Holtec International
Final Safety Analysis Report for the HI-STORM 100 Cask System*

by

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3.	3	CWB	4/15/2012	253466	CB	7/26/2013	305363	AB	3/12/2014	154830
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6.	6	VG	4/15/2012	547279	TH	7/27/2013	874739	RN	3/12/2014	48837
7.	7	VG	4/15/2012	574727	BK	7/27/2013	199808	RN	3/12/2014	398001
8.	8	VG	4/15/2012	227587	VG	7/27/2013	517656	AM	3/12/2014	128836
9.	9	VG	4/15/2012	951243	VG	7/27/2013	166560	RN	3/12/2014	276825
10.	10	VG	4/15/2012	514322	BK	7/27/2013	947187	AB	3/12/2014	720054
11.	11	VG	4/15/2012	577089	IR	7/27/2013	730705	AM	3/12/2014	123376
12.	12	VG	4/24/2012	514303	IR	7/27/2013	419931	AM	3/12/2014	458752
13.	13	VG	4/15/2012	882010	RN	7/27/2013	708860	RN	3/12/2014	635936

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Certificate of Compliance (1014) and Final Safety Analysis Report Matrix

HI-STORM 100 Cask Storage System Final Safety Analysis Report (FSAR) Revision	NRC Certificate of Compliance (CoC) 1014 Amendment No.
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* - These Revisions incorporated changes made via ECO/72.48 process only.

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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.

Through revision 3 of this FSAR, discussions and specific analyses were presented that described and evaluated MPC designs called the MPC-24EF, MPC-32F, and MPC-68FF. These designs contained features required to classify them as secondary containments, permitting transportation of fuel debris under the auspices of 10 CFR 71, and were the only MPC designs allowed to be loaded with fuel debris. Recent changes to 10 CFR 71 have eliminated the need for secondary containment of fuel debris, so the non-F type MPCs (i.e., MPC-24E, MPC-32 and MPC-68) can now accept fuel debris. Any contents that used to require loading into an MPC-24EF, MPC-32F or MPC-68FF may therefore now be loaded in an MPC-24E, MPC-32 or MPC-68, respectively.

Supplements identified by a Roman numeral "I" (i.e. Chapter 1 and Supplement 1.I) have been inserted in the FSAR as placeholders for future use.

1.0.1 Engineering Change Orders

The changes authorized by the Holtec ECOs (with corresponding 10CFR72.48 evaluations, if applicable) listed in the following table are reflected in this Revision of the FSAR.

LIST OF ECO'S AND APPLICABLE 10CFR72.48 EVALUATIONS

Affected Item	ECO Number	72.48 Evaluation or Screening Number
MPC-68/68F/68FF/68M Basket	-	-
MPC-24/24E/24EF Basket	-	-
MPC-32 Basket	-	-
MPC Enclosure Vessel	-	-
HI-STORM Overpack	1024-153	1042
HI-STORM 100U VVM	5014-177 5014-195	N/A N/A
HI-TRAC 100 and 100D Transfer Cask	-	-
HI-TRAC 125 and 125D Transfer Cask	-	-
General FSAR Changes	5014-211 5014-211R1 5014-211R2 5014-212 5014-213 5014-216	N/A N/A N/A N/A N/A 1052

1.0.2 Design Compatibility of Licensed HI-STORM 100 System Components

Each of the licensed HI-STORM 100 System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility.

The following certified HI-TRAC transfer casks have been determined to have design compatibility and may be used with the MPCs and overpacks fabricated in accordance with the CoC #1014 amendments as listed in the table below:

HI-TRAC Design Compatibility			
HI-TRAC Model	Program Number	Serial Number	Applicable CoC Amendments
125	1025	001	0,1,2,3,4,5,6,7,8,9
125	1025	002	0,1,2,3,4,5,6,7,8,9
125D	1025	003	1,2,3,4,5,6,7,8,9
125D	1025	004	1,2,3,4,5,6,7,8,9
125D	1025	005	2,3,4,5,6,7,8,9
125D	1025	007	1,2,3,4,5,6,7,8,9
125D	1025	008	1,2,3,4,5,6,7,8,9
125D	1025	009	2,3,4,5,6,7,8,9
125D	1025	010	3,4,5,6,7,8,9
125D	1025	011	5,6,7,8,9
125D	1025	012	5,6,7,8,9
125D	1025	013	5,6,7,8,9
125D	1025	014	3,4,5,6,7,8,9
125D	1025	015	3,4,5,6,7,8,9
125D	1025	016	5,6,7,8,9
125D	1025	017	7,8,9
125D	1025	019	7,8,9
100	1026	001	0,1,2,3,4,5,6,7,8,9
100D	1026	003	2,3,4,5,6,7,8,9
100D	1026	004	2,3,4,5,6,7,8,9
100D	1026	005	2,3,4,5,6,7,8,9
100D	1026	006	2,3,4,5,6,7,8,9
100D Version IP1	1026	008	4
100D	1026	009	6,7,8,9

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.

Boral™ 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 (or post-irradiation cooling time) for a spent fuel assembly is the time between reactor shutdown and the time the spent fuel assembly is loaded into the MPC.

Critical Characteristic means a feature of a component or assembly that is necessary for the proper safety function of the component or assembly. Critical characteristics of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function.

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, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel

or gross failure of the cladding is expected based on engineering evaluations, 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.

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.

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. The FSAR serves as the Design Report for the HI-STORM 100 System.

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. The FSAR serves as the Design Specification for the HI-STORM 100 System.

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

Equivalent (or Equal) Material is a material with critical characteristics (see definition above) that meet or exceed those specified for the designated material.

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

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

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 is ruptured fuel rods, severed rods, loose fuel pellets, containers or structures that are supporting these loose fuel assembly parts, 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.

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 **H**oltec **I**nternational **Tra**nsfer **C**ask. In this FSAR there are several HI-TRAC transfer casks, the 125-ton standard design HI-TRAC (HI-TRAC-125), the 125-ton dual-purpose lid design (HI-TRAC 125D), the 100-ton HI-TRAC (HI-TRAC-100), the 100-ton dual purpose lid design (HI-TRAC 100D). 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 HI-TRAC transfer cask designs, unless the discussion requires distinguishing among the designs. 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 all overpack designs, including the standard design (HI-STORM 100) and two alternate designs (HI-STORM 100S and HI-STORM 100S Version B). The term "overpack" also applies to those overpacks designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA), unless otherwise clarified.

HI-STORM 100 System consists of any loaded MPC model placed within any design variant of the HI-STORM 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.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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 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 overpack. The trademark is derived from the **metal-con**crete composition of the HI-STORM overpack.

MGDS is an acronym for Mined Geological Disposal System.

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

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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 consisting of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell (the MPC Enclosure Vessel). There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior diameters. The MPC is the confinement boundary for storage conditions.

MPC Transfer means transfer of the MPC between the overpack and the transfer cask which begins when the MPC is lifted off the HI-TRAC bottom lid and ends when the MPC is supported from beneath by the overpack (or the reverse).

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.

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

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), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, instrument tube tie-rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

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.

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 wherein the storage locations are ascribed to two distinct regions each with its own maximum allowable specific heat generation rate. Regionalized fuel loading does not apply to the MPC-68F model.

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 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 Operations 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 the storage system, containing an MPC loaded with CSF stored in *uniform storage,* will actually reject with the ambient environment 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.

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 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

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, 2.III.3.b Structural, 2.III.3.c Thermal	10CFR72.122(b)	2.2
		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
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.5

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
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
		10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a)(1)	9.0
2.3 Safety Protection Systems	--	--	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
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, 2.2.3.10
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
	14.III.2 Cask Decontamination	10CFR72.236(i)	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.3 Financial Assurance & Record Keeping	10CFR72.30	(1)
	14.III.4 License Termination	10CFR72.54	(1)
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 3.V.2.c Structural Materials	10CFR72.24(c)(3)	3.3
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
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4
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 ISG-11, Revision 3	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria ISG-11, Revision 3	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

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
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
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, 6.3.3
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 ISG-18	10CFR72.24(c)(3) 10CFR72.24(1)	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
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3

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
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.236(1)	7.1
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7. 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 ISG-18	10CFR72.104(a)	7.1
7.2.2 Pressurization of Confinement Vessel	--	--	7.1
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.1
7.3.1 Fission Gas Products	--	--	7.1
7.3.2 Release of Contents	ISG-18	--	7.1
NA	--	10CFR72.106(b)	7.1
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			
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

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**
10.3 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
	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

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.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 Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	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

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
	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 SSC 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
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	

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

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 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
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."	Exception: 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.	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.
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."	Clarification: 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.	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.

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
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..."	Clarification: 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.	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.
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..." 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". 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	Exception: The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used as guidance for the material selection and specification, and placement of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack, including clarifications on implementation of this code. ACI 318 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.	Concrete is provided in the HI-STORM overpack primarily 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. To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. The temperature limits for normal conditions are per Paragraph A.4.3 of Appendix A to ACI 349 and temperature limits for

Table 1.0.3 (continued)

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
ACI 359 should comply with the requirements of ACI 349".		off-normal and accident conditions are per Paragraph A.4.2 of Appendix A to ACI 349. Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.
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."	Clarification: 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.	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.
4.IV.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." 4.IV.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." 4.IV.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."	Clarification: As described in Section 4.3, all fuel array types authorized for storage are assigned a single peak fuel cladding temperature limit.	As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak normal fuel cladding temperature limit of 400°C.

Table 1.0.3 (continued)

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
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 operations (e.g., fuel transfer) and the worst-case credible accident."		
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."	Clarification: As discussed in Section 4.4, 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 model.	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.
4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."	Exception: All calculations of fuel assembly effective thermal conductivities use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.	The calculated effective thermal conductivities based on nominal design fuel dimensions have been compared with available literature values [1.0.6] and are demonstrated to be conservative.
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."	Clarification: No additional heat balance is performed or provided.	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,

Table 1.0.3 (continued)

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
		therefore, ensures a heat balance at the surface of the cask.
4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."	<u>Exception:</u> No input or output file listings are provided in Chapter 4.	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.
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.	<u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the fuel assemblies is calculated using maximum weights and minimum densities.	Calculating the volume occupied by the fuel assemblies using maximum weights and minimum densities conservatively overpredicts the volume occupied by the fuel and correspondingly underpredicts the remaining free volume.

Table 1.0.3 (continued)

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
7.V.4 "Confinement Analysis. Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary."	<u>Exception:</u> No confinement analysis is performed and no effluent dose at the controlled area boundary is calculated.	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.,non-destructive examinations and pressure testing). Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. Pursuant to ISG-18, the Holtec MPC is constructed in a manner that supports leakage from the confinement boundary being non-credible. Therefore, no confinement analysis is required.
9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."	<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.	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.

Table 1.0.3 (continued)

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

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
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."	Exception: 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.	The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation. The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures. The composition of the Holtite-A is confirmed by inspection and tests prior to first use. 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.
13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."	Exception: Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.	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 will apply this QA program to all important-to-safety dry storage cask activities. Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation.

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 and HI-STORM 100S Version B overpack designs are variants of the HI-STORM 100 overpack design and have their own drawings in Section 1.5. The "S" suffix indicates an enhanced overpack design, as described later in this section. "Version B" indicates an enhanced HI-STORM 100S overpack design. The HI-STORM 100S and 100S Version B accept the same MPCs and fuel types as the HI-STORM 100 overpack 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 the HI-STORM 100, the HI-STORM 100S, and the HI-STORM 100S Version B. Where necessary, the text distinguishes among the three overpack designs. See Figures 1.1.1A and 1.1.3A for pictorial views of the HI-STORM 100S overpack design. See Figures 1.1.1B and 1.1.3B for pictorial views of the HI-STORM 100S Version B design.

The HI-STORM 100A overpack is a variant of two of the three HI-STORM 100 System overpack designs 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 anchor design is applicable to the HI-STORM 100S overpack design, in which case the assembly would be named HI-STORM 100SA. The HI-STORM 100A anchor design is not applicable to the HI-STORM 100S Version B overpack design. Therefore, the HI-STORM 100S Version B overpack cannot be deployed in the anchored configuration at this time. Hereafter in the text, discussion of HI-STORM 100A applies to both the standard (HI-STORM 100A) and 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 basic overpack design by utilizing different MPCs. The external diameters 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 contain 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, 1.1.1A, and 1.1.1B show the HI-STORM 100 System 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 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 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 overpack 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 external diameter, manufacturing requirements, and handling features, but different in their SNF arrangement details, are designed to fit a common overpack design. 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 of the HI-STORM 100A configuration is provided in Subsection 1.2.1.2.1. The anchored design is applicable to the HI-STORM 100 and the HI-STORM 100S overpack designs only.

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 enclosure vessel is identical in external diameter to those presented in References [1.1.2] and [1.1.3]. However, certain fuel basket models may not be certified for storage or transportation in the HI-STAR 100 System. The Part 71 and 72 CoCs for HI-STAR 100 should be consulted for the MPC models that are certified for that system. 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, and 1.1.3B. 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.

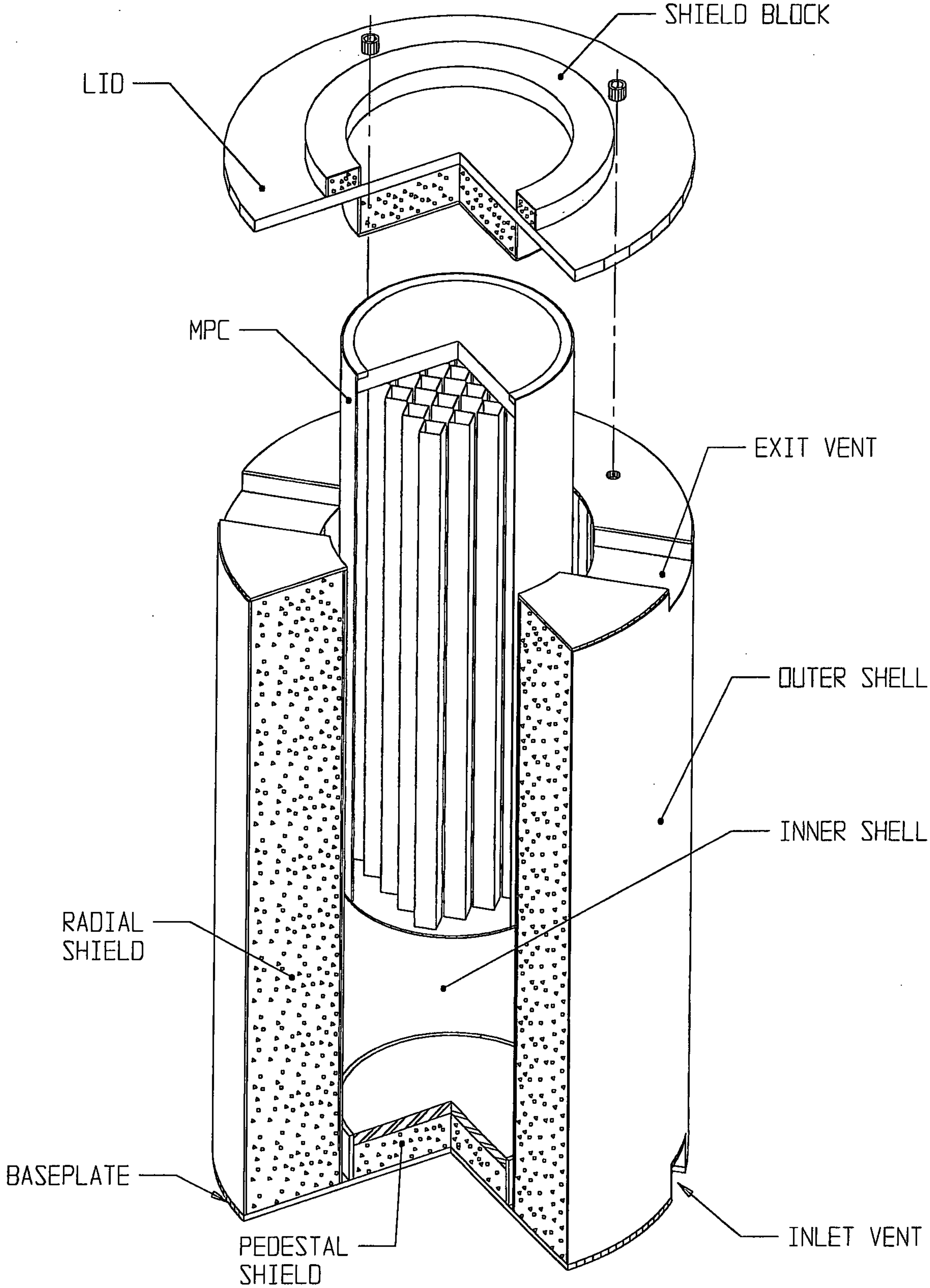

FIGURE 1.1.1; HI-STORM 100 OVERPACK WITH MPC PARTIALLY INSERTED

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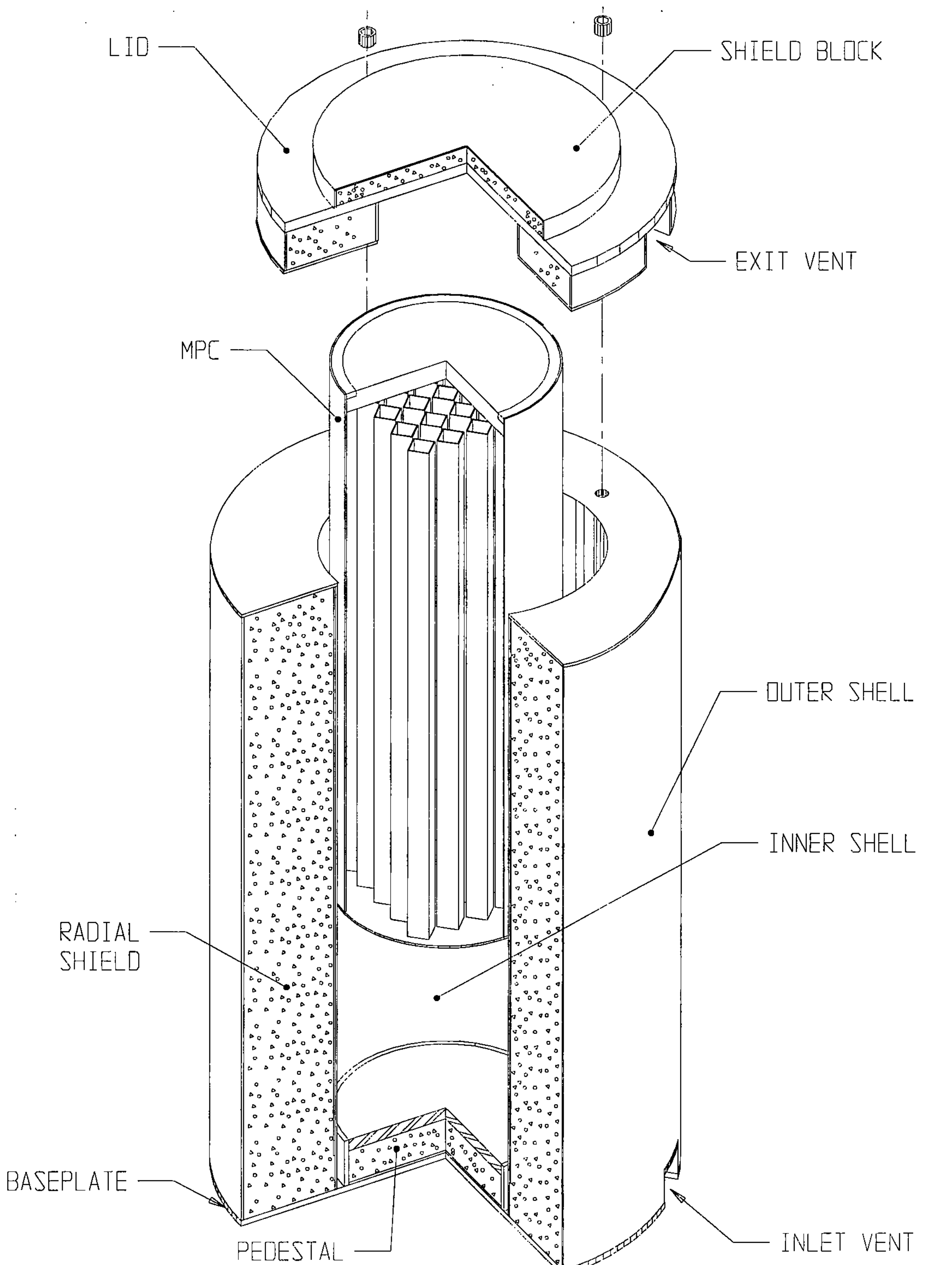

FIGURE 1.1.1A; HI-STORM 100S OVERPACK WITH MPC PARTIALLY INSERTED

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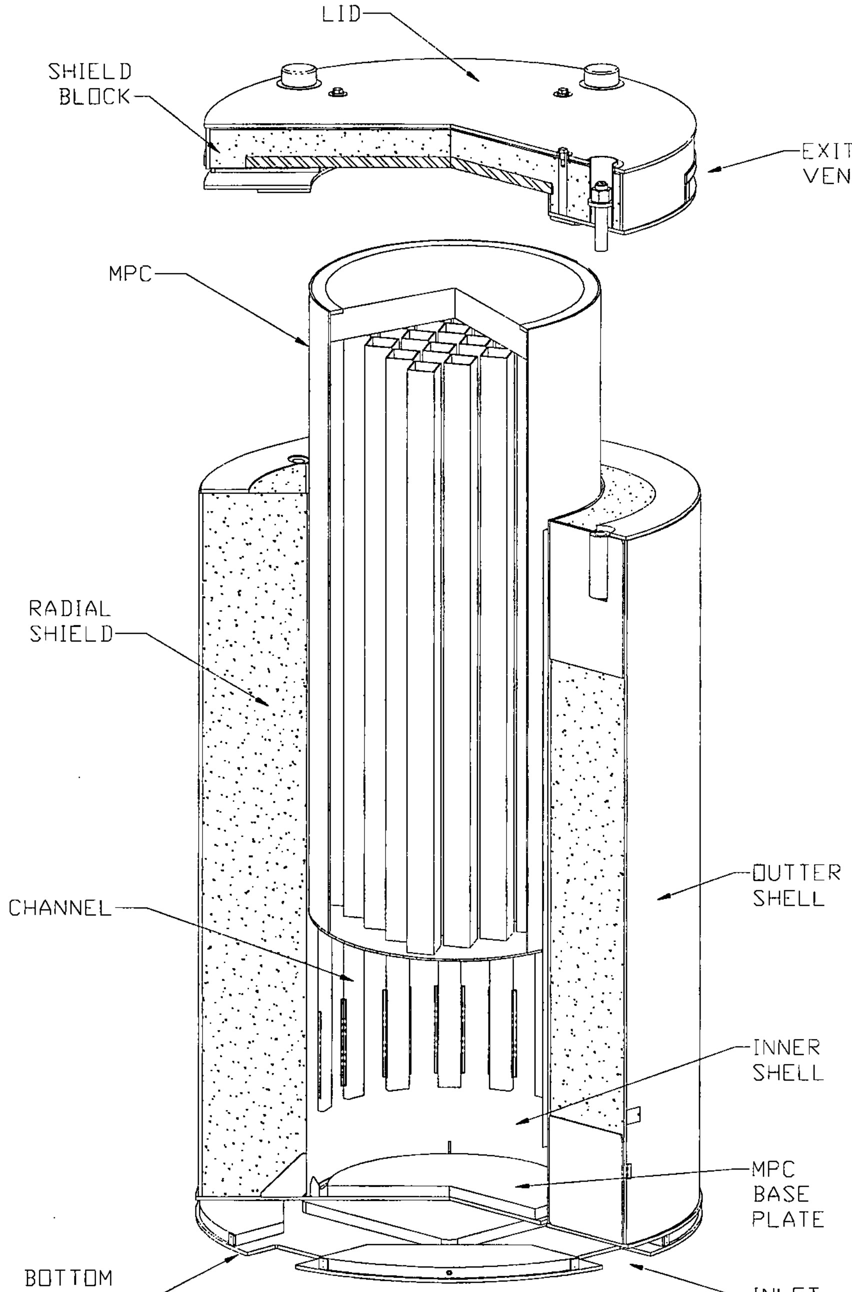

FIGURE 1.1.1B; HI-STORM 100S VERSION B OVERPACK WITH MPC PARTIALLY INSERTED

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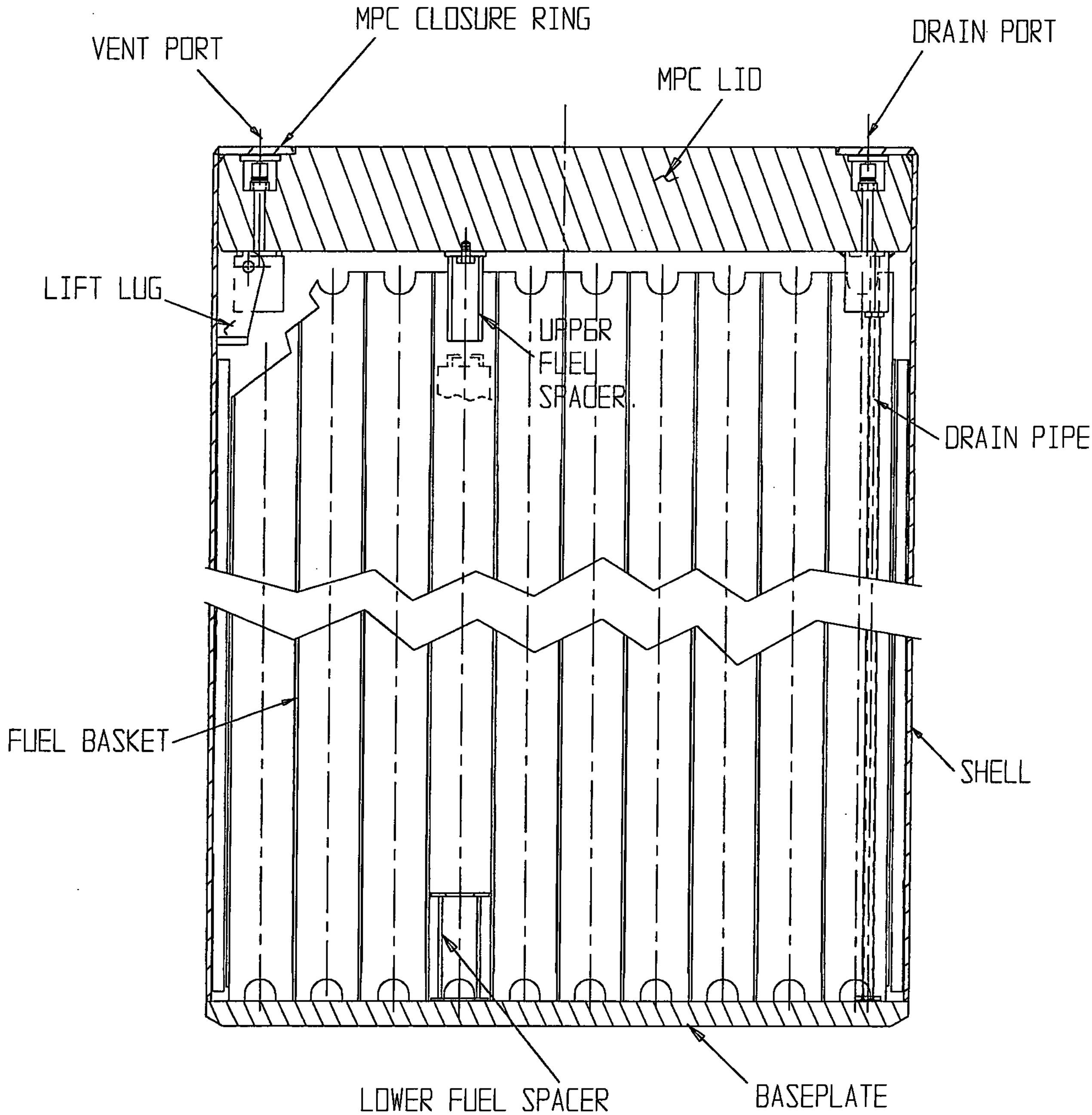

FIGURE 1.1.2; CROSS SECTION ELEVATION VIEW OF MPC

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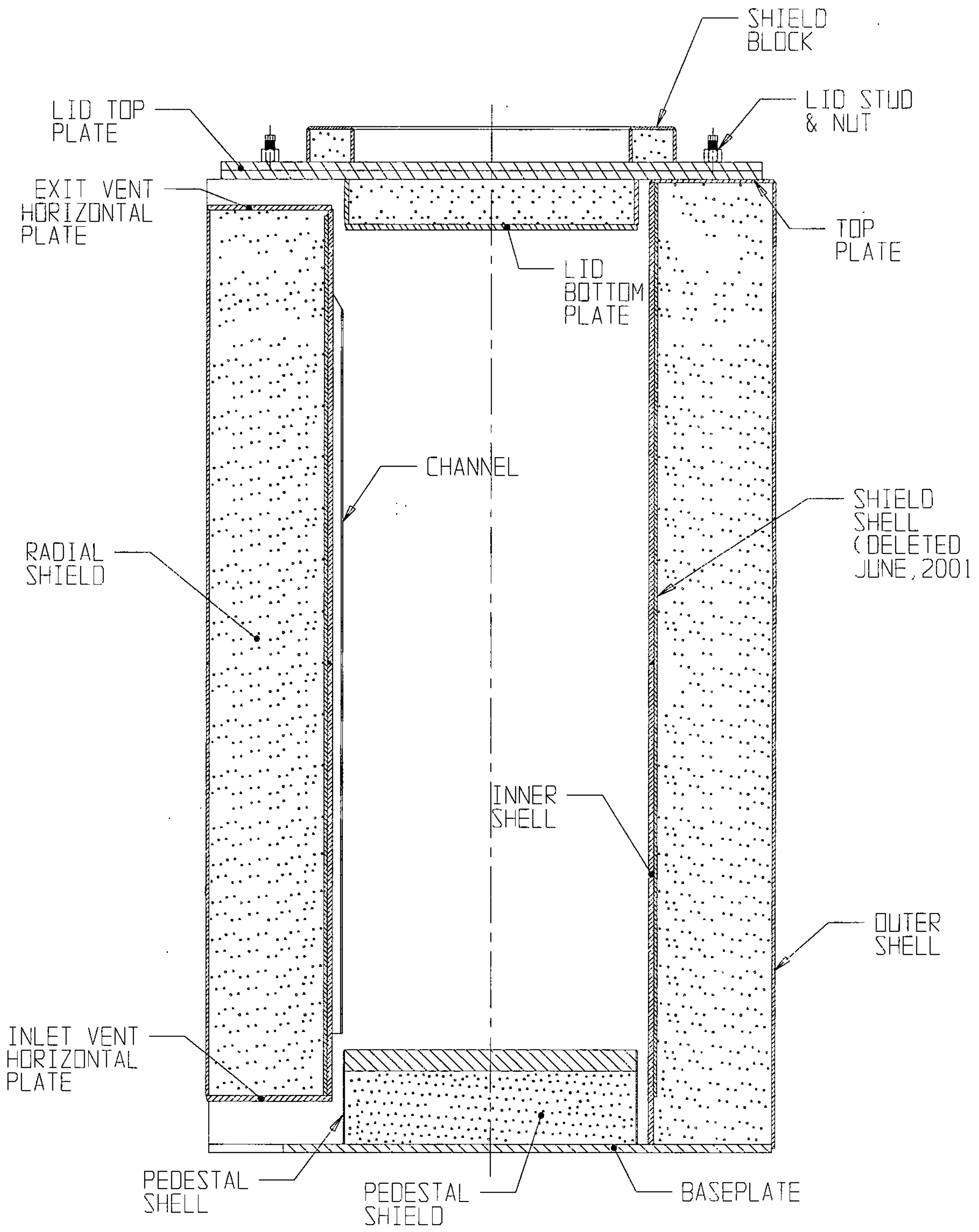

FIGURE 1.1.3; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

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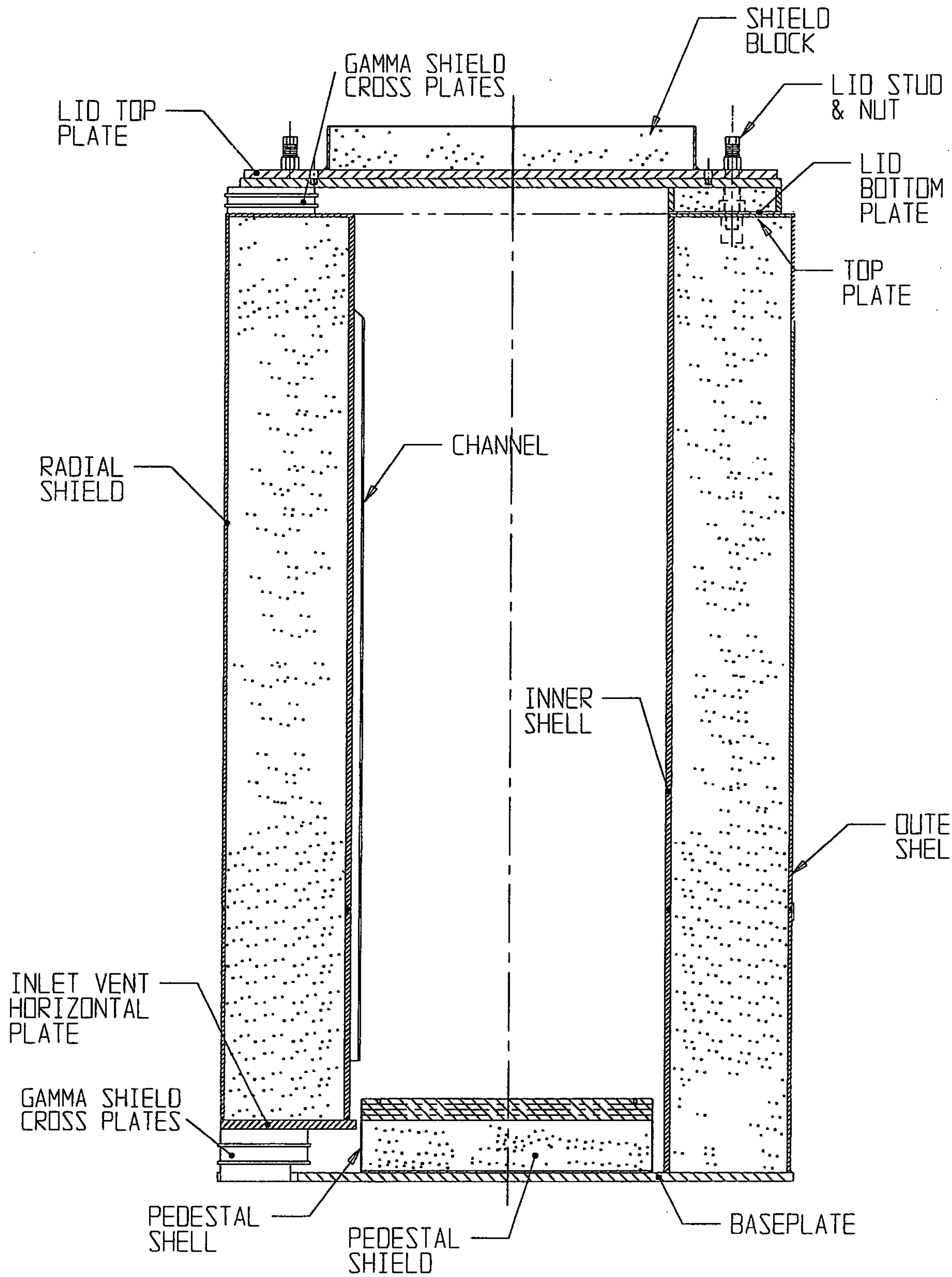

FIGURE 1.1.3A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

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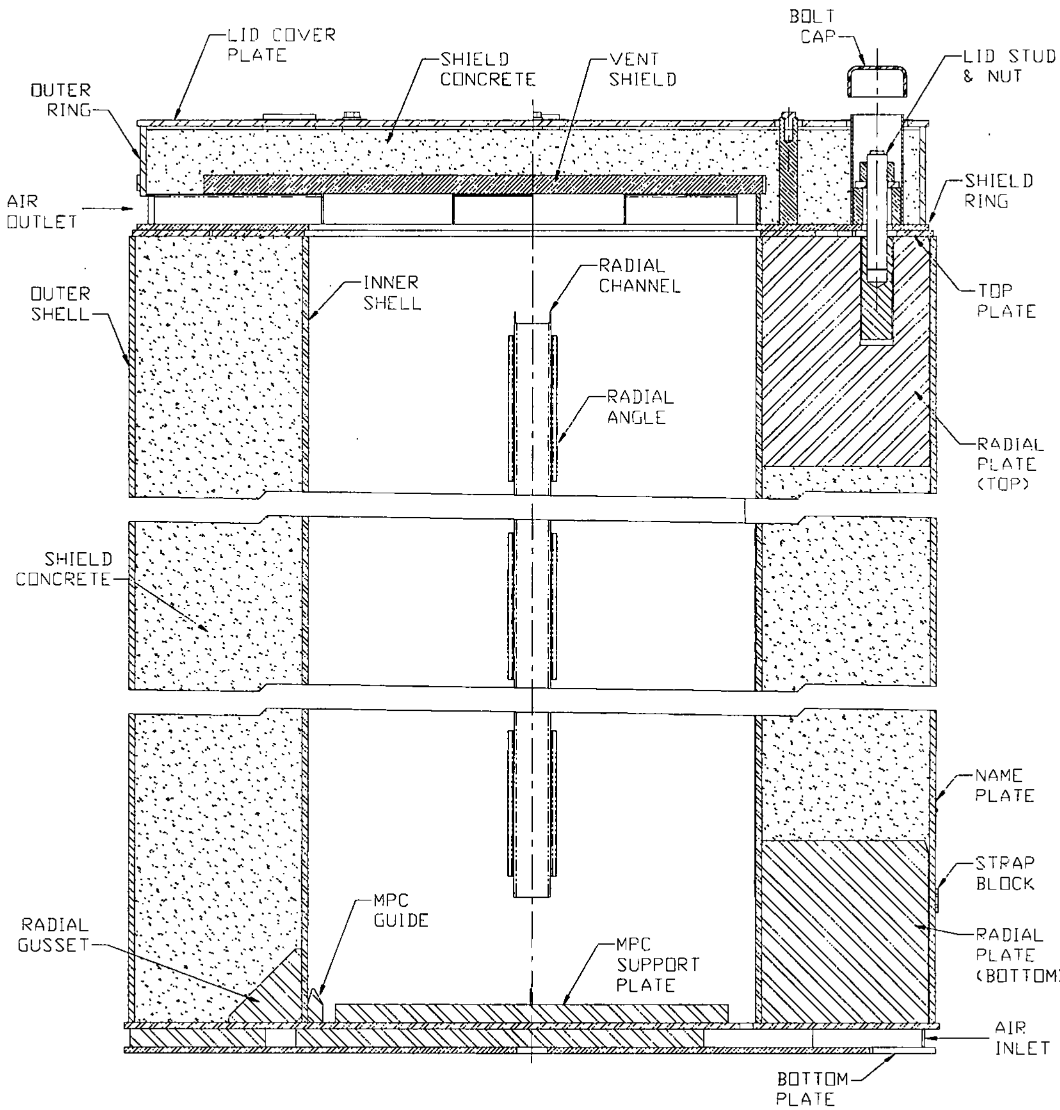

FIGURE 1.1.3B; HI-STORM 100S VERSION B OVERPACK CROSS SECTIONAL ELEVATION VIEW

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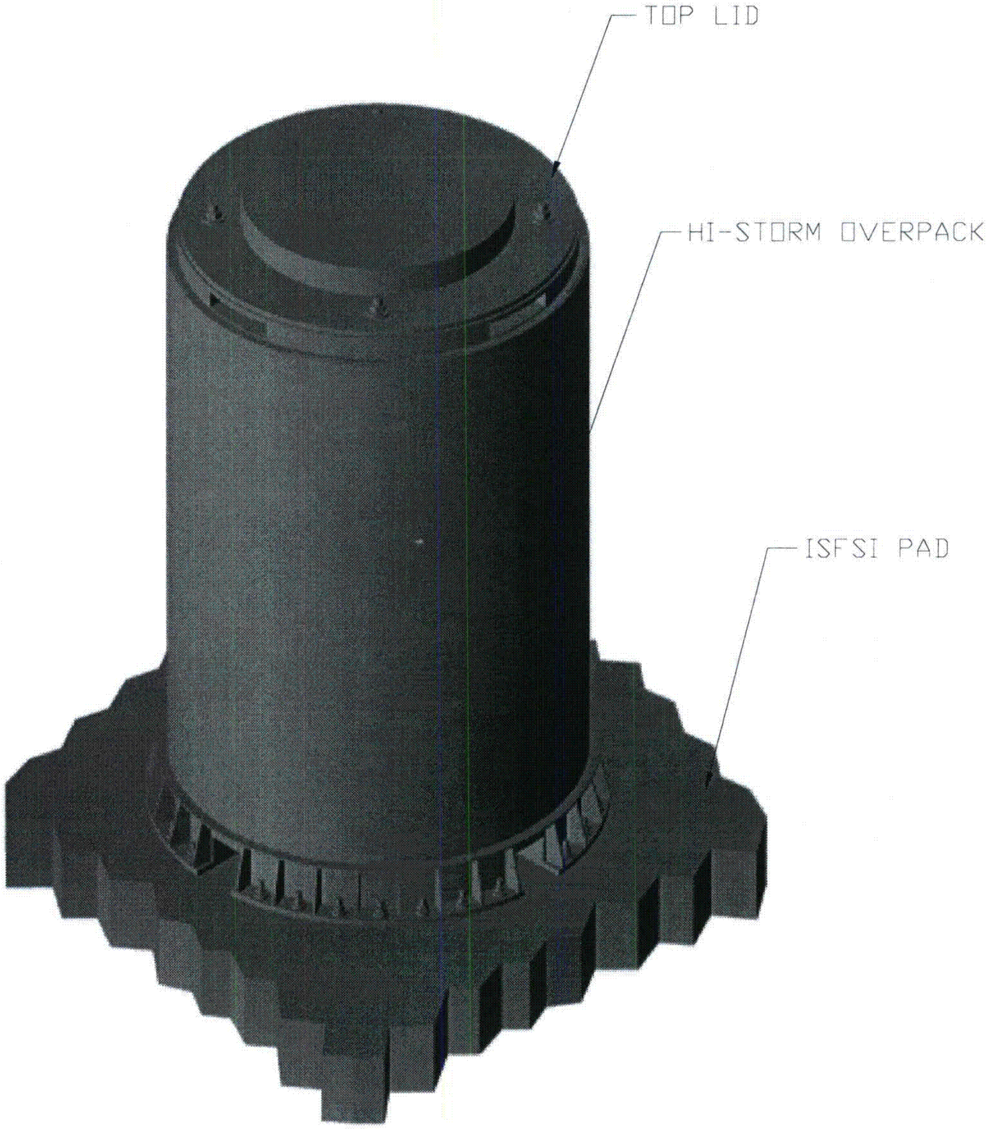

FIGURE 1.1.4; A PICTORAL VIEW OF THE HI-STORM 100A OVERPACK (100SA MODEL SHOWN)

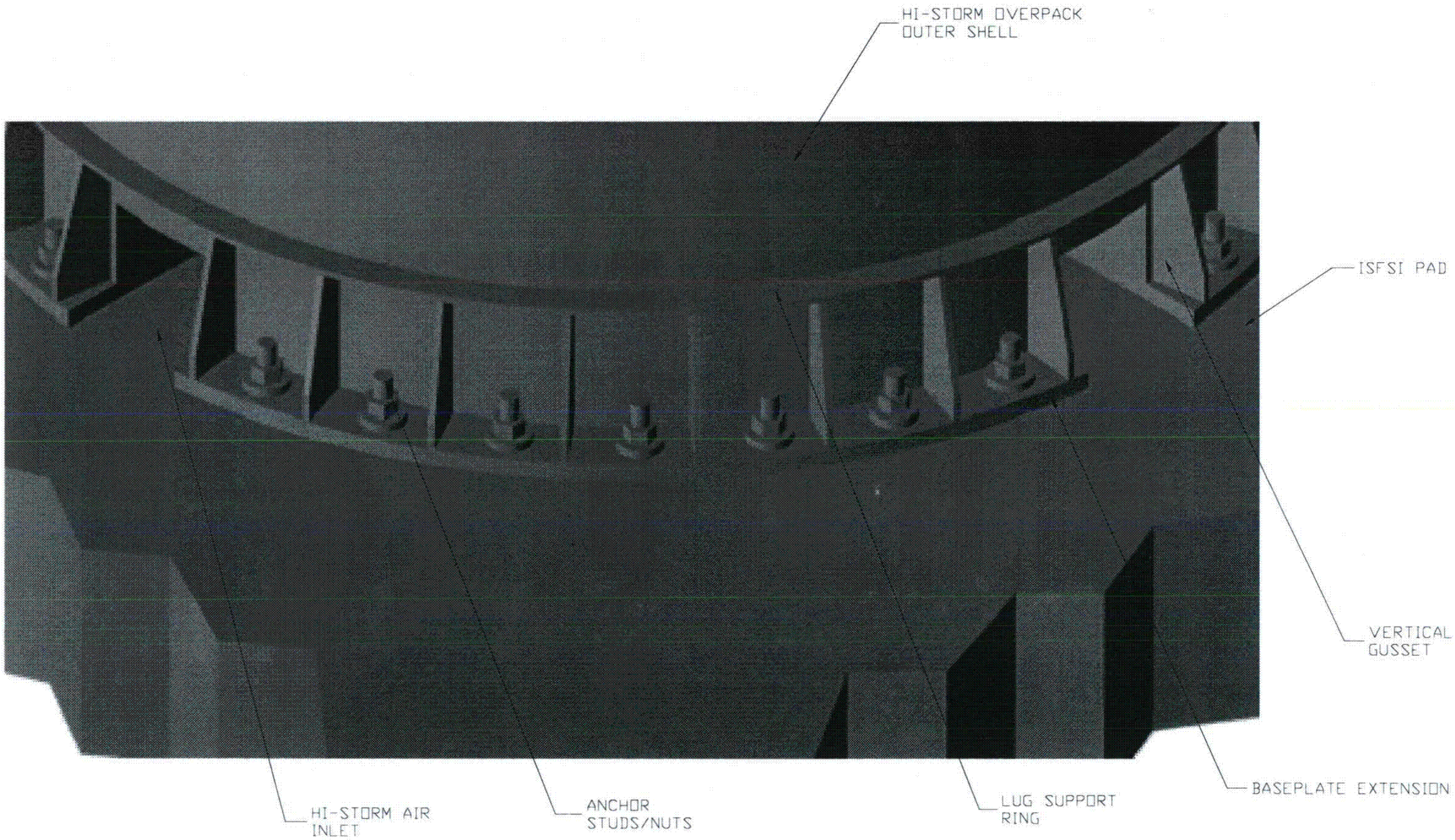

FIGURE 1.1.5; ANCHORING DETAIL FOR THE HI-STORM 100A AND 100SA OVERPACKS

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. Figures 1.2.1 and 1.2.1A provide example cross sectional views of the HI-STORM 100 System with an MPC inserted into HI-STORM 100 and HI-STORM 100S storage overpacks, respectively. Figure 1.1.1B provides similar information for the HI-STORM 100 System using a HI-STORM 100S Version B 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 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 (or other system capable of the same backfill condition)
iii. lifting and handling systems
iv welding equipment
v. transfer vehicles/trailer

All MPCs have identical external diameters. The outer diameter of the MPC is 68-3/8 inches[†] and the maximum overall length is approximately 190-1/2 inches. See Section 1.5 for the 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 system data and parameters for the MPCs.

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

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 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 overpack is 239-1/2 inches.

There are two variants of the HI-STORM 100S overpack, differing from each other 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 assuming standard density 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.

There are also variants of the HI-STORM 100S Version B overpack, differing from each other only in height and weight. The HI-STORM 100S-218 is 218 inches high, and the HI-STORM 100S-229 is 229 inches high. The HI-STORM 100S-229 is approximately 8,700 lbs heavier, including standard density concrete. Hereafter in the text, the versions of the HI-STORM 100S Version B overpack will only be referred to as HI-STORM 100S Version B and will be discussed separately only if the design feature being discussed is different between the overpacks. See Section 1.5 for drawings.

The weight of the overpack without an MPC varies from approximately 135 tons to 160 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 of each MPC is 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 eight MPC models, distinguished by the type and number of fuel assemblies authorized for loading. Section 1.2.3 and Table 1.2.1 summarize the allowable contents for each MPC model. Section 2.1.9 provides the detailed specifications for the contents authorized for storage in the HI-STORM 100 System. Drawings for 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 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, 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 fuel basket 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. 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.

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 to 2-1/2 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 confinement boundary is constructed entirely from stainless steel alloy materials. All MPC components that may come into contact with spent fuel pool water or the ambient environment (with the exception of neutron absorber, aluminum seals on vent and drain port caps, and optional aluminum heat conduction elements) must be constructed from stainless steel alloy materials. 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. Any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed below, except that the steel pieces comprising the MPC shell (i.e., the 1/2" thick cylinder) must be fabricated from the same Alloy X stainless steel type.

- 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 Overpack

The HI-STORM overpacks are rugged, heavy-walled cylindrical vessels. Figures 1.1.3B, 1.2.7, 1.2.8, and 1.2.8A provide cross sectional views of the HI-STORM 100 System, showing all of the overpack designs. The HI-STORM 100A overpack design is an anchored variant of the HI-STORM 100 and -100S designs 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. A minimum of 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 (including Version B) 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 and –100S Version B are integral to the lid assembly and are not in vertical alignment with the inlet ducts. See the drawings in Section 1.5 for details of the overpack air inlet and outlet duct designs. The air inlets and outlets are covered by a 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 air inlets and 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 and –100S Version B overpack designs, 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 radiation traveling through the ducts. The result of scattering the radiation in the ducts is a significant decrease in the local dose rates around the air inlets and 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. For the HI-STORM 100 and –100S overpack designs, 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. There is only one gamma shield cross plate design employed with the HI-STORM 100S Version B overpack design, which has been appropriately considered in the shielding and thermal analyses.

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 HI-STORM 100S Version B overpack design incorporates partial-length radial plates at the top of the overpack to secure the anchor blocks and uses both gussets and partial-length radial plates at the bottom of the overpack for structural stability. Details of this arrangement are shown in the drawings in Section 1.5.

The four anchor blocks are located on 90° arcs around the circumference of the top of the overpack lid. 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 the cracking and crushing under a tip-over accident is not of significant consequence, the deformation characteristics are germane to the analysis of the structural members.

Density and compressive strength are the key parameters that delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal (HI-STORM 100 and –100S overpacks only), and overpack 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 three base HI-STORM overpack designs - HI-STORM 100, HI-STORM 100S, and HI-STORM 100S Version B. The significant differences among the three 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 variants of the HI-STORM 100S overpack design, differing only in height and weight. The HI-STORM 100S(232) is approximately 232 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 211 inches high without the lid installed. The HI-STORM 100S(243) is approximately 243 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 222 inches high without the lid installed. There are also variants of the HI-STORM 100S Version B overpack design, differing only in height and weight. The HI-STORM 100S-218 is approximately 218 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 199 inches

high without the lid installed. The HI-STORM 100S-229 is approximately 229 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and 210 inches high without the lid installed.

The HI-STORM 100S Version B overpack design does not include a concrete-filled pedestal to support the MPC. Instead, the MPC rests upon a steel plate that maintains the MPC sufficiently above the inlet air ducts to prevent direct radiation shine through the ducts. To facilitate this change, the inlet air ducts for the HI-STORM 100S Version B are shorter in height but larger in width. See the drawings in Section 1.5 for details.

The anchored embodiment of the HI-STORM overpack is referred to as HI-STORM 100A or HI-STORM 100SA. The HI-STORM 100S version B overpack design may not be deployed in the anchored configuration at this time. 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, imputes 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 [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 100D and 125D designs.

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 HI-TRAC-125 weight does not exceed 125 tons during any loading or transfer operation. The HI-TRAC-100 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 HI-TRAC-100 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 HI-TRAC-125 to bound the HI-TRAC-100 design. Additionally, as the two HI-TRACs are identical except for a reduced thickness of lead and water, the HI-TRAC-125 has a larger thermal resistance than the smaller and lighter HI-TRAC-100. Therefore, for normal conditions the HI-TRAC-125 thermal analysis bounds that of the HI-TRAC-100. 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 100D and 125D Transfer Casks

As an option to using either of the standard HI-TRAC transfer cask designs, users may choose to use the optional HI-TRAC 100D or 125D designs. Figure 1.2.9A provides a typical cross section of the HI-TRAC-125D with the pool lid installed. The HI-TRAC 100D (figure not shown) is similar to the HI-TRAC 125D except for the top lid (which contains no Holtite). Like the standard designs, the optional designs are 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 100D and 125D designs and the standard designs 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 designs and the optional designs. The optional designs are 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 lids remain on the HI-TRAC 100D and 125D until MPC transfer is required. The HI-STORM mating device is located between, and secured with bolting (if required by seismic analysis), to the top of the HI-STORM overpack and the HI-TRAC 100D or 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 designs and the optional HI-TRAC 100D or 125D designs 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 neutron 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 the HI-STORM 100 and –100S have a thick circular concrete pedestal upon which the MPC rests. This concrete pedestal is not necessary in the HI-STORM 100S Version B overpack design. 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 and HI-TRAC 100D top lids are composed of steel only providing gamma attenuation.

1.2.1.3.1 Fixed Neutron Absorbers

1.2.1.3.1.1 Boral™

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.

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.

Operating experience in nuclear plants with fuel loading of Boral equipped MPCs as well as laboratory test data indicate that the aluminium used in the manufacture of the Boral may react with water, resulting in the generation of hydrogen. The numerous variables (i.e., aluminium particle size, pool temperature, pool chemistry, etc.) that influence the extent of the hydrogen produced make it impossible to predict the amount of hydrogen that may be generated during MPC loading or unloading at a particular plant. Therefore, due to the variability in hydrogen generation from the Boral-water reaction, the operating procedures in Chapter 8 require monitoring for combustible gases and purging the space beneath the MPC lid during loading and unloading operations when an ignition event could occur (i.e., when the space beneath the MPC lid is open to the welding or cutting operation).

1.2.1.3.1.2 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 matrix of 6061 aluminum alloy reinforced with Type 1 ASTM C-750 boron carbide. METAMIC® is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. Typically, the average B_4C particle size is between 10 and 15 microns. As described in the U.S. patents held by METAMIC, Inc.*†, the high performance and reliability of METAMIC® derives from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields excellent and uniform homogeneity.

The powders are carefully blended without binders or other additives that could potentially adversely influence performance. The maximum percentage of B_4C that can be dispersed in the aluminum alloy 6061 matrix is approximately 40 wt.%, although extensive manufacturing and testing experience is limited to approximately 31 wt.%. The blend of powders is isostatically compacted into a green billet under high pressure and vacuum sintered to near theoretical density.

According to the manufacturer, billets of any size can be produced using this technology. The billet is subsequently extruded into one of a number of product forms, ranging from sheet and plate to angle, channel, round and square tube, and other profiles. For the METAMIC® sheets used in the MPCs, the extruded form is rolled down into the required thickness.

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 (Ref. [1.2.11]) 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 metal matrix configuration produced by the powder metallurgy process with a complete absence of open porosity in METAMIC® ensures that its density is essentially equal to the theoretical density.

- 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 corrosion test conditions has been observed.

* U.S. Patent No. 5,965,829, "Radiation Absorbing Refractory Composition".

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

In addition, independent measurements of boron carbide particle distribution show extremely small particle-to-particle distance† and near-perfect homogeneity.

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 makes METAMIC® an acceptable neutron absorber material for use in the 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 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. 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. Additional details pertaining to the qualification and production tests for METAMIC® are summarized in Subsection 9.1.5.3.

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 Boral.

NUREG/CR-5661 (Ref. [1.2.14]) recommends limiting poison material credit to 75% of the minimum ^{10}B loading because of concerns for potential "streaming" of neutrons, and allows for greater percentage credit in criticality analysis "if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented". The value of 75% is characterized in NUREG/CR-5661 as a very conservative value, based on experiments with neutron poison containing relatively large B_4C particles, such as BORAL with an average particle size in excess of 100 microns. METAMIC®, however, has a much smaller particle size of typically between 10 and 15 microns on average. Any streaming concerns would therefore be drastically reduced.

† 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.

Analyses performed by Holtec International show that the streaming due to particle size is practically non-existent in METAMIC®. Further, EPRI's neutron attenuation measurements on 31 and 15 B_4C weight percent METAMIC® showed that METAMIC® exhibits very uniform ^{10}B areal density. This makes it easy to reliably establish and verify the presence and microscopic and macroscopic uniformity of the ^{10}B in the material. Therefore, 90% credit is applied to the minimum ^{10}B areal density in the criticality calculations, i.e. a 10% penalty is applied. This 10% penalty is considered conservative since there are no significant remaining uncertainties in the ^{10}B areal density. In Chapter 9 the qualification and on production tests for METAMIC® to support 90% ^{10}B credit are specified. With 90% credit, the target weight percent of boron carbide in METAMIC® is 31 for all MPCs, as summarized in Table 1.2.8, consistent with the test coupons used in the EPRI evaluations [1.2.11]. The maximum permitted value is 33.0 wt% to allow for necessary fabrication flexibility.

Because METAMIC® is a solid material, there is no capillary path through which spent fuel pool water can penetrate METAMIC® panels and chemically react with aluminum in the interior of the material to generate hydrogen. Any chemical reaction of the outer surfaces of the METAMIC® neutron absorber panels with water to produce hydrogen occurs rapidly and reduces to an insignificant amount in a short period of time. Nevertheless, combustible gas monitoring for METAMIC® -equipped MPCs and purging the space under the MPC lid during welding and cutting operations, is required until sufficient field experience is gained that confirms that little or no hydrogen is released by METAMIC® during these operations.

Mechanical properties of 31 wt.% METAMIC® based on coupon tests of the material in the as-fabricated condition and after 48 hours of an elevated temperature state at 900°F are summarized below from the EPRI report [1.2.11].

Mechanical Properties of 31wt.% B_4C METAMIC		
Property	As-Fabricated	After 48 hours of 900°F Temperature Soak
Yield Strength (psi)	32937 ± 3132	28744 ± 3246
Ultimate Strength (psi)	40141 ± 1860	34608 ± 1513
Elongation (%)	1.8 ± 0.8	5.7 ± 3.1

The required flexural strain of the neutron absorber to ensure that it will not fracture when the supporting basket wall flexes due to the worst case lateral loading is 0.2%, which is the flexural strain of the Alloy X basket panel material. The 1% minimum elongation of 31wt.% B_4C METAMIC® indicated by the above table means that a large margin of safety against cracking exists, so there is no need to perform testing of the METAMIC® for mechanical properties.

EPRI's extensive characterization effort [1.2.11], which was focused on 15 and 31 wt.% B_4C METAMIC® served as the principal basis for a recent USNRC SER for 31wt.% B_4C METAMIC for used in wet storage [1.2.12]. Additional studies on METAMIC® [1.2.13], EPRI's and others work provide the confidence that 31wt.% B_4C METAMIC® will perform its intended function in the

MPCs.

1.2.1.3.1.3 Locational Fixity of 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. 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 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 (only for the HI-STORM 100 and –100S overpack designs). 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 B_4C 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/cm^3 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/cm^3. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm^3 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 B_4C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal B_4C weight percent of 1%.

Design Temperature

The design temperatures of Holtite-A are provided in Table 1.B.1.. The maximum spatial temperatures of Holtite-A under all normal operating conditions must be demonstrated to be below these design temperatures, as applicable.

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 pedestal (HI-STORM 100 and –100S overpack designs only) and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are 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. See Figure 5.3.19 and the drawings in Section 1.5 for more details of the gamma shield cross plate designs for each overpack design.

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 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 affected 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 100D and 125D are 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 shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated.

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 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, 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 (borated if necessary). Based on the MPC model and fuel enrichment, this may be borated water or plant demineralized water (see Section 2.1). 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. 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 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. At the appropriate time in the sequence of activities, based on the type of test performed (hydrostatic or pneumatic), a pressure test of the MPC enclosure vessel is performed.

For MPCs containing all moderate burnup fuel, 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 and as an option for storage of moderate burnup fuel, 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 demoisturizing module described in Appendix 2.B meets the specified limit.

Following moisture removal, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage and provides an inert atmosphere for long-term fuel integrity. 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 100D or 125D, the HI-STORM mating device is placed (bolted if required by generic or site specific seismic evaluation) 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 positioned (bolted if required by generic or site specific seismic evaluation) atop 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 100D and 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 100D and 125D, the transfer cask must first be disconnected from the mating device). For the HI-STORM 100S and HI-STORM 100S Version B, 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 100D or 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 100D and 125D, the mating device is positioned (bolted if required by generic or site specific seismic evaluation) atop 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 100D and 125D do not include pocket trunnions in their designs. 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.

For MPCs containing any HBF and a decay heat load that would yield a peak HBF cladding temperature above the short-term temperature limit, the Supplemental Cooling System (SCS) is required to be operational during the time the loaded and backfilled MPC is in HI-TRAC to ensure fuel cladding temperatures remain within limits. The SCS is discussed in detail in Section 4.5 and the design criteria for the system are provided in Appendix 2.C. The SCS is not required when the MPC is inside the HI-STORM overpack, regardless of decay heat load.

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 100D and 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 and the HI-STORM 100S Version B, 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 100D or 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 100D and 125D, the mating device is installed. Rigging is attached to the MPC lift cleats. For the HI-STORM 100S and HI-STORM 100S Version B with the standard HI-TRAC design, the transfer doors may need to be opened to avoid interfering with the MPC lift cleats. For the HI-TRAC 100D and 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 bolted (if necessary) to the mating device, as applicable. For the HI-TRAC 100D and 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 the HI-TRAC 100D and 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. As required based on the presence of high burnup fuel and the decay heat load, the Supplemental Cooling System is installed and placed into operation.

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 flooded with borated or unborated water, as required. 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-24/24E/24EF (all with lower enriched fuel) and the MPC-68/68F/68FF 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 neutron absorber plates affixed to the fuel cell walls as shown on the drawings in Section 1.5. The minimum ^{10}B areal density specified for the 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, non-destructively examined and pressure 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 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 system data and parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in Table 1.0.1. A summary of the types of fuel authorized for storage in each MPC model is provided below. All fuel assemblies, non-fuel hardware, and neutron sources must meet the fuel specifications provided in Section 2.1. 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 and MPC-24EF

The MPC-24E and MPC-24EF are 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 and MPC-32F

The MPC-32 and MPC-32F are 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-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-68 and MPC-68FF

The MPC-68 and MPC-68FF are 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	8	5 for PWR 3 for BWR
MPC storage capacity[†]:	MPC-24 MPC-24E MPC-24EF	Up to 24 intact ZR or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to four damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24E or MPC-24EF OR
	MPC-32 MPC-32F	Up to 32 intact ZR or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to 8 damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-32 or MPC-32F.

[†] See Section 2.1 for a complete description of authorized cask contents and fuel specifications.

Table 1.2.1 (continued)
KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
MPC storage capacity:	MPC-68F	Up to 4 damaged fuel containers with ZR clad Dresden Unit 1 (D-1) or Humboldt Bay (HB) BWR fuel debris and the complement damaged ZR clad Dresden Unit 1 or Humboldt Bay BWR fuel assemblies in damaged fuel containers or intact Dresden Unit 1 or Humboldt Bay BWR intact fuel assemblies. OR
	MPC-68 MPC-68FF	Up to 68 Dresden Unit 1 or Humboldt Bay intact fuel or damaged fuel and up to 8 damaged fuel containers containing D-1 or HB fuel debris. For other BWR plants, up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris with the complement intact fuel assemblies, up to a total of 68. The number of damaged fuel containers containing BWR fuel debris is limited to eight (8) for all BWR plants.

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 Off-normal/Short-term conditions Accident Conditions	 100 110 200	 100 110 200
Total heat load, max. (kW)	36.9	36.9
Maximum permissible peak fuel cladding temperature: Long Term Normal (°F) Short Term Operations (°F) Off-normal and Accident (°F)	 752 752 or 1058††† 1058	 752 752 or 1058††† 1058

† 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.

††† See Section 4.5 for discussion of the applicability of the 1058°F temperature limit during MPC drying.

Table 1.2.2 (cont'd)

KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
MPC internal environment Helium fill (99.995% fill helium purity) (See Note 2)	(all pressure ranges are at a reference temperature of 70°F)	(all pressure ranges are at a reference temperature of 70°F)
MPC-24 (heat load ≤ 27.77 kW)	≥ 29.3 psig and ≤ 48.5 psig OR 0.1212 +/-10% g-moles/liter	
(heat load > 27.77 kW)	≥ 45.5 psig and ≤ 48.5 psig	
MPC-24E/24EF (heat load ≤ 28.17 kW)	≥ 29.3 psig and ≤ 48.5 psig OR 0.1212 +/-10% g-moles/liter	
(heat load > 28.17 kW)	≥ 45.5 psig and ≤ 48.5 psig	
MPC-68/68F/68FF (heat load ≤ 28.19 kW)		≥ 29.3 psig and ≤ 48.5 psig OR 0.1218 +/-10% g-moles/liter
(heat load >28.19 kW)		≥ 45.5 psig and ≤ 48.5 psig
MPC-32/32F (heat load ≤ 28.74 kW)	≥ 29.3 psig and ≤ 48.5 psig	
(heat load >28.74 kW)	≥ 45.5 psig and ≤ 48.5 psig	
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
Fixed Neutron Absorber ^{10}B Areal Density (g/cm^2) Boral/Metamic	0.0267/0.0223 (MPC-24) 0.0372/0.0310 (MPC-24E, MPC-24EF MPC-32 & MPC-32F)	0.0372/0.0310 (MPC-68 & MPC-68FF) 0.01/NA (MPC-68F) (See Note 1)
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

NOTE:
1. All MPC-68F canisters are equipped with Boral neutron absorber.
2. Refer to Section 4.4.5.1 for detailed information on heat load values.

Table 1.2.3

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Table 1.2.4

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Table 1.2.5

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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, and pressure and leakage 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 100D or 125D)
9	MPC lowered and seated on HI-TRAC transfer lid (not applicable to HI-TRAC 100D or 125D)
9a	HI-STORM mating device secured to top of empty HI-STORM overpack (HI-TRAC 100D and 125D only)
10	HI-TRAC/MPC assembly transferred to atop the 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 100D and 125D)
12	MPC lowered into HI-STORM overpack, and HI-TRAC removed from atop the HI-STORM overpack/mating device
12a	HI-STORM mating device removed (HI-TRAC 100D and 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"
¾ Cr	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 – ½ Mo	SA-193	B7	125	105	t ≤ 2.5"
$2N_i$ – ¾ Cr – ¼ Mo	SA-540	B23 (H-43400)	135	120	
$2N_i$ – ¾ Cr – 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.

Table 1.2.8

METAMIC® DATA FOR HOLTEC MPCs

MPC Type	Min. B-10 areal density required by criticality analysis (g/cm^2)	Nominal Weight Percent of B_4C and Reference *METAMIC®* Panel Thickness			
		100% Credit	90% Credit	75% Credit	Ref. Thickness (inch) (see note)
MPC-24	0.020	27.6	31	37.2	0.075
MPC-68, -68FF, -32, -32F, -24E, and -24EF	0.0279	27.8	31	37.4	0.104

Note: The drawings in Section 1.5 show slightly larger thickness to ensure that the minimum B-10 areal density is conservative under all conditions.

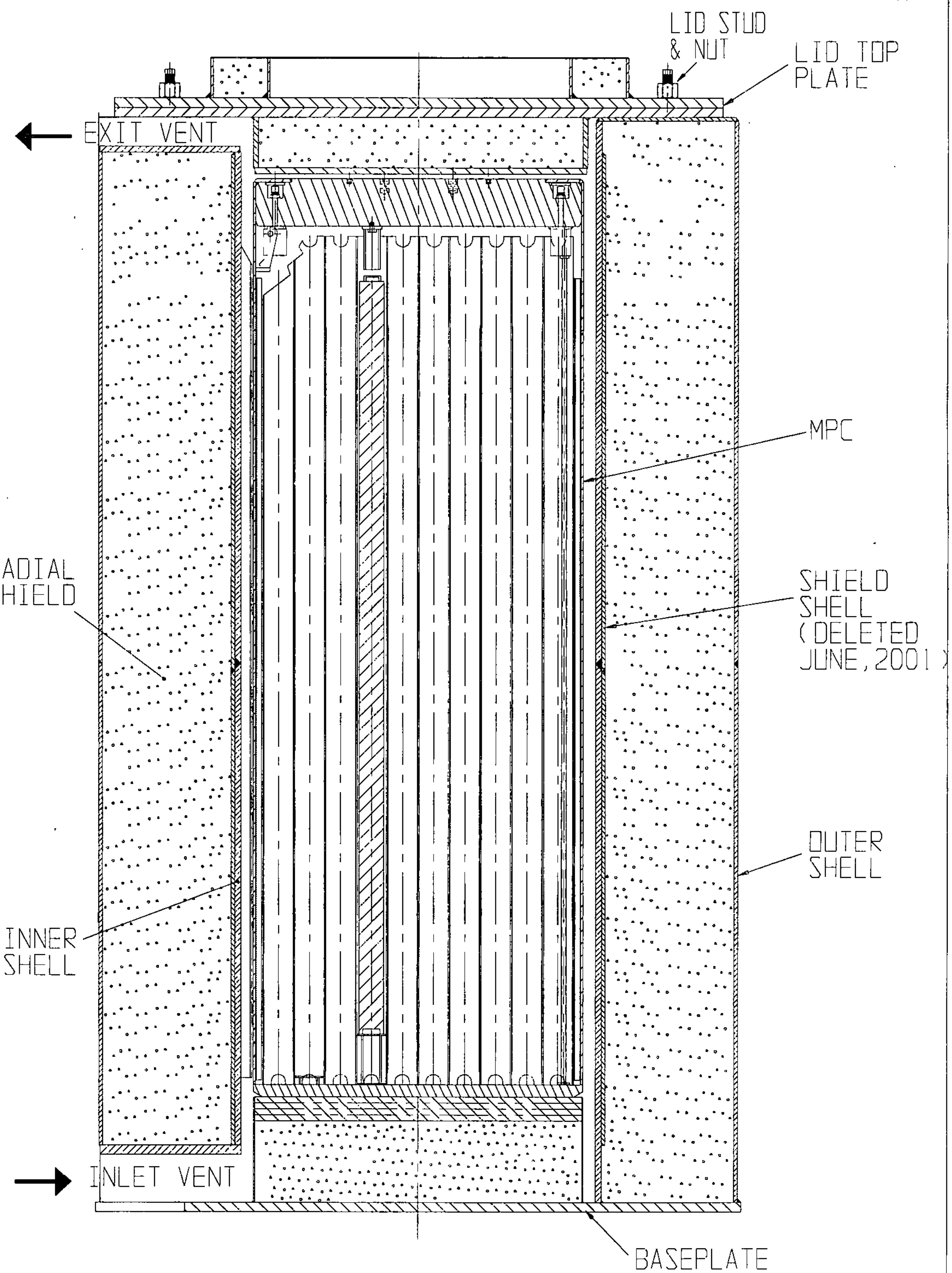

FIGURE 1.2.1; CROSS SECTION VIEW OF THE HI-STORM 100 SYSTEM

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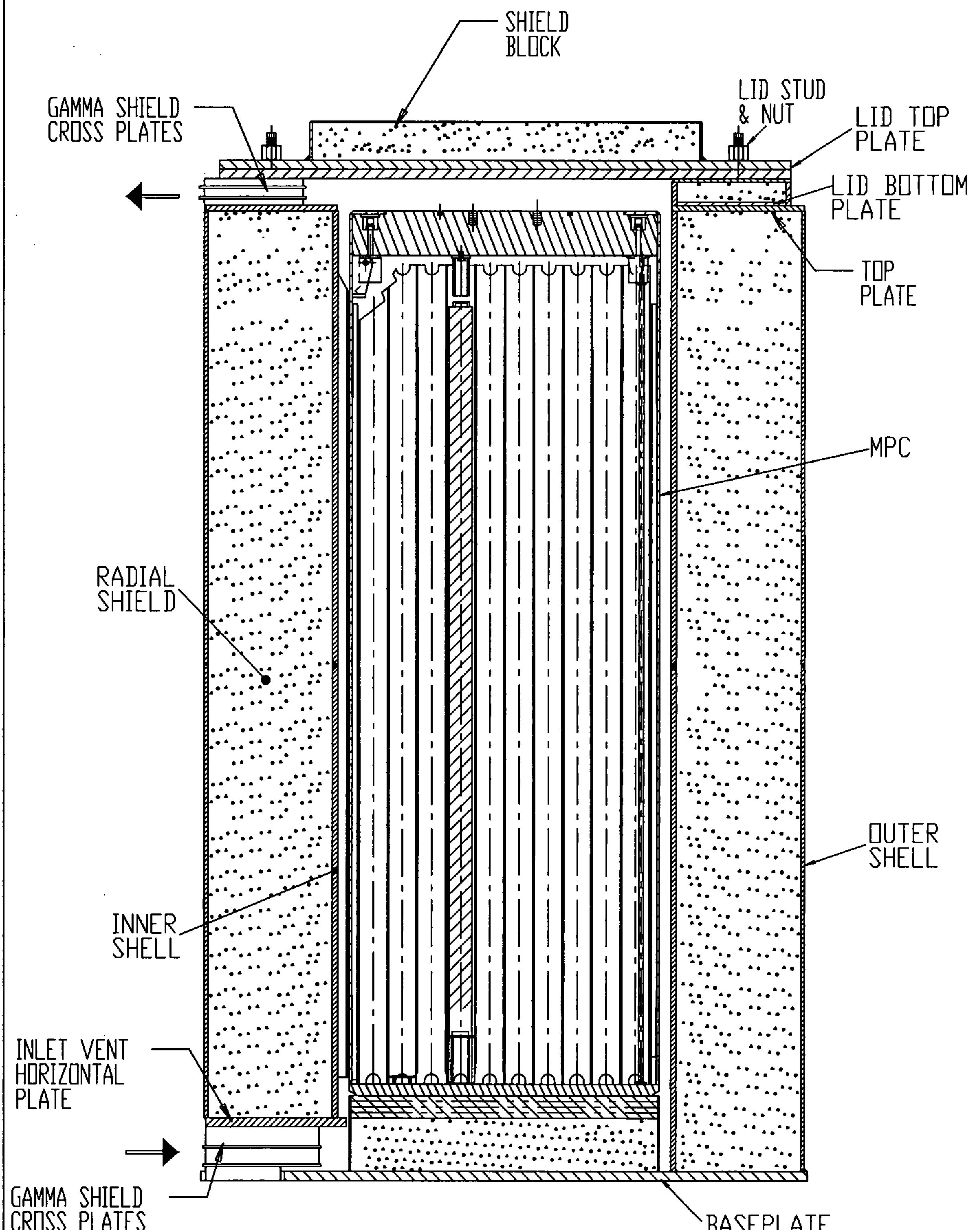

FIGURE 1.2.1A; CROSS SECTION VIEW OF THE HI-STORM 100S SYSTEM

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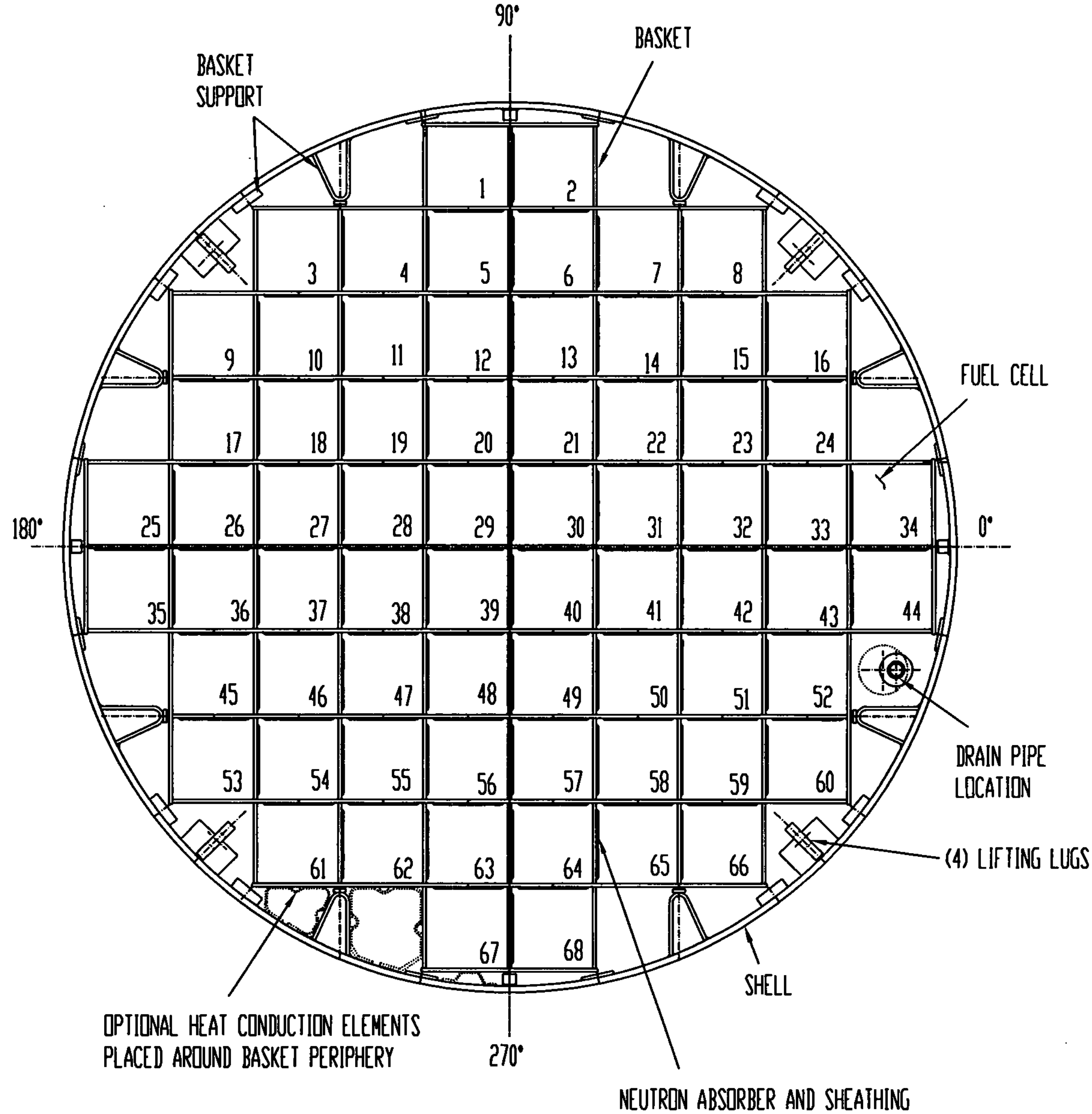

FIGURE 1.2.2; MPC-68/68F/68FF CROSS SECTION VIEW

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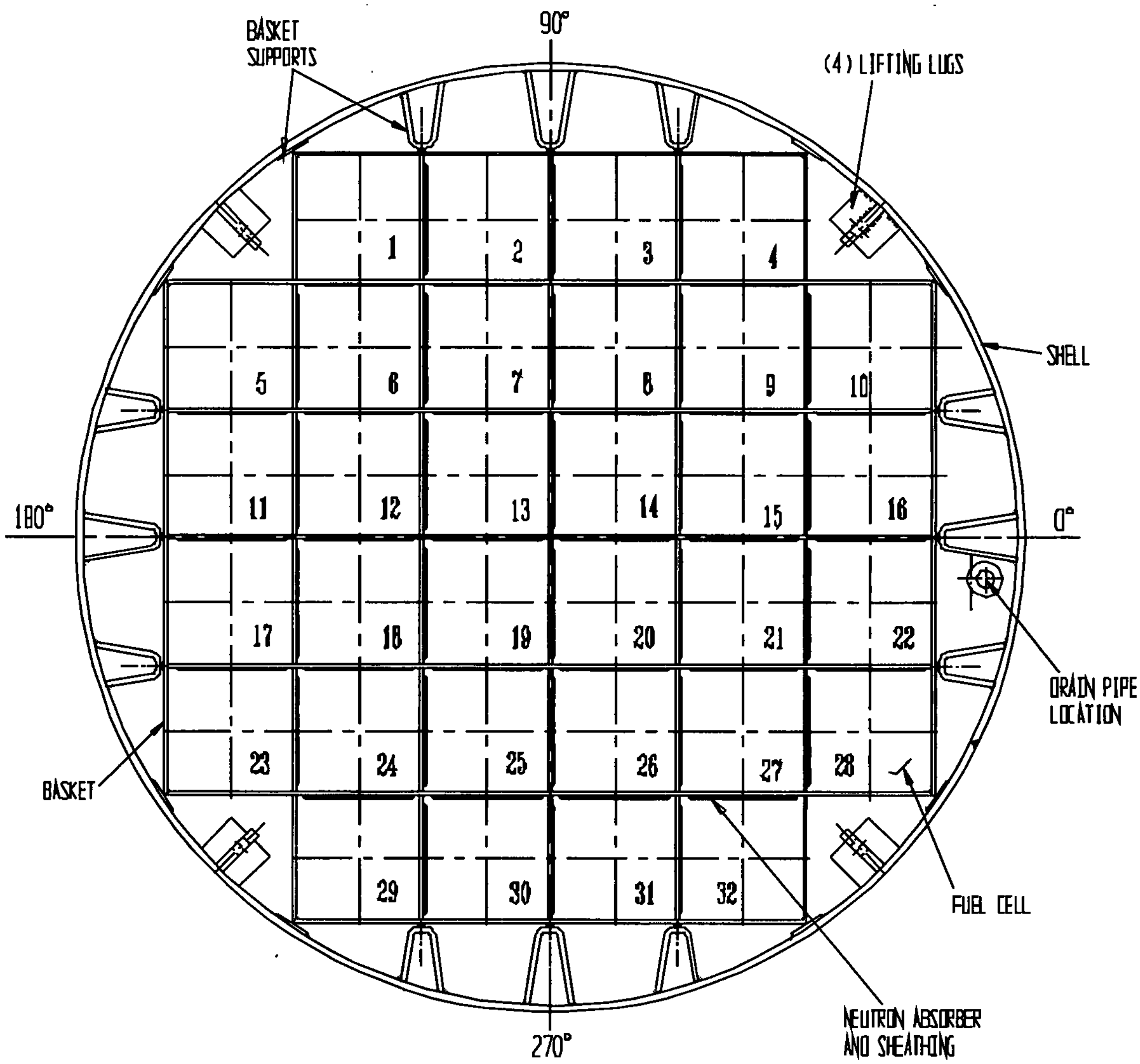

FIGURE 1.2.3; MPC-32/32F CROSS SECTION

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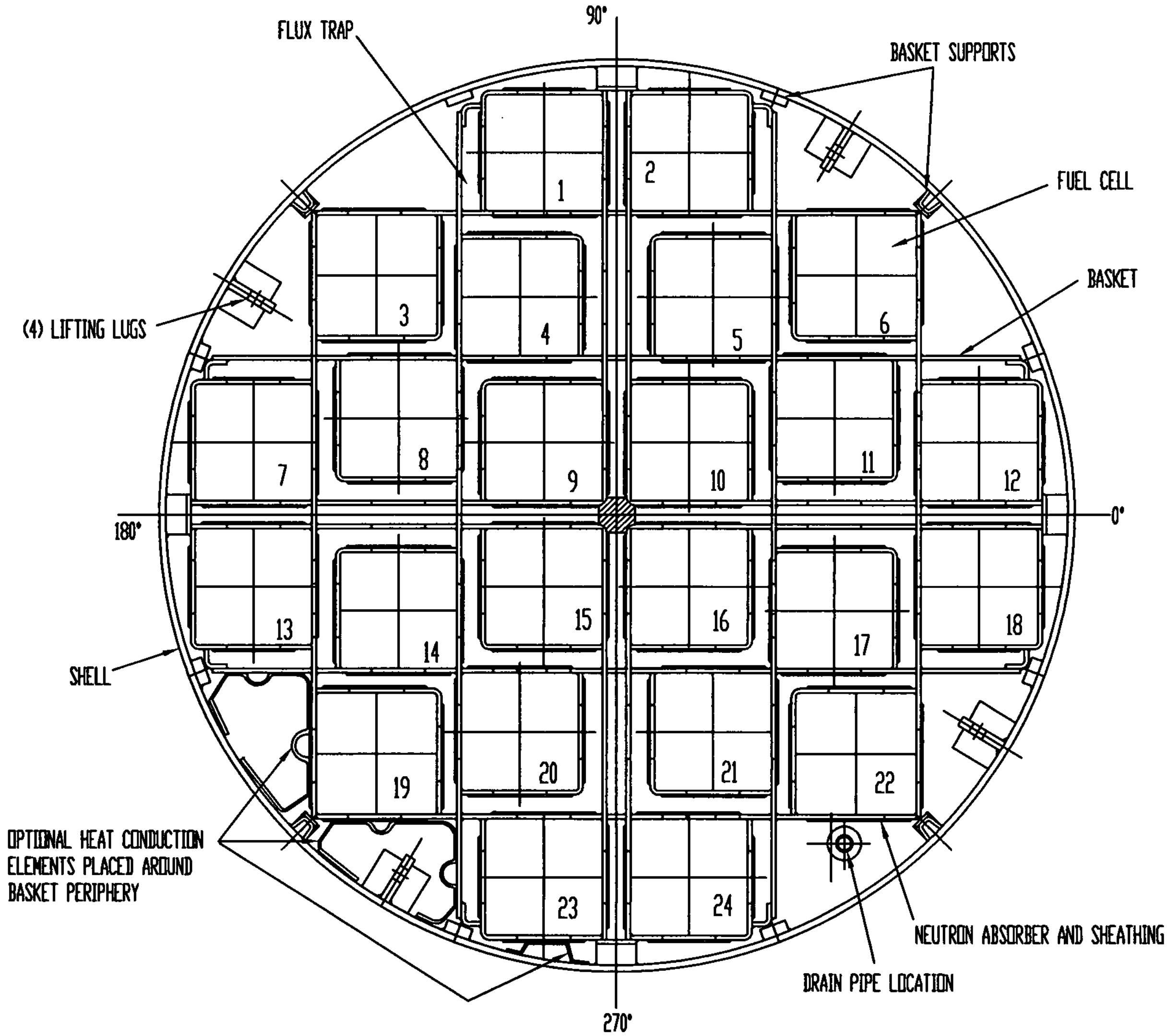

FIGURE 1.2.4j MPC-24/24E/24EF CROSS SECTION VIEW

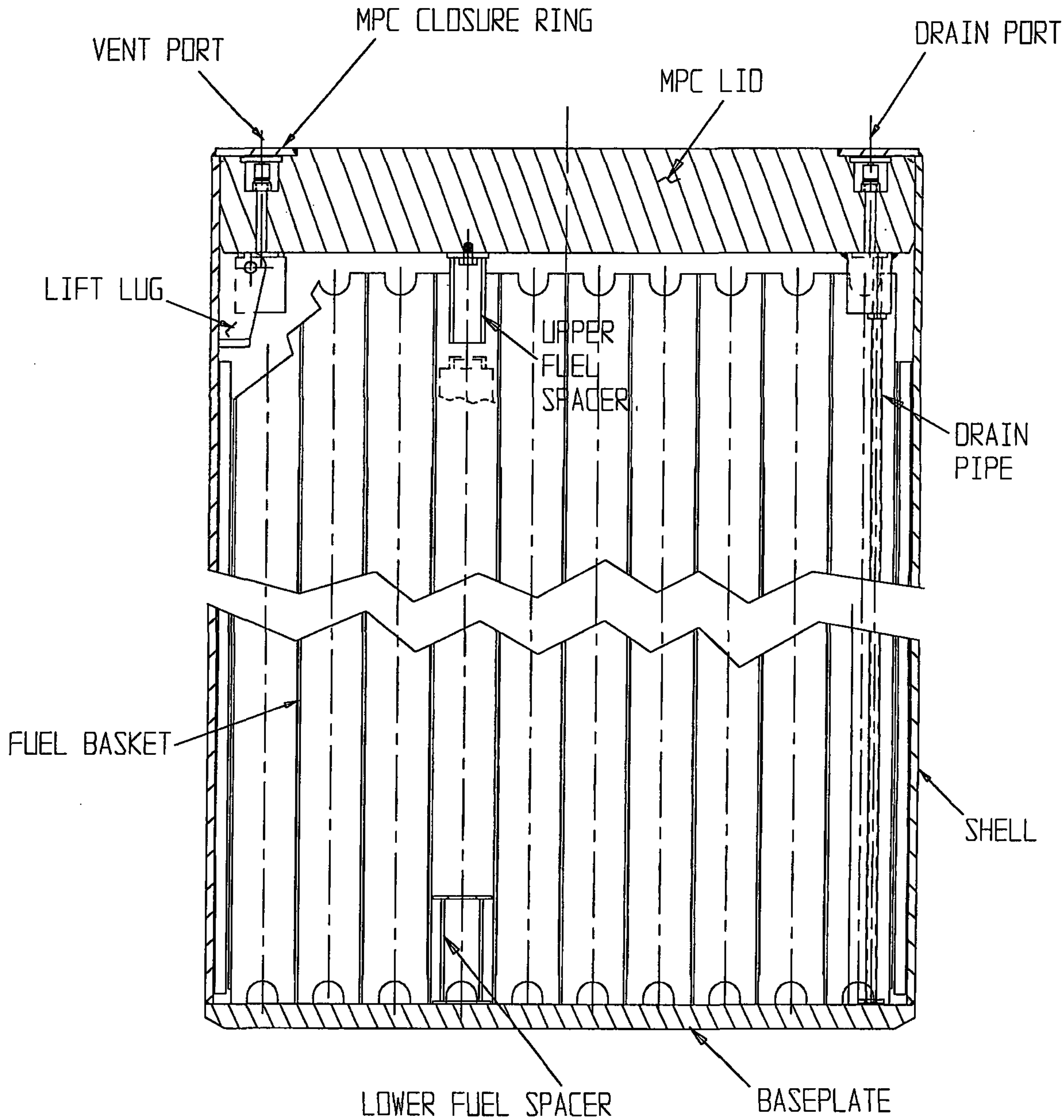

FIGURE 1.2.5; CROSS SECTION ELEVATION VIEW OF MPC

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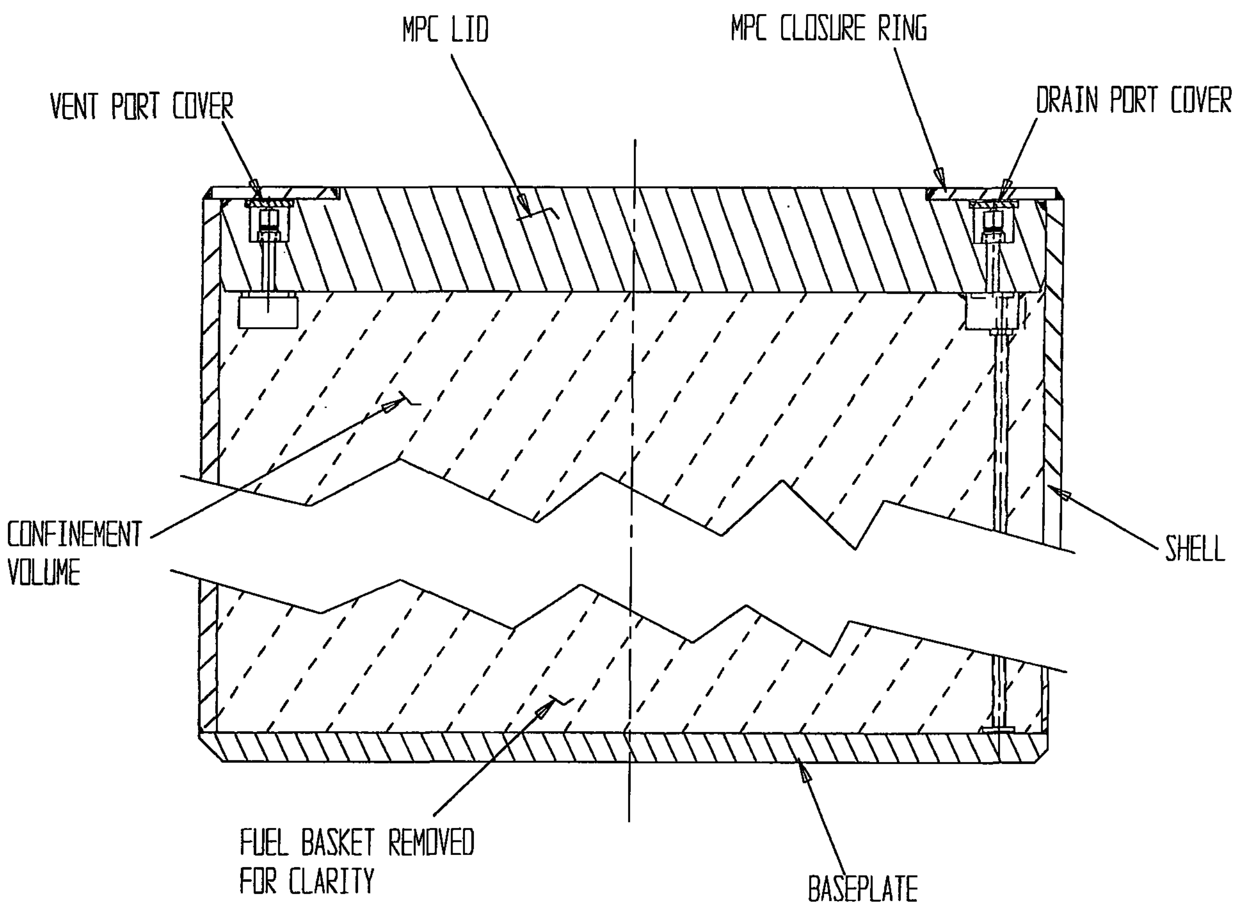

FIGURE 1.2.6; MPC CONFINEMENT BOUNDARY

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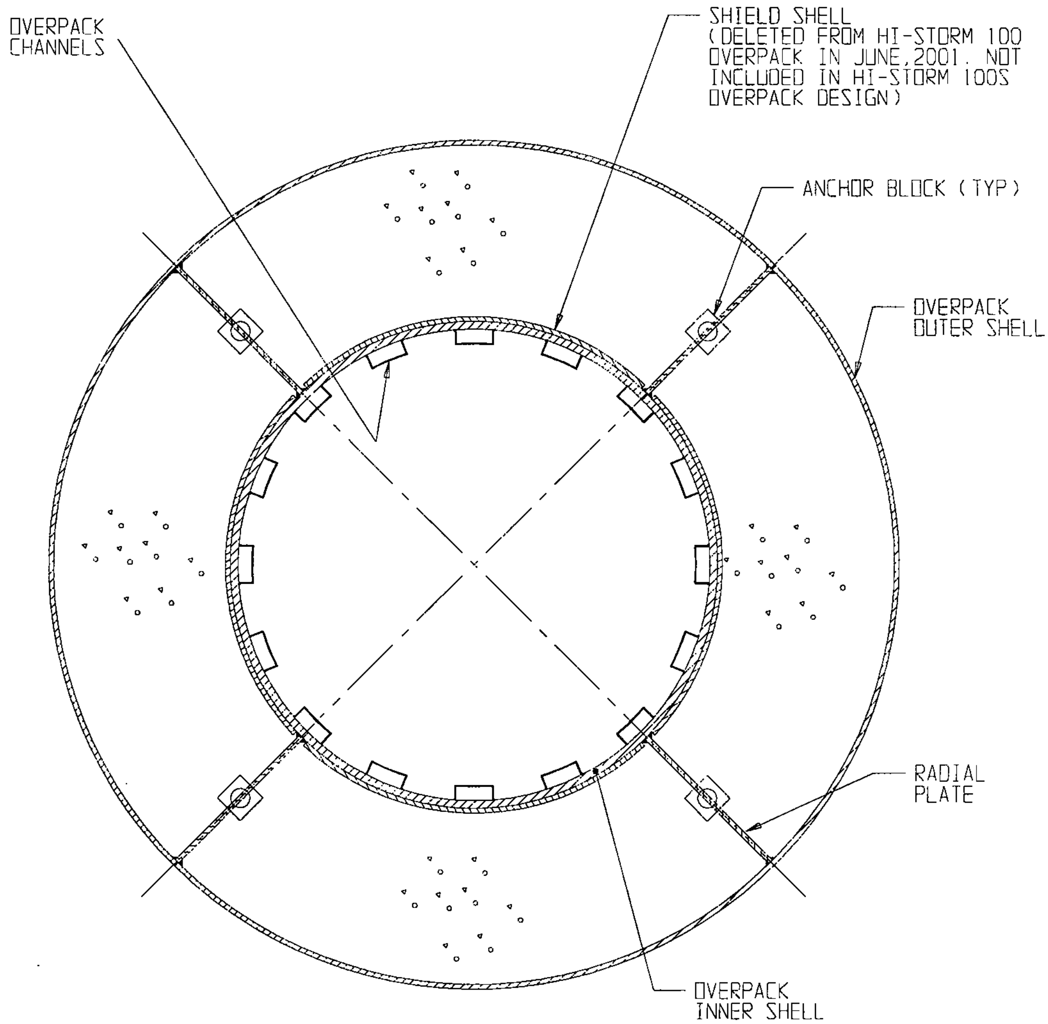

FIGURE 1.2.7; CROSS SECTION OF HI-STORM OVERPACK

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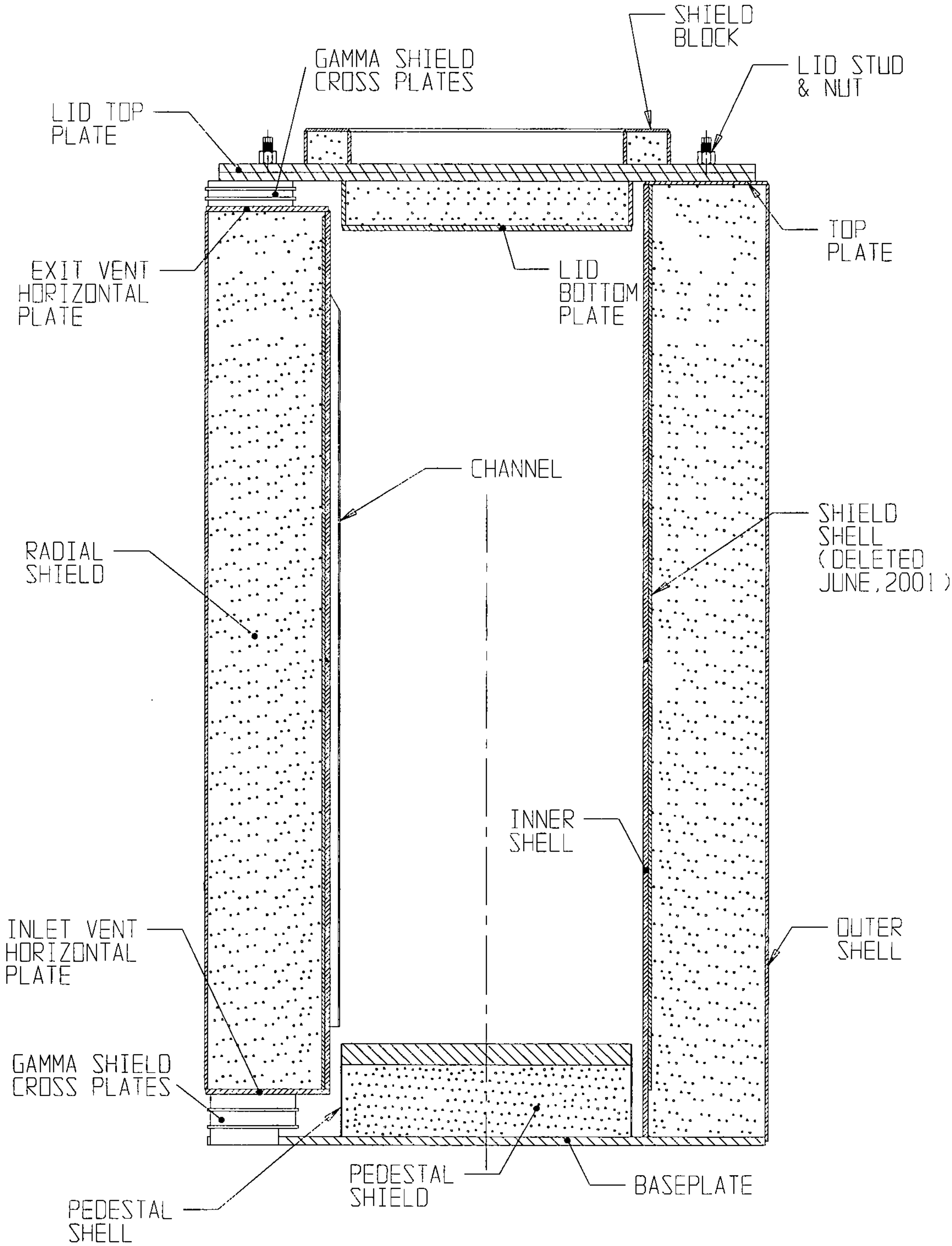

FIGURE 1.2.8; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

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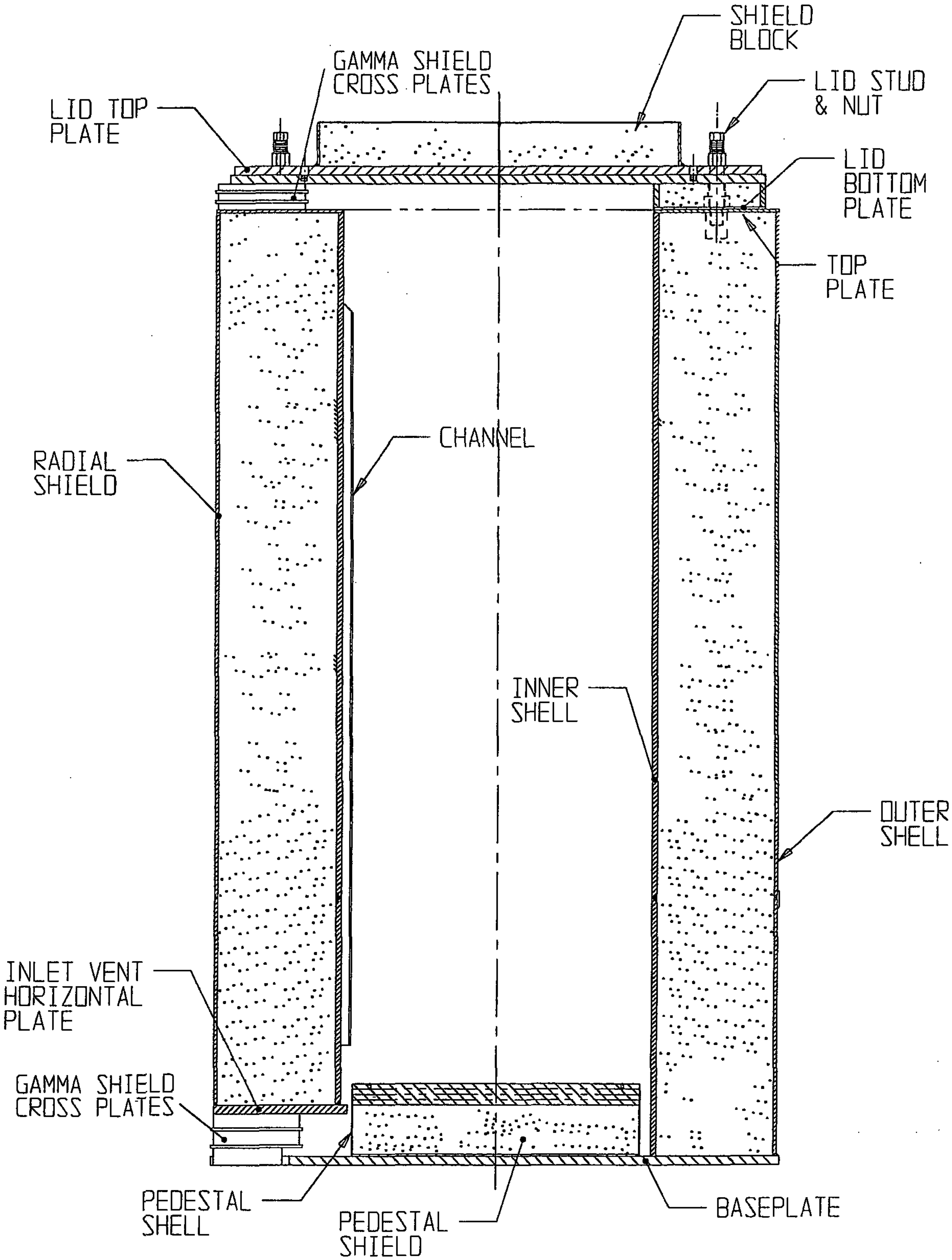

FIGURE 1.2.8A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

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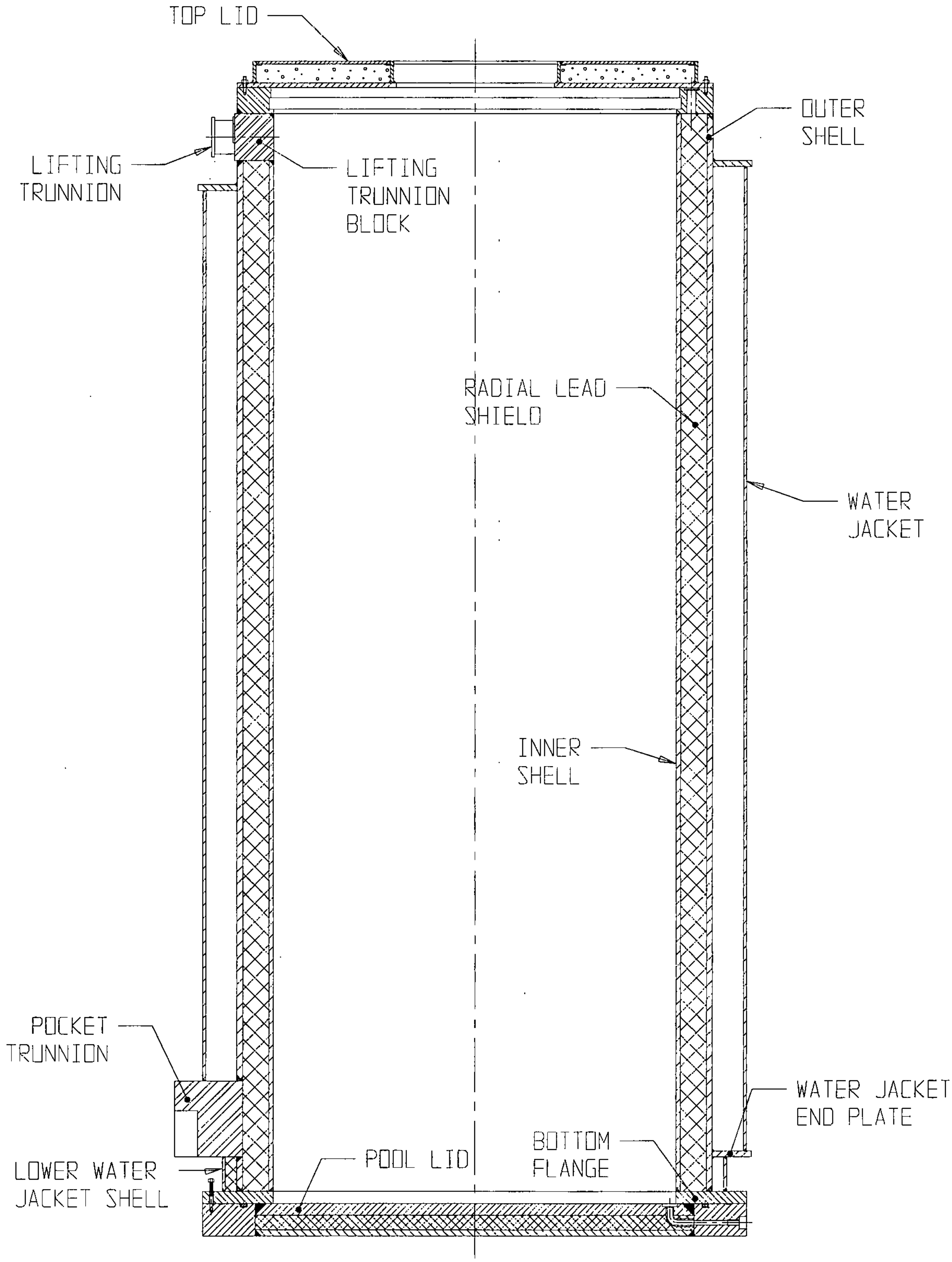

FIGURE 1.2.9; 125TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW

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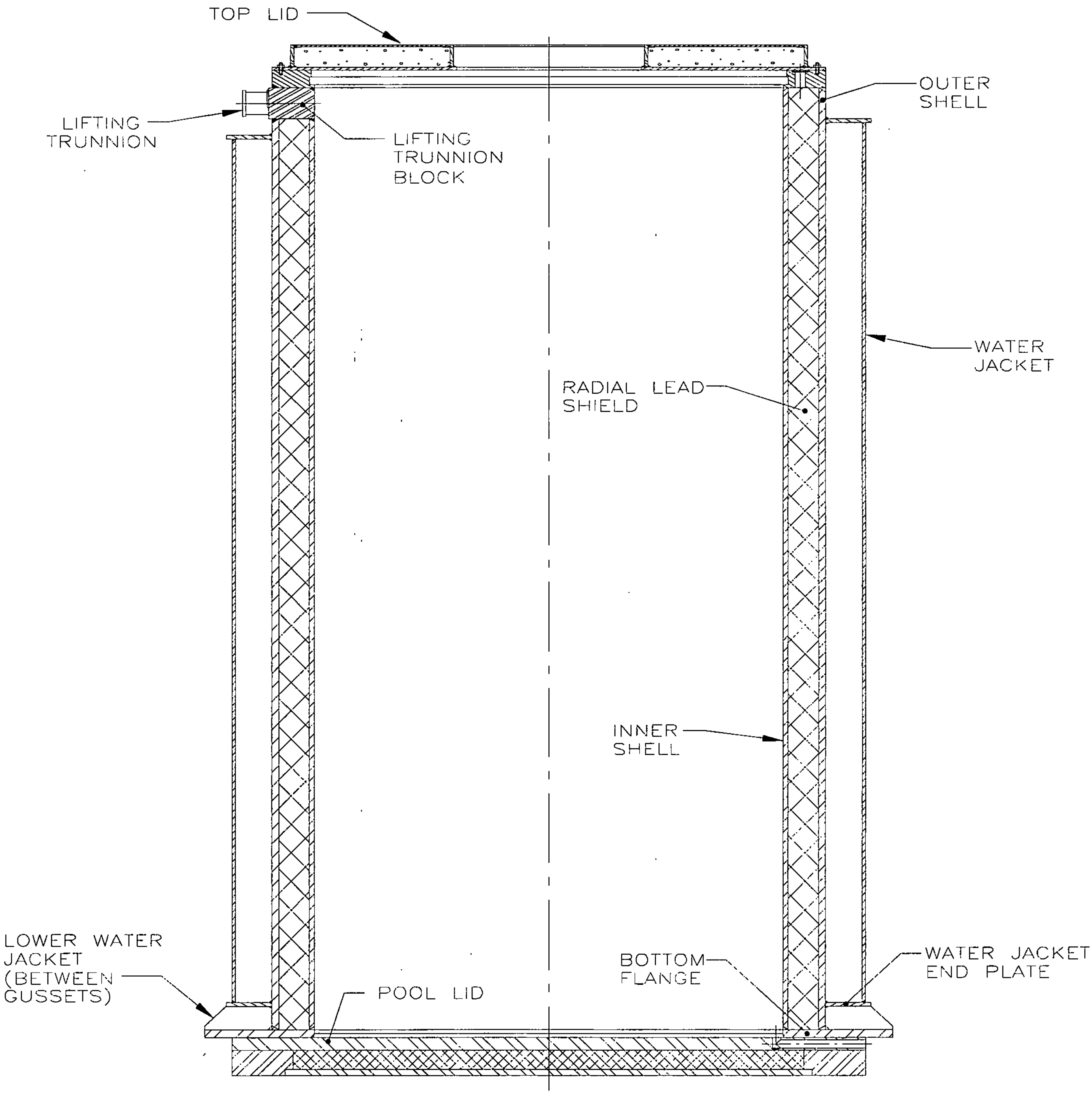

FIGURE 1.2.9A; HI-TRAC 125D TRANSFER CASK CROSS SECTIONAL ELEVATION VIEW

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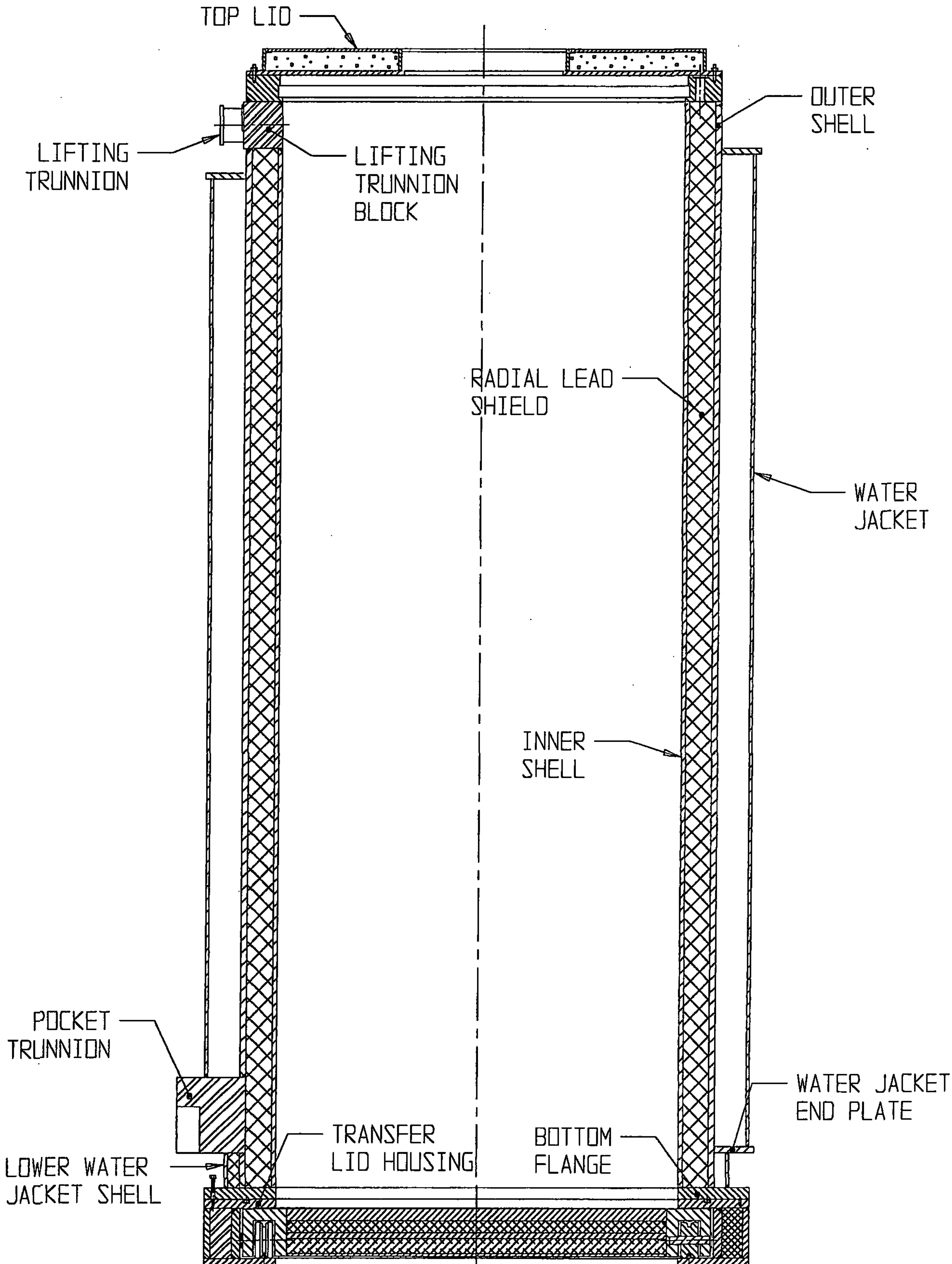

FIGURE 1.2.10; 125TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW

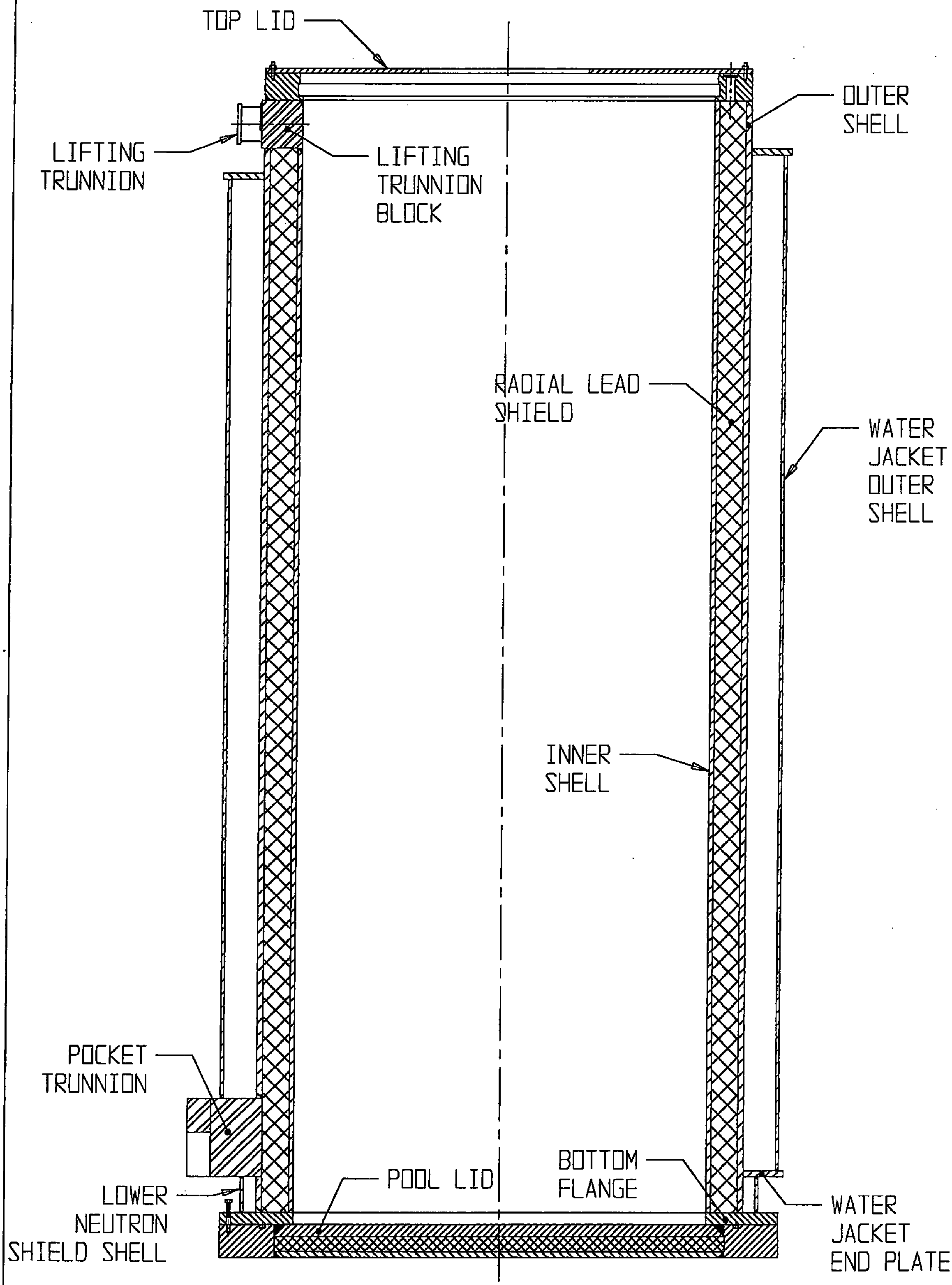

FIGURE 1.2.11; 100 HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW

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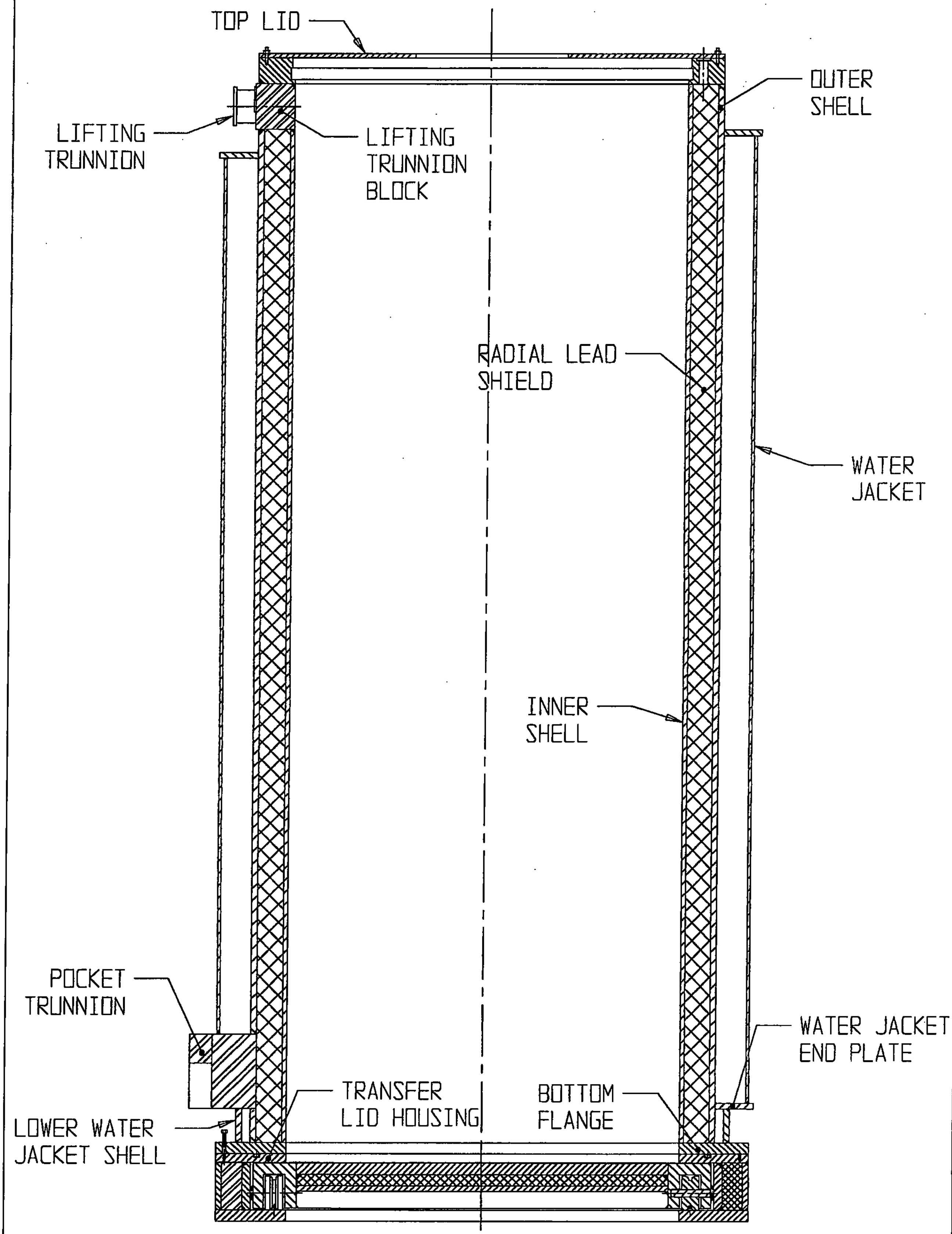

FIGURE 1.2.12; 100 TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW

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FIGURE 1.2.13; DELETED

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FIGURE 1.2.14; DELETED

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FIGURE 1.2.15; DELETED

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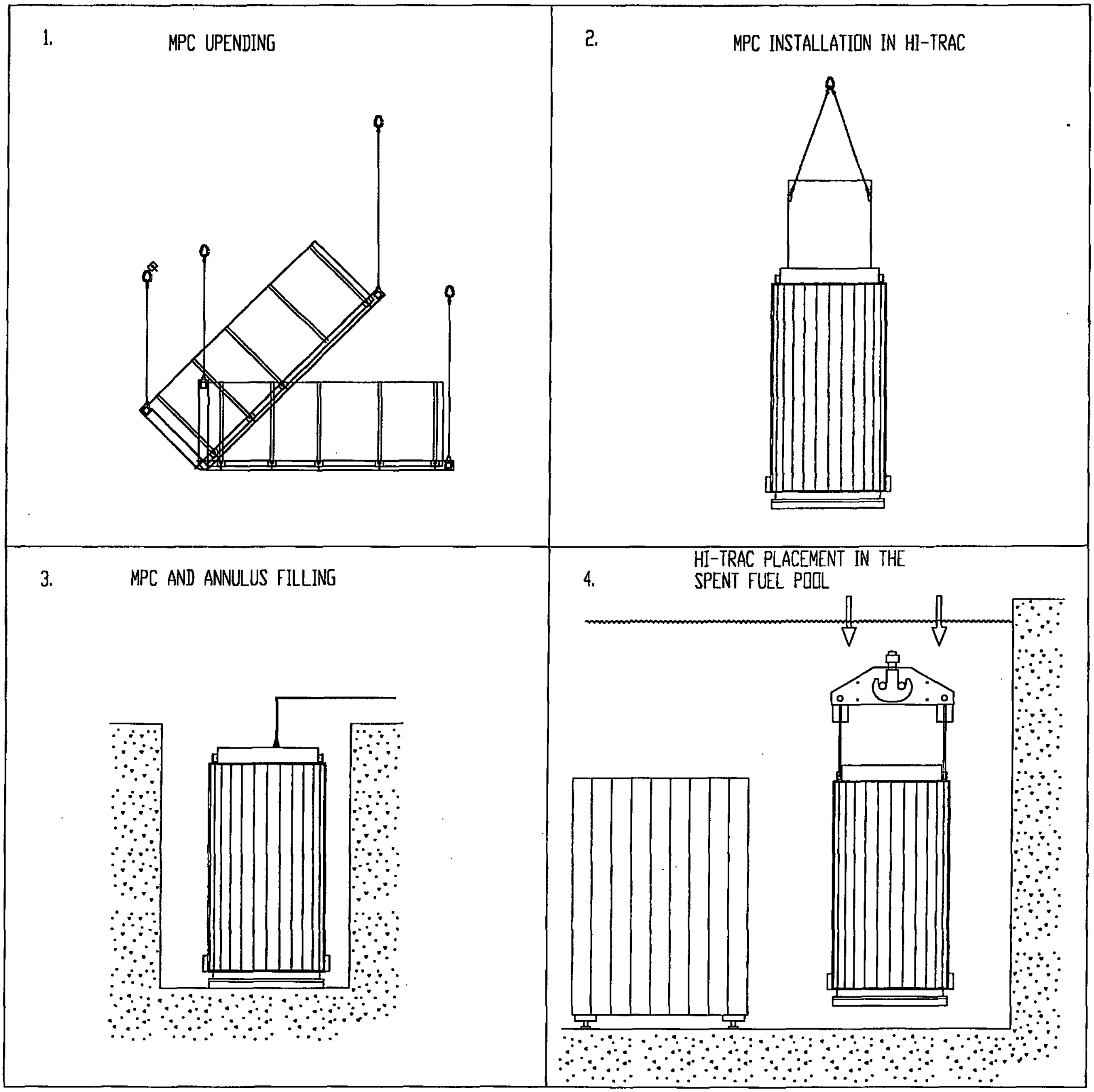

Figure 1.2.16a; Major HI-STORM 100 Loading Operations (Sheet 1 of 6)

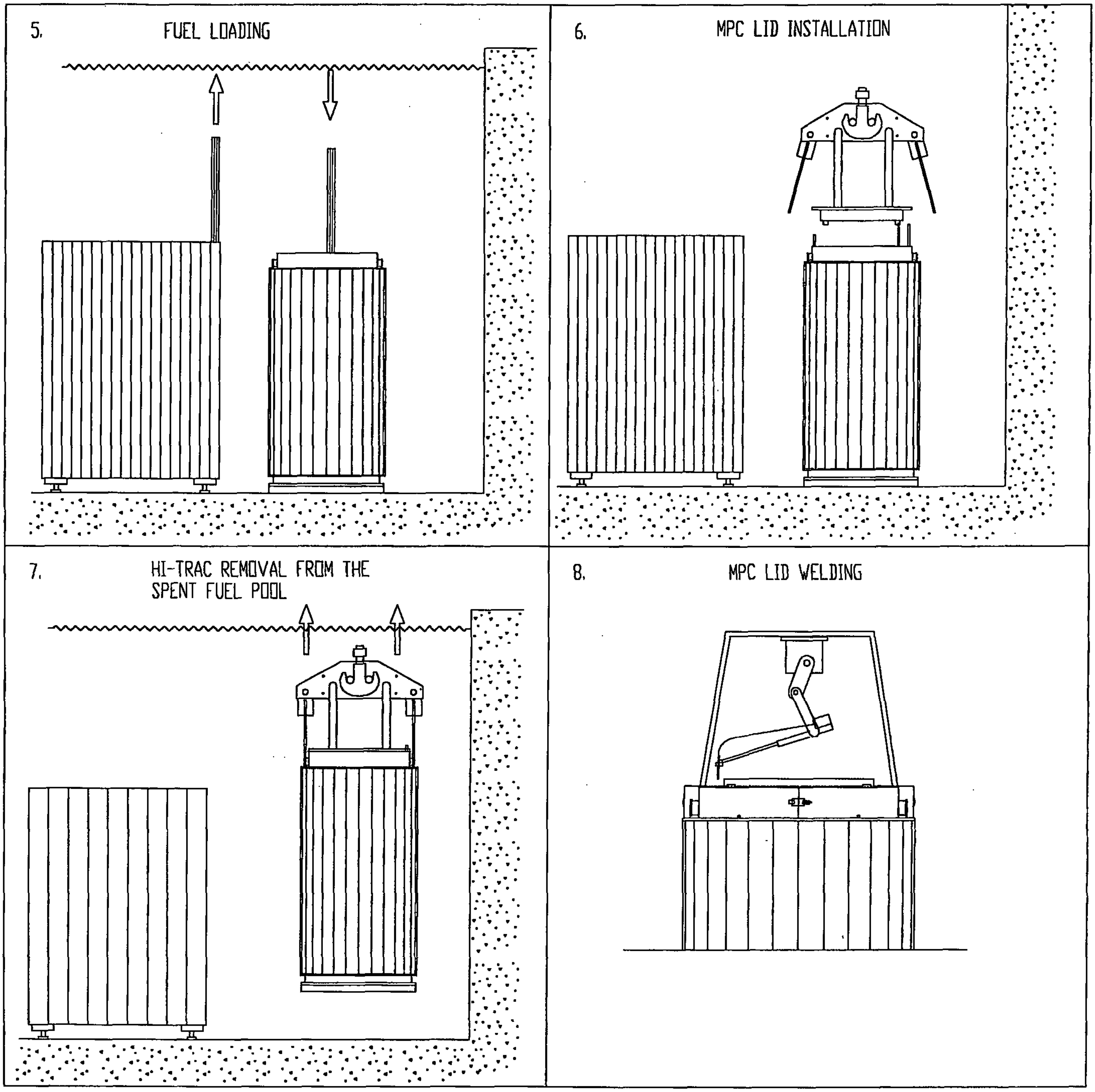

Figure 1.2.16b; Major HI-STORM 100 Loading Operations (Sheet 2 of 6)

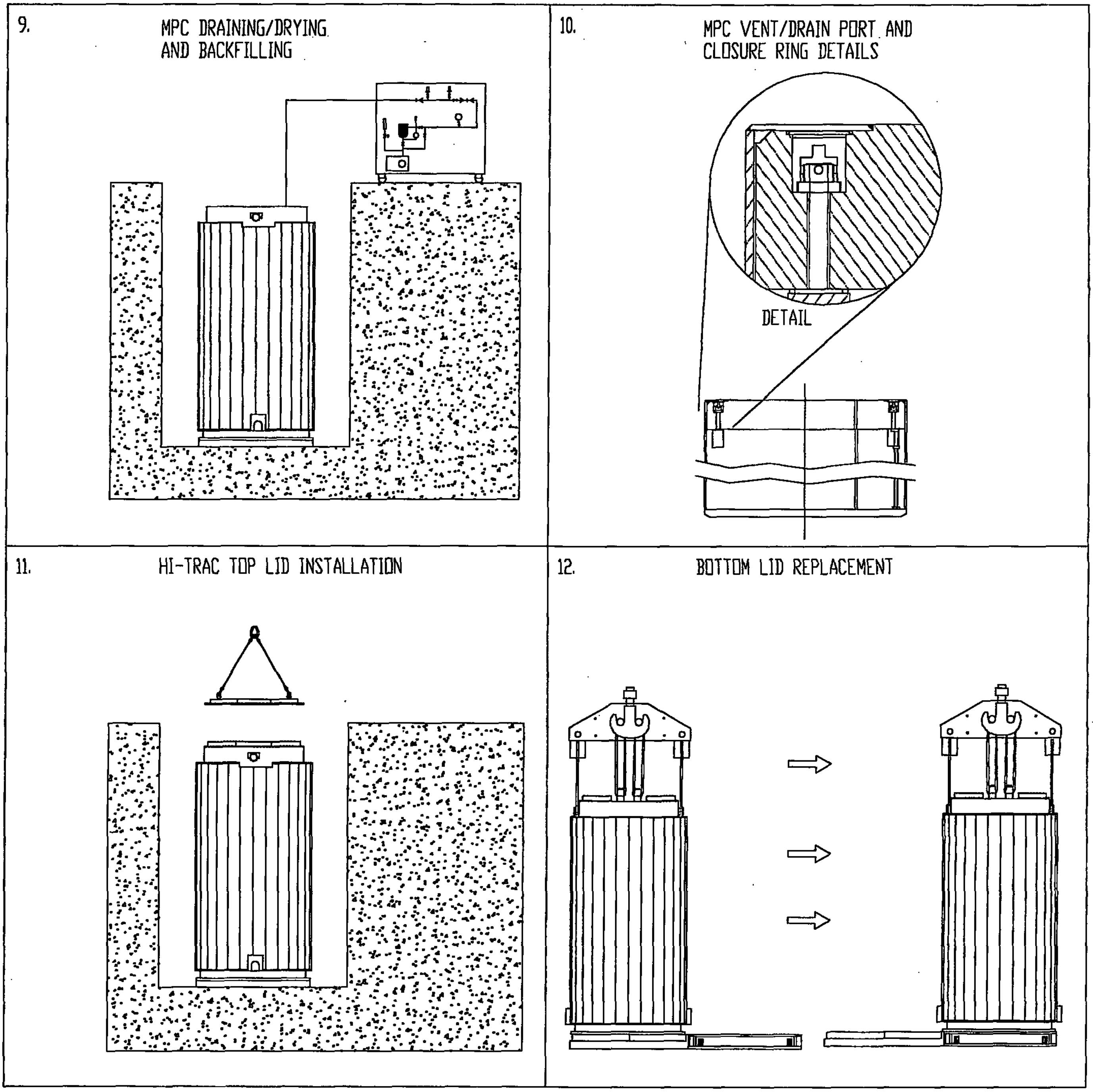

Figure 1.2.16c; Major HI-STORM 100 Loading Operations (Sheet 3 of 6)

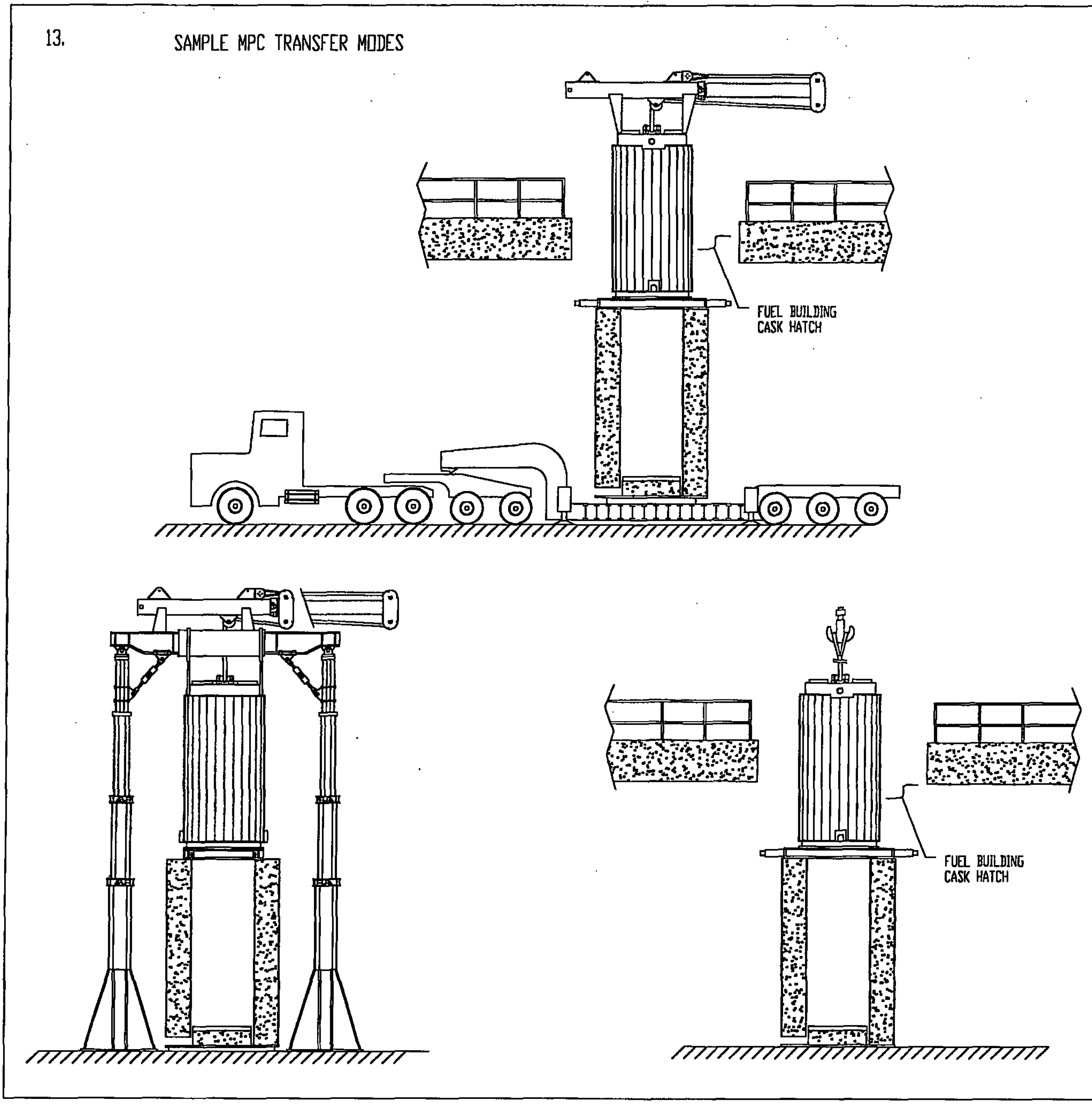

Figure 1.2.16d; Major HI-STORM 100 Loading Operations (Sheet 4 of 6)

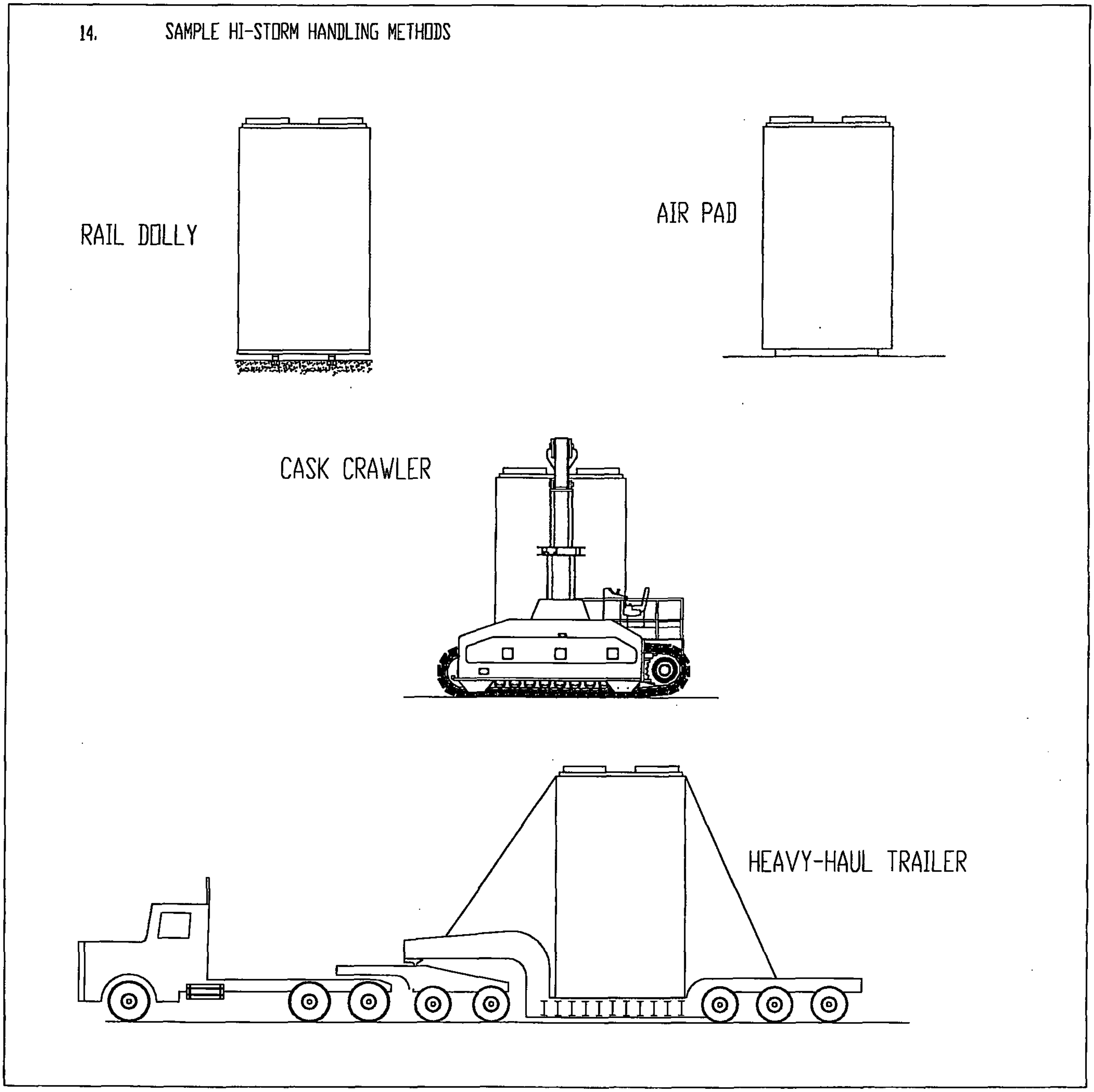

Figure 1.2.16e; Example of HI-STORM 100 Handling Options (Sheet 5 of 6)

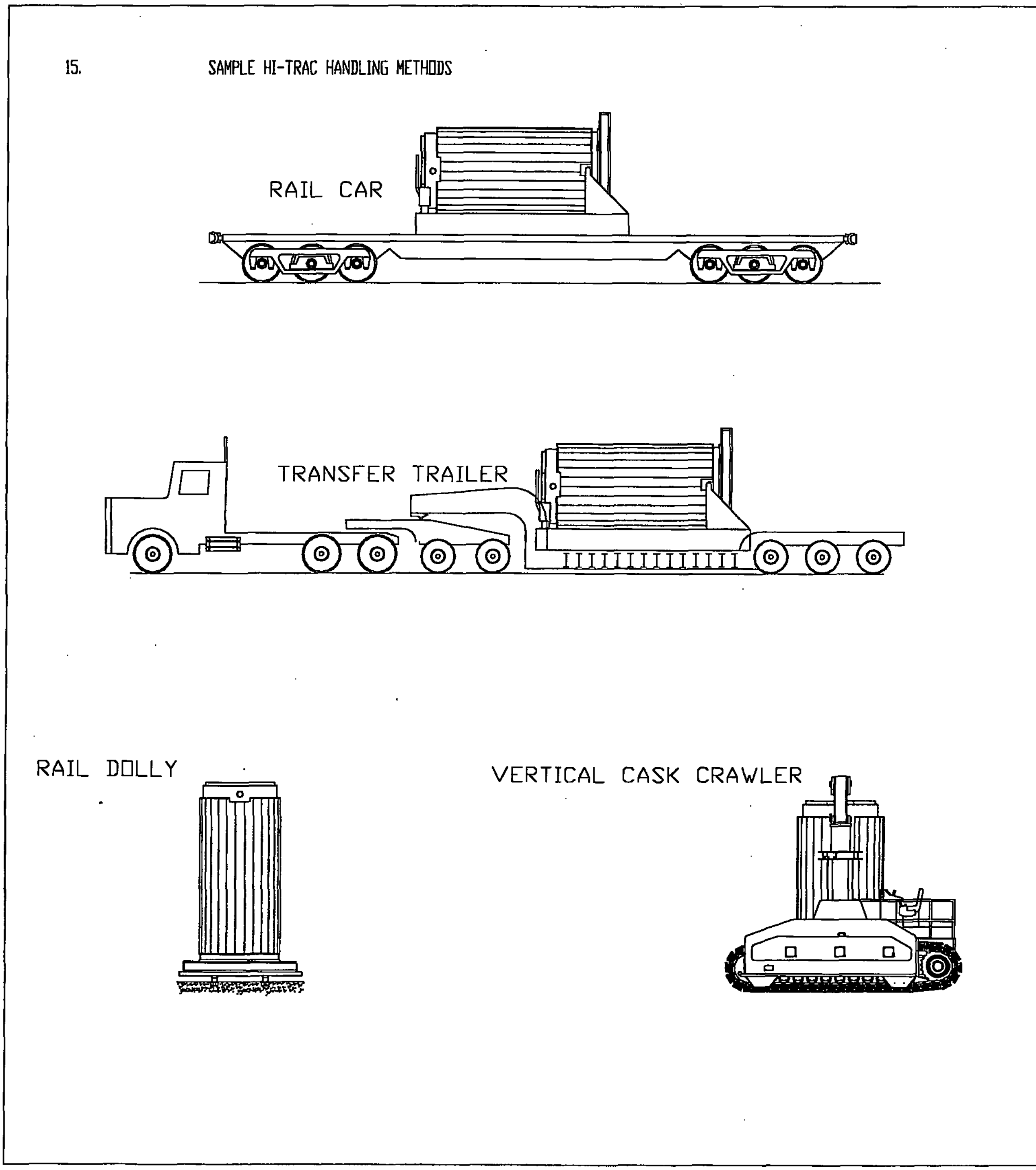

Figure 1.2.16f; Example of HI-TRAC Handling Options (Sheet 6 of 6)

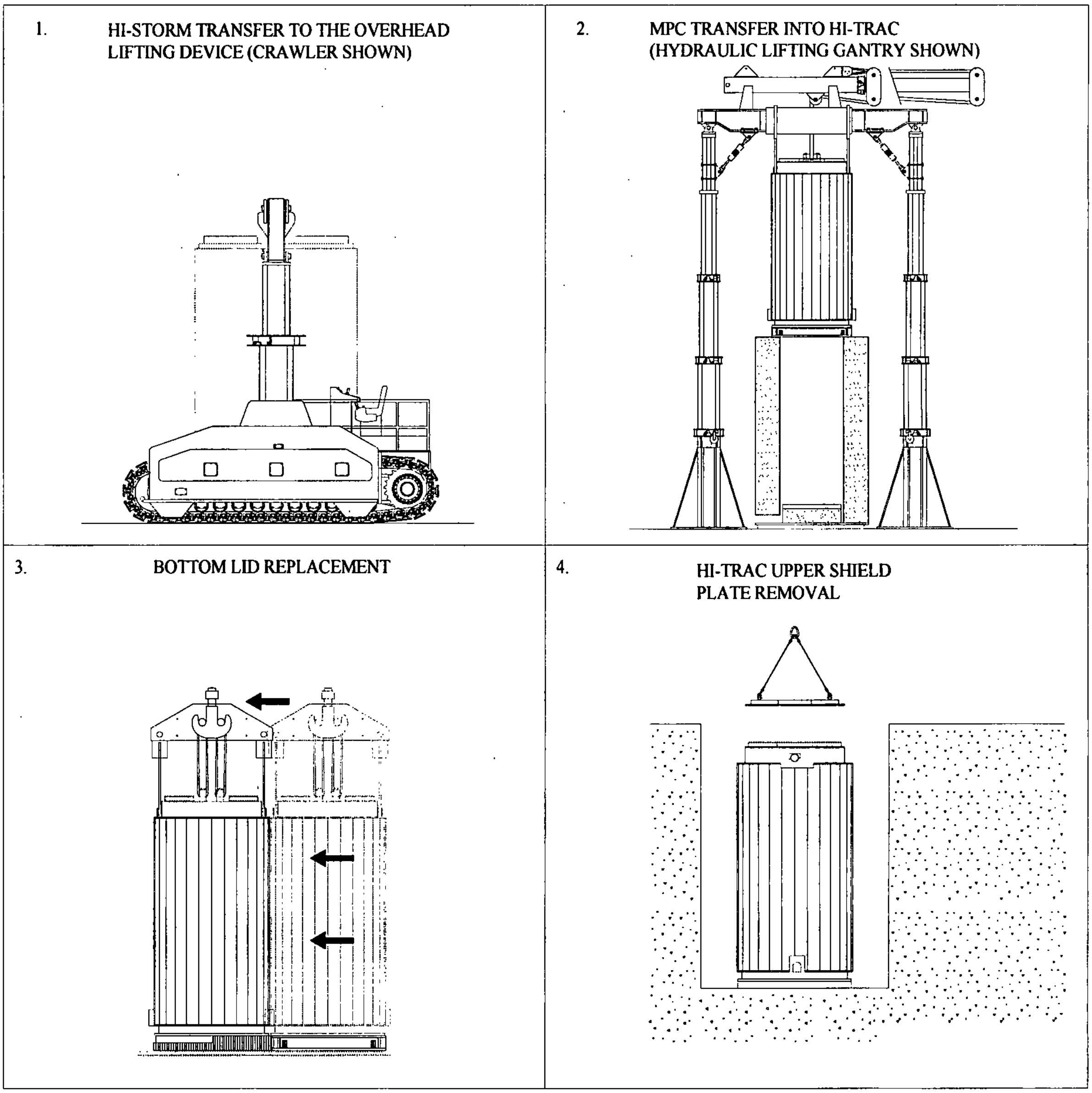

Figure 1.2.17a; Major HI-STORM 100 Unloading Operations (Sheet 1 of 4)

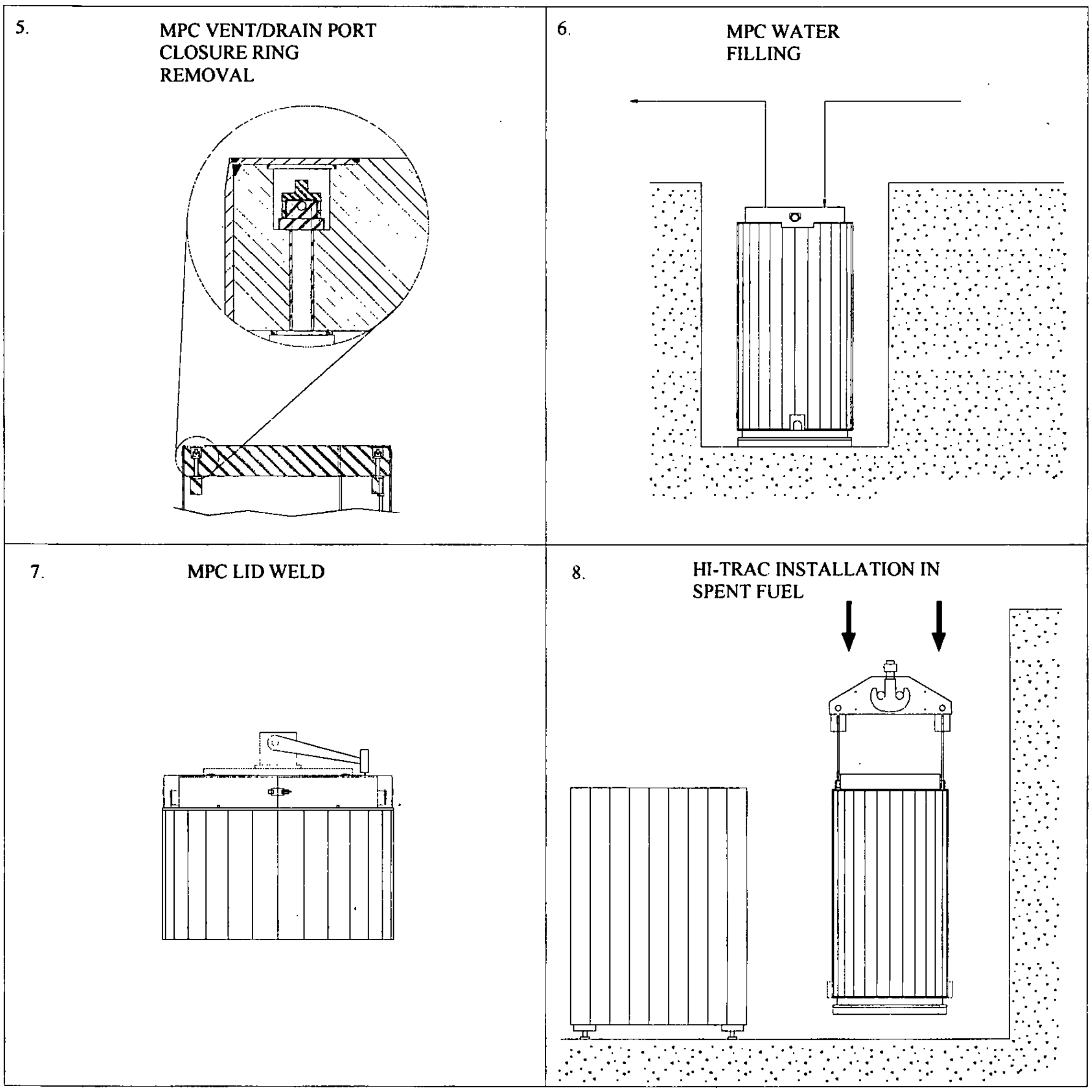

Figure 1.2.17b; Major HI-STORM 100 Unloading Operations (Sheet 2 of 4)

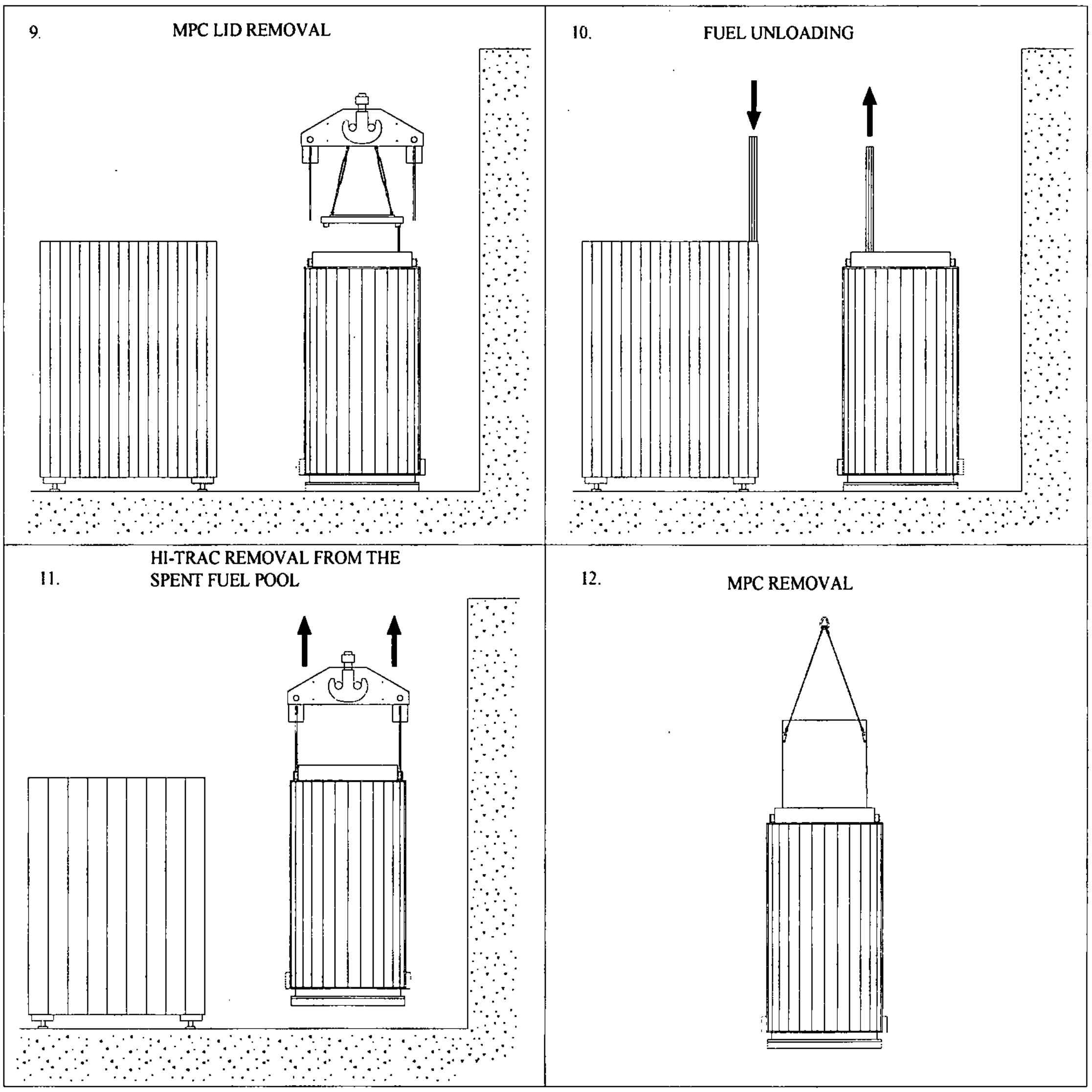

Figure 1.2.17c; Major HI-STORM 100 Unloading Operations (Sheet 3 of 4)

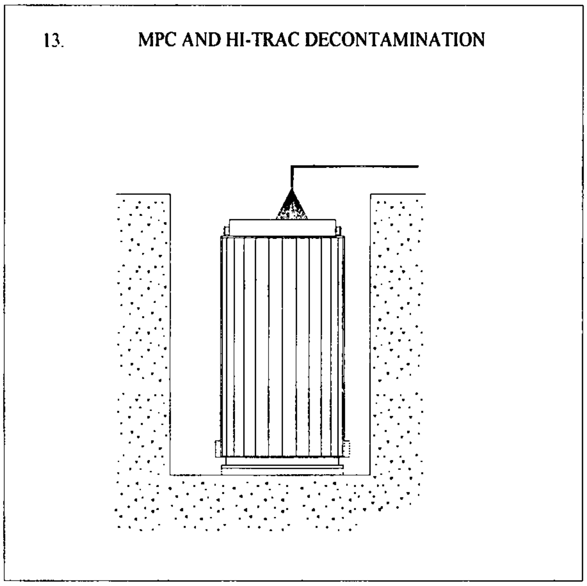

Figure 1.2.17d; Major HI-STORM 100 Unloading Operations (Sheet 4 of 4)

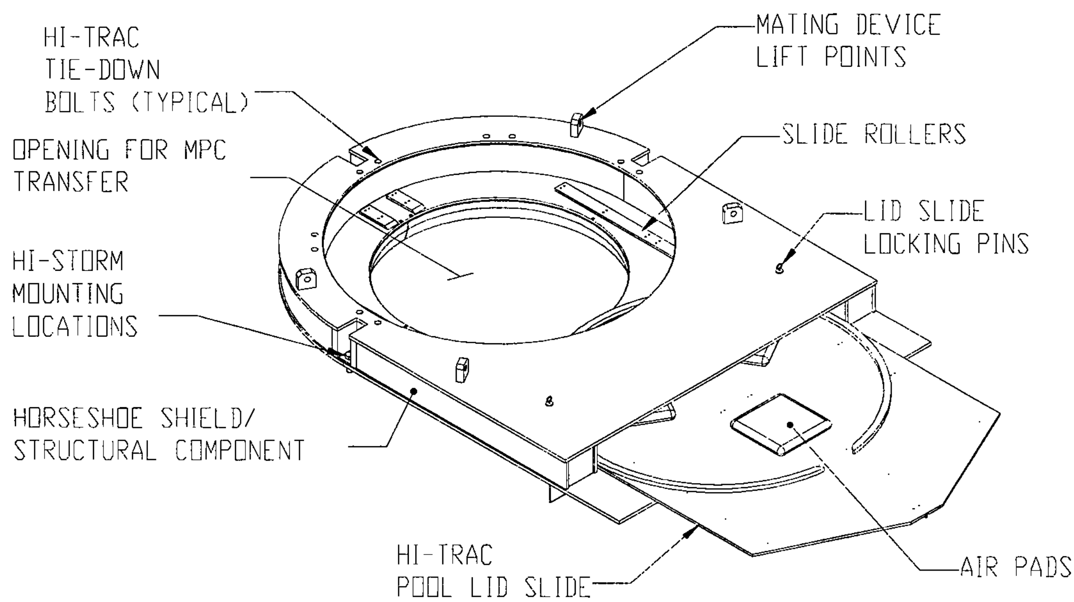

FIGURE 1.2.18; HI-STORM MATING DEVICE

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1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Holtec International is a specialty engineering company with a principal focus on spent fuel storage technologies. Holtec has carried out turnkey wet storage capacity expansions (engineering, licensing, fabrication, removal of existing racks, performance of underwater modifications, volume reduction of the old racks and hardware, installation of new racks, and commissioning of the pool for increased storage capacity) in numerous plants around the world. Over 45 plants in the U.S., Britain, Brazil, Korea, and Taiwan have utilized Holtec's wet storage technology to extend their in-pool storage capacity.

Holtec's corporate engineering consists of experts with advanced degrees (Ph.D.'s) in every discipline germane to the fuel storage technologies, namely structural mechanics, heat transfer, computational fluid dynamics, and nuclear physics. All engineering analyses for Holtec's fuel storage projects (including HI-STORM 100) are carried out in-house.

Holtec International's quality assurance program was originally developed to meet NRC requirements delineated in 10CFR50, Appendix B, and was expanded to include provisions of 10CFR71, Subpart H, and 10CFR72, Subpart G, for structures, systems, and components designated as important to safety. The Holtec quality assurance program, which satisfies all 18 criteria in 10CFR72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of structures, systems, and components important to safety is incorporated by reference into this FSAR as described in Chapter 13.

The HI-STORM 100 System is fabricated by Holtec Manufacturing Division (HMD) of Pittsburgh, Pennsylvania; formerly UST&D. HMD is an N-Stamp holder and a highly respected fabricator of nuclear components. HMD is on Holtec's Approved Vendors List (AVL) and has a quality assurance program meeting 10CFR50 Appendix B criteria. Extensive prototypical fabrication of the MPCs has been carried out at the HMD shop to resolve fixturing and tolerance issues. If another fabricator is to be used for the fabrication of any part of the HI-STORM 100 System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's quality assurance program.

Construction, assembly, and operations on-site may be performed by Holtec or a licensee as the prime contractor. A licensee shall be suitably qualified and experienced to perform selected activities. Typical licensees are technically qualified and experienced in commercial nuclear power plant construction and operation activities under a quality assurance program meeting 10CFR50 Appendix B criteria.

1.4 GENERIC CASK ARRAYS

The HI-STORM 100 System is stored in a vertical configuration. The required center-to-center spacing between the modules (layout pitch) is guided by operational considerations. Tables 1.4.1 and 1.4.2 provide the nominal layout pitch information. Site-specific pitches are determined by practical operation with supporting heat transfer calculations in Chapter 4. The pitch values in Tables 1.4.1 and 1.4.2 are nominal and may be varied to suit the user's specific needs.

Table 1.4.1 provides recommended cask spacing data for array(s) of two by N casks. The pitch between adjacent rows of casks and between each adjacent column of casks are denoted by P_1 and P_2 In Table 1.4.1. There may be an unlimited number of rows. The distance between adjacent arrays of two by N casks (P3) shall be as specified in Table 1.4.1. See Figure 1.4.1 for further clarification. The pattern of required pitches and distances may be repeated for an unlimited number of columns.

For a square array of casks the pitch between adjacent casks may be in accordance with Table 1.4.2. See Figure 1.4.2 for further clarification. The data in Table 1.4.2 provide nominal values for large ISFSIs (i.e., those with hundreds of casks in a uniform layout), where access of feed air to the centrally located casks may become a matter of thermal consideration. From a thermal standpoint, regardless of the size of the ISFSI, the casks should be arrayed in such a manner that the tributary area for each cask (open ISFSI area attributable to a cask) is a minimum of 225 ft^2. Subsection 4.4.1.1.7 provides the detailed thermal evaluation of the required tributary area. For specific sites, a smaller tributary area can be utilized after appropriate thermal evaluations for the site-specific conditions are performed.

Table 1.4.1

CASK LAYOUT PITCH DATA FOR 2 BY N ARRAYS

Orientation	**Nominal Cask Pitch (ft.)**
Between adjacent rows, P1, and adjacent columns, P2	13.5
Between adjacent sets of two columns, P3	38

Table 1.4.2

CASK LAYOUT PITCH DATA FOR SQUARE ARRAYS

Orientation	**Nominal Cask Pitch (ft.)**
Between adjacent casks	18' - 8"

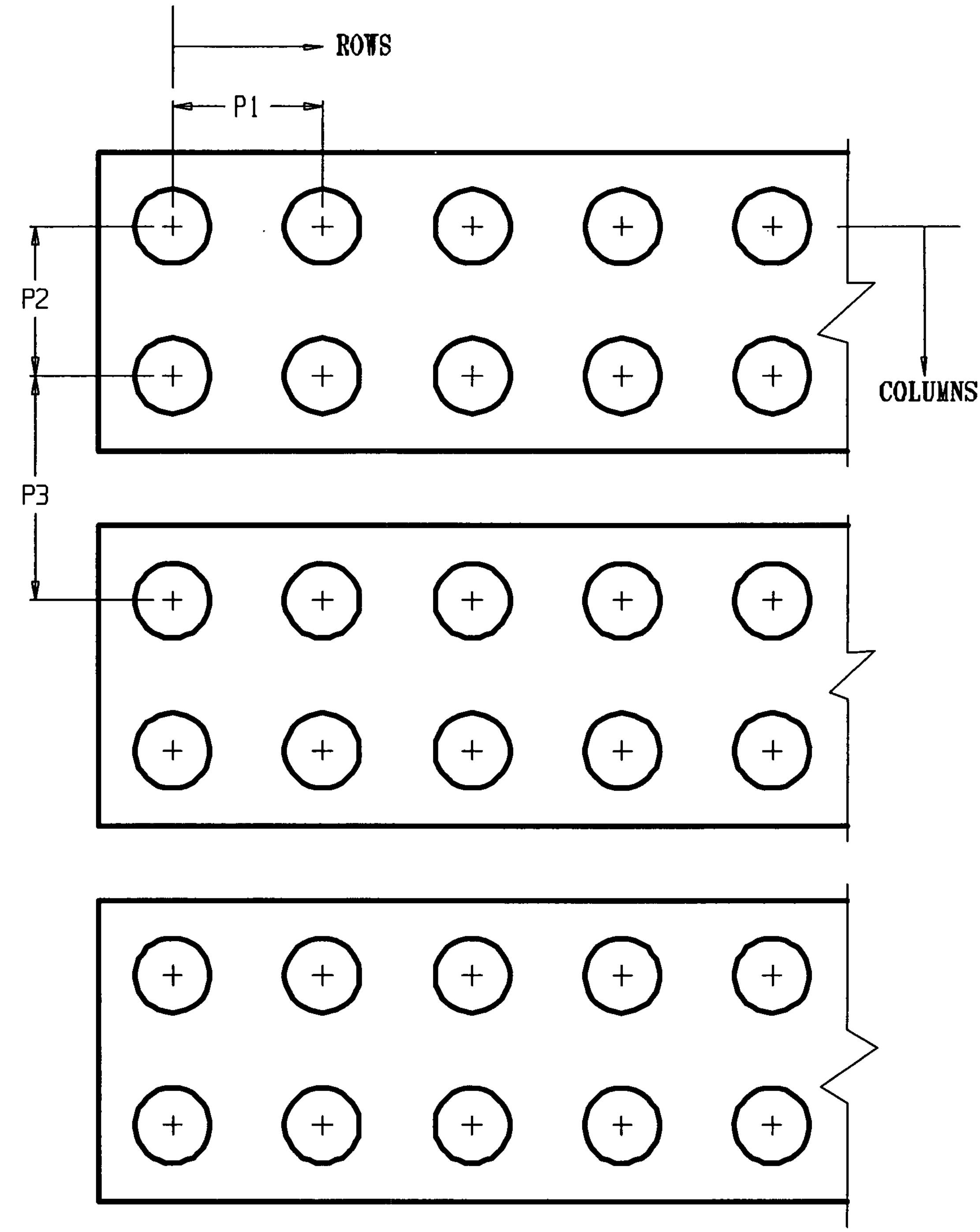

FIGURE 1.4.1; CASK LAYOUT PITCH REQUIREMENTS BASED ON 2 BY N ARRAY(S)

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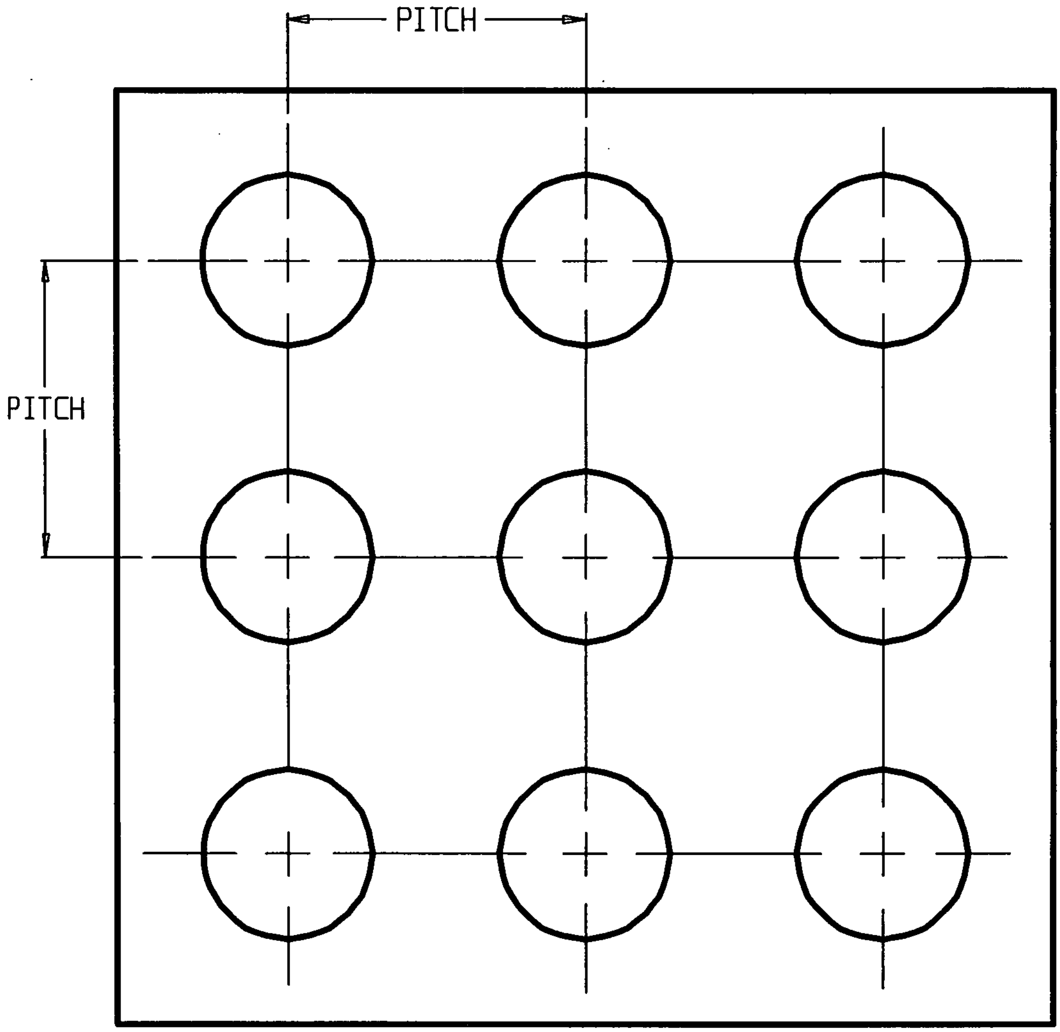

FIGURE 1.4.2; CASK LAYOUT PITCH REQUIREMENTS
BASED ON A SQUARE ARRAY

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1.5 DRAWINGS

The following HI-STORM 100 System drawings and bills of materials are provided on subsequent pages in this subsection:

Drawing Number/Sheet	Description	Rev.
3923	MPC Enclosure Vessel	30
3925	MPC-24E/EF Fuel Basket Assembly	9
3926	MPC-24 Fuel Basket Assembly	12
3927	MPC-32 Fuel Basket Assembly	16
3928	MPC-68/68F/68FF Basket Assembly	16
7195	MPC-68M Basket Assembly	4
1495 Sht 1/6	HI-STORM 100 Assembly	13
1495 Sht 2/6	Cross Section "Z" - "Z" View of HI-STORM	18
1495 Sht 3/6	Section "Y" - "Y" of HI-STORM	12
1495 Sht 4/6	Section "X" -"X" of HI-STORM	13
1495 Sht 5/6	Section "W" -"W" of HI-STORM	15
1561 Sht 1/6	View "A" -"A" of HI-STORM	11
1561 Sht 2/6	Detail "B" of HI-STORM	15
1561 Sht 3/6	Detail of Air Inlet of HI-STORM	11
1561 Sht 4/6	Detail of Air Outlet of HI-STORM	12
3669	HI-STORM 100S Assembly	19
1880 Sht 1/10	125 Ton HI-TRAC Outline with Pool Lid	9
1880 Sht 2/10	125 Ton HI-TRAC Body Sectioned Elevation	10
1880 Sht 3/10	125 Ton HI-TRAC Body Sectioned Elevation "B" - "B"	9
1880 Sht 4/10	125 Ton Transfer Cask Detail of Bottom Flange	10
1880 Sht 5/10	125 Ton Transfer Cask Detail of Pool Lid	10
1880 Sht 6/10	125 Ton Transfer Cask Detail of Top Flange	10
1880 Sht 7/10	125 Ton Transfer Cask Detail of Top Lid	9
1880 Sht 8/10	125 Ton Transfer Cask View "Y" - "Y"	9
1880 Sht 9/10	125 Ton Transfer Cask Lifting Trunnion and Locking Pad	7
1880 Sht 10/10	125 Ton Transfer Cask View "Z" - "Z"	9
1928 Sht 1/2	125 Ton HI-TRAC Transfer Lid Housing Detail	11
1928 Sht 2/2	125 Ton HI-TRAC Transfer Lid Door Detail	10
2145 Sht 1/10	100 Ton HI-TRAC Outline with Pool Lid	8
2145 Sht 2/10	100 Ton HI-TRAC Body Sectioned Elevation	8
2145 Sht 3/10	100 Ton HI-TRAC Body Sectioned Elevation 'B-B'	8
2145 Sht 4/10	100 Ton HI-TRAC Detail of Bottom Flange	7
2145 Sht 5/10	100 Ton HI-TRAC Detail of Pool Lid	6
2145 Sht 6/10	100 Ton HI-TRAC Detail of Top Flange	8
2145 Sht 7/10	100 Ton HI-TRAC Detail of Top Lid	8
2145 Sht 8/10	100 Ton HI-TRAC View Y-Y	8
2145 Sht 9/10	100 Ton HI-TRAC Lifting Trunnions and Locking Pad	5
2145 Sht 10/10	100 Ton HI-TRAC View Z-Z	7

Drawing Number/Sheet	Description	Rev.
2152 Sht 1/2	100 Ton HI-TRAC Transfer Lid Housing Detail	10
2152 Sht 2/2	100 Ton HI-TRAC Transfer Lid Door Detail	8
3187	Lug and Anchoring Detail for HI-STORM 100A	2
BM-1575, Sht 1/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1575, Sht 2/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1880, Sht 1/2	Bill-of-Material for 125 Ton HI-TRAC	9
BM-1880, Sht 2/2	Bill-of-Material for 125 Ton HI-TRAC	7
BM-1928, Sht 1/1	Bill-of-Material for 125 Ton HI-TRAC Transfer Lid	10
BM-2145 Sht 1/2	Bill-of-Material for 100 Ton HI-TRAC	6
BM-2145 Sht 2/2	Bill-of-Material for 100 Ton HI-TRAC	5
BM-2152 Sht 1/1	Bill-of-Material for 100 Ton HI-TRAC Transfer Lid	8
3768	125 Ton HI-TRAC 125D Assembly	12
4116	HI-STORM 100S Version B	25
4128	100 Ton HI-TRAC 100D Assembly	9

[DRAWINGS WITHHELD PER 10 CFR 2.390]

1.6 REFERENCES

[1.0.1] 10CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.

[1.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989.

[1.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.

[1.0.4] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan.†

[1.0.5] American Concrete Institute, "Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)".

[1.0.6] "Spent Nuclear Fuel Effective Thermal Conductivity Report," U.S. Department of Energy Document Identifier BBA000000-01717-5705-00010, Rev. 00, Tables S-1 through S-4.

[1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, 1995 with Addenda through 1997.

[1.1.2] USNRC Docket No. 72-1008, Final Safety Analysis Report for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.

[1.1.3] USNRC Docket No. 71-9261, Safety Analysis Report for Packaging for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.

[1.1.4] 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.

[1.1.5] Deleted.

[1.2.1] U.S. NRC Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket".

† The 1997 edition of ACI-349 is specified for embedment design for deployment of the anchored HI-STORM 100A and HI-STORM 100SA.

[1.2.2] Directory of Nuclear Reactors, Vol. II, Research, Test & Experimental Reactors, International Atomic Energy Agency, Vienna, 1959.

[1.2.3] V.L. McKinney and T. Rockwell III, "Boral: A New Thermal-Neutron Shield", USAEC Report AECD-3625, August 29, 1949.

[1.2.4] Reactor Shielding Design Manual, USAEC Report TID-7004, March 1956.

[1.2.5] "Safety Analysis Report for the NAC Storable Transport Cask", Revision 8, September 1994, Nuclear Assurance Corporation (USNRC Docket No. 71-9235).

[1.2.6] Deleted.

[1.2.7] Materials Handbook, 13th Edition, Brady, G.S. and H.R. Clauser, McGraw-Hill, 1991, Page 310.

[1.2.8] Deleted.

[1.2.9] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, June, 1993.

[1.2.10] Deleted.

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

[1.2.12] "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications," Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., Docket No. 50-313 and 50-368, USNRC, June 2003.

[1.2.13] "Metamic 6061+40% Boron Carbide Metal Matrix Composite Test", California Consolidated Tech. Inc. Report dated August 21, 2001 to NAC International.

[1.2.14] "Recommendations for Preparing the Criticality Safety Evaluation for Transportation Packages," NUREG/CR-5661, USNRC, Dyer and Parks, ORNL.

APPENDIX 1.A: ALLOY X DESCRIPTION

1.A ALLOY X DESCRIPTION

1.A.1 Alloy X Introduction

Alloy X is used within this licensing application to designate a group of stainless steel alloys. Alloy X can be any one of the following alloys:

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

Qualification of structures made of Alloy X is accomplished by using the least favorable mechanical and thermal properties of the entire group for all MPC mechanical, structural, neutronic, radiological, and thermal conditions. The Alloy X approach is conservative because no matter which material is ultimately utilized, the Alloy X approach guarantees that the performance of the MPC will meet or exceed the analytical predictions.

This appendix defines the least favorable material properties of Alloy X.

1.A.2 Alloy X Common Material Properties

Several material properties do not vary significantly from one Alloy X constituent to the next. These common material properties are as follows:

- density
- specific heat
- Young's Modulus (Modulus of Elasticity)
- Poisson's Ratio

The values utilized for this licensing application are provided in their appropriate chapters.

1.A.3 Alloy X Least Favorable Material Properties

The following material properties vary between the Alloy X constituents:

- Design Stress Intensity (S_m)
- Tensile (Ultimate) Strength (S_u)
- Yield Strength (S_y)
- Coefficient of Thermal Expansion (α)
- Coefficient of Thermal Conductivity (k)

Each of these material properties are provided in the ASME Code Section II [1.A.1]. Tables 1.A.1 through 1.A.5 provide the ASME Code values for each constituent of Alloy X along with the least favorable value utilized in this licensing application. The ASME Code only provides values to -20°F. The design temperature of the MPC is -40°F to 725°F as stated in Table 1.2.2. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STORM 100 System have been assumed to be equal to the lowest value stated in the ASME Code. The lone exception is the thermal conductivity. The thermal conductivity decreases with the decreasing temperature. The thermal conductivity value for -40°F is linearly extrapolated from the 70°F value using the difference from 70°F to 100°F.

The Alloy X material properties are the minimum values of the group for the design stress intensity, tensile strength, yield strength, and coefficient of thermal conductivity. Using minimum values of design stress intensity is conservative because lower design stress intensities lead to lower allowables that are based on design stress intensity. Similarly, using minimum values of tensile strength and yield strength is conservative because lower values of tensile strength and yield strength lead to lower allowables that are based on tensile strength and yield strength. When compared to calculated values, these lower allowables result in factors of safety that are conservative for any of the constituent materials of Alloy X. Further discussion of the justification for using the minimum values of coefficient of thermal conductivity is given in Chapter 3. The maximum and minimum values are used for the coefficient of thermal expansion of Alloy X. The maximum and minimum coefficients of thermal expansion are used as appropriate in this submittal. Figures 1.A.1-1.A.5 provide a graphical representation of the varying material properties with temperature for the Alloy X materials.

1.A.4 <u>References</u>

[1.A.1] ASME Boiler & Pressure Vessel Code Section II, 1995 ed. with Addenda through 1997.

Table 1.A.1

ALLOY X AND CONSTITUENT DESIGN STRESS INTENSITY (S_m) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	20.0	20.0	20.0	20.0	20.0
100	20.0	20.0	20.0	20.0	20.0
200	20.0	20.0	20.0	20.0	20.0
300	20.0	20.0	20.0	20.0	20.0
400	18.7	18.7	19.3	18.9	18.7
500	17.5	17.5	18.0	17.5	17.5
600	16.4	16.4	17.0	16.5	16.4
650	16.2	16.2	16.7	16.0	16.0
700	16.0	16.0	16.3	15.6	15.6
750	15.6	15.6	16.1	15.2	15.2
800	15.2	15.2	15.9	14.9	14.9

Notes:

1. Source: Table 2A on pages 314, 318, 326, and 330 of [1.A.1].

2. Units of design stress intensity values are ksi.

Table 1.A.2

ALLOY X AND CONSTITUENT TENSILE STRENGTH (S_u) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
100	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
200	71.0 (66.2)	71.0 (66.2)	75.0 (70.0)	75.0 (70.0)	71.0 (66.2)
300	66.0 (61.5)	66.0 (61.5)	73.4 (68.5)	70.9 (66.0)	66.0 (61.5)
400	64.4 (60.0)	64.4 (60.0)	71.8 (67.0)	67.1 (62.6)	64.4 (60.0)
500	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	64.6 (60.3)	63.5 (59.3)
600	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	63.1 (58.9)	63.1 (58.9)
650	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	62.8 (58.6)	62.8 (58.6)
700	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	62.5 (58.4)	62.5 (58.4)
750	63.1 (58.9)	63.1 (58.9)	71.4 (66.5)	62.2 (58.1)	62.2 (58.1)
800	62.7 (58.5)	62.7 (58.5)	70.9 (66.2)	61.7 (57.6)	61.7 (57.6)

Notes:

1. Source: Table U on pages 437, 439, 441, and 443 of [1.A.1].

2. Units of tensile strength are ksi.

3. The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in parentheses are based on SA-336 forged materials (type F304, F304LN, F316, and F316LN), which are used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.

Table 1.A.3

ALLOY X AND CONSTITUENT YIELD STRESSES (S_y) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	30.0	30.0	30.0	30.0	30.0
100	30.0	30.0	30.0	30.0	30.0
200	25.0	25.0	25.8	25.5	25.0
300	22.5	22.5	23.3	22.9	22.5
400	20.7	20.7	21.4	21.0	20.7
500	19.4	19.4	19.9	19.4	19.4
600	18.2	18.2	18.8	18.3	18.2
650	17.9	17.9	18.5	17.8	17.8
700	17.7	17.7	18.1	17.3	17.3
750	17.3	17.3	17.8	16.9	16.9
800	16.8	16.8	17.6	16.6	16.6

Notes:

1. Source: Table Y-1 on pages 518, 519, 522, 523, 530, 531, 534, and 535 of [1.A.1].

2. Units of yield stress are ksi.

Table 1.A.4

ALLOY X AND CONSTITUENT COEFFICIENT OF THERMAL EXPANSION vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X Maximum	Alloy X Minimum
-40	8.55	8.54	8.55	8.54
100	8.55	8.54	8.55	8.54
150	8.67	8.64	8.67	8.64
200	8.79	8.76	8.79	8.76
250	8.90	8.88	8.90	8.88
300	9.00	8.97	9.00	8.97
350	9.10	9.11	9.11	9.10
400	9.19	9.21	9.21	9.19
450	9.28	9.32	9.32	9.28
500	9.37	9.42	9.42	9.37
550	9.45	9.50	9.50	9.45
600	9.53	9.60	9.60	9.53
650	9.61	9.69	9.69	9.61
700	9.69	9.76	9.76	9.69
750	9.76	9.81	9.81	9.76
800	9.82	9.90	9.90	9.82

Notes:

1. Source: Table TE-1 on pages 590 and 591 of [1.A.1].

2. Units of coefficient of thermal expansion are in./in.-°F x 10^{-6}.

Table 1.A.5

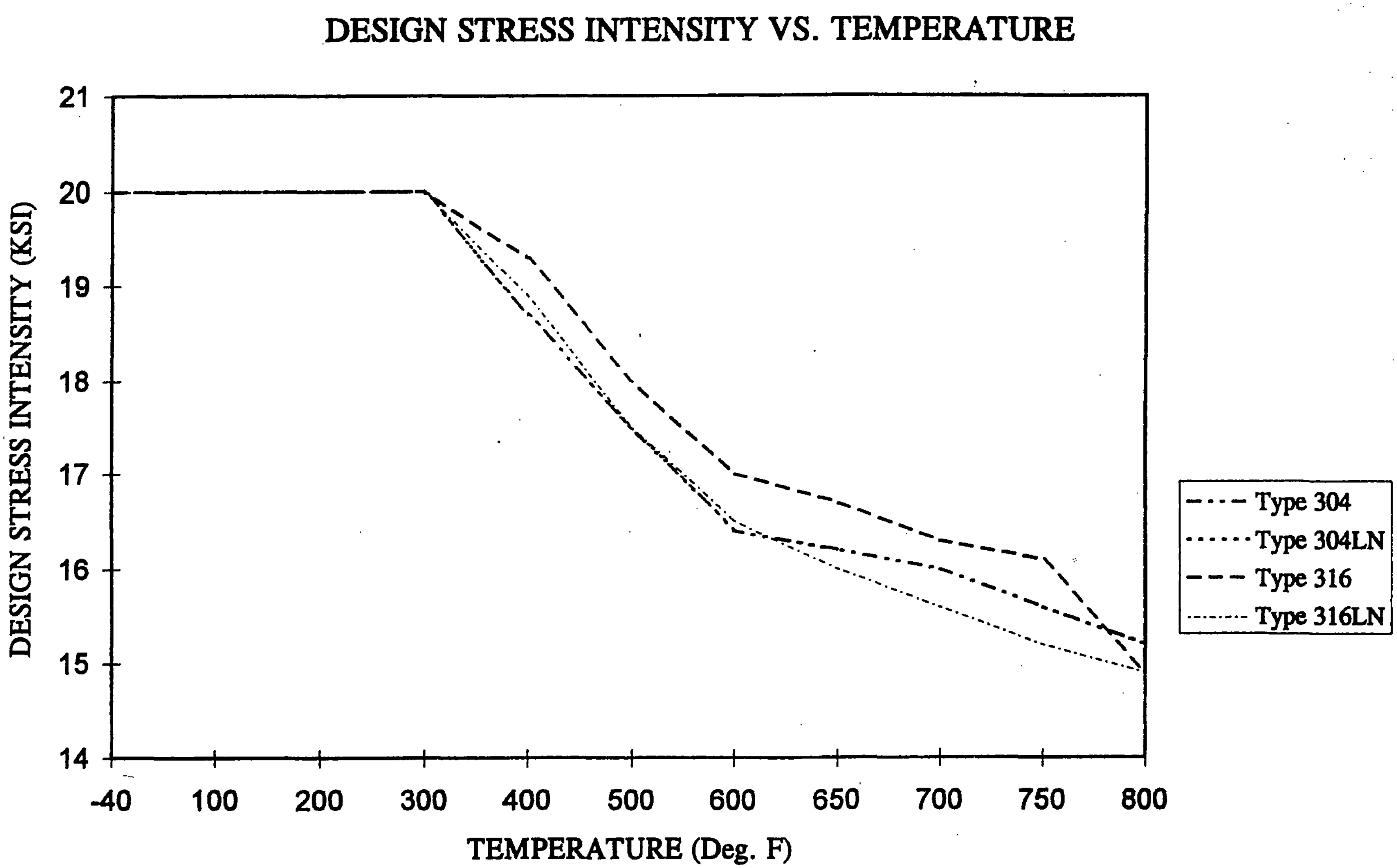
ALLOY X AND CONSTITUENT THERMAL CONDUCTIVITY vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X (minimum of constituent values)
-40	8.23	6.96	6.96
70	8.6	7.7	7.7
100	8.7	7.9	7.9
150	9.0	8.2	8.2
200	9.3	8.4	8.4
250	9.6	8.7	8.7
300	9.8	9.0	9.0
350	10.1	9.2	9.2
400	10.4	9.5	9.5
450	10.6	9.8	9.8
500	10.9	10.0	10.0
550	11.1	10.3	10.3
600	11.3	10.5	10.5
650	11.6	10.7	10.7
700	11.8	11.0	11.0
750	12.0	11.2	11.2
800	12.2	11.5	11.5

Notes:

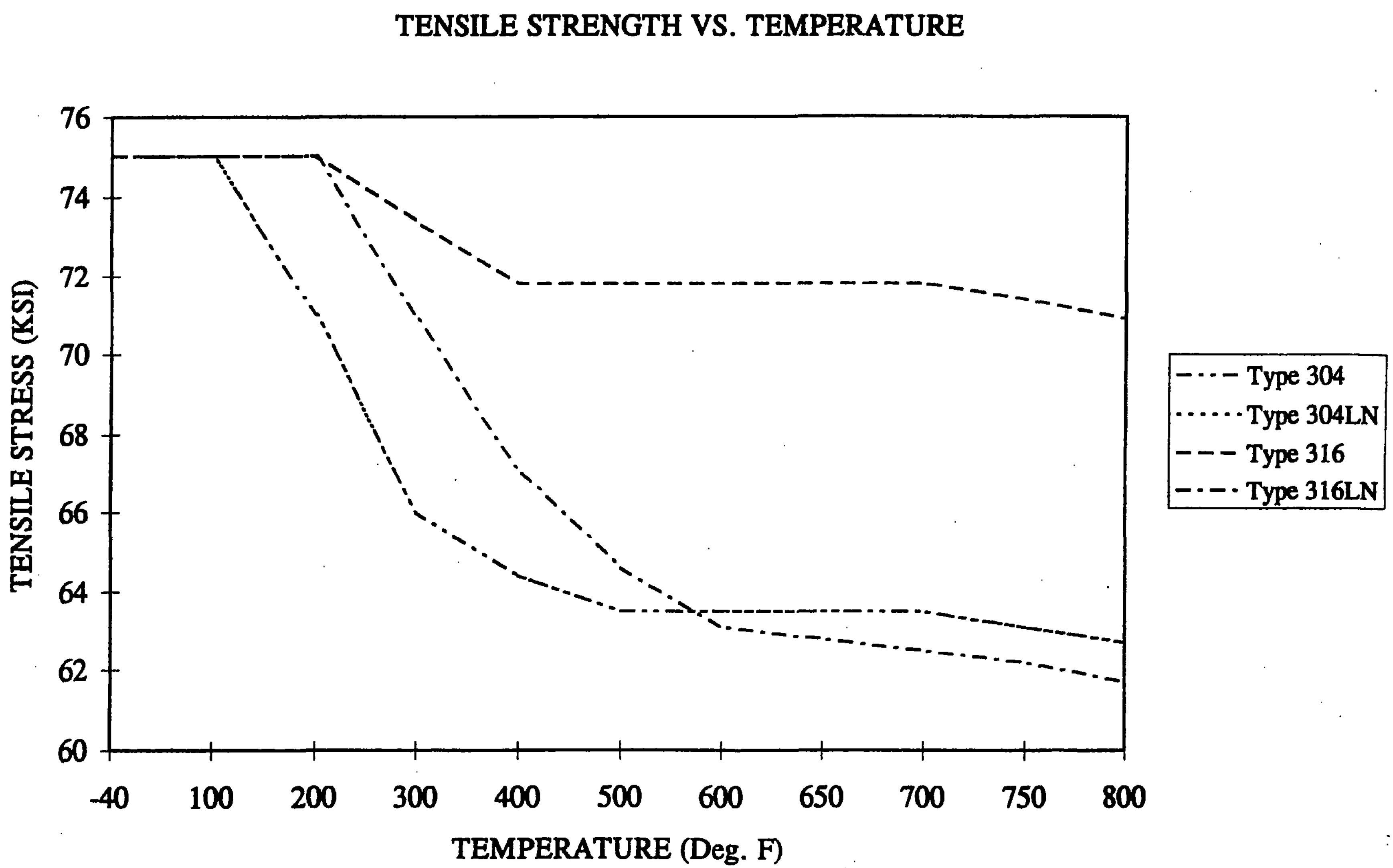
1. Source: Table TCD on page 606 of [1.A.1].

2. Units of thermal conductivity are Btu/hr-ft-°F.



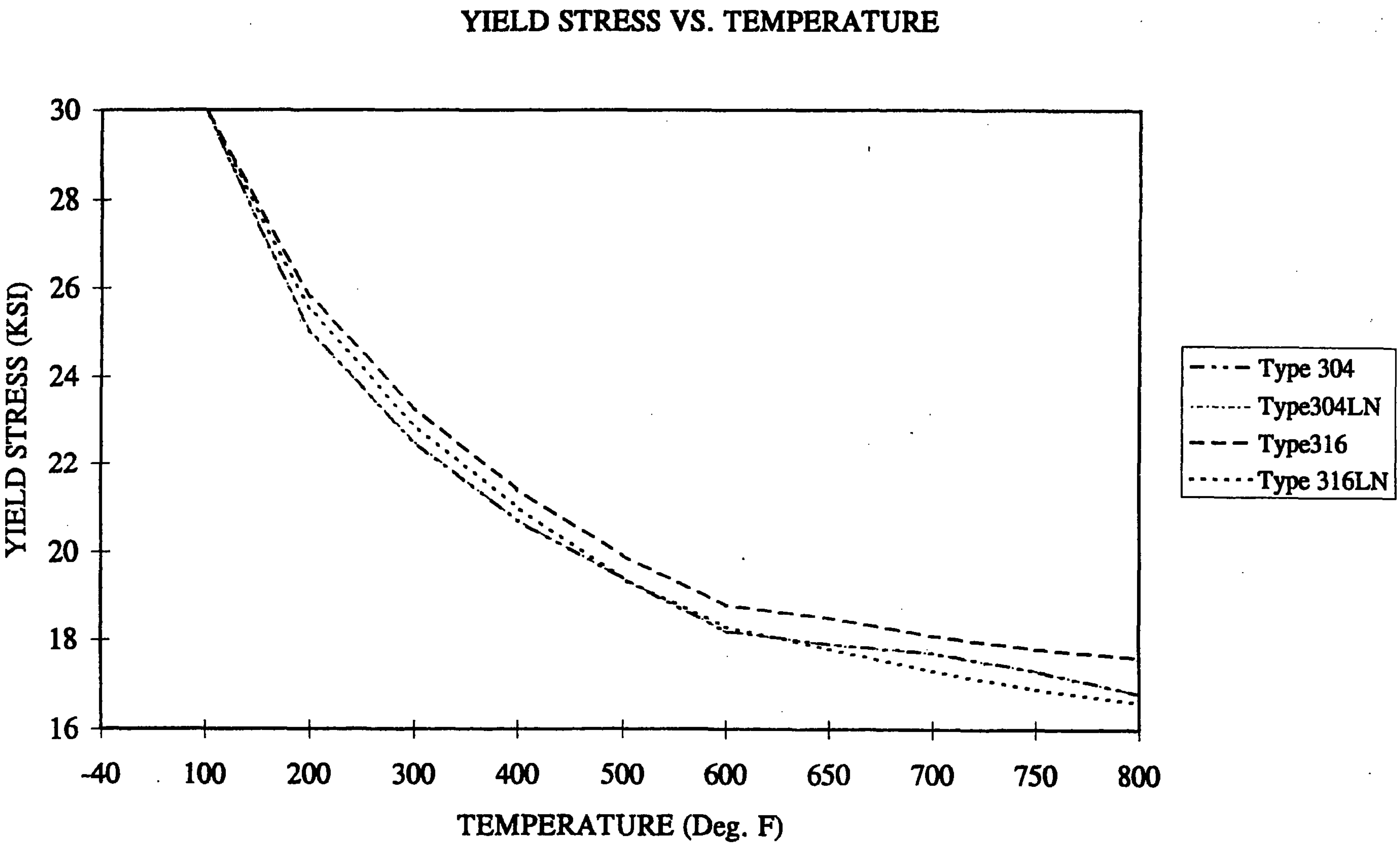
SOURCE: TABLE 1.A.1

FIGURE 1.A.1; DESIGN STRESS INTENSITY VS. TEMPERATURE



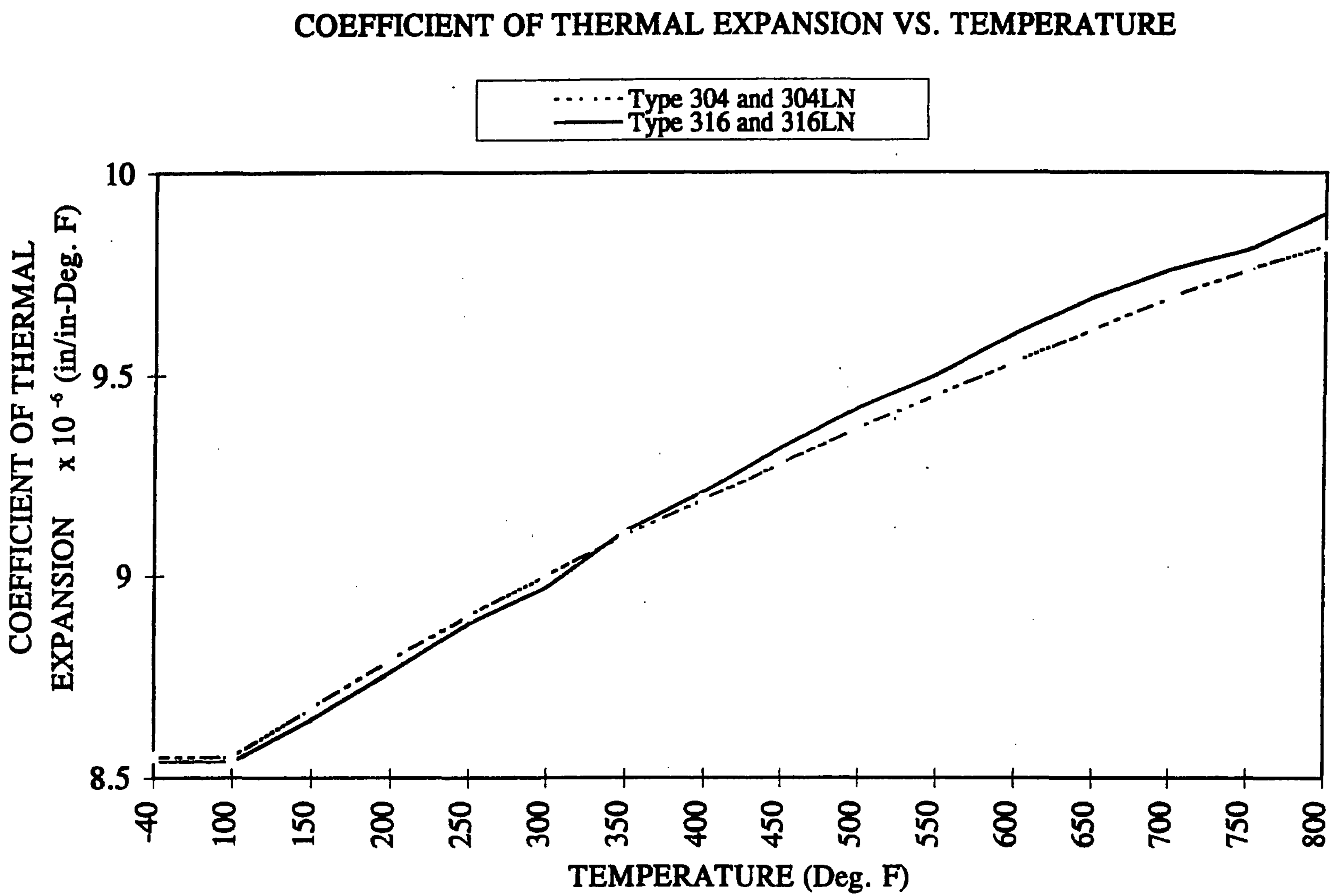
SOURCE: TABLE 1.A.2

FIGURE 1.A.2; TENSILE STRENGTH VS. TEMPERATURE



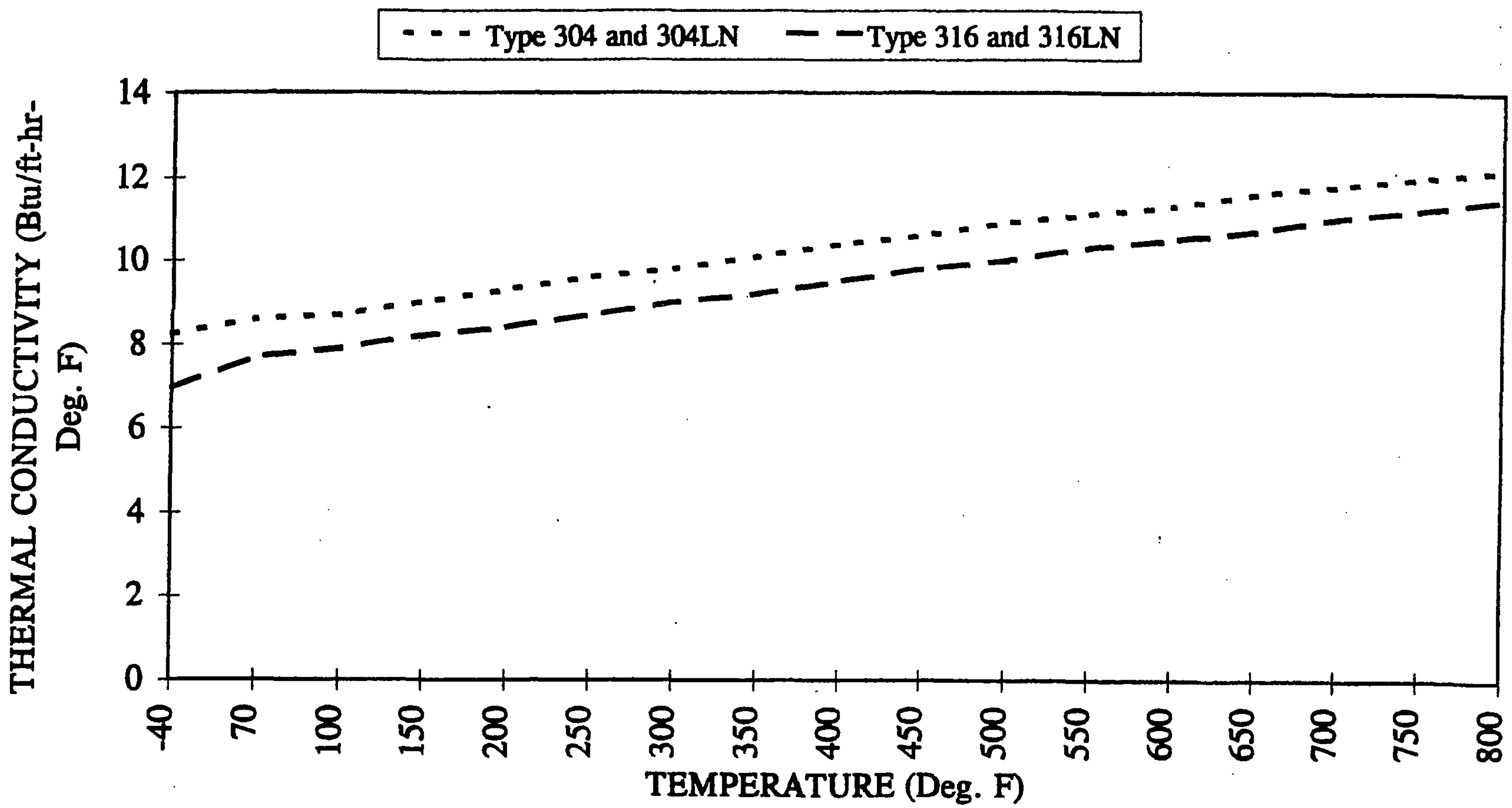
SOURCE: TABLE 1.A.3

FIGURE 1.A.3; YIELD STRESS VS. TEMPERATURE



SOURCE: TABLE 1.A.4 **FIGURE 1.A.4; COEFFICIENT OF THERMAL EXPANSION VS. TEMPERATURE**

THERMAL CONDUCTIVITY VS. TEMPERATURE



SOURCE: TABLE 1.A.5

FIGURE 1.A.5; THERMAL CONDUCTIVITY VS. TEMPERATURE

APPENDIX 1.B: HOLTITE ™ MATERIAL DATA

The information provided in this appendix describes the neutron absorber material, Holtite-A for the purpose of confirming its suitability for use as a neutron shield material in spent fuel storage casks. Holtite-A is one of the family of Holtite neutron shield materials denoted by the generic name Holtite™. It is currently the only solid neutron shield material approved for installation in the HI-TRAC transfer cask. It is chemically identical to NS-4-FR which was originally developed by Bisco Inc. and used for many years as a shield material with B_4C or Pb added.

Holtite-A contains aluminum hydroxide ($Al(OH)_3$) in an epoxy resin binder. Aluminum hydroxide is also known by the industrial trade name of aluminum tri-hydrate or ATH. ATH is often used commercially as a fire-retardant. Holtite-A contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A contains 1% (nominal) by weight B_4C, a chemically inert material added to enhance the neutron absorption property. Pertinent properties of Holtite-A are listed in Table 1.B.1.

The essential properties of Holtite-A are:

1. the hydrogen density (needed to thermalize neutrons),

2. thermal stability of the hydrogen density, and

3. the uniformity in distribution of B_4C needed to absorb the thermalized neutrons.

ATH and the resin binder contain nearly the same hydrogen density so that the hydrogen density of the mixture is not sensitive to the proportion of ATH and resin in the Holtite-A mixture. B_4C is added as a finely divided powder and does not settle out during the resin curing process. Once the resin is cured (polymerized), the ATH and B_4C are physically retained in the hardened resin. Qualification testing for B_4C throughout a column of Holtite-A has confirmed that the B_4C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B_4C is specified in the Holtite-A mixing and pouring procedure as a precaution to assure that the B_4C concentration is always adequate throughout the mixture.

The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity assumed in the shielding analysis includes a 4% reduction to conservatively account for potential weight loss at the design temperatures listed in Table 1.B.1. or an inability to reach theoretical density. Tests on the stability of Holtite-A were performed by Holtec International. The results of the tests are summarized in Holtec Reports HI-2002396, "Holtite-A Development History and Thermal Performance Data" and HI-2002420, "Results of Pre- and Post-Irradiation Test Measurements." The information provided in these reports demonstrates that Holtite-A™ possesses

the necessary thermal and radiation stability characteristics to function as a reliable shielding material in the HI-TRAC transfer cask.

The Holtite-A is encapsulated in the HI-TRAC transfer cask lid and, therefore, should experience a very small weight reduction during the design life of the cask. The data and test results confirm that Holtite-A remains stable under design thermal and radiation conditions, the material properties meet or exceed that assumed in the shielding analysis, and the B_4C remains uniformly distributed with no evidence of settling or non-uniformity.

Based on the information described above, Holtite-A meets all of the requirements for an acceptable neutron shield material.

Table 1.B.1

REFERENCE PROPERTIES OF HOLTITE-A NEUTRON SHIELD MATERIAL

PHYSICAL PROPERTIES	
% ATH	62 nominal
Specific Gravity	1.68 g/cc nominal
Max. Continuous Operating Temperature	300°F
Max. Short-Term Operating Temperature	350°F (Note 1)
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5
wt% Hydrogen	6.0
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	1.0

NOTES:

1. As defined in Section 2.2, all operations involving the HI-TRAC transfer cask are short-term operating conditions. The short-term operating temperature limit is, therefore, the appropriate maximum design temperature for the Holtite-A in the HI-TRAC transfer cask.

APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA
(Total of 1 Page Including This Page)

The information provided in this appendix specifies the paint properties and demonstrates their suitability for use in spent nuclear fuel storage casks.

Thermaline 450 or functionally equivalent paint/coating is specified to coat the overpack to the maximum extent practical and the inner cavity of the HI-TRAC transfer cask. Carboline 890 or functionally equivalent paint/coating is specified to coat external surfaces of the HI-TRAC transfer cask. The paints are suitable for the design temperatures (see Table 2.2.3) and the environment.

APPENDIX 1.D: Requirements on HI-STORM 100 Shielding Concrete

1.D.1 Introduction

The HI-STORM 100 overpack utilizes plain concrete for neutron and gamma shielding. Plain concrete used in the HI-STORM overpack provides only a compressive strength structural function due to the fact that both the primary and secondary load bearing members of the overpack are made of carbon steel. While most of the shielding concrete used in the HI-STORM 100 overpack is installed in the annulus between the concentric structural shells, smaller quantities of concrete are also present in the pedestal shield and the overpack lid. Because plain concrete has little ability to withstand tensile stresses, but is competent in withstanding compressive and bearing loads, the design of the HI-STORM 100 overpack places no reliance on the tension-competence of the shielding concrete.

During normal operations of the HI-STORM, the stresses in the concrete continuum are negligible, arising solely from its self-weight. ACI 318.1-89(92) provides formulas for permissible compressive and bearing stresses in plain concrete, which incorporate a penalty over the corresponding permissible values in reinforced concrete. The formulas for permissible compressive and bearing stresses set forth in ACI 318.1-89(92)are used in calculations supporting this FSAR in load cases involving compression or bearing loads on the overpack concrete. However, since the overpack concrete is designated as an ITS Category B material, it is appropriate to ensure that all "*critical characteristics*" of the concrete, as defined herein, are fully satisfied. During normal storage operations, the overpack concrete is completely enclosed by the overpack steel structure, protecting it from the deleterious effects of direct exposure to the environment, typical of most concrete structures governed by the ACI codes.

The "*critical characteristics*" of the plain concrete in the HI-STORM overpack are: (i) its density and (ii) its compressive strength. This appendix provides the complete set of criteria applicable to the plain concrete in the HI-STORM 100 overpack.

1.D.2 Design Requirements

The primary function of the plain concrete is to provide neutron and gamma shielding. As plain concrete is a competent structural member in compression, the plain concrete's effect on the performance of the HI-STORM overpack under compression loadings is considered and modeled in the structural analyses, as necessary. The formulas for permissible compressive and bearing stresses set forth in ACI 318.1-89(92)are used. However, as plain concrete has very limited capabilities in tension, no tensile strength capability is allotted to the HI-STORM concrete.

The steel structure of the HI-STORM overpack provides the strength to meet all load combinations specified in Chapters 2 and 3, due to the fact that both the primary and secondary load bearing members (as defined in the ASME Code, Section III, Subsection NF-1215) of the HI-STORM overpack are made from carbon steel. Credit for the structural strength of the plain concrete is only taken to enhance the compressive load carrying capability of the concrete in calculations appropriate to handling and transfer operations, and to demonstrate that the HI-STORM 100 System continues to provide functional performance in a post-accident environment. Therefore, the load combinations provided in ACI 349 and NUREG-1536, Table 3-1 are not applicable to the plain concrete in the HI-STORM overpack.

The shielding performance of the plain concrete is maintained by ensuring that the minimum concrete density is met during construction and the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions demonstrate that the plain concrete does not exceed the allowable long term temperature limit provided in Table 1.D.1. Under accident conditions, the bulk of the plain concrete in the HI-STORM overpack does not exceed the allowable short term temperature limit provided in Table 1.D.1. Any portion of the plain concrete, which exceeds the short-term temperature limit under accident conditions, is neglected in the post-accident shielding analysis and in any post-accident structural analysis.

1.D.2.1 Test Results to Support Normal Condition Temperature Limit

Note 3 to Table 1.D.1 references Paragraph A.4.3 of ACI-349, which requires that normal condition temperatures in excess of 150°F bulk and 200°F local must be supported by test data to demonstrate that strength reductions are acceptable and that concrete deterioration does not occur. Such data are described and discussed in this subsection.

With respect to concrete compressive strength at bulk temperatures up to 300°F, test studies for elevated temperatures were performed by Carette and Malhorta [1.D.1] that examined conditions very similar to those of the HI-STORM concrete. Their tests were performed on 4" diameter by 8" long test cylinders. The test condition most closely matching the HI-STORM concrete was: 0.6 water-to-cement ratio, limestone aggregate and 300°F for four months. While the HI-STORM storage period is much greater than 4 months, the investigators state "any major strength loss is found to occur within the first month of exposure." The four-month compressive strength for these conditions was actually determined to be greater than the nominal concrete strengths despite the elevated temperatures. This is attributable to the increase in compressive strength that accompanies concrete aging, which more than offsets the temperature effects.

With respect to concrete shielding performance at local temperatures above 300°F, a report by Schneider and Horvath [1.D.2] examined weight loss of concrete at elevated temperatures. Tests were performed on 12mm diameter by 40 mm long test cylinders in an apparatus called a thermo-balance. A variety of aggregates (i.e., quartz, limestone and basalt) were tested. The test results indicate a worst-case weight loss of 0.666% from 300°F to 390°F for quartz aggregates. This maximum level of weight loss would reduce the concrete density from 2.24 gm/cc to 2.225 gm/cc. If the entire weight loss is attributed to water loss, the corresponding limiting reduction in hydrogen

content is from 0.6% to 0.529%. As discussed in Section 5.3.2, such reductions are negligible with respect to shielding performance.

1.D.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements, drawn from ACI 349-85 and supplemented by the provisions of NUREG 1536 (page 3-21), are intended to ensure that the "*critical characteristics*" of the concrete placed in the HI-STORM overpack comply with the requirements of this Appendix and standard good practice. Two different minimum concrete densities are specified for the overpack concrete, based on the presence or absence of the steel shield shell. The steel shield shell was deleted from the overpack design after the construction of overpack serial number 1024-7.

ACI 349 was developed to govern the design and construction of steel reinforced concrete structures for the entire array of nuclear power plant applications, except for concrete reactor vessels and containment structures. Therefore, ACI 349 contains many requirements not germane to the plain concrete installed in and completely enclosed by the steel HI-STORM overpack structure. For example, the overpack concrete is not exposed to the environment, so provisions in the standard for protecting concrete from the environment would not be applicable to the concrete contained in the overpack.

In accordance with the requirement in Section 3.3 of Appendix B of the HI-STORM 100 CoC, Section 1.D.4, Table 1.D.1 and Table 1.D.2 were developed using the guidance of ACI 349-85, to the extent it needs to be applied to the unique application of placing unreinforced concrete inside the steel enclosure of the HI-STORM overpack. Other concrete standards were used, as appropriate, to provide the controls necessary to assure that the *critical characteristics* of the overpack concrete will be achieved and that the concrete will perform its design function.

1.D.3.1 Essential Requirements for Concrete Supplier and Lab Testing Support

The material used in HI-STORM related concrete shall be procured from suppliers that have been qualified under Holtec QA program through appropriate validation and surveillance. The QA surveillance record on the concrete supplier must be current at the time of concrete placement. Among the many missions of the surveillance program are activities that are crucial to insure that all required *critical characteristics* shall be met such as, all scales used in the batching process are calibrated, delivery trucks are in good working condition, and all aggregate material stored at the facility is segregated. These parameters ensure that the batched concrete is in compliance with the Holtec concrete mix design.

With respect to the test lab services, surveillance of the lab ensures that all equipment used in testing of aggregates and concrete cylinder samples are calibrated. Additionally, inspections are completed on the concrete cylinder storage facilities as well as basic material controls. With these controls in place, the results of any aggregate testing or concrete cylinder testing can be confirmed to be accurate and reliable.

1.D.3.2 Concrete Mix Design and Material Requirements

A concrete mix design shall first be established to determine the necessary recipe to produce a HI-STORM concrete that meets the *critical characteristics* of the HI-STORM as specified in this section. Once the mix design is formulated, actual site testing shall be conducted to confirm the mix design. At the batch plant, the mix design will be used to make concrete for initial testing purposes. This initial batch shall be checked for slump and density. The mix design may be altered as necessary at this time until the desired results are achieved. Additionally, a total of ten cylinders from the final acceptable batch shall be taken for laboratory testing. These cylinders shall be used for compressive strength break test to determine the strength of the concrete mix.

With respect to individual aggregate testing, the provisions from ACI 349 those are germane to the plain concrete installed in and completely enclosed by the steel HI-STORM overpack structure are summarized herein. For example, the overpack concrete is not exposed to the environment, so provisions in the ACI standards for protecting concrete from the environment would not be applicable to the concrete contained in the overpack.

For the standard use local course and fine aggregates supplied by the local batch, a high level of confidence based on continued use in area concrete obviates the need for many of the aggregate testing recommended by ASTM C33. However, certain testing relevant to confirming the acceptability of the aggregate is required by this specification. For both the local fine and course aggregate, laboratory testing shall be carried out to confirm grading per ASTM C33 as well as the test per ASTM C117 to determine materials finer than 200 sieve. A laboratory technician shall also visually inspect the source pile to evaluate the aggregates for any deleterious substances or organic impurities. If this visual inspection reveals any evidence of deleterious substances or organic impurities, additional aggregate testing that addresses deleterious substances per ASTM C33 for both fine and course aggregates as well as organic impurities testing per ASTM C40 for the local fine aggregate shall be conducted.

For the specially supplied dense aggregate that is supplied from an outside source, applicable grading and 200 sieve testing shall be completed.

1.D.4 Construction Requirements

Method of placement of the concrete is important to achieving the desired properties in the concrete. It is imperative to achieve a concrete placement with no voids. In order to accomplish this, procedural steps shall be in place to control the placement technique with respect to lift height and vibratory agitation. The concrete shall be placed in the HI-STORM in two foot (approximate) lifts. Vibration of poured concrete shall be such that the vibrator is inserted and removed in a vertical movement with no dragging of the vibrator through the concrete. Vibrator placement shall be based on the size of the vibrator as detailed in ACI-309R.

The slump of the concrete shall be checked as necessary prior to placement to ensure that the concrete is suitable for pumping.

Appropriate measures shall be taken for hot and cold weather conditions as prescribed by ACI-305R and ACI-306R, respectively.

1.D.5 Testing Requirements

Concrete may be tested for temperature, slump, and density for each truck prior to placement in the HI-STORM for informational purposes. Official samples, as required by the applicable Holtec procedure, shall be taken from the approximate middle of the truck discharge and will become the sample of record for slump, temperature, and density. Additionally, compressive test cylinder samples shall be taken of a quantity to support required break tests as detailed in the governing Holtec procedure and will ensure a representative sample of the concrete is tested in accordance with ACI 349-85. At a minimum one set of samples must be taken for each HI-STORM. Samples taken in the field should be stored as best possible to protect the samples from extreme temperature conditions. Compressive break strengths of the official concrete cylinder samples taken shall be tested for the required minimum concrete strength. The compressive strength of concrete is observed to increase monotonically with the time of curing [1.D.3]. Therefore, break tests resulting in a compressive strength exceeding the minimum required compressive strength may be used as the official concrete break data in lieu of waiting for 28-day breaks.

1.D.6 References

[1.D.1] Carette and Malhorta, "Performance of Dolostone and Limestone Concretes at Sustained High Temperatures," Temperature Effects on Concrete, ASTM STP 858.

[1.D.2] Schneider and Horvath, "Behaviour of Ordinary Concrete at High Temperature," Vienna Technical University – Institute for Building Materials and Fire Protection, Research Report Volume 9.

[1.D.3] Concrete Manual, 8th Edition, US Bureau of Proclamation, Denver, Colorado, 1975.

Table 1.D.1
Requirements for Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density in overpack body (Minimum) (see Table 3.2.1 for information on maximum concrete density)	140 lb/ft^3
Density in lid and pedestal (Minimum) (See Table 3.2.1 for information on maximum concrete density)	140 lb/ft^3 (HI-STORM 100S Version B does not have a concrete-filled pedestal)
Specified Compressive Strength	3,300 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318.1-89(92)
Cement Type and Mill Test Report	Type II; (ASTM C 150 or ASTM C595)
Aggregate Type	Fine and coarse aggregate as required (Note 2)
Nominal Maximum Aggregate Size	1-1/2 (inch)
Water Quality	Deleted
Material Testing	See Note 4.
Admixtures	Deleted
Maximum Water to Cement Ratio	0.5 (Table 4.5.2)
Maximum Water Soluble Chloride Ion Cl in Concrete	1.00 percent by weight of cement (Table 4.5.4) (See Table 1.D.2, Note 1)
Concrete Quality	Deleted
Mixing and Placing	See Note 6.
Consolidation	Deleted
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Through-Thickness Section Average† Temperature Limit Under Long Term Conditions	300°F (See Note 3)
Through-Thickness Section Average† Temperature Limit Under Short Term Conditions	350°F (Appendix A, Paragraph A.4.2)
Aggregate Maximum Value†† of Coefficient of Thermal Expansion (tangent in the range of 70°F to 100°F)	6E-06 inch/inch/°F (NUREG-1536, 3.V.2.b.i.(2)(c)2.b)

† The through-thickness section average is the same quantity as that defined in Paragraph A.4.3 of Appendix A to ACI 349 as the mean temperature distribution. A formula for determining this value, consistent with the inner and outer surface averaging used in this FSAR, is presented in Figure A-1 of the commentary on ACI 349. Use of this quantity as an acceptance criterion is, therefore, in accordance with the governing ACI code.

†† The following aggregate types are a priori acceptable: limestone, marble, basalt, granite, gabbros, or rhyolite. The thermal expansion coefficient limit does not apply when these aggregates are used. Careful consideration shall be given to the potential of long-term degradation of concrete due to chemical reactions between the aggregate and cement selected for HI-STORM overpack concrete.

Table 1.D.1 (continued)
Requirements for Plain Concrete

Notes:

1. Deleted

2. The coarse aggregate shall meet the requirements of ASTM C33 for class designation 1S from Table 3. However, if the requirements of ASTM C33 cannot be met, concrete aggregates that have been shown by special tests or actual service to produce concrete of adequate strength, unit weight, and durability meeting the requirements of Tables 1.D.1 and 1.D.2 are acceptable in accordance with ACI 349 Section 3.3.2. The high-density coarse aggregate percentage of Material Finer than No. 200 Sieve may be increased to 10 % if the material is essentially free of clay or shale.

3. The 300°F long term temperature limit is specified in accordance with Paragraph A.4.3 of Appendix A to ACI 349 for normal conditions considering the very low maximum stresses calculated and discussed in Section 3.4 of this FSAR for normal conditions. In accordance with this paragraph of the governing code, the specified concrete compressive strength is supported by test data and the concrete is shown not to deteriorate, as evidenced by a lack of reduction in concrete density or durability.

4. Tests of materials and concrete, as required, shall be made in accordance with standards of the American Society for Testing and Materials (ASTM) as specified here, to ensure that the *critical characteristics* for the HI-STORM concrete are achieved. ASTM Standards to be used include: C 31-96, C 33-82, C 39-96, C 88-76, C 131-81, C 138-92, C 143-98, C 150-97, C 172-90, C 192-95, C 494-92, C 637-73. More recent approved editions of the referenced standards may be used.

5. Deleted

6. Water and admixtures may be added at the job site to bring both the slump and wet unit weight of the concrete within the mix design limits. Water or admixtures shall not be added to the concrete after placement activities have started. The tolerance for individual and combined aggregate weights in the concrete batch may be outside of tolerances specified in ASTM C94, provided that the wet unit weight of the concrete is tested prior to placement and confirmed to be within the approved range.

Table 1.D.2: Testing Requirements for Plain Concrete

TEST	SPECIFICATION
Compression Test	ASTM C31, ASTM C39, ASTM C192
Unit Weight (Density)	ASTM C138
Maximum Water Soluble Chloride Ion Concentration	Federal Highway Administration Report FHWA-RD-77-85, "Sampling and Testing for Chloride Ion in Concrete" (Note 1)

Notes:

1. If the concrete or concrete aggregates are suspected of containing excessive amounts of chlorides, they will be tested to ensure that their contribution will not cause the water-soluble chloride concentration to exceed the required maximum. Factors to be considered will consist of the source of the aggregates (proximity to a salt water source, brackish area, etc.) and service history of the concrete made from aggregates originating from the same source. No specific tests are required unless the aggregates or water source are suspected of containing an excessive concentration of chloride ions.

SUPPLEMENT 1.I

GENERAL DESCRIPTION OF HI-STORM 100U SYSTEM

1.I.0 GENERAL INFORMATION

The HI-STORM 100U System is an alternative Vertical Ventilated Module (VVM) design to be used with the Holtec International Multi-purpose Canisters (MPCs) for dry storage of spent nuclear fuel at an Independent Spent Fuel Storage Installation (ISFSI). Information pertaining to the HI-STORM 100U System is generally contained in the "I" supplements to each chapter of this FSAR. Certain sections of the main FSAR are also affected and are appropriately modified for continuity with the "I" supplements. Unless superseded or specifically modified by information in the "I" supplements, the information in the main FSAR is applicable to the HI-STORM 100U System. Drawings specific to the HI-STORM 100U VVM are in Subsection 1.I.5. The Glossary has been appropriately augmented to include the terms particular to the HI-STORM 100U VVM.

1.I.1 INTRODUCTION

HI-STORM 100U, like HI-STORM 100[1] and HI-STORM 100S[2], is a vertical, ventilated dry spent fuel storage system engineered to be fully compatible with the presently certified HI-TRAC transfer casks and MPCs. HI-STORM 100U is an underground vertical ventilated module (VVM) designed to accept all MPC models for storage at an ISFSI (see Figure 1.I.1). ISFSIs employing the VVM may be designed for any number of MPCs and expanded to add additional storage modules as the need arises. Each VVM stores one MPC.

The design and operational attributes of the HI-STORM 100U VVM, described in the following paragraphs pursuant to the provisions of 10CFR72.24(b), are subject to intellectual property rights in the U.S. and abroad under the patent laws governing the respective jurisdictions.

1.I.2 GENERAL DESCRIPTION OF HI-STORM 100U SYSTEM

1.I.2.1 HI-STORM 100U Vertical Ventilated Module

The VVM provides for storage of MPCs in a vertical configuration inside a subterranean cylindrical cavity entirely below the top-of-the-grade (TOG) of the ISFSI (Figure 1.I.2 provides identification of the TOG). The MPC Storage Cavity is defined by the Cavity Enclosure Container (CEC), consisting of the Container Shell integrally welded to the Bottom Plate. The top of the Container Shell is stiffened by the Container Flange (a ring shaped flange), which is also integrally welded. As shown in licensing basis drawings provided in Section 1.I.5, all of the constituent parts of the CEC are made of thick low carbon steel plate (See Table 2.I.8 for component materials). In its installed configuration, the CEC is interfaced with the surrounding subgrade for most of its height except for

[1] U.S. Patent No. 6,064,710 dated May 16, 2000.

[2] U.S. Patent No. 6,718,000 dated April 6, 2004.

the top region where it is girdled by the ISFSI pad. The ISFSI pad serves several purposes in the HI-STORM 100U storage system, such as:

- It provides an essentially impervious barrier of reinforced concrete against seepage of water from rain/snow into the subgrade.
- It provides the interface surface for the CEC flange.
- It helps maintain a clean, debris-free region around the VVMs.
- It provides the necessary riding surface for the cask transporter (see Figure 1.I.7).

The ISFSI pad is actually composed of two distinct regions separated by suitably engineered expansion joints. These are referred to as (see Figure 1.I.3):

i. the VVM Interface Pad (VIP) and
ii. the Top Surface Pad (TSP).

As its name implies, the VIP is in close contact with the Container Flange and the upper part of the Container Shell for sealing and shielding purposes. In Figures 1.I.1 and 1.I.2, the elevated portion of the ISFSI pad is the VIP.

The balance of the ISFSI pad, lower in elevation than the VIP, is the top surface pad (TSP). The TSP carries no significant loads except during the movement of the cask transporter over portions of its surface. The substantial difference in the dead load patterns on the two regions of the ISFSI pad warrants that the two regions be physically disconnected so that differential settlement between the two do not produce (undesirable) flexural and shear loadings. Governing codes for the ISFSI pad design and construction are described (see Supplement 2.I) to ensure a high integrity design. Expansion joints are placed between the two pads where necessary to ensure that vertical movements are independent. As discussed in Supplement 3.I, an optional concrete encasement around the coated external surface of the CEC may be added to control the pH at the CEC-to-subgrade interface.

Corrosion mitigation measures commensurate with site-specific conditions are implemented on below-grade external surfaces of the CEC. A corrosion allowance (metal wastage) equal to 1/8" on the external surfaces of the VVM in contact with the subgrade is nevertheless assumed in the structural evaluation in Supplement 3.I. All external and internal surfaces of the VVM are coated with an appropriate surface preservative. The top surfaces of the MPC Bearing Pads are equipped with stainless steel liners so that the MPC is not resting directly on carbon steel components. Details of corrosion mitigation measures are described in Section 3.I.4.

With the Closure Lid removed, the CEC is a closed bottom, open top, thick walled cylindrical vessel that has no penetrations or openings. Thus, groundwater has no path for intrusion into the interior space of the MPC storage cavity. Likewise, any water that may be introduced into the MPC storage cavity through the air passages in the top lid will not drain out on its own. The Bottom Plate of the CEC is round and slightly larger in diameter than the Container Shell to accommodate an all around weld between the plate and the shell.

The Support Foundation has circular VVM Lateral Support Recessed Regions to locate and contain lateral motion of each VVM with respect to the Support Foundation. The VVM Support Foundation and the underlying substrate must be sufficiently strong to prevent significant long-term settlement under the weight of the loaded storage cavities. The appropriate requirements on the Support Foundation's structural strength and the applicable industry code are specified in Supplement 2.I of this FSAR. Like the ISFSI pad above, the Support Foundation is classified as an "interfacing structure" in this FSAR.

The MPC Bearing Pads and the Divider Shell, two parts internal to the CEC, are important to the VVM's thermal performance. The Divider Shell, as its name implies, is a vertical cylindrical shell concentrically situated in the CEC. The Divider Shell creates an outer annular coolant air or intake plenum and an inner annular coolant air space around the MPC. The bottom end of the Divider Shell has cutouts to enable incoming air streaming down the intake plenum to enter the inner coolant air space from around the circumference of the Divider Shell in a symmetric manner (Figures 1.I.2 and 1.I.4). The sectors of the Divider Shell that rest on the CEC Bottom Plate are also the locations where MPC Bearing Pads provide for a Bottom Plenum underneath the MPC for access of coolant air. The cutouts in the Divider Shell are sufficiently tall to ensure that if the cavity were to be filled with water, the bottom region of the MPC would be submerged for several inches before the water level reaches the top edge of the cutouts. This design feature is important to ensure uncompromised thermal performance of the system under any conceivable accidental flooding of the cavity by any means whatsoever. The Divider Shell is laterally restrained in the horizontal plane at its bottom end by the Divider Shell Restraints and rotationally restrained in the horizontal plane by the MPC Bearing Pads. The Divider Shell is not attached to the CEC; this allows convenient removal for decommissioning, for unplanned in-service maintenance, or for any other unforeseeable reason. The Divider Shell's interface with the Closure Lid features a small gap to permit the Divider Shell to expand freely from heating by ventilation air.

In addition to the lateral restraints at the bottom, the Divider Shell is also restrained against lateral movement at the top by the cylindrical protrusion in the Closure Lid. In addition, the Divider Shell is equipped with Upper and Lower MPC Guides. The Upper MPC Guides are radially symmetric and located at the elevation of the MPC's top lid. The Upper MPC Guides serve to guide the MPC down to the Lower MPC Guides and MPC Bearing Pads during the MPC's lowering operation, as well as to limit the MPC's lateral movement relative to the CEC, during an earthquake event, to a fraction of an inch.

The cylindrical surface of the Divider Shell is equipped with insulation to ensure that the heated air streaming up around the MPC in the inner coolant air space causes minimal preheating of the air streaming down the intake plenum. As discussed in Supplement 3.I.4, the insulation material is selected to be water and radiation resistant and non-degradable under accidental wetting.

Finally, the Closure Lid shown in Figure 1.I.6 completes the physical embodiment to the VVM. The Closure Lid is a steel structure filled with shielding concrete. The design of the top lid fulfills the following principal performance objectives:

i. Both the inlet and outlet air passages are located in the Closure Lid, so there are no lateral radiation leakage paths during the MPC lowering or raising operation. The need for shield blocks (necessary to close off vents in some aboveground HI-STORM 100 overpacks) is eliminated.

ii. Both inlet and outlet passages are radially symmetric so that the air cooling action in the system is not affected by the change in the horizontal direction of the wind.

iii. By locating the air inlet at the periphery of the Closure Lid and the air outlet at its top central axis, mixing of entering and exiting air streams is essentially eliminated.

iv. The inlet and outlet air passages are made of "formed and flued" heads (i.e., surfaces of revolution) that serve three major design objectives as noted below.

   a. The curved passages eliminate any direct line of sight to the MPC storage space and serve as an effective means to scatter the photons streaming from the stored fuel.

   b. The curved steel plates significantly increase the load bearing capacity of the Closure Lid much in the manner as a curved beam exhibits considerably greater lateral load bearing capacity in comparison to its straight counterpart. This design feature is a valuable attribute if a "beyond-the-design basis" impact scenario involving a large and energetic missile needs to be evaluated for a particular ISFSI site.

   c. The curved passages, as is well known in classical hydraulics, provide for minimum loss of pressure in the coolant air stream, resulting in a more vigorous ventilation action.

v. The Closure Lid rests on the Container Flange and is gasketed to minimize foreign material intrusion.

vi. The top surface of the Closure Lid is also curved and extended beyond the air inlet perimeter to efficiently drain off rainwater.

vii. The Container Flange restrains the Closure Lid against horizontal movement, during a Design Basis Earthquake event or a tornado missile strike.

viii. The radially symmetric air inlet passage in the lid is geometrically aligned with the annular opening formed between by the Divider Shell and the CEC Shell.

ix. Because the inlet opening extends around the circumference of the Closure Lid, the hydraulic resistance to the incoming airflow, a common limitation in ventilated modules, is minimized. A similar airflow resistance minimization facility is built into the pathway for the exiting air. A circumferentially circumscribing vent opening is

also quite obviously less apt to be completely blocked under even most extreme environmental phenomena involving substantial quantities of debris.

x. To minimize the VVM's height, a portion of the Closure Lid extends into the cylindrical space above the MPC. This cylindrical below-surface extension of the Closure Lid is also made of steel filled with shielding concrete to maximize the blockage of skyward radiation issuing from the MPC.

xi. All inlet and outlet air passages are equipped with screens, as in the aboveground HI-STORM overpacks, to prevent debris, insects, and small animals from entering the VVM. Although the screen is a non-structural member, it is designed for long-term durability and easy maintainability to ensure that its installation, removal, and maintenance are ALARA.

Finally, particular attention is paid to the design of the exit vent assembly (at the top of the outlet air passages in Figure 1.I.2) to ensure that wind-driven rain at up to 45° inclination from the vertical will not have a direct line of sight to the vertically oriented portion of the air passage in the Closure Lid.

xii. As can be seen from the drawings in Section 1.I.5, the Closure Lid is substantially larger in diameter than the Divider Shell in the CEC and the MPC is positioned to be at a significant vertical depth below the top of the Container Flange. These geometric provisions ensure that the Closure Lid will not fall into the MPC storage cavity space and strike the MPC if it were accidentally dropped during its handling. An accidental drop of the MPC, however, can lead to a collision with the top of the Divider Shell. The Divider Shell, if damaged due to a handling accident, can be readily removed and repaired or replaced without affecting any other parts of the VVM. Because the Closure Lid is the only removable heavy load, the carefully engineered design features to facilitate recovery from its accidental drop provide added assurance that a handling accident at the ISFSI will not lead to radiological release. This additional measure against accidental Closure Lid drop does not replace the drop prevention features mandated in this FSAR on heavy load lifting devices such as the cask transporter (illustrated in Figure 1.I.7) that have been a standard and established requirement in the HI-STORM 100 docket.

From a jurisdictional standpoint, the CEC, the Container Flange, and the Closure Lid, constitute the body of the VVM. The Support Foundation on which the VVM rests, however, must be designed to meet certain structural criteria to minimize long-term settlement and physical degradation from aggressive attack of the materials in the surrounding subgrade. Likewise, the Top Surface Pad serves to augment shielding, but is mainly needed to provide a sufficiently stiff roadway for the transporter. Similarly, the VVM Interface Pad (Figure 1.I.2) serves to augment shielding, as a barrier against gravity induced seepage of rain or floodwater around the VVM body, and as a barrier against a missile directed towards the underground portion of the CEC structure. The essential structural requirements applicable to the design of the Support Foundation, the VVM Interface Pad, and the Top Surface Pad for proper functioning of the VVM are provided in Supplement 2.I (Principal Design Criteria). Similarly, typical physical characteristics of the surrounding substrate are provided

in Supplement 2.I. This data is intended to provide guidelines for the design of SSCs proximate to the VVM to ensure that the VVM, regardless of the wide variations in the properties at an ISFSI site, will render its intended function for the duration of its Design Life.

The foregoing description of the VVM clearly indicates that the principal function of the VVM structure is to provide the biological shield and cooling facility. However, for conservatism, stress limits of the "Level A" service condition in Subsection NF of the ASME Code are applied to establish the embedded structural margins of safety in the primary load bearing parts of the VVM under normal conditions of storage. For short term and accident conditions (i.e., earthquakes, missile strike, etc.), the continued functional adequacy of the system is the appropriate criterion. For the VVM, continued functional adequacy under accident or extreme environmental events demands absence of a complete blockage of the ventilation passages and a non-significant amount of loss of shielding. Supplement 2.I provides complete details on the applicable design criteria.

All MPC types certified for storage in the aboveground overpacks can be stored in the below ground VVM. The chief distinguishing features of the VVM are its low profile and subterranean configuration. The Container Shell is buried below the ISFSI Pad for virtually its entire height, resulting in a near complete blockage of laterally emanating radiation from the stored fuel.

In summary, the notable design and operational features of the HI-STORM 100U System are:

i. The MPC is supported on MPC Bearing Pads to provide an inlet air plenum at the bottom of the storage cavity (Figure 1.I.2). The bottom of the MPC, however, will be in contact with water if the cutouts at the bottom of the Divider Shell were to be filled with water cutting off feed air. As long as the MPC is wetted with water, the peak cladding temperature of the stored spent fuel will not exceed the regulatory off-normal condition temperature limit. Thus, the VVM configuration provides a built-in protection against flood events.

ii. Like the HI-STORM 100A and 100SA models, tipover of the canister in storage is not possible.

iii. Although the modules may be closely spaced, as illustrated in Figure 1.I.5, the design permits any MPC located in any cavity to be independently accessed and retrieved using a HI-TRAC transfer cask.

iv. A cask transporter typical of those used in numerous Holtec ISFSI projects for on-site transport of loaded HI-TRACs and HI-STORMs can provide the means to deliver the loaded HI-TRAC to the HI-STORM 100U VVM and to carry out the MPC lowering operation (Figure 1.I.7). The same cask transporter can also be used to remove an MPC from storage and place it in a recipient HI-TRAC transfer cask.

v. To exploit the biological shielding provided by the surrounding soil subgrade, the MPC is entirely situated well below the top-of-grade level. The open plenum above the MPC also acts to boost the ventilation action of the coolant air.

vi. Because the VVM is rendered into an integral part of the subgrade, it cannot be translocated to another ISFSI site. It also cannot be lifted and, therefore, is not subject to the potential for a handling accident.

vii. Removal of water from the bottom of the storage cavity can be carried out by the simple expedient use of a flexible hose inserted through either the inlet or the outlet passageway.

viii. As discussed in Supplement 3.I.4, all exposed surfaces of the VVM are coated with proven surface preservatives that meet the toxicological and extraction test requirements of ANSI/NSF Standard 61.

ix. The VVM is a formed metallic welded structure with a removable Closure Lid. The Closure Lid is also a formed metallic welded structure but filled with shielding concrete. The requirements on the shielding concrete are specified in Appendix 1.D.

As can be readily deduced from the above description of the VVM, the MPC storage cavity (consisting of the Container Shell and Bottom Plate) is at or near ambient temperature during normal operations. The only portions of the VVM in contact with heated ventilation air are the Divider Shell and the domed annular outlet in the Closure Lid, neither of which is in contact with the subgrade soil.

It should be recognized that the depth of the MPC Storage cavity determines the height of the hot air column in the annular region during the system's operation. Therefore, deepening the cavity has the beneficial effect of increasing the quantity of the ventilation air and, thus, enhancing the rate of heat rejection from the stored MPC. Further, lowering the MPC in the MPC Storage cavity will increase the subterranean depth of the radiation source, making the site boundary dose even more miniscule. To ensure that the thermal and shielding performance is the bounding minimum, the top of the MPC is assumed to be at its maximum permissible elevation with respect to the Top-of-the-Grade and the MPC Storage Cavity depth is assumed to be accordingly at its permitted minimum in all thermal and shielding analyses reported in Supplements 4.I and 5.I, respectively, and in the drawings provided in Section 1.I.5. At a specific ISFSI site, the user has the latitude to deepen the VVM cavity and situate the MPC at a deeper depth using the §72.48 process.

The VVM implements seals or gaskets at the Closure Lid. The outer seal is a weather seal (between the Closure Lid and the top of the Divider Shell), which facilitates maintenance by minimizing foreign material intrusion into the MPC storage cavity. The inner seal (between the Closure Lid skirt and the Divider Shell (not shown on the licensing drawing 4501)) provides an enhanced barrier against mixing of inlet and outlet air in the annular space between the Divider Shell and the cylindrical protrusion in the Closure Lid (even though the pressure differential between the two sides is extremely low – less than a few inches of water). The outer seal relies on the weight of the Closure Lid to insure sealing. A polymeric gasket made from EPDM[3] is preferred for this purpose. The inner seal is made of a durable radiation and heat resistant material and designed to have no credible mechanism for significant degradation or detachment from its sealing location. The seals do not

[3] Radiation resistant polymeric gasket materials are available from the Presray and Pawling Corporations, for example.

provide a safety function because their loss during operation would not have an effect on safe operation of the system.

Finally, the physical hardening of the VVM against impulsive and impactive loadings is a major consideration in the embodiment of the HI-STORM 100U System. Quite obviously, the low physical profile of the VVM reduces the probability of impact from a missile or a projectile. In addition, to impute maximum margin against extreme environmental phenomena loads, the Closure Lid is a METCON® (metal/concrete) structure engineered to possess considerably greater strength reserve than that required to prevent design basis missiles from penetrating into the MPC storage cavity, as demonstrated by analysis in Supplement 3.I. Another design consideration is protection against intrusion of rainwater and other liquid matter into the MPC storage cavity. In contrast to typical ventilated modules, the VVM air passages are elevated above the Top-of-the-Grade, providing a physical barrier against the intrusion of any accumulating pool of fluid (including combustibles) on the ISFSI surfaces into the module cavity. A significantly enhanced level of protection against incident missiles and an improved barrier against ingress of rainwater or spilled fluids into the module cavity space, and a design that is ideally configured for a flood event, are among the many distinguishing features of the HI-STORM 100U System.

1.I.2.2 HI-STORM 100U System Sequence of Operations

Fuel loading operations and MPC preparation are identical for the VVM as they are with the other HI-STORM overpack designs. The HI-TRAC transfer cask is used for on-site transport of the loaded MPC from the MPC preparation area to the VVM at the ISFSI. The Closure Lid will have been previously removed from the VVM. The cask transporter carrying the transfer cask and the MPC moves over the top of the open VVM where the HI-STORM mating device (shown beneath the HI-TRAC in Figure 1.I.7) is in place. The MPC inside the transfer cask is lifted slightly by the cask transporter (or an equivalent heavy load handling device) to allow the transfer cask pool lid to be removed. Once the pool lid is removed, the heavy load handling device is used to lower the MPC into the VVM. The transfer cask and mating device are removed from the top of the VVM, the MPC lift connectors are removed, and the VVM Closure Lid is installed. Supplement 8.I provides a more detailed discussion of operations involving the HI-STORM 100U System. (The "mating device" aided MPC transfer operation is an exclusive intellectual property of Holtec International under U.S. Patent No. 6,853,797 B2 dated February 8, 2005.)

1.I.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Same as in Section 1.3.

1.I.4 GENERIC CASK ARRAYS

An ISFSI deploying the HI-STORM 100U System may use an unlimited number of VVMs. The preferred embodiment of the VVM array is a rectangular grid as illustrated in Figure 1.I.5. The minimum pitch between the VVM cavities is shown on the licensing drawing in Subsection 1.I.5. In either or both directions, the spacing can be increased by the site to ensure that any of the commercially available cask transporters can traverse the VVM arrays to provide autonomous access

to each stored MPC. This minimum spacing also serves to provide adequate shielding around each storage cavity.

No limit is placed on the maximum spacing. Multiple VVMs in an ISFSI shall be founded on a continuous support foundation to prevent an unacceptable level of differential settlement between adjacent VVMs and to enhance the seismic response characteristics of the ISFSI.

The design of the expansion joints between the VVM Interface Pad and the Top Surface Pad regions of the ISFSI Pad is guided by the need to physically decouple the settlement of the two regions due to long term creep effects.

Additional VVMs may be built adjacent to existing VVMs without imparting excessive dose to the construction crew, if a sufficient distance to loaded VVMs is kept. To ensure that this distance is kept, a "Radiation Protection Space" (RPS) boundary is specified in the drawing package in Section 1.I.5. This boundary shall not be encroached upon during any site construction effort. Subsection 2.I.6(xii) contains additional requirements on the design and qualification of the RPS to insure that the earthen shielding in the RPS shall be protected against a significant loss due to human error or natural events such as earthquakes and tornado borne missiles.

1.I.5 FIGURES AND DRAWINGS

Figures associated with Supplement 1.I and the licensing drawing package of the HI-STORM 100U VVM, pursuant to the requirements of 10CFR72.24(c)(3), are provided in this subsection. The material in the licensing drawing package in this section contains sufficient information to articulate major design features and general operational characteristics of the HI-STORM 100U VVM. Further, it is intended to serve as the control information to guide the preparation of the documents required to manufacture the components under the company's quality assurance system. Some key document types needed for manufacturing in the factory under the company's fail-safe configuration control protocol are:

- Purchasing Specifications (PSs)
- Manufacturing Drawing Package
- Holtec Standard Procedures (HSPs)
- Holtec Project Procedures (HPPs)
- Bill-of-Materials
- Fabrication and NDE Procedures
- Shop Travelers

Holtec's Quality Assurance Program requires that the entire array of manufacturing documents must remain in complete consonance with the Licensing Drawing Package (and other provisions in this FSAR) at all times.

Drawing Number/Sheet	Description	Rev.
4501	HI-STORM 100U Vertical Ventilated Module	6

[DRAWING WITHHELD PER 10 CFR 2.390]

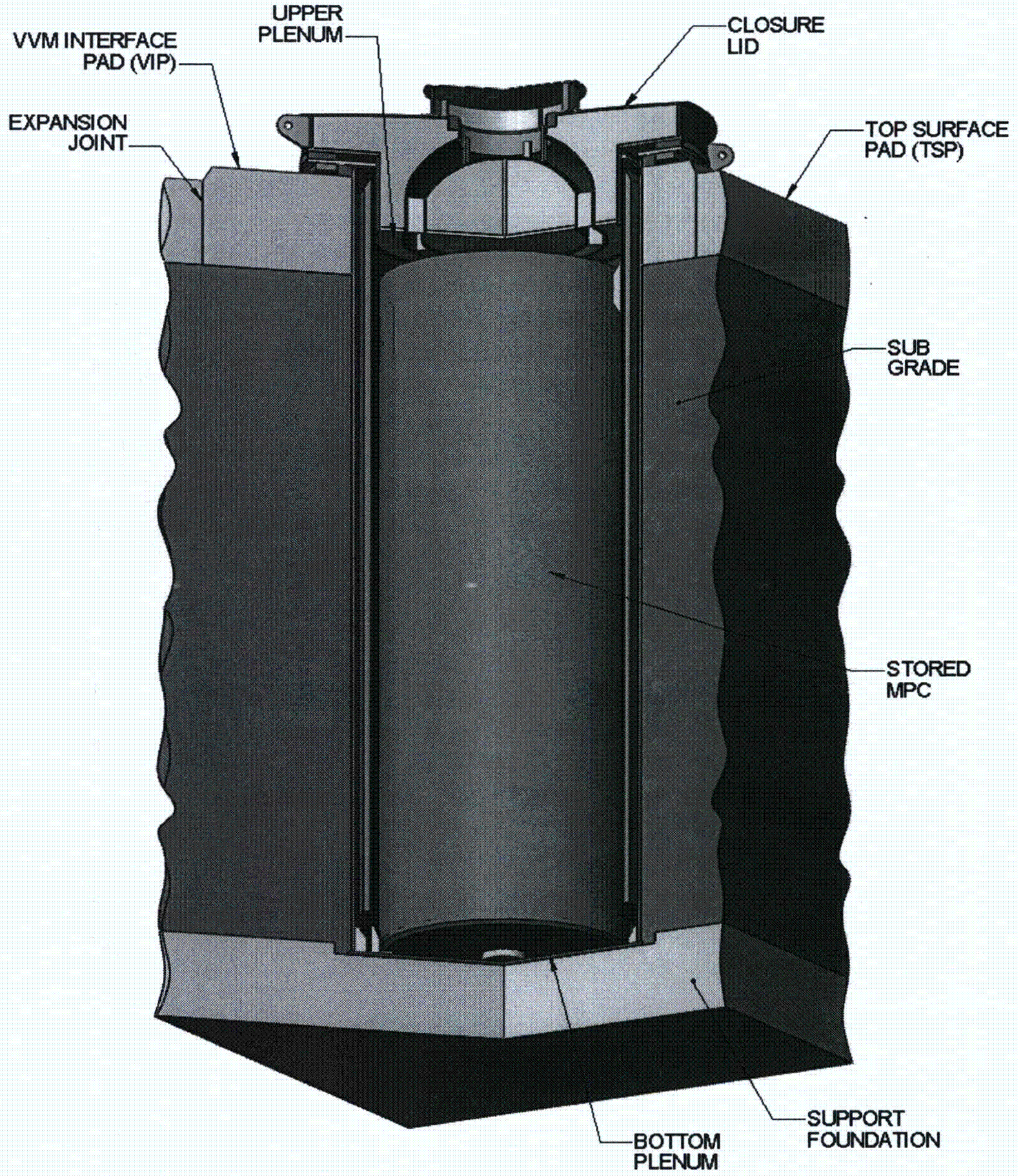


FIGURE 1.I.1: CUT-AWAY VIEW OF HI-STORM 100U SYSTEM)

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

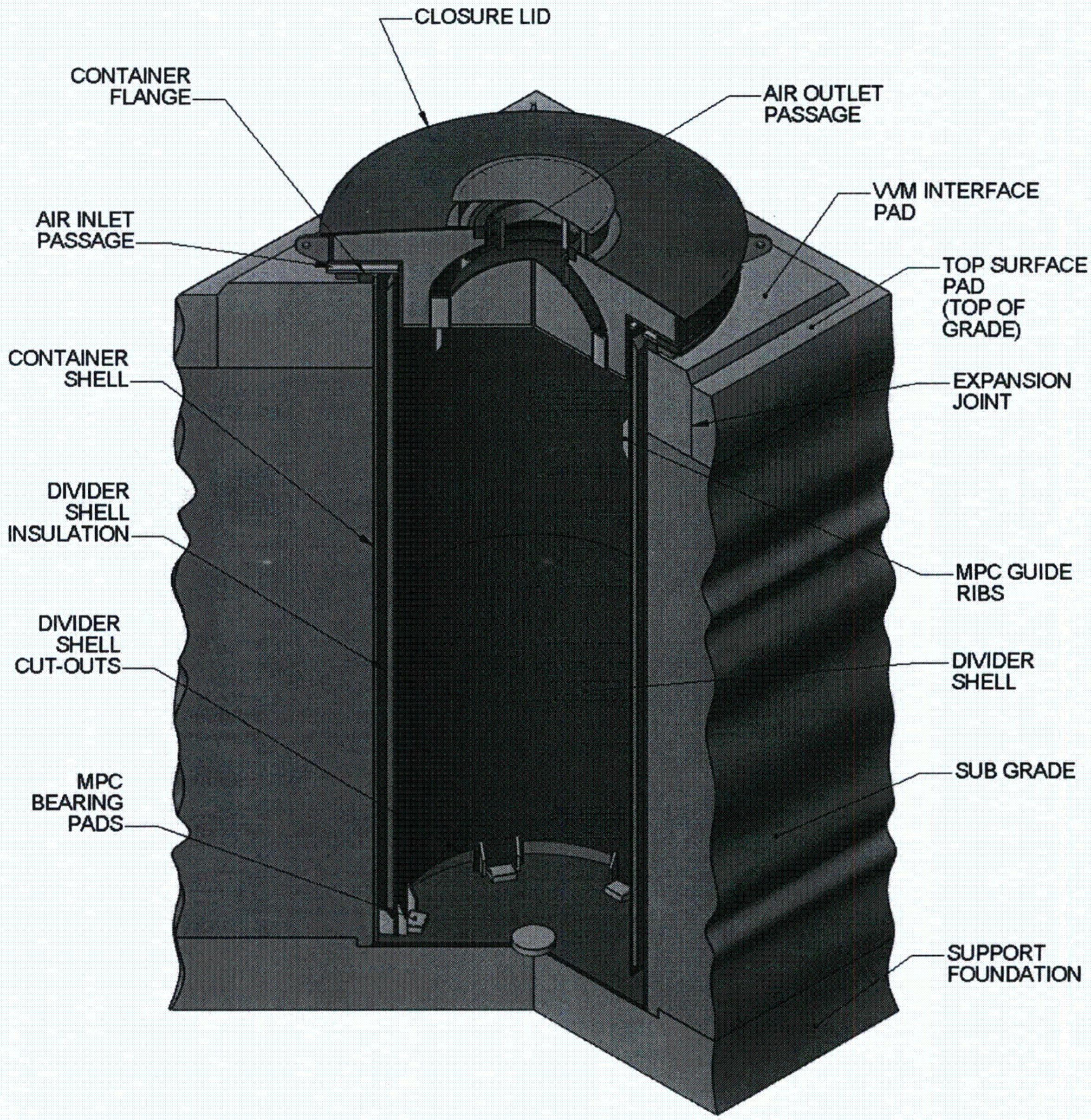


FIGURE 1.I.2: CUT-AWAY VIEW OF THE HI-STORM 100U VVM

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

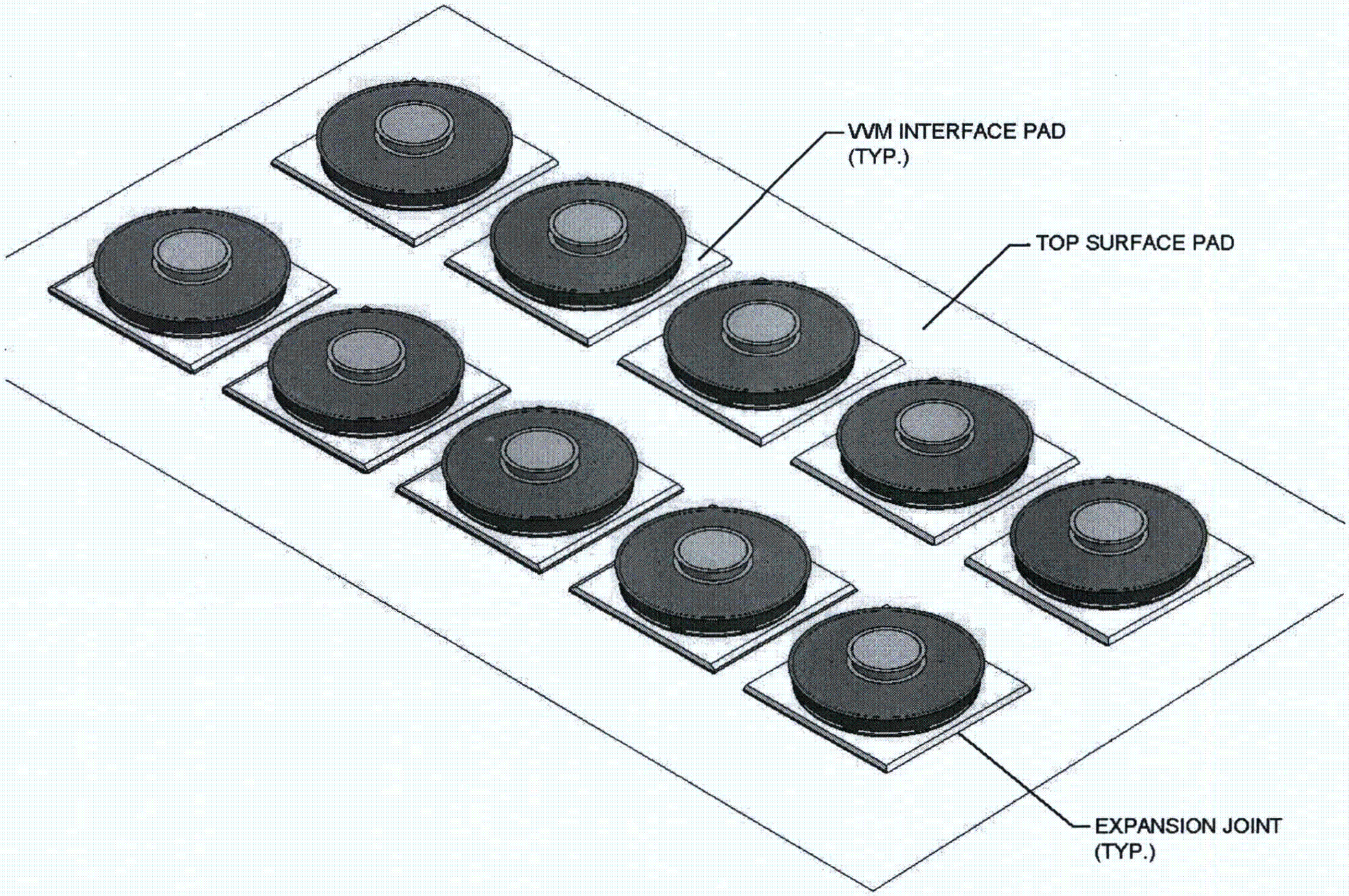


FIGURE 1.I.3: TYPICAL HI-STORM 100U SYSTEM ISFSI 2 x 5 ARRAY

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

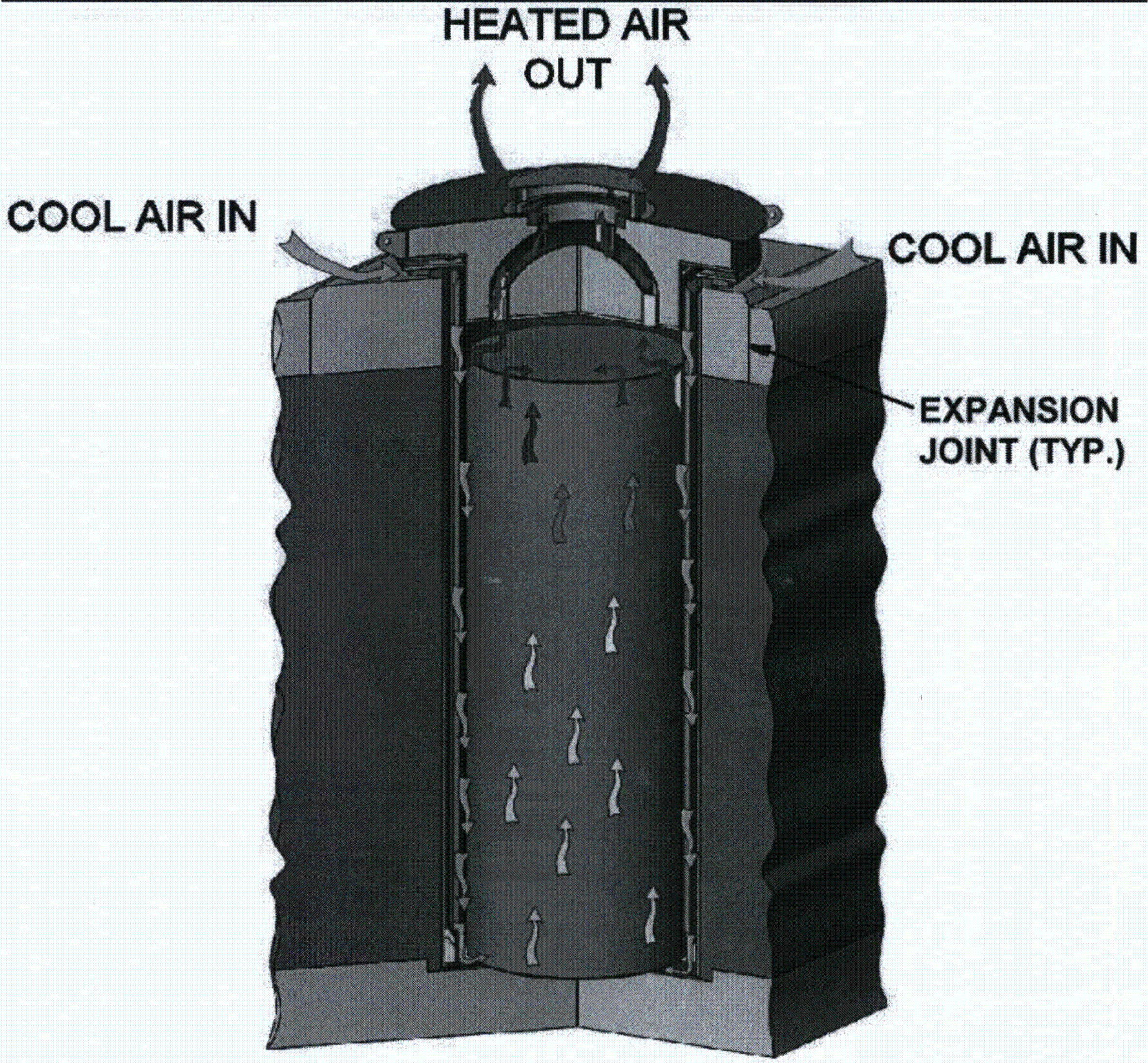


FIGURE 1.I.4: HI-STORM 100U SYSTEM AIR FLOW PATTERN

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

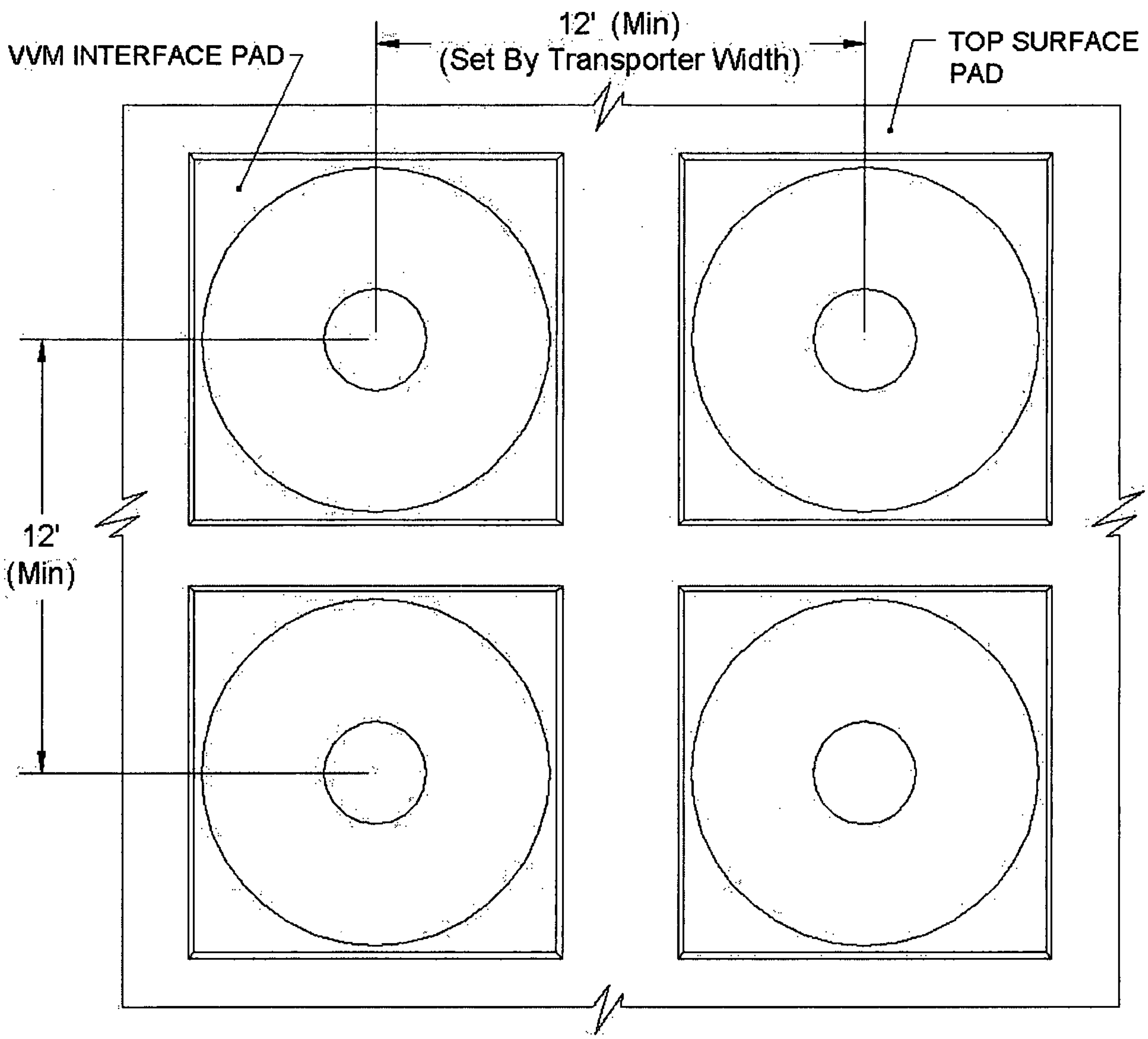


FIGURE 1.I.5: PLAN VIEW OF A 2X2 HI-STORM 100U SYSTEM STORAGE ARRAY

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

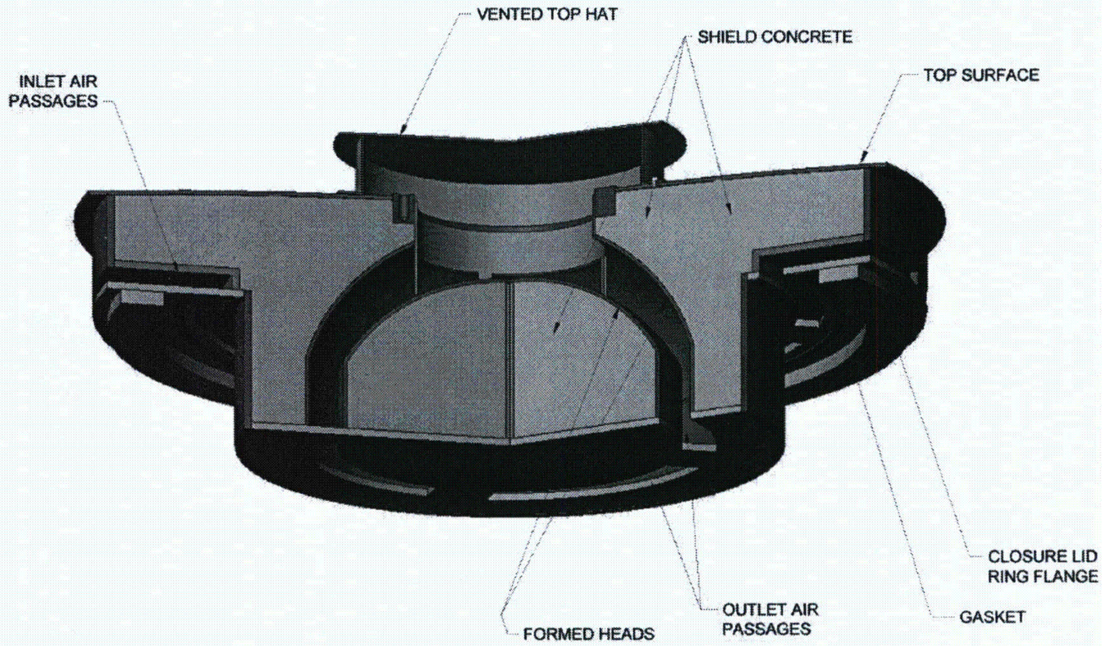


FIGURE 1.I.6; HI-STORM 100U VVM CLOSURE LID GENERAL ARRANGEMENT (SHOWN IN CUT-AWAY VIEW)

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

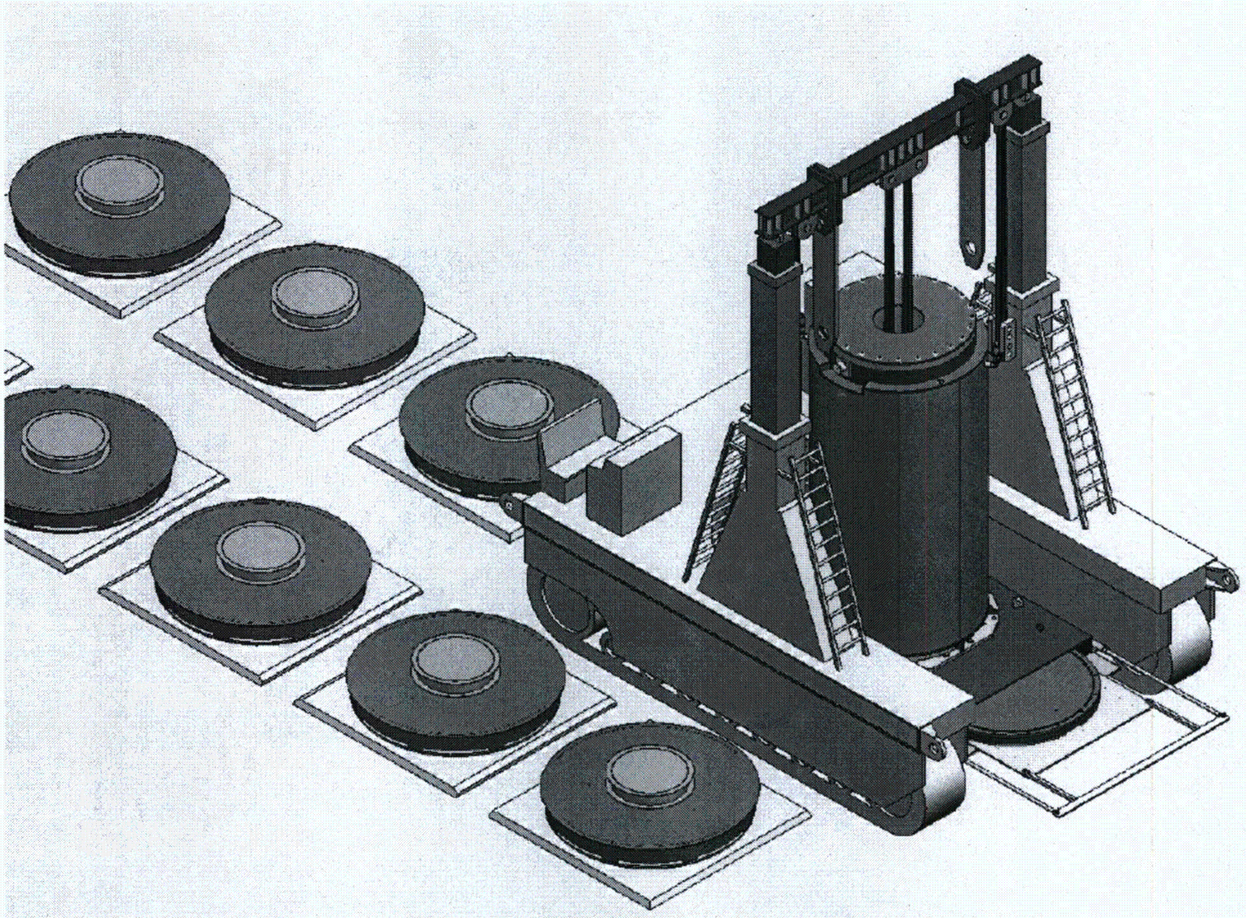


FIGURE 1.I.7; MPC TRANSFER IN A HI-STORM 100U VVM USING A VERTICAL CASK TRANSPORTER

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.I.5.

SUPPLEMENT 1.III: GENERAL DESCRIPTION OF THE MPC-68M

1.III.0 GENERAL INFORMATION

The list of MPC models for use in the HI-STORM 100 System is expanded to include the MPC-68M model which consists of a Metamic-HT BWR fuel basket inside the existing MPC enclosure vessel. Information pertaining to the MPC-68M is contained in the "III" supplements to each chapter of this FSAR. Unless superseded or specifically modified by information in the "III" supplements, the information in the main FSAR is applicable to the HI-STORM 100 System with the MPC-68M.

Each chapter in the HI-STORM FSAR has been updated by the addition of a supplement labeled n.III (n = chapter number) that contains all necessary information in support of Licensing Amendment Request #1014-8 to the HI-STORM 100 CoC. This series of supplements is focused solely on incorporating one specific MPC model containing a new BWR fuel basket design into the HI-STORM 100 system.

1.III.1 INTRODUCTION

The safety evaluation in the "III" Supplements supports the use of an alternate fuel basket, made entirely of Metamic-HT, for use within the current MPC Enclosure Vessel. The canister is referred to as MPC-68M. The same design/service life applied to the current MPC models applies to the MPC-68M.

1.III.2 GENERAL DESCRIPTION OF THE MPC-68M

1.III.2.1 MPC-68M Characteristics

The MPC-68M contains a 68 storage cell BWR fuel basket made of co-planar slotted plates of Metamic-HT (See Figure 1.III.1). The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. The Metamic-HT in the MPC-68M serves as the structural material of the basket and provides the necessary neutron absorption for maintaining the fuel in a sub-critical condition. The following design characteristics of MPC-68M are important to its function:

i. The fuel basket is assembled from a rectilinear gridwork of plates so that there are no bends or radii at the cell corners. 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 accident events, etc.).

ii. Precision extruded aluminum shims (the so-called "*Basket Shims*") are installed in the peripheral space between the fuel basket and the Enclosure Vessel to provide conformal

contact surfaces between the Basket Shims and the fuel basket and between the Basket Shims and the Enclosure Vessel shell. The axial holes in the corner basket shims serve as the passageway for the downwards flow of the helium gas under the thermosiphon action, which is intrinsic to the thermal performance of all MPCs in the HI-STORM 100 system.

iii. To facilitate an effective convective circulation inside the Canister, the Enclosure Vessel is pressurized to the same pressure level specified for the MPC-68 model. See Table 1.III.1.

iv. The fuel basket consists of adjacent square openings (cells) separated by one wall of neutron absorber.

The mechanical and thermophysical properties of Metamic-HT presented in Appendix 1.III.A to this supplement are extracted from the Metamic-HT Sourcebook referenced therein. The basket shims are made from creep resistant aluminum alloy similar to those used in HI-STAR 180 (Docket No. 71-9325) [2.III.6.2]. All design aspects of the MPC enclosure vessel (confinement boundary), including the vent and drain port arrangements in the MPC lid, are unchanged. The MPC-68M may be loaded in all licensed aboveground HI-STORM 100 overpacks.

1.III.2.2 Operational Characteristics

The MPC-68M canister is loaded in exactly the same manner as all other MPCs certified in the HI-STORM 100 docket, and uses the same ancillary equipment, (viz., lift cleats, lift yokes, Lid Welding Machine, Weld Removal Machine, Cask Transporter, Mating Device, Low Profile transporter or Zero Profile Transporter, Canister Upender, Forced Helium Dehydrator, the Hydrostatic pressure test system and the like). The operational characteristics of the HI-STORM overpack are unaffected.

All short-term operations, including draining of the MPC, welding of the lid, drying and filling of the MPC cavity with inert gas, and handling of the MPC remain unchanged from the existing practice, except that the dried and helium filled MPC-68M may reside indefinitely in the HI-TRAC transfer cask without the aid of the supplemental cooling system. The increased thermal conductivity of the Metamic-HT basket maintains the steady state fuel cladding temperatures below the ISG-11, Revision 3 [2.0.8] limits; thus eliminating the need for the supplemental cooling system. All other loading and unloading procedures and operations described in Chapter 8 apply to MPC-68M with the clarifications and limitations presented in Supplement 8.III.

1.III.2.2.1 Criticality Prevention

Metamic-HT is the designated neutron absorber and structural material in the MPC-68M. The properties of Metamic-HT and key characteristics, necessary for ensuring nuclear reactivity control, thermal, and structural performance of the basket, are presented in Appendix 1.III.A.

The entire basket is made of Metamic-HT, incorporating in the fuel basket a much greater B-10 concentration than is available in the fuel baskets designs with "attached" neutron absorber. This accrues three major safety and reliability advantages:

(i) The BWR basket may store high enrichment fuel (i.e., fuel with up to 4.8 wt% U-235 initial planar enrichment) without reliance on gadolinium or burn-up credit.
(ii) The neutron absorber cannot detach from the basket or displace within it.
(iii) Axial movement of the fuel with respect to the basket due to internal clearances has no reactivity consequence because the entire length of the basket contains the same concentration of the B-10 isotope.

During storage, criticality control measures beyond the integral neutron poison in 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 fuel storage space.

1.III.2.2.2 Structural Considerations

Due to the low density of the Metamic-HT material and the optimized fuel basket design, the loaded weight of the MPC-68M (see Supplement 3.III) is less than the loaded weight of the MPC-68. Therefore, the lifting features on the MPC are unchanged and all of the lifting and handling equipment used to lift and handle the loaded MPC-68 may also be used to lift and handle the loaded MPC-68M.

1.III.2.2.3 Thermal Considerations

The MPC-68M, for an identical heat load, is intrinsically capable of more cooling effectiveness than the MPC-68, due to the higher thermal conductivity of Metamic-HT (approximately 1 order of magnitude greater than the thermal conductivity of Alloy X), the use of full length aluminum basket shims, and the hard-anodizing of the basket and basket shim materials to obtain high emissivities. Thus satisfaction of the ISG-11 temperature limits with acceptable margins is assured.

1.III.2.2.4 Shielding Considerations

The MPC68M basket consists of homogeneously dispersed boron carbide (10% minimum by weight). The B-10 areal density of the Metamic-HT panels which make up the basket is consistent with the areal density of the Metamic classic neutron poison panels in the MPC-68, therefore provides equivalent neutron shielding. From an overall shielding perspective the MPC-68M is expected to provide similar, if not better, shielding characteristics as the MPC-68. The MPC enclosure vessel, overpack, and transfer cask shielding properties are not modified.

1.III.2.3 Cask Contents

The MPC-68M is designed to accommodate up to sixty-eight intact BWR fuel assemblies. Up to sixteen damaged fuel containers (DFCs) containing BWR damaged fuel assemblies and/or up to eight DFCs containing fuel debris may be stored in the following fuel storage locations: 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (Figure 1.III.2), with the remaining fuel storage locations filled with intact BWR fuel assemblies. Table 1.2.2 as supplemented by Table 1.III.1 provides the key system data and parameters for the MPC-68M.

1.III.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Same as in Section 1.3.

1.III.4 GENERIC CASK ARRAYS

Same as in Section 1.4.

1.III.5 DRAWINGS

[DRAWINGS WITHHELD PER 10 CFR 2.390]

Table 1.III.1
Key Parameters for MPC-68M

	BWR
MPC internal environment Helium fill (99.995% fill helium purity) (heat load ≤ 28.19 kW) (heat load >28.19 kW)	(all pressure ranges are at a reference temperature of 70°F) > 29.3 psig and < 48.5 psig OR 0.1218 +/-10% g-moles/liter > 45.5 psig and < 48.5 psig
B_4C content in Metamic-HT (wt. %)	As specified on drawing in Section 1.5

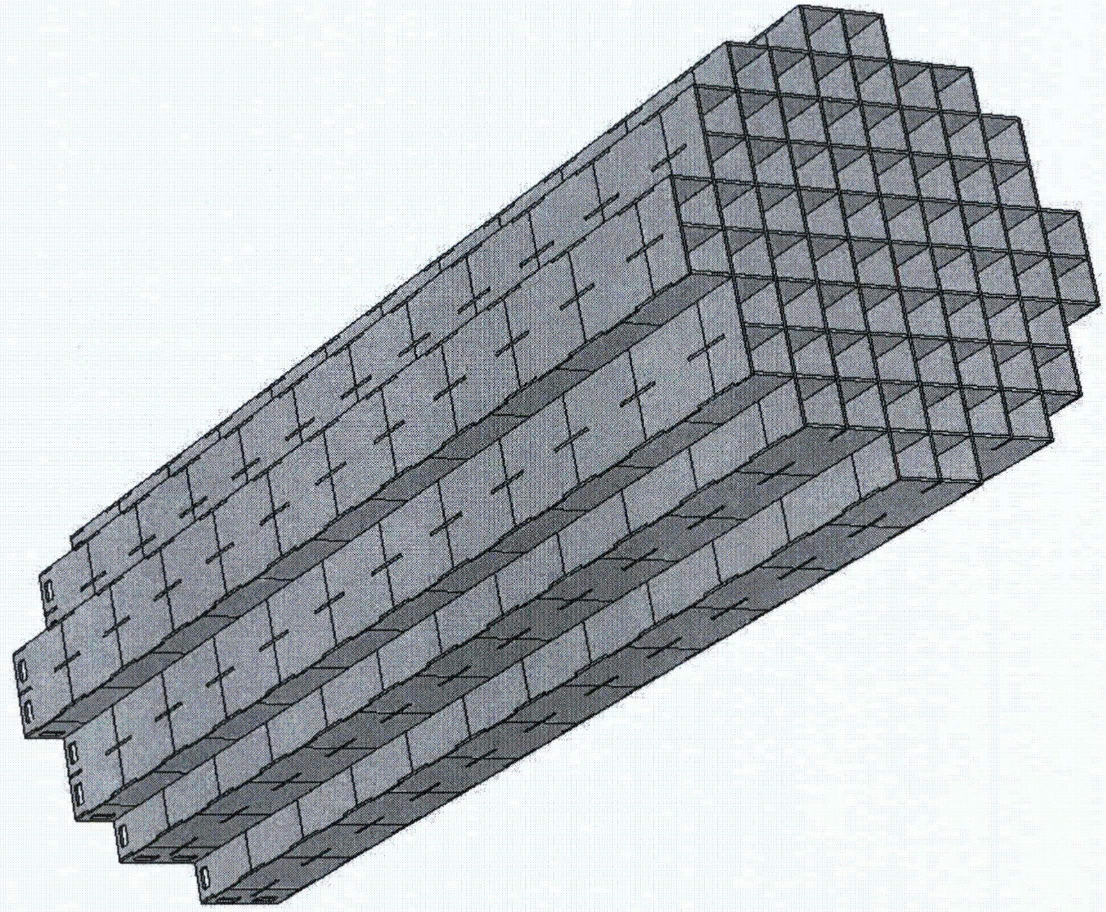


FIGURE 1.III.1: ISOMETRIC VIEW OF THE MPC-68M BASKET
(BASKET SHIMS NOT SHOWN)

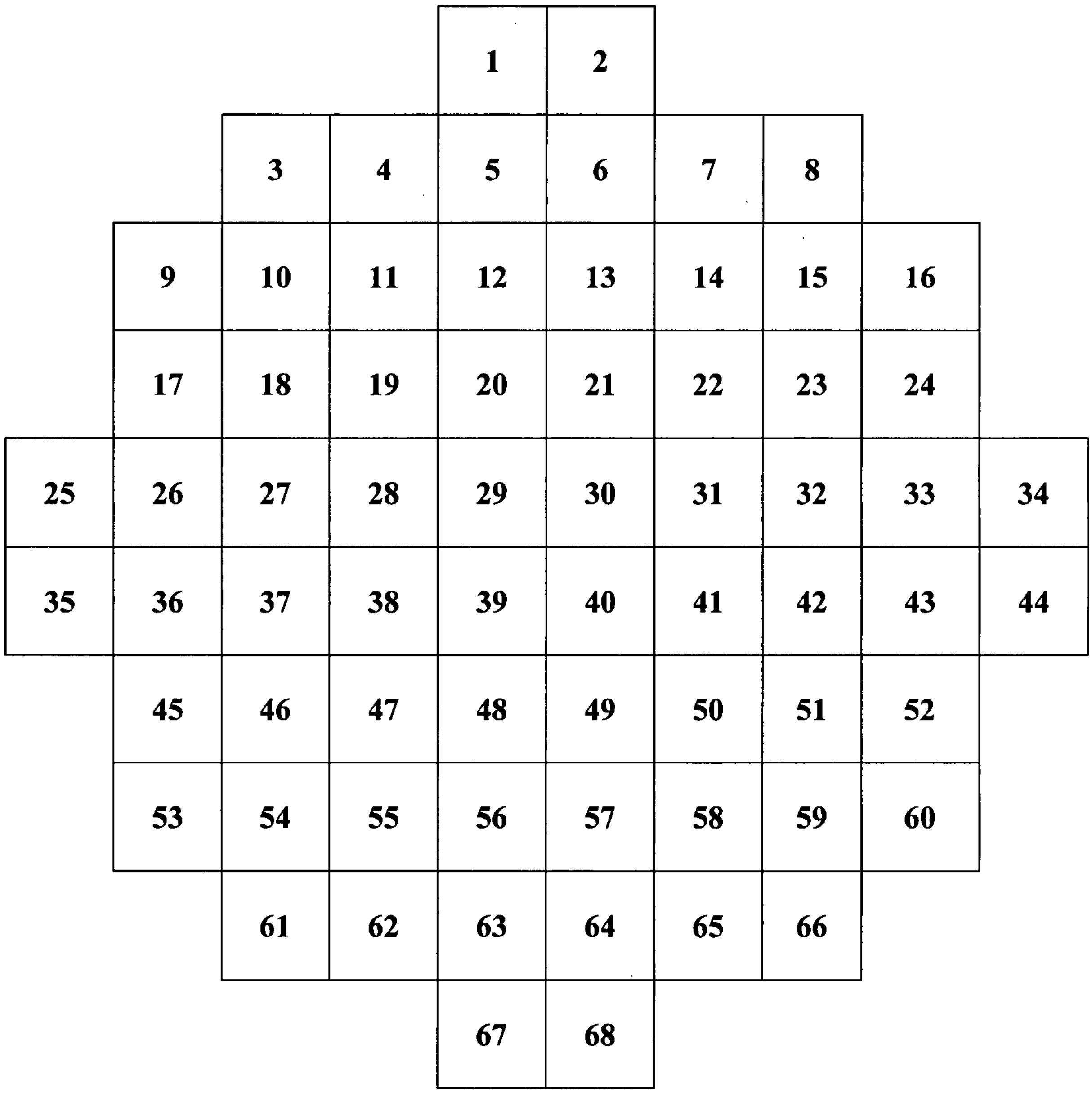


Figure 1.III.2 MPC-68M STORAGE LOCATIONS

APPENDIX 1.III.A: METAMIC-HT

[APPENDIX WITHHELD IN ITS ENTIRETY PER 10 CFR 2.390]

APPENDIX 1.III.B: METAMIC-HT[1] PROPERTIES SUPPORTING MPC-68M SHORT-TERM OPERATIONS AND ACCIDENT EVALUATIONS

The temperature range at which the mechanical and thermo physical properties of Metamic-HT have been provided in Appendix 1.III.A is exceeded under certain short-term operations and accident conditions. To facilitate the evaluation of Metamic-HT integrity, Minimum Guaranteed Values (MGV) of Metamic-HT properties are defined in the same manner as described in Appendix 1.III.A and adopted in this Appendix. Metamic-HT properties germane to structural and thermal evaluation are as follows:

- Ultimate Tensile Strength
- Yield Strength
- Area Reduction
- Young's Modulus
- Thermal Conductivity
- Emissivity
- Specific Heat

Reasonably bounding MGVs of above properties are defined in Table 1.III.B.1 up to 500°C, which comfortably bounds all temperatures.

1.III.B.1 High Temperature Tensile Testing

To characterize the mechanical properties of Metamic-HT for this higher temperature range the Ultimate, Yield, Area Reduction and Young's Modulus of Metamic-HT coupons were tested under bounding test temperatures. The testing was conducted at the Westmoreland testing lab in Youngstown, PA. The test specimens were prepared and tested in accordance with the ASTM standards adopted in the Metamic-HT sourcebook for qualification testing. A total of fifteen coupons were tested at 450°C and 500°C in the as-extruded condition and test results archived in the Metamic-HT Sourcebook [1.III.A.3]. Thermal aging and irradiation effects were not included in the testing as prior testing archived in the Metamic-HT sourcebook have discerned no significant difference due to these effects.

To characterize lower bound strength of Metamic-HT the Minimum Measured Values (MMV) of the above properties were obtained, archived in the Metamic-HT sourcebook and MGV compliance evaluated. In all cases the Metamic-HT properties meet or exceed Minimum Guaranteed Values prescribed in Table 1.III.B.1. The high temperature strength values of Metamic-HT support the following:

- Metamic-HT retains well over 50% of the operating temperature strength properties at a reasonably bounding 450°C accident temperature.

[1] This appendix is abstracted from the Metamic-HT Sourcebook [1.III.A.3].

- ❖ Under extremely high 500°C temperature reasonable values of strength properties are retained.
- ❖ The test data provides reasonable assurance of Metamic-HT integrity under thermally challenging events.

1.III.B.2 High Temperature Thermo-physical Properties

To characterize the thermal properties of Metamic-HT for this higher temperature range the conductivity, emissivity and heat capacity of Metamic-HT coupons were tested under bounding test temperatures. With the exception of emissivity property wherein prior testing adequately covered the high temperature range up to 500°C additional tests were conducted to measure the conductivity and specific heat properties. A total of two specimens were tested for each of the conductivity and specific heat properties in accordance with the ASTM standards adopted in the Metamic-HT Sourcebook [1.III.A.3]. As thermo-physical properties are principally a function of composition the properties were tested in the as-manufactured condition from the extrusion plant. The coupons were tested at 450°C and 500°C and test results archived in the Metamic-HT Sourcebook. The results are evaluated in the following.

Conductivity Measurements

To characterize the lower bound conductivity of Metamic-HT the Minimum Measured Value (MMV) were obtained and MGV compliance confirmed. To discern data trends the MMV conductivity values in the operating temperature range and high temperature range are tabulated below.

Temperature (°C)	205	370	450	500
Conductivity (W/m-°K)	188	187	193	193

The above data supports the observation that Metamic-HT thermal conductivity is essentially constant for the temperature range spanning all operating and accident temperatures. Therefore a single valued lower bound conductivity defined in the MGV tables provides a conservative characterization of Metamic-HT conductivity for evaluation under normal, off-normal and accident conditions.

Emissivity Measurements

Emissivity measurements in the high temperature range at an upper bound 500°C temperature are covered by prior testing reported in the Metamic-HT Sourcebook. The measured emissivity data supports the Table 1.III.B.1 MGV requirement that high temperature Metamic-HT emissivity meets or exceeds $\varepsilon = 0.8$.

Specific Heat Measurements

In accordance with definition of heat capacity as a reference property in Table 1.III.B.1 the mean

value of the measurements were obtained and added to Metamic-HT Sourcebook. To discern data trends the mean heat capacity values in the operating temperature range and high temperature range are tabulated below.

Temperature (°C)	350	450	500
Heat Capacity (J/kg-°K)	1024.2	1129.2	1098.2

The above data supports the observation that the heat capacity of Metamic-HT is a weak function of temperature. To provide a reasonable characterization of heat capacity in the range of 350°C to 500°C a linear function fitting the end points of the range is obtained and added to Table 1.III.B.1.

Table 1.III.B.1:
Reference & Minimum Guaranteed Values (MGVs) of Metamic-HT Mechanical and Thermal Characteristics Supporting Accident Evaluations

	Property	Temperature, °C	Design Value	Type
1.	Yield strength, σ_y (ksi)	450/500	8.5/6	MGV
2.	Tensile strength, σ_u (ksi)	450/500	9/6.5	MGV
3.	Young's Modulus, E (ksi)	450/500	4000/3500	MGV
4.	Area Reduction, A (%)	450/500	9.5/4	MGV
5.	Thermal conductivity, k (W/m°k)	450/500	180/180	MGV
6.	Emissivity (dimensionless), e	$350 \leq T \leq 500$	See Note 1	MGV
7.	Specific Heat, C_p (J/g-°C) (Note 2)	$350 \leq T \leq 500$	Note 3	Reference

Note 1: Emissivity Equation (Hard Anodized Metamic-HT)

$e = 0.2+0.6 \sin[\pi(T-100)/1304]$ $(100°F \leq T \leq 752°F)$

$e = 0.8$ $(T > 752°F)$

Note 2: These properties are reference values (not MGVs). Property variations in the small do not have significant effect on the safety evaluations in which these properties are used. Reference properties are characterized by the mean of the measured data.

Note 3: Heat Capacity Function

$Cp = 1024.2 + 0.493(T-350)$

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

† 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).

welds to the MPC lid and shell, 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.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC.

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, and a Code pressure test.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, pressure testing, and helium leak testing, (performed on the vent and drain port cover plates), 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.

<u>Thermal</u>

The design and operation of the HI-STORM 100 System meets the intent of the review guidance contained in ISG-11, Revision 3 [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 must remain below 400°C (752°F). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400°C (752°F) for high burnup fuel and 570°C (1058°F) for moderate burnup fuel.

iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570°C (1058°F).

iv. For High Burnup Fuel (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 operations is generally set at 400°C (752°F). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570°C (1058°F) because fuel cladding stress is shown to be less than approximately 90 MPa per Reference [2.0.9]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions added to ensure these limits are met (see Section 4.5).

ii. For MPCs containing at least one high burnup fuel (HBF) assembly or if the MPC heat load is greater than the threshold heat load defined in Table 4.5.1, the forced helium dehydration (FHD) method of MPC cavity drying must be used to meet the normal operations 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. For MPCs loaded with one or more high burnup fuel assemblies and the MPC heat load is greater than threshold heat load defined in Table 4.5.4, the Supplemental Cooling System (SCS) is required to ensure fuel cladding temperatures remain below the applicable temperature limit (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C.

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 limits in Table 1.2.2 during canister sealing operations to promote heat transfer and prevent cladding degradation.

The normal condition 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 as specified in Section 2.1.9.

Each MPC model, except MPC-68F, 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 up to a maximum specific heat emission rate, subject to other restrictions, such as location

requirements for damaged fuel containers (DFCs) and fuel with integral non-fuel hardware (e.g., APSR). The second is regionalized fuel loading, wherein the basket is segregated into two regions. Regionalized loading allows for storage of fuel assemblies with higher heat emission rates than would otherwise be authorized for uniform loading. Regionalized loading strategies must also comply with other requirements, such as those for DFCs and non-fuel hardware. Specific fuel assembly cooling time, burnup, and decay heat limits for regionalized loading are presented in Section 2.1.9. The two fuel loading regions are defined by fuel storage location number in Table 2.1.27 (refer to Figures 1.2.2 through 1.2.4). For MPC-68F, only uniform loading is permitted.

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 10.

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 Section 2.1.9. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation, 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 incorporated into the fuel basket assembly, and, for certain MPC models, soluble boron in the MPC water. The minimum specified boron concentration verified during neutron absorber manufacture is further reduced by 25% for criticality analysis for Boral-equipped MPCs and by 10% for METAMIC®-equipped MPCs. No credit is taken for burnup. The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC is limited. Enrichment limits and soluble boron concentration requirements are delineated in Section 2.1.9 consistent with the criticality analysis described in Chapter 6.

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, the Holtec MPC design meets the guidance in Interim Staff Guidance 18 to classify confinement boundary leakage as non-credible. Therefore, no confinement dose analysis is performed. The confinement function of the MPC is verified through pressure testing and helium leak testing on the vent and drain port cover plates, and weld examinations performed in accordance with the acceptance test program in Chapter 9.

Helium leakage testing of the MPC base metal (shell, baseplate and MPC Lid) and MPC shell to baseplate welds and shell to shell weld is performed on the unloaded MPC.

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 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 radial 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-318.1-89 (92) [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 clarified 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 analysis described in Section 3.4.10. 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 through-thickness section average temperature limit for the overpack concrete is established in accordance with Paragraph A.4.3 of Appendix A to ACI 349, which allows the use of elevated temperature limits if test data supporting the compressive strength is available and an evaluation to show no concrete deterioration provided. Appendix 1.D specifies the cement

and aggregate requirements to allow the utilization of the 300°F temperature limit. For short term conditions the through-thickness section average concrete temperature limit of 350°F is specified in accordance with Paragraph A.4.2 of Appendix A to 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 to a real individual beyond 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 Chapter 5. 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.

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 lids of the HI-TRAC 125 and HI-TRAC 125D 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 heat load analyzed for storage operations. When the MPC contains any high burnup fuel assemblies and the MPC decay heat is greater than the threshold heat load defined in Table 4.5.4, the Supplemental Cooling System (SCS) will be required for certain time periods while the MPC is inside the HI-TRAC transfer cask (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C. 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 limited in accordance with the design criteria established for the transfer cask.

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 HI-TRAC 125 or HI-TRAC 100 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 HI-TRAC 100 or 100D. 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 this FSAR, 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 regulations 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"; and 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
Normal Events D	 NF-3322.1, 3324.6
Off-Normal Events D+F	 NF-3322.1, 3324.6 with all stress limits increased by 1.33
Accident-Level Events 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.

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 for core supports (NG-1121)	10CFR72.24(c)(4)	Section 2.0.1
MPC Fuel Basket Supports (Angled Plates)	ASME Code, Section III, Subsection NG for internal structures (NG-1122)	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:	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

† 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
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(l)	Section 2.0.1
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	See Subsection 2.2.2.3	Table 2.2.3
Neutron Poison:			
Neutron Absorber (normal)	800° F	See Table 4.3.1 and Subsection 1.2.1.3.1	Table 2.2.3
Neutron Absorber (accident)	1000° F	See Table 4.3.1 and Subsection 1.2.1.3.1	Table 2.2.3
Canister Drying	≤ 3 torr for ≥ 30 minutes (VDS) ≤ 21°F exiting the demoisturizer for ≥ 30 minutes or a dew point of the MPC exit gas ≤ 22.9°F for ≥ 30 minutes(FHD)	NUREG-1536, ISG-11, Rev. 3	Section 4.5, Appendix 2.B
Canister Backfill Gas	Helium	-	Section 4.4
Canister Backfill	Varies (see Table 1.2.2)	Thermal Analysis	Section 4. 4
Fuel cladding temperature limit for long term storage conditions	752 °F (400 °C)	ISG-11, Rev. 3	Section 4.3
Fuel cladding temperature limit for normal short-term operating conditions (e.g., MPC drying and onsite transport)	752 °F (400 °C), except certain MPCs containing all moderate burnup fuel (MBF) may use 1058°F (570°C) for normal short-term operating conditions	ISG-11, Rev. 3	Sections 4.3 and 4.5
Fuel cladding temperature limit for	1058° F (570 °C)	ISG-11, Rev. 3	Sections 2.0.1 and 4.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Off-Normal and Accident Events			
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
Leak Testing:			
Welds Tested	MPC shell to shell and MPC shell to baseplate welds (Fabrication). Port covers-to-MPC lid (field)	ISG-25 ISG-18	Section 9.1
Base Metals Tested	MPC shell, MPC baseplate,	ISG-25	Section 9.1

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	MPC lid (Fabrication). MPC vent and drain port cover plates (Field).		
Medium	Helium	ANSI N14.5	Section 9.1
Max. Leak Rate	Leaktight	ANSI N14.5	Section 9.1
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
Pressure Testing:			
Minimum Test Pressure	125 psig (hydrostatic) 120 psig (pneumatic)	-	Sections 8.1 and 9.1
Welds Tested	MPC Lid-to-Shell, MPC Shell seams, MPC Shell-to-Baseplate	-	Sections 8.1 and 9.1
Medium	Water or helium	-	Section 8.1 and Chapter 9
Retrievability:			
Normal and Off-normal: Post (design basis) Accident	No Encroachment on Fuel Assemblies	10CFR72.122(f) & (l)	Sections 3.4 and 3.1.2
Criticality:		10CFR72.124 & 10CFR72.236(c)	
Method of Control	Fixed Borated Neutron Absorber, Geometry, and Soluble Boron	-	Section 2.3.4
Min. ^{10}B Loading (Boral/METAMIC®)	0.0267/0.0223 g/cm^2 (MPC-24) 0.0372/0.0310 g/cm^2 (MPC-68, MPC-68FF, MPC-24E,MPC-24EF, MPC-32 and MPC-32F) 0.01 g/cm^2 (MPC-68F)	-	Sections 2.1.8 and 6.1
Minimum Soluble Boron	Varies (see Tables 2.1.14 and 2.1.16)	Criticality Analysis	Sections 2.1.9 and 6.1
Max. k_{eff}	0.95	-	Sections 6.1 and 2.3.4
Min. Burnup	0.0 GWd/MTU (fresh fuel)	-	Section 6.1
Radiation Protection/Shielding:		10CFR72.126, & 10CFR72.128(a)(2)	

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
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 and Section 2.1.9
Type of Cladding	ZR and Stainless Steel	-	Section 2.1.9
Fuel Condition	Intact, Damaged, and Debris	-	Sections 2.1.2, 2.1.3, and 2.1.9
PWR Fuel Assemblies:			
Type/Configuration	Various	-	Section 2.1.9
Max. Burnup	68,200 MWD/MTU	-	Sections 2.1.9 and 6.2
Max. Enrichment	Varies by fuel design	-	Table 2.1.3 and Section 2.1.9
Max. Decay Heat/ MPC†:	36.9 kW	-	Section 4.4
Minimum Cooling Time:	3 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)	-	Section 2.19
Max. Fuel Assembly Weight:	1,720 lb. for fuel assemblies that	-	Section 2.1.9

† Section 2.1.9.1 describes the decay heat limits per assembly

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
(including non-fuel hardware and DFC, as applicable)	do not require fuel spacers, otherwise 1,680 lb.		
Max. Fuel Assembly Length: (Unirradiated Nominal)	176.8 in.	-	Section 2.1.9
Max. Fuel Assembly Width (Unirradiated Nominal)	8.54 in.	-	Section 2.1.9
BWR Fuel Assemblies:			
Type	Various	-	Sections 2.1.9 and 6.2
Max. Burnup	65,000 MWD/MTU	-	Section 2.1.9
Max. Enrichment	Varies by fuel design	-	Section 2.1.9, Table 2.1.4
Max. Decay Heat/ MPC†.	36.9 kW	-	Section 4.4
Minimum Cooling Time:	3 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)		Section 2.1.9
Max. Fuel Assembly Weight:			
w/channels and DFC, as applicable	730 lb.	-	Section 2.1.9
Max. Fuel Assembly Length (Unirradiated Nominal)	176.5in.	-	Section 2.1.9
Max. Fuel Assembly Width (Unirradiated Nominal)	5.85 in.	-	Section 2.1.9
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

† Section 2.1.9.1 describes 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%	NUREG-1536	Sections 2.2.1.3
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3
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	N/A	ISG-18	Sections 2.2.2.4 and 7.1
Partial Blockage of Overpack Air Inlets	50% of Air Inlets Blocked	-	Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10%	NUREG-1536	Sections 2.2.2.1
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1
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
Fill Gases	100%	NUREG-1536	Sections 2.2.3.8

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Fission Gases	30%	NUREG-1536	Sections 2.2.3.8
Particulates & Volatiles	See Table 7.3.1	-	Sections 2.2.3.9
Confinement Boundary Leakage	None	ISG-18 / ANSI N 14.5	Sections 2.2.3.9 and 7.1
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 clarified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Fabrication	ACI 349 as clarified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Compressive Strength	ACI 318.1-89 (92)as clarified 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)	90,000 lb. (MPC- 32)	R.G. 3.61	Table 3.2.1
Max. Empty Overpack Assembled with Top Lid (150 pcf concrete/200pcf concrete)	270,000/320,000 lb.	R.G. 3.61	Table 3.2.1
Max. MPC/Overpack (150 pcf concrete/200pcf concrete)	360,000/410,000 lb.	R.G. 3.61	Table 3.2.1
Design Cavity Pressures	N/A	-	Section 2.2.1.3

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

Type	Criteria	Basis	FSAR Reference
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
Thermal:			
Maximum Design Temperatures:			
Concrete			
Through-Thickness Section Average (Normal)	300° F	ACI 349, Appendix A (Paragraph A.4.3)	Section 2.0.2, and Tables 1.D.1 and 2.2.3
Through-Thickness Section Average (Off-Normal and Accident)	350° F	ACI 349 Appendix A (Paragraph A.4.2)	Section 2.0.2, and Tables 1.D.1 and 2.2.3
Steel Structure (other than lid bottom and top plates) Lid Bottom and Top Plates	350° F 450°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	10CFR72.122(f) & (l)	Section 3.4
Accident			Section 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:		10CFR72.126 & 10CFR72.128(a)(2)	
Overpack (Normal/Off-Normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10

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

Type	Criteria	Basis	FSAR Reference
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) ≤ 50 rem. 15 rem lens dose. 50 rem shallow dose to skin or extremity.	10CFR72.106	Sections 5.1.2, 7.3, and 10.1
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1.9
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†:		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	250,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.4 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:			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	ASME Code	ASME Code	Section 3.4.3

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

Type	Criteria	Basis	FSAR Reference
Acceptance Criteria	Level A		
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
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	50% of Air Inlets 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

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

Type	Criteria	Basis	FSAR Reference
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. acceleration 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
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) 102,000 lb. (HI-TRAC 100D) 146,000 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 100D and 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):	250,000 lb. (HI-TRAC 125 and 125D) 200,000 lb. (HI-TRAC 100 and 100D)	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

† 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
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Section 2.0.3
	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.) (long term) 350°F (max.) (short term)	Test Data	Appendix 1.B and 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	10CFR72.122(f) & (l)	Section 3.4
After Design-basis (Postulated) Accident			Section 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:		10CFR72.126 & 10CFR72.128(a)(2)	
Transfer Cask (Normal/Off-Normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10

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

Type	Criteria	Basis	FSAR Reference
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature:	80 ° 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
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

Item	Free-Standing	Anchored	Comments
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	$G_H + \mu G_V \leq \mu$ (see note 1 below)	$G_H \leq 2.12$ AND $G_V \leq 1.5$	
6. Required minimum value of cask to pad static coefficient of friction (μ, must be confirmed by testing if a value greater than 0.53 is used).	Greater than or equal to 0.53 (per Table 2.2.9).	Not applicable	
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.	

Note 1 – GH and GV may be the coincident values of the instantaneous horizontal and vertical accelerations, and the inequality shall be evaluated at each time step.

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 lengths of the multi-purpose canisters have been set at dimensions which permit 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 lengths.

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. Section 2.1.9 summarizes the authorized contents for the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 and meets the other limits specified in Section 2.1.9 is acceptable for storage in the HI-STORM 100 System. Tables 2.1.3 and 2.1.4 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 thermal. 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 ZR or stainless steel (SS) clad fuel assemblies with the characteristics listed in Tables 2.1.17 through 2.1.24.

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 Section 2.1.9 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 within the decay heat, burnup, and cooling time limits specified in Section 2.1.9 for intact fuel assemblies.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in Table 1.0.1.

Damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with mesh screens having between 40x40 and 250x250 openings per inch, for storage in the HI-STORM 100 System (see Figures 2.1.1 and 2.1.2B, C, and D). The MPC-24, MPC-24EF, MPC-32 and MPC-32F are designed to accommodate PWR damaged fuel and fuel debris. The MPC-68, 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 and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in Section 2.1.9. 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. 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 to qualify as a secondary containment, which used to be required under hypothetical accident conditions of transportation under 10 CFR 71. 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 specified in Section 2.1.9. 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 to 2-1/2 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 Section 2.1.9.

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 and the decay heat removal capabilities of the HI-STORM 100 System. No attempt is made to link the maximum allowable decay heat per fuel assembly with burnup, enrichment, or cooling time. Rather, the decay heat per fuel assembly is adjusted to yield peak fuel cladding temperatures with an allowance for margin to the temperature limit.

To ensure the permissible fuel cladding temperature limits are not exceeded, Section 2.1.9 specifies the allowable decay heat per assembly for each MPC model. For both uniform and regionalized loading of moderate and high burnup fuel assemblies, the allowable decay heat per assembly is presented in Section 2.1.9.

Section 2.1.9 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 the heat transfer characteristics of the fuel assemblies. The bounding fuel assembly design for thermal calculations for each fuel type is

provided in Table 2.1.5.

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.

Except for MPC-68F, 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, such as those for 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 some storage locations with lower heat emitting fuel assemblies in the remaining 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 is provided in Table 2.1.27. Regionalized fuel loading is not permitted in MPC-68F.

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. Section 2.1.9 provides the procedure for determining 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 Section 2.1.9.

Non-fuel hardware, as defined in Table 1.0.1, has been evaluated and is authorized for storage in the PWR MPCs as specified in Section 2.1.9.

2.1.8 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68, MPC-68F, MPC-68FF, 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 the criticality analysis is conservatively established below the minimum values shown in Table 2.1.15. For Boral, the value used in the analysis is 75% of the minimum value, while for METAMIC, it is 90% of the minimum value. This is consistent with NUREG-1536 [2.1.5] which suggests a 25% reduction in ^{10}B areal density credit when subject to standard acceptance tests, and which allows a smaller reduction when more comprehensive tests of the areal density are performed.

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 provide the required soluble boron concentrations for these MPCs.

2.1.9 Summary of Authorized Contents

Tables 2.1.3, 2.1.4, 2.1.12, and 2.1.17 through 2.1.29 together specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM 100 System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR. Fuel classified as damaged fuel assemblies or fuel debris must be stored in damaged fuel containers for storage in the HI-STORM 100 System.

Tables 2.1.17 through 2.1.24 are the baseline tables that specify the fuel assembly limits for each of the MPC models, with appropriate references to the other tables in this section for certain other limits. Tables 2.1.17 through 2.1.24 refer to Section 2.1.9.1 for ZR-clad fuel limits on minimum cooling time, maximum decay heat, and maximum burnup for uniform and regionalized fuel loading.

2.1.9.1 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel

Each ZR-clad fuel assembly and any PWR integral non-fuel hardware (NFH) to be stored in the

HI-STORM 100 System must meet the following limits, in addition to meeting the physical limits specified elsewhere in this section, to be authorized for storage in the HI-STORM 100 System. The contents of each fuel storage location (fuel assembly and NFH) to be stored must be verified to have, as applicable:

- A decay heat less than or equal to the maximum allowable value.
- An assembly average enrichment greater than or equal to the minimum value used in determining the maximum allowable burnup.
- A burnup less than or equal to the maximum allowable value.
- A cooling time greater than or equal to the minimum allowable value.

The maximum allowable ZR-clad fuel storage location decay heat values are determined using the methodology described in Section 2.1.9.1.1 or 2.1.9.1.2 depending on whether uniform fuel loading or regionalized fuel loading is being implemented†. The total permissible MPC heat load, for both uniform and regionalized loading, is determined in the following two subsections 2.1.9.1.1 and 2.1.9.1.2. The decay heat limits are independent of burnup, cooling time, or enrichment and are based strictly on the thermal analysis described in Chapter 4. Decay heat limits must be met for all contents in a fuel storage location (i.e., fuel and PWR non-fuel hardware, as applicable).

The maximum allowable average burnup per fuel storage location is determined by calculation as a function of minimum enrichment, maximum allowable decay heat, and minimum cooling time from 3 to 20 years, as described in Section 2.1.9.1.3.

Section 12.2 describes how compliance with these limits may be verified, including practical examples.

2.1.9.1.1 Uniform Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.1.26 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model in aboveground storage*. Even if the limits in Table 2.1.26 are met, the user must follow the instructions in the next section to calculate Q_{CoC} to determine if certain operational steps are required per the CoC. If the user needs to load fuel assemblies with a decay heat higher than the limits in Table 2.1.26, a regionalized loading pattern discussed in the next section may be considered.

† Note that the stainless steel-clad fuel decay heat limits apply to all fuel in the MPC, if a mixture of stainless steel and ZR-clad fuel is stored in the same MPC. The stainless steel-clad fuel assembly decay heat limits may be found in Table 2.1.17 through 2.1.24

* Maximum allowable heat loads in 100U underground storage are defined in Supplement 2.I; however the discussion in Section 2.1.9.1 also applies to the 100U.

2.1.9.1.2 <u>Design Heat Load for ZR-Clad Fuel</u>

The discussion in this section provides the approach to determine the maximum permitted per cell heat load for long term-storage in a regionalized pattern. In addition, this section also provides the approach to determine the allowed per cell heat load for those operations that are dependent on the total MPC heat load. These include helium backfill pressure, supplemental cooling, drying method, and time requirements for clearing blockage on HI-STORM inlet vents.

The Design Basis heat load for the aboveground HI-STORM System, Q_d, is 34 kW. Q_d is based on the assumption that every storage cell in the MPC is generating an equal amount of heat. In other words, the specific heat generation rate, q, of each storage location is considered equal. Thus, in an MPC with n storage locations,

$$Q_d = n\, q \qquad \text{Equation a}$$

In reality, however, the population of SNF and associated NFH loaded in the MPC invariably has unequal decay heat. If we consider the loaded decay heat in a cell as r, and r_i denotes the loaded decay heat in location i, then the aggregate MPC heat load, Q_t, is given by a simple summation, i.e.,

$$Q_t = \sum_{i}^{n} r_i \qquad \text{Equation b}$$

For purposes of the CoC compliance for operations where the total MPC heat load needs to be calculated, the total MPC heat load is,

$$Q_{CoC} = r_{max}\, n \qquad \text{Equation c}$$

where r_{max} is the largest value of r_i in the population of SNF loaded in the MPC, i.e.,

$$r_{max} = \max \text{ of } [r_i,\ i = 1,2\ldots n] \qquad \text{Equation d}$$

In all cases, the aggregate MPC heat load, Q_t, is less than or equal to Q_{CoC} or Q_d. In some cases this difference can be quite large. This scenario can be illustrated by considering the example of a MPC-32 that has 31 cells containing 0.5kW and one cell containing 1 kW. The aggregate MPC heat load is (31)(0.5) + 1 = 16.5kW. However, because r_{max} = 1 kW, Q_{CoC} = (32)(1) = 32kW. Thus, $Q_{CoC} >> Q_t$. This condition prevails in all loaded MPCs to a varying degree.

Even though Q_t may be significantly less than Q_{CoC}, Q_{CoC} must be used to establish certain operational procedures and the pre-analyzed conditions. The aggregate total heat load Q_t may be used for time to boil calculations, when determining if the air mass flow rate test on a loaded system, per Condition 9 of the CoC, needs to be performed and when considering if time limits need to be applied to vacuum drying. It should be noted that equation c is used to determine Q_{CoC} when following the heat load limits for uniform loading (Table 2.1.26).

To utilize more of the design basis heat load and allow for more loading flexibility, the MPC is divided into two regions. This is referred to as “regionalized loading”. The inner region (Region 1) and the outer region (Region 2) have maximum permitted heat load of q_1 and q_2, respectively. The maximum permitted values of q_1 and q_2 are related through the ratio X, where $X = q_1/q_2$. The ratio X is NOT the ratio of the maximum values of the as-loaded storage locations in Region 1 and Region 2.The case where q_1 and q_2 are equal (X= 1) is referred to as “uniform storage”. Q_{CoC} for regionalized loading is computed by:

$$Q_{CoC} = n_1 q_1 + n_2 q_2 \qquad \text{Equation e}$$

where n_1 and n_2 are the number of cells in Regions 1 and 2, respectively.

A functional relationship between Q and X was determined by performing an iterative thermal analysis. This led to the functional relationship Q(X):

$$Q(X) = \frac{2Q_d}{1+X^y} \qquad \text{Equation f}$$

where X is a value greater than or equal to 0.5 and less than or equal to 3 and where y is also a function of X as defined below:

$$y(X) = \frac{0.23}{X^{0.1}} \qquad \text{Equation g}$$

Table 2.1.30 is provided to give a list of permissible q_1 and q_2 for discrete values of X for all MPC types. The table was determined using the following approach:

(i) Choose a value of X in the permissible range ($0.5 \leq X \leq 3$)

(ii) Calculate q_2 using the following equation:

$$q_2 = \frac{2 \times Q_d}{(1+X^y) \times (n_1 \times X + n_2)} \qquad \text{Equation h}$$

where:

$y = 0.23/X^{0.1}$
q_2 = Maximum allowable decay heat per fuel storage location in Region 2 (kW)
Q_d= Maximum uniform storage MPC decay heat (34 kW)
X = Ratio of q_1 to q_2 chosen in Step (i)
n_1 = Number of fuel storage locations in Region 1 from Table 2.1.27
n_2 = Number of fuel storage locations in Region 2 from Table 2.1.27

(iii) Calculate q_1 using the following equation:

$$q_1 = X \times q_2 \qquad \text{Equation i}$$

Using the steps provided above we find for X=2 that q_1 = 1.43 kW and q_2 = 0.715 kW for MPC-32. The user can follow Table 2.1.30 for discrete values of X to determine q_1 and q_2 or calculate q_1 and q_2 for a specific value of X using the steps above. It should be noted that equation e is used to determine Q_{CoC} when following the heat load limits for regionalized loading.

It should be emphasized that the variable two-region scheme of storage does not introduce any new complication in the dry storage implementation. As compared to uniform loading in MPC-32, where q = 1.0625 kW for all cells, the regionalized loading gives the user the flexibility to load the MPC with more varying heat loads. It is noted that for X < 1 Q_{CoC} is greater than Q_d, for X = 1 Q_{CoC} equals Q_d, and for X >1 Q_{CoC} is less than Q_d. For ALARA and regardless of which loading pattern is used, a plant should always seek to preferentially locate the fuel with the higher heat loads toward the center of the MPC. If the need arises to place younger fuel into dry storage a regionalized pattern with X <1 may be more appropriate.

2.1.9.1.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel

The maximum allowable ZR-clad fuel assembly average burnup varies with the following parameters, based on the shielding analysis in Chapter 5:

- Minimum required fuel assembly cooling time
- Maximum allowable fuel assembly decay heat
- Minimum fuel assembly average enrichment

The calculation described in this section is used to determine the maximum allowable fuel assembly burnup for minimum cooling times between 3 and 20 years, using maximum decay heat and minimum enrichment as input values. This calculation may be used to create multiple burnup versus cooling time tables for a particular fuel assembly array/class and different minimum enrichments. The allowable maximum burnup for a specific fuel assembly may be calculated based on the assembly's particular enrichment and cooling time.

(i) Choose a fuel assembly minimum enrichment, E_{235}.

(ii) Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below:

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$$

Equation j

Where:

Bu = Maximum allowable assembly average burnup (MWD/MTU)

q = Maximum allowable decay heat per fuel storage location determined in Section

2.1.9.1.1 or 2.1.9.1.2 (kW)

E_{235} = Minimum fuel assembly average enrichment (wt. % ^{235}U)
(e.g., for 4.05 wt. %, use 4.05)

A through G = Coefficients from Tables 2.1.28 or 2.1.29 for the applicable fuel assembly array/class and minimum cooling time.

2.1.9.1.4 Other Considerations

In computing the allowable maximum fuel storage location decay heats and fuel assembly average burnups, the following requirements apply:

- Calculated burnup limits shall be rounded down to the nearest integer

- Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR fuel must be reduced to be equal to these values.

- Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a minimum cooling time of 4.5 years may be interpolated between those burnups calculated for 4 and 5 years.

- ZR-clad fuel assemblies must have a minimum enrichment, as defined in Table 1.0.1, greater than or equal to the value used in determining the maximum allowable burnup per Section 2.1.9.1.3 to be authorized for storage in the MPC.

- When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any PWR non-fuel hardware, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

Section 12.2.10 provides a practical example of determining fuel storage location decay heat, burnup, and cooling time limits and verifying compliance for a set of example fuel assemblies.

2.1.9.1.5 Supplemental Cooling Threshold Heat Loads

Fuel loading operations involving the handling of High Burnup Fuel (HBF) in a dewatered MPC emplaced in a HI-TRAC transfer cask require additional cooling under certain thermal loads to address reduced heat dissipation relative to the normal storage condition. To address this requirement the Supplemental Cooling System (SCS) defined in Appendix 2.C is mandated under threshold heat loads defined in Section 4.5 and Table 2.1.30. The specific design of a SCS must accord with site-specific needs and resources, including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the SCS are satisfied by plant-specific designs are set forth in Appendix 2.C.

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™
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 (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	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 365	≤ 412	≤ 438	≤ 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 Note 5) (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.) (Note 8)	≤ 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)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 473	≤ 473	≤ 473	≤ 495	≤ 495	≤ 495
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit – see Note 5) (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.) (Note 8)	≤ 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**	**15x15I**	**16x16 A**
Clad Material (Note 2)	SS	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 495	≤ 495	≤ 448
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)	N/A (Note 9)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit – see Note 5) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0 (Note 9)	≤ 5.0
No. of Fuel Rod Locations	204	208	216	236
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.413	≥ 0.382
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.367	≤ 0.3350
Fuel Pellet Dia. (in.) (Note 8)	≤ 0.3825	≤ 0.3622	≤ 0.360	≤ 0.3255
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.550	≤ 0.506
Active Fuel length (in.)	≤ 144	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	9 (Note 10)	5 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0140	≥ 0.0350

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

Fuel Assembly Array and Class	**17x17A**	**17x17 B**	**17x17 C**
Clad Material (Note 2)	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 433	≤ 474	≤ 480
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	≤ 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, 24EF, 32 or 32F with soluble boron credit – see Note 5) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	264	264	264
Fuel Clad O.D. (in.)	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.) (Note 8)	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 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. 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 to account for manufacturer's tolerances.

4. Each guide tube replaces four fuel rods.

5. Soluble boron concentration per Tables 2.1.14 and 2.1.16, as applicable.

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.% ^{235}U.

8. Annular fuel pellets are allowed in the top and bottom 12" of the active fuel length.

9. This fuel assembly array/class can only be loaded in MPC-32.

10. One Instrument Tube and eight Guide Bars (Solid Zr).

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)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 198	≤ 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)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 190	≤ 190	≤ 190	≤ 191	≤ 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	≤ 0.055	≤ 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)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 180	≤ 182	≤ 182	≤ 183	≤ 183	≤ 164
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)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 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."

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. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2)

5. This assembly class contains 74 total 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	BWR	PWR
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Array/Class 10x10A)	B&W 15x15 (Array/Class 15x15F)
Shielding	GE 7x7	B&W 15x15
Thermal-Hydraulic	GE-12/14 10x10	W 17x17 OFA

Tables 2.1.6 through 2.1.8

INTENTIONALLY DELETED

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 V5H	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 to 2-1/2 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 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 to 2-1/2 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 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
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	$\leq$ 18
Decay Heat Per Thoria Canister	$\leq$ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time $\geq$ 18 years and average burnup $\leq$ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	$\leq$ 27 kg/canister
Fuel Cladding O.D.	$\geq$ 0.412 inches
Fuel Cladding I.D.	$\leq$ 0.362 inches
Fuel Pellet O.D.	$\leq$ 0.358 inches
Active Fuel Length	$\leq$ 111 inches
Canister Weight	$\leq$ 550 lbs., including Thoria Rods
Canister Material	Type 304 SS

Table 2.1.13

[INTENTIONALLY DELETED]

Table 2.1.14

Soluble Boron Requirements for 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	All fuel assemblies classified as intact fuel assemblies and 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. %	≥ 300
MPC-24E/24EF	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.%	≥ 600

[1]Refer to Table 2.1.3 for these enrichments.

Table 2.1.15

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

MPC MODEL	MINIMUM ^{10}B LOADING (g/cm^2)	
	Boral Neutron Absorber Panels	METAMIC Neutron Absorber Panels
MPC-24	0.0267	0.0223
MPC-24E and MPC-24EF	0.0372	0.0310
MPC-32/32F	0.0372	0.0310
MPC-68 and MPC-68FF	0.0372	0.0310
MPC-68F	0.01	N/A (Note 1)

Notes:

1. All MPC-68F canisters are equipped with Boral neutron absorber panels.

Table 2.1.16

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

Fuel Assembly Array/Class	**All Intact Fuel Assemblies**		**One or More Damaged Fuel Assemblies or Fuel Debris**	
	Max. Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G/I	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A	1,400	2,000	1,500	2,300
17x17A/B/C	1,900	2,600	2,100	2,900

Note:

1. For maximum initial enrichments between 4.1 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.1 wt% and 5.1 wt% ^{235}U.

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 8 years and $\leq$ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq$ 710 Watts
Non-Fuel Hardware Burnup and Cooling Time	As specified in Table 2.1.25
Fuel Assembly Length	$\leq$ 176.8 in. (nominal design)
Fuel Assembly Width	$\leq$ 8.54 in. (nominal design)
Fuel Assembly Weight	$\leq$ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise $\leq$ 1,680 lbs (including non-fuel hardware)
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies. ▪ Damaged fuel assemblies and fuel debris are not permitted for loading in MPC-24. ▪ One NSA is authorized to be loaded with a fuel assembly in fuel storage location 9, 10, 15, or 16. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location. ▪ APSRs may be loaded with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16 ▪ CRAs, RCCAs and/or CEAs may be stored with fuel assemblies in fuel cell locations 4, 5, 8 through 11, 14 through 17, 20, and/or 21. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Table 2.1.18

[INTENTIONALLY DELETED]

Table 2.1.19

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	VALUE (Notes 1 and 2)			
Fuel Type(s)	Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels	Uranium oxide, BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)	Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels	Mixed Oxide (MOX) BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)
Cladding Type	ZR	ZR	ZR	ZR
Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable array/class	As specified in Table 2.1.4 for the applicable array/class	As specified in Table 2.1.4 for array/class 6x6B	As specified in Table 2.1.4 for array/class 6x6B
Post-irradiation Cooling Time, and Average Burnup per Assembly	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.
Decay Heat Per Fuel Storage Location	≤ 115 Watts	≤ 115 Watts	≤ 115 Watts	≤ 115 Watts
Fuel Assembly Length	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)
Fuel Assembly Width	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)
Fuel Assembly Weight	≤ 400 lbs, (including channels)	≤ 550 lbs, (including channels and DFC)	≤ 400 lbs, (including channels)	≤ 550 lbs, (including channels and DFC)

Table 2.1.19 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to four (4) DFCs containing Dresden Unit 1 or Humboldt Bay uranium oxide or MOX fuel debris. The remaining fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable: - uranium oxide BWR intact fuel assemblies - MOX BWR intact fuel assemblies - uranium oxide BWR damaged fuel assemblies in DFCs - MOX BWR damaged fuel assemblies in DFCs - up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12. ▪ Stainless steel channels are not permitted. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

2. Only fuel from the Dresden Unit 1 and Humboldt Bay plants are permitted for storage in the MPC-68F.

Table 2.1.20

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E AND MPC-24EF

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class	Uranium oxide PWR damaged fuel assemblies and/or fuel debris meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time, and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including DFC and non-fuel hardware)

Table 2.1.20 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E AND MPC-24EF

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies. ▪ One NSA is permitted for loading with a fuel assembly in fuel storage location 9, 10, 15, or 16. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location. ▪ APSRs may be loaded with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16. ▪ CRAs, RCCAs and/or CEAs may be stored with fuel assemblies in fuel cell locations 4, 5, 8 through 11, 14 through 17, 20, and/or 21. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

Table 2.1.21

[INTENTIONALLY DELETED]

Table 2.1.22

LIMITS FOR MATERIAL TO BE STORED IN MPC-68 AND MPC-68FF

PARAMETER	**VALUE (Note 1)**	
Fuel Type	Uranium oxide or MOX BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels.	Uranium oxide or MOX BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, in DFCs.
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class
Maximum Initial Planar Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable fuel assembly array/class	Planar Average: ≤ 2.7 wt% ^{235}U for array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A; ≤ 4.0 wt% ^{235}U for all other array/classes Rod: As specified in Table 2.1.4
Post-irradiation cooling time and average burnup per Assembly	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4.
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts
Fuel Assembly Length	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design) All Other array/classes: ≤ 176.5 in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design) All Other array/classes: ≤ 176.5 in. (nominal design)

Table 2.1.22 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68 AND MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Assembly Width	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design) All Other array/classes: ≤ 5.85 in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design) All Other array/classes: ≤ 5.85 in. (nominal design)
Fuel Assembly Weight	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 400 lbs. (including channels) All Other array/classes: ≤ 730 lbs. (including channels)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC) All Other array/classes: ≤ 730 lbs. (including channels and DFC)
Other Limitations	▪ For assembly/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A, up to 68 intact fuel assemblies or damaged fuel assemblies in DFCs may be stored. Fuel debris in DFCs may be stored in up to 8 locations. A Dresden Unit 1 Thoria Rod Container may be stored in one location. ▪ For all other array/classes, up to 16 DFCs containing damaged fuel assemblies and/or up to eight (8) DFCs containing fuel assemblies classified as fuel debris may be stored. DFCs shall be located only in fuel cell locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68, with the balance comprised of intact fuel assemblies meeting the above specifications, up to a total of 68. ▪ SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.	

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.
2. Array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU or MWD/MTIHM, and a decay heat ≤ 115 Watts.
3. Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a decay heat ≤ 183.5 Watts.
4. SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.

Table 2.1.23

[INTENTIONALLY DELETED]

Table 2.1.24

LIMITS FOR MATERIAL TO BE STORED IN MPC-32 AND MPC-32F

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class	Uranium oxide, PWR damaged fuel assemblies and fuel debris in DFCs meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3	As specified in Table 2.1.3
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including DFC and non-fuel hardware)

Table 2.1.24 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32 AND MPC-32F

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32. ▪ One NSA is permitted for loading with a fuel assembly in fuel storage location 13, 14, 19, or 20. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, or APSRs may only be loaded with fuel assemblies in fuel cell locations 7, 8, 12-15, 18-21, 25 and/or 26. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

Table 2.1.25

NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, 3, and 8)

Post-irradiation Cooling Time (yrs)	Inserts (Note 4) Maximum Burnup (MWD/MTU)	NSA or Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)	Control Component (Note 6) Maximum Burnup (MWD/MTU)	APSR Maximum Burnup (MWD/MTU)
≥ 3	≤ 24,635	N/A (Note 7)	N/A	N/A
≥ 4	≤ 30,000	≤ 20,000	N/A	N/A
≥ 5	≤ 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ 60,000	≤ 45,000	-	≤ 83,000
≥ 9	-	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 90,000	-	-
≥ 13	-	≤ 180,000	-	-
≥ 14	-	≤ 630,000	-	-

NOTES:

1. Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.
2. Linear interpolation between points is permitted, except that NSA or Guide Tube Hardware and APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 14 years and ≥ 11 years, respectively.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts.
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
7. N/A means not authorized for loading at this cooling time.
8. Non-fuel hardware burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post-irradiation.

Table 2.1.26
DESIGN HEAT EMISSION RATES
(UNIFORM LOADING, ZR-CLAD, ABOVEGROUND STORAGE[1])

MPC	Decay Heat (kW)	
	Per Intact Fuel Assembly[2]	Per MPC
MPC-24/24E/24EF	1.416	34
MPC-32/32F	1.062	34
MPC-68/68FF	0.5	34
	Per Damaged Fuel Assembly or Fuel Debris	Per MPC with Damaged Fuel Assembly or Fuel Debris
MPC-24E/24EF	≤ 1.114	≤ 26.7
MPC-32/32F	≤ 0.718	≤ 23
MPC-68/68FF	≤ 0.393	≤ 26.7

[1] Maximum allowable heat loads in 100U underground storage are defined in Supplement 2.I
[2] This limit applies to each storage cell and should include decay heat from any NFH

Table 2.1.27

MPC FUEL STORAGE REGIONS

MPC	Number of Storage Cells		Storage Cell IDs**	
	Inner Region (n_1)	Outer Region (n_2)	Inner Region	Outer Region
MPC-24/24E/24EF	12	12	4, 5 8 through 11 14 through 17 20 and 21	All other locations
MPC-32/32F	12	20	7, 8, 12 through 15, 18 through 21, 25 and 26	All other locations
MPC-68/68FF	32	36	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All other locations

** See Figures 1.2.2 through 1.2.4 for storage cell numbering

Table 2.1.28

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
≥ 3	19311.5	275.367	-59.0252	-139.41	2851.12	-451.845	-615.413
≥ 4	33865.9	-5473.03	851.121	-132.739	3408.58	-656.479	-609.523
≥ 5	46686.2	-13226.9	2588.39	-150.149	3871.87	-806.533	-90.2065
≥ 6	56328.9	-20443.2	4547.38	-176.815	4299.19	-927.358	603.192
≥ 7	64136	-27137.5	6628.18	-200.933	4669.22	-1018.94	797.162
≥ 8	71744.1	-34290.3	9036.9	-214.249	4886.95	-1037.59	508.703
≥ 9	77262	-39724.2	11061	-228.2	5141.35	-1102.05	338.294
≥ 10	82939.8	-45575.6	13320.2	-233.691	5266.25	-1095.94	-73.3159
≥ 11	86541	-49289.6	14921.7	-242.092	5444.54	-1141.6	-83.0603
≥ 12	91383	-54456.7	17107	-242.881	5528.7	-1149.2	-547.579
≥ 13	95877.6	-59404.7	19268	-240.36	5524.35	-1094.72	-933.64
≥ 14	97648.3	-61091.6	20261.7	-244.234	5654.56	-1151.47	-749.836
≥ 15	102533	-66651.5	22799.7	-240.858	5647.05	-1120.32	-1293.34
≥ 16	106216	-70753.8	24830.1	-237.04	5647.63	-1099.12	-1583.89
≥ 17	109863	-75005	27038	-234.299	5652.45	-1080.98	-1862.07
≥ 18	111460	-76482.3	28076.5	-234.426	5703.52	-1104.39	-1695.77
≥ 19	114916	-80339.6	30126.5	-229.73	5663.21	-1065.48	-1941.83
≥ 20	119592	-86161.5	33258.2	-227.256	5700.49	-1100.21	-2474.01

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14B						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	18036.1	63.7639	-24.7251	-130.732	2449.87	-347.748	-858.192
≥ 4	30303.4	-4304.2	598.79	-118.757	2853.18	-486.453	-459.902
≥ 5	40779.6	-9922.93	1722.83	-138.174	3255.69	-608.267	245.251
≥ 6	48806.7	-15248.9	3021.47	-158.69	3570.24	-689.876	833.917
≥ 7	55070.5	-19934.6	4325.62	-179.964	3870.33	-765.849	1203.89
≥ 8	60619.6	-24346	5649.29	-189.701	4042.23	-795.324	1158.12
≥ 9	64605.7	-27677.1	6778.12	-205.459	4292.35	-877.966	1169.88
≥ 10	69083.8	-31509.4	8072.42	-206.157	4358.01	-875.041	856.449
≥ 11	72663.2	-34663.9	9228.96	-209.199	4442.68	-889.512	671.567
≥ 12	74808.9	-36367	9948.88	-214.344	4571.29	-942.418	765.261
≥ 13	78340.3	-39541.1	11173.8	-212.8	4615.06	-957.833	410.807
≥ 14	81274.8	-42172.3	12259.9	-209.758	4626.13	-958.016	190.59
≥ 15	83961.4	-44624.5	13329.1	-207.697	4632.16	-952.876	20.8575
≥ 16	84968.5	-44982.1	13615.8	-207.171	4683.41	-992.162	247.54
≥ 17	87721.6	-47543.1	14781.4	-203.373	4674.3	-988.577	37.9689
≥ 18	90562.9	-50100.4	15940.4	-198.649	4651.64	-982.459	-247.421
≥ 19	93011.6	-52316.6	17049.9	-194.964	4644.76	-994.63	-413.021
≥ 20	95567.8	-54566.6	18124	-190.22	4593.92	-963.412	-551.983

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14C						
	A	B	C	D	E	F	G
≥ 3	18263.7	174.161	-57.6694	-138.112	2539.74	-369.764	-1372.33
≥ 4	30514.5	-4291.52	562.37	-124.944	2869.17	-481.139	-889.883
≥ 5	41338	-10325.7	1752.96	-141.247	3146.48	-535.709	-248.078
≥ 6	48969.7	-15421.3	2966.33	-163.574	3429.74	-587.225	429.331
≥ 7	55384.6	-20228.9	4261.47	-180.846	3654.55	-617.255	599.251
≥ 8	60240.2	-24093.2	5418.86	-199.974	3893.72	-663.995	693.934
≥ 9	64729	-27745.7	6545.45	-205.385	3986.06	-650.124	512.528
≥ 10	68413.7	-30942.2	7651.29	-216.408	4174.71	-702.931	380.431
≥ 11	71870.6	-33906.7	8692.81	-218.813	4248.28	-704.458	160.645
≥ 12	74918.4	-36522	9660.01	-218.248	4283.68	-696.498	-29.0682
≥ 13	77348.3	-38613.7	10501.8	-220.644	4348.23	-702.266	-118.646
≥ 14	79817.1	-40661.8	11331.2	-218.711	4382.32	-710.578	-236.123
≥ 15	82354.2	-42858.3	12257.3	-215.835	4405.89	-718.805	-431.051
≥ 16	84787.2	-44994.5	13185.9	-213.386	4410.99	-711.437	-572.104
≥ 17	87084.6	-46866.1	14004.8	-206.788	4360.3	-679.542	-724.721
≥ 18	88083.1	-47387.1	14393.4	-208.681	4420.85	-709.311	-534.454
≥ 19	90783.6	-49760.6	15462.7	-203.649	4403.3	-705.741	-773.066
≥ 20	93212	-51753.3	16401.5	-197.232	4361.65	-692.925	-964.628

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15A/B/C						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	15037.3	108.689	-18.8378	-127.422	2050.02	-242.828	-580.66
≥ 4	25506.6	-2994.03	356.834	-116.45	2430.25	-350.901	-356.378
≥ 5	34788.8	-7173.07	1065.9	-124.785	2712.23	-424.681	267.705
≥ 6	41948.6	-11225.3	1912.12	-145.727	3003.29	-489.538	852.112
≥ 7	47524.9	-14770.9	2755.16	-165.889	3253.9	-542.7	1146.96
≥ 8	52596.9	-18348.8	3699.72	-177.17	3415.69	-567.012	1021.41
≥ 9	56055.4	-20837.1	4430.93	-192.168	3625.93	-623.325	1058.61
≥ 10	59611.3	-23402.1	5179.52	-195.105	3699.18	-626.448	868.517
≥ 11	62765.3	-25766.5	5924.71	-195.57	3749.91	-627.139	667.124
≥ 12	65664.4	-28004.8	6670.75	-195.08	3788.33	-628.904	410.783
≥ 13	67281.7	-29116.7	7120.59	-202.817	3929.38	-688.738	492.309
≥ 14	69961.4	-31158.6	7834.02	-197.988	3917.29	-677.565	266.561
≥ 15	72146	-32795.7	8453.67	-195.083	3931.47	-681.037	99.0606
≥ 16	74142.6	-34244.8	9023.57	-190.645	3905.54	-663.682	10.8885
≥ 17	76411.4	-36026.3	9729.98	-188.874	3911.21	-663.449	-151.805
≥ 18	77091	-36088	9884.09	-188.554	3965.08	-708.55	59.3839
≥ 19	79194.5	-37566.4	10477.5	-181.656	3906.93	-682.4	-117.952
≥ 20	81600.4	-39464.5	11281.9	-175.182	3869.49	-677.179	-367.705

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15D/E/F/H/I						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	14376.7	102.205	-20.6279	-126.017	1903.36	-210.883	-493.065
≥ 4	24351.4	-2686.57	297.975	-110.819	2233.78	-301.615	-152.713
≥ 5	33518.4	-6711.35	958.544	-122.85	2522.7	-371.286	392.608
≥ 6	40377	-10472.4	1718.53	-144.535	2793.29	-426.436	951.528
≥ 7	46105.8	-13996.2	2515.32	-157.827	2962.46	-445.314	1100.56
≥ 8	50219.7	-16677.7	3198.3	-175.057	3176.74	-492.727	1223.62
≥ 9	54281.2	-19555.6	3983.47	-181.703	3279.03	-499.997	1034.55
≥ 10	56761.6	-21287.3	4525.98	-195.045	3470.41	-559.074	1103.3
≥ 11	59820	-23445.2	5165.43	-194.997	3518.23	-561.422	862.68
≥ 12	62287.2	-25164.6	5709.9	-194.771	3552.69	-561.466	680.488
≥ 13	64799	-27023.7	6335.16	-192.121	3570.41	-561.326	469.583
≥ 14	66938.7	-28593.1	6892.63	-194.226	3632.92	-583.997	319.867
≥ 15	68116.5	-29148.6	7140.09	-192.545	3670.39	-607.278	395.344
≥ 16	70154.9	-30570.1	7662.91	-187.366	3649.14	-597.205	232.318
≥ 17	72042.5	-31867.6	8169.01	-183.453	3646.92	-603.907	96.0388
≥ 18	73719.8	-32926.1	8596.12	-177.896	3614.57	-592.868	46.6774
≥ 19	75183.1	-33727.4	8949.64	-172.386	3581.13	-586.347	3.57256
≥ 20	77306.1	-35449	9690.02	-173.784	3636.87	-626.321	-205.513

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 16x16A						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	16226.8	143.714	-32.4809	-136.707	2255.33	-291.683	-699.947
≥ 4	27844.2	-3590.69	444.838	-124.301	2644.09	-411.598	-381.106
≥ 5	38191.5	-8678.48	1361.58	-132.855	2910.45	-473.183	224.473
≥ 6	46382.2	-13819.6	2511.32	-158.262	3216.92	-532.337	706.656
≥ 7	52692.3	-18289	3657.18	-179.765	3488.3	-583.133	908.839
≥ 8	57758.7	-22133.7	4736.88	-199.014	3717.42	-618.83	944.903
≥ 9	62363.3	-25798.7	5841.18	-207.025	3844.38	-625.741	734.928
≥ 10	66659.1	-29416.3	6993.31	-216.458	3981.97	-642.641	389.366
≥ 11	69262.7	-31452.7	7724.66	-220.836	4107.55	-681.043	407.121
≥ 12	72631.5	-34291.9	8704.8	-219.929	4131.5	-662.513	100.093
≥ 13	75375.3	-36589.3	9555.88	-217.994	4143.15	-644.014	-62.3294
≥ 14	78178.7	-39097.1	10532	-221.923	4226.28	-667.012	-317.743
≥ 15	79706.3	-40104	10993.3	-218.751	4242.12	-670.665	-205.579
≥ 16	82392.6	-42418.9	11940.7	-216.278	4274.09	-689.236	-479.752
≥ 17	84521.8	-44150.5	12683.3	-212.056	4245.99	-665.418	-558.901
≥ 18	86777.1	-45984.8	13479	-204.867	4180.8	-621.805	-716.366
≥ 19	89179.7	-48109.8	14434.5	-206.484	4230.03	-648.557	-902.1
≥ 20	90141.7	-48401.4	14702.6	-203.284	4245.54	-670.655	-734.604

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17A						
	A	B	C	D	E	F	G
≥ 3	15985.1	3.53963	-9.04955	-128.835	2149.5	-260.415	-262.997
≥ 4	27532.9	-3494.41	428.199	-119.504	2603.01	-390.91	-140.319
≥ 5	38481.2	-8870.98	1411.03	-139.279	3008.46	-492.881	388.377
≥ 6	47410.9	-14479.6	2679.08	-162.13	3335.48	-557.777	702.164
≥ 7	54596.8	-19703.2	4043.46	-181.339	3586.06	-587.634	804.05
≥ 8	60146.1	-24003.4	5271.54	-201.262	3830.32	-621.706	848.454
≥ 9	65006.3	-27951	6479.04	-210.753	3977.69	-627.805	615.84
≥ 10	69216	-31614.7	7712.58	-222.423	4173.4	-672.33	387.879
≥ 11	73001.3	-34871.1	8824.44	-225.128	4238.28	-657.259	101.654
≥ 12	76326.1	-37795.9	9887.35	-226.731	4298.11	-647.55	-122.236
≥ 13	78859.9	-40058.9	10797.1	-231.798	4402.14	-669.982	-203.383
≥ 14	82201.3	-43032.5	11934.1	-228.162	4417.99	-661.61	-561.969
≥ 15	84950	-45544.6	12972.4	-225.369	4417.84	-637.422	-771.254
≥ 16	87511.8	-47720	13857.7	-219.255	4365.24	-585.655	-907.775
≥ 17	90496.4	-50728.9	15186	-223.019	4446.51	-613.378	-1200.94
≥ 18	91392.5	-51002.4	15461.4	-220.272	4475.28	-636.398	-1003.81
≥ 19	94343.9	-53670.8	16631.6	-214.045	4441.31	-616.201	-1310.01
≥ 20	96562.9	-55591.2	17553.4	-209.917	4397.67	-573.199	-1380.64

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17B/C						
	A	B	C	D	E	F	G
≥ 3	14738	47.5402	-13.8187	-127.895	1946.58	-219.289	-389.029
≥ 4	25285.2	-3011.92	350.116	-115.75	2316.89	-319.23	-220.413
≥ 5	34589.6	-7130.34	1037.26	-128.673	2627.27	-394.58	459.642
≥ 6	42056.2	-11353.7	1908.68	-150.234	2897.38	-444.316	923.971
≥ 7	47977.6	-15204.8	2827.4	-173.349	3178.25	-504.16	1138.82
≥ 8	52924	-18547.6	3671.08	-183.025	3298.64	-501.278	1064.68
≥ 9	56465.5	-21139.4	4435.67	-200.386	3538	-569.712	1078.78
≥ 10	60190.9	-23872.7	5224.31	-203.233	3602.88	-562.312	805.336
≥ 11	63482.1	-26431.1	6035.79	-205.096	3668.84	-566.889	536.011
≥ 12	66095	-28311.8	6637.72	-204.367	3692.68	-555.305	372.223
≥ 13	67757.4	-29474.4	7094.08	-211.649	3826.42	-606.886	437.412
≥ 14	70403.7	-31517.4	7807.15	-207.668	3828.69	-601.081	183.09
≥ 15	72506.5	-33036.1	8372.59	-203.428	3823.38	-594.995	47.5175
≥ 16	74625.2	-34620.5	8974.32	-199.003	3798.57	-573.098	-95.0221
≥ 17	76549	-35952.6	9498.14	-193.459	3766.52	-556.928	-190.662
≥ 18	77871.9	-36785.5	9916.91	-195.592	3837.65	-599.45	-152.261
≥ 19	79834.8	-38191.6	10501.9	-190.83	3812.46	-589.635	-286.847
≥ 20	81975.5	-39777.2	11174.5	-185.767	3795.78	-595.664	-475.978

Table 2.1.29

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 7x7B & 10x10F†						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	26409.1	28347.5	-16858	-147.076	5636.32	-1606.75	1177.88
≥ 4	61967.8	-6618.31	-4131.96	-113.949	6122.77	-2042.85	-96.7439
≥ 5	91601.1	-49298.3	17826.5	-132.045	6823.14	-2418.49	-185.189
≥ 6	111369	-80890.1	35713.8	-150.262	7288.51	-2471.1	86.6363
≥ 7	126904	-108669	53338.1	-167.764	7650.57	-2340.78	150.403
≥ 8	139181	-132294	69852.5	-187.317	8098.66	-2336.13	97.5285
≥ 9	150334	-154490	86148.1	-193.899	8232.84	-2040.37	-123.029
≥ 10	159897	-173614	100819	-194.156	8254.99	-1708.32	-373.605
≥ 11	166931	-186860	111502	-193.776	8251.55	-1393.91	-543.677
≥ 12	173691	-201687	125166	-202.578	8626.84	-1642.3	-650.814
≥ 13	180312	-215406	137518	-201.041	8642.19	-1469.45	-810.024
≥ 14	185927	-227005	148721	-197.938	8607.6	-1225.95	-892.876
≥ 15	191151	-236120	156781	-191.625	8451.86	-846.27	-1019.4
≥ 16	195761	-244598	165372	-187.043	8359.19	-572.561	-1068.19
≥ 17	200791	-256573	179816	-197.26	8914.28	-1393.37	-1218.63
≥ 18	206068	-266136	188841	-187.191	8569.56	-730.898	-1363.79
≥ 19	210187	-273609	197794	-182.151	8488.23	-584.727	-1335.59
≥ 20	213731	-278120	203074	-175.864	8395.63	-457.304	-1364.38

†Array/Class 10x10F for MPC-68M only.

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8B						
	A	B	C	D	E	F	G
≥ 3	28219.6	28963.7	-17616.2	-147.68	5887.41	-1730.96	1048.21
≥ 4	66061.8	-10742.4	-1961.82	-123.066	6565.54	-2356.05	-298.005
≥ 5	95790.7	-53401.7	19836.7	-134.584	7145.41	-2637.09	-298.858
≥ 6	117477	-90055.9	41383.9	-154.758	7613.43	-2612.69	-64.9921
≥ 7	134090	-120643	60983	-168.675	7809	-2183.3	-40.8885
≥ 8	148186	-149181	81418.7	-185.726	8190.07	-2040.31	-260.773
≥ 9	159082	-172081	99175.2	-197.185	8450.86	-1792.04	-381.705
≥ 10	168816	-191389	113810	-195.613	8359.87	-1244.22	-613.594
≥ 11	177221	-210599	131099	-208.3	8810	-1466.49	-819.773
≥ 12	183929	-224384	143405	-207.497	8841.33	-1227.71	-929.708
≥ 13	191093	-240384	158327	-204.95	8760.17	-811.708	-1154.76
≥ 14	196787	-252211	169664	-204.574	8810.95	-610.928	-1208.97
≥ 15	203345	-267656	186057	-208.962	9078.41	-828.954	-1383.76
≥ 16	207973	-276838	196071	-204.592	9024.17	-640.808	-1436.43
≥ 17	213891	-290411	211145	-202.169	9024.19	-482.1	-1595.28
≥ 18	217483	-294066	214600	-194.243	8859.35	-244.684	-1529.61
≥ 19	220504	-297897	219704	-190.161	8794.97	-10.9863	-1433.86
≥ 20	227821	-318395	245322	-194.682	9060.96	-350.308	-1741.16

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8C/D/E						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	28592.7	28691.5	-17773.6	-149.418	5969.45	-1746.07	1063.62
≥ 4	66720.8	-12115.7	-1154	-128.444	6787.16	-2529.99	-302.155
≥ 5	96929.1	-55827.5	21140.3	-136.228	7259.19	-2685.06	-334.328
≥ 6	118190	-92000.2	42602.5	-162.204	7907.46	-2853.42	-47.5465
≥ 7	135120	-123437	62827.1	-172.397	8059.72	-2385.81	-75.0053
≥ 8	149162	-152986	84543.1	-195.458	8559.11	-2306.54	-183.595
≥ 9	161041	-177511	103020	-200.087	8632.84	-1864.4	-433.081
≥ 10	171754	-201468	122929	-209.799	8952.06	-1802.86	-755.742
≥ 11	179364	-217723	137000	-215.803	9142.37	-1664.82	-847.268
≥ 12	186090	-232150	150255	-216.033	9218.36	-1441.92	-975.817
≥ 13	193571	-249160	165997	-213.204	9146.99	-1011.13	-1119.47
≥ 14	200034	-263671	180359	-210.559	9107.54	-694.626	-1312.55
≥ 15	205581	-275904	193585	-216.242	9446.57	-1040.65	-1428.13
≥ 16	212015	-290101	207594	-210.036	9212.93	-428.321	-1590.7
≥ 17	216775	-299399	218278	-204.611	9187.86	-398.353	-1657.6
≥ 18	220653	-306719	227133	-202.498	9186.34	-181.672	-1611.86
≥ 19	224859	-314004	235956	-193.902	8990.14	145.151	-1604.71
≥ 20	228541	-320787	245449	-200.727	9310.87	-230.252	-1570.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9A						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30538.7	28463.2	-18105.5	-150.039	6226.92	-1876.69	1034.06
≥ 4	71040.1	-16692.2	1164.15	-128.241	7105.27	-2728.58	-414.09
≥ 5	100888	-60277.7	24150.1	-142.541	7896.11	-3272.86	-232.197
≥ 6	124846	-102954	50350.8	-161.849	8350.16	-3163.44	-91.1396
≥ 7	143516	-140615	76456.5	-185.538	8833.04	-2949.38	-104.802
≥ 8	158218	-171718	99788.2	-196.315	9048.88	-2529.26	-259.929
≥ 9	172226	-204312	126620	-214.214	9511.56	-2459.19	-624.954
≥ 10	182700	-227938	146736	-215.793	9555.41	-1959.92	-830.943
≥ 11	190734	-246174	163557	-218.071	9649.43	-1647.5	-935.021
≥ 12	199997	-269577	186406	-223.975	9884.92	-1534.34	-1235.27
≥ 13	207414	-287446	204723	-228.808	10131.7	-1614.49	-1358.61
≥ 14	215263	-306131	223440	-220.919	9928.27	-988.276	-1638.05
≥ 15	221920	-321612	239503	-217.949	9839.02	-554.709	-1784.04
≥ 16	226532	-331778	252234	-216.189	9893.43	-442.149	-1754.72
≥ 17	232959	-348593	272609	-219.907	10126.3	-663.84	-1915.3
≥ 18	240810	-369085	296809	-219.729	10294.6	-859.302	-2218.87
≥ 19	244637	-375057	304456	-210.997	10077.8	-425.446	-2127.83
≥ 20	248112	-379262	309391	-204.191	9863.67	100.27	-2059.39

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9B						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30613.2	28985.3	-18371	-151.117	6321.55	-1881.28	988.92
≥ 4	71346.6	-15922.9	631.132	-128.876	7232.47	-2810.64	-471.737
≥ 5	102131	-60654.1	23762.7	-140.748	7881.6	-3156.38	-417.979
≥ 6	127187	-105842	51525.2	-162.228	8307.4	-2913.08	-342.13
≥ 7	146853	-145834	79146.5	-185.192	8718.74	-2529.57	-484.885
≥ 8	162013	-178244	103205	-197.825	8896.39	-1921.58	-584.013
≥ 9	176764	-212856	131577	-215.41	9328.18	-1737.12	-1041.11
≥ 10	186900	-235819	151238	-218.98	9388.08	-1179.87	-1202.83
≥ 11	196178	-257688	171031	-220.323	9408.47	-638.53	-1385.16
≥ 12	205366	-280266	192775	-223.715	9592.12	-472.261	-1661.6
≥ 13	215012	-306103	218866	-231.821	9853.37	-361.449	-1985.56
≥ 14	222368	-324558	238655	-228.062	9834.57	3.47358	-2178.84
≥ 15	226705	-332738	247316	-224.659	9696.59	632.172	-2090.75
≥ 16	233846	-349835	265676	-221.533	9649.93	913.747	-2243.34
≥ 17	243979	-379622	300077	-222.351	9792.17	1011.04	-2753.36
≥ 18	247774	-386203	308873	-220.306	9791.37	1164.58	-2612.25
≥ 19	254041	-401906	327901	-213.96	9645.47	1664.94	-2786.2
≥ 20	256003	-402034	330566	-215.242	9850.42	1359.46	-2550.06

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9C/D						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30051.6	29548.7	-18614.2	-148.276	6148.44	-1810.34	1006
≥ 4	70472.7	-14696.6	-233.567	-127.728	7008.69	-2634.22	-444.373
≥ 5	101298	-59638.9	23065.2	-138.523	7627.57	-2958.03	-377.965
≥ 6	125546	-102740	49217.4	-160.811	8096.34	-2798.88	-259.767
≥ 7	143887	-139261	74100.4	-184.302	8550.86	-2517.19	-275.151
≥ 8	159633	-172741	98641.4	-194.351	8636.89	-1838.81	-486.731
≥ 9	173517	-204709	124803	-212.604	9151.98	-1853.27	-887.137
≥ 10	182895	-225481	142362	-218.251	9262.59	-1408.25	-978.356
≥ 11	192530	-247839	162173	-217.381	9213.58	-818.676	-1222.12
≥ 12	201127	-268201	181030	-215.552	9147.44	-232.221	-1481.55
≥ 13	209538	-289761	203291	-225.092	9588.12	-574.227	-1749.35
≥ 14	216798	-306958	220468	-222.578	9518.22	-69.9307	-1919.71
≥ 15	223515	-323254	237933	-217.398	9366.52	475.506	-2012.93
≥ 16	228796	-334529	250541	-215.004	9369.33	662.325	-2122.75
≥ 17	237256	-356311	273419	-206.483	9029.55	1551.3	-2367.96
≥ 18	242778	-369493	290354	-215.557	9600.71	659.297	-2589.32
≥ 19	246704	-377971	302630	-210.768	9509.41	1025.34	-2476.06
≥ 20	249944	-382059	308281	-205.495	9362.63	1389.71	-2350.49

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9E/F						
	A	B	C	D	E	F	G
≥ 3	30284.3	26949.5	-16926.4	-147.914	6017.02	-1854.81	1026.15
≥ 4	69727.4	-17117.2	1982.33	-127.983	6874.68	-2673.01	-359.962
≥ 5	98438.9	-58492	23382.2	-138.712	7513.55	-3038.23	-112.641
≥ 6	119765	-95024.1	45261	-159.669	8074.25	-3129.49	221.182
≥ 7	136740	-128219	67940.1	-182.439	8595.68	-3098.17	315.544
≥ 8	150745	-156607	88691.5	-193.941	8908.73	-2947.64	142.072
≥ 9	162915	-182667	109134	-198.37	8999.11	-2531	-93.4908
≥ 10	174000	-208668	131543	-210.777	9365.52	-2511.74	-445.876
≥ 11	181524	-224252	145280	-212.407	9489.67	-2387.49	-544.123
≥ 12	188946	-240952	160787	-210.65	9478.1	-2029.94	-652.339
≥ 13	193762	-250900	171363	-215.798	9742.31	-2179.24	-608.636
≥ 14	203288	-275191	196115	-218.113	9992.5	-2437.71	-1065.92
≥ 15	208108	-284395	205221	-213.956	9857.25	-1970.65	-1082.94
≥ 16	215093	-301828	224757	-209.736	9789.58	-1718.37	-1303.35
≥ 17	220056	-310906	234180	-201.494	9541.73	-1230.42	-1284.15
≥ 18	224545	-320969	247724	-206.807	9892.97	-1790.61	-1381.9
≥ 19	226901	-322168	250395	-204.073	9902.14	-1748.78	-1253.22
≥ 20	235561	-345414	276856	-198.306	9720.78	-1284.14	-1569.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9G						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	35158.5	26918.5	-17976.7	-149.915	6787.19	-2154.29	836.894
≥ 4	77137.2	-19760.1	2371.28	-130.934	8015.43	-3512.38	-455.424
≥ 5	113405	-77931.2	35511.2	-150.637	8932.55	-4099.48	-629.806
≥ 6	139938	-128700	68698.3	-173.799	9451.22	-3847.83	-455.905
≥ 7	164267	-183309	109526	-193.952	9737.91	-3046.84	-737.992
≥ 8	182646	-227630	146275	-210.936	10092.3	-2489.3	-1066.96
≥ 9	199309	-270496	184230	-218.617	10124.3	-1453.81	-1381.41
≥ 10	213186	-308612	221699	-235.828	10703.2	-1483.31	-1821.73
≥ 11	225587	-342892	256242	-236.112	10658.5	-612.076	-2134.65
≥ 12	235725	-370471	285195	-234.378	10604.9	118.591	-2417.89
≥ 13	247043	-404028	323049	-245.79	11158.2	-281.813	-2869.82
≥ 14	253649	-421134	342682	-243.142	11082.3	400.019	-2903.88
≥ 15	262750	-448593	376340	-245.435	11241.2	581.355	-3125.07
≥ 16	270816	-470846	402249	-236.294	10845.4	1791.46	-3293.07
≥ 17	279840	-500272	441964	-241.324	11222.6	1455.84	-3528.25
≥ 18	284533	-511287	458538	-240.905	11367.2	1459.68	-3520.94
≥ 19	295787	-545885	501824	-235.685	11188.2	2082.21	-3954.2
≥ 20	300209	-556936	519174	-229.539	10956	2942.09	-3872.87

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 10x10A/B/G†						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥ 4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥ 5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥ 6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥ 7	135115	-123721	63588.9	-171.747	8060.23	-2485.59	73.6219
≥ 8	148721	-151690	84143.9	-190.26	8515.81	-2444.25	-63.4649
≥ 9	160770	-177397	104069	-197.534	8673.6	-2101.25	-331.046
≥ 10	170331	-198419	121817	-213.692	9178.33	-2351.54	-472.844
≥ 11	179130	-217799	138652	-209.75	9095.43	-1842.88	-705.254
≥ 12	186070	-232389	151792	-208.946	9104.52	-1565.11	-822.73
≥ 13	192407	-246005	164928	-209.696	9234.7	-1541.54	-979.245
≥ 14	200493	-265596	183851	-207.639	9159.83	-1095.72	-1240.61
≥ 15	205594	-276161	195760	-213.491	9564.23	-1672.22	-1333.64
≥ 16	209386	-282942	204110	-209.322	9515.83	-1506.86	-1286.82
≥ 17	214972	-295149	217095	-202.445	9292.34	-893.6	-1364.97
≥ 18	219312	-302748	225826	-198.667	9272.27	-878.536	-1379.58
≥ 19	223481	-310663	235908	-194.825	9252.9	-785.066	-1379.62
≥ 20	227628	-319115	247597	-199.194	9509.02	-1135.23	-1386.19

†Array/Class 10x10G for MPC-68M only.

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 10x10C						
	A	B	C	D	E	F	G
≥ 3	31425.3	27358.9	-17413.3	-152.096	6367.53	-1967.91	925.763
≥ 4	71804	-16964.1	1000.4	-129.299	7227.18	-2806.44	-416.92
≥ 5	102685	-62383.3	24971.2	-142.316	7961	-3290.98	-354.784
≥ 6	126962	-105802	51444.6	-164.283	8421.44	-3104.21	-186.615
≥ 7	146284	-145608	79275.5	-188.967	8927.23	-2859.08	-251.163
≥ 8	162748	-181259	105859	-199.122	9052.91	-2206.31	-554.124
≥ 9	176612	-214183	133261	-217.56	9492.17	-1999.28	-860.669
≥ 10	187756	-239944	155315	-219.56	9532.45	-1470.9	-1113.42
≥ 11	196580	-260941	174536	-222.457	9591.64	-944.473	-1225.79
≥ 12	208017	-291492	204805	-233.488	10058.3	-1217.01	-1749.84
≥ 13	214920	-307772	221158	-234.747	10137.1	-897.23	-1868.04
≥ 14	222562	-326471	240234	-228.569	9929.34	-183.47	-2016.12
≥ 15	228844	-342382	258347	-226.944	9936.76	117.061	-2106.05
≥ 16	233907	-353008	270390	-223.179	9910.72	360.39	-2105.23
≥ 17	244153	-383017	304819	-227.266	10103.2	380.393	-2633.23
≥ 18	249240	-395456	321452	-226.989	10284.1	169.947	-2623.67
≥ 19	254343	-406555	335240	-220.569	10070.5	764.689	-2640.2
≥ 20	260202	-421069	354249	-216.255	10069.9	854.497	-2732.77

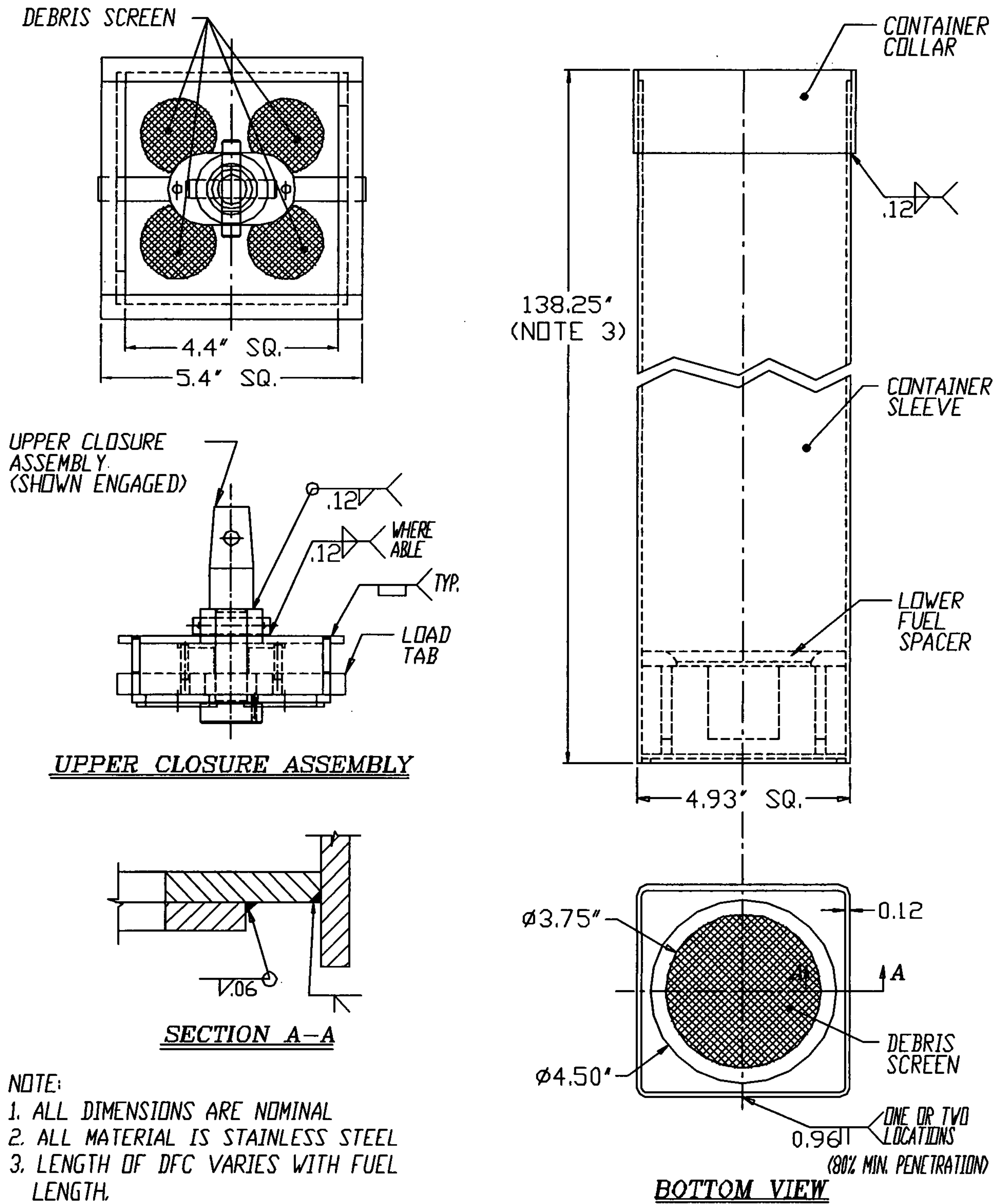
Table 2.1.30

MPC Regionalized Loading Heat Load Limits (q_1 and q_2)[1] for Discrete Values of X

X	MPC-24		MPC-32		MPC-68	
	q_1 (kW)	q_2 (kW)	q_1 (kW)	q_2 (kW)	q_1 (kW)	q_2 (kW)
0.5	1.025	2.050	0.710	1.419	0.354	0.710
0.6	1.128	1.880	0.796	1.327	0.392	0.653
0.7	1.216	1.737	0.873	1.248	0.424	0.606
0.8	1.292	1.615	0.943	1.178	0.453	0.566
0.9	1.358	1.509	1.005	1.117	0.478	0.531
1	1.416	1.416	1.062	1.062	0.500	0.500
1.1	1.468	1.334	1.114	1.012	0.519	0.472
1.2	1.513	1.261	1.161	0.968	0.537	0.447
1.3	1.554	1.195	1.205	0.926	0.552	0.425
1.4	1.590	1.136	1.245	0.889	0.567	0.405
1.5	1.623	1.082	1.282	0.854	0.579	0.386
1.6	1.653	1.033	1.316	0.822	0.591	0.369
1.7	1.680	0.988	1.347	0.792	0.602	0.354
1.8	1.705	0.947	1.377	0.765	0.612	0.340
1.9	1.728	0.909	1.405	0.739	0.621	0.326
2	1.748	0.874	1.430	0.715	0.629	0.314
2.1	1.767	0.841	1.454	0.692	0.637	0.303
2.2	1.785	0.811	1.477	0.671	0.644	0.292
2.3	1.801	0.783	1.498	0.651	0.650	0.282
2.4	1.816	0.756	1.518	0.632	0.656	0.273
2.5	1.829	0.731	1.537	0.614	0.662	0.265
2.6	1.842	0.708	1.554	0.597	0.667	0.256
2.7	1.854	0.686	1.571	0.581	0.672	0.249
2.8	1.865	0.666	1.587	0.566	0.677	0.241
2.9	1.875	0.646	1.602	0.552	0.681	0.235
3	1.885	0.628	1.616	0.538	0.685	0.228

*See Table 2.1.27 for the number of storage cells (n) in each region for the specific MPC type listed.

1 Under SCS mandatory conditions evaluated in HI-TRAC operations Section 4.5, the storage cell heat loads tabulated herein are limited by the heat load reduction factor defined in Table 4.5.4.



NOTE:
1. ALL DIMENSIONS ARE NOMINAL
2. ALL MATERIAL IS STAINLESS STEEL
3. LENGTH OF DFC VARIES WITH FUEL LENGTH.

FIGURE 2.1.1; DAMAGED FUEL CONTAINER FOR DRESDEN UNIT-1/ HUMBOLDT BAY SNF

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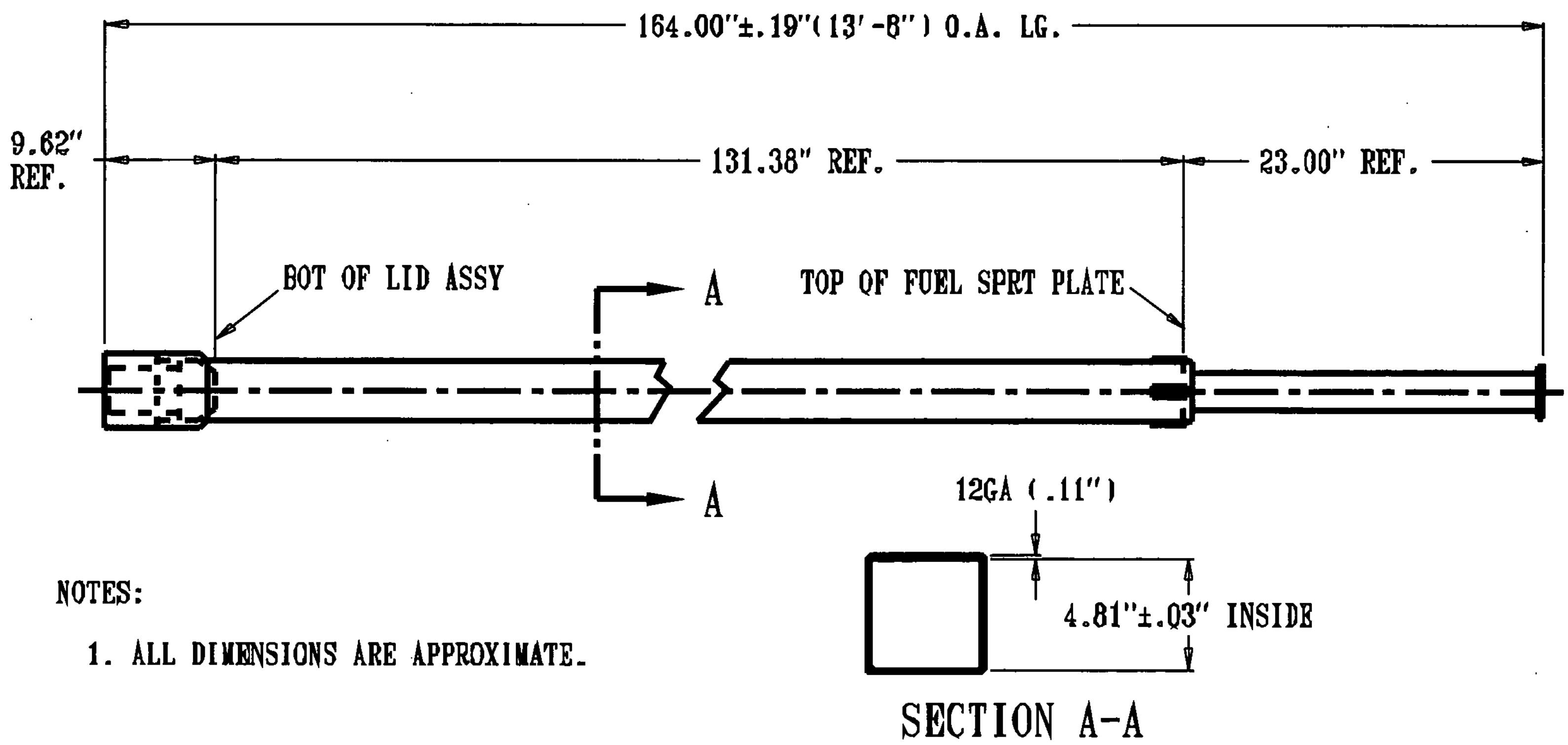

NOTES:

1. ALL DIMENSIONS ARE APPROXIMATE.

FIGURE 2.1.2; TN DAMAGED FUEL CANISTER FOR DRESDEN UNIT-1

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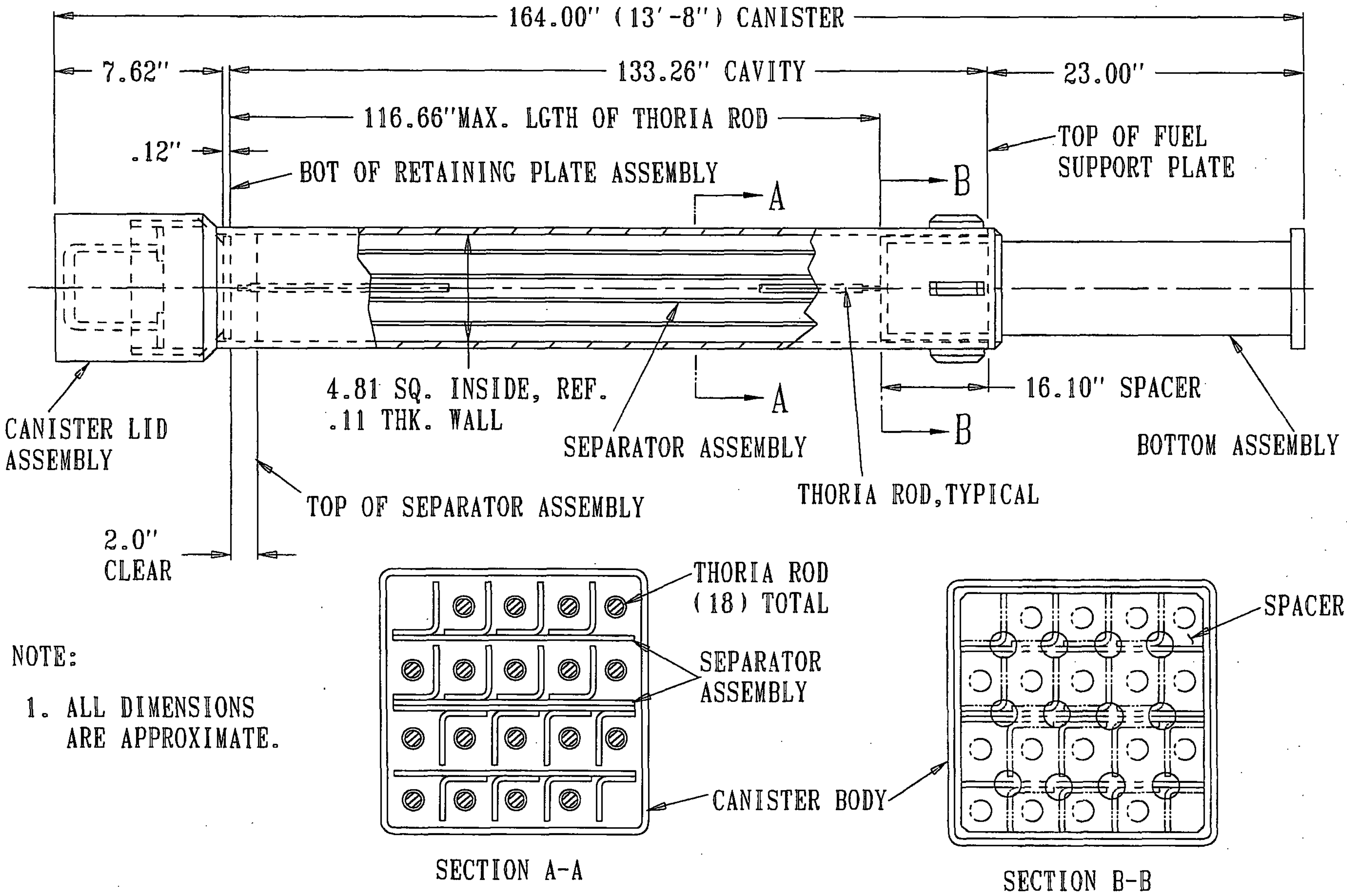

FIGURE 2.1.2A; TN THORIA ROD CANISTER FOR DRESDEN UNIT-1

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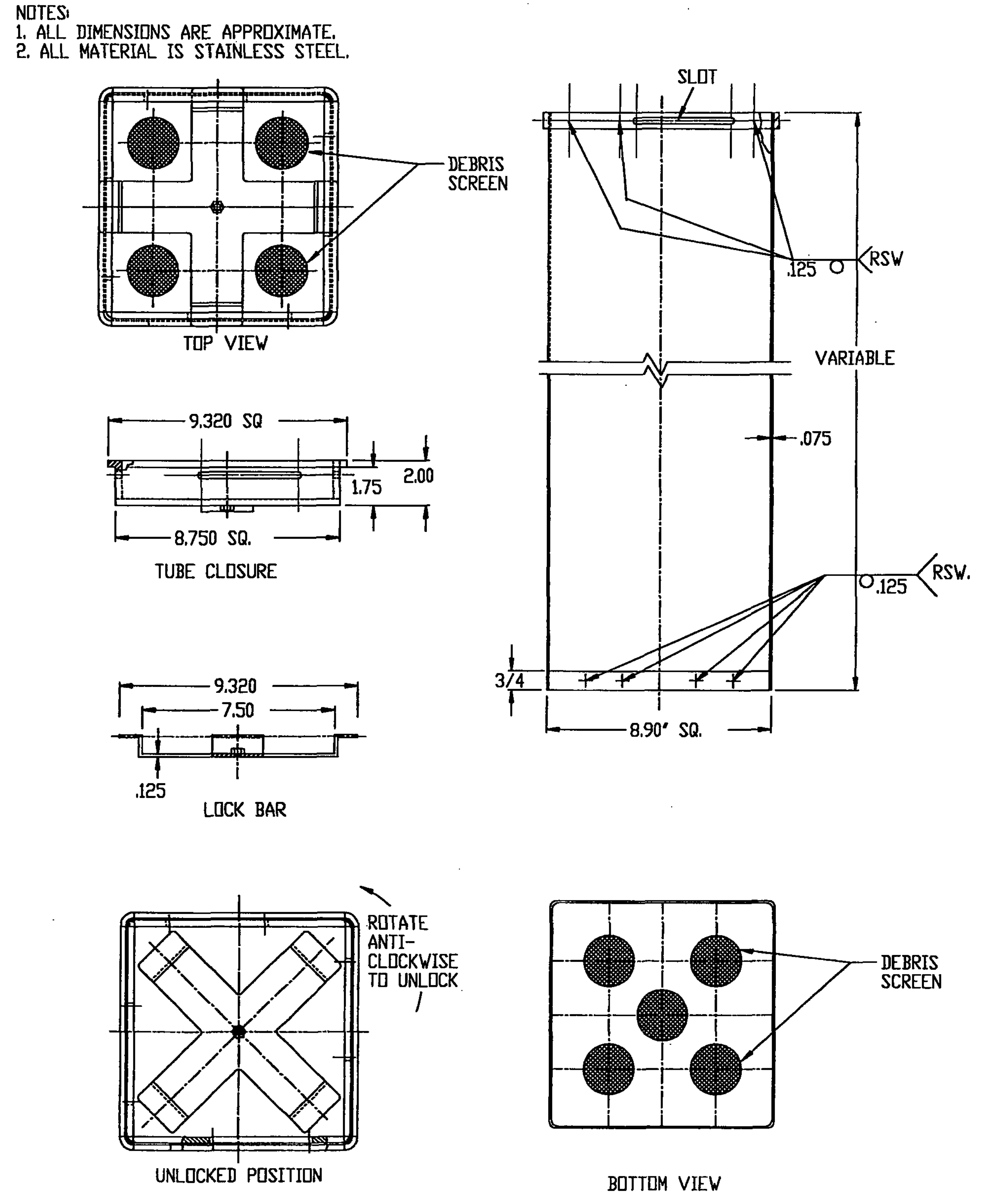

FIGURE 2.1.2B; HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-24E/24EF

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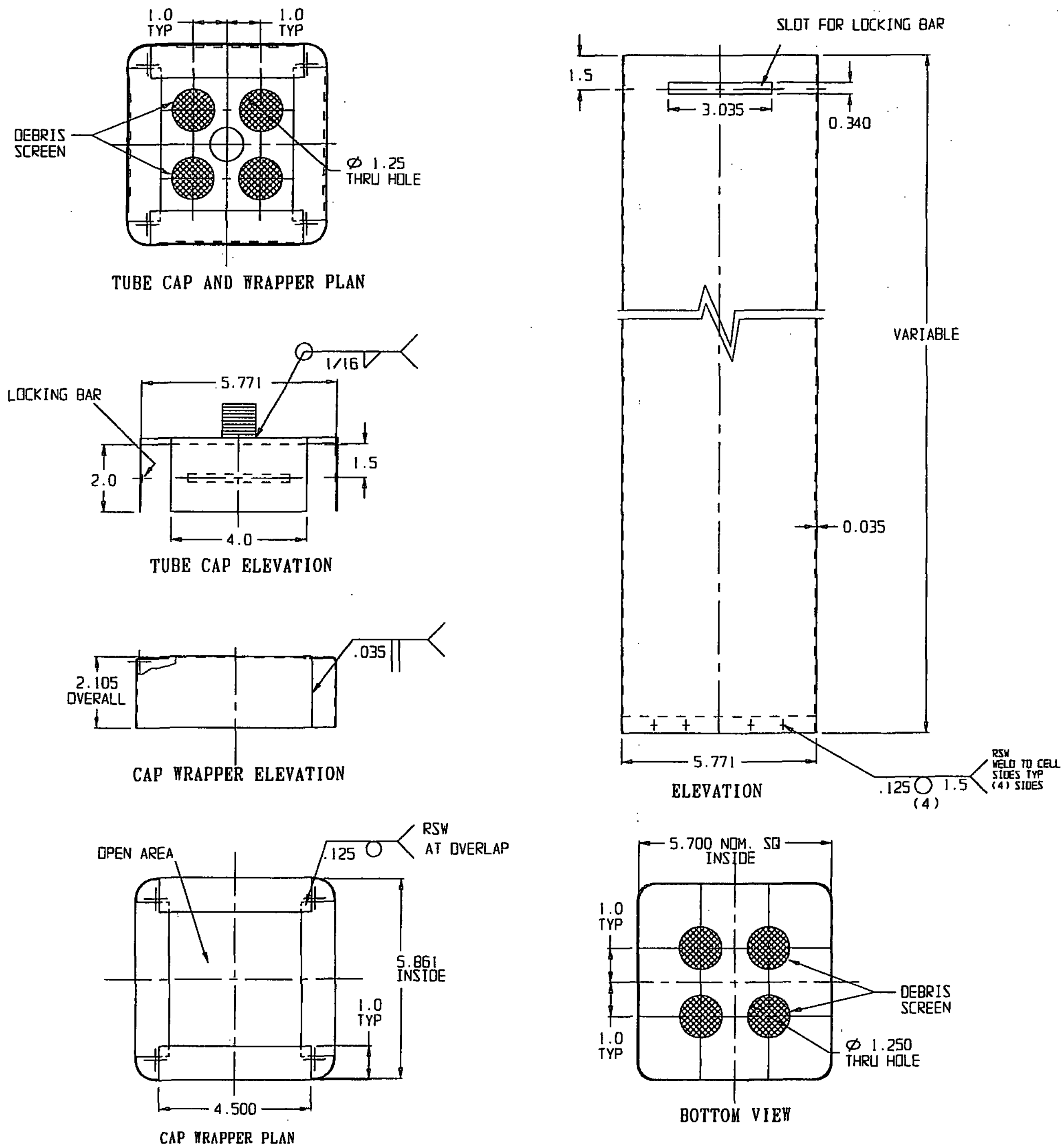

FIGURE 2.1.2C; HOLTEC DAMAGED FUEL CONTAINER FOR BWR SNF IN MPC-68/68FF

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

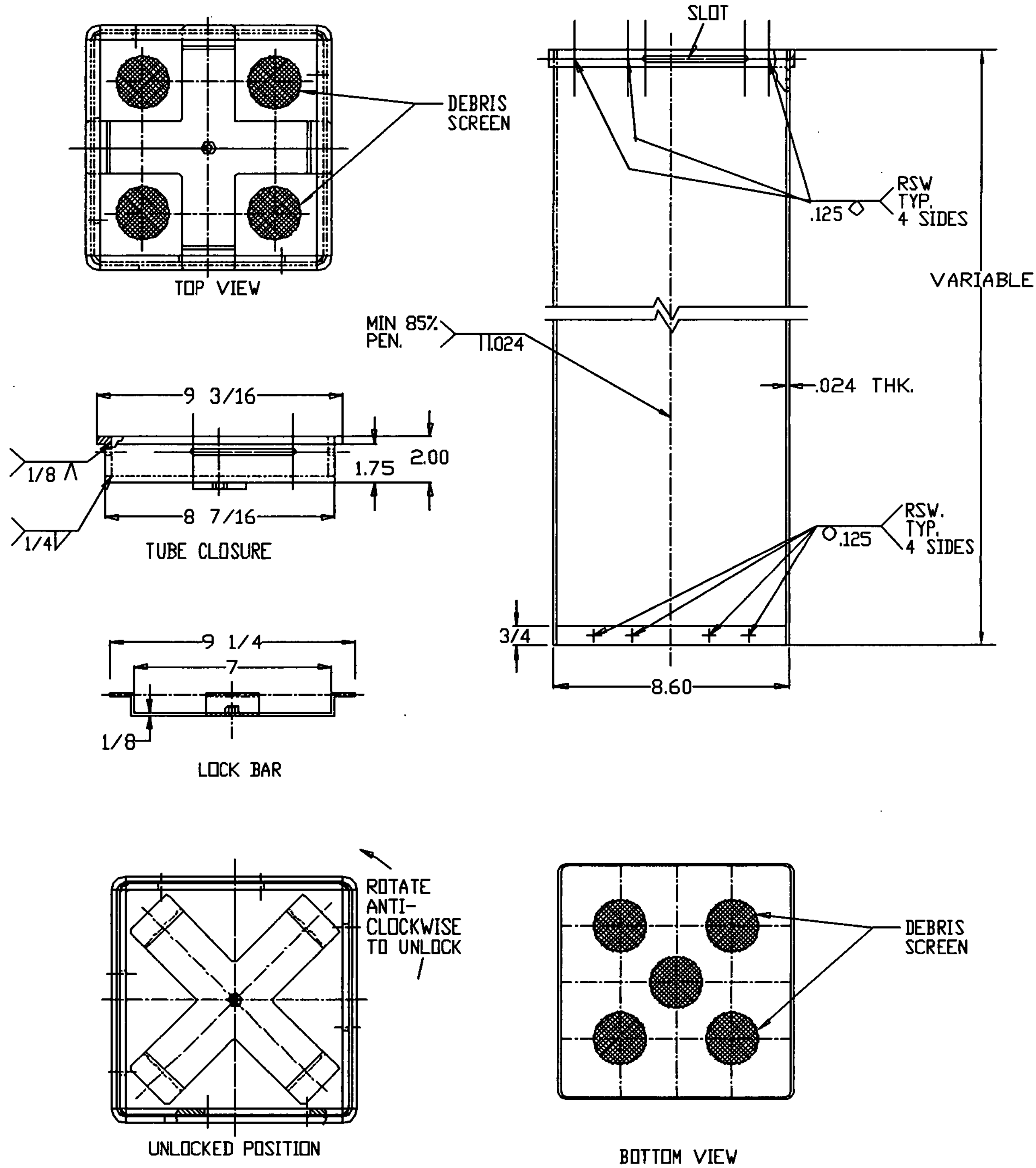

FIGURE 2.1.2D; HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-32/32F

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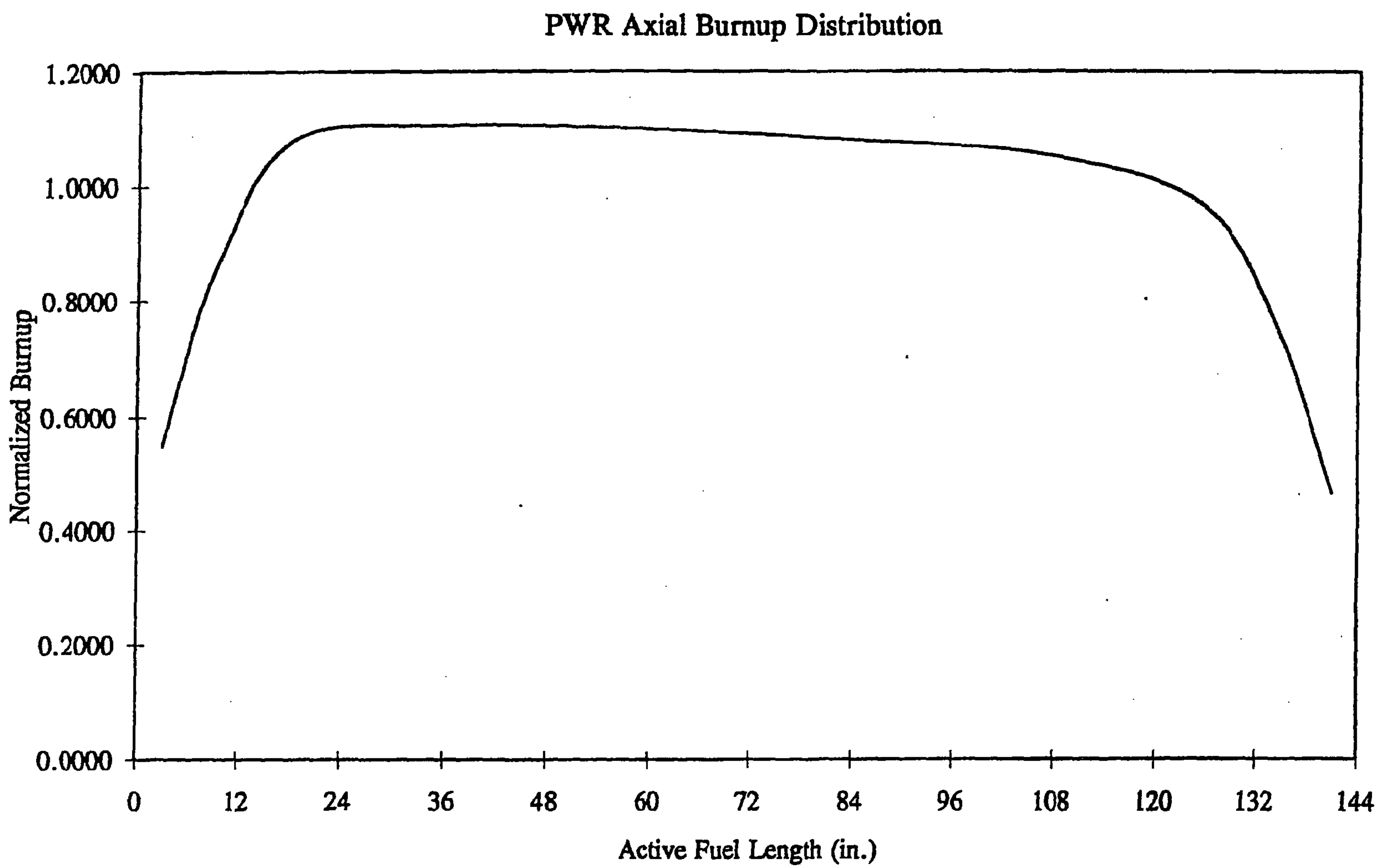

Figure 2.1.3; PWR Axial Burnup Profile with Normalized Distribution

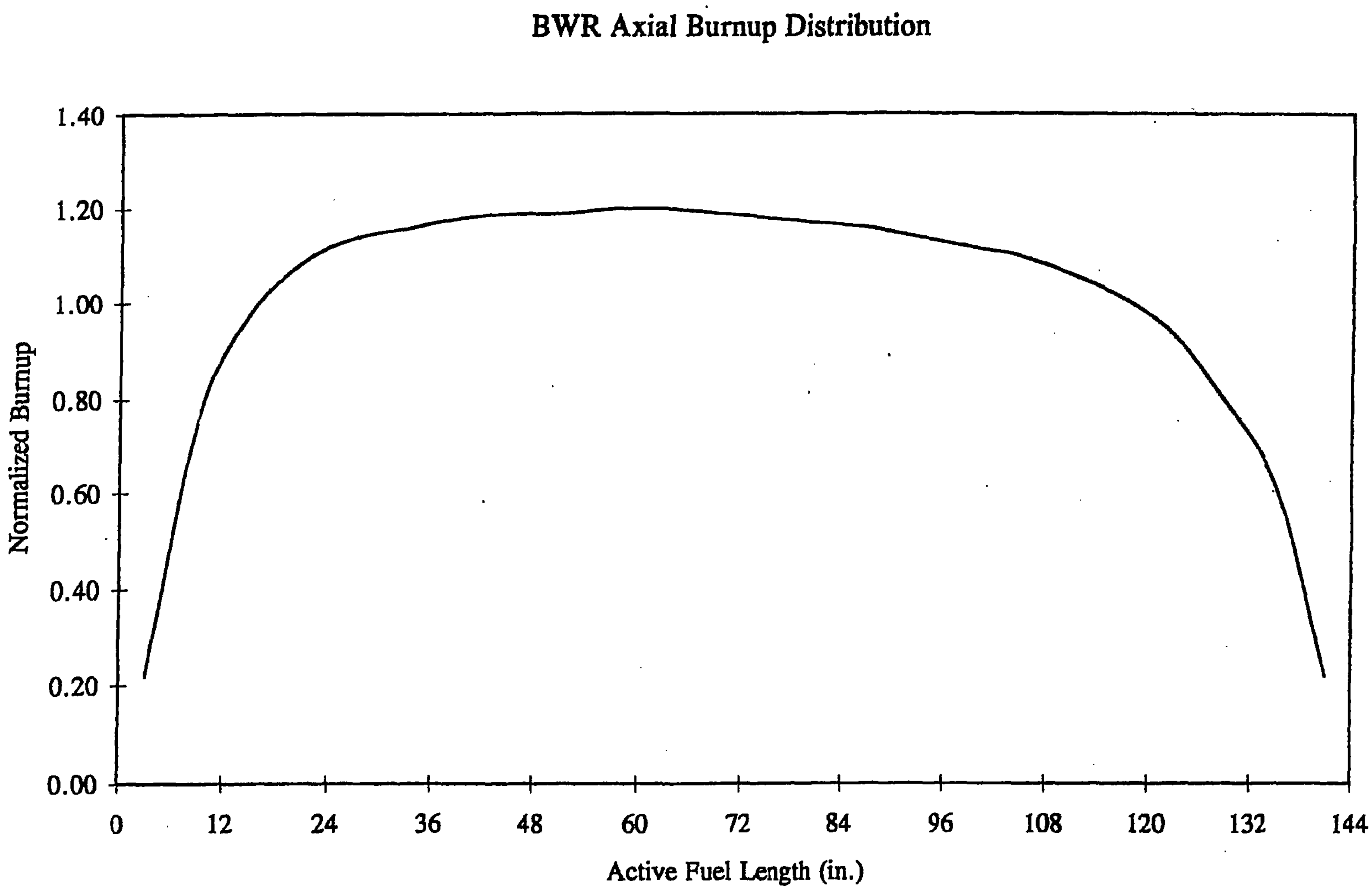

Figure 2.1.4; BWR Axial Burnup Profile with Normalized Distribution

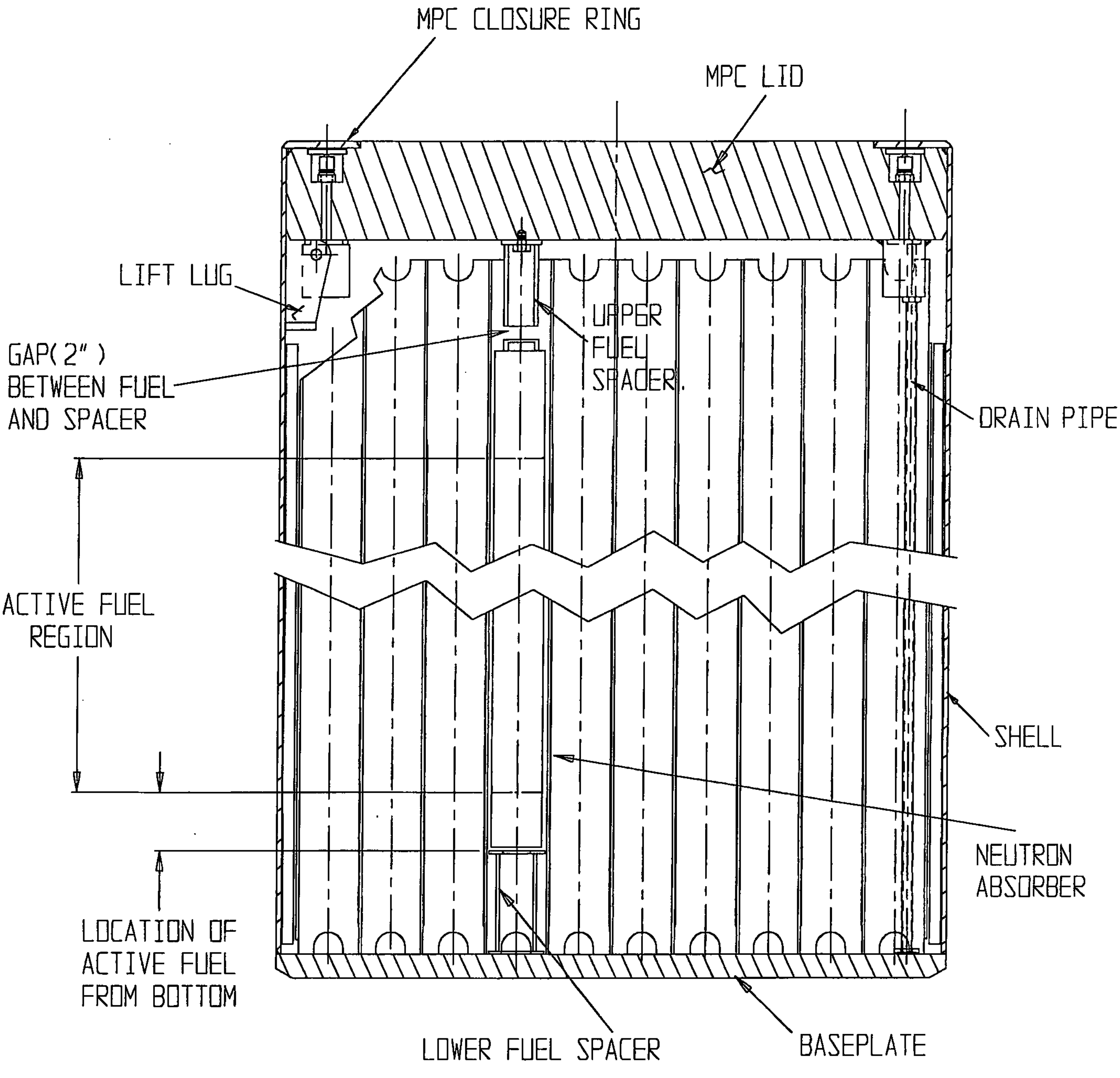

FIGURE 2.1.5; MPC WITH UPPER AND LOWER FUEL SPACERS

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FIGURE 2.1.6

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FIGURE 2.1.7; DELETED

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FIGURE 2.1.8; DELETED

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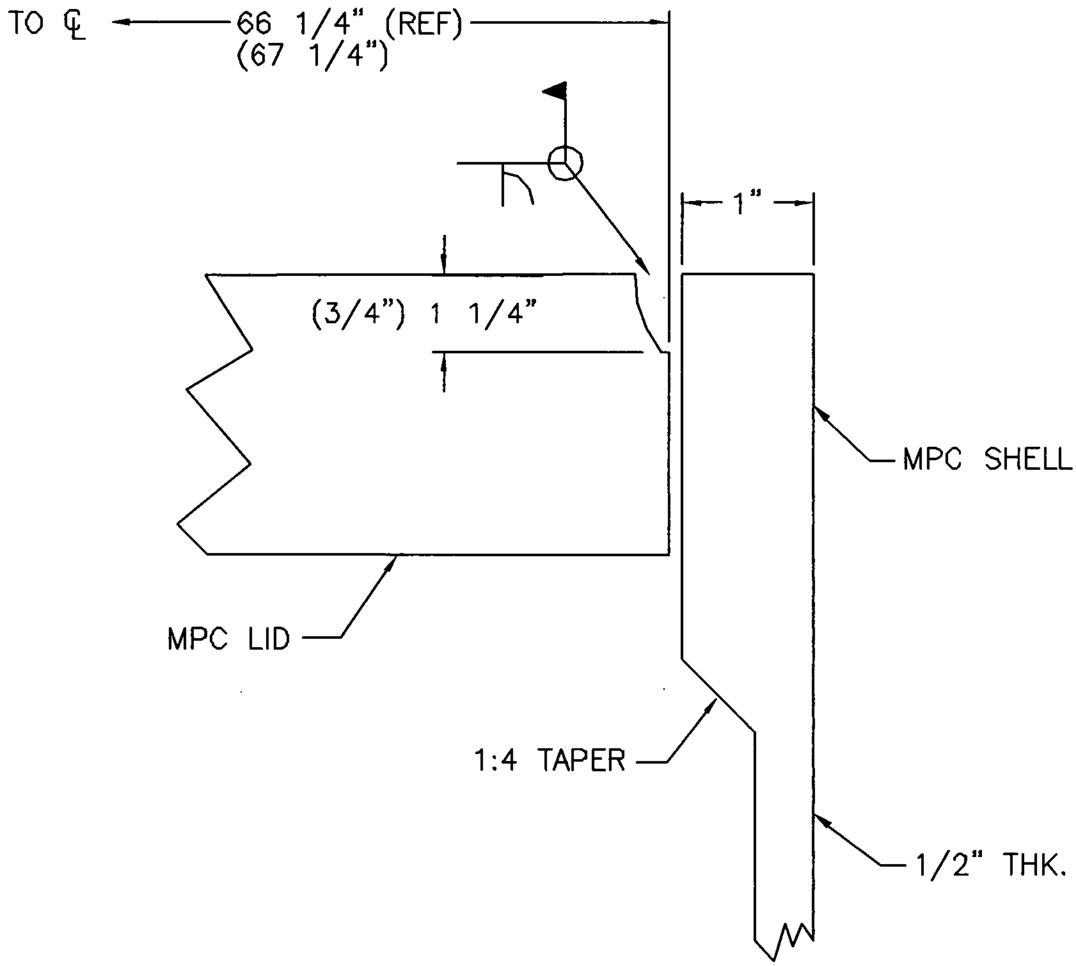

NOTES: 1. Standard MPC dimensions in parentheses.
2. Standard MPC shell thickness is 1/2" along its entire length.
3. Figure is not to scale.

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

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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, Malfunction of Forced Helium Dehydrator System, Supplemental Cooling System Power Failure

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, Supplemental Cooling System Operational Failure

Short-Term Operations: This loading condition is defined to accord with ISG-11, Revision 3 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 special lifting 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., H^3, 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., H^3, 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 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 3 [2.0.8]. For moderate burnup fuel, the previously licensed PCT limit of 570°C (1058°F) may be used [2.0.9] (see also Section 4.5).

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., H^3, Kr, and Xe) released in accordance with NUREG-1536.

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 condition design temperatures, which apply to long-term storage and short-term normal operating conditions (e.g., MPC drying operations and onsite transport operations), 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 temperature, which may exist during a transient event (examples of such instances are the overpack blocked air duct off-normal event and fire accident). 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 the transient event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects during or immediately after, a transient event.

The off-normal/accident design temperatures for stainless steel and carbon steel components are chosen such that the material's ultimate tensile strength does not fall below 30% of its room temperature value, based on data in published references [2.2.12 and 2.2.13]. This ensures that the material will not fail due to creep rupture during these short duration transient events.

2.2.2.4 Leakage of One Seal

The MPC enclosure vessel is designed to have no credible leakage under all normal, off-normal, and hypothetical accident conditions of storage.

The confinement boundary is defined by the MPC shell, baseplate, MPC lid, port cover plates, closure ring, and associated welds. MPC shell welds and shell to baseplate weld are subject to helium leakage testing. 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 pressure tested, and volumetrically examined or multi-pass liquid penetrant examined. The vent and drain port cover plates are subject to liquid penetrant examination and helium leakage testing. These inspection and testing techniques are performed to verify the integrity of the confinement boundary.

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 defined in Table 2.0.2 as 50% blockage of the four air inlets. Because the overpack air inlets and outlets are covered by screens, located 90° apart, and inspected routinely (or alternatively, exit vent air temperature monitored), significant blockage of all vents by blowing debris, animals, etc. is very unlikely. To demonstrate the inherent thermal stability of the HI-STORM 100 System all four air inlets are assumed to be 50% blocked.

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 100D and 125D transfer cask designs do 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.

2.2.2.7 Malfunction of Forced Helium Dehydrator (FHD)

The FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, the FHD system is equipped with active components requiring external power for normal operation.

Initiating events of FHD malfunction are: (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs and heat dissipation in the MPC transitions to natural convection cooling.

Although the FHD System is monitored during its operation, stoppage of FHD operations does not require actions to restore forced cooling for adequate heat dissipation. This is because the condition of natural convection cooling evaluated in Section 4.5 shows that the fuel temperatures remain below off-normal limits. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

2.2.2.8 Supplemental Cooling System (SCS) Power Failure

The SCS system is a fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation. The SCS is normally operated from an external source of power such as from site utilities or a feed from a heavy haul vehicle carrying the HI-TRAC. Occasional interruption in power supply is possible.

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 special lifting devices 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 special 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, and the number of fuel assemblies being stored (full or partially loaded MPC). For site-specific conditions including impact surfaces not encompassed by Table 2.2.9 or handling a partially loaded MPC, 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 100D and 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, and the number of fuel assemblies being stored (full or partially loaded MPC). For site-specific conditions including impact surfaces not encompassed by Table 2.2.9 or handling a partially loaded MPC, 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 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 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-STORM 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, HI-STORM 100S, and HI-STORM 100S Version B 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, HI-STORM 100S, and HI-STORM 100S Version B 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 System 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 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 temperature limits specified in ISG-11, Rev. 3 [2.0.9].

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. The damaged fuel container is equipped with 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 overpack. Blockage of the air inlets is addressed in Subsection 2.2.3.13.

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

† 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.

ZPA provided in Table 2.2.8 is the vector sum of two horizontal earthquakes.

For anchored casks, the limit on zero period accelerations (ZPA) 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., H^3, Kr, and Xe) released in accordance with NUREG-1536. All 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. Section 7.1 provides a discussion as to why leakage of any magnitude from the MPC is not credible, based on the materials and methods of fabrication and inspection.

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, or appropriate preventive measures are taken to preclude internal explosive events (see Section 1.2.1.3.1.1). The MPC is composed of stainless steel, neutron absorber material, and prior to CoC Amendment 2, possibly optional aluminum alloy 1100 heat conduction elements. For these materials, and considering the protective measures taken during loading and unloading operations there is no credible 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 are 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 off-normal/accident 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 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.

2.2.3.16 Supplemental Cooling System (SCS) Failure

The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation.

Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that

renders it inoperable for an extended duration is postulated. Possible causes of SCS failure are: (a) Simultaneous loss of external and backup power, or (b) Complete loss of annulus water from an uncontrolled leak or line break.

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 IX. The latest effective editions of ASME Section V and IX may be used, provided a written reconciliation of the later edition against the 1995 Edition, including addenda, is performed by the certificate holder. 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 318.1-85(92) 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 318.1-85(92) 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 alternatives to the ASME Code for the HI-STORM 100 System and the justification for those 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, this document refers to those SARs, as applicable, if the MPC loadings for storage in the HI-STORM 100 System do not exceed those computed in the 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. Tables 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/Short-Term	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)

* The overpack is also qualified to sustain without tip-over a lateral impulse load of 60 psi (differential pressure for 85 milliseconds maximum) [3.4.5].

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		
Inside Building Short-Term Operations (3-Day Average)	110	110°F with no insolation
Outside Building Short-Term Operations (3-Day Average)	90	90°F with insolation

Note:

1. Handling operations with the loaded HI-STORM 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.

Table 2.2.3

DESIGN TEMPERATURES

HI-STORM 100 Component	Long Term, Normal Condition Design Temperature Limits (Long-Term Events) (° F)	Off-Normal and Accident Condition Temperature Limits (Short-Term Events)† (° F)
MPC shell	500	775
MPC basket	725	950
MPC Neutron Absorber	800	1000
MPC lid	550	775
MPC closure ring	400	775
MPC baseplate	400	775
HI-TRAC inner shell	400	800
HI-TRAC pool lid/transfer lid	350	800
HI-TRAC top lid	400	800
HI-TRAC top flange	400	700
HI-TRAC pool lid seals	350	N/A
HI-TRAC bottom lid bolts	350	800
HI-TRAC bottom flange	350	800
HI-TRAC top lid neutron shielding	300	350
HI-TRAC radial neutron shield	307	N/A
HI-TRAC radial lead gamma shield	350	600
Remainder of HI-TRAC	350	800
Fuel Cladding	752	752 or 1058 (Short Term Operations)†† 1058 (Off-Normal and Accident Conditions)
Overpack concrete	300	350
Overpack Lid Top and Bottom Plate	450	800
Remainder of overpack steel structure	350	800

† 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).

†† Normal short term operations includes MPC drying and onsite transport per Reference [2.0.8]. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel as discussed in Reference [2.0.9]. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F. See also Section 4.3.

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 (One-piece design and top portion of optional two-piece design)	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 core support structures (NG-1121)	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	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

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 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 $\geq$ 75 ksi, yield strength $\geq$ 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)
Shielding	Bottom portion of optional two-piece MPC lid design	B	Non-code	Alloy X or Carbon Steel	See Appendix 1.A for Alloy X, Table 3.3.6 for Carbon Steel	Stainless Steel coating when using Carbon Steel	Stainless Steel when using Carbon Steel
Structural Integrity	Upper Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Sheathing	A	Non-code	Alloy X	See Appendix 1.A	Aluminum/SS	NA
Structural Integrity	Shims	NITS	Non-code (shims, welded directly to angle or parallel plate basket supports, are ASME Section II)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Angled Plate or Parallel Plates with connecting end shim)	A	ASME Section III; Subsection NG internal structures (NG-1122)	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

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 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 $\geq$ 75 ksi, yield strength $\geq$ 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	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 II	NA	NA
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

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 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)
Operations	Damaged Fuel Container	C	ASME Section III; Subsection NG	S/S (Primarily 304 S/S)	See Appendix 1.A	NA	NA
Operations	Drain Line Guide Tube	NITS	Non-code	S/S	NA	NA	NA
Operations	Vent and Drain Tube, Optional	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	N/A
Operations	Threaded Disc, Plug Adjustment	C	Non-code	S/S	Per ASME Section II	N/A	N/A
Operations	Vent and Drain Plug	C	Non-code	Aluminum	N/A	N/A	N/A
Operations	Thread Shield Cap	NITS	Non-code	Aluminum	N/A	N/A	N/A
Operations	Retaining Ring	NITS	Non-code	S/S	N/A	N/A	N/A

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 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 $\geq$ 75 ksi, yield strength $\geq$ 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 100S Version B) and Shield Block Shell (100S)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70 (SA515-70 not permitted for 100S Version B)	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

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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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	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
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

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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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)
Thermal	Heat Shield	B	N/A	C/S	N/A	See Note 5	N/A
Thermal	Heat Shield Ring	B	N/A	C/S	N/A	See Note 5	N/A
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 (Cover) Plate, including shear ring (100 and 100S)	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
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 (stud) Per ASME Section II (nut)	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, SA350-LF2, or SA203E	See Table 3.3.3	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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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	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	NA
Structural Integrity	Channel Mounts	B	ASME Section III; Subsection NF	SA36 or equivalent	Per ASME Section II	See Note 5	NA
Shielding	Pedestal Platform	B	Non-Code	SA36 or equivalent	NA	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
Operations	Exit & Inlet Screen Frame	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Temperature Element & Associated Temperature Monitoring Equipment	C	Non-code	NA	NA	NA	NA
Operations	Screen	NITS	Non-code	Mesh Wire	NA	NA	NA
Operations	Paint	NITS	Non-code	Per Appendix 1.C	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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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	100S Version B Base Bottom Plate	B	ASME III; Subsection NF	Carbon Steel	See Table 3.3.6	See Note 5	NA
Structural Integrity	100S Version B Base Spacer Block	B	Non-code	Carbon Steel	See Table 3.3.6	See Note 5	NA
Shielding	100S Version B Base Shield Block	B	Non-code	Carbon Steel	NA	See Note 5	NA
Structural Integrity	100S Version B Base Top Plate	B	ASME III; Subsection NF	SA 516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	100S Version B Base MPC Support	B	Non-code	Carbon Steel	See Table 3.3.6	See Note 5	NA
Shielding	100S Version B Lid Outer Ring	B	ASME III; Subsection NF	SA516-70 or SA36	See Table 3.3.2 or Table 3.3.6	See Note 5	NA
Operations	100S Version B Lid Vent Duct	NITS	Non-code	Carbon Steel	NA	See Note 5	NA
Structural Integrity	100S Version B Lid Inner Ring	B	ASME III; Subsection NF	Carbon Steel	See Table 3.3.6	See Note 5	NA
Operations	100S Version B Lid Stud Pipe	NITS	Non-code	Carbon Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Spacer	NITS	Non-code	Carbon Steel	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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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	100S Version B Lid Lift Block	B	ASME III; Subsection NF	SA36	See Table 3.3.6	See Note 5	NA
Shielding	100S Version B Lid Vent Shield	B	Non-code	Carbon Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Washer	C	Non-code	Stainless Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Cap	NITS	Non-code	PVC	NA	See Note 5	NA
Structural Integrity	100S Version B Radial Gusset	B	ASME III; Subsection NF	SA 516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	100S Version B Lid Closure Bolt and Closure Bolt Handle	B (bolt) NITS (bolt Handle)	ASME Section II	SA 193-B7 (bolt) C/S (bolt handle)	See Table 3.3.4 (bolt) NA (bolt handle)	Threads to have cadmium coating (or similar lubricant for corrosion protection)	NA
Structural Integrity	100S Version B Lid Top (Cover) Plate	B	ASME Section III; Subsection NF	SA516-70 or SA36	See Table 3.3.2 or Table 3.3.6	See Note 5	NA
Structural Integrity	100S Version B Shear Ring	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 Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
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 or S/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 100D and 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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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 3.3.2 or 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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lifting Trunnion	A	ANSI N14.6	SB637 (N07718) or SA564-630H1100 (For HI-TRAC125D only)	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 N14.6	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

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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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	Bottom Flange Gussets (HI-TRAC 100D and 125D only)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Top Lid Stud or bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA
Operations	Top Lid Nut	B	ASME Section III; Subsection NF	SA194-2H	Per ASME Section II	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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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

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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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)
Structural Integrity	Door Side Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
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 the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

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; core support structures (NG-1121)	Section III, Subsection NG, NG-3300 and NG-3200; core support structures (NG-1121)	Section III, Subsection NG, NG-4000; core support structures (NG-1121)	Section III, Subsection NG, NG-5000 and Section V; core support structures (NG-1121)
HI-TRAC Trunnions	Section II, Section III, Subsection NF, NF-2000	ANSI N14.6	Section III, Subsection NF, NF-4000	See Chapter 9
MPC basket supports (Angled Plates)	Section II, Section III, Subsection NG, NG-2000; internal structures (NG-1122)	Section III, Subsection NG, NG-3300 and NG-3200; internal structures (NG-1122)	Section III, Subsection NG, NG-4000; internal structures (NG-1122)	Section III, Subsection NG, NG-5000 and Section V; internal structures (NG-1122)
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 318.1-89(92) 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 acceleration for HI-STORM	Accident	$G_H + 0.53G_V = 0.53$ †† (HI-STORM 100, 100S, and 100S Version B) $G_H = 2.12$; $G_V = 1.5$ (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††† (HI-STORM 100 and 100S), OR By Users (HI-STORM 100A)
HI-TRAC Transfer Cask Horizontal Lift Height Limit (in.)	Accident	42†††

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

††† For ISFSI and subgrade design parameter Sets A and B and fully loaded MPC. 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 used. If for a specific ISFSI a higher value of the coefficient of friction is used, it 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)

		SERVICE CONDITION	
STRESS CATEGORY	**DESIGN + LEVEL A**	**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)	P_i’
Off-Normal Pressure (External)†	P_o
Off-Normal Temperature	T’
Off-Normal HI-TRAC Handling	H’

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 ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure.	Subsection NCA	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.	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. 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.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC basket supports and lift lugs	NB-1130	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. 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.	The MPC basket supports (nonpressure - retaining structural attachment) and lift lugs (nonstructural attachments (relative to the function of lifting a loaded MPC) that are 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.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC, MPC basket assembly, HI-STORM overpack, and HI-TRAC transfer cask	NB-3100 NG-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	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.
MPC	NB-3350	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.	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. 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.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure	NB-4120 NG-4120 NF-4120	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.	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. 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).
MPC, HI-STORM overpack steel structure, HI-TRAC transfer cask steel structure	NB-4220 NF-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	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.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, 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.
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 closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
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 pressure tested as defined in Chapter 9. Accessibility for leakage inspections precludes a Code compliant pressure test. Since the shell welds of the MPC cannot be checked for leakage during this pressure test, the shop leakage test to 10^{-7} ref cc/sec (as described in Chapter 9) provides reasonable assurance as to its leak tightness. All MPC 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 PT, the maximum undetectable flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection results, including relevant findings

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			(indications) shall be made a permanent part of the user's records by video, photographic, of other means which provide an equivalent 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 and the closure ring welds are 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.
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.	The HI-STORM 100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NG-2000 requirements.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	NG-4420	NG-4427(a) requires a fillet weld in any single continuous weld may be less that 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.	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. 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).

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package 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.
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 margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have large safety factors during lifting. Likewise, the top lid plate to lid shell weld has a large structural margin under the inertia loads imposed during a non-mechanistic tipover event.

Table 2.2.15 (continued)
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
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. 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.
HI-STORM Overpack and HI-TRAC Transfer Cask	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting	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 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). 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.

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 the HI-STAR SAR under docket number 71-9261, which does not impose any off-normal condition loadings.

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 pressure tested and/or inspected by the liquid penetrant method (see Section 9.1).

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, helium leakage testing and pressure 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 that maximizes thermal resistance (see Section 2.1.6). Additional margin is built into the calculated cask cooling rate by using the design basis fuel assembly that offers maximum resistance to MPC internal helium circulation.

- 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. Ancillary equipment is classified as the highest ITS classification of any of its constituent parts/components. The ITS classification of the ancillary equipment consituent parts/components are determined and documented in the quality program documents such as purchase specifications. 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 using devices not integral to structures 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 may be any of the following types to effectuate the cask manipulations and MPC transfers:

1. Stand-alone, aboveground
2. Underground, combined with a mobile lifting device
3. Underground, combined with a cask transporter (e.g., crawler)

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 100D and 125D designs, 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. The numbers in brackets {} after each design criterion indicate which of the 3 types of CTF design they apply to.

A 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). {1, 2}

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). {1, 2}

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 100D and 125D designs would include the mating device and no transfer lid). {1, 2, 3}

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 100D and 125D designs would include the mating device and no transfer lid). {1, 2, 3}

e. MPC transfer between the HI-TRAC transfer cask and the HI-STORM overpack. {1, 2, 3}

f. MPC transfer between the HI-TRAC transfer cask and the HI-STAR 100 overpack. {1, 2, 3}

ii. Other Functional Requirements:

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

a. Devices and areas to support installation and removal of the HI-STORM overpack lid. {1, 2, 3}

b. Devices and areas to support installation and removal of the HI-STORM 100 overpack vent shield block inserts. {1, 2, 3}

c. Devices and areas to support installation and removal of the HI-STAR 100 closure plate. {1, 2, 3}

d. Devices and areas to support installation and removal of the HI-STAR 100 transfer collar. {1, 2, 3}

e. Features to support positioning and alignment of the HI-STORM overpack and the HI-TRAC transfer cask. {1, 2, 3}

f. Features to support positioning and alignment of the HI-STAR 100 overpack and the HI-TRAC transfer cask. {1, 2, 3}

g. Areas to support jacking of a loaded HI-STORM overpack for insertion of a translocation device underneath. {1, 2, 3}

h. Devices and areas to support placement of an empty MPC in the HI-TRAC transfer cask or HI-STAR 100 overpack. {1, 2, 3}

i. Devices and areas to support receipt inspection of the MPC, HI-TRAC transfer cask, HI-STORM overpack, and HI-STAR overpack. {1, 2, 3}

j. Devices and areas to support installation and removal of the HI-STORM mating device (HI-TRAC 100D and 125D only). {1, 2, 3}

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 typical terms not defined elsewhere in this FSAR and used in this specification are provided below. Not all parts defined in this paragraph apply to all CTF designs.

- 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. {1, 2, 3}

- 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). {1}

- 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.[†] {1, 2, 3}

- Lifter Mount: A beam-like structure (part of the CTF structure) that supports the HI-TRAC and MPC lifter(s). {1}

- 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. {1}

- Mobile lifting devices: A mobile lifting device is a device defined in ASME B30.5-1994, Mobile and Locomotive Cranes. A mobile lifting device 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. {2}

- 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. {1}

- Pier: The portion of the reinforced concrete foundation which projects above the concrete floor of the CTF. {1}

- 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 and/or is designed in accordance with ANSI N14.6 and employs redundant drop protection features. {1, 2, 3}

- 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. {1,2}

- Vertical Cask Transporter: A device which is capable of performing the CTF functions as well as transporting the transfer

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

cask and overpack to and from the CTF. A vertical cask transporter may be used in lieu of the CTF structure, HI-TRAC lifter, and/or an MPC lifter provided all requirements set forth in this subsection are satisfied. {3}

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 the Holtec quality assurance program. {1,2,3}

B. Environmental and Design Conditions

i. Lowest Service Temperature (LST): Unless otherwise specified, the LST for the CTF is 0°F (consistent with the specification for the HI-TRAC transfer cask in Subsection 3.1.2.3). Based on its local meteorological data, a host nuclear site may use a higher LST value for the Cask Transfer Facility.{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. {1}

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 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 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.

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 analysis:

- The missile has a planform area of 20 sq. ft. and impact

force characteristics consistent with the HI-TRAC missile impact analysis.

- The large missile can strike the CTF 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 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 and, therefore, need not be considered. {1, 2, 3}

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. {1, 2, 3}

v. Lightning: Meteorological data for the region surrounding the ISFSI site shall be used to specify the applicable lightning input to the CTF for personnel safety evaluation purposes. {1, 2, 3}

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, then such a loading, with due consideration of the short duration of CTF operations, shall be defined for the CTF. {1, 2, 3}

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. {1, 2, 3}

viii. Design Temperature: Unless a lower value can be justified for a specific site, all material properties used in the stress analysis of the CTF structure shall utilize a reference design temperature of 150°F. {1, 2, 3}

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*.{1, 2, 3}

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:

a. Mobile lifting devices at the CTF shall conform to the guidelines of Section 5.1.1 of NUREG-0612 with the exception that mobile lifting devices 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 lifting device used shall have a minimum safety factor of two over the allowable load table for the lifting device 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.{2}

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

c. Vertical cask transporters shall be designed in accordance with ANSI N14.6 and shall employ redundant drop protection features. {3}

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. {1}

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. {1}

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. {1}

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. {1}

e. The design of the lifting system shall ensure that the lift platform (or lift frame) is held essentially horizontal at all times and that the symmetrically situated axial members are symmetrically loaded. {1,3}

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. {1, 2, 3}

g. The HI-TRAC lifter and MPC lifter shall possess design features to avoid excessive side-sway of the payload during lifting operations. {1, 2, 3}

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. {1, 2, 3}

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. {1, 2, 3}

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. {1, 2, 3}

v. Specific Requirements for Mobile Lifting Devices and Vertical Cask Transporters:

A mobile lifting device, 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). {2}

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 lifted load does not exceed 110% of the loaded MPC weight.{2,3}

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 developing a thru-wall breach resulting in loss of confinement or experiencing a deceleration in excess of its design basis deceleration specified in this FSAR.{2}

vi. Lift Height Limitation: The HI-TRAC lift heights shall be governed by the Technical Specifications.{1,2,3}

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. {1, 2, 3}

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). {1}

E. Materials and Failure Modes

i. Acceptable Materials and Material Properties: All materials used in the design of the CTF shall be ASME or ASTM approved or equal, consistent with the ITS category of the part (see discussion in subsection 2.I.0). Reinforced concrete, if used, shall comply with the provisions of ACI 318 (89). The material property and allowable stress values for all steel structures 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. {1, 2, 3}

ii. Brittle Fracture: All materials used for structural components of the CTF structure and the lift platform, designated as primary load bearing, shall meet the fracture toughness requirements for ASME Section III, Subsection NF, and Class 3 structures (NF-2300). If required, Charpy impact test temperature shall be equal to the designated LST or lower. {1, 2, 3}

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. {1, 2, 3}

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. {1, 2, 3}

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). {1, 2, 3}

G. Miscellaneous Components

Hoist rings, turnbuckles, slings, and other appurtenances which are in the load path during heavy load handling at the CTF shall have enhanced margin of safety per ANSI N14.6 or be single-failure-proof. {1, 2, 3}

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. {1}

I. Foundation

The design of the CTF structure foundation and piers, including load combinations, shall be in accordance with ACI-318(89). {1}

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.{1,2}

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. {1}

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). {1}

iii. Exterior aprons shall be of a material that will not be damaged by the vertical cask crawler tracks, if a crawler is used. {1}

L. Facility Floor

i. The facility floor shall be sufficiently flat to allow optimum handling of casks with a translocation device. {1}

ii. Any floor penetrations, in areas where translocation device operations may occur, shall be equipped with flush inserts. {1}

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. {1}

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. {1, 2, 3}

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. {1, 2, 3}

iii. The design of the connector brackets shall ensure that the HI-TRAC transfer cask is fully secured against slippage during MPC transfer operations. {1, 2, 3}

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). If the HI-STORM/HI-TRAC stack is laterally supported such that the fundamental natural frequency of the beam mode vibration of the stack is in the rigid range of the horizontal acceleration time histories (or the corresponding response spectra), then the dynamic time history solution converges to the static solution using the ZPA (see Glossary, Table 1.0.1) of the corresponding time history. In such a case, a time history analysis or a static analysis may be performed since the maximum predicted responses from both solutions are identical. {1, 2, 3}

O. Design Life

The Cask Transfer Facility shall be constructed to have a minimum design life of 40 years.{1}

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]. {1}

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. {1, 2, 3}

R. Documentation Requirements

i. O&M Manual: An Operations and Maintenance Manual shall be prepared which contains, at minimum, the following items of information: {1, 2, 3}

- Maintenance Drawings
- Operating Procedures

ii. Design Report: if required by the safety classification, 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 the Holtec QA program. {1, 2, 3}

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 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 or 2.1.16, as applicable.

b. Prevention of fresh water entering the MPC internals.

Administrative controls 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 72.106 for dose at the controlled area boundary. 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 permitted for storage as described in Section 2.1.9. 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 Section 2.1.9.

- No single cask will contain fuel and non-fuel hardware at the limiting burnups and cooling times allowed by Section 2.1.9.

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 A design objective for the maximum average radial surface dose rate has been established as 300 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 175 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, several transfer cask designs are provided (i.e., HI-TRAC 125, HI-TRAC 100, HI-TRAC 100D and HI-TRAC 125D). The two 125 ton HI-TRAC provide better shielding than the HI-TRAC 100D and 125D 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.

2.3.5.3 Radiological Alarm System

There are no credible events that could result in release of radioactive materials or increases in direct radiation above the requirements of 10CFR72.106.

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 $\geq$ 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.

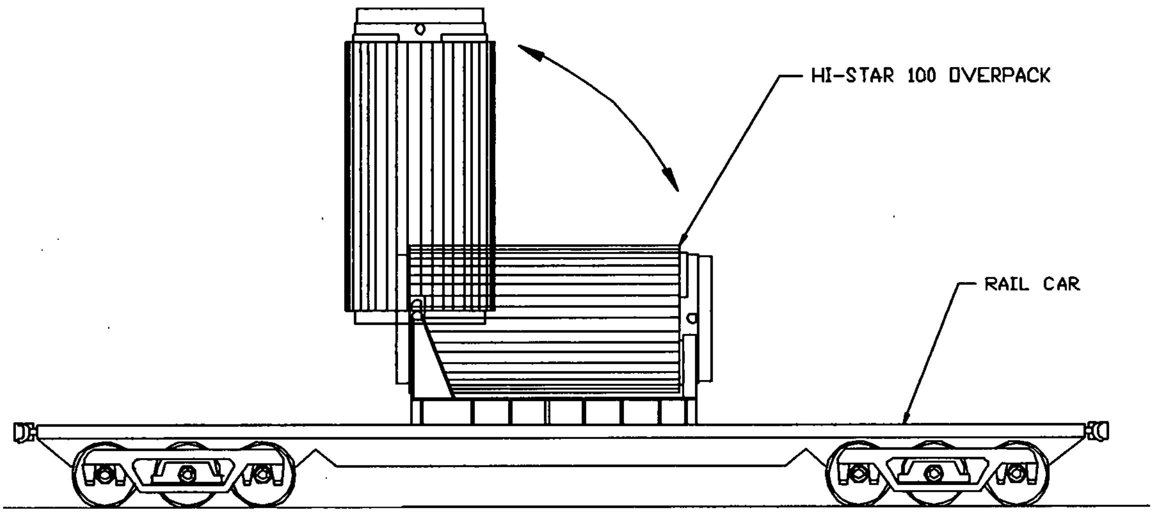

FIGURE 2.3.1; HI-STAR 100 UPENDING AND DOWNENDING ON A RAIL CAR

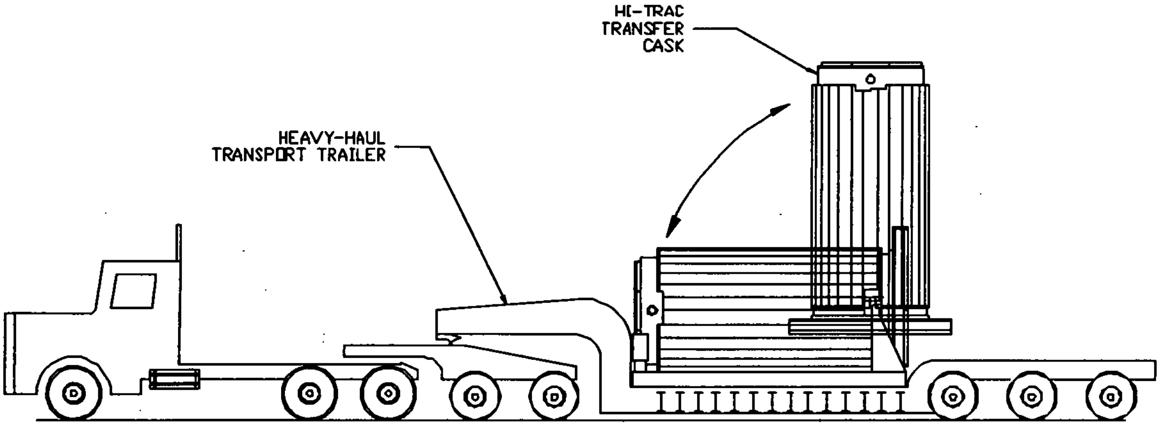

FIGURE 2.3.2; HI-TRAC UPENDING AND DOWNENDING ON A HEAVY-HAUL TRANSPORT TRAILER

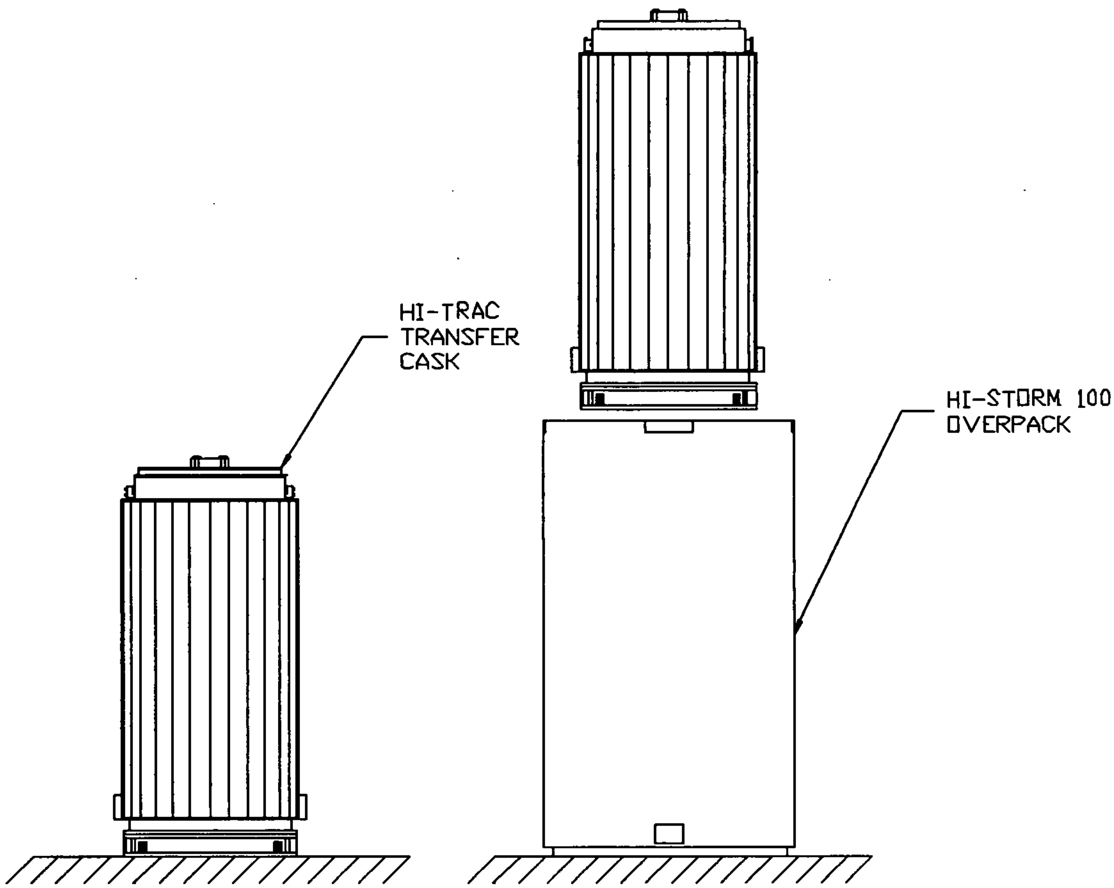

FIGURE 2.3.3; HI-TRAC PLACEMENT ON HI-STORM 100 FOR MPC TRANSFER OPERATIONS

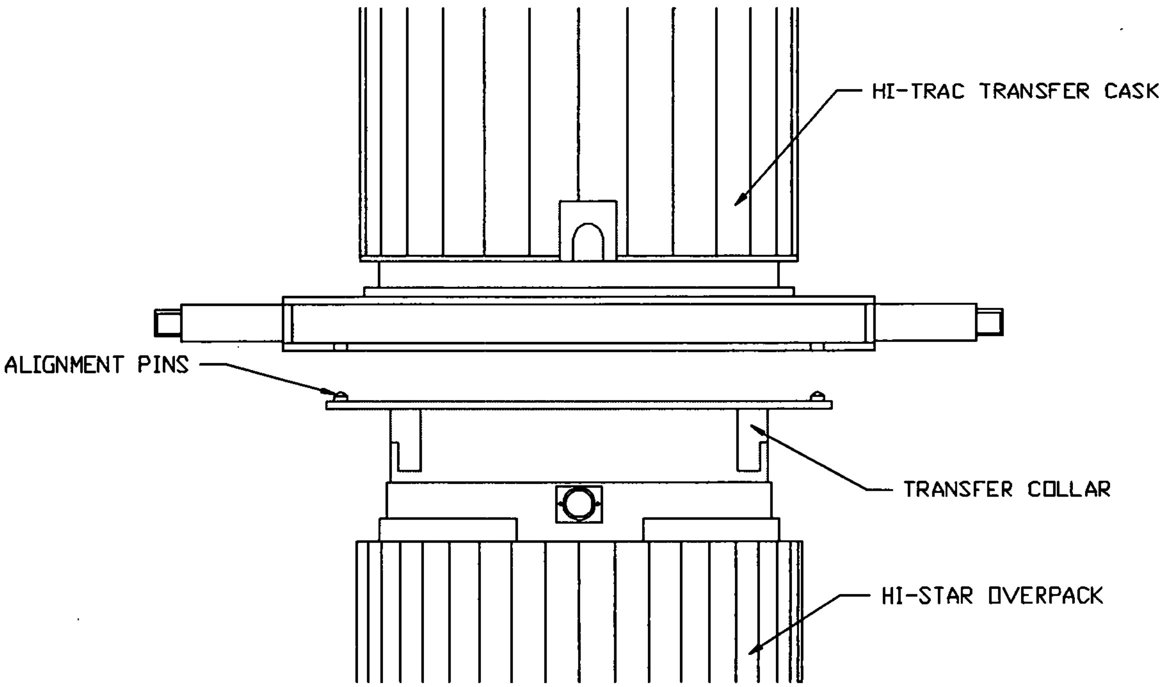

FIGURE 2.3.4; HI-TRAC PLACEMENT ON HI-STAR 100 FOR MPC TRANSFER OPERATIONS

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 internment 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 in areas where they could be exposed to spent fuel pool water or the ambient environment. 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, 4.8% enrichment, 70,000 MWD/MTU, and 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 68,200 MWD/MTU.

- Material quantities based on the drawings 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	6.82e-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

2.5 REGULATORY COMPLIANCE

Chapter 2 provides the principal design criteria related to structures, systems, and components important to safety. These criteria include specifications regarding the fuel, as well as, external conditions that may exist in the operating environment during normal and off-normal operations, accident conditions, and natural phenomena events. The chapter has been written to provide sufficient information to allow verification of compliance with 10CFR72, NUREG-1536, and Regulatory Guide 3.61. A more detailed evaluation of the design criteria and an assessment of compliance with those criteria is provided in Chapters 3 through 13.

2.6 REFERENCES

[2.0.1] American Concrete Institute, "Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)".

[2.0.2] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan†

[2.0.3] Deleted.

[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," USNRC Interim Staff Guidance-11, Revision 3, November 17, 2003.

[2.0.9] USNRC Memorandum from Christopher L. Brown to M. Wayne Hodges, "Scoping Calculations for Cladding Hoop Stresses in Low Burnup Fuel," dated January 29, 2004.

[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.

† The 1997 edition of ACI-349 is specified for embedment design for deployment of the anchored HI-STORM 100A and HI-STORM 100SA.

[2.1.4] Deleted.

[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.

[2.1.7] S.E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks", SAND-89-0018, Sandia National Laboratory, Oct., 1989.

[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.

[2.2.3] ANSI N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More", June 1993.

[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." B.W. Wilson, PG&E (September 1985, Revision 1).

[2.2.12] D. Peckner and I.M. Bernstein, "Handbook of Stainless Steels," McGraw-Hill Book Company, 1977.

[2.2.13] "Nuclear Systems Materials Handbook," Oak Ridge National Laboratory, TID 26666, Volume 1.

[2.2.14] Deleted.

[2.2.15] Deleted.

[2.2.16] Crane Manufacturer's Association of America (CMAA), Specification #70, 1988, Section 3.3.

APPENDIX 2.A

GENERAL DESIGN AND CONSTRUCTION REQUIREMENTS FOR THE ISFSI PAD FOR HI-STORM 100A

2.A.1 General Comments

As stated in Section 2.0.4, an ISFSI slab that anchors a spent fuel storage cask should be classified as "important to safety." This classification of the slab follows from the provisions of 10CFR72, which require that the cask system retain its capacity to store spent nuclear fuel in a safe configuration subsequent to a seismic or other environmental event. Since the slab for anchored HI-STORM deployment is designated as ITS, the licensee is required to determine whether the reactor site parameters, including earthquake intensity and large missiles, are enveloped by the cask design bases. The intent of the regulatory criteria is to ensure that the slab meets all interface requirements of the cask design and the geotechnical characteristics of the ISFSI site.

This appendix provides general requirements for design and construction of the ISFSI concrete pad as an ITS structure, and also establishes the framework for ensuring that the ISFSI design bases are clearly articulated. The detailed design of the ISFSI pad for anchored HI-STORM deployment shall comply with the technical provisions set forth in this appendix.

2.A.2 General Requirements for ISFSI Pad

1. Consistent with the provisions of NUREG-1567 [2.0.6], all concrete work shall comply with the requirements of ACI-349-85 [2.0.2].

2. All reinforcing steel shall be manufactured from high strength billet steel conforming to ASTM designation A615 Grade 60.

3. The ISFSI owner shall develop appropriate mixing, pouring, reinforcing steel placement, curing, testing, and documentation procedures to ensure that all provisions of ACI 349-85 [2.0.2] are met.

4. The placement, depth, and design and construction of the slab shall take into account the depth of the frost line at the ISFSI location. The casks transmit a very small amount of heat into the cask pad through conduction. The American Concrete Institute guidelines on reinforced concrete design of ground level slabs to minimize thermal and shrinkage induced cracking shall be followed.

5. General Requirements for Steel Embedment: The steel embedment, excluding the pre-tensioned anchorage studs, is required to follow the provisions stipulated in ACI 349-85 [2.0.2], Appendix B "Steel Embedment" and the associated Commentary on Appendix B, as applicable. Later editions of this Code may be used provided a written reconciliation is performed. An example of one acceptable embedment configuration is provided in Figure 2.A.1. Site-specific embedment designs may vary from this example, depending on the geotechnical characteristics of the site-specific foundation. The embedment designer shall consider any current, relevant test data in designing the pad embedment for HI-STORM 100A and HI-STORM 100SA.

6. The ISFSI owner shall ensure that pad design analyses, using interface loads provided in this report, demonstrate that all structural requirements of NUREG-1567 and ACI-349-85 are satisfied.

7. Unless the load handling device is designed in accordance with ANSI N14.6 and incorporates redundant drop protection features, the ISFSI owner shall ensure that a permissible cask carry height is computed for the site-specific pad/foundation configuration such that the design basis deceleration set forth in this FSAR are not exceeded in the event of a handling accident involving a vertical drop.

8. The ISFSI owner shall ensure that the pad/foundation configuration provides sufficient safety margins for overall kinematic stability of the cask/pad/foundation assemblage.

9. The ISFSI owner shall ensure that the site-specific seismic inputs, established at the top surface of the ISFSI pad, are bounded by the seismic inputs used as the design basis for the attachment components. If required, the ISFSI owner shall perform additional analyses to ensure that the site-specific seismic event or durations greater than the design basis event duration analyzed in this report, do not produce a system response leading to structural safety factors (defined as allowable stress (load) divided by calculated stress (load)) less than 1.0. Table 2.0.5 and Table 2.2.8 provide the limiting values of ZPAs in the three orthogonal directions that must not be exceeded at an ISFSI site (on the pad top surface) to comply with the general CoC for the HI-STORM 100A (and 100SA) System.

10. An ISFSI pad used to support anchored HI-STORM overpacks, unlike the case of free standing overpacks, may experience tensile (vertically upward) anchorage forces in addition to compression loads. The reinforcing steel (pattern and quantity) must be selected to meet the demands of the anchorage forces under seismic and other environmental conditions that involve destabilizing loadings (such as the large tornado missile defined in this FSAR).

2.A.3 Steel Embedment for Anchored Casks

Figure 2.A.1 shows a typical fastening arrangement for the HI-STORM 100A System. The details of the rebars in the pad (which are influenced by the geotechnical characteristics of the foundation and its connection to the underlying continuum) are not shown in Figure 2.A.1. Representative dimensions of the embedment and anchorage system are provided in Table 2.A.1.

The embedment detail illustrated in Figure 2.A.1 is designed to resist a load equal to the ASME Code, Section III Appendix F Level D load capacity of the cask anchor studs. The figure does not show the additional reinforcement required to ensure that tensile cracking of concrete is inhibited (see Figure B-4 in the Commentary ACI-349R-97) as this depends on the depth chosen for the ITS ISFSI pad concrete. The ACI Code contemplates ductile failure of the embedment steel and requires that the ultimate load capacity of the steel embedment be less than the limit pullout strength of the concrete surrounding the embedment that resists the load transferred from the cask anchor stud. If this criterion cannot be assured, then additional reinforcement must be added to inhibit concrete cracking (per Subsection B.4.4 of Appendix B of ACI-349-97).

The anchor stud receptacle described in Figure 2.A.1 is configured so that the cask anchor studs (which interface with the overpack baseplate as well as the pad embedment per Table 2.0.5 and are designed in accordance with ASME Section III, Subsection NF stress limits), sits flush with the ISFSI top surface while the cask is being positioned. Thus, a translocation device such as an "air pad" (that requires a flat surface) can be used to position the HI-STORM overpack at the designated location. Subsequent to positioning of the cask, the cask anchor stud is raised, the anchor stud nut installed, and the anchor stud preload applied. The transfer of load from the cask anchor stud to the embedment is through the bearing surface of the lower head of the cask anchor stud and the upper part of the anchor stud receptacle shown in the figure. The members of the anchoring system illustrated in Figure 2.A.1, as well as other geometries developed by the ISFSI designer, must meet the following criteria:

i. The weakest structural link in the system shall be in the ductile member. In other words, the tension capacity of the anchor stud/anchor receptacle group (based on the material ultimate strengths) shall be less than the concrete pull-out strength (computed with due recognition of the rebars installed in the pad).

ii. The maximum ratio of embedment plus cask anchor stud effective tensile stiffness to the effective compressive stiffness of the embedment plus concrete shall not exceed 0.25 in order to ensure the effectiveness of the pre-load.

iii. The maximum axial stress in the cask anchor studs under normal and seismic conditions shall be governed by the provisions of ASME Section III Subsection NF (1995).

iv. The load-bearing members of the HI-STORM 100A anchorage system shall be considered important-to-safety. This includes the following components shown in Figure 2.A.1: anchor stud and nut, top ring, upper collar, anchor receptacle, and anchor ring.

For sites with lower ZPA DBE events, compared to the limiting ZPAs set down in this FSAR, the size of the anchor studs and their number can be appropriately reduced. However, the above three criteria must be satisfied in all cases.

Table 2.A.1

Typical Embedment and Anchoring Data*

Nominal diameter of the anchor stud, (inch)	2
Thickness of the embedment ring, (inch)	2
I.D. of the embedment ring, (inch)	130
Anchor receptacle: Upper Position O.D. and I.D. (inch) Lower portion O.D. and I.D. (inch)	 O.D.: 2.5 / I.D.: 2.125 (min.) O.D.: 4.875 / I.D.: 3.625 (min.)
Depth of anchor receptacle collar, d, (inch)	2.5
Free fall height of the anchor stud, h_e,(inch)	8
Representative Materials of Construction are as follows:†	
Anchor Studs:	Per Table 2.0.4
Anchor Receptacle:	Low carbon steel such as A-36, A-105
Top Ring, Upper Collar, Anchor Ring:	Low carbon steel such as A-36, SA-516-Gr. 70

* Refer to Figure 2.A.1

† The ISFSI designer shall ensure that all permanently affixed embedment parts (such as the anchor receptacle) made from materials vulnerable to deleterious environmental effects (e.g. low carbon steel) are protected through the use of suitably engineered corrosion barrier. Alternatively, the selected material of construction must be innately capable of withstanding the long term environmental conditions at the ISFSI site.

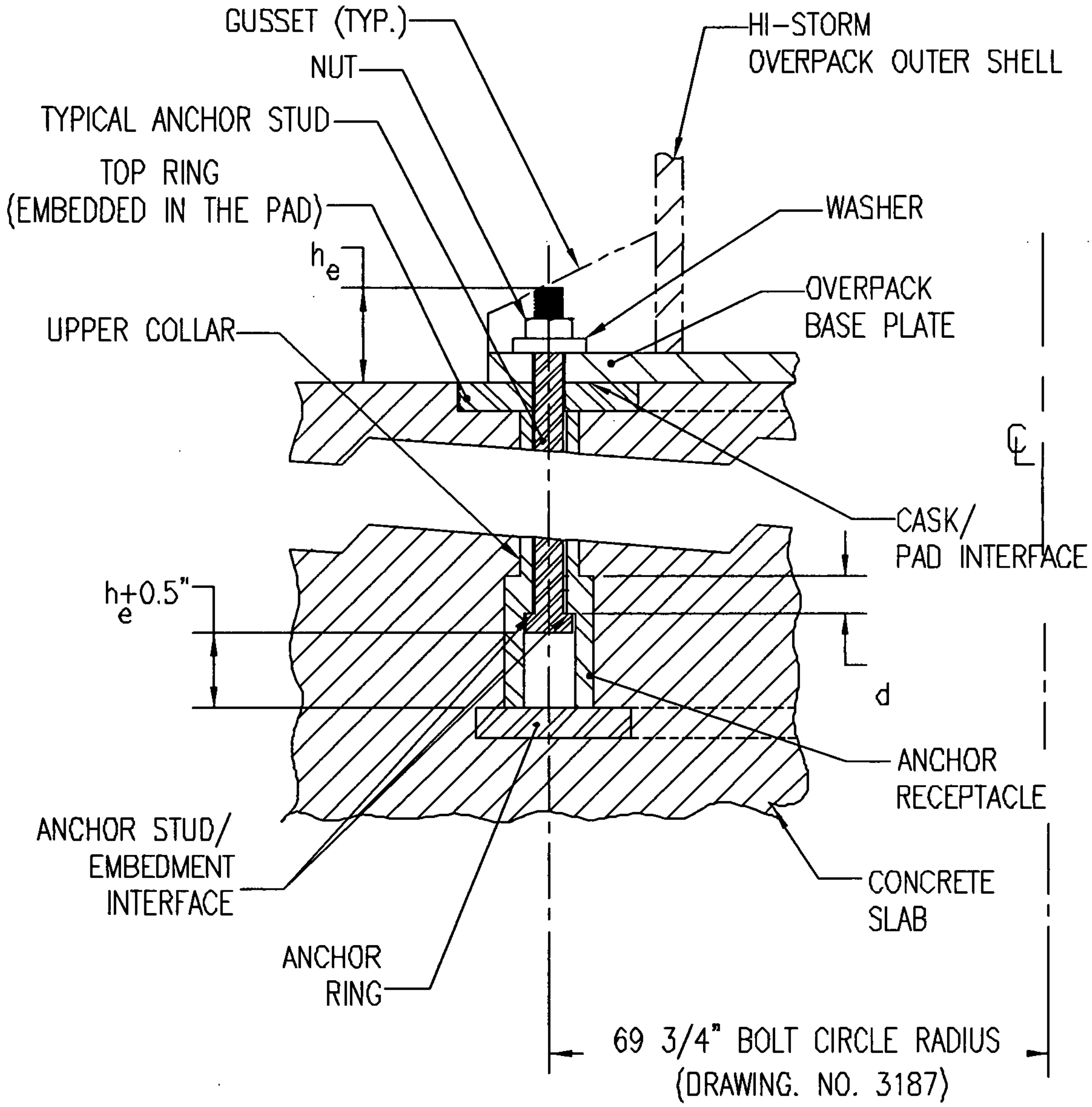

Figure 2.A.1;
Typical HI-STORM/ISFSI pad Fastening Detail

Note: Rebars in the ISFSI pad and sub-surface soil/rock continuum not shown.

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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 MPCs containing at least one high burnup fuel assembly. The FHD method of moisture removal is optional for all other MPCs.

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 $\leq$ 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 demoisturizer 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 demoisturizer 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 demoisturizer 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 gas has met the specified temperature or dew point value and duration criteria. 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 turbulated flow regime in the MPC cavity.

iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. The limit will be met if the gas temperature at the demoisturizer outlet is verified by measurement to remain ≤ 21°F for ≥ 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain ≤ 22.9°F for ≥ 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 demoisturizer module shall be configured to be introduced into its helium conditioning function <u>after</u> 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 demoisturizer 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 demoisturizer 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.

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 demoisturizer 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.

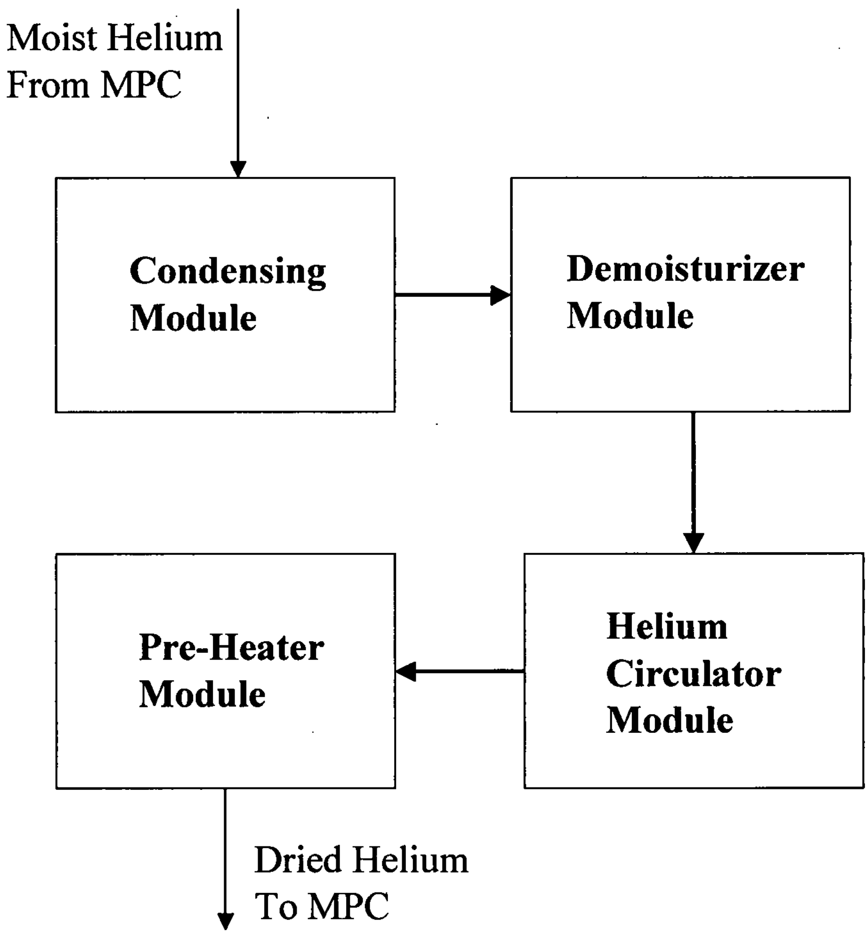


FIGURE 2.B.1: SCHEMATIC OF THE FORCED HELIUM DEHYDRATION SYSTEM

Appendix 2.C

The Supplemental Cooling System

2.C.1 Purpose

The Supplemental Cooling System (SCS) will be utilized, as necessary, to maintain the peak fuel cladding temperature below the limit set forth in Chapter 2 of the FSAR during normal short-term operations (as defined in Section 2.2).

2.C.2 General Description and Requirements

The SCS is a system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. During normal SCS operation, heat is removed by a coolant from the HI-TRAC annulus and rejected to the heat sink (ambient air). The SCS shall be designed to meet the following criteria:

(i) If the system uses water as the coolant, the system is sized to limit the coolant temperature to below 180°F under steady-state conditions for the design basis heat load at an ambient air temperature of 110°F. Active components (i.e., pump or air-cooler fan) are powered by electric motors with a backup power supply for uninterrupted operation.

(ii) The system will utilize a contamination-free fluid medium in contact with the external surfaces of the MPC and inside surfaces of the HI -TRAC transfer cask to minimize corrosion. Figure 2.C.1 shows a typical P&ID for a SCS.

(iii) The number of active components in the SCS will be minimized.

(iv) All passive components such as tubular heat exchangers, manually operated valves and fittings shall be designed to applicable standards (TEMA, ANSI).

2.C.3 Thermal/Hydraulic Design Criteria

(i) The heat dissipation capacity of the SCS shall be equal to or greater than the minimum necessary to ensure that the peak cladding temperature of High-Burnup fuel assemblies is below the ISG-11, Rev. 3 limit of 400°C (752°F). All heat transfer surfaces in any heat exchangers shall be assumed to be fouled to the maximum limits specified in a widely used heat exchange equipment standard such as the Standards of Tubular Exchanger Manufacturers Association.

(ii) The coolant utilized to extract heat from the MPC shall be either high purity water or air. Anti-freeze may be used to prevent water from freezing if warranted by operating conditions.

2.C.4 Mechanical Requirements

(i) All pressure boundaries (as defined in the ASME Boiler and Pressure Vessel Code, Section VIII Division 1) shall have pressure ratings that are greater than the maximum system operating pressure by at least 15 psi.

(ii) All ASME Code components shall comply with Section VIII Division 1 of the ASME Boiler and Pressure Vessel Code.

(iii) Prohibited Materials

The following materials will not be in contact with the system coolant in the SCS.

- Lead
- Mercury
- Sulfur
- Saran
- Silastic L8-53
- Cadmium
- Tin
- Antimony
- Bismuth
- Mischmetal
- Neoprene or similar gasket materials made of halogen containing elastomers
- Phosphorus
- Zinc
- Copper and Copper Alloys
- Rubber-bonded asbestos
- Nylon
- Magnesium oxide (e.g., insulation)
- Materials that contain halogens in amounts exceeding 75 ppm

(iv) Not Used.

(v) The SCS skid shall be equipped with appropriate lifting lugs to permit its handling by the plant's lifting devices in full compliance with NUREG-0612 provisions.

2.C.5 Regulatory Requirements

The SCS is classified as Important-to-Safety Category B.

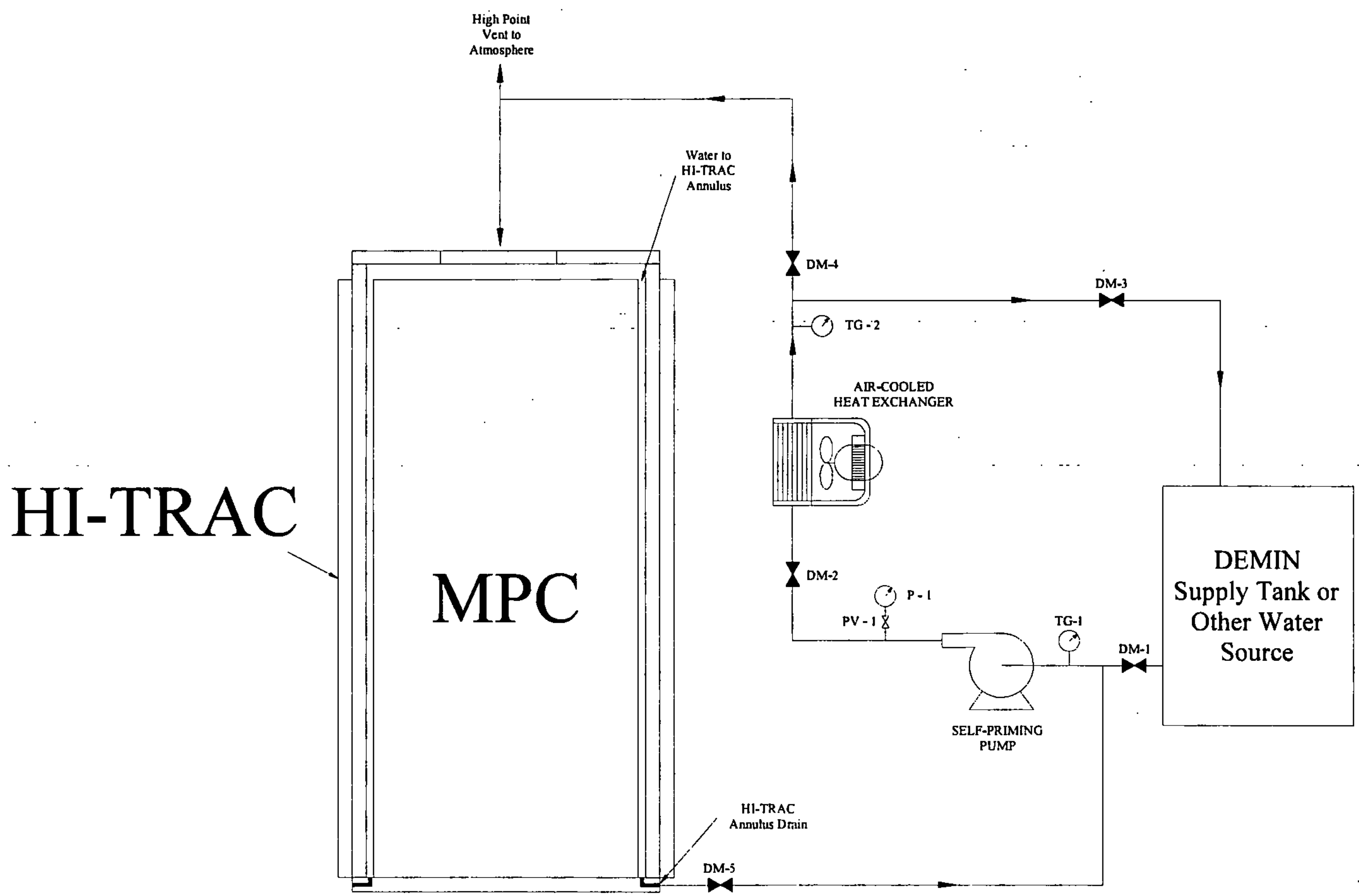


FIGURE 2.C.1: SUPPLEMENTAL COOLING SYSTEM EXAMPLE P&I DIAGRAM

SUPPLEMENT 2.I
PRINCIPAL DESIGN CRITERIA FOR THE HI-STORM 100U SYSTEM

2.I.0 OVERVIEW OF THE PRINCIPAL DESIGN CRITERIA

General

A description of the HI-STORM 100U VVM is provided in Supplement 1.I. Because the HI-STORM 100U System uses the same MPCs, transfer cask, and ancillary equipment as the aboveground systems, the design criteria presented in Table 2.0.1 for the MPC, and Table 2.0.3 for the HI-TRAC transfer cask provide the basis for setting down the applicable criteria in this supplement with due recognition of the advances in the analysis methodologies over the past decade. The applicable loads, the affected parts under each loading condition, and the applicable structural acceptance criteria are compiled in this supplement to provide a complete framework for the required qualifying analyses in Supplement 3.I. Information consistent with the regulatory requirements related to shielding, thermal performance, confinement, radiological, and operational considerations is also provided. The licensing drawing of the HI-STORM 100 System 100U VVM, provided in Section 1.I.5, along with Table 2.I.2 herein provide information on all necessary critical characteristics to define the "100U" storage system. The constituents of the HI-STORM 100U ISFSI fall into two broad categories, namely:

i. VVM components
ii. ISFSI structures

The safety analyses documented in Supplement 3.I address both the VVM components and the ISFSI Structures. The ISFSI Structures consist of:

i. The Support Foundation Pad (SFP)
ii. The Top Surface Pad (TSP),
iii. The VVM Interface Pad (VIP), and
iv. The Retaining Wall, if used at the site.

Figure 2.I.5 depicts the subgrade and undergrade nomenclature for the ISFSI. The density and shear wave velocities of these are given in Table 2.I.2. The following are a description of the areas shown in Figure 2.I.5 which contribute to the analysis of the ISFSI.

i. Space A is the lateral subgrade space, in and around the VVMs, which may be excavated and refilled with engineered fill.
ii. Space B is the lateral subgrade that extends by the amount W around the ISFSI where W is the characteristic dimension of the ISFSI.
iii. Space C is the undergrade below the SFP extending 100 feet below the bottom of the SFP.
iv. Space D is the undergrade surrounding Space C extending 100 feet below the bottom of the SFP.

Structural

All required information on the design bases and criteria for the VVM are compiled in this supplement to fulfill the requirements of 10CFR72.24(c)(3) and 72.44(d). Table 2.I.1 contains a detailed listing of the information and its location in this FSAR corresponding to each relevant requirement in 10CFR72 with reference to the VVM. The VVM structure described in Supplement 1.I is designed for all applicable normal, off-normal, extreme environmental phenomena, and accident condition loadings pursuant to 10CFR72.24(c), 72.122(b) and 72.122(c).

The subgrade surrounding the VVM, the SFP on which the VVM is founded, and the VIP are categorized as "interfacing SSCs", while the TSP and the retaining wall (if used) are categorized as "proximate structures". All of these structures are classified as important-to-safety (ITS) (see Table 2.I.8) and are included in the analyses in Supplement 3.I, and in other supplements as applicable. Table 2.I.2 defines the essential design requirements for these structures. ACI-318 (2005) [2.I.5] is specified as the governing code for the design qualification of the SFP, the VIP, the TSP, and the retaining wall (if used) using the load combinations specified in Table 2.I.11. The seismic qualification of the storage system is performed in Supplement 3.I using the design data of the ISFSI.

In addition to defining the design details of the ISFSI components and structures, the material types used in the VVM are also identified in Table 2.I.8. Material designations used by ASTM and ASME for various product forms are, however, subject to change as these material certifying organizations publish periodic updates of their standards. Material designations adopted by the International Standards Organization (ISO) also affect the type of steels and steel alloys available from suppliers around the world. Therefore, it is necessary to provide for the ability in this FSAR to substitute materials with equivalent materials in the manufacture of the equipment governed by this FSAR.

As defined in this FSAR, the term "Equivalent Material" has a specific meaning: Equivalent materials are those that can be substituted for each other without adversely affecting the safety function of the SSC (system, structure, and component) in which the substitution is made. Substitution by an equivalent material can be made after the equivalence in accordance with the provisions of this FSAR has been established.

The concept of material equivalence explained above has been previously used in this FSAR to qualify four different austenitic stainless steel alloys (ASME SA240 Types 304, 304LN, 316, and 316LN) to serve as candidate MPC basket materials.

The equivalence of materials is directly tied to the notion of *critical characteristics*. A critical characteristic of a material is a material property whose value must be specified and controlled to ensure an SSC will render its intended function. The numerical value of the critical characteristic invariably enters in the safety evaluation of an SSC and therefore its range must be guaranteed. To ensure that the safety calculation is not adversely affected, material properties such as Yield Strength, Ultimate Strength and Elongation must be specified as *minimum* guaranteed values in

VVM Components. However, there are certain properties where both minimum and maximum acceptable values are required. In this category lie VVM Component properties such as specific gravity and thermal expansion coefficient.

Table 2.I.10 lists the array of material properties typically required in safety evaluation of an SSC in dry storage and transport applications. The required value of each applicable property, guided by the safety evaluation defines the critical characteristics of the material. The subset of applicable properties for a material depends on the role played by the material. The role of a material in the SSC is divided into three categories:

Type	Technical Area of Applicability
S	Those needed to ensure structural compliance
T	Those needed to ensure compliance with thermal (temperature limits)
R	Those needed to ensure radiation (criticality and shielding) compliance

The material properties listed in Table 2.I.10 are the ones that may apply in a dry storage or transport application.

To summarize, the following procedure shall be used to establish acceptable equivalent materials for a particular application.

Criterion i: Functional Adequacy:
Evaluate the guaranteed critical characteristics of the equivalent material against the values required to be used in safety evaluations. The required values of each critical characteristic must be met by the minimum (or maximum) guaranteed values (MGVs of the selected material).

Criterion ii: Chemical and Environmental Compliance:
Perform the necessary evaluations and analyses to ensure the candidate material will not excessively corrode or otherwise degrade in the operating environment.

A material from another designation regime that meets Criteria (i) and (ii) above is deemed to be an acceptable material, and hence, equivalent to the candidate material.

Equivalent materials as an alternative to the U.S. national standards materials (e.g., ASME, ASTM, ANSI) shall not be used for the Confinement Boundary materials. For other ITS materials, recourse to equivalent materials shall be made only in the extenuating circumstances where the designated material in this FSAR is not readily available.

As can be ascertained from its definition in the glossary, the *critical characteristics* of the material used in a subcomponent depend on its function. The Closure Lid, for example, serves as a shielding device and as a physical barrier to protect the MPC against loadings under all service conditions, including the Extreme Environmental phenomena. Therefore, the critical characteristics of steel used in the lid are its strength (yield and ultimate), ductility, and fracture resistance.

The appropriate critical characteristics for structural components of the VVM, therefore, are:

i. Material yield strength, σ_y
ii. Material ultimate strength, σ_u
iii. Elongation, ε
iv. Charpy impact strength at the lowest service temperature for the part, C_i

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than the minimum values used in the qualifying analyses in this FSAR. The above *critical characteristics* apply to all materials used in the structural parts of the CEC. Table 2.I.9 provides guidance for the critical characteristics associated with the steels used in the VVM.

In the event that one or more of the *critical characteristics* of the replacement material is slightly lower than the original material, then the use of the §72.48 process is necessary to ensure that all regulatory predicates for the material substitution are fully satisfied.

In addition to the design configuration, the maximum magnitude of Design Basis Earthquake for the "100U" ISFSI is also specified in this FSAR. A three-dimensional non-linear time-history solution procedure implemented on LS-DYNA is used in Supplement 3.I to qualify the ISFSI including the storage system. This same three-dimensional non-linear time-history solution procedure may be used to perform safety evaluation under 10CFR72.212 at a host site, as indicated in Paragraph 2.I.6(v). Likewise, the loadings from the extreme environmental phenomena, defined in the main body of Chapter 2, are considered in Supplement 3.I. Site specific loadings that deviate from those analyzed in Supplement 3.I are subject to 72.212 safety evaluations in the manner of all HI-STORM models.

To serve their intended functions, the CEC and Closure Lid shall ensure physical protection, biological shielding, and allow the retrieval of the MPC under all conditions of storage (10 CFR 72.122(l)). Because the VVM is an in-ground structure, drops and tip-over of the VVM are not credible events and, therefore, do not warrant analysis. The load cases germane to establishing the structural adequacy of the VVM pursuant to 10 CFR 72.24(c) are compiled in Table 2.I.5. The physical characteristics of the MPCs intended for storage in the VVMs are presented in the main body of Chapter 1.

The design bases and criteria provided in this supplement are intended to quantify the safety margins in the VVM design with respect to all applicable loadings that follow from the provisions of 10CFR72.24(c)(3), §72.122(b) and §72.122(c).

Thermal

The engineered thermal performance of the HI-STORM 100U system is essentially equivalent to its aboveground counterparts under quiescent conditions. Ambient air enters from a circumferential opening provided in the Closure Lid. The intake air flows downward through an annular passage or

intake plenum formed between the CEC and the Divider Shell. At the bottom of the intake plenum the air turns inwards through openings or cutouts provided in the Divider Shell bottom and rises up through an annular gap formed between the MPC and the Divider Shell. Heat is dissipated from the MPC to this upward rising column of air. The rising air column enters the curved flow passages engineered in the Closure Lid and exhausts from the top through a large central opening (see Figure 1.I.4). To minimize the heating of the downward flowing inlet air and the upward column of heated air, the divider shell is insulated on its outside surface. The *critical characteristic* of the insulation is specified in Table 2.I.1. This thermal insulation material is required to meet the service conditions (temperature and humidity) for the design life of the VVM. Because the thermal performance of the HI-STORM 100U relies on buoyancy-driven convection of air and because of the relative proximity of the inlet and outlet vents to each other, the effect of wind on its thermal performance is also considered.

The allowable long-term and short term section-average temperature limits for concrete (used in the Closure Lid) are established in Appendix 1.D. Section-average temperature limits for structural steel in the VVM are provided in Table 2.I.8.

The VVM is designed for extreme cold conditions, as discussed in Subsection 2.2.2.2. The safety of structural steel material used for the VVM from brittle fracture is discussed in Subsection 3.1.2.3.

The VVM is designed to reject the maximum allowable heat load as defined below in a reliable and testable manner consistent with its important-to-safety designation (10CFR72.128(a)(4)).

The maximum permissible HI-STORM 100U heat load Q(X) is a function of the parameter "X" defined as the ratio of the maximum permissible inner region assembly heat load q_1, and outer region assembly heat load q_2. The inner and outer fuel storage regions are defined in Table 2.1.27. The functional relationship Q(X) is presented below:

$$Q(X) = 2 \cdot \alpha \cdot Q_d / (1 + X^y) \text{ where } y = 0.23/X^{0.1}$$

Q_d is the maximum heat load where X=1 (34kW) and α is a penalty factor for underground storage discussed in Supplement 4.I.

Shielding

The off-site dose for normal operating conditions to any real individual beyond 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 on site-specific conditions (e.g., the ISFSI design and proximity to the controlled area boundary, and the number and arrangement of loaded storage casks at the ISFSI), the determination and comparison of ISFSI doses to these limits are necessarily site-specific. Dose rates from the HI-STORM 100U System are provided in Supplement 5.I. 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 for the specific VVM array in accordance with 10CFR72.212.

The VVM is designed to limit the dose rates for all MPCs to ALARA values. The VVM is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The underground location of the VVM significantly reduces the radiation from the ISFSI at the site boundary compared to an aboveground cask. The calculated VVM dose rates are discussed in Supplement 5.I, which also discusses dose rates during site construction next to an operating ISFSI.

The dose rate calculations presented in Supplement 5.I conservatively use a lower density for the subgrade than is specified in Table 2.I.2. For dose rate calculation at a particular ISFSI, the spatial average of the actual subgrade density shall be used.

Criticality

The VVM does not perform any criticality control function. The MPCs provide criticality control for all design basis normal, off-normal and postulated accident conditions, as discussed in Chapter 6.

Confinement

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

Operations

MPC preparation for storage and onsite transport of the MPC in the HI-TRAC transfer cask is the same for the VVM as for the aboveground overpack designs. The cask transporter is used to move the loaded transfer cask to the ISFSI and to transfer the MPC into the VVM. Generic operating instructions for the use of the HI-STORM 100U System that parallel those for the aboveground overpack are provided in Supplement 8.I.

Acceptance Tests and Maintenance

The fabrication acceptance bases and maintenance program to be applied to the VVM are described in Supplement 9.I. Application of these requirements will assure that the VVM is fabricated and maintained in a manner that satisfies the design criteria defined in this FSAR.

Decommissioning

Decommissioning considerations for the HI-STORM 100U System, including the VVM, are addressed in Section 2.I.11.

2.I.1 SPENT FUEL TO BE STORED

There is no difference in the authorized contents of the HI-STORM 100U VVM and the aboveground HI-STORM systems. The information in Section 2.1 is applicable.

2.I.2 HI-STORM 100U VVM COMPONENTS, ISFSI STRUCTURES, AND CORROSION MITIGATION MEASURES

The VVM is engineered for below-grade storage for the duration of its design life, and is designed to withstand normal, off-normal, and extreme environmental phenomena as well as accident conditions of storage with appropriate margins of safety.

As discussed in Supplement 1.I, the VVM Components are (see Figure 1.I.2):

1. The MPC Cavity Enclosure Container (CEC), and
2. The Closure Lid

The CEC is comprised of the following subcomponents:

1. Container Shell (a cylindrical enclosure shell)
2. Bottom Plate
3. Container Flange (a top ring flange)
4. Divider Shell with insulation and MPC Guides
5. MPC bearing pads

The Closure Lid consists of:

1. The integral steel weldment (filled with shielding concrete), and
2. The removable vent screen assemblies (inlet and outlet).

The structural limit criteria imposed on the VVM Components are selected to comply with the provisions of 10CFR72, with an embedded large margin of safety. Table 2.I.1 provides the principal design criteria applicable to the VVM Components. The specifications of the materials of construction for the load bearing and non-load bearing parts are provided in Table 2.I.8 along with their maximum permissible temperature for different conditions of storage.

The interfacing SSCs, the proximate structures, and corrosion mitigation measures germane to the design of a HI-STORM 100U ISFSI are:

i) The SFP that supports the weight of the loaded VVMs.

ii) The ISFSI pad consisting of the VIP which provides a water seepage barrier against rainwater and melting snow and also acts as a missile barrier, and the TSP which serves as a water seepage barrier as well as the riding surface for the transporter.

iii) The lateral subgrade (natural or engineered fill) surrounding the CEC (Space "A" in Figure 2.I.5).

iv) The impressed current cathodic protection system (ICCPS) that may be used as a corrosion mitigation measure for the CEC in accordance with the Technical Specifications.

v) The concrete encasement that may be used as a corrosion mitigation measure for the CEC in accordance with Technical Specifications. Reference is made to Figure 2.I.3 for typical concrete encasement of the CEC.

vi) The retaining wall used to protect the soil column in the Radiation Protection Space from excavation activities adjacent to the ISFSI.

Each of these is discussed below:

i. <u>The Support Foundation Pad (SFP) (Interfacing Structure)</u>

The structural requirements on the SFP are focused on providing a robust support to the CEC structure (for shear and compression), and to limit the long-term settlement of the SFP. The minimum structural design requirements on the SFP are provided in Table 2.I.2 and the licensing drawing in 1.I.5.

ACI-318(2005) is the prescribed Code for SFP design. As specified in ACI-318(2005), the applicable loads on the SFP are:

1. Dead load (from the TSP, the VIP, the loaded VVM, and the mass of soil above the SFP).
2. Live load (from the loaded vertical cask transporter bearing on the TSP).
3. Seismic load (the additional inertia load, in excess of the dead weight, live load transmitted to the SFP from the loaded VVM and the transporter under the ISFSI's DBE event).
4. Long-term differential settlement.

The load combinations for the structural analysis of the SFP pursuant to ACI-318(2005) are provided in Table 2.I.11.

Of the above loads, the effect of long-term settlement on the SFP is treated together with the Dead load. The standard approach to compute the long-term settlement is provided in [2.I.6]. This methodology, which is based on classical soil mechanics and is utilized in the structural analysis in Supplement 3.I, is summarized below.

1. Compute the total long-term settlement, "d", of the subgrade under the SFP (Space C) over the Design Life assuming that the total load "P" (modeled as a uniform pressure at the top of the subgrade) is equivalent to that produced by the SFP fully populated with loaded VVMs for the entire life using the methodology in [2.I.6].

2. Determine an "effective" elastic spring constant "K" of Space C that emulates the cumulative settlement:

$$K = P/d.$$

3. Using the spring constant computed above, which accounts for the effect of long-term settlement under static loading, an appropriate elastic modulus is defined for the soil column under the SFP. The degraded soil modulus so defined is used in the finite element model of the SFP to evaluate the pad flexure under the factored dead load.

The maximum permitted settlement of the SFP is limited to the value specified in Table 2.I.2. Remedial measures such as pilings must be used if the Table 2.I.2 limit cannot be met.

In the structural qualification of the SFP, the loading from the seismic event is computed using the dynamic elastic modulus corresponding to the minimum strain wave velocity of the subgrades specified in Table 2.I.2.

ii. <u>VVM Interface Pad (VIP) (Interfacing Structure) and Top Surface Pad (TSP) (Proximate Structure)</u>

The VIP portion of the ISFSI Pad serves no structural function in supporting the VVM. However, it girdles the Container Shell and underlies the Container Flange to form a leak tight interface, and directs water away from the CEC. The principal functions of the TSP are to provide the riding surface for the loaded transporter and also to enable rainwater to be channeled away from the storage arrays and into the site's storm drain system. The TSP is isolated from the VIP by appropriately located expansion joints to isolate the CEC from any unbalanced loads imparted by the transporter. Similarly, an expansion joint between the CEC and the VIP is incorporated to permit differential movement between the two. The licensing drawing in Section 1.I.5 provides details for the expansion joint and typical drainage and sealing details. Because the sealing is visible and accessible, re-sealing, when and if necessary, is easily accomplished. Thus, continued sealing is assured. A specific brand of sealant is noted on the expansion joint detail, but there are several equivalent* proven sealant materials commercially available that are ideal for this application and the expected ambient conditions.

In summary, the design objectives for the VIP and the TSP are: to provide a leak tight interface and to provide a sufficiently inflexible travel surface for the loaded transporter, respectively. The top surface of the VIP, as shown in the 100U licensing drawing package, also serves to keep rain water away from the VVM. The minimum structural design requirements on the VIP and TSP are provided in Table 2.I.2 and the licensing drawing in 1.I.5. The applicable loads on the TSP and VIP are:

1. Dead load (Self weight including settlement effects) (TSP and VIP)
2. Live load (Weight of a loaded cask transporter) (TSP only)

* The definition of the term "equivalent" is provided in the Glossary.

3. Seismic Load (Inertia load from the concrete pad and the transporter under the ISFSI's DBE event) (TSP only).

The applicable load combinations for the structural analysis of the VIP and TSP pursuant to ACI-318(2005) are provided in Table 2.I.11.

The effect of settlement is incorporated in the stress analysis of the TSP using the same procedure as the SFP discussed above. As in the case of the SFP stress analysis, the settlement of the TSP from Dead load (self-weight) relative to the SFP over their Design Life is computed and incorporated in the stress analysis. The maximum permissible settlement of the TSP with respect to the SFP is required to be limited to the value in Table 2.I.2.

The design of the TSP together with the lateral subgrade must also satisfy the allowable bearing capacity requirement of ACI 360R-06 [2.I.8] for slabs on grade. In particular, the total load imparted by the TSP on the lateral subgrade, including the live load and seismic load from the transporter, shall be less than 50 percent of the allowable bearing capacity thereof when the load is applied uniformly.

iii. Lateral Subgrade (Interfacing SSC)

The physical characteristics of the subgrade surrounding the VVM vary from site-to-site. Further, an ISFSI owner may elect to excavate the natural subgrade and replace it with an engineered fill of an appropriate density and composition to fulfill shielding demand. While the surrounding subgrade may not provide a structural support function to the CEC structure, as an interfacing SSC, it plays a role in the loading applied to the CEC under certain scenarios, namely:

a. during an earthquake event;
b. during movement of the cask transporter along the Top Surface Pad;
c. normal storage condition from the natural overburden or under the state of maximum soil saturation (hydraulic buoyancy).

During a seismic event, the subgrade surrounding the VVM may exert a time-varying lateral pressure loading on the Container Shell, which, in principle, may ovalize the Container Shell and possibly bend it like a beam.

During the movement of the cask transporter, which is loaded with the transfer cask (see Supplement 8.I for operational details), the vertical load of the cask transporter results in a lateral pressure on the Container Shell. Although the lateral pressure is apt to be quite small due to the physical restriction on how close to the Container Shell the transporter can ride, mandatory limits on the lateral separation and subgrade properties are necessary to ensure a design with adequate safety margins. Accordingly, the minimum average density and the minimum shear wave velocity in the lateral subgrade surrounding the VVMs have been specified in Table 2.I.2.

The soil overburden pressure on the Container Shell is the third loading category which must be evaluated. Also, the condition of maximum soil saturation applies a hydrostatic pressure on the CEC. The maximum value depends on the depth of the MPC storage cavity and the effective density of the saturated soil.

iv. <u>Impressed Current Cathodic Protection System (ICCPS) (Corrosion Mitigation Measure)</u>

If an ICCPS is required by the technical specifications, it shall be implemented in accordance with the requirements in Supplement 3.I, Subsection 3.I.4.1 and appropriate references. The following general design procedure may be followed:

1. Select the current density to be applied.
2. Compute the total current required to achieve the selected current density.
3. Design the ground bed system or distributed anode system.
4. Select a rectifier of proper voltage and current output.
5. Design all electrical circuits, fittings, and switchgear in accordance with good electrical practice.
6. Locate the cathodic protection test stations.
7. Prepare the necessary drawings and specifications for the project.

An example design is provided in this subsection for illustrative purposes and should not be interpreted as implying to present the best design or the only possible design. Because there are a multitude of ISFSI variables that will bear upon the design of the ICCPS for a particular site including differing ISFSI layouts, certain simplifying assumptions are made throughout the example. The example provides the user with insight on the types of design decisions that will need to be made. For example, because of possible shielding effects between CECs, as well as other SSC obstructions, the design implements a layout with closely distributed anodes to provide more uniform current distribution. Also, the example design implements closed loop electrical connections such that if the wire/cable is severed at any one place, electrical continuity is maintained to all anodes. Another item to be considered during the design phase is whether or not a test station is needed for each and every CEC.

Figure 2.I.1 presents an example ICCPS design layout for a 2x6 Array of VVMs. The ICCPS consists of the following four main subsystems/components:

1) Rectifier
2) Anodes
3) Test Stations
4) Wires and Cables

Figure 2.I.2 presents an example ICCPS test station.

The following is an example computation for determining the required current (approximate dimensions and quantities are used) as applicable to Figure 2.I.1:

Assume a CEC length (determined from "top of grade" to bottom of CEC bottom plate): 219.5 in.
CEC outside diameter: 86 in.
CEC condition: exterior is coated
Coating efficiency: 91.5% (i.e. 8.5% of the coated CEC surface is considered bare metal)
Cathodic Protection: Rectifier and distributed Natural Graphite Anodes with carbonaceous backfill
Soil resistivity: 4,000 ohm/cm^2
Current density: 1 mA/ft^2 exposed metal
Outside area of each CEC: 59,300 in^2 (412 ft^2)
Total area for an array of twelve CECs: 4,944 ft^2
Bare CEC metal exposed: 4,944 ft^2 x 0.085 or 420 ft^2
Current required: 420 ft^2 x 1 mA/ft^2 or 420 mA

The following is additional data applicable to Figure 2.I.1.

Approximate Anode quantity: 11
Approximate Anode size: 5 in dia. x 120 in. long
Approximate Backfill quantity: 6,000 lbs of carbonaceous backfill

The total number of anodes required is determined primarily by the total current requirements of the CEC metal to be protected and the optimum current density of the anode material selected.

Graphite is a semi-consumable anode. Graphite typically has experienced corrosion rates of 1.5 to 2.16 lbs /amp year [2.I.3] or as determined by experiment, 0.08 grams per square meter of anode per amp-hour of current (at 30 C, 40 mA/cm^2 anode current density) [2.I.4]. A computed anode life of less than 40 years is acceptable as long as appropriate measures are taken to facilitate the replacement of anodes during the design phase and appropriate maintenance planning measures are implemented. Use of carbonaceous backfill should be considered since it can substantially lengthen the anode life. Inert (non-consumable) platinized anodes may also be considered.

v. Concrete Encasement (Corrosion Mitigation Measure)

If concrete encasement is used, it shall be implemented in accordance with the requirements in Supplement 3.I, Subsection 3.I.4.1 and appropriate references.

The following points shall also be taken into consideration:

- The effect of the concrete encasement on the ICCPS, if an ICCPS is also implemented.
- The concrete encasement should not interfere with the settlement of the TSP (which provides the transporter support surface) without appropriate evaluation.

vi. Retaining Wall

Because the subgrade within and around an operating 100U ISFSI serves a principal shielding

function, it is essential that any excavation activity adjacent to the ISFSI (e.g., to build an extension of the ISFSI), must not disturb the soil in the Radiation Protection Space (RPS) shown in the licensing drawings (Section 1.I.5).

The extent of the RPS is set down to ensure, with sufficient margin of safety, that the ISFSI will continue to meet all relevant safety criteria under all applicable conditions of storage including normal, off-normal, extreme environmental phenomena and accident conditions. For example, the RPS must provide sufficient buffer so that design basis projectiles (large, medium, and penetrant missiles) will not access an MPC stored in a VVM cavity. In this case, as explained in Supplement 3.I, the incident missile is assumed to act when a deep cavity has been excavated contiguous to the RPS and the direction of action of the missile is oriented to achieve maximum penetration of the sub-grade towards the CEC shell.

A retaining wall at the edge of or beyond the RPS is recommended if an excavation activity is planned adjacent to the RPS boundary while the ISFSI is in active service. The retaining wall, as shown in the licensing drawing, shall be keyed to the TSP and connected using dowels to the SFP so that it is laterally restrained from movement but does not transmit any bending moment to the SFP or the TSP. The minimum structural design requirements on the retaining wall are provided in Table 2.I.2 and the licensing drawing in 1.I.5.The applicable load combinations for the structural analysis of the retaining wall pursuant to ACI-318(2005) are provided in Table 2.I.11.

When a retaining wall is installed on one or more sides of the 100U ISFSI, excavation activities associated with the construction of a new underground ISFSI can be performed directly adjacent to the retaining wall(s) at depths above the bottom surface of the existing SFP. Soil excavations below the bottom surface of the existing SFP shall be treated as though no retaining walls are installed and, therefore, are subject to the limitations of the following paragraph.

For the case where a retaining wall is not installed, no excavation activities associated with the construction of a new underground ISFSI shall take place within a distance from the RPS equal to ten times the planned excavation depth. Alternatively, the Excavation Exclusion Zone (EEZ), defined as the minimum distance from the centerline of a VVM located on the periphery of the ISFSI to where the effect of DBE is sufficiently attenuated such that a full depth excavation will not cause collapse of the lateral sub-grade at the RPS boundary during an earthquake, can be determined by a site specific seismic analysis. If a retaining wall is installed at or beyond the RPS then the wall becomes the EEZ boundary, but only for excavation depths above the bottom surface of the existing SFP.

2.I.3 Service Conditions and Applicable Loads

The categories of loads on the HI-STORM 100U VVM are identified below. They parallel those for the aboveground systems.

- Normal Condition: dead weight, handling of the Closure Lid, soil overburden pressure from subgrade, live load due to cask transporter movement, snow loads, and buoyancy effect of

water saturation of surrounding subgrade and foundation. Most normal condition loadings occur at an ambient temperature denoted as the "normal storage condition temperature"; however, for calculations involving the Closure Lid, a higher temperature is assumed when the VVM carries a loaded MPC since the Closure Lid outlet ducts will be subject to heated air.

- Off-Normal Condition: elevated ambient temperature and partial blockage of air inlets.

- Extreme Environmental Phenomena and Accident Condition: handling accidents, fire, tornado, flood, earthquake, explosion, lightning, burial under debris, 100% blockage of air inlets, extreme environmental temperature, 100% fuel rod rupture, and an accident during construction in the vicinity of a loaded ISFSI.

The design basis magnitudes of the above loads, as applicable, are provided in Tables 2.I.1 and 2.I.5, and are discussed further in the following subsections. Applicable loads for an MPC contained in a VVM or for a HI-TRAC that services a VVM are identical to those already identified in the main body of Chapter 2 and, therefore, are not repeated or discussed within this supplement. However, recognizing that the support of an MPC in a VVM is different from the support provided in an above ground HI-STORM, the design basis dynamic analysis model includes the fuel assemblies, the fuel basket, and the enclosure vessel so that the loads described above are properly distributed within the VVM.

2.I.4 Normal Condition Operating Parameters and Loads

i. Dead Load

The HI-STORM 100U System must withstand the static loads due to the weight of each of its components. As the support provided by the subgrade and the VVM Interface Pad from lateral friction is apt to be negligible, the weight of the Closure Lid is assumed to bear on the Container Flange and the Container Shell; the load to the VVM Support Foundation is transferred through direct bearing action.

ii. Handling Loads

The only instance of a handling load occurs during emplacement or removal of the Closure Lid while the CEC contains a loaded MPC. To provide defense-in-depth, Closure Lid lifting attachments shall meet the design requirements of ANSI N14.6 [2.2.3].

Lift locations for the CEC and the Divider Shell are used for lifting only during construction, and possibly during maintenance and decommissioning of the VVM with no loaded MPC present; therefore, these lifting locations are not subject to the defense-in-depth measures of NUREG-0612. They are therefore considered as a part of the site construction safety plan, site-specific maintenance program, or site decommissioning plan, as applicable, and as such are treated as being outside the scope of this FSAR.

iii. Live Loads

a. Subgrade Pressure Due to Transporter Movement

The properties of the surrounding subgrade and the presence of a loaded cask transporter affect the state of stress in the subgrade continuum. This stress field may produce a lateral compressive load on the Container Shell, which acts together with the effect from soil overburden.

b. MPC Transfer Operation

The VVM must withstand the weight of the loaded HI-TRAC transfer cask and the mating device during MPC transfer operations. Bounding weights for these components are used in the qualifying analysis.

iv. Ambient Temperature

The HI-STORM 100U System is analyzed for the same maximum yearly average ambient air temperature as that used for the aboveground overpacks. This normal operating condition temperature bounds all locations in the continental United States.

v. Snow

An appropriately conservative snow load on the Closure Lid is considered as a potential bounding case (see Table 2.I.1).

vi. Long-Term Settlement

There is no mechanism for an appreciable long-term settlement of Support Foundation Pad from loaded VVMs because the equivalent density of the loaded VVM is nearly equal to the density of the removed subgrade. Therefore, at an ISFSI site, depending on the density of the subgrade, there may be a small mismatch between the mass of the subgrade displaced by the loaded VVM leading to a minor amount of long-term settlement.

The TSP and the VIP are founded on well conditioned subgrade that is typically installed after the CECs are emplaced on the SFP. In addition to its own weight, the sole long-term load is the dead weight of the pads (TSP and VIP) which is evidently insufficient to cause appreciable long-term settlement in a subgrade continuum installed to meet or exceed a specific shear wave velocity and density criterion (see Table 2.I.2). Therefore, the long-term settlement of the TSP and the VIP relative to the SFP is expected to be small. Limiting allowable values of settlement for the SFP and the TSP have been specified in Table 2.I.2 for a conservative stress analysis of the TSP and SFP under the load combinations including the generic Design Basis Earthquake .

The effect of long-term settlement on the SFP and TSP shall be considered as a concurrent load with Dead load in all load combinations in the manner described in Section 2.I.2.

2.I.5 Off-Normal Condition Design Criteria

i. Elevated Ambient Air Temperature

The HI-STORM 100U System must be able to reject the design basis heat load under short-term conditions of elevated ambient air temperature.

ii. Partial Blockage of Inlet Air Ducts

The HI-STORM 100U System must withstand 50% blockage of the inlet air flow area without exceeding allowable temperature and pressure limits.

2.I.6 Environmental Phenomena and Accident Condition Criteria

The extreme environmental phenomena and accident conditions applicable to the HI-STORM 100U System are listed below. The loadings apply to either or both the VVM components and ISFSI structures.

i. Handling Accidents (Drops and Tipover)

Because the VVM is situated underground and cannot be moved, drop and tipover events are not credible accidents for this design. The Closure Lid, as discussed in Supplement 1.I, cannot strike the MPC lid due to geometry constraints if it were to undergo a free fall. Further, because the load handling device and lifting equipment are required to meet the defense-in-depth criteria set down in this FSAR, the drop of the Closure Lid or transfer cask during handling operation is termed non-credible (as is the case for the aboveground HI-STORM system MPC transfer operations at the ISFSI).

ii. Fire

The VVM must withstand the effects of a fire that consumes the maximum volume of fuel permitted to be in the fuel tank of the cask transporter. The duration of the fire for the VVM is conservatively assumed to be the same as that used for the aboveground overpacks. As is the case for aboveground overpacks, the fuel is assumed to spill, surround one storage system and burn until it is depleted. Because the VVM is configured to have a surrounding built-in step or spill barrier (see Figure 1.I.3), the spilled fuel will collect and burn over the Top Surface Pad, also referred to as Top-of-Grade (see Figure 1.I.2). Therefore, the location of fuel combustion will be somewhat removed from the CEC. Also, the natural grade in the TSP surface, engineered to direct the rainwater away from the VVMs, will do the same to the spilled fuel, further ameliorating the thermal consequence of the fire to the stored SNF.

The closed-end geometry of the MPC storage cavity ensures that a sustained combustion of the fuel, even if it were to be hypothesized to enter the VVM cavity, is not possible.

The loss of shielding effectiveness due to heat up of the concrete and the surrounding SSCs is primarily due to vaporization of the small amount of volatiles, including the contained moisture present in the concrete. This reduction in shielding is small and is permitted under the regulations. Therefore, the fire analysis of the VVM is focused on determining safety against a structural collapse due to elevation in the structure's metal temperature.

The sole effect of fire on the VVM structure is to raise the metal temperature of the structural members surrounding the shielding concrete in the Closure Lid. The analysis for the fire event accordingly seeks to establish that the load bearing structure will not be weakened by the rise in its metal temperature (and a consequent reduction in the yield and ultimate strength) and result in its structural collapse.

iii. Tornado

The HI-STORM 100U System is protected from the effects of a tornado and accompanying missiles by virtue of its underground configuration. The only VVM component that warrants evaluation for the effects of a tornado-induced missile strike is the Closure Lid, which is made of a steel weldment with encased concrete.

The HI-STORM 100U System is inherently stable under tornado missile impact. The impact of a large missile (1800kg Automobile) is evaluated to determine whether the Closure Lid continues to maintain its required shielding function. Penetration and perforation issues associated with the Closure Lid due to intermediate missiles that constitute the Extreme Environmental Phenomena loads for the HI-STORM 100U system are also addressed. The Closure Lid is analyzed for penetration of a solid steel cylinder traveling at a high speed consistent with the characteristics of the intermediate missile listed in Table 2.2.5. As there is no direct line of sight to the MPC, small missiles are not considered. Also, since a tornado is a short duration event, the effect of extremely high tornado winds on the thermal performance of the VVM would be negligible due to the system's thermal inertia. Therefore, the effect of tornado wind on the thermal performance of the HI-STORM 100U system is not analyzed.

iv. Flood

As discussed in Subsection 1.I.2, the HI-STORM 100U System is engineered to be flood resistant. However, even though the potential water ingress passages are elevated in the HI-STORM 100U (in contrast to the pad level inlet ducts in typical ventilated overpacks), submersion flooding that fills all or a portion of the ducts could occur at certain ISFSI sites located in flood zones. The MPC is designed to withstand 125 feet of water submergence. The VVM will clearly withstand this static head of water above the surface of the ISFSI

because all structural members either are not subject to any pressure differential from the flood or are backed by the subgrade, which resists the flood water directly. Full or partial submergence of the MPC is not a concern from a thermal perspective, as discussed in Supplement 1.I, because heat removal is enhanced by the floodwater.

The most severe flooding event from a thermal perspective would be the partial filling of the intake plenum such that airflow is blocked but the MPC is not submerged in water. To mitigate the consequences of this event, the height of the Divider Shell cutouts is purposely located well above the bottom elevation of the MPC. Therefore, if the flood level is just high enough to block air flow, the lower portion of the MPC will be submerged in water. The wetted MPC bottom region serves as an efficient means of heat rejection to the floodwater. This accident event is described in Supplement 11.I.

v. Earthquake

As explained herein and in Subsection 3.I.4.7, the generic seismic loading for the HI-STORM 100U system is established using the combination of an earthquake and soil subgrade properties that maximize the severity of the inertia forces on the ISFSI structures and components.

As required by 10CFR72.102(f), the Design Basis Earthquake for the ISFSI must be specified. For the HI-STORM 100U system, a generic Design Basis Earthquake is specified with horizontal and vertical ZPAs intended to envelope the site-specific DBEs at all U.S. plant sites (See Table 2.I.2). For purposes of the generic seismic analysis in this FSAR, the Design Basis Earthquake for the HI-STORM 100U system is defined by two sets of response spectra specified at the SFP bottom surface elevation and at the TSP top surface elevation, as shown in Figure 2.I.4. These two spectra sets together exhibit the severity of the earthquake experienced by the ISFSI structures and VVM Components and are henceforth referred to as the governing spectra. The two sets of response spectra are obtained from the two-step SHAKE/LS-DYNA seismic response analyses performed using a lower-bound soil shear wave velocity profile (see Figure 2.I.6). This lower bound profile was established in [2.I.10] based on the geotechnical data of typical U.S. nuclear power plant sites. To develop the governing spectra, the input seismic acceleration time history for the SHAKE analysis is derived from the Regulatory Guide 1.60 seismic response spectrum and designated as the rock outcrop motion. The synthetic time history complies with the response spectrum and power density enveloping criteria in SRP 3.7.1 in NUREG-0800, Rev 2. The input acceleration time history is scaled to yield ground surface ZPAs (at the top of grade elevation) specified in Table 2.I.2. The average strain-compatible shear wave velocities of the soil column obtained from the SHAKE analysis are used to specify the minimum shear wave velocity values in Table 2.I.2. The ZPAs of the rock outcropping acceleration that yielded the governing spectra are 0.538 g's for the horizontal direction and 0.483 g's for the vertical direction.

The soil model for the subsequent LS-DYNA seismic response analysis uses the average

strain-compatible wave velocities obtained from the SHAKE analysis (i.e., minimum shear wave velocity values in Table 2.I.2) to define the structural characteristics of the soil layers above and below the SFP elevation (see Figure 2.I.5 for sub-grade and under-grade space nomenclature). The acceleration time history at the soil column bottom surface, also obtained from the above-mentioned SHAKE analysis, is used as the input seismic motion for the LS-DYNA seismic response analysis performed in Supplement 3.I. The response spectrum plots shown in Figure 2.I.4 are the results of the LS-DYNA seismic response analysis (in the absence of the ISFSI). The same soil model and input seismic motion used in the LS-DYNA seismic response analysis is used for the LS-DYNA Soil-Structure Interaction (SSI) analysis (with the ISFSI included in the model) in Supplement 3.I.

The combination of weak soil properties and strong earthquake, as specified in Table 2.I.2 and Figure 2.I.4 for the structural evaluation of the underground ISFSI, has been selected to ensure that the Design Basis Earthquake response spectra at the ISFSI location will uniformly envelope those at most U.S. nuclear plants and that the Design Basis structural evaluation for the "100U" system is performed conservatively based on the lower bound support from the sub-grade and the under-grade. Thus, the HI-STORM 100U system can be deployed in most U.S. nuclear power plant sites without the need for a site-specific analysis to satisfy the requirements of 72.212. Specifically, a candidate 100U ISFSI site will be exempt from a detailed SSI analysis if the seismic response analysis for the site (using SHAKE or similar program) can demonstrate that the following two criteria are met:

1. The site's response spectra at both TSP and SFP elevations are enveloped by the Design Basis Earthquake response spectra shown in Figures 2.I.4-A and 2.I.4-B, respectively;

2. The soil properties of the candidate site are greater than the minimum values specified in Table 2.I.2.

In order to satisfy the first criterion, the site must consider multiple time history sets as input to the seismic response analysis based on the guidelines set forth in SRP 3.7.1 [2.I.12] and ASCE 4-98 [2.I.11]. The site's response spectra at both the TSP and SFP elevations must be bounded by the Design Basis Earthquake response spectra in Figure 2.I.4 for all acceleration time histories sets used as input.

For the case where only one of the above two criteria is not satisfied, a site-specific evaluation under 10CFR72.212 is permitted. Typical scenarios that warrant a site specific evaluation are discussed below:

<u>Scenario A</u>: The site's response spectra are not completely enveloped by the Design Basis Earthquake response spectra in Figure 2.I.4. However, the site's overall earthquake strength, represented by the resultant ZPA (see Table 2.I.2 for definition) is bounded by that of the Design Basis Earthquake at both TSP and SFP elevations.

While the ZPA represents the strength of the earthquake (in terms of the maximum value of

the seismic acceleration time history), the shape of the seismic response spectrum is affected by many factors such as the overall stiffness of the site and the stiffness profile of soil layers. Therefore, for the same input seismic time history at the base of the soil column, a stiffer site could have a peak response that is not enveloped by the Design Basis Earthquake response spectrum (as demonstrated in the SHAKE parametric study results presented in Table 2.I.4, where the only difference between the two analyzed cases is the stiffness (i.e., shear wave velocity) of the soil column). Although it is expected that the 100U system would exhibit a greater safety margin against the earthquake loading at the stiffer subgrade/undergrade site, a site-specific evaluation under 10CFR72.212 is the appropriate vehicle to confirm the structural integrity in this situation.

Scenario B: The strain compatible wave velocity of the soil in Space B and/or Space D of the ISFSI site (see Figure 2.I.5) is less than the required minimum value specified in Table 2.I.2.

Typically, Spaces B and D (in Figure 2.I.5) contain native soils whose properties are not affected by the ISFSI construction. More importantly, the loaded VVMs are not directly supported by the soil in the two spaces. Therefore, it is reasonable to assume that a small reduction of soil stiffness in these two spaces would not significantly modify the structural response of the VVM system. Structural compliance through a site specific analysis is assured if the ZPA of the DBE is well below the Design Basis value set down in this FSAR (Figure 2.I.4).

The site-specific safety analysis, if performed, shall follow the methodology set down in Supplement 3.I. In addition, since the soil and rock configuration varies from site to site, the total depth of the soil model for site-specific analysis shall be determined following the guideline in Section 3.3.3.2 of ASCE 4-98 [2.I.11]. Uncertainties in SSI analysis for a candidate 100U ISFSI site shall be accounted for by varying the best estimate low strain shear modulus of the substrates between the best estimate values times (1+c) and the best estimate value divided by (1+c). If sufficient, adequate soil investigation data is available, the mean and standard deviation of the low strain shear modulus shall be established for every soil layer. The value of c may be established so that it will cover the mean plus or minus one standard deviation for every layer; however, the minimum value for c shall be no less than 0.5. If sufficient data is not available to determine a statistically meaningful mean and standard deviation, then the value for c shall be no less than 1.0.

The qualification of the ISFSI under the system's DBE event involves the following safety determinations:

1. Compliance of the VVM components (Divider shell, CEC shell, etc.) to the applicable stress/deformation limits specified in Table 2.I.6.

2. Strength compliance of the ISFSI reinforced concrete structures under ACI -318(2005) load combinations listed in Table 2.I.11.

A candidate 100U ISFSI site that does not meet the requirement discussed above for seismic qualification shall not be allowed for the consideration of a 100U general license.

vi. Explosion

The HI-STORM 100U System must withstand the pressure pulse due to a design basis explosion event. The effect of overpressure due to an explosion near the VVM is evaluated. The overpressure design value applied to the Closure Lid outer shell surface is intended to bound all credible explosion events because no combustible material is permitted to be stored near the VVM, and all materials of construction are engineered to be compatible with the operating environment. However, site-specific explosion scenarios that are not evidently bounded by the design basis explosion load considered herein (see Table 2.I.1) shall be evaluated under the provisions of 10CFR72.212.

vii. Lightning

The HI-STORM 100U System must withstand a lightning strike without a significant loss in its shielding capability. The effect of a lightning strike on the VVM is the same as that described for the aboveground overpack design, even though the likelihood of a lightning strike on the VVM is lower due to its low height above grade. Lightning is treated as an Extreme Environmental Phenomena event in Supplement 11.I. Because of its non-significant structural effect on the VVM, it is not considered as a load that warrants analysis in Supplement 3.I.

viii. Burial Under Debris

The burial under debris event for the HI-STORM 100U System is bounded by the evaluation performed for the aboveground overpacks, as discussed in Supplement 4.I.

ix. 100% Blockage of Air Inlets

The blockage of the entire inlet air flow area is analyzed as an accident event and is described in Supplement 11.I and analyzed in Supplement 4.I.

x. Extreme Environmental Temperature

An extremely high ambient air temperature is analyzed as an extreme environmental event and is described in Supplement 11.I and analyzed in Supplement 4.I.

xi. 100% Fuel Rod Rupture

This loading condition is specific to the MPC thermal evaluation and treated in Supplement 11.I.

2.I.7 Codes, Standards, and Practices to Ensure Regulatory Compliance

There is no U.S. or international code that is sufficiently comprehensive to provide a completely prescriptive set of requirements for the design, manufacturing, and structural qualification of the VVM. The various sections of the ASME Codes, however, contain a broad range of specifications that can be assembled to provide a complete set of requirements for the design, analysis, shop manufacturing, and field erection of the VVM. The portions of the ASME Codes that are invoked for the various elements of the VVM design, analysis, and manufacturing activities are summarized in Table 2.I.3.

The ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NF Class 3, 1995 Edition, with Addenda through 1997 [2.2.1], is the applicable code to determine stress limits for the metallic structural components of the VVM when required by the acceptance criteria listed in Table 2.I.5. Table 2.I.3 summarizes considerations for design, fabrication, materials, and inspection. The permitted material types and their permissible temperature limits for long-term use are listed in Table 2.I.8. Manufacturing requirements are set down in licensing and design drawings.

ACI-318(2005) [2.I.5] is the applicable reference code to establish applicable limits on unreinforced concrete (in the Closure Lid), which is subject to secondary structural loadings. Appendix 1.D contains the design, construction, and testing criteria applicable to the plain concrete in the VVM's Closure Lid. The load combinations applicable to the TSP, SFP, and the retaining wall, pursuant to ACI-318(2005) are summarized in Table 2.I.11. Since the VIP carries no load except for the self-weight and is thicker than the TSP, the structural evaluation of the VIP is not necessary. Applicable sections of ACI-318(2005) should be used in the design of the interfacing SSCs and proximate structures.

The selection of the ISFSI site shall be made with due consideration of the potential of liquefaction. The host plant's criteria with respect to liquefaction for siting the Part 50 structures shall be used.

As mandated by 10CFR72.24(c)(3) and §72.44(d), Holtec International's quality assurance program requires all constituent parts of an SSC subject to NRC's certification under 10CFR72 to be assigned an ITS category appropriate to its function in the control and confinement of radiation. The ITS designations for the constituent parts of the HI-STORM 100U VVM, using the guidelines of NUREG-CR/6407 [2.0.5], are provided in Table 2.I.8.

The aggregate of the citations from the codes, standards, and generally recognized industry publications invoked in this FSAR, supplemented by the commitments in Holtec's quality assurance procedures, provide the necessary technical framework to ensure that the as-installed VVM would meet the intent of §72.24(c), §72.120(a) and §72.236(b). As required by Holtec's QA Program

(discussed in Chapter 13), all operations on ITS components must be performed under QA validated written procedures and specifications that are in compliance with the governing citations of codes, standards, and practices set down in this FSAR. For activities that may be performed by others, such as site construction work to install the VVM, Holtec International requires that all activities be formalized in procedures and subject to the CoC holder's as well as the ISFSI owner's review and approval.

An ITS designation is also applied to the interfacing SSCs (such as the Support Foundation), which requires that all quality assurance measures set down in Holtec's Quality Assurance Procedure Manual be complied with by the entity performing the site construction work. In this manner, the compliance of the as-built VVMs with its engineered safety margins under all design basis scenarios of loading is assured.

2.I.8 Service Limits

No new service limits are defined for the HI-STORM 100U System beyond those described in Subsection 2.2.5.

2.I.9 Loads and Acceptance Criteria

Subsections 2.I.4, 2.I.5, and 2.I.6 describe the loadings for normal, off-normal, and extreme environmental phenomena and accident conditions, respectively, for the HI-STORM 100U System. Tables 2.I.1 and 2.I.2, respectively, provide the design loads and seismic load parameters in terms of ZPA values for a bounding analysis using the methodology of Subsection 3.I.4.7.

Bounding load cases that are significant to the structural performance of the VVM and require evaluation are compiled in Table 2.I.5 using information provided in Sections 2.I.4, 2.I.5, and 2.I.6. Supplement 3.I contains a description of the evaluations, establishes the evaluation methodology, and provides evaluation results that demonstrate compliance of the VVM to the applicable load cases and acceptance criteria described below. The load cases and acceptance criteria are explained in subsequent paragraphs and summarized in Table 2.I.5. Table 2.I.6 summarizes the acceptance criteria for the CEC and internals under extreme environmental events.

Each loading case in Table 2.I.5 is distinct in respect of the sub-component of the VVM that it affects most significantly. The acceptance criteria consist of demonstrating that (i) radiation shielding does not degrade under normal and off-normal conditions of storage loadings, (ii) the system does not deform under credible loading conditions in a manner that would jeopardize the subcritical condition or retrievability of the MPC, and (iii) the MPC maintains confinement. For accident condition loadings, any permissible degradation in shielding must be shown to result in dose rates sufficiently low to permit recovery of the MPC from the damaged cask, including unloading if necessary, and loss of function must be readily visible, apparent or detectable.

The above set of criteria, extracted from NUREG-1536, is further particularized in a more conservative form for each applicable loading case in this subsection.

Load Case 01: Buoyant Force

This loading case pertains to the scenario wherein a VVM has been built, but the Closure Lid and MPC are not yet installed. Strictly speaking, this condition is not important to storage safety because the MPC is not present. However, considerations of long-term service life warrant that a minimum weight CEC, subject to the maximum buoyant force of water under an assumed hypothetical condition of submergence in water with a head equal to the length of the CEC, does not float. This evaluation sets a minimum additional weight (usually on a temporary cover) that will be set in place during construction to protect the CEC from construction debris, to provide for construction worker safety, and to insure that the CEC does not suffer uplift from buoyant forces. In addition, the Bottom Plate of the CEC must have sufficient flexural strength such that under a buoyant uplift pressure, its primary bending stress intensity remains below the ASME Level D allowable stress intensity at the reference metal temperature (assumed to be same as the extreme environmental condition temperature specified in Table 2.I.1 of this FSAR).

Load Case 02: Dead Load plus Design Basis Explosion Pressure

The dead weight loading, explained in Paragraph 2.I.4(i) is accentuated by the design basis explosion loading defined in Paragraph 2.I.6(vi). The explosion load is stated in terms of an equivalent static pressure. The affected sub-components are:

a. The Container Shell, subjected to a compressive state of stress under the combined effect of dead weight of the Closure Lid and surface pressure on the Closure Lid under the explosion event.

b. The Closure Lid, subject to self-weight and the Closure Lid surface pressure under the explosion event.

Other VVM components are not in the direct path of this loading. The explosion pressure envelops other mechanical loads such as snow and flood. Load Case 02, therefore, is a bounding load combination that conservatively subsumes a number of normal and extreme environmental phenomena loads. As this load case is intended to bound any normal condition, Level A stress limits are applicable to this case based on reference metal temperatures that bound all mechanical loading scenarios.

Load Case 03: Tornado Missile Impact

The Closure Lid is the only exposed portion of the VVM. Therefore, the tornado-borne missile strikes must be postulated to occur on the lid. The only other affected VVM part is the Container Flange, which prevents lateral sliding of the lid.

When subject to a tornado missile strike, the Closure Lid must not be dislodged, resulting in a direct line of sight from the top of the MPC to the outside. For the intermediate missile, the Closure Lid

must resist full penetration. Finally, any CEC deformation from the compressive axial impulse due to the missile strike must not prevent MPC retrievability.

Load Case 04: Design Basis Seismic Event

The Design Basis Seismic Event is classified as an extreme environmental phenomenon. As such the Level D service condition limits are applicable to the Code components, such as the MPC Enclosure Vessel. The MPC Enclosure Vessel and fuel basket have been qualified to a 60g deceleration limit in the HI-STAR 100 (Docket Nos. 72-1008, 71-9261); this deceleration exceeds the expected deceleration from a seismic event. However, to ensure an accurate structural evaluation of the VVM, the evaluation of the response of the VVM to the design basis seismic event shall include a detailed model of the MPC, the fuel basket, and the contained fuel; this model, referred to as the Design Basis Seismic Model, should capture impacts between the fuel and the fuel basket, between the fuel basket and the MPC, and between the MPC and applicable components of the VVM.

The CEC shell is subject to performance-based limits, which require that the deformation of the CEC does not prevent MPC retrievability, does not cause loss of MPC confinement, and that the system remains subcritical. This is accomplished by demonstrating that after the seismic event, permanent ovalization of the Container Shell and/or Divider Shell does not result in a geometry that precludes retrievability of the MPC and that the impact loadings on the MPC due to its rattling inside the CEC do not cause a breach of the MPC confinement boundary.

The Divider Shell's sole function is to direct the airflow inside the CEC cavity and to hold MPC Guides that serve to restrain the MPC from excessive rattling motion during an earthquake event. The MPC guides welded to the Divider Shell are subject to compressive impacts from the "hard points" on the MPC (the approximately 2.5-inch thick baseplate at the bottom and the 9.5-inch thick lid at the top). The MPC tubular guides are engineered to serve as "impact limiters" to minimize the local plastic strains in the MPC Confinement Boundary.

Finally, because the MPC Enclosure Vessel is designed to meet ASME Section III, Subsection "NB" (Class 1) stress intensity limits, and the earthquake is categorized as a "Level D" event, the primary stress intensities in the MPC Enclosure Vessel must meet Level D limits. The primary stress intensity in the MPC shell is the maximum longitudinal flexural stress intensity, which is compared against the primary membrane stress intensity limit for the material (Alloy X) at the applicable service temperature. The fuel basket is a multi-flange 3-D beam structure, designed to meet the stress limits of Subsection "NG" of the Code. The maximum longitudinal primary stress intensity in the basket, calculated from the 3-D fuel basket/fuel assembly model, must be less than the corresponding Level D condition limit at the service temperature. In addition to the primary stress based limits it is also necessary to demonstrate that the transverse bending stress in any panel normalized over the length of the fuel basket is less than the Level D primary stress limit.

The limits on the primary stresses in the MPC components for the DBE condition are also applicable to other Level D (faulted) events. Dynamic analysis using a 3-D detailed model of the MPC (which includes the Confinement Boundary, the internal fuel basket, and the fuel assemblies inside the

basket) is the vehicle for performing the structural qualification. In addition to the primary stress limits, the local strain in the Confinement Boundary due to the impact between the MPC and the MPC guides under the Design Basis Earthquake requires evaluation.

Table 2.I.5 summarizes the above discussion in tabular form.

Load Case 05: Closure Lid Handling

The Closure Lid lifting attachments shall meet the strength limits of ANSI N14.6 for heavy load handling. The metal load bearing parts shall satisfy the requirements of Reg. Guide 3.61 for primary stresses near the lifting locations and shall satisfy ASME NF Level A limits away from the lifting locations.

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.

Load Case 06: Design Basis Fire Event

The exposed portion of the VVM, namely the Closure Lid, will experience the heat input and temperature rise under the fire event. The balance of the VVM, because of its underground location, will be subject to only a secondary temperature increase.

It is required to demonstrate that the structural collapse of the Closure Lid cannot occur due to the reduction of its structural material's (low carbon steel) strength at the elevated temperatures from the fire.

Load Case 07: CEC Loading From Surrounding Subgrade

The CEC is subject to a lateral pressure from the soil in the non-seismic condition. This pressure is affected by the presence of a loaded cask transporter adjacent to the CEC. The CEC must be shown to provide adequate resistance to this loading.

This load case tends to ovalize the CEC; the maximum primary membrane plus bending stress is limited to the material yield strength under normal conditions of storage.

In evaluating the structural safety margins in Supplement 3.I for the load cases described above, design data for the interfacing SSCs presented in Table 2.I.2 is used as applicable.

2.I.10 Safety Protection Systems

The HI-STORM 100U System, featuring the VVM with the stored MPC, provides for confinement, criticality control, and heat removal for the stored spent nuclear fuel in the manner of the aboveground overpacks. The VVM provides better shielding and protection from environmental

events, such as tornado missiles, because of its underground configuration. The information in Section 2.3 also applies to the HI-STORM 100U System, with the recognition that the air ventilation system is modified. Instead of the ambient air entering through inlet ducts at the bottom, the cooling air enters the circumferentially symmetric passage at the top of the VVM and is directed to the bottom of the VVM cavity along a radially symmetric annulus (Figure 1.I.4). However, the mechanism of heat transfer from the MPC to the cooling air is identical to the aboveground overpack designs.

The HI-STORM 100U System is completely passive requiring no active components or instrumentation to perform its design functions. Temperature monitoring or scheduled visual verification of the integrity of the air passages is used to verify continued operability of the VVM heat removal system, as set down in the system's Technical Specification.

2.I.11 Decommissioning Considerations

The HI-STORM 100U VVM is specifically engineered to facilitate convenient decommissioning. As discussed in Supplement 1.I, the component most proximate to the active fuel and, hence, likely to be the most activated, is the Divider Shell. The Divider Shell is not welded to the CEC structure; therefore, it can be conveniently removed for decommissioning. The CEC structure can be removed by excavating the surrounding subgrade. Alternatively, the cavity can be filled with suitable fill materials and the CEC left in place. While the above discussion is unique to the VVM design, the information in Section 2.4 pertaining to decommissioning of other HI-STORM models is also applicable to the VVM. Even if the decision is made to dispose of all activated material, the VVM, due to differences in its geometry and construction (particularly, use of the native soil as the biological shield to the extent possible) will result in less steel and concrete to be disposed of. In the aggregate, it is estimated that less material will need to be disposed of to decommission a VVM ISFSI in comparison to an ISFSI containing aboveground overpacks.

Finally, the activation estimate in Table 2.4.1 for the aboveground overpack inner shell is conservatively applicable to the VVM steel shell enclosure.

2.I.12 Regulatory Compliance

Pursuant to the guidance provided in NUREG-1536, the foregoing material in this supplement provides:

i. a complete set of principal design criteria for the VVM as mandated by 10CFR72.24I(1), §72.24(c)(2), §72.120(a) and §72.236(b);

ii. a clear identification of VVM structural parts subject to a fully articulated design subject to certification under 10CFR72 and of interfacing SSCs;

iii. the required set of limiting critical characteristics of the interfacing SSCs to ensure that the VVM will render its intended function under all design basis scenarios of operation;

iv. a complete set of requirements premised on well-recognized codes and standards to govern the design and analysis (to establish safety margins) and manufacturing of the VVM; and

v. a table containing cross-reference between the applicable 10CFR72 requirements and the location in this FSAR where the fulfillment of each specific requirement is demonstrated.

It is noted that the requirements of 10CFR72 do not preclude the use of an underground storage system such as the HI-STORM 100U. The VVM concept, while not specifically mentioned in the regulatory guidance literature associated with implementing the requirements in 10CFR72 (i.e., NUREG-1536), meets and exceeds the intent of the guidance in that it provides an enhanced protection of the stored spent nuclear fuel and a significantly reduced site boundary dose, enables a more convenient handling operation, and presents a much smaller target for missiles/projectiles compared to an aboveground storage system.

2.I.13 References

The references in Section 2.6 apply to the VVM to the extent that they are appropriate for use with an underground system.

[2.I.1] NACE Standard RP0104-2004 "The Use of Coupons for Cathodic Protection Monitoring Applications", NACE International.

[2.I.2] NACE Standard TM0101-2001 "Measurement Techniques Related to Criteria for Cathodic Protection on Underground or Submerged Metallic Tank Systems", NACE International.

[2.I.3] Federal Construction Council Technical Report No. 32, Cathodic Protection As Applied to Underground Metal Structures", National Academy of Sciences – National Research Council, Publication 741, 1959.

[2.I.4] Rabah, M.A., et al., "Electrochemical Wear of Graphite Anodes during Electrolysis of Brine," *Carbon*, Vol. 29, No. 2, pp. 165-171, 1991.

[2.I.5] ACI-318(2005), Building Code Requirements for Structural Concrete (ACI 318-05) and Commentary (ACI 318R-05), Chapter 22, American Concrete Institute, 2005.

[2.I.6] Holtec Position Paper, DS-338, "A Methodology to Compute the Equivalent Elastic Properties of the Subgrade Continuum to Incorporate the Effect of Long- Term Settlement," A.I. Soler and C. Bullard (2010) (Holtec Proprietary)

[2.I.7] Basic Soils Engineerings, B.H.Hough, Second Edition.

[2.I.8] ACI 360R-06, Design of Slabs on Grade, American Concrete Institute, 2006.

[2.I.9] "2009 International Building Code," International Code Council, Inc.

[2.I.10] NUREG/CR-6865, "Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Storage Systems," U.S. Nuclear Regulatory Commission, February 2005.

[2.I.11] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 2000.

[2.I.12] NUREG-0800, SRP 3.7.1, "Seismic Design Parameters", USNRC, Revision 3, March 2007.

[2.I.13] N.M. Newmark, "Seismic Design Criteria for Structures and Facilities: Trans-Alaska Pipeline System," proceedings of U.S. national Conference on Earthquake Engineering, Ann Arbor, Michigan, June 18-20, 1975.

[2.I.14] USNRC Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Responses Analysis," Revision 2, July 2006.

TABLE 2.I.1
LOADS, CRITERIA, APPLICABLE REGULATIONS, REFERENCE CODES, AND STANDARDS FOR THE VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Life:		
Design Life	40 yrs, Section 3.I.4	-
License Life	20 yrs, Section 3.I.4	10CFR72.42(a) & 10CFR72.236(g)
Structural:		
Design & Fabrication Codes: Foundation Pad; VVM Interface Pad and Top Surface Pad	ACI-318(2005)	10CFR 72.24
Unreinforced Concrete Stress Limits (Closure Lid)	Applicable Sections of ACI-318(2005)	10CFR72.24(c)(4)
Structural Steel	Section 2.I.7, Tables 2.I.5, 2.I.6	10CFR72.24(c)(4)
VVM Closure Lid Dead Weight†:	Table 3.I.1	R.G. 3.61
Design Internal Pressure	Atmospheric, Supplement 1.I	Ventilated Module
Response and Degradation Limits	Section 3.I.4	10CFR72.122(b), (c)
Corrosion Allowance	1/8" on surfaces directly in contact with subgrade	Standard industry practice
Thermal:		
Maximum Design Temperatures:		
Closure Lid Concrete		
Through-Thickness Section Average (Normal)	Table 1.D.1	ACI 349-85, Appendix A, (Paragraph A.4.3)
Through-Thickness Section Average (Off-Normal and Accident)	Table 1.D.1	ACI 349-85, Appendix A, (Paragraph A.4.2)
Structural Steel	Table 2.I.8	ASME Code, Section II, Part D
VVM Divider Shell Thermal Insulation	Heat transfer resistance ≥ 4 hr-ft²-°F/Btu. Must be stable at temperatures ≤ 800°F	N/A
Confinement:	N/A, Provided by MPC; Supplement 7.I	10CFR72.128(a)(3) and 10CFR72.236(d) & (e)
Retrievability: Normal/Off-Normal/Accident	No damage that precludes MPC retrieval or threatens subcriticality of fuel. MPC maintains confinement,	10CFR72.122(f), (h), (1), & (l)

† All weights listed in Table 3.I.1 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.I.1 (continued)
LOADS, CRITERIA, APPLICABLE REFERENCE REGULATION/CODES AND STANDARDS FOR HI-STORM 100U VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
	Supplement 3.I	
Criticality:	N/A; Provided by MPC; Supplement 6.I	10CFR72.124 and 10CFR72.128(a)(2)
Radiation Protection/Shielding:		
Normal/Off-Normal	Provide capability to meet controlled area boundary dose limits under 10CFR72 for all normal and off-normal conditions; Supplement 5.I	10CFR72.104 and 10CFR72.212
	Ensure dose rates on and around the VVM during MPC transfer and lid installation operations are ALARA; Supplement 10.I	10CFR20
Accident or Conditions of Extreme Environmental Phenomena	Meet controlled area boundary dose limits in regulations for all accidents; Supplement 5.I	10CFR72.106
Design Bases:		
Spent Fuel Specification	Table 2.0.1; Section 2.I.1	10CFR72.236(a)
Normal Design Event Conditions:		
Ambient Outside Temperature:	-	-
Max. Yearly Average	80°F; Subsection 2.2.1.4	ANSI/ANS 57.9
Live Load†:		
Loaded HI-TRAC 125D and Mating Device	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Dry Loaded MPC	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Cask Transporter	Table 3.I.1, Subsection 2.I.9	-
Handling:	Subsection 2.I.4	-
VVM Closure Lid Lift Points	Subsection 3.I.4	NUREG-0612 ANSI N14.6
Minimum Temperature During Closure Lid Handling Operations	0°F; Subsection 2.2.1.2	ANSI/ANS 57.9
Snow and Ice Load	100 lb/ft^2; Subsection 2.I.4	ASCE 7-88
Wet/Dry Loading	Dry; Supplement 1.I, 8.I	-

† Weights listed in Table 3.I.1 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.I.1 (continued)
LOADS, CRITERIA, APPLICABLE REFERENCE REGULATION/CODES AND STANDARDS FOR HI-STORM 100U VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Storage Orientation	Vertical; Supplement 1.I	-
Off-Normal Design Event Conditions:		
Ambient Temperature:	Subsection 2.I.5	-
Minimum	-40°F; Subsection 2.2.2.2	ANSI/ANS 57.9
Maximum	100°F; Subsection 2.2.2.2	ANSI/ANS 57.9
Partial Blockage of Air Inlets	50% blockage of air inlet flow area; Supplement 4.I	-
Design Basis Accident Events and Conditions:		
Drop Cases:		
End Drop	Not credible; Subsection 2.I.6	In-ground VVM is not lifted
Tipover	Not credible; Subsection 2.I.6	In-ground VVM is constrained by subgrade and foundation
Fire:	-	-
Duration	217 seconds; Supplement 11.I	10CFR72.122(c)
Temperature	1475°F; Supplement 11.I	10CFR72.122(c)
Fuel Rod Rupture	See Table 2.0.1; Subsection 2.2.3.8	-
Air Flow Blockage	100% blockage of air inlet flow area; Subsection 2.I.6	10CFR72.128(a)(4)
Explosive Overpressure External Differential Pressure	10 psi steady state; Subsection 2.I.6 and Table 2.2.1	10CFR72.128(a)(4)
Extreme Environmental Phenomenon Events and Conditions:		
Flood:	Subsection 2.I.6	-
Height	125 ft	R.G. 1.59
Velocity	N/A; Supplement 1.I	In-ground VVM is not subject to tipover or sliding. Loads on the Closure Lid are bounded by missile impact loads.
Max. Earthquake	Table 2.I.2, Figure 2.I.4	10CFR72.102(f)
Tornado:	Subsection 2.I.6	-

TABLE 2.I.1 (continued)
LOADS, CRITERIA, APPLICABLE REFERENCE REGULATION/CODES AND STANDARDS FOR HI-STORM 100U VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Tornado-Borne Missiles:		
i. Automobile	Ensure shielding, subcriticality and retrievability MPC maintains confinement Subsection 2.I.6 and Supplement 3.I	NUREG-1536
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
ii. Rigid Solid Steel Cylinder (intermediate tornado missile)	Ensure shielding, subcriticality and retrievability, MPC maintains confinement	NUREG-1536
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
iii. Steel Sphere	Subsection 2.I.6	NUREG-1536 In-ground VVM has no penetrations that provide line-of-sight to MPC
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
Burial Under Debris	Maximum decay heat load and adiabatic heat-up; Subsection 2.I.6	-
Lightning	Bounded by aboveground evaluation (resistance heat-up); Subsection 2.I.6	In-ground VVM contains less metal
Extreme Environmental Temp.	125°F; Subsection 2.I.6 and Table 2.2.2	-
Load Cases for Structural Qualification:	Subsection 2.I.9 and Table 2.I.5	ANSI/ANS 57.9 and NUREG-1536

TABLE 2.I.2
DESIGN DATA FOR HI-STORM 100U ISFSI

	Item	Value(Minimum or nominal, as applicable)	Comment
1.	Support Foundation Pad, VVM Interface Pad and Top Support Pad, and Retaining Wall	▪ Minimum Concrete density = 145 lb/ft^3 ▪ Minimum concrete compressive strength @ ≤ 28 days = 4,500 psi ▪ Grade 60 Rebar - Minimum yield strength of rebar = 60,000 psi; rebar is #11@9" (each face, each direction) ▪ Minimum concrete cover on rebar per section 7.7.1 of ACI-318(05)	See Licensing Drawings in Section 1.I.5 for detailed concrete pad/wall thickness.
2.	Depth averaged density of subgrade in Space A (see Figure 2.I.5), lb/ft^3	120	A lower average density value may be used in shielding analysis in Supplement 5.I for conservatism.
3.	Depth averaged density of subgrade in Space B (see Figure 2.I.5), lb/ft^3	110	A lower average density value may be used in shielding analysis in Supplement 5.I for conservatism.
4.	Depth depth averaged density of subgrade in Space C (see Figure 2.I.5), lb/ft^3	120	Not required for shielding.
5.	Depth depth averaged density of subgrade in Space D (see Figure 2.I.5), lb/ft^3	120	This space will typically contain native soil. Not required for shielding.
6.	Lower bound, strain compatible effective shear wave velocity in Space A, V ft/sec (see Notes 1 and 2)	500	This space will typically contain engineered fill.
7.	Lower bound, strain compatible effective shear wave velocity in Space B, V ft/sec (see Notes 1 and 2)	450	This space will typically contain native soil.
8.	Lower bound, strain compatible effective shear wave velocity in Space C, V ft/sec (see Notes 1 and 2)	485	This space may be remediated with vertical reinforcement such as pilings to enhance V.

TABLE 2.I.2 (continued)
DESIGN DATA FOR HI-STORM 100U ISFSI

	Item	Value	Comment
9.	Lower bound, strain compatible effective shear wave velocity in Space D, V ft/sec (see Notes 1 and 2)	485	This space will typically contain native soil.
10.	Design Basis Earthquake	Ground surface spectra per Figure 2.I.4-A with horizontal ZPA, a_H and vertical ZPA, a_V as: a_H = 1.0g a_V = 0.75g and foundation surface spectra per Figure 2.I.4-B.	Horizontal and vertical spectra shown in Figures 2.I.4-A and 2.I.4-B are based on 5% damping. Following the Newmark 100-40-40 response combination technique [2.I.13] endorsed by the Regulatory Guide 1.92 [2.I.14], the *resultant ZPA* for a 3-D earthquake site is defined as: $a_R = a_1+0.4a_2+0.4a_3$, where a_1, a_2 and a_3 are the site's ZPAs in three orthogonal directions and $a_1 \geq a_2 \geq a_3$. Hence, the DBE *resultant ZPAs* at ground surface and foundation surface elevations are 1.3 g's (=1.0×1.0g's + 0.4×0.75 g's) and 1.228 g's (=1.0×0.94g's + 0.4×0.72 g's), respectively.
11.	Maximum permissible long-term settlement of the SFP	0.2 inches	
12.	Maximum permissible long-term settlement of the TSP with respect to the SFP	0.4 inches	

TABLE 2.I.2 (continued)
DESIGN DATA FOR HI-STORM 100U ISFSI

Note 1:
Strain compatible shear wave velocities in each space at an ISFSI site (see Figure 2.I.5) shall be computed using the guidance provided in Section 16 of the International Building Code, 2009 Edition [2.I.9]. The equivalent wave velocity is defined so that the wave transit time for an equivalent homogeneous material of the same total depth is the same as the actual layered substrate.

$$V = \frac{d}{\sum \frac{d_i}{v_i}}$$

d_i = thickness of i^{th} layer within the region (ft.);
v_i = strain compatible shear wave velocity of i^{th} layer within the region (ft./sec.);
d = total thickness of substrate region (e.g. 20', 80')
V = Equivalent Strain Compatible Shear Wave Velocity for substrate thickness "d".

Note 2:
The lower bound, strain compatible effective shear wave velocities at a particular site must account for the potential variability (i.e., uncertainty) in the site soil properties in accordance with Section 3.7.2 of NUREG-0800. This means that a site must demonstrate that, when the lower bound values for shear wave velocity (based on the site soil investigation data) are used as input, the free-field site response analysis yields a strain compatible effective shear wave velocity greater than the minimum value provided in this table for each space. The lower bound shear wave velocities used as input to the free-field site response analysis shall be determined using the following formula:

$$V_{LB} = \frac{V_{BE}}{\sqrt{1 + COV}}$$

where V_{BE} is the best estimate shear wave velocity for a given soil layer based on the soil investigation data, and *COV* is the coefficient of variation for the site soil properties. For well-investigated sites, the *COV* should be no less than 0.5. For sites that are not well investigated, the *COV* shall be set equal to 1.0.

TABLE 2.I.3
REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS

	Item	Code Paragraph†	Explanation and Applicability
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The "intervening elements" are termed interfacing SSCs in this FSAR.
3.	Certification of material	NF-2130(b) and (c)	Materials shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2400	-
6.	Welding procedure	Section IX	-
7.	Welding material	Section II	-
8.	Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3424	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 plate and shell supports and for linear supports are applicable for Closure Lid and Container Shell, respectively.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	Applies to the Divider Shell and Container Shell
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Storage of Welding Materials	NF-4411	-
19.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural welds
20.	Backing Strips, Peening	NF-4421 NF-4422	Applies to structural and non-structural welds
21.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
22.	Non-Destructive Examination	NF-5360	Invokes Section V
23.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	-

† All references to the ASME Code refer to applicable sections of the 1995 edition with addenda through 1997.

TABLE 2.I.4
SHAKE PARAMETRIC STUDY OF THE EFECT OF SUBGRADE PROPERTIES ON SOIL RESPONSES AT 100U ISFSI TOP & BOTTOM ELEVATIONS

Elevation & Direction	Acceleration Response	Value (g's)	
		Lower Bound Shear Wave Velocity Profile (see Figure 2.I.6)	Upper Bound Shear Wave Velocity Profile (see Figure 2.I.6)
TSP Top Surface Horizontal Direction	ZPA	1.008	0.897
	Peak	3.851	4.040
SFP Bottom Surface Horizontal Direction	ZPA	0.945	0.790
	Peak	3.590	3.848
TSP Top Surface Vertical Direction	ZPA	0.751	0.539
	Peak	3.912	2.377
SFP Bottom Surface Vertical Direction	ZPA	0.724	0.523
	Peak	3.674	2.314

TABLE 2.I.5
LOAD CASES AND ACCEPTANCE CRITERION APPLICABLE TO VVM COMPONENTS

Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data		Acceptance Criterion
			Magnitude of Loading (Ref. Table I.D.)	Value of Coincident Metal Temperature used (Deg. F)	
01	Condition with no MPC or Closure Lid installed; buoyancy from a water head equal to the distance between TOG and TOF.	• Temporary Cover • CEC Bottom Plate	Buoyant Force From CEC Displaced Volume < 8 psi	125 125	The minimum weight of the anti-buoyancy cover is 16,000lb. Maximum primary bending stress intensity in the CEC Bottom Plate must be below Level D limit.
02	Normal operation condition; dead load plus design basis explosion pressure	• Container Shell structure • Closure Lid	2.1.1; 3.I.1 2.1.1	125 350	Primary stresses do not exceed applicable Level A stress limits of ASME Subsection NF (or Level D limits with explosion)
03	Design basis missile	Closure Lid	2.1.1 and 2.2.5	350	Closure Lid does not collapse, is not dislodged from the cavity, and is not perforated by the missile.
04	Design basis earthquake	Container Shell	Figure 2.I.6	125	After the DBE event, MPC retrievability, subcriticality and confinement must not be compromised. Additional criteria for the CEC and its contents are defined in Table 2.I.6.
05	Closure lid handling	Lid Lift Lugs; all metal structure in Lid	1.15 x Closure Lid Weight (From Table 3.I.1)	125	ANSI N14.6 limits based on yield or ultimate strength including magnified inertia loads. Meet Reg. Guide 3.61 and Level A limits as applicable. (see Section 2.I.9)

TABLE 2.I.5 (continued)
LOAD CASES AND ACCEPTANCE CRITERION APPLICABLE TO VVM COMPONENTS

Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data		Acceptance Criterion
			Magnitude of Loading (Ref. Table I.D.)	Limiting Value of Coincident Metal Temperature (Deg. F)	
06	Design basis fire	Closure Lid	2.I.1	800	The Closure Lid structure does not collapse under its dead weight due to elevated metal temperatures.
07	CEC loading from subgrade	Container Shell	Calculated in 3.I	125	Service A stress limit for NF Class 3 plate and shell structure for the maximum "body extensive" membrane plus bending stress (body extensive defined as the region whose characteristic dimension exceeds 2.5 SQRT (R*T), where R and T are, respectively, the radius and thickness of the CEC shell.

Note 1. Structural loads and acceptance criteria for each load case are further explained in Section 2.I.9.
Note 2: Materials of construction are identified in Table 2.I.8.
Note 3: Design attributes of the VVM are explained in Section 1.I.2 and details are presented in the drawings in Section 1.I.5.
Note 4: The limiting value of coincident metal temperature is used to establish material properties and allowable stress (or stress intensity) when applicable.

TABLE 2.I.6
ACCEPTANCE CRITERIA FOR THE HI-STORM 100U VVM AND INTERNALS UNDER EXTREME ENVIRONMENTAL CONDITIONS

Component	Calculated Value	Allowable Limit
CEC Container Shell and Divider Shell	Radial gap between CEC Shell and Divider Shell Insulation after the seismic event	Nominal Gap (based on OD of Divider Shell Insulation and ID of CEC Shell) must remain open at end of event.
MPC Guides	Maximum compressive load	Minimum of limiting buckling load or ultimate load
MPC Shell	Longitudinal flexural stress intensity in shell wall from bending of the MPC shell as a beam. The local true strain in the MPC shell in the region of MPC guide/MPC impact.	ASME Level D primary membrane stress intensity limit The local strain from impact must be less than 10%, which has been established as a conservative limit in [3.1.31]
MPC Fuel Basket	Primary flexural stress intensity in basket panel from bending of the fuel basket as a beam	ASME Level D primary membrane stress intensity limit
MPC Fuel Basket	Maximum transverse bending stress in most heavily loaded basket panel, averaged over the panel length	ASME Level D primary membrane + bending stress intensity limit

Table 2.I.7
Intentionally Deleted

TABLE 2.I.8
PERMISSIBLE MATERIALS FOR HI-STORM 100U VVM COMPONENTS AND ISFSI STRUCTURES

	Primary Function	Part	ITS Category	Material (note6)	Normal Storage (Long-Term Limit)	Max. Permissible Temperature (°F) Off-normal, extreme environmental phenomena, and accident conditions	Special Surface Finish/ Coating (note 1)	Interfacing Matl. (if dissimilar)
1	Shielding	Closure Lid Concrete	C	Shielding Concrete per Appendix 1.D (note 2)	300 (note 3)	350 (note 3)	NA	Steel
2	Shielding	Closure Lid Steel	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Concrete/Elastomer
3	Structural	CEC (Container Shell, Bottom Plate and Container Flange)	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Subgrade/Concrete
4	Thermal	Insulation	C	Commercial	800	800	NA	Steel
5	Thermal	Inlet/Outlet Vent Screens and associated hardware	NITS	Carbon steel, stainless steel, aluminum, a polymeric fabric capable of 400°F (min.) service temperature or commercial	800 (note 4) if all metallic 400 otherwise	800 (note 4) if all metallic 400 otherwise	(note 5)	variable
6	Thermal	Outlet Vent Cover and associated hardware	NITS	Carbon steel, stainless steel, aluminum or commercial	800 (note 4)	800 (note 4)	(note 5)	variable
7	Thermal	Divider Shell and Divider Shell Restraints	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Insulation

TABLE 2.I.8 (continued)
PERMISSIBLE MATERIALS FOR HI-STORM 100U VVM COMPONENTS AND ISFSI STRUCTURES

	Primary Function	Part	ITS Category	Material (note 6)	Max. Permissible Temperature (°F)		Special Surface Finish/ Coating (note 1)	Interfacing Matl. (if dissimilar)
					Normal Storage (Long-Term Limit)	Off-normal, extreme environmental phenomena, and accident conditions		
8	Structural	Upper and Lower MPC Guides	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	-
9	Structural	MPC Bearing Pads	C	Carbon Steel (with stainless steel liners)	800 (note 4)	800 (note 4)	(note 5)	Stainless steel
10	Shielding and Physical Protection to the CEC	VVM Interface Pad (VIP)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	Steel
11	Shielding and Physical Protection	Top Surface Pad (TSP)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	—
12	Shielding and Physical Protection	Subgrade Surrounding the VVMs	B	Engineered fill, natural soil, or treated soil	150	350	N/A	Steel or Concrete
13	Structural Support	Support Foundation Pad (SFP)	C	Reinforced Concrete per ACI-318 (2005)	150	350	N/A	Soil, rock, mud mat, piling, etc., as appropriate
14	Shielding and Physical Protection	Retaining Wall (if used)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	—

Note 1 Materials identified by a supplier's trademark may be replaced with an equivalent product after an appropriate evaluation of acceptability.
Note 2 All requirements are identical to the shielding concrete in aboveground HI-STORMs.
Note 3 Limit per Appendix 1.D.
Note 4 Permissible temperature limit from ASME Code, Section II, is used as guidance to define all long and short-term loading limits. The metal temperature limits do not apply to the fire event (see Subsection 2.I.6).
Note 5 Surface preservative per Subsection 3.I.4.
Note 6 Materials listed as "or equivalent" may be replaced with "equivalent materials" as defined in Table 1.0.1. The critical characteristics for these materials are given in Table 2.I.9.

TABLE 2.I.9
CRITICAL CHARACTERISTICS OF EQUIVALENT MATERIALS USED IN THE VVM COMPONENTS

Designated Material	Item	Critical Characteristic
ASTM A515 or A516, Gr. 70	Yield Strength	Yield strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures. Applicable Code year is the same as used for the above ground HI-STORM.
	Ultimate Strength	Ultimate strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures. Applicable Code year is the same as used for the above ground HI-STORM.
	Elongation	Elongation must equal or exceed value(s) for 515/516 Gr. 70
	Charpy Impact	Values that measure resistance to impact must equal or exceed corresponding values for 515/516 Gr. 70.

TABLE 2.I.10 CRITICAL CHARACTERISTICS OF MATERIALS REQUIRED FOR SAFETY EVALUATION OF STORAGE AND TRANSPORT SYSTEMS				
	Property	**Type**	**Purpose**	**Bounding Acceptable Value**
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of $\delta_{min.}$, %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at ambient conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum allowable creep rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Thermal conductivity (minimum averaged value in the range of ambient to maximum service temperature, t_{max})	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
8.	Minimum Emissivity	T	To ensure that the thermal calculations are performed conservatively	Min.
9.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
10.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. and Max.
11.	Boron-10 Content	R	To control reactivity	Min.

TABLE 2.I.11 LOAD COMBINATIONS FOR THE TOP SURFACE PAD, VVM INTERFACE PADS, SUPPORT FOUNDATION PAD, AND THE RETAINING WALL PER ACI-318 (2005)	
Load Combination	
LC-1	1.4D
LC-2	1.2D + 1.6L
LC-3	1.2D + E + L

where:
D: Dead Load including long-term differential settlement effects.
L: Live Load
E: DBE for the Site

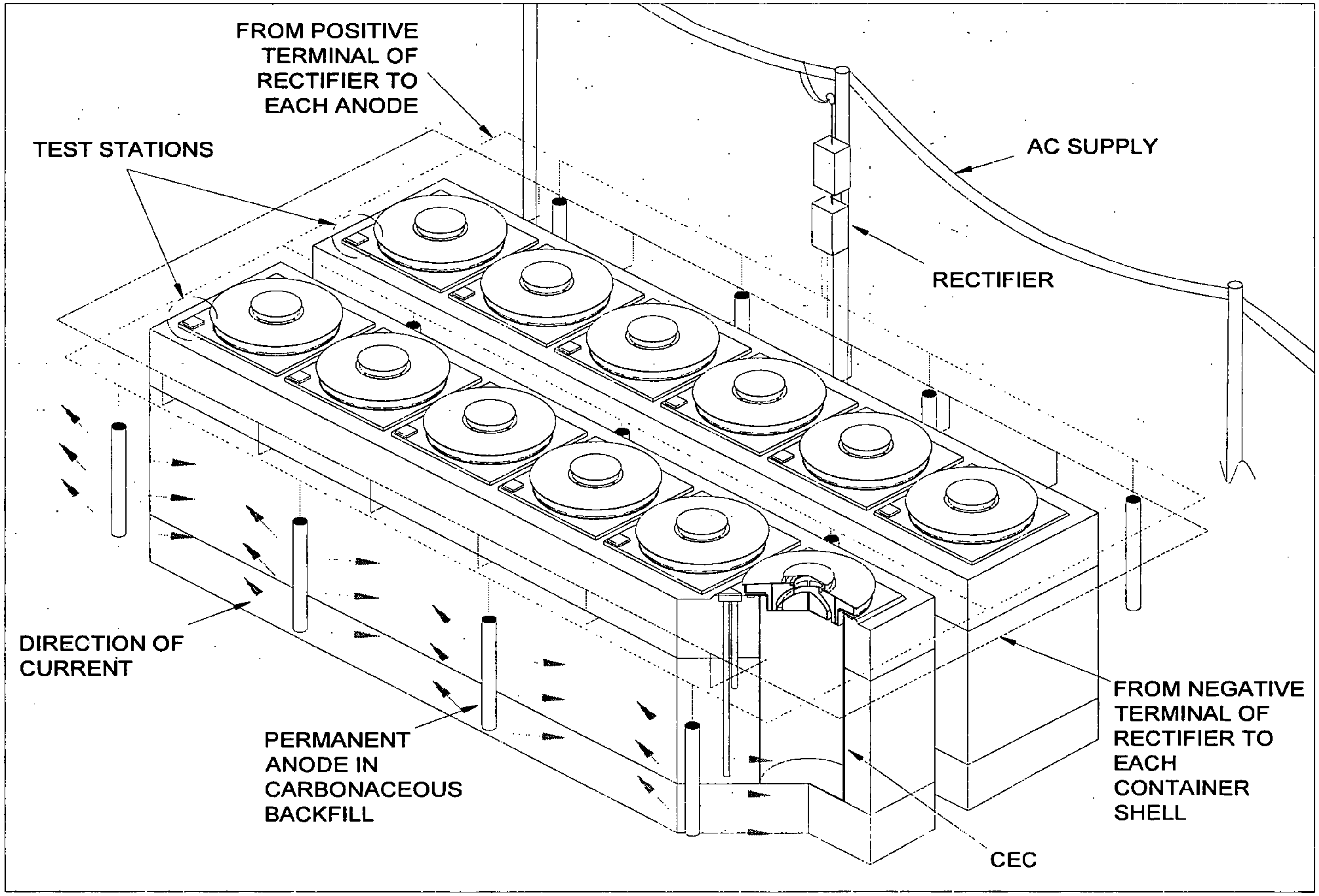


FIGURE 2.I.1: HI-STORM 100U SYSTEM EXAMPLE ICCPS DESIGN – 2 X 6 ARRAY DESIGN LAYOUT*

* The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Expansion joints between the VVM Interface Pad and the Top Surface Pad are not shown in this figure.

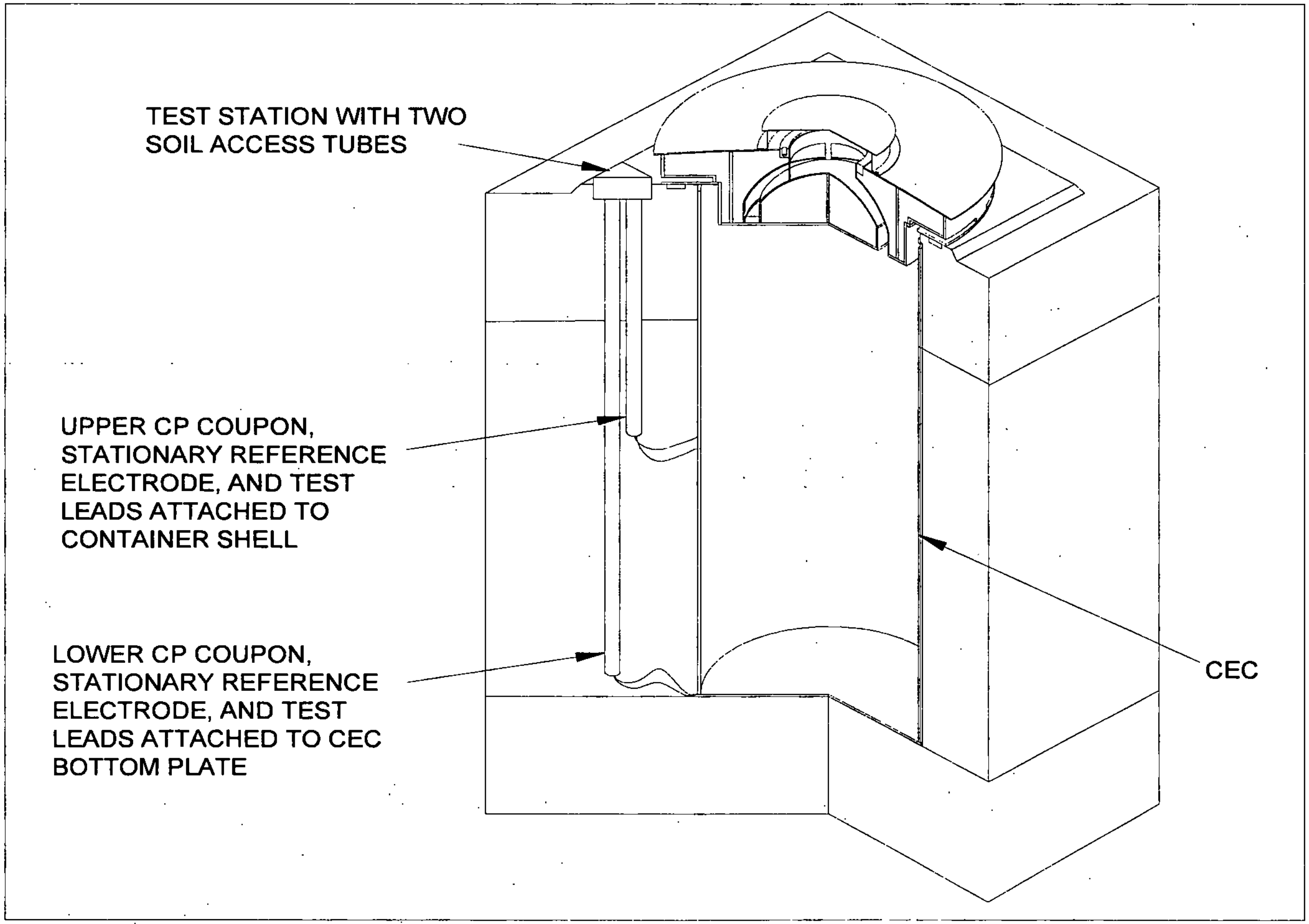


FIGURE 2.I.2: HI-STORM 100U SYSTEM EXAMPLE ICCPS DESIGN – TEST STATION*

*The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Expansion joints between VVM Interface Pad and Top Surface Pad are omitted from this figure.

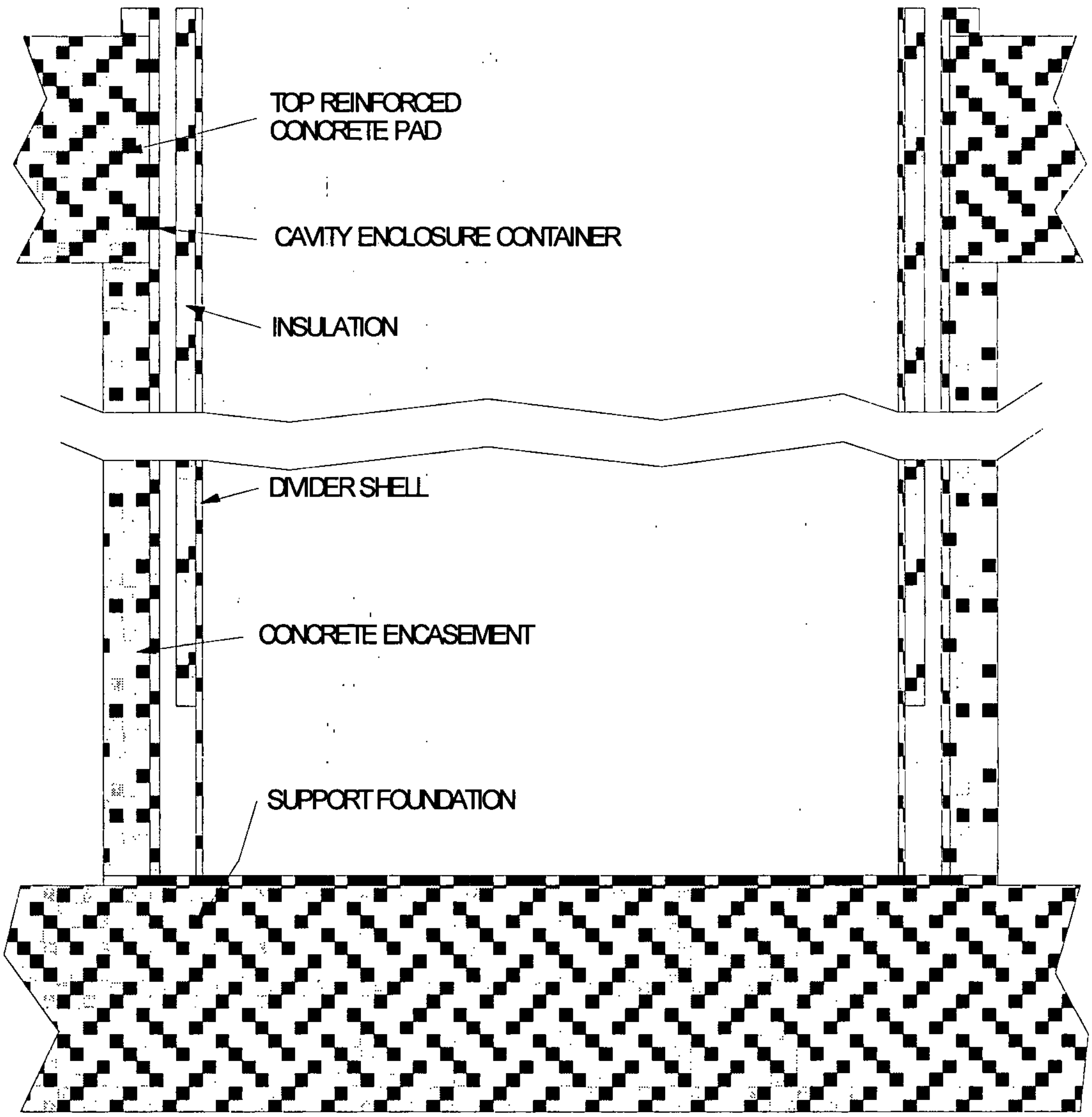
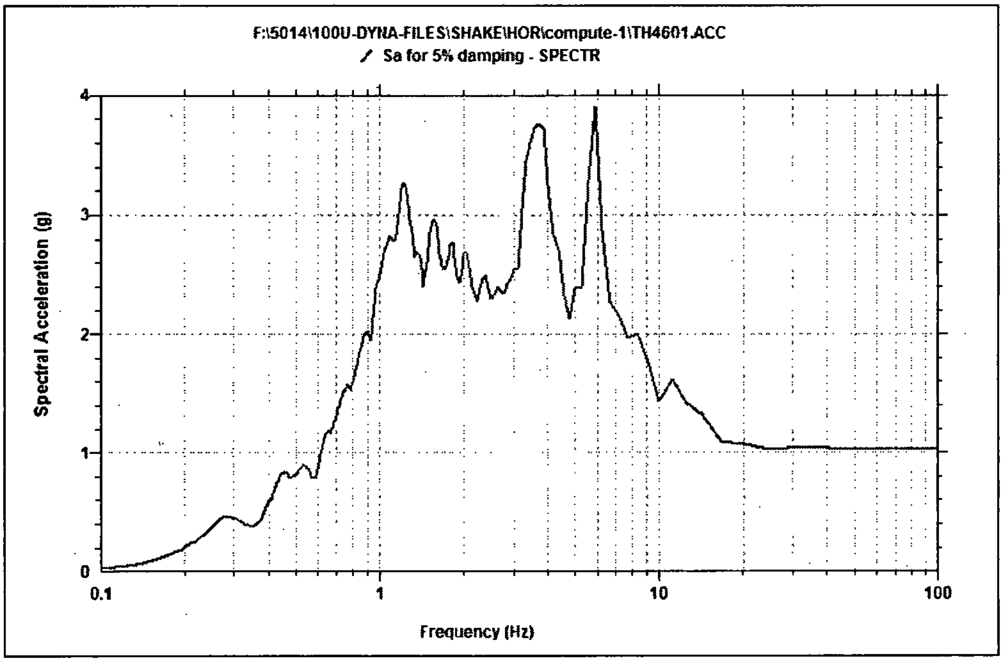
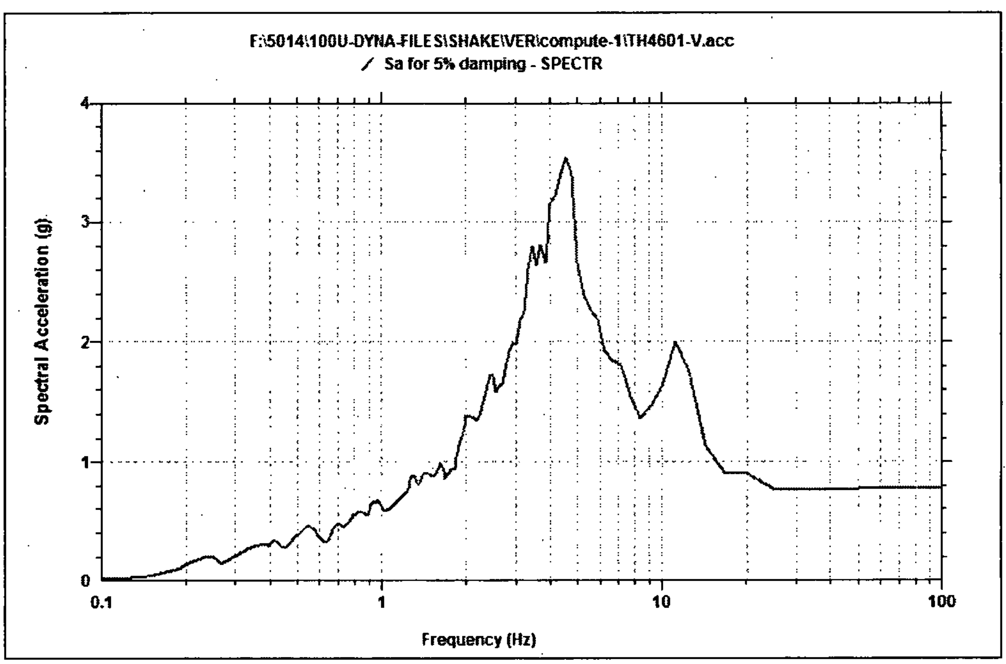


FIGURE 2.I.3: TYPICAL CONCRETE ENCASEMENT OF THE CEC

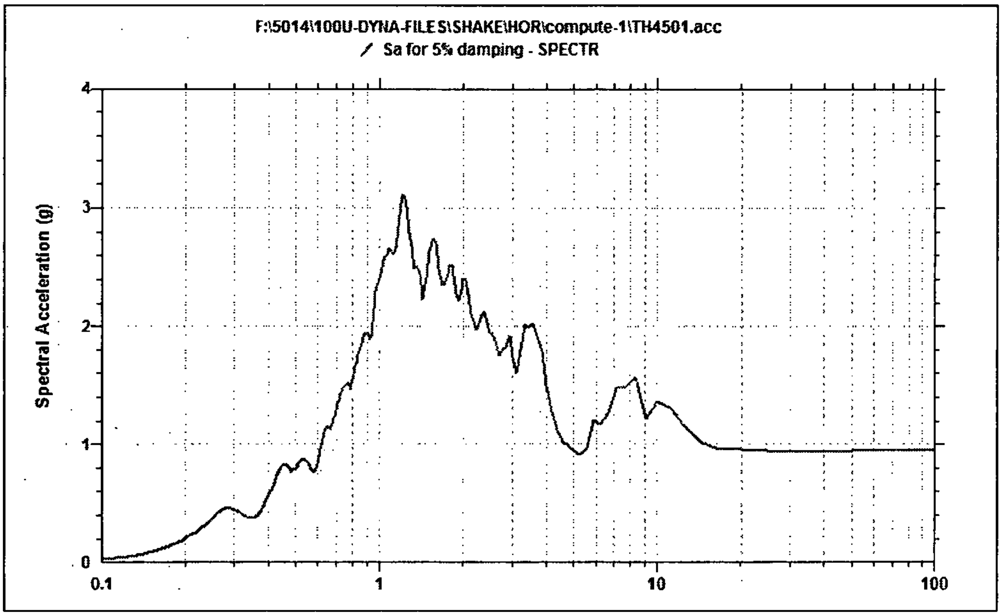


(a)

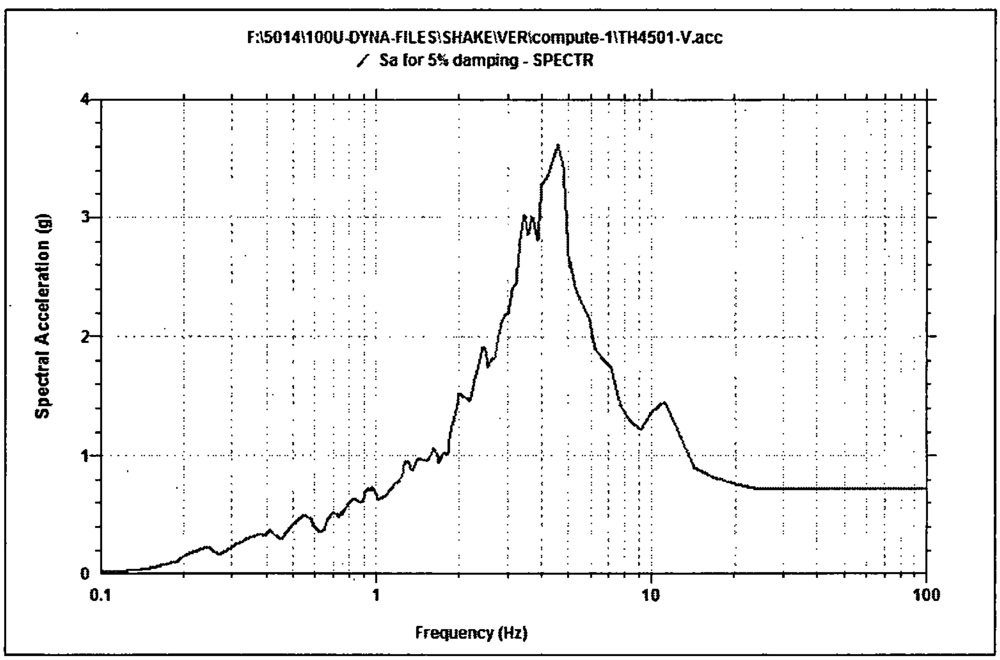


(b)

FIGURE 2.I.4-A: DESIGN BASIS SPECTRUM AT THE GROUND SURFACE (TOP OF TSP) ELEVATION
(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION



(a)



(b)

FIGURE 2.I.4-B: DESIGN BASIS SPECTRUM AT THE 100U FOUNDATION SURFACE (BOTTOM OF SFP) ELEVATION
(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION

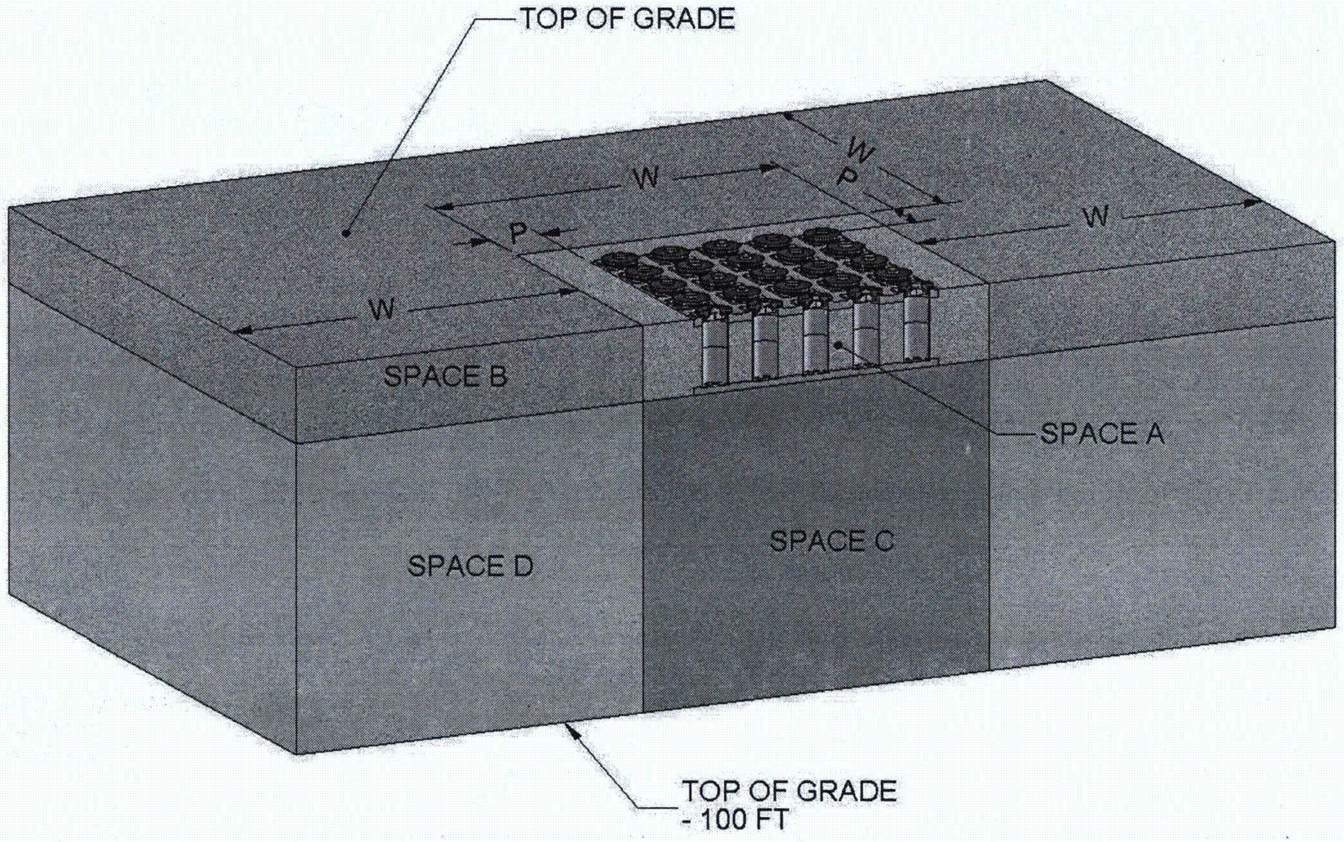


FIGURE 2.I.5 SUB-GRADE AND UNDER-GRADE SPACE NOMENCLATURE

Note: The figure shows a 5 by 5 array with a slice through the centerline of the first row of VVMs facing the reader. Space A is the lateral subgrade space in and around the VVMs which may be excavated and refilled with engineered fill. Space B is the lateral subgrade that extends by the amount W around the ISFSI where W is the characteristic dimension of the ISFSI. Space C is the undergrade below the SFP. Space D is the undergrade surrounding Space C. P is the distance to the Retaining wall.

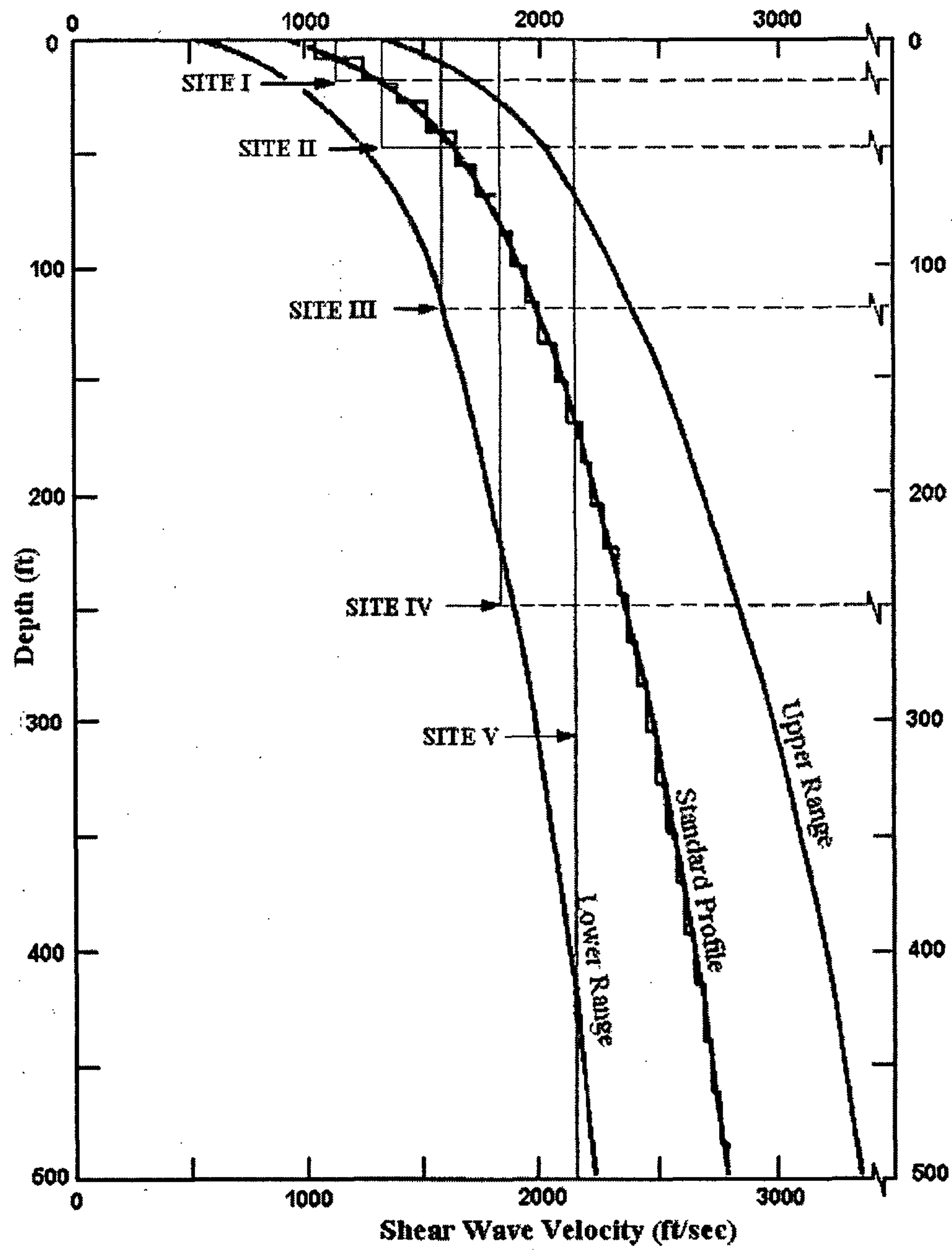


FIGURE 2.I.6 TYPICAL SHEAR WAVE VELOCITY PROFILES FOR NUCLEAR POWER PLANT SITES (REPRODUCED FROM FIGURE I-1 OF [2.I.10])

SUPPLEMENT 2.III

PRINCIPAL DESIGN CRITERIA FOR THE MPC-68M

2.III.0 OVERVIEW OF THE PRINCIPAL DESIGN CRITERIA

A general description of the MPC-68M is provided in Supplement 1.III. This supplement specifies the loading conditions and associated design criteria applicable to the MPC-68M fuel basket. The loads, loading conditions, and design criteria presented in Chapter 2 are applicable to the HI-STORM 100 System using the MPC-68M unless otherwise specified in this supplement. The drawing package for the MPC-68M fuel basket is provided in Section 1.5. The safety classifications of Metamic-HT basket and basket shims are ITS-A and NITS, respectively.

The design criteria pertaining to the HI-STORM overpack and the HI-TRAC transfer cask are completely unaffected by the incorporation of the MPC-68M. Likewise, the structural demands on the MPC Enclosure Vessel (whose design remains unchanged) are also unaffected. The design criteria in this supplement pertain to the loading conditions that bear upon the fuel basket's function and performance.

2.III.0.1 MPC-68M Design Criteria

i. Structural

The fuel basket is designed to meet a more stringent displacement limit under mechanical loadings than those implicit in the stress limits of the ASME code (see Section 2.III.2). The basket shims are designed to remain below the yield limit of the selected aluminum alloy. Fuel basket welds are designed and fabricated in accordance with Supplement 9.III and the drawing package in Section 1.5. Fuel basket structural welds are designed to the minimum weld strength specified in the drawing package in Section 1.5. Metamic-HT is a Holtec proprietary (non-ASME code) material. The *critical characteristics* and the attainment of the required critical characteristics through a comprehensive qualification process and production testing are discussed in Appendix A to Supplement 1.III with acceptance criteria established in Supplement 9.III.

All normal and off-normal conditions (including pressures) for the MPC-68M are the same as those described in Section 2.2. All loads on the HI-STORM 100 overpack and HI-TRAC transfer cask described in Section 2.2 remain applicable when using the MPC-68M.

The main acceptance criterion for the evaluation of accident conditions on the MPC-68M fuel basket is for the basket structure to maintain the fuel contents in a subcritical configuration. The structural design criteria for the MPC-68M basket are provided in Table 2.III.4.

The fuel basket material is subject to the requirements in Appendix 1.III.A and is designed to a specific (lateral) deformation limit of its walls under accident conditions of loading (credible and non-mechanistic). The basis for the lateral deflection limit in the active fuel region, θ, is provided in [2.III.6.1] as

$$\theta = \frac{\delta}{w}$$

where δ is defined as the maximum total deflection sustained by the basket panels under the loading event and w is the nominal inside (width) dimension of the storage cell. The limiting value of θ is provided in Table 2.III.4. The above deflection-based criterion has been used previously in the HI-STAR 180 Transportation Package [2.III.6.2] to qualify similar Metamic-HT fuel baskets.

ii. Thermal

The design and operation of the HI-STORM 100 System with the MPC-68M must meet the intent of the review guidance contained in ISG-11, Revision 3 [2.0.8] as described in Subsection 2.0.1.

All applicable material design temperature limits in Section 2.2 and 4.3 continue to apply to the MPC-68M. Temperature limits of MPC-68M fuel basket and basket shim materials are specified in Table 4.III.2.

The MPC-68M is designed for both uniform and regionalized fuel loading strategies as described in Subsection 2.0.1. The regions for the MPC-68M are given in Table 2.III.1.

iii. Shielding

Same as Subsection 2.0.1.

iv. Criticality

Same as Subsection 2.0.1 with the clarifications herein.

Criticality control is maintained by the geometric spacing of the fuel assemblies and spatially distributed B-10 isotope in the Metamic-HT. No soluble boron is required in the MPC-68M water. The minimum specified boron concentration in the Metamic-HT purchasing specification must be met in every lot of the material manufactured. No credit is taken for burnup. Enrichment limits are delineated in Table 2.III.2.

v. Confinement

Same as Subsection 2.0.1

vi. Operations

Same as Subsection 2.0.1. Generic operating procedures for the HI-STORM 100 System with MPC-68M are provided in Chapter 8 with certain limitations and clarifications provide in Supplement 8.III.

vii. Acceptance Tests and Maintenance

Same as Subsection 2.0.1. The acceptance criteria for the HI-STORM 100 System with MPC-68M are provided in Chapter 9 and Supplement 9.III.

vi. Decommissioning

The MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM 100 System is addressed in Section 2.III.4.

2.III.1 SPENT FUEL TO BE STORED

Table 2.1.22 and the limitations/clarification in this supplement provide the limits for material to be stored in the MPC-68M. All BWR fuel assembly array/classes which are authorized for the MPC-68 are authorized in the MPC-68M except fuel assembly array/classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D, and 10x10E. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for the fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt.% U-235 are authorized in the MPC-68M. The maximum planar-average initial enrichments acceptable for loading in the MPC-68M, for each fuel assembly array/class given in Table 2.1.4, are provided in Table 2.III.2. Table 2.III.3 provides the description of two new fuel assembly array/classes which are added as acceptable contents to the MPC-68M only, 10x10F and 10x10G. No credit is taken for fuel burnup or integral poisons such as gadolinia for any fuel assembly array/class. The maximum allowable initial enrichment for fuel assemblies are consistent with the criticality analysis described in Supplement 6.III. See Table 2.1.29 for assembly cooling time-dependent coefficients for 10x10F and 10x10G.

Fuel classified as damaged fuel assemblies or fuel debris will be loaded into damaged fuel containers (DFCs) for storage in the MPC-68M. The appropriate thermal and criticality analyses have been performed to account for damaged fuel and fuel debris and are described in Supplements 4.III and 6.III, respectively. The restrictions on the number and location of damaged fuel containers authorized for loading in the MPC-68M is the same as MPC-68 (see Section 1.III.2.3 and Figure 1.III.2). Non-fuel hardware is not applicable to all the BWR fuel classes/arrays.

The heat generation rate, axial burnup distribution, and all other bounding radiological, thermal, and criticality parameters specified for MPC-68 are used to ensure the performance of the HI-STORM SYSTEM with the MPC-68M.

2.III.2 MPC-68M DESIGN LOADINGS

Design loadings in Section 2.2 apply to the HI-STORM 100 System using the MPC-68M.

2.III.3 SAFETY PROTECTION SYSTEMS

Same as Section 2.3.

2.III.4 DECOMMISSIONING CONSIDERATION

Same as Section 2.4.

2.III.5 REGULATORY COMPLIANCE

Same as Section 2.5.

2.III.6 REFERENCES

[2.III.6.1] Holtec Proprietary Position Paper DS-331, "Structural Acceptance Criteria for the Metamic-HT Fuel Basket", (USNRC Docket No. 71-9325).

[2.III.6.2] HI-STAR 180 Transportation Package, USNRC Docket No. 71-9325.

Table 2.III.1: MPC-68M FUEL STORAGE REGIONS

Number of Storage Cells		Storage Cell IDs*	
Inner Region (n_1)	Outer Region (n_2)	Inner Region	Outer Region
32	36	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All other locations
* See Figure 1.III.2 for storage cell numbering			

Table 2.III.2: BWR FUEL ASSEMBLY INITIAL ENRICHMENTS FOR LOADING IN MPC-68M (Note 1)

Fuel Assembly Array and Class	Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 3)
7x7 B	4.8
8x8 B	4.8
8x8 C	4.8
8x8 D	4.8
8x8 E	4.8
8x8 F	4.5 (Note 2)
9x9 A	4.8
9x9 B	4.8
9x9 C	4.8
9x9 D	4.8
9x9 E	4.5 (Note 2)
9x9 F	4.5 (Note 2)
9x9 G	4.8
10x10 A	4.8
10x10 B	4.8
10x10 C	4.8

Notes:

1. All other fuel assembly array/class specifications from Table 2.1.4 apply.

2. Fuel assemblies classified as damaged fuel assemblies are limited to 4.0 wt.% U-235.

3. For MPC-68M 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 the enrichment of the damaged assembly.

Table 2.III.3: BWR FUEL ASSEMBLY CHARACTERISTICS FOR LOADING IN MPC-68M (Note 1)

Fuel Assembly Array and Class	**10x10F**	**10x10G**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 188
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 8)	4.7 (Note 7)	4.6 (Note 7)
Initial Rod Maximum Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 4)	96/84
Fuel Clad O.D. (in.)	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150
No. of Water Rods (Note 6)	2	5 (Note 5)
Water Rod Thickness (in.)	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.120	≤ 0.060

Table 2.III.3 (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."

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

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

6. These rods may also be sealed at both ends and contain ZR material in lieu of water.

7. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% 235U for the 10x10F array/class and 4.0 wt.% 235U for the 10x10G array/class.

8. For MPC-68M 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 the enrichment of the damaged assembly.

Table 2.III.4 STRUCTURAL DESIGN CRITERIA FOR THE FUEL BASKET	
PARAMETER	ALLOWABLE VALUE
Minimum service temperature	-40°F
Maximum total (lateral) deflection in the active fuel region, θ (dimensionless)	0.005