300</i>	<i>1,900</i>	<i>1,500</i>	<i>2,300</i>
<i>17x17A/B/C</i>	<i>1,900</i>	<i>2,600</i>	<i>2,100</i>	<i>2,900</i>

- NOTES:
 1. ALL DIMENSIONS ARE IN INCHES AND ARE APPROXIMATE.
 2. ALL MATERIAL IS STAINLESS STEEL.

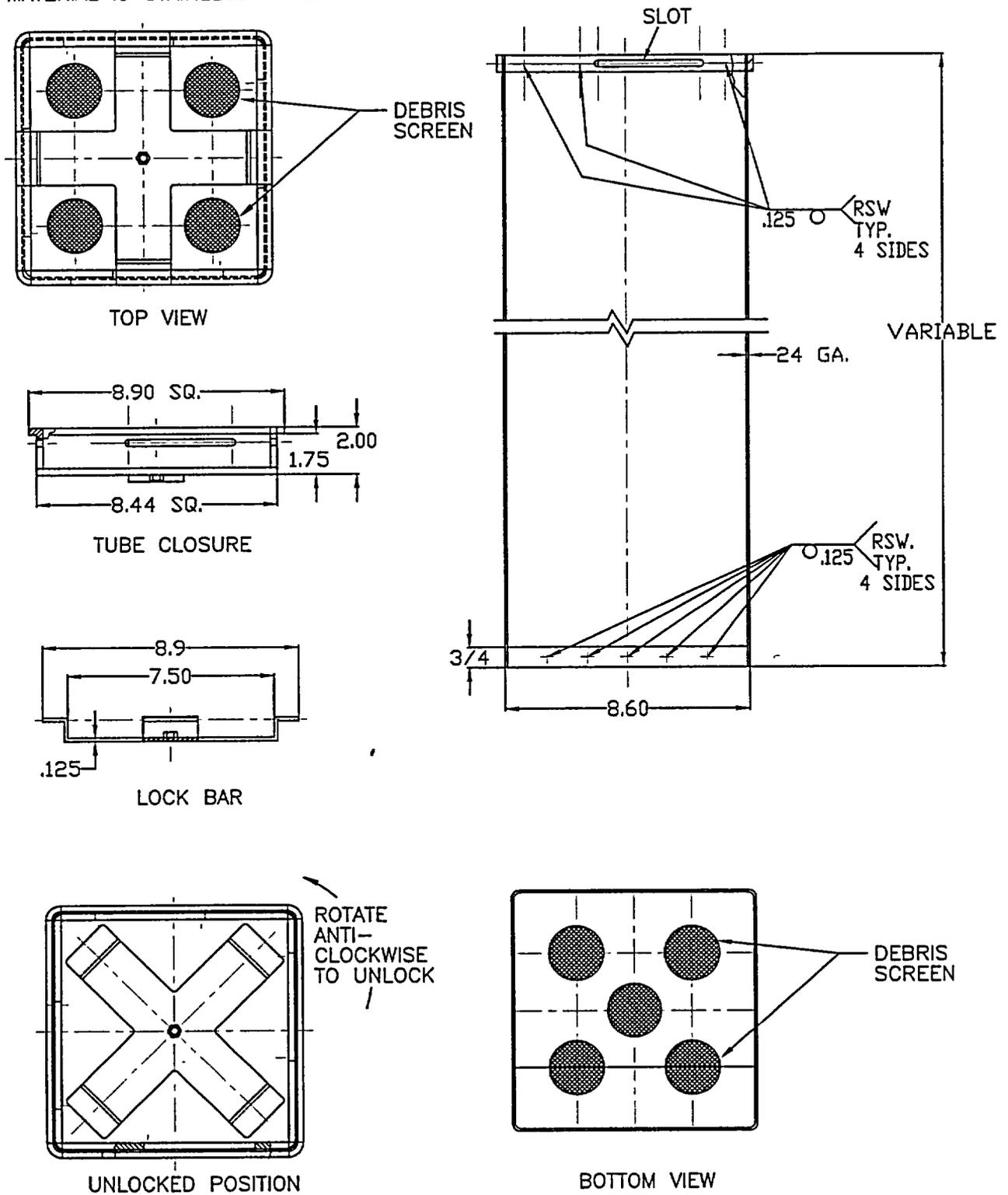
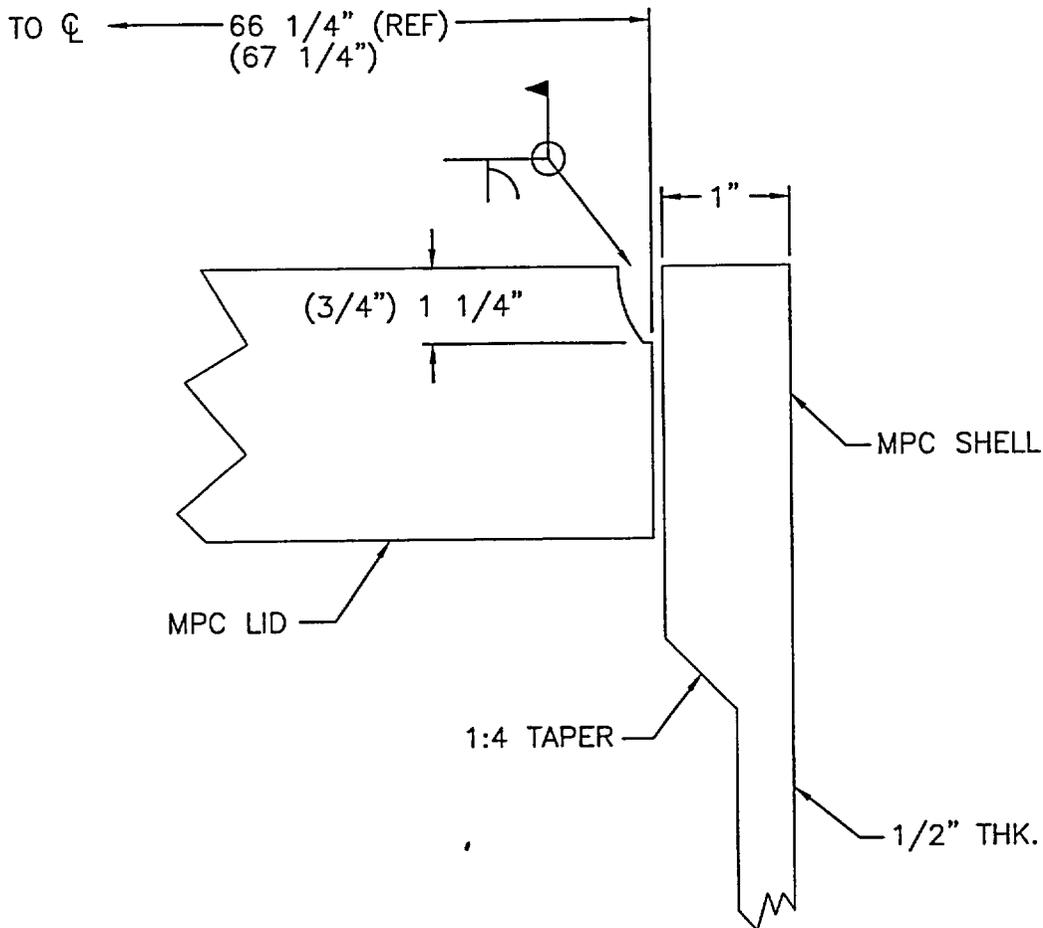


FIGURE 2.1.2D; HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-32/32F



- NOTES: 1. Standard MPC dimensions in parentheses.
 2. Standard MPC shell thickness is 1/2" along its entire length.
 3. Figure is not to scale.

Figure 2.1.9; Fuel Debris MPC ("F" Model)

2.2 HI-STORM 100 DESIGN CRITERIA

The HI-STORM 100 System is engineered for unprotected outside storage for the duration of its design life. Accordingly, the cask system is designed to withstand normal, off-normal, and environmental phenomena and accident conditions of storage. Normal conditions include the conditions that are expected to occur regularly or frequently in the course of normal operation. Off-normal conditions include those infrequent events that could reasonably be expected to occur during the lifetime of the cask system. Environmental phenomena and accident conditions include events that are postulated because their consideration establishes a conservative design basis.

Normal condition loads act in combination with all other loads (off-normal or environmental phenomena/accident). Off-normal condition loads and environmental phenomena and accident condition loads are not applied in combination. However, loads that occur as a result of the same phenomena are applied simultaneously. For example, the tornado winds loads are applied in combination with the tornado missile loads.

In the following subsections, the design criteria are established for normal, off-normal, and accident conditions for storage. Loads that require consideration under each condition are identified and the design criteria discussed. Based on consideration of the applicable requirements of the system, the following loads are identified:

Normal (Long-Term Storage) Condition: Dead Weight, Handling, Pressure, Temperature, Snow

Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Partial Blockage of Air Inlets, Off-Normal Handling of HI-TRAC

Accident Condition: Handling Accident, Tip-Over, Fire, Partial Blockage of MPC Basket Vent Holes, Tornado, Flood, Earthquake, Fuel Rod Rupture, Confinement Boundary Leakage, Explosion, Lightning, Burial Under Debris, 100% Blockage of Air Inlets, Extreme Environmental Temperature

Short-Term Operating Conditions: *This loading condition is defined to accord with ISG-11, Revision 2 guidance [2.0.8]. This includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC transfer cask.*

Each of these conditions and the applicable loads are identified with applicable design criteria established. Design criteria are deemed to be satisfied if the specified allowable limits are not exceeded.

2.2.1 Normal Condition Design Criteria

2.2.1.1 Dead Weight

The HI-STORM⁺ 100 System must withstand the static loads due to the weights of each of its components, including the weight of the HI-TRAC with the loaded MPC atop the storage overpack.

2.2.1.2 Handling

The HI-STORM 100 System must withstand loads experienced during routine handling. Normal handling includes:

- i. vertical lifting and transfer to the ISFSI of the HI-STORM overpack with loaded MPC
- ii. lifting, upending/downending, and transfer to the ISFSI of the HI-TRAC with loaded MPC in the vertical or horizontal position
- iii. lifting of the loaded MPC into and out of the HI-TRAC, HI-STORM, or HI-STAR overpack

The loads shall be increased by 15% to include any dynamic effects from the lifting operations as directed by CMAA #70 [2.2.16].

Handling operations of the loaded HI-TRAC transfer cask or HI-STORM overpack are limited to working area ambient temperatures greater than or equal to 0°F. This limitation is specified to ensure that a sufficient safety margin exists before brittle fracture might occur during handling operations. Subsection 3.1.2.3 provides the demonstration of the adequacy of the HI-TRAC transfer cask and the HI-STORM overpack for use during handling operations at a minimum service temperature of 0° F.

Lifting attachments and devices shall meet the requirements of ANSI N14.6[†] [2.2.3].

2.2.1.3 Pressure

The MPC internal pressure is dependent on the initial volume of cover gas (helium), the volume of fill gas in the fuel rods, the fraction of fission gas released from the fuel matrix, the number of fuel rods assumed to have ruptured, and temperature.

The normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 1% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H³, Kr, and Xe) released in accordance with NUREG-1536.

[†] Yield and ultimate strength values used in the stress compliance demonstration per ANSI N14 6 shall utilize confirmed material test data through either independent coupon testing or material suppliers= CMTR or COC, as appropriate. To ensure consistency between the design and fabrication of a lifting component, compliance with ANSI N14 6 in this FSAR implies that the guidelines of ASME Section III, Subsection NF for Class 3 structures are followed for material procurement and testing, fabrication, and for NDE during manufacturing.

Table 2.2.1 provides the design pressures for the HI-STORM 100 System.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container, it is conservatively assumed that 100% of the fuel rods are ruptured with 100% of the rod fill gas and 30% of the significant radioactive gases (e.g., H₂, Kr, and Xe) released for both normal and off-normal conditions. For PWR assemblies stored with non-fuel hardware, it is assumed that 100% of the gasses in the non-fuel hardware (e.g., BPRAs) is also released. This condition is bounded by the pressure calculation for design basis intact fuel with 100% of the fuel rods ruptured in all of the fuel assemblies. It is shown in Chapter 4 that the accident condition design pressure is not exceeded with 100% of the fuel rods ruptured in all of the design basis fuel assemblies. Therefore, rupture of 100% of the fuel rods in the damaged fuel assemblies or fuel debris will not cause the MPC internal pressure to exceed the accident design pressure.

The MPC internal design pressure under accident conditions is discussed in Subsection 2.2.3.

The HI-STORM overpack and MPC external pressure is a function of environmental conditions which may produce a pressure loading. The normal and off-normal condition external design pressure is set at ambient standard pressure (1 atmosphere).

The HI-STORM overpack is not capable of retaining internal pressure due to its open design, and, therefore, no analysis is required or provided for the overpack internal pressure.

The HI-TRAC is not capable of retaining internal pressure due to its open design and, therefore, ambient and hydrostatic pressures are the only pressures experienced. Due to the thick steel walls of the HI-TRAC transfer cask, it is evident that the small hydrostatic pressure can be easily withstood; no analysis is required or provided for the HI-TRAC internal pressure. However, the HI-TRAC water jacket does experience internal pressure due to the heat-up of the water contained in the water jacket. Analysis is presented in Chapter 3 that demonstrates that the design pressure in Table 2.2.1 can be withstood by the water jacket and Chapter 4 demonstrates by analysis that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device set at the design pressure is provided, which ensures the pressure will not be exceeded.

2.2.1.4 Environmental Temperatures

To evaluate the long-term effects of ambient temperatures on the HI-STORM 100 System, an upper bound value on the annual average ambient temperatures for the continental United States is used. The normal temperature specified in Table 2.2.2 is bounding for all reactor sites in the contiguous United States. The "normal" temperature set forth in Table 2.2.2 is intended to ensure that it is greater than the annual average of ambient temperatures at any location in the continental United States. In the northern region of the U.S., the design basis "normal" temperature used in this FSAR will be exceeded only for brief periods, whereas in the southern U.S., it may be straddled daily in summer months. Inasmuch as the sole effect of the "normal" temperature is on the computed fuel cladding temperature to establish long-term fuel integrity, it should not lie below the time averaged yearly mean for the ISFSI site. Previously licensed cask systems have employed lower "normal" temperatures (viz. 75° F in Docket 72-1007) by utilizing national meteorological data

Likewise, within the thermal analysis, a conservatively assumed soil temperature of the value specified in Table 2.2.2 is utilized to bound the annual average soil temperatures for the continental United States. The 1987 ASHRAE Handbook (HVAC Systems and Applications) reports average earth temperatures, from 0 to 10 feet below grade, throughout the continental United States. The highest reported annual average value for the continental United States is 77° F for Key West, Florida. Therefore, this value is specified in Table 2.2.2 as the bounding soil temperature.

Confirmation of the site-specific annual average ambient temperature and soil temperature is to be performed by the licensee, in accordance with 10CFR72.212. The annual average temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours to establish the normal condition temperatures in the HI-STORM 100 System.

2.2.1.5 Design Temperatures

The ASME Boiler and Pressure Vessel Code (ASME Code) requires that the value of the vessel design temperature be established with appropriate consideration for the effect of heat generation internal or external to the vessel. The decay heat load from the spent nuclear fuel is the internal heat generation source for the HI-STORM 100 System. The ASME Code (Section III, Paragraph NCA-2142) requires the design temperature to be set at or above the maximum through thickness mean metal temperature of the pressure part under normal service (Level A) condition. Consistent with the terminology of NUREG-1536, we refer to this temperature as the "Design Temperature for Normal Conditions". Conservative calculations of the steady-state temperature field in the HI-STORM 100 System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STORM component temperatures at or below the normal condition design temperatures for the HI-STORM 100 System defined in Table 2.2.3.

Maintaining fuel rod cladding integrity is also a design consideration. The ~~maximum-fuel rod peak cladding temperature (PCT) limits for the long-term storage and short-term normal operating conditions meet the intent of the guidance in ISG-11, Revision 2 [2.0.8]. For moderate burnup fuel, the previously licensed PCT limit of 570°C (1058°F) may be used up to certain threshold heat loads provided the fuel cladding hoop stress is shown by analysis to be acceptable (see Section 4.5).~~ ~~calculated by the DCCG (Diffusion Controlled Cavity Growth) methodology outlined in the LLNL report [2.2.14] in accordance with NUREG 1536. However, for conservatism, the PNL methodology outlined in PNL report [2.0.3] produces a lower fuel cladding temperature, which is used to establish the permissible fuel cladding temperature limits, which are used to determine the allowable fuel decay heat load. Maximum fuel rod stainless steel cladding temperature limits recommended in EPRI report [2.2.13] are greater than the long term allowable Zircaloy fuel cladding temperature limits. However, in this FSAR the long term Zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel clad fuel. The short term temperature limits for Zircaloy and stainless steel cladding are taken from references [2.2.15] and [2.2.13], respectively. A detailed description of the maximum fuel rod cladding temperature limits determination is provided in Section 4.3.~~

2.2.1.6 Snow and Ice

The HI-STORM 100 System must be capable of withstanding pressure loads due to snow and ice. ASCE 7-88 (formerly ANSI A58.1) [2.2.2] provides empirical formulas and tables to compute the effective design pressure on the overpack due to the accumulation of snow for the contiguous U.S. and Alaska. Typical calculated values for heated structures such as the HI-STORM 100 System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure loading is set at a level in Table 2.2.8 which bounds the ASCE 7-88 recommendation.

2.2.2 Off-Normal Conditions Design Criteria

As the HI-STORM 100 System is passive, loss of power and instrumentation failures are not defined as off-normal conditions. The off-normal condition design criteria are defined in the following subsections.

A discussion of the effects of each off-normal condition is provided in Section 11.1. Section 11.1 also provides the corrective action for each off-normal condition. The location of the detailed analysis for each event is referenced in Section 11.1.

2.2.2.1 Pressure

The HI-STORM 100 System must withstand loads due to off-normal pressure. The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H³, Kr, and Xe) released in accordance with NUREG-1536. For conservatism, the MPC normal internal design pressure bounds both normal and off-normal conditions. Therefore, the normal and off-normal condition MPC internal pressures are set equal for analysis purposes.

2.2.2.2 Environmental Temperatures

The HI-STORM 100 System must withstand off-normal environmental temperatures. The off-normal environmental temperatures are specified in Table 2.2.2. The lower bound temperature occurs with no solar loads and the upper bound temperature occurs with steady-state insolation. Each bounding temperature is assumed to persist for a duration sufficient to allow the system to reach steady-state temperatures.

Limits on the peaks in the time-varying ambient temperature at an ISFSI site is recognized in the FSAR in the specification of the off-normal temperatures. The lower bound off-normal temperature is defined as the minimum of the 72-hour average of the ambient temperature at an ISFSI site. Likewise, the upper bound off-normal temperature is defined by the maximum of 72-hour average of the ambient temperature. The lower and upper bound off-normal temperatures listed in Table 2.2.2 are intended to cover all ISFSI sites in the continent U.S. The 72-hour average of temperature used in the definition of the off-normal temperature recognizes the considerable thermal inertia of the HI-

STORM 100 storage system which reduces the effect of undulations in instantaneous temperature on the internals of the multi-purpose canister.

2.2.2.3 Design Temperatures

In addition to the normal design temperature, we also define an "off-normal/accident condition temperature" pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61. This is, in effect, the short-term temperature which may exist during a transition state or a transient event (examples of such instances are short-term temperature excursion during canister vacuum drying and backfilling operations (transition state) and fire (transient event)). The off-normal/accident design temperatures of Table 2.2.3 are set down to bound the maximax (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during a short-term event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects during or immediately after, a short-term event.

2.2.2.4 Leakage of One Seal

The HI-STORM 100 System must withstand leakage of one seal in the radioactive material confinement boundary.

The confinement boundary is defined by the MPC shell, baseplate, MPC lid, port cover plates, and closure ring. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root (if more than one weld pass is required) and final weld passes. In addition to liquid penetrant examination, the MPC lid-to-shell weld is leakage tested, hydrostatic tested, and volumetrically examined or multi-pass liquid penetrant examined. The vent and drain port cover plates are leakage tested in addition to the liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary.

Although leakage of one seal is not a credible accident because the MPC confinement boundary does not employ seals, a non-mechanistic leak of the confinement boundary is analyzed as an accident event in Chapter 11.

2.2.2.5 Partial Blockage of Air Inlets

The HI-STORM 100 System must withstand the partial blockage of the overpack air inlets. This event is conservatively defined as a complete blockage of two (2) of the four air inlets. Because the overpack air inlets and outlets are covered by fine mesh steel screens, located 90° apart, and inspected routinely (or alternatively, exit vent air temperature monitored), it is unlikely that all vents could become blocked by blowing debris, animals, etc. during normal and off-normal operations. Two of the air inlets are conservatively assumed to be completely blocked to demonstrate the inherent thermal stability of the HI-STORM 100 System.

2.2.2.6 Off-Normal HI-TRAC Handling

During upending and/or downending of the HI-TRAC 100 or HI-TRAC 125 transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation. The HI-TRAC 125D transfer cask design does not include pocket trunnions. Therefore, the entire load is held by the lifting trunnions.

If the lifting device cables begin to "go slack" while upending or downending the HI-TRAC 100 or HI-TRAC 125, the eccentricity of the pocket trunnions would immediately cause the cask to pivot, restoring tension on the cables. Nevertheless, the pocket trunnions are conservatively analyzed to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to demonstrate that the pocket trunnions in the standard HI-TRAC design possess sufficient strength to support the increased load under this off-normal condition.

2.2.3 Environmental Phenomena and Accident Condition Design Criteria

Environmental phenomena and accident condition design criteria are defined in the following subsections.

The minimum acceptance criteria for the evaluation of the accident conditions are that the MPC confinement boundary maintains radioactive material confinement, the MPC fuel basket structure maintains the fuel contents subcritical, the stored SNF can be retrieved by normal means, and the system provides adequate shielding.

A discussion of the effects of each environmental phenomenon and accident condition is provided in Section 11.2. The consequences of each accident or environmental phenomenon are evaluated against the requirements of 10CFR72.106 and 10CFR20. Section 11.2 also provides the corrective action for each event. The location of the detailed analysis for each event is referenced in Section 11.2.

2.2.3.1 Handling Accident

The HI-STORM 100 System must withstand loads due to a handling accident. Even though the loaded HI-STORM 100 System will be lifted in accordance with approved, written procedures and may use lifting equipment which complies with ANSI N14.6-1993 [2.2.3], certain drop events are considered herein to demonstrate the defense-in-depth features of the design.

The loaded HI-STORM overpack will be lifted so that the bottom of the cask is at a height less than the vertical lift limit (see Table 2.2.8) above the ground. For conservatism, the postulated drop event assumes that the loaded HI-STORM 100 overpack falls freely from the vertical lift limit height before impacting a thick reinforced concrete pad. The deceleration of the cask must be maintained below 45 g's. Additionally, the overpack must continue to suitably shield the radiation emitted from the loaded MPC. The use of lifting devices designed in accordance with ANSI N14.6 having redundant drop protection features to lift the loaded overpack will eliminate the lift height limit. The

lift height limit is dependent on the characteristics of the impacting surface which are specified in Table 2.2.9. For site-specific conditions, which are not encompassed by Table 2.2.9, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the analyses in Appendix 3.A and shall be reviewed by the Certificate Holder.

The loaded HI-TRAC will be lifted so that the lowest point on the transfer cask (i.e., the bottom edge of the cask/lid assemblage) is at a height less than the calculated horizontal lift height limit (see Table 2.2.8) above the ground, when lifted horizontally outside of the reactor facility. For conservatism, the postulated drop event assumes that the loaded HI-TRAC falls freely from the horizontal lift height limit before impact.

Analysis is provided that demonstrates that the HI-TRAC continues to suitably shield the radiation emitted from the loaded MPC, and that the HI-TRAC end plates (top lid and transfer lid for HI-TRAC 100 and HI-TRAC 125 and the top lid and pool lid for HI-TRAC 125D) remain attached. Furthermore, the HI-TRAC inner shell is demonstrated by analysis to not deform sufficiently to hinder retrieval of the MPC. The horizontal lift height limit is dependent on the characteristics of the impacting surface which are specified in Table 2.2.9. For site-specific conditions, which are not encompassed by Table 2.2.9, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the *methodology described in this FSAR analyses in Appendix 3.A* and shall be reviewed by the Certificate Holder. The use of lifting devices designed in accordance with ANSI N14.6 having redundant drop protection features during horizontal lifting of the loaded HI-TRAC outside of the reactor facilities eliminate the need for a horizontal lift height limit.

The loaded HI-TRAC, when lifted in the vertical position outside of the Part 50 facility shall be lifted with devices designed in accordance with ANSI N14.6 and having redundant drop protection features unless a site-specific analysis has been performed to determine a lift height limit. For vertical lifts of HI-TRAC with suitably designed lift devices, a vertical drop is not a credible accident for the HI-TRAC transfer cask and no vertical lift height limit is required to be established. Likewise, while the loaded HI-TRAC is positioned atop the HI-STORM 100 overpack for transfer of the MPC into the overpack (outside the Part 50 facility), the lifting equipment will remain engaged with the lifting trunnions of the HI-TRAC transfer cask or suitable restraints will be provided to secure the HI-TRAC. This ensures that a tip-over or drop from atop the HI-STORM 100 overpack is not a credible accident for the HI-TRAC transfer cask. The design criteria and conditions of use for MPC transfer operations from the HI-TRAC transfer cask to the HI-STORM 100 overpack at a Cask Transfer Facility are specified in the HI-STORM 100 CoC, Appendix B, Section 3.5 and in Subsection 2.3.3.1 of this FSAR.

The loaded MPC is lowered into the HI-STORM or HI-STAR overpack or raised from the overpack using the HI-TRAC transfer cask and a MPC lifting system designed in accordance with ANSI N14.6 and having redundant drop protection features. Therefore, the possibility of a loaded MPC falling freely from its highest elevation during the MPC transfer operations into the HI-STORM or HI-STAR overpacks is not credible.

The magnitude of loadings imparted to the HI-STORM 100 System due to drop events is heavily influenced by the compliance characteristics of the impacted surface. Two "pre-approved" concrete pad designs for storing the HI-STORM 100 System are presented in Table 2.2.9. Other ISFSI pad designs may be used provided the designs are reviewed by the Certificate Holder to ensure that impactive and impulsive loads under accident events such as cask drop and non-mechanistic tip-over are less than the design basis limits when analyzed using the methodologies established in this FSAR.

2.2.3.2 Tip-Over

The free-standing HI-STORM 100 System is demonstrated by analysis to remain kinematically stable under the design basis environmental phenomena (tornado, earthquake, etc.). However, the HI-STORM 100 Overpack and MPC shall also withstand impacts due to a hypothetical tip-over event. The structural integrity of a loaded HI-STORM 100 System after a tip-over onto a reinforced concrete pad is demonstrated by analysis. The cask tip-over is not postulated as an outcome of any environmental phenomenon or accident condition. The cask tip-over is a non-mechanistic event.

The ISFSI pad for deploying a free-standing HI-STORM overpack must possess sufficient structural stiffness to meet the strength limits set forth in the ACI Code selected by the ISFSI owner. At the same time, the pad must be sufficiently compliant such that the maximum deceleration under a tip-over event is below the limit set forth in Table 3.1.2 of this FSAR.

During original licensing for the HI-STAR 100 System, a single set of ISFSI pad and subgrade design parameters (now labeled Set A) was established. Experience has shown that achieving a maximum concrete compressive strength (at 28 days) of 4,200 psi can be difficult. Therefore, a second set of ISFSI pad and subgrade design parameters (labeled Set B) has been developed. The Set B ISFSI parameters include a thinner concrete pad and less stiff subgrade, which allow for a higher concrete compressive strength. Cask deceleration values for all design basis drop and tipover events with the HI-STORM 100 and HI-STORM-100S overpacks have been verified to be less than or equal to the design limit of 45 g's for both sets of ISFSI pad parameters.

The original set and the new set (Set B) of acceptable ISFSI pad and subgrade design parameters are specified in Table 2.2.9. Users may design their ISFSI pads and subgrade in compliance with either parameter Set A or Set B. Alternatively, users may design their site-specific ISFSI pads and subgrade using any combination of design parameters resulting in a structurally competent pad that meets the provisions of ACI-318 and also limits the deceleration of the cask to less than or equal to 45 g's for the design basis drop and tip-over events for the HI-STORM 100 and HI-STORM 100S overpacks. The structural analyses for site-specific ISFSI pad design shall be performed using methodologies consistent with those described in this FSAR, as applicable.

If the HI-STORM 100 cask is deployed in an anchored configuration (HI-STORM 100A), then tip-over of the cask is structurally precluded along with the requirement of target compliance, which warrants setting specific limits on the concrete compressive strength and subgrade Young's Modulus. Rather, at the so-called high seismic sites (ZPAs greater than the limit set forth in the CoC for free standing casks), the ISFSI pad must be sufficiently rigid to hold the anchor studs and maintain the integrity of the fastening mechanism embedded in the pad during the postulated seismic

event. The ISFSI pad must be designed to minimize a physical uplift during extreme environmental event (viz., tornado missile, DBE, etc.). The requirements on the ISFSI pad to render the cask anchoring function under long-term storage are provided in Section 2.0.4.

2.2.3.3 Fire

The possibility of a fire accident near an ISFSI site is considered to be extremely remote due to the absence of significant combustible materials. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM 100 overpack or HI-TRAC transfer cask while it is being moved to the ISFSI.

The HI-STORM 100 System must withstand temperatures due to a fire event. The HI-STORM overpack and HI-TRAC transfer cask fire accidents for storage are conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible transporter fuel. The HI-STORM overpack and HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Table 2.2.8 provides the fire durations for the HI-STORM

100 overpack and HI-TRAC transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475° F in accordance with 10CFR71.73.

The accident condition design temperatures for the HI-STORM 100 System, and the fuel rod cladding limits are specified in Table 2.2.3. The specified fuel cladding temperature limits are based on the short-term temperature limit specified in reports [2.2.13 and 2.2.15].

2.2.3.4 Partial Blockage of MPC Basket Vent Holes

The HI-STORM 100 System is designed to withstand reduction of flow area due to partial blockage of the MPC basket vent holes. As the MPC basket vent holes are internal to the confinement barrier, the only events that could partially block the vents are fuel cladding failure and debris associated with this failure, or the collection of crud at the base of the stored SNF assembly. The HI-STORM 100 System maintains the SNF in an inert environment with fuel rod cladding temperatures below accepted values (Table 2.2.3). Therefore, there is no credible mechanism for gross fuel cladding degradation during storage in the HI-STORM 100. For the storage of damaged BWR fuel assemblies or fuel debris, the assemblies and fuel debris will be placed in damaged fuel containers prior to placement in the MPC. The damaged fuel container is equipped with fine mesh screens which ensure that the damaged fuel and fuel debris will not escape to block the MPC basket vent holes. In addition, each MPC will be loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities reported in an Empire State Electric Energy Research Corporation Report [2.2.6], a layer of crud of conservative depth is assumed to partially block the MPC basket vent holes. The crud depths for the different MPCs are listed in Table 2.2.8.

2.2.3.5 Tornado

The HI-STORM 100 System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STORM 100 System are consistent with NRC Regulatory Guide 1.76 [2.2.7], ANSI 57.9 [2.2.8], and ASCE 7-88 [2.2.2]. Table 2.2.4 provides the wind speeds and pressure drops which the HI-STORM 100 overpack must withstand while maintaining kinematic stability. The pressure drop is bounded by the accident condition MPC external design pressure.

The kinematic stability of the HI-STORM overpack, and continued integrity of the MPC confinement boundary, while within the storage overpack or HI-TRAC transfer cask, must be demonstrated under impact from tornado-generated missiles in conjunction with the wind loadings. Standard Review Plan (SRP) 3.5.1.4 of NUREG-0800 [2.2.9] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile that deforms on impact (large missile); a rigid missile to test penetration resistance (penetrant missile); and a small rigid missile of a size sufficient to pass through any openings in the protective barriers (micro-missile). SRP 3.5.1.4 suggests an automobile for a large missile, a rigid solid steel cylinder for the penetrant missile, and a solid sphere for the small rigid missile, all impacting at 35% of the maximum horizontal wind speed of the design basis tornado. Table 2.2.5 provides the missile data used in the analysis, which is based on the above SRP guidelines. The effects of a large tornado missile are considered to bound the effects of a light general aviation airplane crashing on an ISFSI facility.

During horizontal handling of the loaded HI-TRAC transfer cask outside the Part 50 facility, tornado missile protection shall be provided to prevent tornado missiles from impacting either end of the HI-TRAC. The tornado missile protection shall be designed such that the large tornado missile cannot impact the bottom or top of the loaded HI-TRAC, while in the horizontal position. Also, the missile protection for the top of the HI-TRAC shall be designed to preclude the penetrant missile and micro-missile from passing through the penetration in the HI-TRAC top lid, while in the horizontal position. With the tornado missile protection in place, the impacting of a large tornado missile on either end of the loaded HI-TRAC or the penetrant missile or micro-missile entering the penetration of the top lid is not credible. Therefore, no analyses of these impacts are provided.

2.2.3.6 Flood

The HI-STORM 100 System must withstand pressure and water forces associated with a flood. Resultant loads on the HI-STORM 100 System consist of buoyancy effects, static pressure loads, and velocity pressure due to water velocity. The flood is assumed to deeply submerge the HI-STORM 100 System (see Table 2.2.8). The flood water depth is based on the hydrostatic pressure which is bounded by the MPC external pressure stated in Table 2.2.1.

It must be shown that the MPC does not collapse, buckle, or allow water in-leakage under the hydrostatic pressure from the flood.

The flood water is assumed to be nonstagnant. The maximum allowable flood water velocity is determined by calculating the equivalent pressure loading required to slide or tip over the HI-STORM 100 System. The design basis flood water velocity is stated in Table 2.2.8. Site-specific

safety reviews by the licensee must confirm that flood parameters do not exceed the flood depth, slide, or tip-over forces.

If the flood water depth exceeds the elevation of the top of the HI-STORM overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the HI-STORM 100 Overpack. Blockage of the air inlets is addressed in Subsection 2.2.3.12.

Most reactor sites are hydrologically characterized as required by Paragraph 100.10(c) of 10CFR100 and further articulated in Reg. Guide 1.59, "Design Basis Floods for Nuclear Power Plants" and Reg. Guide 1.102, "Flood Protection for Nuclear Power Plants." It is assumed that a complete characterization of the ISFSI's hydrosphere including the effects of hurricanes, floods, seiches and tsunamis is available to enable a site-specific evaluation of the HI-STORM 100 System for kinematic stability. An evaluation for tsunamis[†] for certain coastal sites should also be performed to demonstrate that sliding or tip-over will not occur and that the maximum flood depth will not be exceeded.

Analysis for each site for such transient hydrological loadings must be made for that site. It is expected that the plant licensee will perform this evaluation under the provisions of 10CFR72.212.

2.2.3.7 Seismic Design Loadings

The HI-STORM 100 System must withstand loads arising due to a seismic event and must be shown not to tip over during a seismic event. Subsection 3.4.7 contains calculations based on conservative static "incipient tipping" calculations which demonstrate static stability. The calculations in Section 3.4.7 result in the values reported in Table 2.2.8, which provide the maximum horizontal zero period acceleration (ZPA) versus vertical acceleration multiplier above which static incipient tipping would occur. This conservatively assumes the peak acceleration values of each of the two horizontal earthquake components and the vertical component occur simultaneously. The maximum horizontal ZPA provided in Table 2.2.8 is the vector sum of two horizontal earthquakes.

For anchored casks, the limit on zero period accelerations is set by the structural capacity of the sector lugs and anchoring studs. Table 2.2.8 provides the limits for HI-STORM 100A for the maximum vector sum of two horizontal earthquake peak ZPA's along with the coincident limit on the vertical ZPA.

2.2.3.8 100% Fuel Rod Rupture

The HI-STORM 100 System must withstand loads due to 100% fuel rod rupture. For conservatism, 100 percent of the fuel rods are assumed to rupture with 100 percent of the fill gas and 30% of the significant radioactive gases (e.g., H³, Kr, and Xe) released in accordance with NUREG-1536. All

† A tsunami is an ocean wave from seismic or volcanic activity or from submarine landslides. A tsunami may be the result of nearby or distant events. A tsunami loading may exist in combination with wave splash and spray, storm surge and tides.

of the fill gas contained in non-fuel hardware, such as Burnable Poison Rod Assemblies (BPRAs) is also assumed to be released in analyzing this event.

2.2.3.9 Confinement Boundary Leakage

No credible scenario has been identified that would cause failure of the confinement system. To demonstrate the overall safety of the HI-STORM 100 System, the largest test leakage rate for the confinement boundary plus 50% for conservatism is assumed as the maximum credible confinement boundary leakage rate and 100 percent of the fuel rods are assumed to have failed. Under this accident condition, doses to an individual located at the boundary of the controlled area are calculated.

2.2.3.10 Explosion

The HI-STORM 100 System must withstand loads due to an explosion. The accident condition MPC external pressure and overpack pressure differential specified in Table 2.2.1 bounds all credible external explosion events. There are no credible internal explosive events since all materials are compatible with the various operating environments, as discussed in Section 3.4.1. The MPC is composed of stainless steel, ~~Boral~~ *neutron absorber material*, and, *prior to CoC Amendment 2, possibly optional* aluminum alloy 1100 *heat conduction elements*, ~~all of which have a long proven history of use in fuel pools at nuclear power plants~~. For these materials there is no credible cause for an internal explosive event.

2.2.3.11 Lightning

The HI-STORM 100 System must withstand loads due to lightning. The effect of lightning on the HI-STORM 100 System is evaluated in Chapter 11.

2.2.3.12 Burial Under Debris

The HI-STORM 100 System must withstand burial under debris. Such debris may result from floods, wind storms, or mud slides. Mud slides, blowing debris from a tornado, or debris in flood water may result in duct blockage, which is addressed in Subsection 2.2.3.13. The thermal effects of burial under debris on the HI-STORM 100 System is evaluated in Chapter 11. Siting of the ISFSI pad shall ensure that the storage location is not located near shifting soil. Burial under debris is a highly unlikely accident, but is analyzed in this FSAR.

2.2.3.13 100% Blockage of Air Inlets

For conservatism, this accident is defined as a complete blockage of all four bottom air inlets. Such a blockage may be postulated to occur during accident events such as a flood or tornado with blowing debris. The HI-STORM 100 System must withstand the temperature rise as a result of 100% blockage of the air inlets and outlets. The fuel cladding temperature must be shown to remain below the short term temperature limit specified in Table 2.2.3.

2.2.3.14 Extreme Environmental Temperature

The HI-STORM 100 System must withstand extreme environmental temperatures. The extreme accident level temperature is specified in Table 2.2.2. The extreme accident level temperature occurs with steady-state insolation. This temperature is assumed to persist for a duration sufficient to allow the system to reach steady-state temperatures. The HI-STORM 100 Overpack and MPC have a large thermal inertia. Therefore, this temperature is assumed to persist over three days (3-day average).

2.2.3.15 Bounding Hydraulic, Wind, and Missile Loads for HI-STORM 100A

In the anchored configuration, the HI-STORM 100A System is clearly capable of withstanding much greater lateral loads than a free-standing overpack. Coastal sites in many areas of the world, particularly the land mass around the Pacific Ocean, may be subject to severe fluid inertial loads. Several publications [2.2.10, 2.2.11] explain and quantify the nature and source of such environmental hazards.

It is recognized that a lateral fluid load may also be accompanied by an impact force from a fluid borne missile (debris). Rather than setting specific limits for these loads on an individual basis, a limit on the static overturning base moment on the anchorage is set. This bounding overturning moment is given in Table 2.2.8 and is set at a level that ensures that structural safety margins on the sector lugs and on the anchor studs are essentially equal to the structural safety margins of the same components under the combined effect of the net horizontal and vertical seismic load limits in Table 2.2.8. The ISFSI owner bears the responsibility to establish that the lateral hydraulic, wind, and missile loads at his ISFSI site do not yield net overturning moments, when acting separately or together, that exceed the limit value in Table 2.2.8. If loadings are increased above those values for free-standing casks, their potential effect on the other portions of the cask system must be considered.

2.2.4 Applicability of Governing Documents

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition, with Addenda through 1997 [2.2.1], is the governing code for the structural design of the MPC, the metal structure of the HI-STORM 100 overpack, and the HI-TRAC transfer cask, *except for sections V and LX*. The MPC enclosure vessel and fuel basket are designed in accordance with Section III, Subsections NB Class 1 and NG Class 1, respectively. The metal structure of the overpack and the HI-TRAC transfer cask are designed in accordance with Section III, Subsection NF Class 3. The ASME Code is applied to each component consistent with the function of the component.

ACI 349 is the governing code for the plain concrete in the HI-STORM 100 overpack. ACI 318-95 is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited during structural analysis. Appendix 1.D provides the sections of ACI 349 and ACI 318-95 applicable to the plain concrete.

Table 2.2.6 provides a summary of each structure, system and component (SSC) of the HI-STORM 100 System that is identified as important to safety, along with its function and governing Code. Some components perform multiple functions and in those cases, the most restrictive Code is

applied. In accordance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components", and according to importance to safety, components of the HI-STORM 100 System are classified as A, B, C, or NITS (not important to safety) in Table 2.2.6. Section 13.1 provides the criteria used to classify each item. The classification of necessary auxiliary equipment is provided in Table 8.1.6.

Table 2.2.7 lists the applicable governing Code for material procurement, design, fabrication and inspection of the components of the HI-STORM 100 System. The ASME Code section listed in the design column is the section used to define allowable stresses for structural analyses.

Table 2.2.15 lists the ~~exceptions~~ *alternatives* to the ASME Code for the HI-STORM 100 System and the justification for those ~~exceptions~~ *alternatives*.

The MPC *enclosure vessel and certain fuel basket designs* utilized in the HI-STORM 100 System *are* identical to the MPC *components* described in the SARs for the HI-STAR 100 System for storage (Docket 72-1008) and transport (Docket 71-9261). To avoid unnecessary repetition of the large numbers of stress analyses, ~~attention is directed in this document~~ *refers to those SARs, as applicable, i*to establish that the MPC loadings for storage in the HI-STORM 100 System do not exceed those computed in the ~~referenced applications~~ *HI-STAR documents*. Many of the loadings in the HI-STAR applications envelope the HI-STORM loadings on the MPC, and, therefore, a complete re-analysis of the MPC is not provided in the FSAR. *Certain individual MPC analyses may have been required to license a particular MPC fuel basket design for HI-STORM that was not previously licensed for HI-STAR. These unique analyses are summarized in the appropriate location in this FSAR.*

Table 2.2.16 provides a summary comparison between the loading elements. Table 2.2.16 shows that most of the loadings remain unchanged and several are less than the HI-STAR loading conditions. In addition to the magnitude of the loadings experienced by the MPC, the application of the loading must also be considered. Therefore, it is evident from Table 2.2.16 that the MPC stress limits can be ascertained to be qualified a priori if the HI-STAR analyses and the thermal loadings under HI-STORM storage are not more severe compared to previously analyzed HI-STAR conditions. In the analysis of each of the normal, off-normal, and accident conditions, the effect on the MPC is evaluated and compared to the corresponding condition analyzed in the HI-STAR 100 System SARs [2.2.4 and 2.2.5]. If the HI-STORM loading is greater than the HI-STAR loading or the loading is applied differently, the analysis of its effect on the MPC is evaluated in Chapter 3.

2.2.5 Service Limits

In the ASME Code, plant and system operating conditions are commonly referred to as normal, upset, emergency, and faulted. Consistent with the terminology in NRC documents, this FSAR utilizes the terms normal, off-normal, and accident conditions.

The ASME Code defines four service conditions in addition to the Design Limits for nuclear components. They are referred to as Level A, Level B, Level C, and Level D service limits, respectively. Their definitions are provided in Paragraph NCA-2142.4 of the ASME Code. The four levels are used in this FSAR as follows:

- a. Level A Service Limits: Level A Service Limits are used to establish allowables for normal condition load combinations.
- b. Level B Service Limits: Level B Service Limits are used to establish allowables for off-normal condition load combinations.
- c. Level C Service Limits: Level C Service Limits are not used.
- d. Level D Service Limits: Level D Service Limits are used to establish allowables for accident condition load combinations.

The ASME Code service limits are used in the structural analyses for definition of allowable stresses and allowable stress intensities. Allowable stresses and stress intensities for structural analyses are tabulated in Chapter 3. These service limits are matched with normal, off-normal, and accident condition loads combinations in the following subsections.

The MPC confinement boundary is required to meet Section III, Class 1, Subsection NB stress intensity limits. Table 2.2.10 lists the stress intensity limits for the Levels A, B, C, and D service limits for Class 1 structures extracted from the ASME Code (1995 Edition). The limits for the MPC fuel basket, required to meet the stress intensity limits of Subsection NG of the ASME Code, are listed in Table 2.2.11. Table 2.2.12 lists allowable stress limits for the steel structure of the HI-STORM overpack and HI-TRAC which are analyzed to meet the stress limits of Subsection NF, Class 3. Only service levels A, B, and D requirements, normal, off-normal, and accident conditions, are applicable.

2.2.6 Loads

Subsections 2.2.1, 2.2.2, and 2.2.3 describe the design criteria for normal, off-normal, and accident conditions, respectively. Table 2.2.13 identifies the notation for the individual loads that require consideration. The individual loads listed in Table 2.2.13 are defined from the design criteria. Each load is assigned a symbol for subsequent use in the load combinations.

The loadings listed in Table 2.2.13 fall into two broad categories; namely, (i) those that primarily affect kinematic stability, and (ii) those that produce significant stresses. The loadings in the former category are principally applicable to the overpack. Tornado wind (W'), earthquake (E), and tornado-borne missile (M) are essentially loadings which can destabilize a cask. Analyses reported in Chapter 3 show that the HI-STORM 100 overpack structure will remain kinematically stable under these loadings. Additionally, for the missile impact case (M), analyses that demonstrate that the overpack structure remains unbreached by the postulated missiles are provided in Chapter 3.

Loadings in the second category produce global stresses that must be shown to comply with the stress intensity or stress limits, as applicable. The relevant loading combinations for the fuel basket, the MPC, the HI-TRAC and the HI-STORM overpack are different because of differences in their function. For example, the fuel basket does not experience a pressure loading because it is not a pressure vessel. The specific load combination for each component is specified in Subsection 2.2.7.

2.2.7 Load Combinations

To demonstrate compliance with the design requirements for normal, off-normal, and accident conditions of storage, the individual loads, identified in Table 2.2.13, are combined into load combinations. In the formation of the load combinations, it is recognized that the number of combinations requiring detailed analyses is reduced by defining bounding loads. Analyses performed using bounding loads serve to satisfy the requirements for analysis of a multitude of separately identified loads in combination.

For example, the values established for internal and external pressures (P_i and P_o) are defined such that they bound other surface-intensive loads, namely snow (S), tornado wind (W'), flood (F), and explosion (E'). Thus, evaluation of pressure in a load combination established for a given storage condition enables many individual load effects to be included in a single load combination.

Table 2.2.14 identifies the combinations of the loads that are required to be considered in order to ensure compliance with the design criteria set forth in this chapter. Table 2.2.14 presents the load combinations in terms of the loads that must be considered together. A number of load combinations are established for each ASME Service Level. Within each loading case, there may be more than one analysis that is required to demonstrate compliance. Since the breakdown into specific analyses is most applicable to the structural evaluation, the identification of individual analyses with the applicable loads for each load combination is found in Chapter 3. Table 3.1.3 through 3.1.5 define the particular evaluations of loadings that demonstrate compliance with the load combinations of Table 2.2.14.

For structural analysis purposes, Table 2.2.14 serves as an intermediate classification table between the definition of the loads (Table 2.2.13 and Section 2.2) and the detailed analysis combinations (Tables 3.1.3 through 3.1.5).

Finally, it should be noted that the load combinations identified in NUREG-1536 are considered as applicable to the HI-STORM 100 System. The majority of load combinations in NUREG-1536 are directed toward reinforced concrete structures. Those load combinations applicable to steel structures are directed toward frame structures. As stated in NUREG-1536, Page 3-35 of Table 3-1, "Table 3-1 does not apply to the analysis of confinement casks and other components designed in accordance with Section III of the ASME B&PV Code." Since the HI-STORM 100 System is a metal shell structure, with concrete primarily employed as shielding, the load combinations of NUREG-1536 are interpreted within the confines and intent of the ASME Code.

2.2.8 Allowable Stresses

The stress intensity limits for the MPC confinement boundary for the design condition and the service conditions are provided in Table 2.2.10. The MPC confinement boundary stress intensity limits are obtained from ASME Code, Section III, Subsection NB. The stress intensity limits for the MPC fuel basket are presented in Table 2.2.11 (governed by Subsection NG of Section III). The steel structure of the overpack and the HI-TRAC meet the stress limits of Subsection NF of ASME Code, Section III for plate and shell components. Limits for the Level D condition are obtained from

Appendix F of ASME Code, Section III for the steel structure of the overpack. The ASME Code is not applicable to the HI-TRAC transfer cask for accident conditions, service level D conditions. The HI-TRAC transfer cask has been shown by analysis to not deform sufficiently to apply a load to the MPC, have any shell rupture, or have the top lid, pool lid, or transfer lid (as applicable) detach.

The following definitions of terms apply to the tables on stress intensity limits; these definitions are the same as those used throughout the ASME Code:

- S_m : Value of Design Stress Intensity listed in ASME Code Section II, Part D, Tables 2A, 2B and 4
- S_y : Minimum yield strength at temperature
- S_u : Minimum ultimate strength at temperature

Table 2.2.1

DESIGN PRESSURES

Pressure Location	Condition	Pressure (psig)
MPC Internal Pressure	Normal	100
	Off-Normal	100-110
	Accident	200
MPC External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	60
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	10 (differential pressure for 1 second maximum) or 5 (differential pressure steady state)
HI-TRAC Water Jacket	Normal	60
	Off-normal	60
	Accident	N/A (Under accident conditions, the water jacket is assumed to have lost all water thru the pressure relief valves)

Table 2.2.2

ENVIRONMENTAL TEMPERATURES

Condition	Temperature (°F)	Comments
HI-STORM 100 Overpack		
Normal Ambient (Bounding Annual Average)	80	
Normal Soil Temperature (Bounding Annual Average)	77	
Off-Normal Ambient (3-Day Average)	-40 and 100	-40°F with no insolation 100°F with insolation
Extreme Accident Level Ambient (3-Day Average)	125	125°F with insolation starting at steady-state off-normal high environment temperature
HI-TRAC Transfer Cask		
Normal (Bounding Annual Average)	100	
Off-Normal (3-Day Average)	0 and 100	0° F with no insolation 100° F with insolation

Note:

1. Handling operations with the loaded HI-STORM 100 overpack and HI-TRAC transfer cask are limited to working area ambient temperatures greater than or equal to 0°F as specified in Subsection 2.2.1.2 and the Design Features section of Appendix B to the CoC. .

Table 2.2.3 (continued)

DESIGN TEMPERATURES

HI-STORM 100 Component	Normal Condition Design Temp. (Long-Term Events) (° F)	Off-Normal and Accident Condition Temp. Limits (Short-Term Events) [†] (° F)
MPC shell	450	775
MPC basket	725	950
MPC Boral Neutron Absorber	800	950
MPC lid	550	775
MPC closure ring	400	775
MPC baseplate	400	775
MPC Heat Conduction Elements	725	950
HI-TRAC inner shell	400	600
HI-TRAC pool lid/transfer lid	350	700
HI-TRAC top lid	400	700
HI-TRAC top flange	400	700
HI-TRAC pool lid seals	350	N/A
HI-TRAC bottom lid bolts	350	700
HI-TRAC bottom flange	350	700
HI-TRAC top lid neutron shielding	300	300
HI-TRAC radial neutron shield	307	N/A
HI-TRAC radial lead gamma shield	350	600
Remainder of HI-TRAC	350	700
High Burnup CSF Cladding	752	1058
Moderate Burnup CSF Cladding	752 (long-term storage) AND 752 or 1058 for short term operations ^{††}	1058
Zircaloy fuel cladding (5-year cooled) [†]	-691(PWR) -740(BWR)	1058
Zircaloy fuel cladding (6-year cooled) [†]	-676(PWR) -712(BWR)	1058
Zircaloy fuel cladding (7-year cooled) [†]	-635(PWR) -669(BWR)	1058
Zircaloy fuel cladding (10-year cooled) [†]	-625(PWR) -658(BWR)	1058
Zircaloy fuel cladding (15-year cooled) [†]	-614(PWR)	1058

[†] For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the ISFSI fire event, the maximum temperature limit for ASME Section 1 equipment is appropriate (850°F in Code Table 1A).

^{††} See Section 4.5 for discussion of the 1058°F temperature limit during short-term operations.

Table 2.2.3 (continued)

DESIGN TEMPERATURES

HI-STORM 100 Component	Normal Condition Design Temp. (Long-Term Events) (° F)	Off-Normal and Accident Condition Temp. Limits (Short-Term Events) [†] (° F)
cooled) ¹	-646(BWR)	
Zircaloy fuel cladding (5-year cooled) ²	679 (PWR) 740 (BWR)	1058
Zircaloy fuel cladding (6-year cooled) ²	660 (PWR) 712 (BWR)	1058
Zircaloy fuel cladding (7-year cooled) ²	635 (PWR) 669 (BWR)	1058
Zircaloy fuel cladding (10-year cooled) ²	621 (PWR) 658 (BWR)	1058
Zircaloy fuel cladding (15-year cooled) ²	611 (PWR) 646 (BWR)	1058
Overpack outer shell	350	600
Overpack concrete	200	350
Overpack inner shell	350	400
Overpack Lid Top Plate	350 450	550
Remainder of overpack steel structure	350	400

NOTES: 1. Moderate Burnup Fuel
 2. High Burnup Fuel (see Table 4.A.2)

Table 2.2.4

TORNADO CHARACTERISTICS

Condition	Value
Rotational wind speed (mph)	290
Translational speed (mph)	70
Maximum wind speed (mph)	360
Pressure drop (psi)	3.0

Table 2.2.5

TORNADO-GENERATED MISSILES

Missile Description	Mass (kg)	Velocity (mph)
Automobile	1800	126
Rigid solid steel cylinder (8 in. diameter)	125	126
Solid sphere (1 in. diameter)	0.22	126

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
MPC^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Confinement	Shell	A	ASME Section III; Subsection NB	Alloy X ⁽⁵⁾	See Appendix 1.A	NA	NA
Confinement	Baseplate	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Lid	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Closure Ring	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Port Cover Plates	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Basket Cell Plates	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Boral Neutron Absorber	A	Non-code	NA	NA	NA	Aluminum/SS
Shielding	Drain and Vent Shield Block	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Shielding	Plugs for Drilled Holes	NITS	Non-code	SA 193B8 (or equivalent)	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Alloy X	See Appendix 1.A	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
 - 5) For details on Alloy X material, see Appendix 1.A.
 - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength ≥ 75 ksi, yield strength ≥ 30 ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
MPC^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Sheathing	A	Non-code	Alloy X	See Appendix 1.A	Aluminum/SS	NA
Structural Integrity	Shims	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Angled Plates)	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Structural Form	Basket Supports (Flat Plates)	NITS	Non-Code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug	C	NUREG-0612	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug Baseplate	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Bolt	NITS	Non-code	A193-B8 (or equiv.)	Per ASME Section II	NA	NA
Structural Integrity	Upper Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Stainless Steel. See Note 6	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Vent Shield Block Spacer	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Vent and Drain Tube	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	NA
Operations	Vent & Drain Cap	C	Non-code	S/S	Per ASME Section	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety
 - 5) For details on Alloy X material, see Appendix 1.A.
 - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength ≥ 75 ksi, yield strength ≥ 30 ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
MPC^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
					II		
Operations	Vent & Drain Cap Seal Washer	NITS	Non-code	Aluminum	NA	NA	Aluminum/SS
Operations	Vent & Drain Cap Seal Washer Bolt	NITS	Non-code	Aluminum	NA	NA	NA
Operations	Reducer	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Drain Line	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Damaged Fuel Container	C	ASME Section III; Subsection NG	Primarily 304 S/S	See Appendix 1.A	NA	NA
Operations	Drain Line Guide Tube	NITS	Non-code	S/S	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
 - 5) For details on Alloy X material, see Appendix 1.A.
 - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength ≥ 75 ksi, yield strength ≥ 30 ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
OVERPACK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl (if dissimilar)
Shielding	Radial Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Block Ring (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Shielding	Lid Shield Ring (100S) and Shield Block Shell (100S)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Shielding	Shield Block Shell (100)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Shielding	Pedestal Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Lid Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Shell (eliminated from design 6/01)	B	See Note 6	SA516-70	See Table 3.3.2	NA	NA
Shielding	Shield Block	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Gamma Shield Cross Plates & Tabs	C	Non-code	SA240-304	NA	NA	NA
Structural Integrity	Baseplate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.3	See Note 5	NA
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Concrete Form	Pedestal Shell	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~, the *Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
 - 6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
OVERPACK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Concrete Form	Pedestal Plate (100) Pedestal Baseplate (100S)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Table 3.3.2	NA
Structural Integrity	Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inlet Vent Vertical & Horizontal Plates	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Thermal	Exit Vent Horizontal Plate (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Thermal	Exit Vent Vertical/Side Plate	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Plate, including shear ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Top Plate, including shear ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
 - 6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
OVERPACK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Lid Stud & Nut	B	ASME Section II	SA564-630 or SA 193-B7 (stud) SA 194-2H (nut)	See Table 3.3.4	Threads to have cadmium coating (or similar lubricant for corrosion protection)	NA
Structural Integrity	100S Lid Washer	B	Non-Code	SA240-304	Per ASME Section II	NA	NA
Structural Integrity	Bolt Anchor Block	B	ASME Section III; Subsection NF ANSI N14.6	SA350-LF3 Or SA203E	See Table 3.3.3	See Note 5	NA
Structural Integrity	Channel	B	ASME Section III; Subsection NF	SA516-70 (galvanized) or SA240-304	See Table 3.3.2 or Table 3.3.1	See Note 5 (not applicable to SA240-304)	NA
Structural Integrity	Channel Mounts	B	ASME Section III; Subsection NF	A36 or equivalent	Per ASME Section II	See Note 5	NA
Structural Integrity	Pedestal Platform	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Storage Marking Nameplate	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Exit Vent Screen Sheet	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Drain Pipe	NITS	Non-code	C/S or S/S	NA	See Note 5	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13, the Holtec QA Program. NITS stands for Not Important to Safety.
 - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
 - 6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
OVERPACK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Operations	Exit & Inlet Screen Frame	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Temperature Element & Associated Temperature Monitoring Equipment	B	Non-code	NA	NA	NA	NA
Operations	Screen	NITS	Non-code	Mesh Wire	NA	NA	NA
Operations	Paint	NITS	Non-code	Thermaline 450 or equivalent	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13, the Holtec QA Program~~. NITS stands for Not Important to Safety.
 - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
 - 6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl (if dissimilar)
Shielding	Radial Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Pool Lid Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Top Lid Shielding	B	Non-code	Holtite	NA	NA	NA
Shielding	Plugs for Lifting Holes	NITS	Non-code	C/S	NA	NA	
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial Ribs	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket Enclosure Shell Panels (HI-TRAC 100 and 125)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket Enclosure Shell Panels (HI-TRAC 125D)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Flange	B	ASME Section III; Subsection NF	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lower Water Jacket Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA

- Notes.
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Pool Lid Outer Ring	B	ASME Section III; Subsection NF	SA516-70 or SA 203E or SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Pool Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Outer Ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Inner Ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Fill Port Plugs	C	ASME Section III; Subsection NF	Carbon Steel	See Table 3.3.2	See Note 5	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 *the Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Pool Lid Bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA
Structural Integrity	Lifting Trunnion Block	B	ASME Section III; Subsection NF ANSI 14.6	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lifting Trunnion	A	ANSI N14.6	SB637 (N07718)	See Table 3.3.4	NA	NA
Structural Integrity	Pocket Trunnion (HI-TRAC 100 and HI-TRAC 125 only)	B	ASME Section III; Subsection NF ANSI 14.6	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Dowel Pins	B	ASME Section III; Subsection NF	SA564-630	See Table 3.3.4	NA	SA350-LF3
Structural Integrity	Water Jacket End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Pool Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Lifting Block	C	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Bottom Flange Gussets (HI-TRAC 125D only)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	NA	NA
Operations	Top Lid Stud or bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Operations	Top Lid Nut	B	ASME Section III; Subsection NF	SA194-2H	NA	NA	NA
Operations	Pool Lid Gasket	NITS	Non-code	Elastomer	NA	NA	NA
Operations	Lifting Trunnion End Cap (HI-TRAC 100 and HI-TRAC 125 only)	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	End Cap Bolts (HI-TRAC 100 and HI-TRAC 125 only)	NITS	Non-code	SA193-B7	See Table 3.3.4	NA	NA
Operations	Drain Pipes	NITS	Non-code	SA106	NA	NA	NA
Operations	Drain Bolt	NITS	Non-code	SA193-B7	See Table 3.3.4	NA	NA
Operations	Couplings, Valves and Vent Plug	NITS	Non-code	Commercial	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER LID (HI-TRAC 100 and HI-TRAC 125 ONLY) ^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Shielding	Side Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Door Lead Shield	B	Non-code	Lead	NA	NA	
Shielding	Door Shielding	B	Non-code	Holtite	NA	NA	NA
Structural Integrity	Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Intermediate Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lead Cover Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lead Cover Side Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Middle Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Wheel Housing	B	ASME Section III; Subsection NF	SA516-70 (SA350-LF3)	See Table 3.3.2 (Table 3.3.3)	See Note 5	NA
Structural Integrity	Door Interface Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Side Plate	B	ASME Section III;	SA516-70	See Table 3.3.2	See Note 5	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Top surface of doors to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
 HI-TRAC TRANSFER LID (HI-TRAC 100 and HI-TRAC 125 ONLY) ^(1,2)

Primary Function	Component ⁽³⁾	Safety Class ⁽⁴⁾	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
			Subsection NF				
Structural Integrity	Wheel Shaft	C	ASME Section III; Subsection NF	SA 193-B7	36 (yield)	See Note 5	NA
Structural Integrity	Lid Housing Stiffener	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Lock Bolt	B	ASME Section III; Subsection NB	SA193-B7	See Table 3.3.4	NA	NA
Structural Integrity	Door End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lifting Lug and Pad	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Wheel Track	C	ASME Section III; Subsection NF	SA-36	36 (yield)	See Note 5	NA
Operations	Door Handle	NITS	Non-code	C/S or S/S	NA	See Note 5	NA
Operations	Door Wheels	NITS	Non-code	Forged Steel	NA	NA	NA
Operations	Door Stop Block	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Door Stop Block Bolt	C	Non-code	SA193-B7	See Table 3.3.4	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
 - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
 - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
 - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
 - 5) All external surfaces to be painted with Carboline 890. Top surface of doors to be painted with Thermaline 450.

Table 2.2.7

HI-STORM 100 ASME BOILER AND PRESSURE VESSEL CODE APPLICABILITY

HI-STORM 100 Component	Material Procurement	Design	Fabrication	Inspection
Overpack steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3200	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5350, NF-5360 and Section V
Anchor Studs for HI-STORM 100A	Section II, Section III, Subsection NF, NF-2000*	Section III, Subsection NF, NF-3300	NA	NA
MPC confinement boundary	Section II, Section III, Subsection NB, NB-2000	Section III, Subsection NB, NB-3200	Section III, Subsection NB, NB-4000	Section III, Subsection NB, NB-5000 and Section V
MPC fuel basket	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V
HI-TRAC Trunnions	Section II, Section III, Subsection NF, NF-2000	ANSI 14.6	Section III, Subsection NF, NF-4000	See Chapter 9
MPC basket supports	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V
HI-TRAC steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3300	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5360 and Section V
Damaged fuel container	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V
Overpack concrete	ACI 349 as specified by Appendix 1.D	ACI 349 and ACI 318-95 as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D

* Except impact testing shall be determined based on service temperature and material type.

Table 2.2.8

ADDITIONAL DESIGN INPUT DATA FOR NORMAL, OFF-NORMAL, AND
ACCIDENT CONDITIONS

Item	Condition	Value
Snow Pressure Loading (lb./ft ²)	Normal	100
Constriction of MPC Basket Vent Opening By Crud Settling (Depth of Crud, in.)	Accident	0.85 (MPC-68) 0.36 (MPC-24 and MPC-32)
Cask Environment During the Postulated Fire Event (Deg. F)	Accident	1475
HI-STORM Overpack Fire Duration (seconds)	Accident	217
HI-TRAC Transfer Cask Fire Duration (minutes)	Accident	4.8
Maximum submergence depth due to flood (ft)	Accident	125
Flood water velocity (ft/s)	Accident	15
Interaction Relation for Horizontal & Vertical ZPA (Zero Period Acceleration) for HI-STORM	Accident	$G_H + 0.53G_V = 0.53^{\dagger\dagger}$ (HI-STORM 100 and 100S) $G_H = 2.12; G_V = 1.5$ (HI-STORM 100A)
Net Overturning Moment at base of HI-STORM 100A (ft-lb)	Accident	18.7×10^6
HI-STORM 100 Overpack Vertical Lift Height Limit (in.)	Accident	$11^{\dagger\dagger\dagger}$ (HI-STORM 100 and 100S), OR By Users (HI-STORM 100A)
HI-TRAC Transfer Cask Horizontal Lift Height Limit (in.)	Accident	$42^{\dagger\dagger\dagger}$

^{††} See Subsection 3.4.7.1 for definition of G_H and G_V . The coefficient of 0.53 may be increased based on testing described in Subsection 3.4.7.1

^{†††} For ISFSI and subgrade design parameter Sets A and B. Users may also develop a site-specific lift height limit.

Table 2.2.9

EXAMPLES OF ACCEPTABLE ISFSI PAD DESIGN PARAMETERS

PARAMETER	PARAMETER SET "A" †	PARAMETER SET "B"
Concrete thickness, t_p , (inches)	≤ 36	≤ 28
Concrete Compressive Strength (at 28 days), f_c' , (psi)	$\leq 4,200$	$\leq 6,000$ psi
Reinforcement Top and Bottom (both directions)	Reinforcing bar shall be 60 ksi Yield Strength ASTM Material	Reinforcing bar shall be 60 ksi Yield Strength ASTM Material
Subgrade Effective Modulus of Elasticity ^{††} (measured prior to ISFSI pad installation), E, (psi)	$\leq 28,000$	$\leq 16,000$

NOTE: A static coefficient of friction of ≥ 0.53 between the ISFSI pad and the bottom of the overpack shall be verified by test. The test procedure shall follow the guidelines included in the Sliding Analysis in Subsection 3.4.7.1.

† The characteristics of this pad are identical to the pad considered by Lawrence Livermore Laboratory (see Appendix 3.A).

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

Table 2.2.10
MPC CONFINEMENT BOUNDARY STRESS INTENSITY LIMITS
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220)[†]

STRESS CATEGORY	DESIGN	LEVELS A & B	LEVEL D ^{††}
Primary Membrane, P_m	S_m	N/A ^{†††}	AMIN ($2.4S_m, .7S_u$)
Local Membrane, P_L	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Primary Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
Average Shear Stress ^{††††}	$0.6S_m$	$0.6S_m$	$0.42S_u$

[†] Stress combinations including F (peak stress) apply to fatigue evaluations only.

^{††} Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.

^{†††} No Specific stress limit applicable.

^{††††} Governed by NB-3227.2 or F-1331.1(d).

Table 2.2.11

MPC BASKET STRESS INTENSITY LIMITS
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NG-3220)

STRESS CATEGORY	DESIGN	LEVELS A & B	LEVEL D [†]
Primary Membrane, P_m	S_m	S_m	AMIN ($2.4S_m, .7S_u$) ^{††}
Primary Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Primary Membrane plus Primary Bending plus Secondary	N/A ^{†††}	$3S_m$	N/A

[†] Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.

^{††} Governed by NB-3227 2 or F-1331.1(d).

^{†††} No specific stress intensity limit applicable.

Table 2.2.12
 STRESS LIMITS FOR DIFFERENT
 LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE OVERPACK AND HI-TRAC
 (ELASTIC ANALYSIS PER NF-3260)

STRESS CATEGORY	DESIGN + LEVEL A	SERVICE CONDITION	
		LEVEL B	LEVEL D [†]
Primary Membrane, P_m	S	1.33S	AMAX ($1.2S_y$, $1.5S_m$) but $< .7S_u$
Primary Membrane, P_m , plus Primary Bending, P_b	1.5S	1.995S	150% of P_m
Shear Stress (Average)	0.6S	0.6S	$< 0.42S_u$

Definitions:

S = Allowable Stress Value for Table 1A, ASME Section II, Part D.

S_m = Allowable Stress Intensity Value from Table 2A, ASME Section II, Part D

S_u = Ultimate Strength

[†] Governed by Appendix F, Paragraph F-1332 of the ASME Code, Section III.

Table 2.2.13

NOTATION FOR DESIGN LOADINGS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS

NORMAL CONDITION	
LOADING	NOTATION
Dead Weight	D
Handling Loads	H
Design Pressure (Internal) [†]	P _i
Design Pressure (External) [†]	P _o
Snow	S
Operating Temperature	T
OFF-NORMAL CONDITION	
Loading	Notation
Off-Normal Pressure (Internal) ^{II}	P _i
Off-Normal Pressure (External) ^{II}	P _o
Off-Normal Temperature	T'
Off-Normal HI-TRAC Handling	H'

[†] Internal Design Pressure P_i bounds the normal and off-normal condition internal pressures. External Design Pressure P_o bounds off-normal external pressures. Similarly, Accident pressures P_i^{*} and P_o^{*}, respectively, bound actual internal and external pressures under all postulated environment phenomena and accident events.

Table 2.2.13 (continued)

NOTATION FOR DESIGN LOADINGS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS

ACCIDENT CONDITIONS	
LOADING	NOTATION
Handling Accident	H'
Earthquake	E
Fire	T*
Tornado Missile	M
Tornado Wind	W'
Flood	F
Explosion	E*
Accident Pressure (Internal)	P _i *
Accident Pressure (External)	P _o *

Table 2.2.14
 APPLICABLE LOAD CASES AND COMBINATIONS FOR EACH CONDITION AND COMPONENT^{†, ††}

CONDITION	LOADING CASE	MPC	OVERPACK	HI-TRAC
Design (ASME Code Pressure Compliance)	1	P_i, P_o	N/A	N/A
Normal (Level A)	1	D, T, H, P_i	D, T, H	$D, T^{†††}, H, P_{i, (water\ jacket)}$
	2	D, T, H, P_o	N/A	N/A
Off-Normal (Level B)	1	D, T', H, P_i	D, T', H	$N/A^{†††}$ (H' pocket trunnion)
	2	D, T', H, P_o	N/A	N/A
Accident (Level D)	1	D, T, P_i, H'	D, T, H'	D, T, H'
	2	D, T^*, P_i^*	N/A	N/A
	3	$D, T, P_o^{††††}$	$D, T, P_o^{††††}$	$D, T, P_o^{††††}$
	4	N/A	$D, T, (E, M, F, W')^{††††}$	$D, T, (M, W')^{††††}$

[†] The loading notations are given in Table 2.2.13. Each symbol represents a loading type and may have different values for different components. The different loads are assumed to be additive and applied simultaneously.

^{††} N/A stands for "Not Applicable."

^{†††} T (normal condition) for the HI-TRAC is 100°F and $P_{i, (water\ jacket)}$ is 60 psig and, therefore, there is no off-normal temperature or load combination because Load Case 1, Normal (Level A), is identical to Load Case 1, Off-Normal (Level B). Only the off-normal handling load on the pocket trunnion is analyzed separately.

^{††††} P_o^* bounds the external pressure due to explosion.

^{†††††} (E, M, F, W') means loads are considered separately in combination with D, T. E and F not applicable to HI-TRAC.

Table 2.2.15

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
<p><i>MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure.</i></p>	<p><i>Subsection NCA</i></p>	<p><i>General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.</i></p>	<p><i>Because the MPC, overpack, and transfer cask are not ASME Code stamped vessels, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the HI-STORM 100 System as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program.</i></p> <p><i>Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.</i></p>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
<i>MPC basket supports and lift lugs</i>	NB-1130	<p><i>NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material.</i></p> <p><i>NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.</i></p>	<i>The MPC basket supports (nonpressure-retaining structural attachment) and lift lugs (nonstructural attachments used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.</i>
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.

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
<i>MPC, MPC basket assembly, HI-STORM overpack, and HI-TRAC transfer cask</i>	<i>NB-3100 NG-3100 NF-3100</i>	<i>Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.</i>	<i>These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.</i>
<i>MPC</i>	<i>NB-3350</i>	<i>NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.</i>	<p><i>Due to MPC basket-to-shell interface requirements, the MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.</i></p> <p><i>From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.</i></p>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
<p><i>MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure</i></p>	<p><i>NB-4120 NG-4120 NF-4120</i></p>	<p><i>NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.</i></p>	<p><i>In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of lead are not, unless explicitly stated by the Code, defined as heat treatment operations.</i></p> <p><i>For the steel parts in the HI-STORM 100 System components, the duration for which a part exceeds the off-normal temperature limit defined in Chapter 2 of the FSAR shall be limited to 24 hours in a particular manufacturing process (such as the HI-TRAC lead pouring process).</i></p>
<p><i>MPC, HI-STORM overpack steel structure, HI-TRAC transfer cask steel structure</i></p>	<p><i>NB-4220 NF-4220</i></p>	<p><i>Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.</i></p>	<p><i>The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.</i></p>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
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.
<i>MPC basket assembly</i>	NG-4420	<i>NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length</i>	<i>Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis.</i>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
			<i>From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).</i>
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT examination alone is used, at a minimum, it will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be hydrostatically tested as defined in Chapter 8. Accessibility for leakage inspections preclude a Code compliant hydrostatic test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
			<p>PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded.</p> <p>The inspection process <i>results</i>, including <i>relevant</i> findings (indications), shall be made a permanent part of the user'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 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 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.</p>
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.	<i>The</i> HI-STORM 100 System <i>is</i> 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.

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
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.
<i>MPC Basket Assembly</i>	<i>NG-4420</i>	<i>NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.</i>	<i>Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis.</i>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
			<i>From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).</i>
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System will is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package will is to be in conformance with Holtec's QA program.
Overpack Steel Structure	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.
HI-TRAC Steel Structure	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.

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
Overpack Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member thickness of 1" or greater is loaded to transmit loads in the through thickness direction.	The large margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large and warrant an exemption. The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have <i>large safety factors during lifting</i> a minimum safety factor of greater than 12 during lifting. Likewise, the top lid plate to lid shell weld has a <i>large structural margin under the inertia loads imposed during a non-mechanistic tipover event. safety factor greater than 6 under 45g's.</i>
Overpack Steel Structure	NF-3256 NF-3266	Provides requirements for welded joints.	Welds for which no structural credit is taken are identified as "Non-NF" welds in the design drawings by an "***". These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX. <i>Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.</i>
HI-STORM Overpack and HI-TRAC Transfer Cask	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting	<i>These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The overpack and transfer cask bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances</i>

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
			<p><i>between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).</i></p> <p><i>Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.</i></p>

Table 2.2.16

COMPARISON BETWEEN HI-STORM MPC LOADINGS WITH HI-STAR MPC LOADINGS[†]

Loading Condition	Difference Between MPC Loadings Under HI-STAR and HI-STORM Conditions
Dead Load	Unchanged
Design Internal Pressure (normal, off-normal, & accident)	Unchanged
Design External Pressure (normal, off-normal, & accident)	HI-STORM normal and off-normal external pressure is ambient which is less than the HI-STAR 40 psig. The accident external pressure is unchanged.
Thermal Gradient (normal, off-normal, & accident)	Determined by analysis in Chapters 3 and 4
Handling Load (normal)	Unchanged
Earthquake (accident)	Inertial loading increased less than 0.1g's (for free-standing overpack designs)
Handling Load (accident)	HI-STORM vertical and horizontal deceleration loadings are less than those in HI-STAR, but the HI-STORM cavity inner diameter is different and therefore the horizontal loading on the MPC is analyzed in Chapter 3.

[†] HI-STAR MPC loadings are those specified in HI-STAR SARs under Docket Numbers 71-9261 and 72-1008.

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

The HI-STORM 100 System is engineered to provide for the safe long-term storage of spent nuclear fuel (SNF). The HI-STORM 100 will withstand all normal, off-normal, and postulated accident conditions without any uncontrolled release of radioactive material or excessive radiation exposure to workers or members of the public. Special considerations in the design have been made to ensure long-term integrity and confinement of the stored SNF throughout all cask operating conditions. The design considerations which have been incorporated into the HI-STORM 100 System to ensure safe long-term fuel storage are:

1. The MPC confinement barrier is an enclosure vessel designed in accordance with the ASME Code, Subsection NB with confinement welds inspected by radiography (RT) or ultrasonic testing (UT). Where RT or UT is not possible, a redundant closure system is provided with field welds which are hydrostatically tested, helium leakage tested and inspected by the liquid penetrant method.
2. The MPC confinement barrier is surrounded by the HI-STORM overpack which provides for the physical protection of the MPC.
3. The HI-STORM 100 System is designed to meet the requirements of storage while maintaining the safety of the SNF.
4. The SNF once initially loaded in the MPC does not require opening of the canister for repackaging to transport the SNF.
5. The decay heat emitted by the SNF is rejected from the HI-STORM 100 System through passive means. No active cooling systems are employed.

It is recognized that a rugged design with large safety margins is essential, but that is not sufficient to ensure acceptable performance over the service life of any system. A carefully planned oversight and surveillance plan which does not diminish system integrity but provides reliable information on the effect of passage of time on the performance of the system is essential. Such a surveillance and performance assay program will be developed to be compatible with the specific conditions of the licensee's facility where the HI-STORM 100 System is installed. The general requirements for the acceptance testing and maintenance programs are provided in Chapter 9. Surveillance requirements are specified in the Technical Specifications in Appendix A to the CoC.

The structures, systems, and components of the HI-STORM 100 System designated as important to safety are identified in Table 2.2.6. Similar categorization of structures, systems, and components, which are part of the ISFSI, but not part of the HI-STORM 100 System, will be the responsibility of

the 10CFR72 licensee. For HI-STORM 100A, the ISFSI pad is designated ITS, Category C as discussed in Subsection 2.0.4.1.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity which the HI-STORM 100 System must confine originates from the spent fuel assemblies and, to a lesser extent, the contaminated water in the fuel pool. This radioactivity is confined by multiple confinement barriers.

Radioactivity from the fuel pool water is minimized by preventing contact, removing the contaminated water, and decontamination.

An inflatable seal in the annular gap between the MPC and HI-TRAC, and the elastomer seal in the HI-TRAC pool lid prevent the fuel pool water from contacting the exterior of the MPC and interior of the HI-TRAC while submerged for fuel loading. The fuel pool water is drained from the interior of the MPC and the MPC internals are dried. The exterior of the HI-TRAC has a painted surface which is decontaminated to acceptable levels. Any residual radioactivity deposited by the fuel pool water is confined by the MPC confinement boundary along with the spent nuclear fuel.

The HI-STORM 100 System is designed with several confinement barriers for the radioactive fuel contents. Intact fuel assemblies have cladding which provides the first boundary preventing release of the fission products. Fuel assemblies classified as damaged fuel or fuel debris are placed in a damaged fuel container which restricts the release of fuel debris. The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, shell, lid, closure ring, and port cover plates.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, internal change, or external natural phenomena. The MPC is designed to endure normal, off-normal, and accident conditions of storage with the maximum decay heat loads without loss of confinement. Designed in accordance with the ASME Code, Section III, Subsection NB, with certain NRC-approved alternatives, the MPC confinement boundary provides assurance that there will be no release of radioactive materials from the cask under all postulated loading conditions. Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity. Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7. MPC field weld examinations, hydrostatic testing, and helium leak testing are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 9, to verify the confinement boundary.

2.3.2.2 Cask Cooling

To facilitate the passive heat removal capability of the HI-STORM 100, several thermal design criteria are established for normal and off-normal conditions. They are as follows:

- The heat rejection capacity of the HI-STORM 100 System is deliberately understated by conservatively determining the design basis fuel. The decay heat value in Table 2.1.6 is developed by computing the decay heat from the design basis fuel assembly which produces the highest heat generation rate for a given burnup. Additional margin is built into the calculated cask cooling rate by using a design basis fuel assembly which offers maximum resistance to the transmission of heat (minimum thermal conductivity).
- The MPC fuel basket is formed by a honeycomb structure of stainless steel plates with full-length edge-welded intersections, which allows the unimpaired conduction of heat.
- The MPC confinement boundary ensures that the helium atmosphere inside the MPC is maintained during normal, off-normal, and accident conditions of storage and transfer. The MPC confinement boundary maintains the helium confinement atmosphere below the design temperatures and pressures stated in Table 2.2.3 and Table 2.2.1, respectively.
- The MPC thermal design maintains the fuel rod cladding temperatures below the values stated in Chapter 4 such that fuel cladding is not degraded during the long term storage period.
- The HI-STORM is optimally designed with cooling vents and an MPC to overpack annulus which maximize air flow, while providing superior radiation shielding. The vents and annulus allow cooling air to circulate past the MPC removing the decay heat.

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Equipment

Design criteria for the HI-STORM 100 System are described in Section 2.2. The HI-STORM 100 System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized outside of the reactor facility's 10CFR Part 50 structures may be broken down into two broad categories, namely Important to Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", provides guidance for the determination of a component's safety classification. Certain

ancillary equipment (such as trailers, rail cars, skids, portable cranes, transporters, or air pads) are not required to be designated as ITS for most ISFSI implementations, if the HI-STORM 100 is designed to withstand the failure of these components.

The listing and ITS designation of ancillary equipment in Table 8.1.6 follows NUREG/CR-6407. ITS ancillary equipment utilized in activities that occur outside the 10CFR Part 50 structure shall be engineered to meet all functional, strength, service life, and operational safety requirements to ensure that the design and operation of the ancillary equipment is consistent with the intent of this Safety Analysis Report. The design for these components shall consider the following information, as applicable:

1. Functions and boundaries of the ancillary equipment
2. The environmental conditions of the ISFSI site, including tornado-borne missile, tornado wind, seismic, fire, lightning, explosion, ambient humidity limits, flood, tsunami and any other environmental hazards unique to the site.
3. Material requirements including impact testing requirements
4. Applicable codes and standards
5. Acceptance testing requirements
6. Quality assurance requirements
7. Foundation type and permissible loading
8. Applicable loads and load combinations
9. Pre-service examination requirements
10. In-use inspection and maintenance requirements
11. Number and magnitude of repetitive loading significant to fatigue
12. Insulation and enclosure requirements (on electrical motors and machinery)
13. Applicable Reg. Guides and NUREGs.
14. Welding requirements
15. Painting, marking, and identification requirements
16. Design Report documentation requirements
17. Operational and Maintenance (O&M) Manual information requirements

All design documentation shall be subject to a review, evaluation, and safety assessment process in accordance with the provisions of the QA program described in Chapter 13.

Users may effectuate the inter-cask transfer of the MPC between the HI-TRAC transfer cask and either the HI-STORM 100 or the HI-STAR 100 overpack in a location of their choice, depending upon site-specific needs and capabilities. For those users choosing to perform the MPC inter-cask transfer outside of a facility governed by the regulations of 10 CFR Part 50 (e.g., fuel handling or reactor building), a Cask Transfer Facility (CTF) is required. The CTF is a stand-alone facility located on-site, near the ISFSI that incorporates or is compatible with lifting devices designed to lift a loaded or unloaded HI-TRAC transfer cask, place it atop the overpack, and transfer the loaded MPC to or

from the overpack. The detailed design criteria which must be followed for the design and operation of the CTF are set down in Paragraphs A through R below.

The inter-cask transfer operations consist of the following potential scenarios of MPC transfer:

- Transfer between a HI-TRAC transfer cask and a HI-STORM overpack
- Transfer between a HI-TRAC transfer cask and a HI-STAR 100 overpack

In both scenarios, the standard design HI-TRAC is mounted on top of the overpack (HI-STAR 100, HI-STORM 100, HI-STORM 100S) and the MPC transfer is carried out by opening the transfer lid doors located at the bottom of the HI-TRAC transfer cask and by moving the MPC vertically to the cylindrical cavity of the recipient cask. For the HI-TRAC 125D design, the MPC transfer is carried out in a similar fashion, except that there is no transfer lid involved - the pool lid is removed while the transfer cask is mounted atop the HI-STORM overpack with the HI-STORM mating device located between the two casks (see Figure 1.2.18). However, the devices utilized to lift the HI-TRAC cask to place it on the overpack and to vertically transfer the MPC may be of stationary or mobile type.

The specific requirements for the CTF employing stationary and mobile lifting devices are somewhat different. The requirements provided in the following specification for the CTF apply to both types of lifting devices, unless explicitly differentiated in the text.

1. General Specifications:

- i. The cask handling functions which may be required of the Cask Transfer Facility include:
 - a. Upending and downending of a HI-STAR 100 overpack on a flatbed rail car or other transporter (see Figure 2.3.1 for an example).
 - b. Upending and downending of a HI-TRAC transfer cask on a heavy-haul transfer trailer or other transporter (see Figure 2.3.2 for an example)
 - c. Raising and placement of a HI-TRAC transfer cask on top of a HI-STORM 100 overpack for MPC transfer operations (see Figure 2.3.3 for an example of the cask arrangement with the standard design HI-TRAC transfer cask. The HI-TRAC 125D design would include the mating device and no transfer lid).
 - d. Raising and placement of a HI-TRAC transfer cask on top of a HI-STAR 100 overpack for MPC transfer operations (see Figure 2.3.4

for an example of the cask arrangement with the standard design HI-TRAC transfer cask. The HI-TRAC 125D design would include the mating device and no transfer lid).

- e. MPC transfer between the HI-TRAC transfer cask and the HI-STORM overpack.
- f. MPC transfer between the HI-TRAC transfer cask and the HI-STAR 100 overpack.

ii. Other Functional Requirements:

The CTF should possess facilities and capabilities to support cask operations such as :

- a. Devices and areas to support installation and removal of the HI-STORM overpack lid.
- b. Devices and areas to support installation and removal of the HI-STORM 100 overpack vent shield block inserts.
- c. Devices and areas to support installation and removal of the HI-STAR 100 closure plate.
- d. Devices and areas to support installation and removal of the HI-STAR 100 transfer collar.
- e. Features to support positioning and alignment of the HI-STORM overpack and the HI-TRAC transfer cask.
- f. Features to support positioning and alignment of the HI-STAR 100 overpack and the HI-TRAC transfer cask.
- g. Areas to support jacking of a loaded HI-STORM overpack for insertion of a translocation device underneath.
- h. Devices and areas to support placement of an empty MPC in the HI-TRAC transfer cask or HI-STAR 100 overpack
- i. Devices and areas to support receipt inspection of the MPC, HI-TRAC transfer cask, HI-STORM overpack, and HI-STAR

overpack.

- j. Devices and areas to support installation and removal of the HI-STORM mating device (HI-TRAC 125D only).

iii. Definitions:

The components of the CTF covered by this specification consist of all structural members, lifting devices, and foundations which bear all or a significant portion of the dead load of the transfer cask or the multi-purpose canister during MPC transfer operations. The definitions of key terms not defined elsewhere in this FSAR and used in this specification are provided below. The following terms are used to define key components of the CTF.

- Connector Brackets: The mechanical part used in the load path which connects to the cask trunnions. A fabricated weldment, slings, and turnbuckles are typical examples of connector brackets.
- CTF structure: The CTF structure is the stationary, anchored portion of the CTF which provides the required structural function to support MPC transfer operations, including lateral stabilization of the HI-TRAC transfer cask and, if required, the overpack, to protect against seismic events. The MPC lifter, if used in the CTF design, is integrated into the CTF structure (see Lifter Mount).
- HI-TRAC lifter(s): The HI-TRAC lifter is the mechanical lifting device, typically consisting of jacks or hoists, that is utilized to lift a loaded or unloaded HI-TRAC to the required elevation in the CTF so that it can be mounted on the overpack.[†]
- Lifter Mount: A beam-like structure (part of the CTF structure) that supports the HI-TRAC and MPC lifter(s).
- Lift Platform: The lift platform is the intermediate structure that transfers the vertical load of the HI-TRAC transfer cask to the HI-TRAC lifters.
- Mobile crane: A mobile crane is a device defined in ASME B30.5-1994, Mobile and Locomotive Cranes. A mobile crane may be used in lieu of the HI-TRAC lifter and/or an MPC lifter provided all requirements set forth in this subsection are satisfied.

[†] The term overpack is used in this specification as a generic term for the HI-STAR 100 and the various HI-STORM overpacks.

- MPC lifter: The MPC lifter is a mechanical lifting device, typically consisting of jacks or hoists, that is utilized to vertically transfer the MPC between the HI-TRAC transfer cask and the overpack.
- Pier: The portion of the reinforced concrete foundation which projects above the concrete floor of the CTF.
- Single-Failure-Proof (SFP): A single-failure-proof handling device is one wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria given in of NUREG-0612.
- Translocation Device: A low vertical profile device used to laterally position an overpack such that the bottom surface of the overpack is fully supported by the top surface of the device. Typical translocation devices are air pads and Hillman rollers.

iv. Important to Safety Designation:

All components and structures which comprise the CTF shall be given an ITS category designation in accordance with a written procedure which is consistent with NUREG/CR-6407 and Chapter 13 of this FSAR.

B. Environmental and Design Conditions

- i. Lowest Service Temperature (LST): The LST for the CTF is 0°F (consistent with the specification for the HI-TRAC transfer cask in Subsection 3.1.2.3).
- ii. Snow and Ice Load, S: The CTF structure shall be designed to withstand the dead weight of snow and ice for unheated structures as set forth in ASCE 7-88 [2.2.2] for the specific ISFSI site.
- iii. Tornado Missile, M, and Tornado Wind, W': The tornado wind and tornado-generated missile data applicable to the HI-STORM 100 System (Tables 2.2.4 and 2.2.5) will be used in the design of the CTF structure unless existing site design basis data or a probabilistic risk assessment (PRA) for the CTF site with due consideration of short operation durations indicates that a less severe tornado missile impact or wind loading on the CTF structure can be postulated. The PRA analysis can be performed in the manner of the EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology

Computer Code Manual". USNRC Reg. Guide 1.117 and Section 2.2.3 of NUREG-800 may be used for guidance in

establishing the appropriate tornado missile and wind loading for the CTF structure.

The following additional clarifications apply to the large tornado missile (4,000 lb. automobile) in Tables 2.2.4 and 2.2.5 in the CTF structure analysis:

- The missile has a planform area of 20 sq. ft. and impact force characteristics set forth in Appendix 3.AN (Section 3.AN.3).
- The large missile can strike the CTF structure in any orientation up to an elevation of 15 feet.

If the site tornado missile data developed by the ISFSI owner suggests that tornado missiles of greater kinetic energies than that postulated in this FSAR (Table 2.2.4 and 2.2.5) should be postulated for CTF during its use, then the integrity analysis of the CTF structure shall be carried out under the site-specific tornado missiles. This situation would also require the HI-TRAC transfer cask and the overpack to be re-evaluated under the provisions of 10CFR72.212 and 72.48.

The wind speed specified in this FSAR (Tables 2.2.4 and 2.2.5), likewise, shall be evaluated for their applicability to the site. Lower or higher site-specific wind velocity, compared to the design basis values cited in this FSAR shall be used if justified by appropriate analysis, which may include PRA.

Intermediate penetrant missile and small missiles postulated in this FSAR are not considered to be a credible threat to the functional integrity of the CTF structure and, therefore, need not be considered.

- iv. Flood: The CTF will be assumed to be flooded to the highest elevation for the CTF facility determined from the local meteorological data. The flood velocity shall be taken as the largest value defined for the ISFSI site.
- v. Lightning: Meteorological data for the region surrounding the ISFSI site shall be used to specify the applicable lightning input to the CTF structure for personnel safety evaluation purposes.

- vi. Water Waves (Tsunami, Y): Certain coastal CTF sites may be subject to sudden, short duration waves of water, denoted in the literature by various terms, such as tsunami. If the applicable meteorological data for the CTF site indicates the potential of such water-borne loadings on the CTF structure, then such a loading,

with due consideration of the short duration of CTF operations, shall be defined for the CTF structure.

- vii. Design Basis Earthquake (DBE), E: The DBE event applicable to the CTF facility pursuant to 10CFR100, Appendix A, shall be specified. The DBE should be specified as a set of response spectra or acceleration time-histories for use in the CTF structural and impact consequence analyses.

- viii. Design Temperature: All material properties used in the stress analysis of the CTF structure shall utilize a reference design temperature of 150°F.

C. Heavy Load Handling:

- i. Apparent dead load, D*: The dead load of all components being lifted shall be increased in the manner set forth in Subsection 3.4.3 to define the Apparent Dead Load, D*.
- ii. NUREG-0612 Conformance:

The Connector Bracket, HI-TRAC lifter, and MPC lifter shall comply with the guidance provided in NUREG-0612 (1980) for single failure proof devices. Where the geometry of the lifting device is different from the configurations contemplated by NUREG-0612, the following exceptions apply:

- A1. Mobile cranes at the CTF shall conform to the guidelines of Section 5.1.1 of NUREG-0612 with the exception that mobile cranes shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes", in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes". The mobile crane used shall have a minimum safety factor of two over the allowable load table for the crane in accordance with Section 5.1.6(1)(a) of NUREG-0612, and shall be capable of stopping and holding the load during a DBE event.

B2. Section 5.1.6(2) of NUREG-0612 specifies that new cranes should be designed to meet the requirements of NUREG-0554. For mobile cranes, the guidance of Section 5.1.6(2) of NUREG-0612 does not apply.

iii. Defense-in-Depth Measures:

- a. The lift platform and the lifter mount shall be designed to ensure that the stresses produced under the apparent dead load, D^* , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.
- b. The CTF structure shall be designed to ensure that the stresses produced in it under the apparent dead load, D^* , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.
- c. Maximum deflection of the lift platform and the lifter mount under the apparent dead load shall comply with the limits set forth in CMAA-70.
- d. When the HI-TRAC transfer cask is stacked on the overpack, HI-TRAC shall be either held by the lifting device or laterally restrained by the CTF structure. Furthermore, when the HI-TRAC transfer cask is placed atop the overpack, the overpack shall be laterally restrained from uncontrolled movement, if required by the analysis specified in Subsection 2.3.3.1.N.
- e. The design of the lifting system shall ensure that the lift platform (or lift frame) is held horizontal at all times and that the symmetrically situated axial members are symmetrically loaded.
- f. In order to minimize occupational radiation exposure to ISFSI personnel, design of the MPC lifting attachment (viz., sling) should not require any human activity inside the HI-TRAC cylindrical space.
- g. The HI-TRAC lifter and MPC lifter shall possess design features to avoid side-sway of the payload during lifting operations.
- h. The lifter (HI-TRAC and MPC) design shall ensure that any electrical malfunction in the motor or the power supply will not lead to an

uncontrolled lowering of the load.

- i. The kinematic stability of HI-TRAC or HI-STORM standing upright in an unrestrained configuration (if such a condition exists during the use of the CTF) shall be analytically evaluated and ensured under all postulated extreme environmental phenomena loadings for the CTF facility.

iv. Shielding Surety:

The design of the HI-TRAC and MPC lifters shall preclude the potential for the MPC to be removed, completely or partially, from the cylindrical space formed by the HI-TRAC and the underlying overpack.

v. Specific Requirements for Mobile Cranes:

A mobile crane, if used in the CTF in the role of the HI-TRAC lifter or MPC lifter is governed in part by ANSI/ASME N45.2.15 with technical requirements specified in ANSI B30.5 (1994).

When lifting the MPC from an overpack to the HI-TRAC transfer cask, limit switches or load limiters shall be set to ensure that the mobile crane is prevented from lifting loads in excess of 110% of the loaded MPC weight.

An analysis of the consequences of a potential MPC vertical drop which conforms to the guidelines of Appendix A to NUREG-0612 shall be performed. The analysis shall demonstrate that a postulated drop would not result in the MPC experiencing a deceleration in excess of its design basis deceleration specified in this FSAR.

- vi. Lift Height Limitation: The HI-TRAC lift heights shall be governed by the Technical Specifications.
- vii. Control of Side Sway: Procedures shall provide provisions to ensure that the load is lifted essentially vertically with positive control of the load. Key cask lifting and transfer procedures, as determined by the user, should be reviewed by the Certificate Holder before their use.

D. Loads and Load Combinations for the CTF Structure

The applicable loadings for the CTF have been summarized in paragraph B in the preceding. A stress analysis of the CTF structure shall be performed to demonstrate compliance with the Subsection NF stress limits for Class 3 linear structures for the service condition germane to each load combination. Table 2.3.2 provides the load combinations (the symbols in Table 2.3.2 are defined in the preceding text and in Table 2.2.13).

E. Materials and Failure Modes

- i. Acceptable Materials and Material Properties: All materials used in the design of the CTF shall be ASTM approved or equal, consistent with the ITS category of the part. Reinforced concrete, if used, shall comply with the provisions of ACI 318 (89). The material property and allowable stress values for all steel structurals shall be taken from the ASME and B&PV Code, Section II, wherever such data is available; otherwise, the data provided in the ASTM standards shall be used.
- ii. Brittle Fracture: All structural components in the CTF structure and the lift platform designated as primary load bearing shall have an NDTT equal to 0°F or lower (consistent with the ductile fracture requirements for ASME Section III, Subsection NF, Class 3 structures).
- iii. Fatigue: Fatigue failure modes of primary structural members in the CTF structure whose failure may result in uncontrolled lowering of the HI-TRAC transfer cask or the MPC (critical members) shall be evaluated. A minimum factor of safety of 2 on the number of permissible loading cycles on the critical members shall apply.
- iv. Buckling: For all critical members in the CTF structure (defined above), potential failure modes through buckling under axial compression shall be considered. The margin of safety against buckling shall comply with the provisions of ASME Section III, Subsection NF, for Class 3 linear structures.

F. CTF Pad

A reinforced concrete pad in conformance with the specification for the ISFSI pad set forth in this FSAR (see Table 2.2.9) may be used in the region of the CTF where the overpack and HI-TRAC are stacked for MPC transfer. Alternatively, the pad may be designed using the guidelines of ACI-318(89).

G. Miscellaneous Components

Hoist rings, turnbuckles, slings, and other appurtenances which are in the load path during heavy load handling at the CTF shall be single-failure-proof.

H. Structural Welds

All primary structural welds in the CTF structure shall comply with the specifications of ASME Section III for Class 3 NF linear structures.

I. Foundation

The design of the CTF structure foundation and piers, including load combinations, shall be in accordance with ACI-318(89).

J. Rail Access

The rail lines that enter the Cask Transfer Facility shall be set at grade level with no exposed rail ties or hardware other than the rail itself.

K. Vertical Cask Crawler/Translocation Device Access (If Required)

- i. The cask handling bay in the CTF shall allow access of a vertical cask crawler or translocation device carrying a transfer cask or overpack. The building floor shall be equipped with a smooth transition to the cask travel route such that the vertical cask crawler tracks do not have to negotiate sharp lips or slope transitions and the translocation devices have a smooth transition. Grading of exterior aprons shall be no more than necessary to allow water drainage.
- ii. If roll-up doors are used, the roll up doors shall have no raised threshold that could damage the vertical cask crawler tracks (if a crawler is used).
- iii. Exterior aprons shall be of a material that will not be damaged by the vertical cask crawler tracks, if a crawler is used.

L. Facility Floor

- i. The facility floor shall be sufficiently flat to allow optimum handling of casks with a translocation device.

- ii. Any floor penetrations, in areas where translocation device operations may occur, shall be equipped with flush inserts.
- iii. The rails, in areas where translocation device operations may occur shall be below the finish level of the floor. Flush inserts, if necessary, shall be sized for installation by hand.

M. Cask Connector Brackets

- i. Primary lifting attachments between the cask and the lifting platform are the cask connector brackets. The cask connector brackets may be lengthened or shortened to allow for differences in the vehicle deck height of the cask delivery vehicle and the various lifting operations. The connector brackets shall be designed to perform cask lifting, upending and downending functions. The brackets shall be designed in accordance with ANSI N14.6 [Reference 2.2.3] and load tested at 300% of the load applied to them during normal handling.
- ii. The connector brackets shall be equipped with a positive engagement to ensure that the cask lifting attachments do not become inadvertently disconnected during a seismic event and during normal cask handling operations.
- iii. The design of the connector brackets shall ensure that the HI-TRAC transfer cask is fully secured against slippage during MPC transfer operations.

N. Cask Restraint System

A time-history analysis of the stacked overpack/HI-TRAC transfer cask assemblage under the postulated ISFSI Level D events in Table 2.3.2 shall be performed to demonstrate that a minimum margin of safety of 1.1 against overturning or kinematic instability exists and that the CTF structure complies with the applicable stress limits (Table 2.3.2) and that the maximum permissible deceleration loading specified in the FSAR is not exceeded. If required to meet the minimum margin of safety of 1.1, a cask restraining system shall be incorporated into the design of the Cask Transfer Facility to provide lateral restraint to the overpack (HI-STORM or HI-STAR 100).

O. Design Life

The Cask Transfer Facility shall be constructed to have a minimum design life of 40 years.

P. Testing Requirements

In addition to testing recommended in NUREG-0612 (1980), a structural adequacy test of the CTF structure at 125% of its operating load prior to its first use in a cask loading campaign shall be performed. This test should be performed in accordance with the guidance provided in the CMAA Specification 70 [2.2.16].

Q. Quality Assurance Requirements

All components of the CTF shall be manufactured in full compliance with the quality assurance requirements applicable to the ITS category of the component as set forth in *the Holtec QA program*. ~~Chapter 13 of this FSAR.~~

R. Documentation Requirements

i O&M Manual: An Operations and Maintenance Manual shall be prepared which contains, at minimum, the following items of information:

- Maintenance Drawings
- Operating Procedures

ii Design Report: A QA-validated design report documenting full compliance with the provisions of this specification shall be prepared and archived for future reference in accordance with the provisions of ~~Chapter 13 of this FSAR~~ *the Holtec QA program*.

2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STORM 100 System, instrumentation which is important to safety is not necessary. No instrumentation is required or provided for HI-STORM 100 storage operations, other than normal security service instruments and TLDs.

However, in lieu of performing the periodic inspection of the HI-STORM overpack vent screens, temperature elements may be installed in two of the overpack exit vents to continuously monitor the air

temperature. If the temperature elements and associated temperature monitoring instrumentation are used, they shall be designated important to safety as specified in Table 2.2.6.

The temperature elements and associated temperature monitoring instrumentation provided to monitor the air outlet temperature shall be suitable for a temperature range of -40°F to 500°F. At a minimum, the temperature elements and associated temperature monitoring instrumentation shall be calibrated for the temperatures of 32°F (ice point), 212°F (boiling point), and 449°F (melting point of tin) with an accuracy of +/- 4°F.

2.3.4 Nuclear Criticality Safety

The criticality safety criteria stipulates that the effective neutron multiplication factor, k_{eff} , including statistical uncertainties and biases, is less than 0.95 for all postulated arrangements of fuel within the cask under all credible conditions.

2.3.4.1 Control Methods for Prevention of Criticality

The control methods and design features used to prevent criticality for all MPC configurations are the following:

- a. Incorporation of permanent neutron absorbing material (~~Bora~~^{Bora™}) in the MPC fuel basket walls.
- b. Favorable geometry provided by the MPC fuel basket

Additional control methods used to prevent criticality for the MPC-24, MPC-24E, and MPC-24EF (all with higher enriched fuel), and the MPC-32 and MPC-32F are the following:

- a. Loading of PWR fuel assemblies must be performed in water with a minimum boron content as specified in Table 2.1.14.
- b. Prevention of fresh water entering the MPC internals.

Administrative controls specified as Technical Specifications and Approved Contents are provided in Appendices A and B to the CoC, respectively, and shall be used to ensure that fuel placed in the HI-STORM 100 System meets the requirements described in Chapters 2 and 6. All appropriate criticality analyses are presented in Chapter 6.

2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criticality analyses performed in Chapter 6. Because biases and uncertainties are explicitly evaluated in the analysis, it is not necessary to introduce additional contingency for error.

2.3.4.3 Verification Analyses

In Chapter 6, critical experiments are selected which reflect the design configurations. These critical experiments are evaluated using the same calculation methods, and a suitable bias is incorporated in the reactivity calculation.

2.3.5 Radiological Protection

2.3.5.1 Access Control

As required by 10CFR72, uncontrolled access to the ISFSI is prevented through physical protection means. A peripheral fence with an appropriate locking and monitoring system is a standard approach to limit access. The details of the access control systems and procedures, including division of the site into radiation protection areas, will be developed by the licensee (user) of the ISFSI utilizing the HI-STORM 100 System.

2.3.5.2 Shielding

The shielding design is governed by 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive doses in excess of the limits given in Table 2.3.1 for normal, off-normal, and accident conditions.

The objective of shielding is to assure that radiation dose rates at key locations are *as low as practical in order to maintain occupational doses to operating personnel As Low As Reasonably Achievable (ALARA) and to meet the requirements of 10 CFR 72.104 and 10 CFR 106 for dose at the controlled area boundary.* ~~below acceptable levels for these locations.~~ Three locations are of particular interest in the storage mode:

- immediate vicinity of the cask
- restricted area boundary
- controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded overpack are important in consideration of occupational exposure. *Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on the contents of the BWR and PWR MPCs allowed by the CoC.*

Actual dose rates in operation will be lower than those reported in Chapter 5 for the following reasons:

- *The shielding evaluation model has a number of conservatisms, as discussed in Chapter 5.*
- *No single cask will likely contain design basis fuel in each fuel storage location and the full compliment of non-fuel hardware allowed by the CoC.*
- *No single cask will contain fuel and non-fuel hardware at the limiting burnups and cooling times allowed by the CoC.*

Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM 100 System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask user. A design objective for the maximum average radial surface dose rate has been established as 60 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 60 mrem/hr. The average dose rate at the top of the overpack is limited to below 60 mrem/hr. Chapter 5 of this FSAR presents the analyses and evaluations to establish HI-STORM 100 compliance with these design objectives.

Because of the passive nature of the HI-STORM 100 System, human activity related to the system is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 10. Chapter 10 also provides information concerning temporary shielding which may be utilized to reduce the personnel dose during loading, unloading, transfer, and handling operations. The estimated occupational doses for personnel comply with the requirements of 10CFR20.

For the loading and unloading of the HI-STORM overpack with the MPC, three transfer cask designs are provided (i.e., HI-TRAC 125, HI-TRAC 100, and HI-TRAC 125D). The two 125 ton HI-TRAC provide better shielding than the 100 ton HI-TRAC due to the increased shielding thickness and corresponding greater weight. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would normally dictate that the 125 ton HI-TRAC should be used. However, sites may not be capable of utilizing the 125 ton HI-TRAC due to crane capacity limitations, floor loading limitations, or other site-specific considerations. As with other dose reduction-based plant activities, individual users who cannot accommodate the 125 ton HI-TRAC should perform a cost-benefit analysis of the actions (e.g., plant modifications) that would be necessary to use the 125 ton HI-TRAC. The cost of the action(s) would be weighed against the value of the projected reduction in radiation exposure, and a decision made based on each plant's particular ALARA implementation philosophy.

Dose rates at the restricted area and site boundaries shall be in accordance with applicable regulations. Licensees shall demonstrate compliance with 10CFR72.104 and 10CFR72.106 for the actual fuel being stored, the ISFSI storage array, and the controlled area boundary distances.

The analyses presented in Chapters 5, 10, and 11 demonstrate that the HI-STORM 100 System is capable of meeting the above radiation dose limits and design objectives.

2.3.5.3 Radiological Alarm System

There are no credible events which could result in release of radioactive materials or increases in direct radiation above the requirements of 10CFR72.106. In addition, the non-mechanistic release as the result of a hypothetical accident is described in Chapter 7, and results in a dose to an individual at the controlled area boundary of a very small magnitude. Therefore, radiological alarm systems are not necessary.

2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM 100 System. No such materials would be stored within an ISFSI. However, for conservatism we have analyzed a hypothetical fire accident as a bounding condition for HI-STORM 100. An evaluation of the HI-STORM 100 System in a fire accident is discussed in Chapter 11.

Small overpressures may result from accidents involving explosive materials which are stored or transported near the site. Explosion is an accident loading condition considered in Chapter 11.

Table 2.3.1

RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

BOUNDARY OF CONTROLLED AREA (m) (minimum)	100
NORMAL AND OFF-NORMAL CONDITIONS:	
Whole Body (mrem/yr)	25
Thyroid (mrem/yr)	75
Any Other Critical Organ (mrem/yr)	25
DESIGN BASIS ACCIDENT:	
TEDE (rem)	5
DDE + CDE to any individual organ or tissue (other than lens of the eye) (rem)	50
Lens dose equivalent (rem)	15
Shallow dose equivalent to skin or any extremity (rem)	50

Table 2.3.2

Load Combinations[†] and Service Condition Definitions for the CTF Structure

Load Combination	Service Condition for Section III of the ASME Code for Definition of Allowable Stress	Comment
D*	Level A	All primary load bearing members must satisfy Level A stress limits.
D+S	Level A	
D+M ^{††} +W' D+F D+E or D+Y	Level D	Factor of safety against overturning shall be ≥ 1.1

[†] The reinforced concrete portion of the CTF structure shall also meet factored combinations of the above loads set forth in ACI-318(89).

^{††} This load may be reduced or eliminated based on a PRA for the CTF site.

2.4 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STORM 100 System. The HI-STORM 100 System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site.

The MPC is being licensed for transport off-site in the HI-STAR 100 dual-purpose cask system (Reference Docket No. 71-9261). No further handling of the SNF stored in the MPC is required prior to transport to a licensed centralized storage facility or licensed repository.

The MPC which holds the SNF assemblies is engineered to be suitable as a waste package for permanent interment in a deep Mined Geological Disposal System (MGDS). The materials of construction permitted for the MPC are known to be highly resistant to severe environmental conditions. No carbon steel, paint, or coatings are used or permitted in the MPC. Therefore, the SNF assemblies stored in the MPC should not need to be removed. However, to ensure a practical, feasible method to defuel the MPC, the top of the MPC is equipped with sufficient gamma shielding and markings locating the drain and vent locations to enable semiautomatic (or remotely actuated) boring of the MPC lid to provide access to the MPC vent and drain. The circumferential welds of the MPC lid closure ring can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the overpack consists of steel and concrete rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the overpack, and the latter reused for storage of other MPCs.

In either case, the overpack would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

If the MPC needs to be opened and separated from the SNF before the fuel is placed into the MGDS, the MPC interior metal surfaces will be decontaminated using existing mechanical or chemical methods. This will be facilitated by the MPC fuel basket and interior structures' smooth metal surfaces designed to minimize crud traps. After the surface contamination is removed, the MPC radioactivity will be diminished significantly, allowing near-surface burial or secondary applications at the licensee's facility.

It is also likely that both the overpack and MPC, or extensive portions of both, can be further decontaminated to allow recycle or reuse options. After decontamination, the only radiological hazard the HI-STORM 100 System may pose is slight activation of the HI-STORM 100 materials caused by irradiation over a 40-year storage period.

Due to the design of the HI-STORM 100 System, no residual contamination is expected to be left behind on the concrete ISFSI pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last overpack is removed.

To evaluate the effects on the MPC and HI-STORM overpack caused by irradiation over a 40-year storage period, the following analysis is provided. Table 2.4.1 provides the conservatively determined quantities of the major nuclides after 40 years of irradiation. The calculation of the material activation is based on the following:

- Beyond design basis fuel assemblies (B&W 15x15, 3-74.8% enrichment, 47,50070,000 MWD/MTU, and ~~eight~~ five-year cooling time) stored for 40 years. *A constant source term for 40 years was used with no decrease in the neutron source term. This bounds the source term associated with the limiting PWR burnup of 75,000 MWD/MTU if an appropriate decrease in neutron source with fuel age were to be accounted for.*
- Material quantities based on the Design Ddrawings in Section 1.5.
- A constant flux equal to the initial loading condition is conservatively assumed for the full 40 years.
- Material activation is based on MCNP-4A calculations.

As can be seen from the material activation results presented in Table 2.4.1, the MPC and HI-STORM overpack activation is very low, even including the very conservative assumption of a constant flux for 40 years. The results for the concrete in the HI-STORM overpack can be conservatively applied to the ISFSI pad. This is extremely conservative because the overpack shields most of the flux from the fuel and, therefore, the ISFSI pad will experience a minimal flux.

In any case, the HI-STORM 100 System would not impose any additional decommissioning requirements on the licensee of the ISFSI facility per 10CFR72.30, since the HI-STORM 100 System could eventually be shipped from the site.

Table 2.4.1
MPC ACTIVATION

Nuclide	Activity After 40-Year Storage (Ci/m ³)
⁵⁴ Mn	2.20e-3
⁵⁵ Fe	3.53e-3
⁵⁹ Ni	2.91e-6
⁶⁰ Co	3.11e-4
⁶³ Ni	9.87e-5
Total	6.15e-3

HI-STORM OVERPACK ACTIVATION

Nuclide	Activity After 40-Year Storage (Ci/m ³)
Overpack Steel	
⁵⁴ Mn	3.62e-4
⁵⁵ Fe	7.18e-3
Total	7.18e-3
Overpack Concrete	
³⁹ Ar	3.02e-6
⁴¹ Ca	2.44e-7
⁵⁴ Mn	1.59e-7
⁵⁵ Fe	2.95e-5
Total	3.43e-5

2.6 REFERENCES

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- [2.0.3] Levy, et al., "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy - Clad Fuel Rods in Inert Gas," Pacific Northwest Laboratory, PNL-6189, 1987.
- [2.0.4] NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," USNRC, Washington, D.C. Rev. 1 (1986).
- [2.0.5] J.W. McConnell, A.L. Ayers, and M.J. Tyacke, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Component According to Important to Safety," Idaho Engineering Laboratory, NUREG/CR-6407, INEL-95-0551, 1996.
- [2.0.6] NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities, March 2000
- [2.0.7] ASME Code, Section III, Subsection NF and Appendix F, and Code Section II, Part D, Materials, 1995, with Addenda through 1997.
- [2.0.8] *"Cladding Considerations for the Transportation and Storage of Spent Fuel Interim Staff Guidance-11, Revision 2, July 30, 2002.*
- [2.1.1] ORNL/TM-10902, "Physical Characteristics of GE BWR Fuel Assemblies", by R.S. Moore and K.J. Notz, Martin Marietta (1989).
- [2.1.2] U.S. DOE SRC/CNEAF/96-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1994, Feb. 1996.
- [2.1.3] Deleted.
- [2.1.4] Deleted.

[†]The 1997 edition of ACI-349 is specified for ISFSI pad and embedment design for deployment of the anchored HI-STORM 100A and HI-STORM 100SA.

- [2.1.5] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.
- [2.1.6] DOE Multi-Purpose Canister Subsystem Design Procurement. Specification.
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- [2.1.8] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.
- [2.2.1] ASME Boiler & Pressure Vessel Code, American Society of Mechanical Engineers, 1995 with Addenda through 1997.
- [2.2.2] ASCE 7-88 (formerly ANSI A58.1), "Minimum Design Loads for Buildings and Other Structures", American Society of Civil Engineers, New York, NY, 1990.
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- [2.2.4] Holtec Report HI-2012610, "Final Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 72-1008, latest revision , .
- [2.2.5] Holtec Report HI-951251, "Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 71-9261, latest revision.
- [2.2.6] "Debris Collection System for Boiling Water Reactor Consolidation Equipment", EPRI Project 3100-02 and ESEERCO Project EP91-29, October 1995.
- [2.2.7] Design Basis Tornado for Nuclear Power Plants, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, April 1974.
- [2.2.8] ANSI/ANS 57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (dry type)", American Nuclear Society, LaGrange Park, Illinois.
- [2.2.9] NUREG-0800, SRP 3.5.1.4, USNRC, Washington, DC.
- [2.2.10] United States Nuclear Regulatory Commission Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", August 1973 and Rev. 1, April 1976.

- [2.2.11] "Estimate of Tsunami Effect at Diablo Canyon Nuclear Generating Station, California."
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Appendix 2.B The Forced Helium Dehydration (FHD) System

2.B.1 System Overview

The Forced Helium Dehydration (FHD) system is used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump *or an inert gas* has been expelled in the water blowdown operation. The FHD system is required to be used for *any MPCs containing all moderate burnup fuel if the heat load is greater than certain values (see Section 4.5) or if the MPC contains at least one high burnup fuel assembly. The FHD method of moisture removal* and is optional for MPCs containing all moderate burnup fuel assemblies.

Expelling the water from the MPC using a conventional pump *or a water displacement method using inert gas* would remove practically all of the contained water except for the small quantity remaining on the MPC baseplate below the bottom of the drain line and an even smaller adherent amount wetting the internal surfaces. A skid-mounted, closed loop dehydration system will be used to remove the residual water from the MPC such that the partial pressure of the trace quantity of water vapor in the MPC cavity gas is brought down to ≤ 3 torr. The FHD system, engineered for this purpose, shall utilize helium gas as the working substance.

The FHD system, schematically illustrated in Figure 2.B.1, can be viewed as an assemblage of four thermal modules, namely, (i) the condensing module, (ii) the demister module, (iii) the helium circulator module and (iv) the pre-heater module. The condensing module serves to cool the helium/vapor mixture exiting the MPC to a temperature well below its dew point such that water may be extracted from the helium stream. The condensing module is equipped with suitable instrumentation to provide a direct assessment of the extent of condensation that takes place in the module during the operation of the FHD system. The demister module, engineered to receive partially cooled helium exiting the condensing module, progressively chills the recirculating helium gas to a temperature that is well below the temperature corresponding to the partial pressure of water vapor at 3 torr.

The motive energy to circulate helium is provided by the helium circulator module, which is sized to provide the pressure rise necessary to circulate helium at the requisite rate. The last item, labeled the pre-heater module, serves to pre-heat the flowing helium to the desired temperature such that it is sufficiently warm to boil off any water present in the MPC cavity.

The pre-heater module, in essence, serves to add supplemental heat energy to the helium gas (in addition to the heat generated by the stored SNF in the MPC) so as to facilitate rapid conversion of water into vapor form. The heat input from the pre-heater module can be adjusted in the manner of a conventional electric heater so that the recirculating helium entering the MPC is sufficiently dry and hot to evaporate water, but not unduly hot to place unnecessary thermal burden on the condensing module.

The FHD system described in the foregoing performs its intended function by continuously removing water entrained in the MPC through successive cooling, moisture removal and reheating of the working substance in a closed loop. In a classical system of the FHD genre, the

moisture removal operation occurs in two discrete phases. In the beginning of the FHD system's operation (Phase 1), the helium exiting the MPC is laden with water vapor produced by boiling of the entrained bulk water. The condensing module serves as the principal device to condense out the water vapor from the helium stream in Phase 1. Phase 1 ends when all of the bulk water in the MPC cavity is vaporized. At this point, the operation of the FHD system moves on to steadily lowering the relative humidity and bulk temperature of the circulating helium gas (Phase 2). The demister module, equipped with the facility to chill flowing helium, plays the principal role in the dehydration process in Phase 2.

2.B.2 Design Criteria

The design criteria set forth below are intended to ensure that design and operation of the FHD system will drive the partial pressure of the residual vapor in the MPC cavity to ≤ 3 torr if the temperature of helium exiting the demister has met the value and duration criteria provided in the HI-STORM technical specifications. The FHD system shall be designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

- i. The temperature of helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.
- ii. The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).
- iii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulent flow regime in the MPC cavity.
- iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demister outlet is $\leq 21^\circ\text{F}$ for a period of 30 minutes.

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

- i. The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point of 120°F or less.
- ii. The demister module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in Section 2.B.1) has been completed.
- iii. The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

- iv. The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

2.B.3 Analysis Requirements

The design of the FHD system shall be subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this FSAR.

- i. System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.
- ii. System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demister modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach ≤ 3 torr if the temperature of the helium gas exiting the demister is predicted to be at a maximum of 21°F for 30 minutes.
- iii. Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

2.B.4 Acceptance Testing

The first FHD system designed and built for the MPC drying function required by HI-STORM's technical specifications shall be subject to confirmatory testing as follows:

- a. A representative quantity of water shall be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.
- b. The MPC cavity drying test shall be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).
- c. The drain and vent line RVOAs on the MPC lid shall be connected to the terminals located in the pre-heater and condensing modules of the FHD system, respectively.

- d. The FHD system shall be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demoinsturizer module has been at or below 21°F for thirty minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

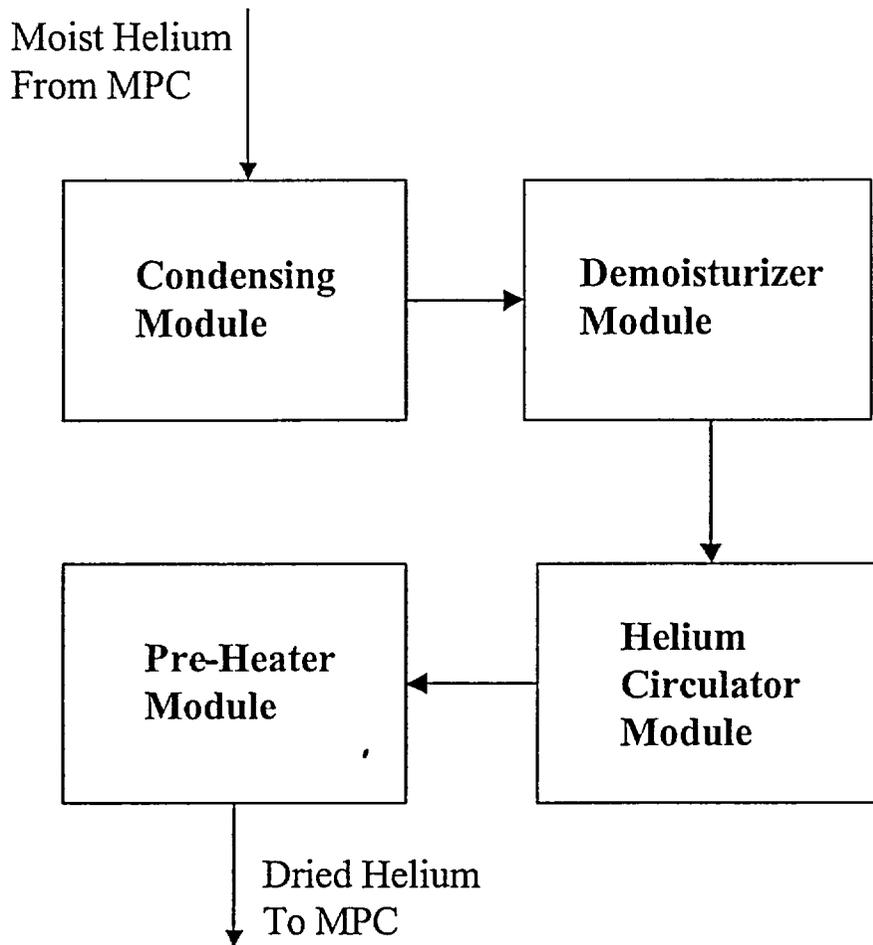


FIGURE 2.B.1: SCHEMATIC OF THE FORCED HELIUM DEHYDRATION SYSTEM