FINAL SAFETY ANALYSIS REPORT

for the

HOLTEC INTERNATIONAL

STORAGE AND TRANSFER
OPERATION

REINFORCED MODULE CASK SYSTEM

(HI-STORM 100 CASK SYSTEM)

DOCKET 72-1014

VOLUME I OF II

HOLTEC
INTERNATIONAL

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Sincerely,

Brian Gutherman, P.E.
Licensing Manager

Approval:

K.P. Singh, Ph.D, P.E.
President and CEO

cc: Ms. Marissa Bailey, USNRC (w/2 hard copies of FSAR Rev. 0)
Mr. E. William Brach, USNRC (w/o encl.)

Document ID: 5014395

Enclosure: Revision 0 of the HI-STORM 100 System Final Safety Analysis Report.

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July 19, 2000

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Subject: USNRC Docket No. 72-1014; TAC No. L22221
HI-STORM 100 Certificate of Compliance 1014
HI-STORM FSAR, Revision 0

References: 1. Holtec Project 5014
2. NRC Letter, E.W. Brach, to Holtec International, K.P. Singh, dated May 4, 2000

Dear Sir:

On May 4, 2000 the Nuclear Regulatory Commission issued the final 10 CFR 72 Certificate of Compliance (CoC) and Safety Evaluation Report for the HI-STORM 100 dry cask spent fuel storage system. The CoC became effective on May 31, 2000. In accordance with 10 CFR 72.248(a)(1), enclosed herewith please find one copy of Holtec Report No. HI-2002444, Revision 0, comprising the HI-STORM 100 System Final Safety Analysis Report (FSAR). The HI-STORM FSAR is identical to HI-STORM TSAR, Revision 10 with the following clarifications:

1. The term "TSAR" is changed to "FSAR" throughout, except where reference is made to the HI-STAR 100 System, where the term "TSAR" is still correct.

2. A small number of typographical errors have been corrected.

3. An editorial clarification is made to Subsection 3.4.7.1 to conform to a change made in Chapter 2 (Table 2.2.9) during final licensing to require verification by testing of the coefficient of friction between the HI-STORM 100 cask and the ISFSI pad. Since testing is required, the requirement for a "broom finish" on the ISFSI pad concrete has been made optional at the user's discretion.

4. Figure 5.1.4 has been revised to more accurately depict the HI-TRAC 100 lid design.

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3.B-44	0	Fig. 3.D.5a	0
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3.AK-18	0		3.AN-11	0
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5.6-1	0		5.C-40	0
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5.B-2	0		5.C-46	0
5.B-3	0		5.C-47	0
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6.1-6	0	6.2-45	0
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6.2-32	0	Fig. 6.4.2	0
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7.1-4	0	7.A-21	0
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7.2-5	0	7.A-30	0
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7.2-7	0	7.A-32	0
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7.3-3	0	7.A-35	0
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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 application seeks 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 applicant. This report is also suitable for incorporation into a site-specific Safety Analysis Report which may be submitted by an applicant for the 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 requirements. 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

† 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).

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.

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 safety 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 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.

This document is exactly the same as Revision 10 of the HI-STORM Topical Safety Analysis Report (TSAR) with the following clarifications:

- typographical errors have been corrected,

- pagination may be slightly different, and

- the term "TSAR" has been replaced with the term "FSAR" or "SAR" when referring to the HI-STORM document (references to the HI-STAR TSAR remain).

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.

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.

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, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. A damaged fuel assembly's inability to be handled by normal means may be due to mechanical damage and must not be due to fuel cladding damage.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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

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

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

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

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

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

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

Fuel Debris refers to ruptured fuel rods, severed rods, and loose fuel pellets with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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. The HI-TRAC is equipped with a pair of lifting trunnions and pocket trunnions to lift and downend/upend the HI-TRAC with a loaded MPC. HI-TRAC is an acronym for **H**oltec **I**nternational **Tra**nsfer **C**ask. In this submittal there are two HI-TRAC transfer casks, the 125 ton HI-TRAC (HI-TRAC-125) and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a maximum crane capacity of 100 tons. The term HI-TRAC is used as a generic term to refer to both the 125 ton and 100 ton HI-TRAC.

HI-STORM 100 overpack or storage overpack means the cask which 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.

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

Holtite™ is a trade name denoting an approved neutron shield material for use in the HI-STORM 100 System. In this application, Holtite-A is the only approved neutron shield material.

Holtite™-A is a commercially available neutron shield material developed by Bisco, Inc., and currently sold under the trade name NS-4-FR. The neutron shield material is specified with a minimum B_4C loading of 1 weight percent. An equivalent neutron shield material with equivalent neutron shielding properties and composition, but not sold under the trade name NS-4-FR, may be used.

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.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

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

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.

METCON™ is a trade name for the HI-STORM 100 overpack. The trademark is derived from the **metal-con**crete composition of the HI-STORM 100 overpack.

MGDS is an acronym for Mined Geological Disposal System.

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions. The MPCs used as part of the HI-STORM 100 System are identical to the HI-STAR 100 MPCs evaluated in the HI-STAR 100 storage (Docket No. 72-1008) and transport (Docket No. 71-9261) applications.

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

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

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

PWR is an acronym for pressurized water reactor.

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

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.

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.

TSAR is an acronym for Topical Safety Analysis Report (10CFR72).

ZPA is an acronym for zero period acceleration.

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 Characteristics	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

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. 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
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

Table 1.0.2 (continued)

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

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
NA	2.III.3.c Thermal	10CFR72.128(a)(4)	2.3.2.2, 4.0
NA	2.III.3.f Operating Procedures	10CFR72.24(f) 10CFR72.128(a)(5)	10.0, 8.0
		10CFR72.236(h)	8.0
		10CFR72.24 (l)(2)	1.2.1, 1.2.2
		10CFR72.236(l)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122 (l) 10CFR72.236 (g) 10CFR72.122 (f) 10CFR72.128 (a)(1)	9.0
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a)(2)	2.3.5.2
		10CFR72.128(a)(3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
		10CFR72.236(e)	2.3.2.1
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h)(4) 10CFR72.122(i) 10CFR72.128(a)(1)	2.3.5

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
2.3.4 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
	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

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
3.2 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
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions	--	3.4.3, Appendices 3.E, 3.AE, 3AC, 3.D
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4, Appendices 3.I, 3.U, 3.V, 3.W

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
3.4.5 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)	4.1
		10CFR72.236(h)	
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
N/A	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
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 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
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configurations	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 Calculational Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4

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

Table 1.0.2 (continued)

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

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

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. 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
	8.III.3 Radioactive Effluent Control	10CFR72.24(l)(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(l)(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

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.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.8 Ready Retrieval	10CFR72.122(l)	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
	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(l)	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)]

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
9.2 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)(l)	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) 10CCR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	10.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3

Table 1.0.2 (continued)

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

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
NA	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	

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. 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) 10CFR72.122(b)(2) 10CFR72.122(b)(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(l)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122(h)(4)	7.1.4

Table 1.0.2 (continued)

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

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
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)(l)	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 12.III.2.b Enrichment 12.III.2.c Burnup 12.III.2.d Minimum Acceptable Cooling Time 12.III.2.f Maximum Spent Fuel Loading Limit	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.g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2.e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.5 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SCC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
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)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Notes:

(1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application.

(2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission's acceptance of the tests defined in Chapter 9.

(3) Not applicable to HI-STORM 100 System. The functional adequacy of all important to safety components is demonstrated by analyses.

(4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.

(5) The stated requirement is not applicable to the HI-STORM 100 System. No monitoring is required for accident conditions.

"--" There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.

"NA" There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3
HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	**Alternate Method to Meet NUREG-1536 Intent**	**Justification**
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 80EF on the annual average ambient temperatures for the United States.	The 80EF 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 80EF. 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 80EF 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.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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 TSAR, 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 type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses are required to demonstrate that the deceleration limit of 45g's is not exceeded.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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 ACI 359 should comply with the requirements of ACI 349".	<u>Exception:</u> The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used for the material selection and specification, and construction of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack. ACI 318-95 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.	Concrete is provided in the HI-STORM overpack solely for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose. To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as specified in Appendix 1.D. In addition, the temperature limits for normal and off-normal condition from ACI 349 will be imposed. 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."	<u>Clarification:</u> The HI-STORM overpack steel structure is designed in accordance with the ASME B&PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.	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.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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." 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."	Clarification: As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak fuel cladding temperature limit.	As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak fuel cladding temperature limit. All major variations in fuel parameters are considered in the determination of the peak fuel cladding temperature limits. Minor variations in fuel parameters within an array type are bounded by the conservative determination of the peak fuel cladding temperature limit.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
4.V.4.a, Page 4-6, Para. 3 "applicants seeking NRC approval of specific internal convection models should propose, in the SAR, a comprehensive test program to demonstrate the adequacy of the cask design and validation of the convection models."	Exception: The natural convection model described in Subsection 4.4.1.1.5 is based on classical correlations for natural convection in differentially heated cavities which have been validated by many experimental studies. Therefore, no additional test program is proposed.	Many experimental studies of this mechanism have been performed by others and reported in open literature sources. As discussed in Subsection 4.4.1.1.5, natural convection has been limited to the relatively large MPC basket to shell peripheral gaps. Subsection 4.4.1.1.5 provides sufficient references to experiments which document the validity of the classical correlation used in the analysis.
4.V.4.a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."	Clarification: As discussed in Subsection 4.4.2, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based update of Wooten-Epstein (described in Subsection 4.4.1.1.2)	The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature *difference* conservatively maximizes the basket wall temperature.
4.V.4.b, Page 4-7, Para. 2 "if the thermal model is axisymmetric or three-dimensional, the longitudinal thermal conductivity should generally be limited to the conductivity of the cladding (weighted fractional area) within the fuel assembly."	Clarification: As described in Subsection 4.4.1.1.4, the axial thermal conductivity of the fuel basket is set equal to the cross-sectional thermal conductivity.	Due to the large number of gaps in the cross-sectional heat transfer paths, use of the fuel basket cross-sectional thermal conductivity for the axial thermal conductivity severely underpredicts the axial thermal conductivity of the fuel basket region. This imposed axial thermal conductivity restriction is even more limiting than that imposed by this requirement of NUREG-1536.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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, described in Subsection 4.4.1.1.2, use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.	Within Subsection 4.4.1.1.2, the calculated effective thermal conductivities based on nominal design fuel dimensions are compared with available literature values and are demonstrated to be conservative by a substantial margin.
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, 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."	Exception: 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.

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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.	Exception: All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) are calculated using maximum weights and minimum densities.	Calculating the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) using maximum weights and minimum densities conservatively overpredicts the volume occupied by the internal components and correspondingly underpredicts the remaining free volume.
7.V.4.c, Page 7-7, Para. 2 and 3 "Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145 is not typically applied." and "Note that for an instantaneous release (and instantaneous exposure), the time that an individual remains at the controlled area boundary is not a factor in the dose calculation."	Exception: As described in Section 7.3, in lieu of an instantaneous release, the assumed leakage rate is set equal to the leakage rate acceptance criteria ($5x10^{-6}$ atm-cm^3/s) plus 50% for conservatism, which yields $7.5x10^{-6}$ atm-cm^3/s. Because the release is assumed to be a leakage rate, the individual is assumed to be at the controlled area boundary for 720 hours. Additionally, the atmospheric dispersion factors of Regulatory Guide 1.145 are applied.	The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g., helium leakage, hydrostatic, and volumetric weld inspection). The NRC letter to Holtec International dated 9/15/97, Subject: Supplemental Request for Additional Information - HI-STAR 100 Dual Purpose Cask System (TAC No. L22019), RAI 7.3 states "use the verified confinement boundary leakage rate in lieu of the assumption that the confinement boundary fails."

Table 1.0.3 (continued)

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

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."	Clarification: 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.
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.

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

The HI-STORM 100 overpack is constructed from a combination of steel and concrete, both of which are materials with long, proven histories of usage in nuclear applications. HI-STORM 100 incorporates and combines many desirable features of previously-approved concrete and metal module designs. In essence, the HI-STORM 100 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.

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

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

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

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

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

The modularity of the HI-STORM 100 System accrues several advantages. Different MPCs, identical in exterior dimensions, manufacturing requirements, and handling features, but different in their SNF arrangement details, are designed to fit a common overpack. Even though the different MPCs have fundamentally identical design and manufacturing attributes, qualification of HI-STORM 100 requires consideration of the variations in the characteristics of the MPCs. In most cases, however, it is possible to identify the most limiting MPC geometry and the specific loading condition for the safety evaluation, and the detailed analyses are then carried out for that bounding condition. In those cases where this is not possible, multiple parallel analyses are performed.

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

For example:

- The HI-STORM 100 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 100 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 the ISFSI storage pad.

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 a MPC.

The MPC is identical to those presented in References [1.1.2] and [1.1.3]. Referencing these submittals avoids repetition of information on the MPCs which is comprehensively set forth in the above-mentioned Holtec International applications 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 100 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. A cross sectional view of the HI-STORM 100 storage overpack is presented in Figure 1.1.3. 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.

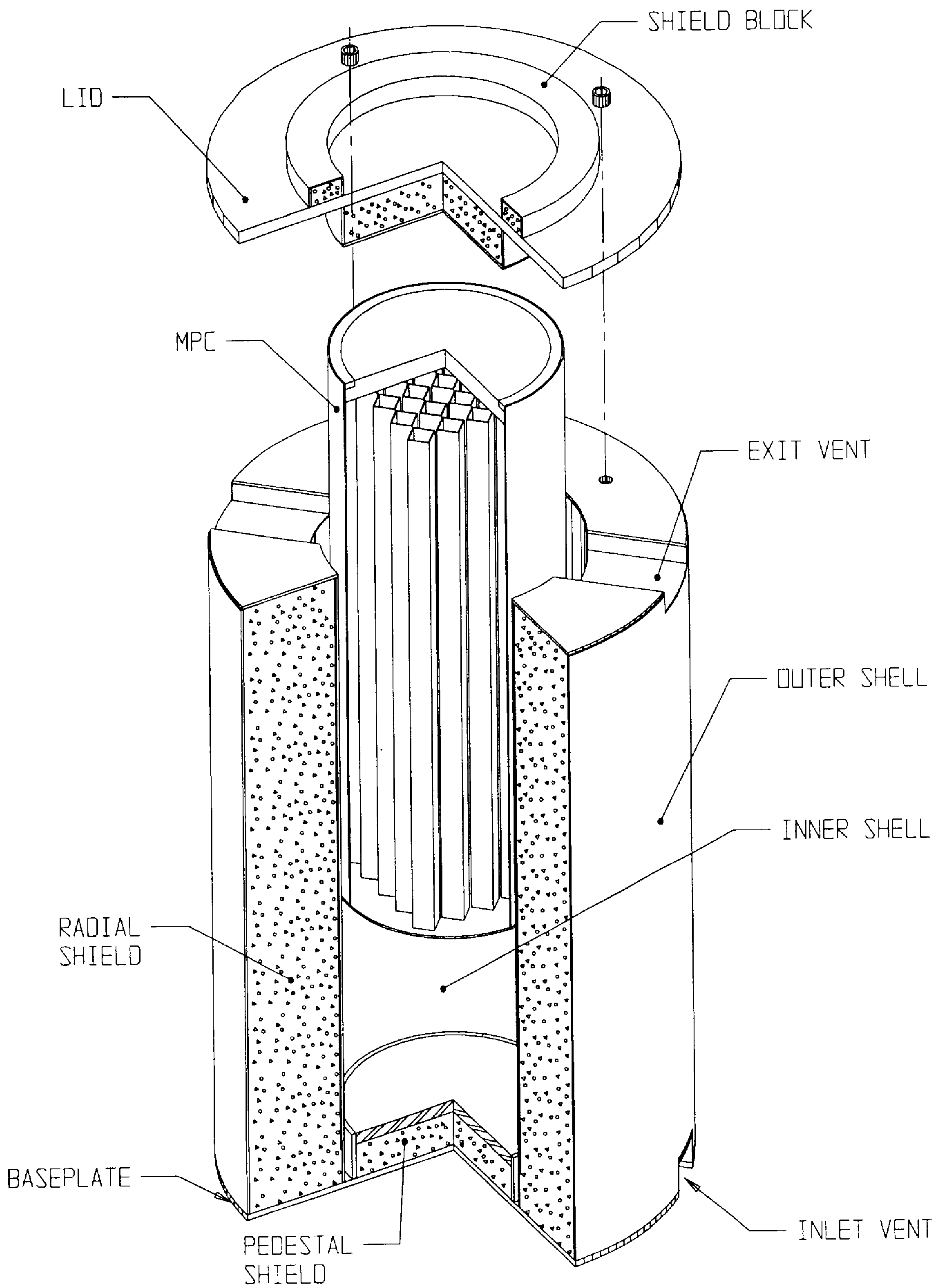

FIGURE 1.1.1; HI-STORM 100 OVERPACK WITH MPC PARTIALLY INSERTED

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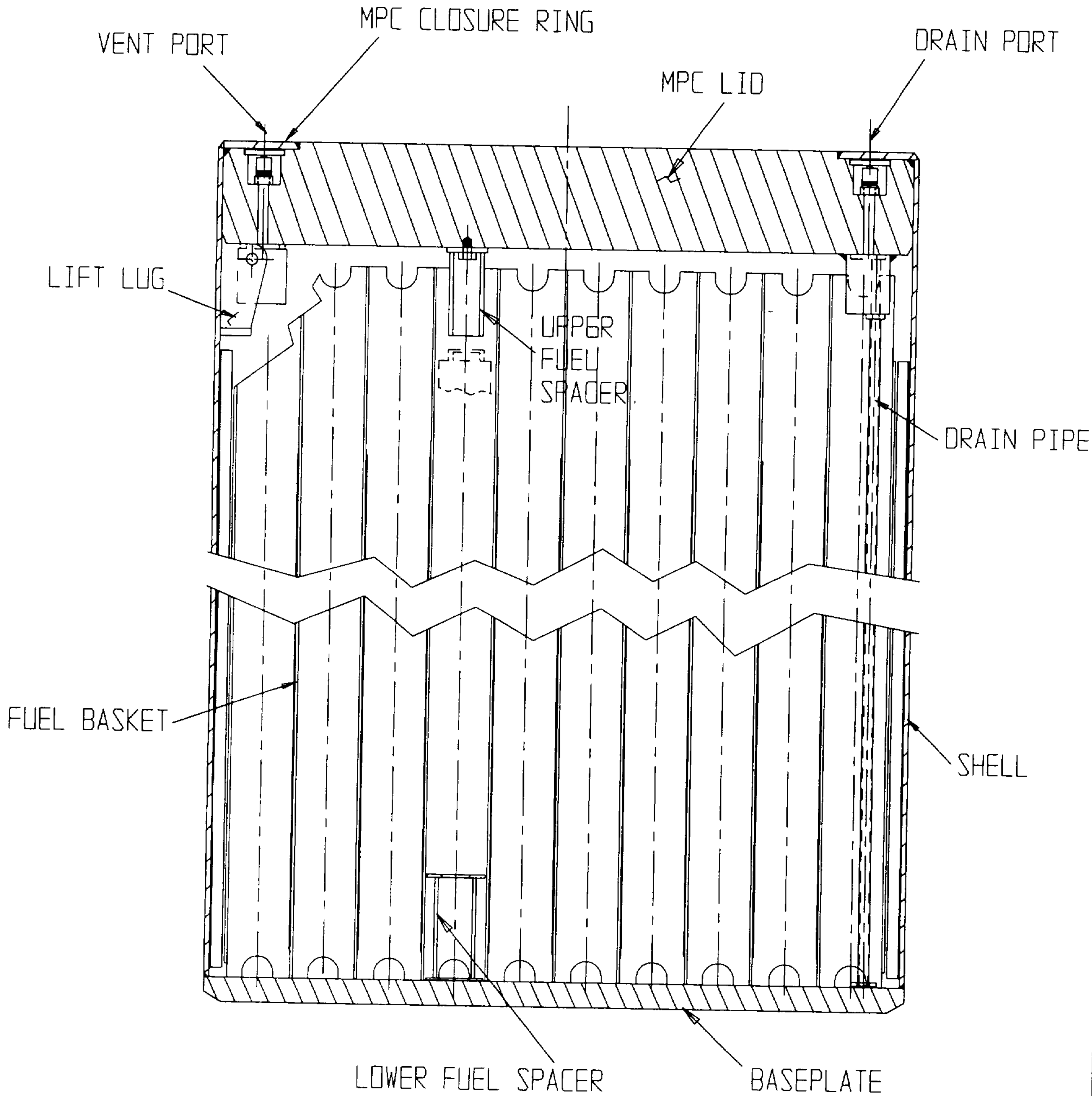

FIGURE 1.1.2; CROSS SECTION ELEVATION VIEW OF MPC

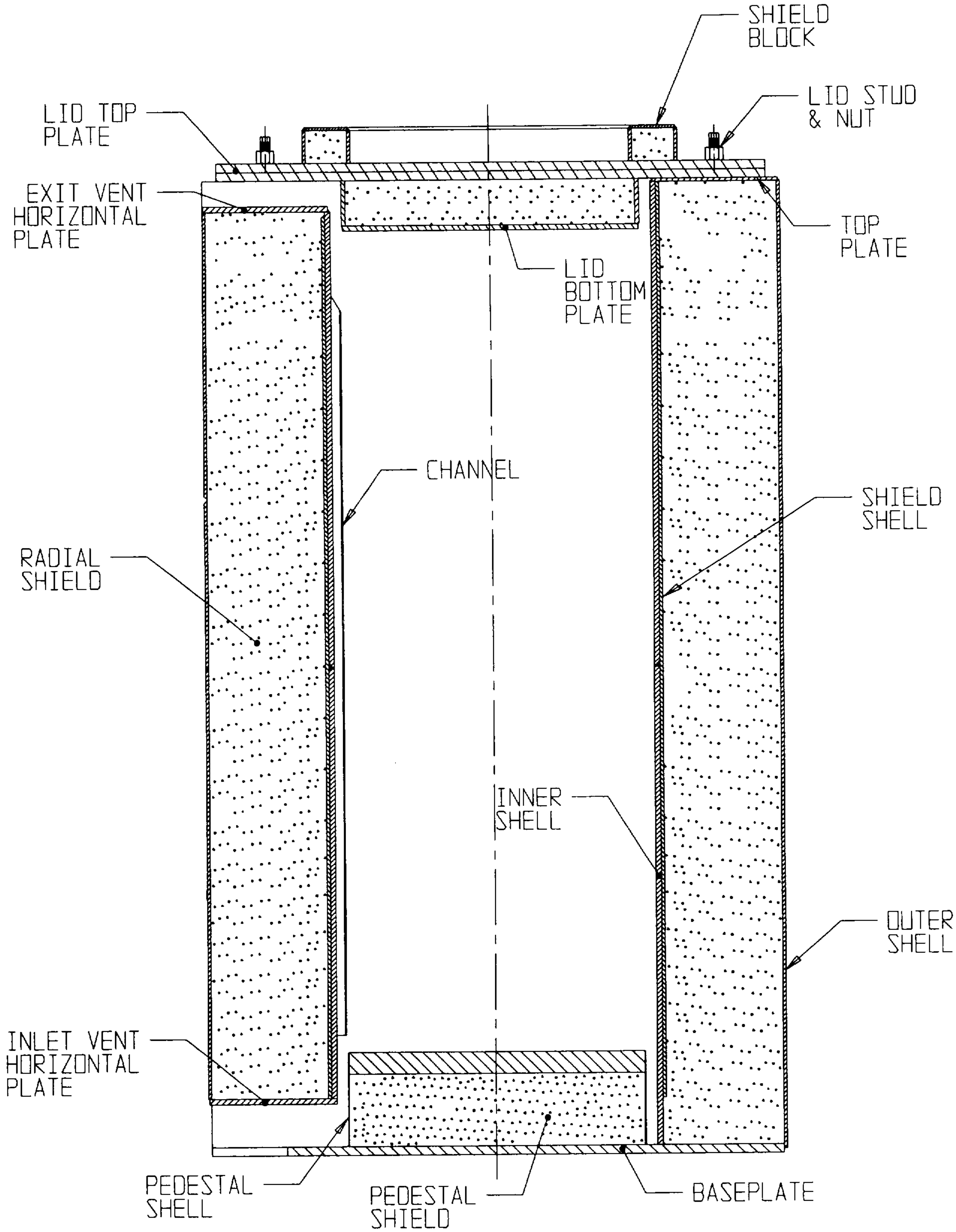

FIGURE 1.1.3; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

1.2.1 System Characteristics

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

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

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

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

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

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

A single 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 and the overall height is 239-1/2 inches. See Section 1.5 for the detailed design drawings. The weight of the overpack without an MPC is approximately 135 tons. See Table 3.2.1 for the detailed weights.

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

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

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

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

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

A description of each of the components is provided in the following sections, along with information with respect to its fabrication and safety features. This discussion is supplemented with the full set of Design Drawings and Bills-of-Material 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 and 1.2.4. The outer diameter and cylindrical height of each MPC are fixed. Each spent fuel MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring, as depicted in the MPC cross section elevation view, Figure 1.2.5. The number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. There are three MPC models, distinguished by the type and number of fuel assemblies authorized for loading. The MPC-24 is designed to store up to 24 intact PWR fuel assemblies. The MPC-68 is designed to store up to 68 intact or damaged BWR fuel assemblies. The MPC-68F is designed to store up to 68 intact or damaged BWR fuel assemblies and up to four BWR fuel assemblies classified as fuel debris. Design Drawings for all of the MPCs are provided in Section 1.5.

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

The construction features of the PWR MPC-24 and the BWR MPC-68 are similar. However, the PWR MPC-24 canister in Figure 1.2.4, which is designed for high-enriched PWR fuel, differs in construction from the MPC-68 in one important aspect: the fuel storage cells are physically separated from one another by a "flux trap", for criticality control. All MPC baskets are formed from an array of plates welded to each other, such that a honeycomb structure is created which resembles a multiflanged, closed-section beam in its structural characteristics.

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, heat conduction elements are installed. These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes which enable a snug fit in the confined spaces and ease of installation. The heat conduction elements are installed along the full length of the MPC basket to create a nonstructural thermal connection which facilitates heat transfer from the basket to shell. In their installed 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 edge-welded to the MPC outer shell. This plate is equipped with vent and drain ports which are utilized to remove moisture and air from the MPC, and backfill the MPC with a specified mass of inert gas (helium). The vent and drain ports are covered and seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by threaded holes in the MPC lid.

To maintain a constant exterior axial length between the MPC-24 and MPC-68, the thickness of the MPC-24 lid is ½ inch thinner than the MPC-68 lid to accommodate the longest PWR fuel assembly which is approximately a ½ inch longer than the longest BWR fuel assembly. For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type.

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

Alloy X is a material which 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 which 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, this application requests approval for use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this application) may be one of the following materials (only a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC).

- 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 which 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.

Other alloy materials which are identified to be more suitable by the DOE for the MGDS in the future and which are also bounded by the Alloy X properties set forth in Appendix 1.A can be used in the MPC after an amendment to this FSAR is approved.

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

1.2.1.2 Overpacks

1.2.1.2.1 HI-STORM 100 Overpack (Storage)

The HI-STORM 100 overpack is a rugged, heavy-walled cylindrical vessel. Figures 1.2.7 and 1.2.8 provide cross sectional views of the HI-STORM 100 System. 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 attached to its underside and top 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. Stainless steel shims are attached to channels to allow the proper inner diameter dimension to be obtained and to provide a guiding surface for MPC insertion and removal.

The storage overpack has air ducts to allow for passive natural convection cooling of the contained MPC. Four air inlets and four air outlets are located at the lower and upper extremities of the overpack, respectively. The air inlets and outlets are covered by a fine mesh screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, provided in this FSAR, evaluates the effects of partial and complete blockage of the air ducts.

The four air inlets and four air outlets are penetrations through the thick concrete shielding provided by the HI-STORM overpack. Within the air inlets and outlets, an array of gamma shield cross plates are installed. These gamma shield cross plates are designed to scatter any particles traveling through the ducts. The result of scattering the particles in the ducts is a significant decrease in the local dose rates around the four air inlets and four air outlets. The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized.

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 (see Figure 1.2.7). The four anchor blocks are located on 90E centers. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, and Hilman rollers. Slings or other suitable devices mate with lifting lugs which 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 and compressive strength. The concrete shall be in accordance with the requirements specified in Appendix 1.D.

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

Density and compressive strength are the key parameters which delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal, and HI-STORM lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.2.6] 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" [1.2.10]. 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.

1.2.1.2.2 HI-TRAC (Transfer Cask)

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 a HI-TRAC with the pool lid installed.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC. The top lid 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 HI-TRAC 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 which 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. The transfer lid incorporates two sliding doors which 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.

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

1.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure the radiation and exposure requirements in 10CFR72.104 and 10CFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the SNF in the MPC for ALARA considerations during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and Boral™ neutron poison 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 100 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 underside of the lid, and a thick circular concrete pedestal upon which the MPC rests. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shield ring atop 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.

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. 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 125 ton HI-TRAC transfer lid, the neutron shield material, Holtite-A, is also provided. The 125 ton HI-TRAC top lid is composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The 100 ton HI-TRAC top lid is composed of steel only providing gamma attenuation.

1.2.1.3.1 Boral Neutron Absorber

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 recently licensed Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

As indicated in Tables 1.2.3-1.2.5, 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 in compliance with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

1.2.1.3.2 Neutron Shielding

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

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

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

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

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

Neutron shielding in the 125 ton HI-TRAC transfer cask in the axial direction is provided by Holtite-A within the top lid and transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer commercially available under the trade name NS-4-FR (or equivalent) and will be specified with a minimum B_4C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties. 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.

<u>Density</u>

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.

<u>Hydrogen</u>

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

<u>Boron Carbide</u>

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (minimum) 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 minimum B_4C weight percent of 1%.

<u>Design Temperature</u>

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

Thermal Conductivity

Table 1.B.1 lists the thermal conductivity of Holtite-A specified by the manufacturer.

The Holtite-A neutron shielding material is stable below the design temperature for the long term and provides excellent shielding properties for neutrons. Technical papers provided in Appendix 1.B validate the neutron shield material's long-term stability within the design temperature and the material's ability to resist the effects of a fire accident. Holtite-A has been utilized in similar applications and has been licensed for use in a transportation cask under Docket No. 71-9235 and for storage in the HI-STAR 100 overpack under Docket No. 72-1008.

1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7). Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the baseplate and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are installed in the ducts (Figure 1.2.8) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets.

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

1.2.1.4 Lifting Devices

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

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The anchor blocks are integrally welded to the overpack radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate. Studs are threaded into the anchor blocks to secure the lid and provide for lifting. These four studs provide for direct attachment of lifting devices which, along with a specially-designed lift rig to ensure a vertical lift, allow lifting by a crane or similar equipment.

The lift rig shall be designed to lift a fully-loaded storage overpack with margins of safety specified in ANSI N14.6 [1.2.9].

A bottom lift of the HI-STORM 100 storage overpack is effected by the insertion of four hydraulic jacks underneath the inlet vent horizontal plates (Figure 1.2.1). A slot in the overpack baseplate allows the hydraulic jacks to be placed underneath the inlet vent horizontal plate. The hydraulic jacks lift the loaded overpack to a sufficient height to allow air pads to be placed or removed from under the overpack baseplate.

The HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket 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 lifting trunnions feature a locking plate, which is placed onto the trunnion shaft and bolted to the HI-TRAC external surface to prevent the lifting trunnion from backing out.

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 mass to be placed in the MPC internal cavity as a function of the free space. As the fill pressure is highly dependent on the MPC internal temperature, which increases because of the decay heat and the vacuum drying process, it is more accurate to measure the mass placed in the MPC internal cavity rather than pressure.

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

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

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

1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STORM 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the Design Drawings in Section 1.5. A summary of the 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 spent fuel pool water or plant demineralized water. HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

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

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid and around the mid-height circumference of HI-TRAC to ensure that the dose rates are within expected values. The Automated Welding System baseplate shield 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). Liquid penetrant examinations are performed on the root and final passes. A volumetric examination is also performed on the MPC lid-to-shell weld. The water level is raised to the top of the MPC and the weld is hydrostatically tested. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid-to-shell) to verify weld integrity and to ensure that required leakage rates are within acceptance criteria. The water level is raised to the top of the MPC again and then 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. The volume of water displaced from the MPC is measured to determine the free volume inside the MPC. This information is used to determine the helium backfill requirements for the MPC.

The Vacuum Drying System (VDS) is connected to the MPC and is used to remove all 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.

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

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

Two or four stays (depending on the site crane hook configuration) are installed between the MPC lift cleats and the lift yoke main pins. The stays secure the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. The HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and stays 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 positioned in the truck bay with the lid removed and the vent duct shield inserts installed. The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. The doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent shield inserts. The MPC lift cleats and slings are removed from atop the MPC. 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 positioned in the cask transfer facility with the lid removed and the vent duct shield inserts installed. 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. The cradle is equipped with rotation trunnions which engage the HI-TRAC 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.

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

After the loaded HI-TRAC arrives at the cask transfer facility, the HI-TRAC is upended by a crane if the HI-TRAC is in a horizontal orientation. The loaded HI-TRAC is then placed, using the crane located in the transfer area, on top of HI-STORM. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. The doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent duct shield inserts. The MPC lift cleats and slings are removed from atop the MPC. 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.

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. However, the loaded overpack may be lifted from the top through the lid studs 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.

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 vent duct shield inserts are installed, and the MPC lift cleats are attached to the MPC. The MPC lift slings are attached to the MPC lift cleats. HI-TRAC is raised and positioned on top of HI-

STORM. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the HI-TRAC transfer lid doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM.

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

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC stays, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water. The annulus shield is installed to protect the annulus from debris produced

from the lid removal process. Similarly, HI-TRAC top surfaces are covered with a protective fire-retarding blanket.

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

The inflatable annulus seal is installed and pressurized. 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 pumped back into the spent fuel pool. 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 and MPC-68 do not rely on soluble boron credit during loading or the assurance that water cannot enter the MPC during storage to meet the stipulated criticality limits.

The MPC-68 basket is equipped with Boral with a minimum ^{10}B areal density of 0.0372 g/cm^2.

The MPC-24 basket is equipped with Boral with a minimum ^{10}B areal density of 0.0267 g/cm^2. Due to the lower reactivity of the fuel to be stored in the MPC-68F as specified by the Technical Specifications in Chapter 12, the MPC-68F is equipped with Boral with a minimum ^{10}B areal density of 0.01 g/cm^2.

1.2.2.3.2 Chemical Safety

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

1.2.2.3.3 Operation Shutdown Modes

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

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is seal welded and leak tested. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, a thermocouple may be utilized to monitor the air temperature of the HI-STORM overpack exit vent in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 and the Technical Specifications in Chapter 12 for additional details.

1.2.2.3.5 Maintenance Technique

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

1.2.3 Cask Contents

The HI-STORM 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key design parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1 and the Technical Specifications.

At this time, failed fuel assemblies discharged from Dresden Unit 1 and Humboldt Bay reactors have been evaluated and this application requests approval of these two types of damaged fuel assemblies and fuel debris as contents for storage in the MPC-68. Damaged fuel assemblies and fuel debris shall be placed in damaged fuel containers prior to loading into the MPC to facilitate handling and contain loose components. Any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68, may be stored in the standard MPC-68. The MPC-68 design to store fuel debris is almost identical to the MPC-68 design to store intact or damaged fuel, the sole difference being the former requires a lower minimum B^{10} areal density in the Boral. Therefore, an MPC-68 which is to store damaged fuel containers with fuel assemblies classified as fuel debris must be designated during fabrication to ensure the proper minimum B^{10} areal density criteria is applied. To distinguish an MPC-68 which is fabricated to store damaged fuel containers with fuel assemblies classified as fuel debris, the MPC shall be designated as an "MPC-68F".

Up to 4 damaged fuel containers with fuel assemblies classified as fuel debris and meeting the requirements in the Technical Specifications may be stored within an MPC-68F. The quantity of damaged fuel containers with fuel debris is limited to meet the off-site transportation requirements of 10CFR71, specifically, 10CFR71.63(b).

Table 1.2.1

KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	3	1 for PWR 2 for BWR
MPC storage capacity:	MPC-24	Up to 24 intact zircaloy or stainless steel clad PWR fuel assemblies. Control components and non-fuel hardware are not authorized for loading.
	MPC-68 MPC-68F	Any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68 in the MPC-68 OR Up to 4 damaged fuel containers with zircaloy clad BWR fuel debris and the complement damaged zircaloy clad BWR fuel assemblies in damaged fuel containers or intact fuel assemblies within an MPC-68F.

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 conditions Accident Conditions	 100 100 125	 100 100 125
Total heat load, max. (kW)	20.88 (MPC-24)	21.4 (MPC-68)
Maximum permissible peak fuel cladding temperature: Normal (°F) Short Term & Accident (°F)	 See Table 2.2.3 1058°	 See Table 2.2.3 1058°
MPC internal environment Helium fill (g-moles/l of free space)	 0.1212 (MPC-24)	 0.1218 (MPC-68 & MPC-68F)
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	<0.95	<0.95
Boral ^{10}B Areal Density (g/cm^2)	0.0267 (MPC-24)	0.0372 (MPC-68) 0.01 (MPC-68F)
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

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

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

Table 1.2.3

BORAL EXPERIENCE LIST
DOMESTIC PRESSURIZED WATER REACTORS

Plant	Utility
Donald C. Cook	American Electric Power
Indian Point 3	New York Power Authority
Maine Yankee	Maine Yankee Atomic Power
Salem 1,2	Public Service Electric and Gas
Sequoyah 1,2	Tennessee Valley Authority
Yankee Rowe	Yankee Atomic Power
Zion 1,2	Commonwealth Edison Company
Byron 1,2	Commonwealth Edison Company
Braidwood 1,2	Commonwealth Edison Company
Three Mile Island I	GPU Nuclear
Sequoyah (rerack)	Tennessee Valley Authority
D.C. Cook (rerack)	American Electric Power
Maine Yankee	Maine Yankee Atomic Power Company
Connecticut Yankee	Northeast Utilities Service Company
Salem Units 1 & 2 (rerack)	Public Service Electric & Gas Company

Table 1.2.4

BORAL EXPERIENCE LIST
DOMESTIC BOILING WATER REACTORS

Plant	Utility
Browns Ferry 1,2,3	Tennessee Valley Authority
Brunswick 1,2	Carolina Power & Light
Clinton	Illinois Power
Dresden 2,3	Commonwealth Edison Company
Duane Arnold Energy Center	Iowa Electric Light and Power
J.A. FitzPatrick	New York Power Authority
E.I. Hatch 1,2	Georgia Power Company
Hope Creek	Public Service Electric and Gas
Humboldt Bay	Pacific Gas and Electric Company
LaCrosse	Dairyland Power
Limerick 1,2	Philadelphia Electric Company
Monticello	Northern States Power
Peachbottom 2,3	Philadelphia Electric Company
Perry 1,2	Cleveland Electric Illuminating
Pilgrim	Boston Edison Company
Susquehanna 1,2	Pennsylvania Power & Light
Vermont Yankee	Vermont Yankee Atomic Power
Hope Creek	Public Service Electric and Gas Company
Shearon Harris Pool B	Carolina Power & Light Company
Duane Arnold	Iowa Electric Light and Power
Pilgrim	Boston Edison Company
LaSalle Unit 1	Commonwealth Edison Company
Millstone Point Unit One	Northeast Utilities Service Company

Table 1.2.5

BORAL EXPERIENCE LIST
FOREIGN PLANTS

INTERNATIONAL INSTALLATIONS USING BORAL	
COUNTRY	PLANT(S)
France	12 PWR Plants
South Africa	Koeberg 1,2
Switzerland	Beznau 1,2 Gosgen
Taiwan	Chin-Shan 1,2 Kuosheng 1,2
Mexico	Laguna Verde Units 1,2
Korea	Ulchin Units 1, 2
Brazil	Angra 1
United Kingdom	Sizewell B

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 PTexamined, hydrostatically tested, and leak tested
5	MPC dewatered, vacuum dried, 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
9	MPC lowered and seated on HI-TRAC transfer lid
10	HI-TRAC/MPC assembly transferred to atop HI-STORM overpack
11	MPC weight supported by rigging and transfer lid doors opened
12	MPC lowered into HI-STORM overpack, HI-TRAC transfer lid doors closed, and HI-TRAC removed from atop HI-STORM overpack
13	HI-STORM overpack lid installed and bolted in place
14	HI-STORM overpack placed in storage at the ISFSI pad

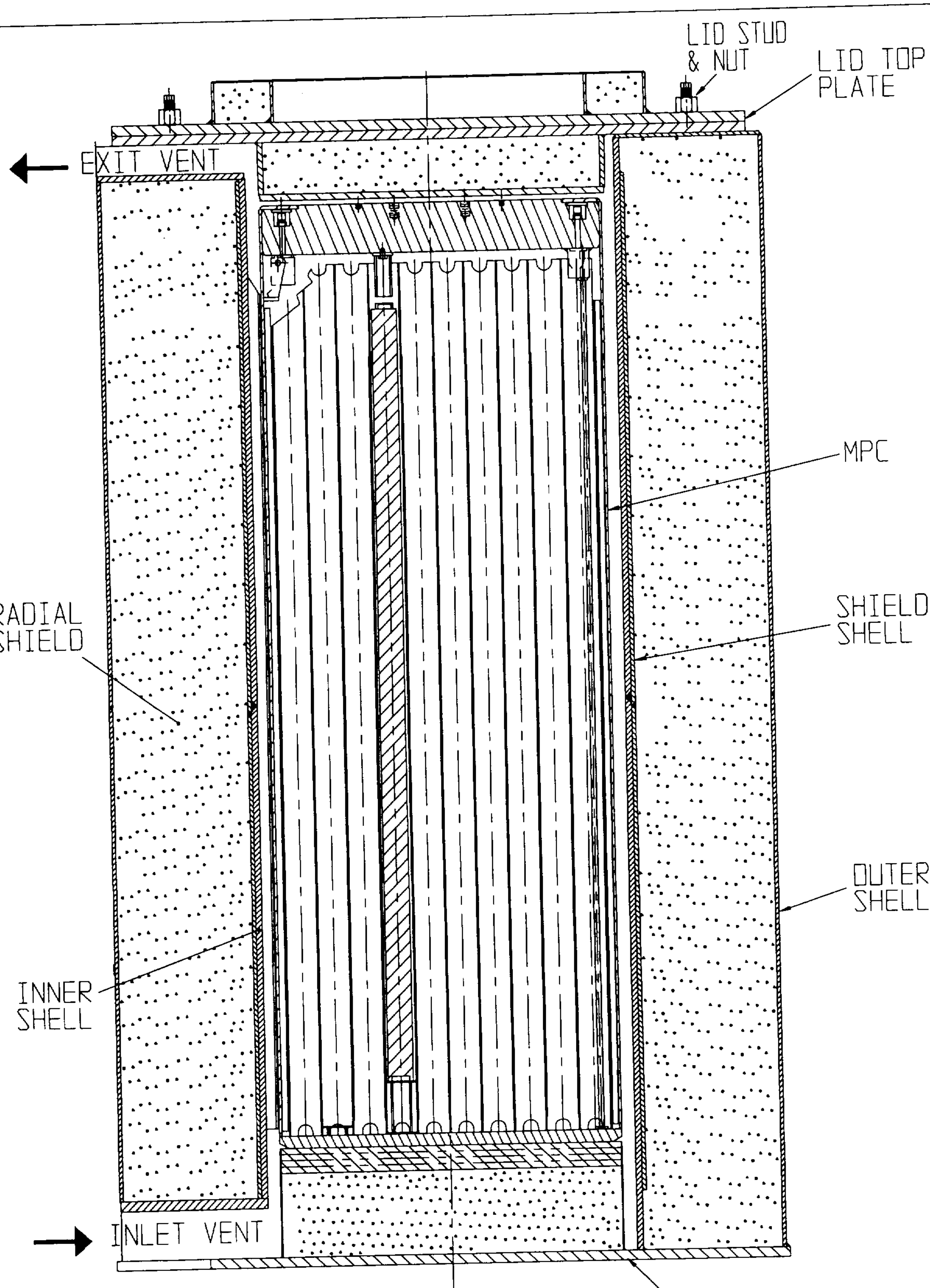

FIGURE 1.2.1; CROSS SECTION VIEW OF THE HI-STORM 100 SYSTEM

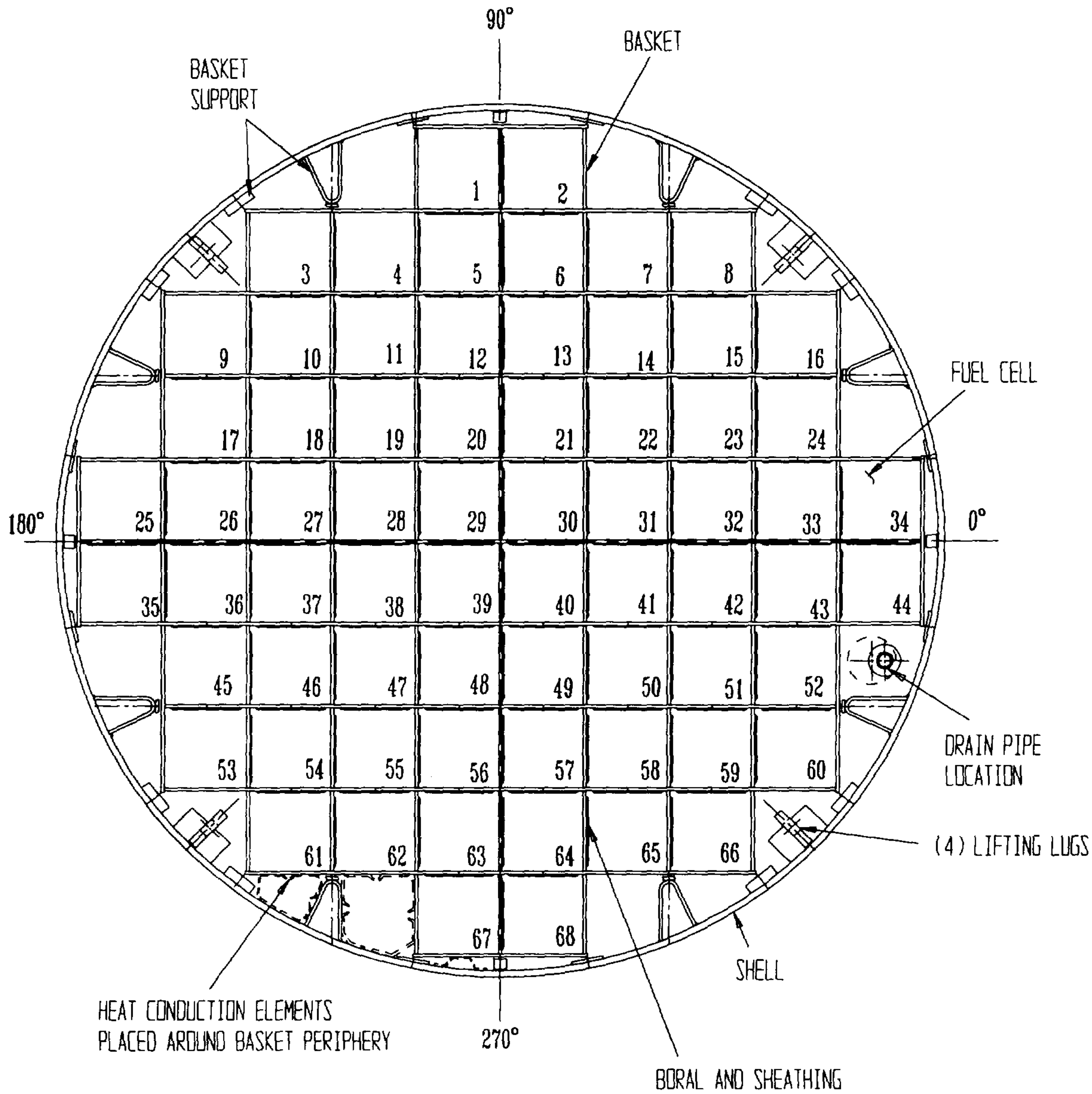

FIGURE 1.2.2; MPC-68 CROSS SECTION VIEW

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FIGURE 1.2.3; DELETED

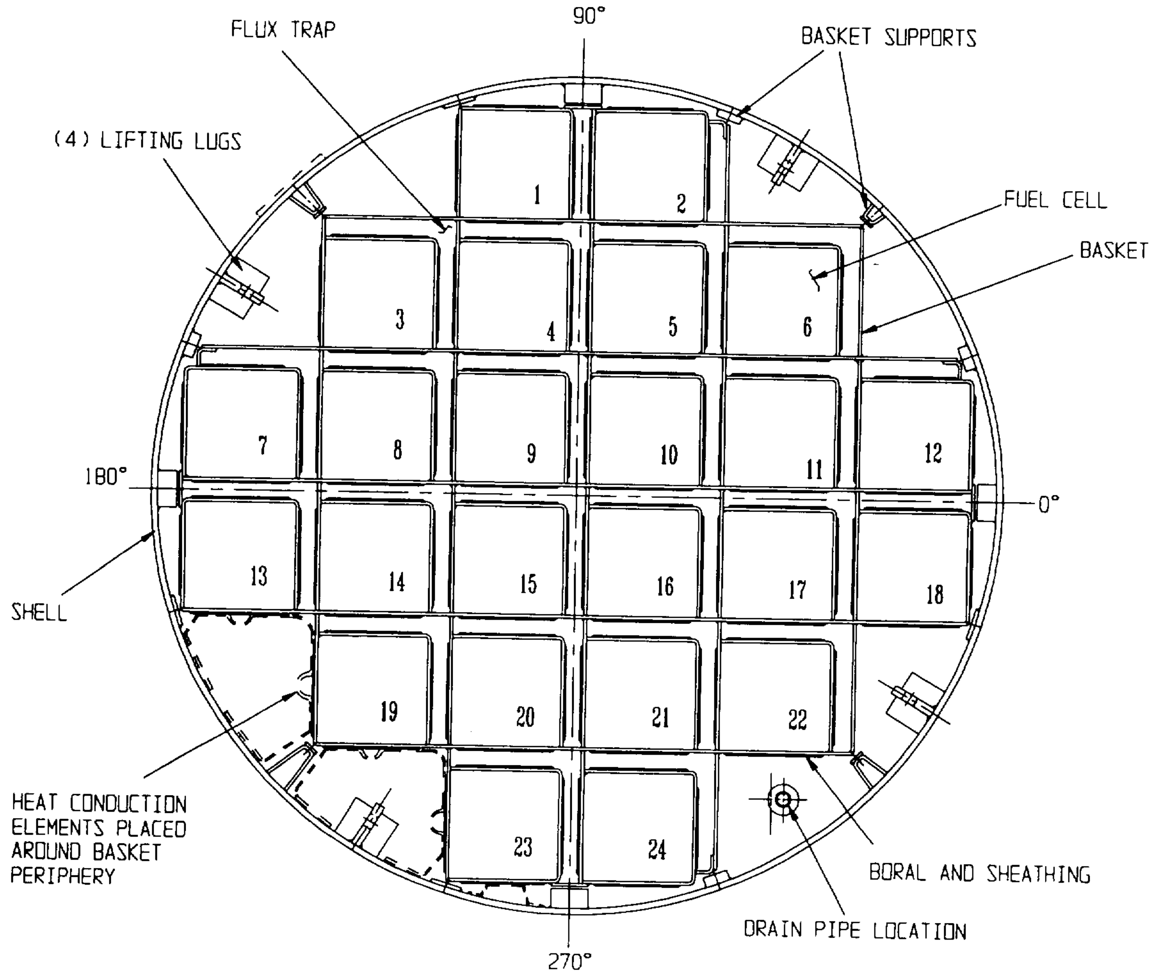

FIGURE 1.2.4; MPC-24 CROSS SECTION VIEW

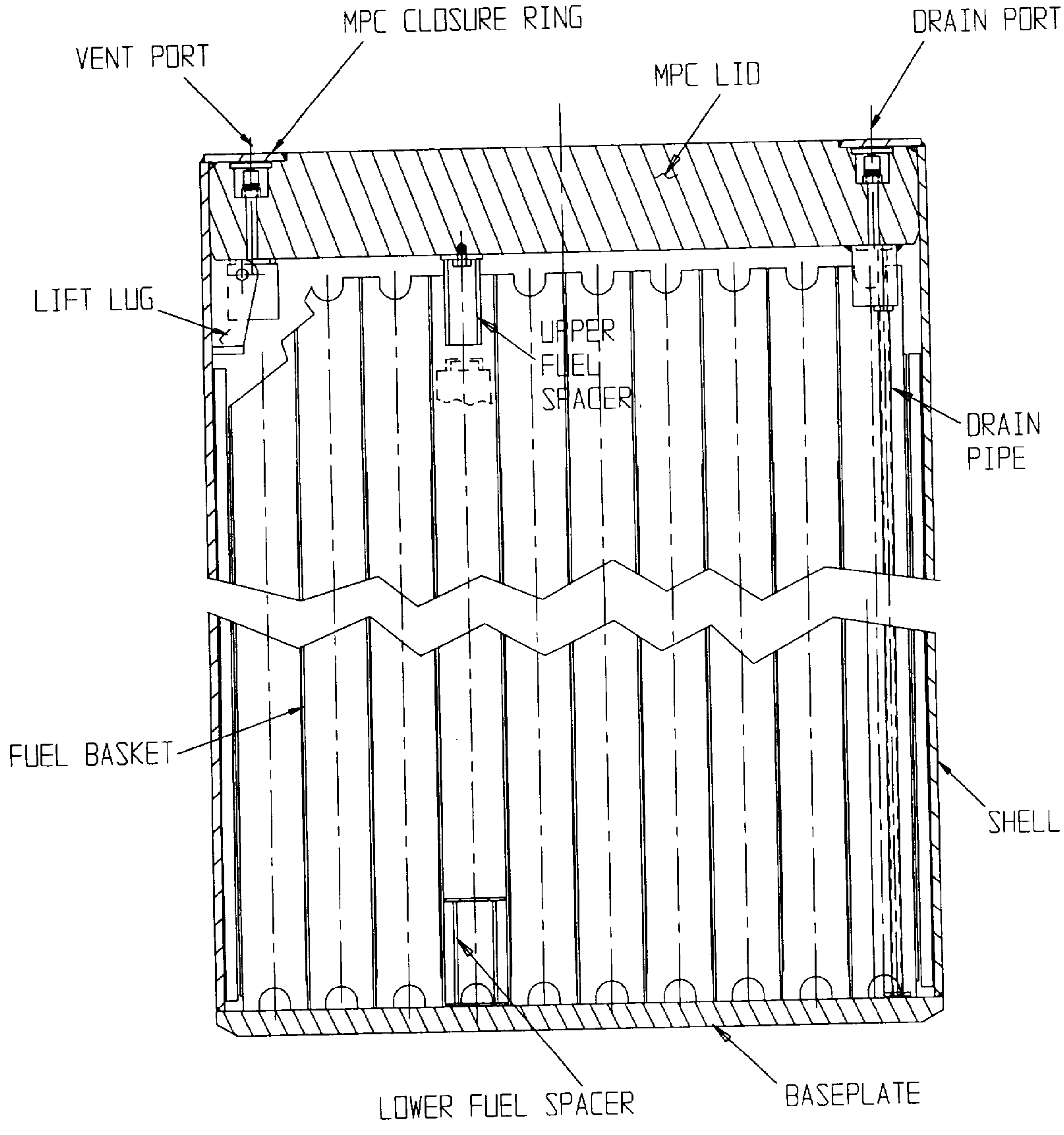

FIGURE 1.2.5; CROSS SECTION ELEVATION VIEW OF MPC

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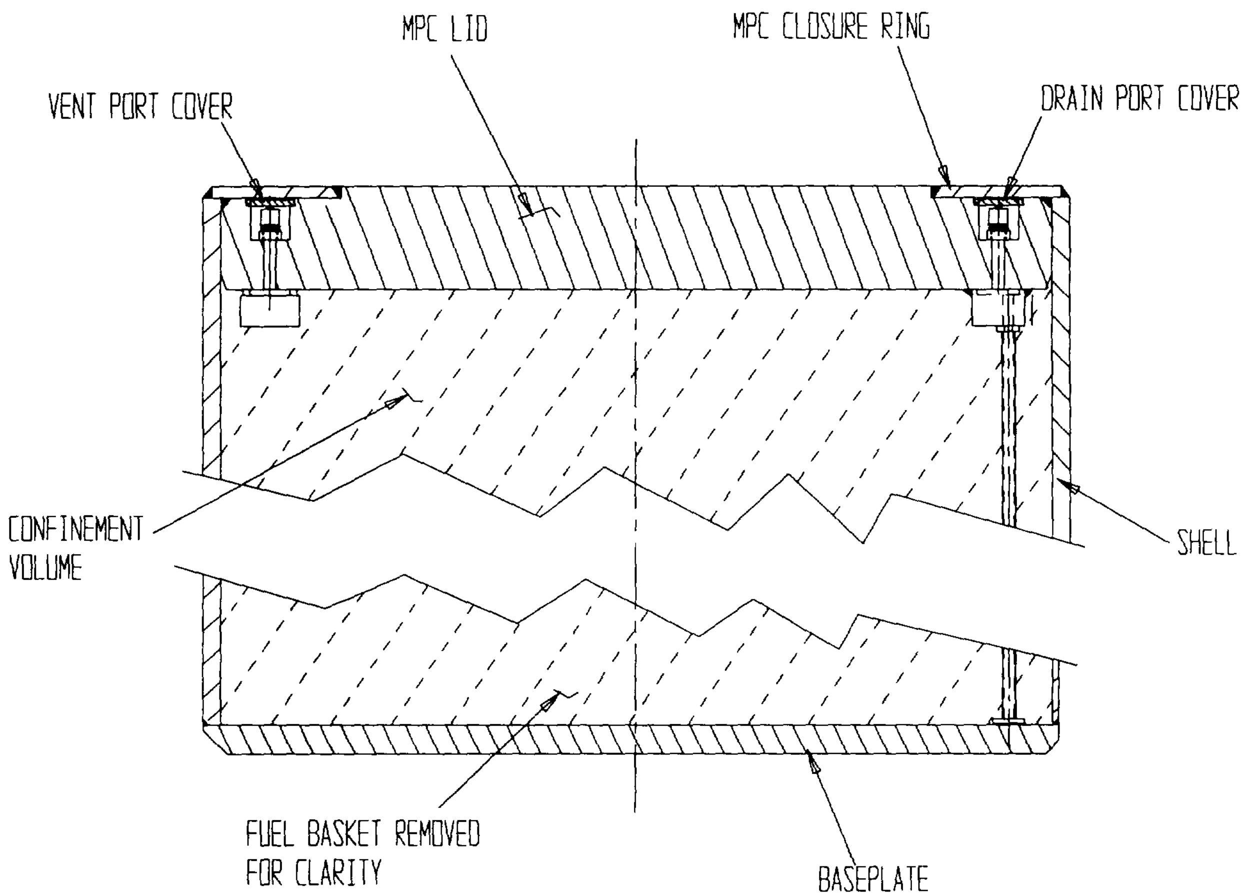

FIGURE 1.2.6; MPC CONFINEMENT BOUNDARY

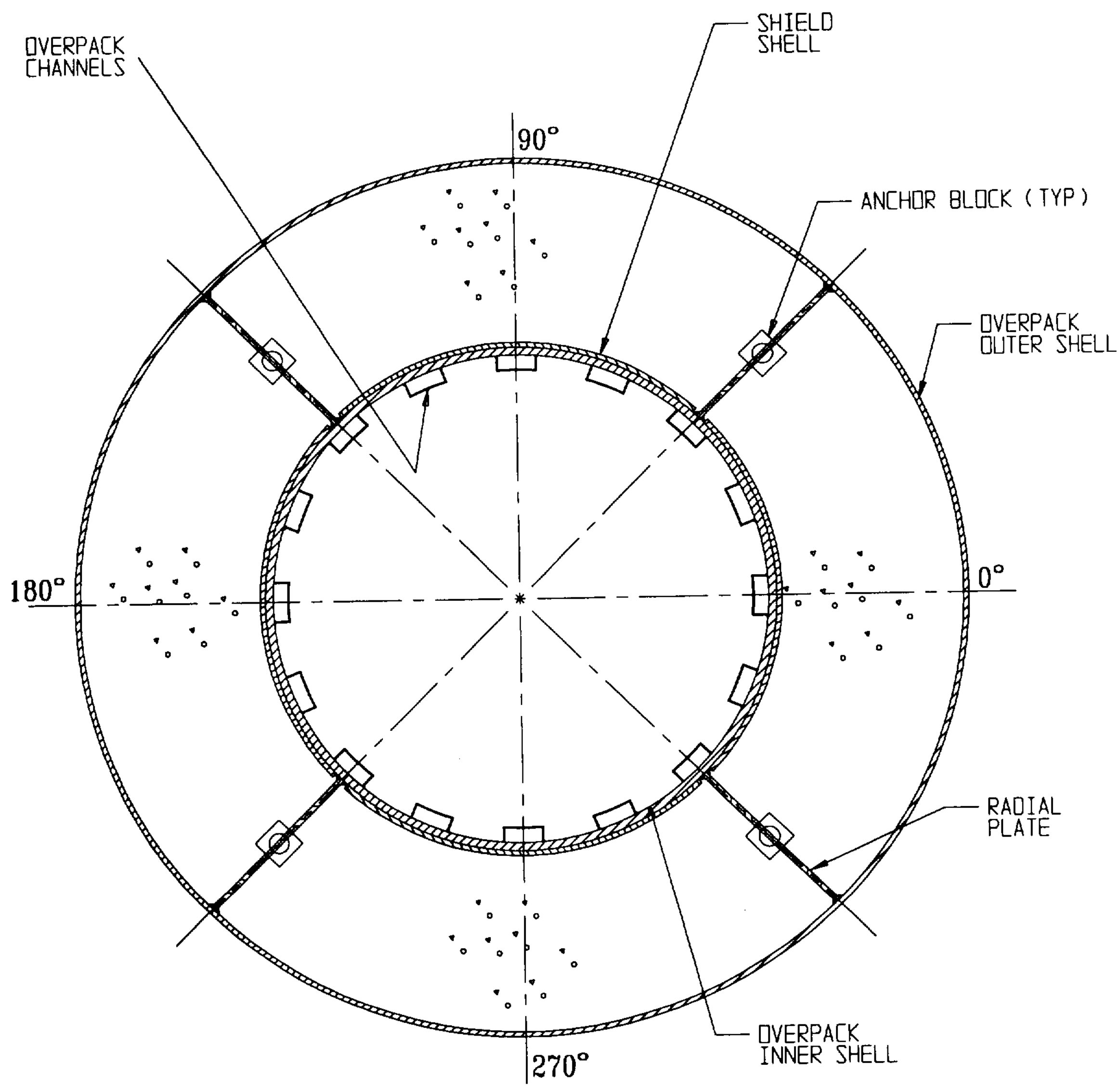

FIGURE 1.2.7; CROSS SECTION OF HI-STORM OVERPACK

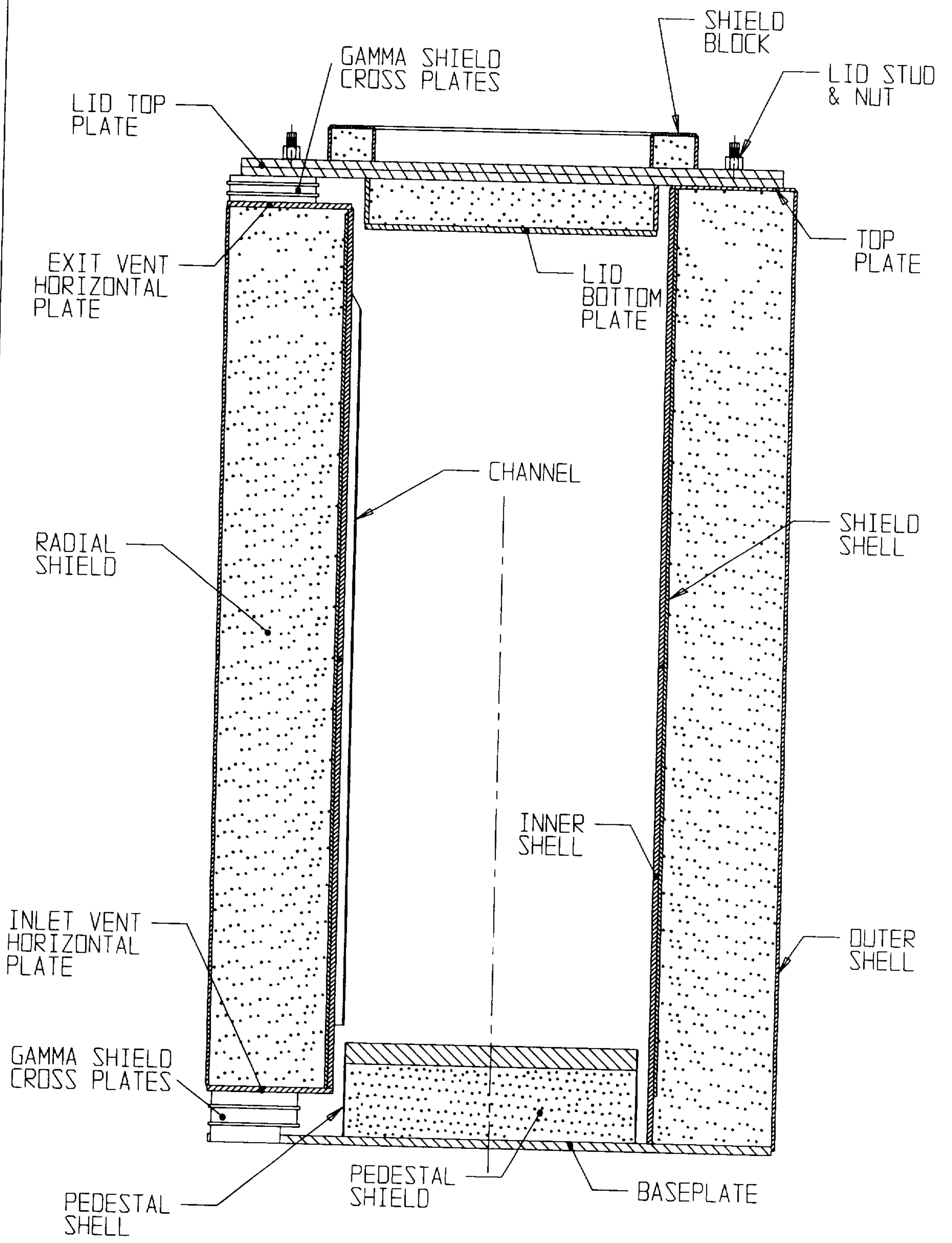

FIGURE 1.2.8; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

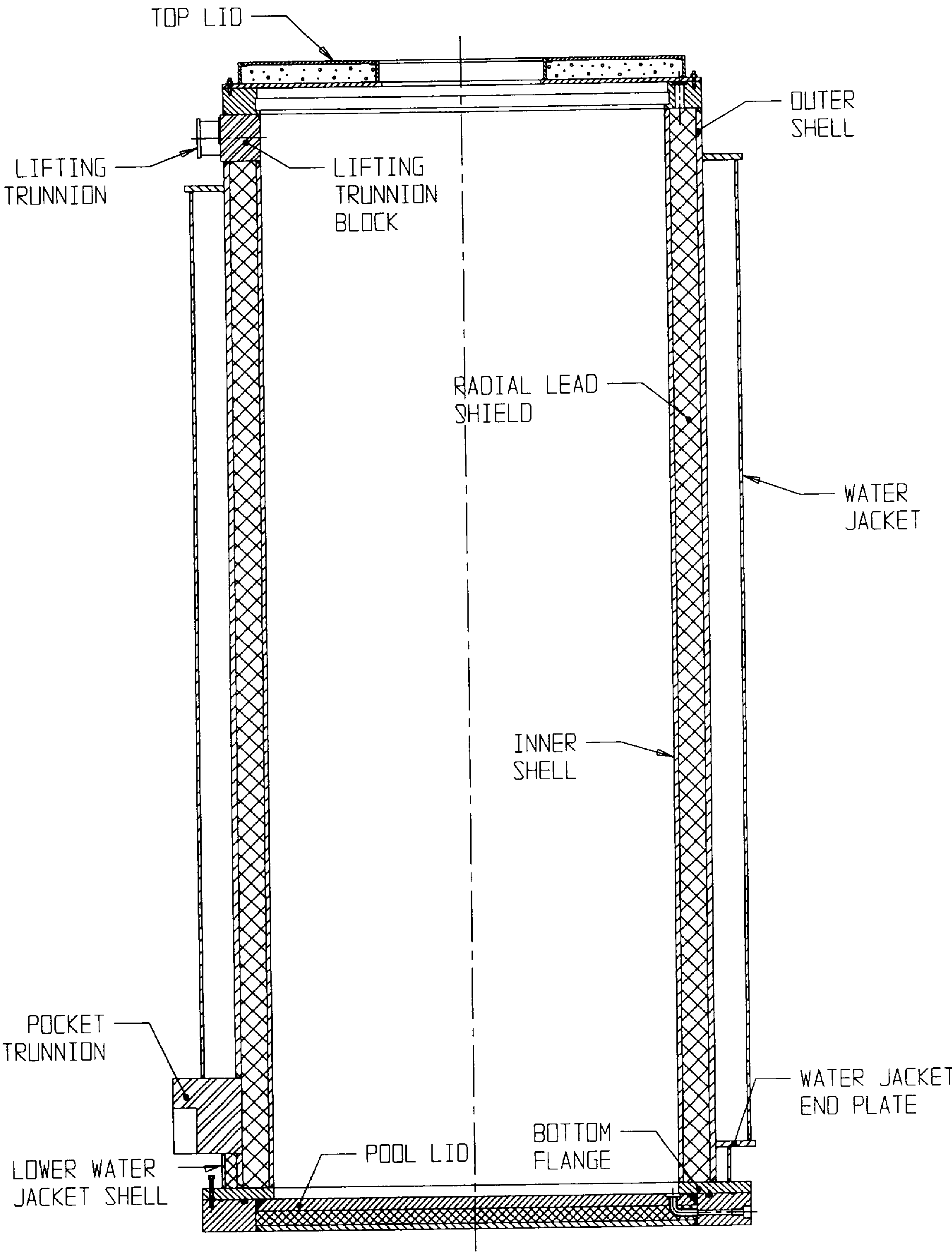

FIGURE 1.2.9; 125TON HI-TRAC TRANSFER CASK WIT POOL LID CROSS SECTIONAL ELEVATION VIEW

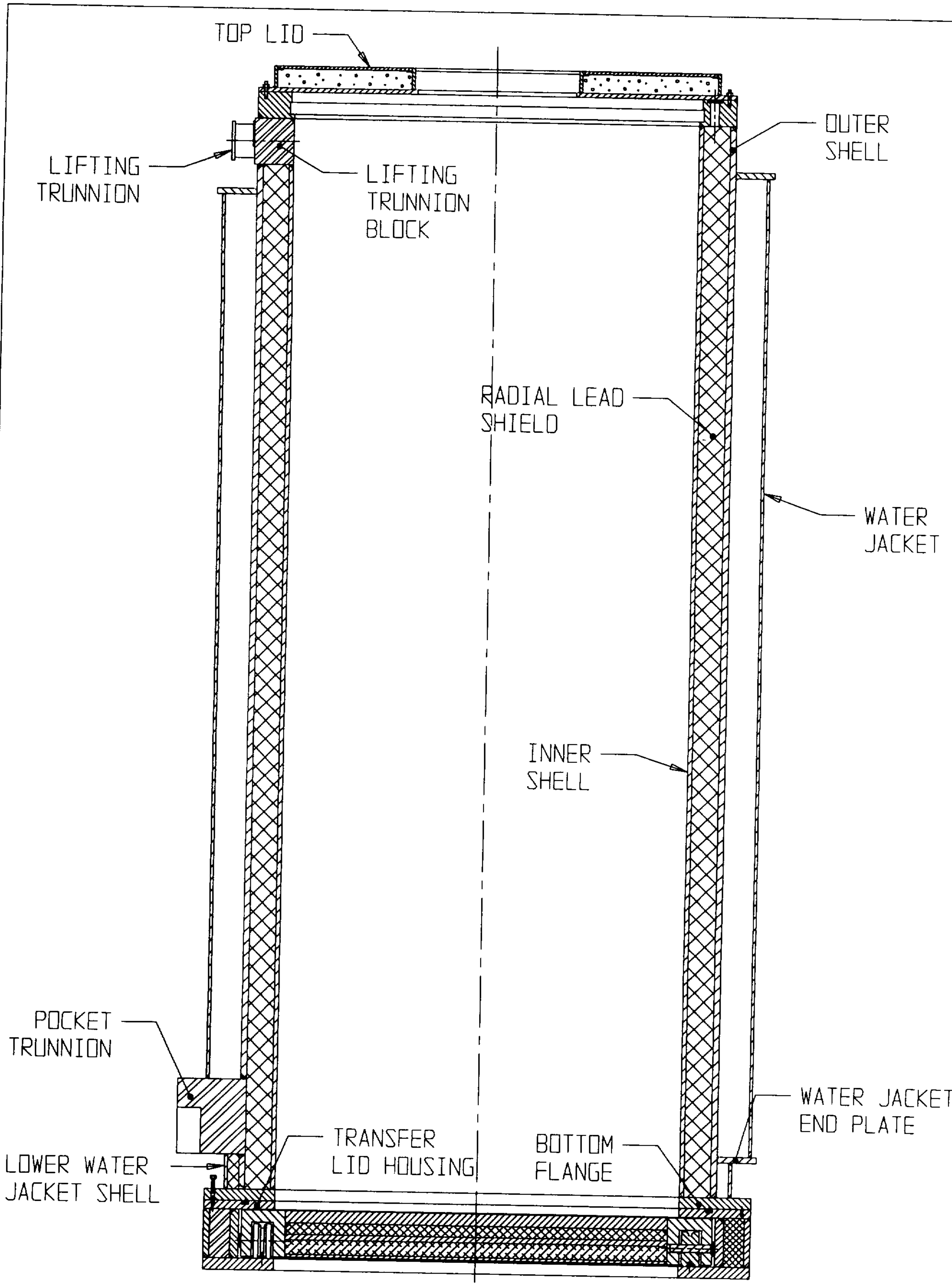

FIGURE 1.2.10; 125TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW

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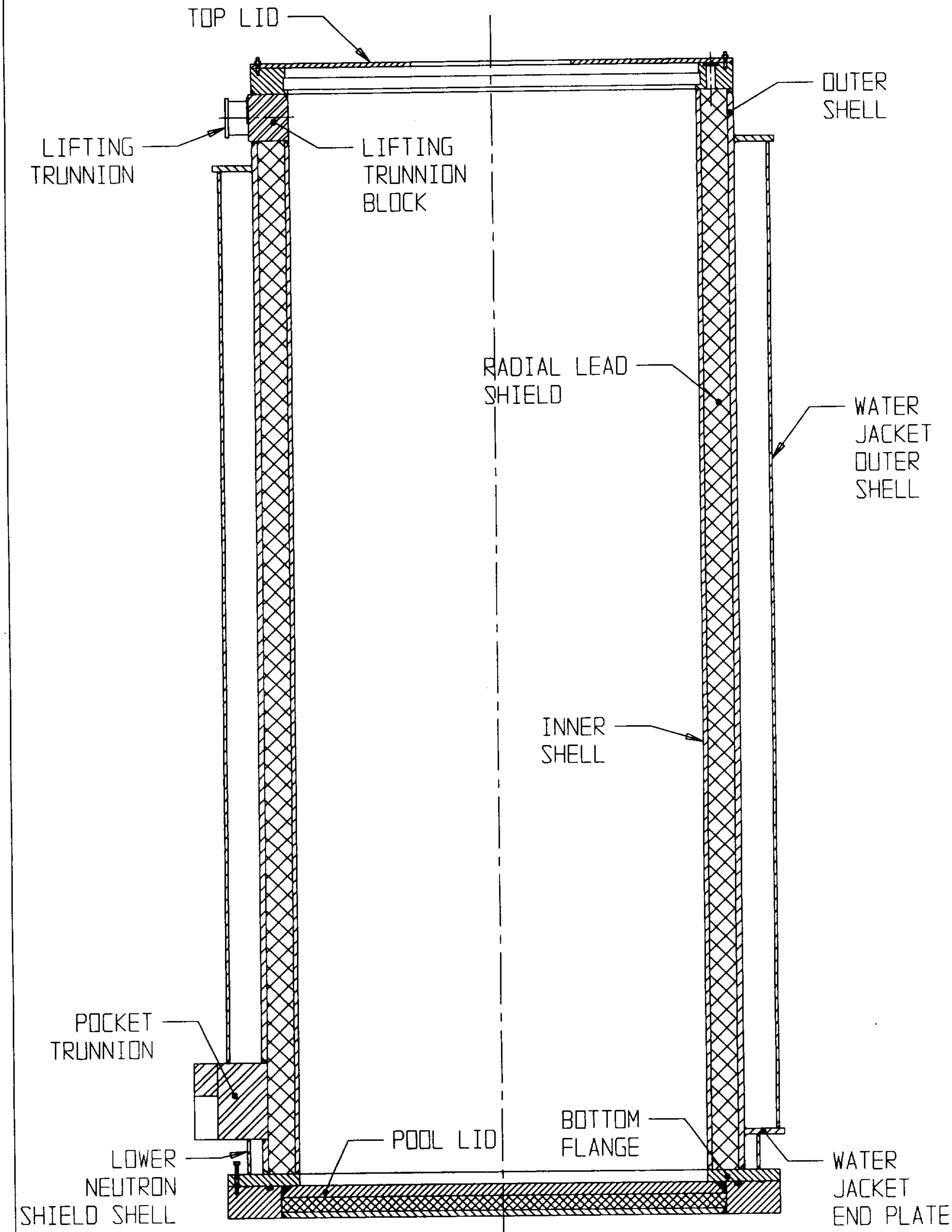

FIGURE 1.2.11; 100 HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW

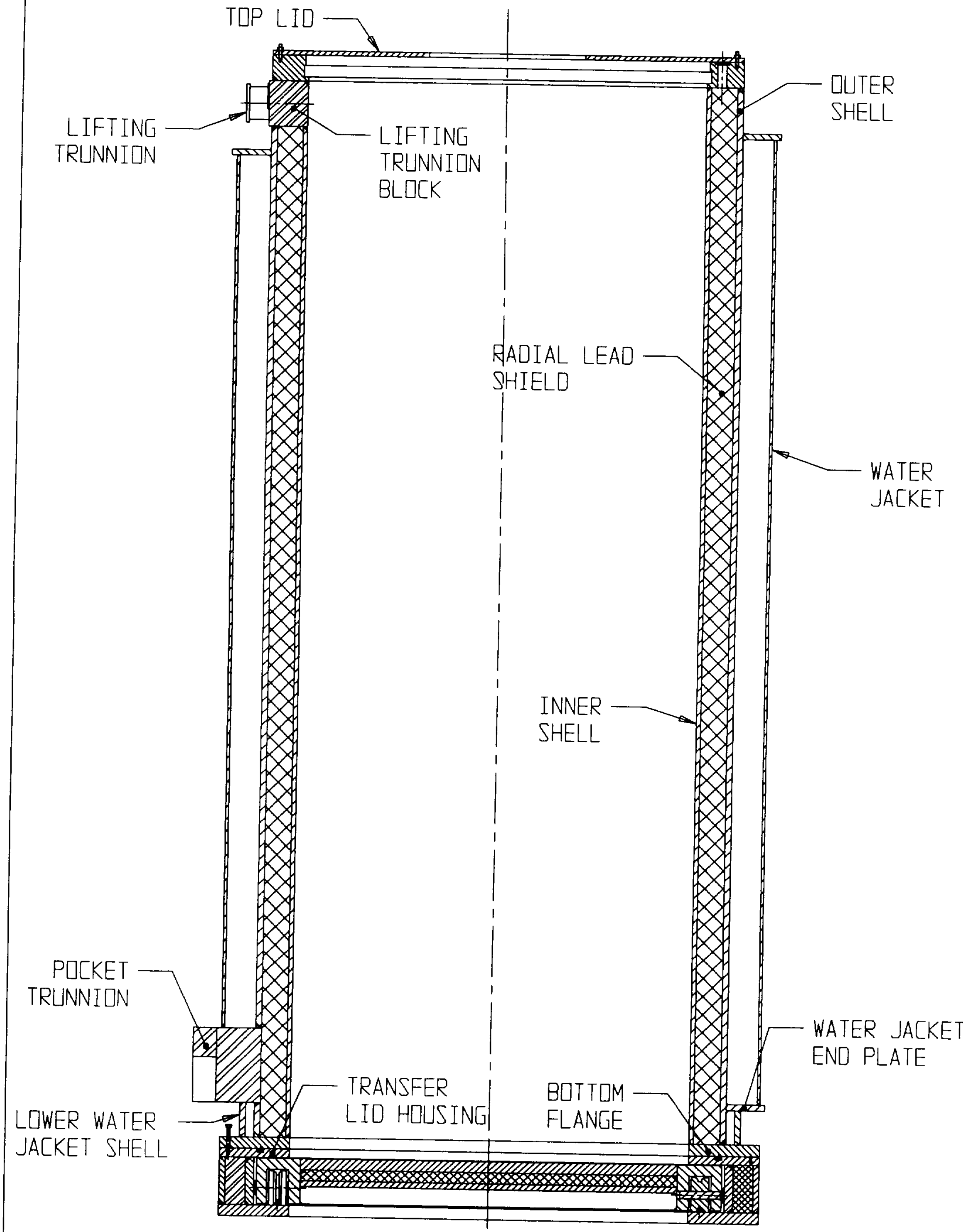

FIGURE 1.2.12; 100 TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW

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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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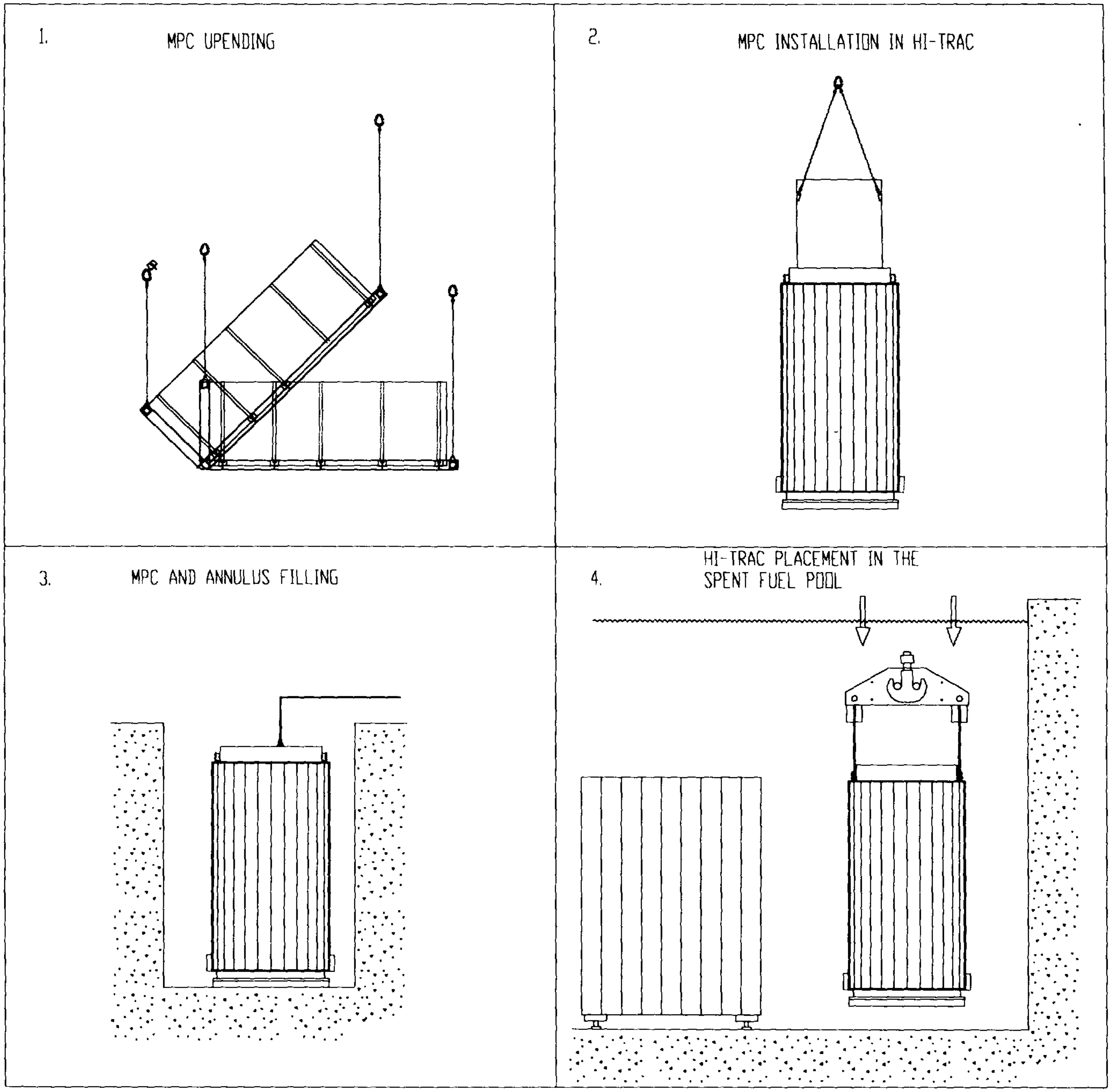


Figure 1.2.16a; Major HI-STORM 100 Loading Operations (Sheet 1 of 6)

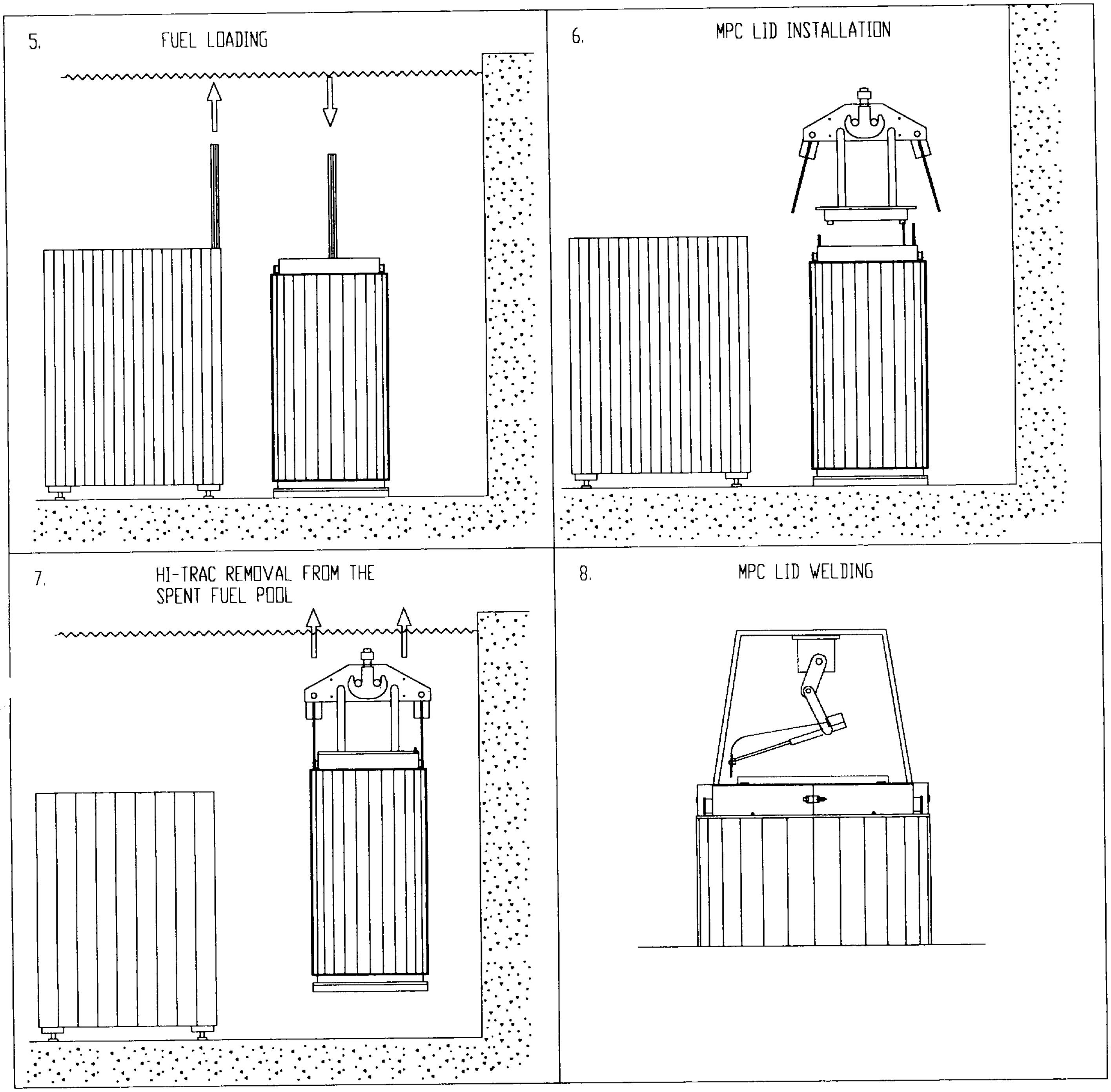


Figure 1.2.16b; Major HI-STORM 100 Loading Operations (Sheet 2 of 6)

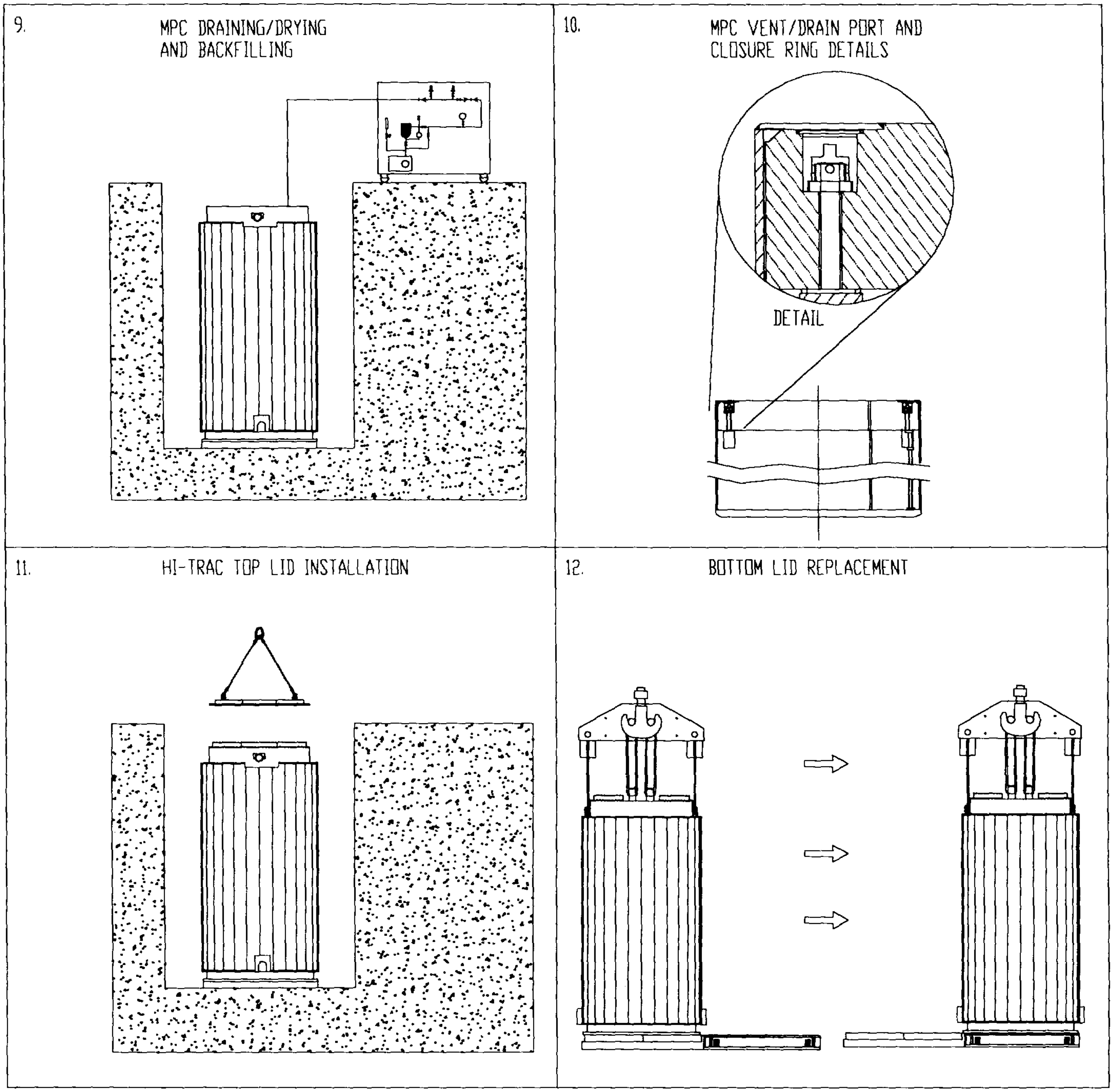


Figure 1.2.16c; Major HI-STORM 100 Loading Operations (Sheet 3 of 6)

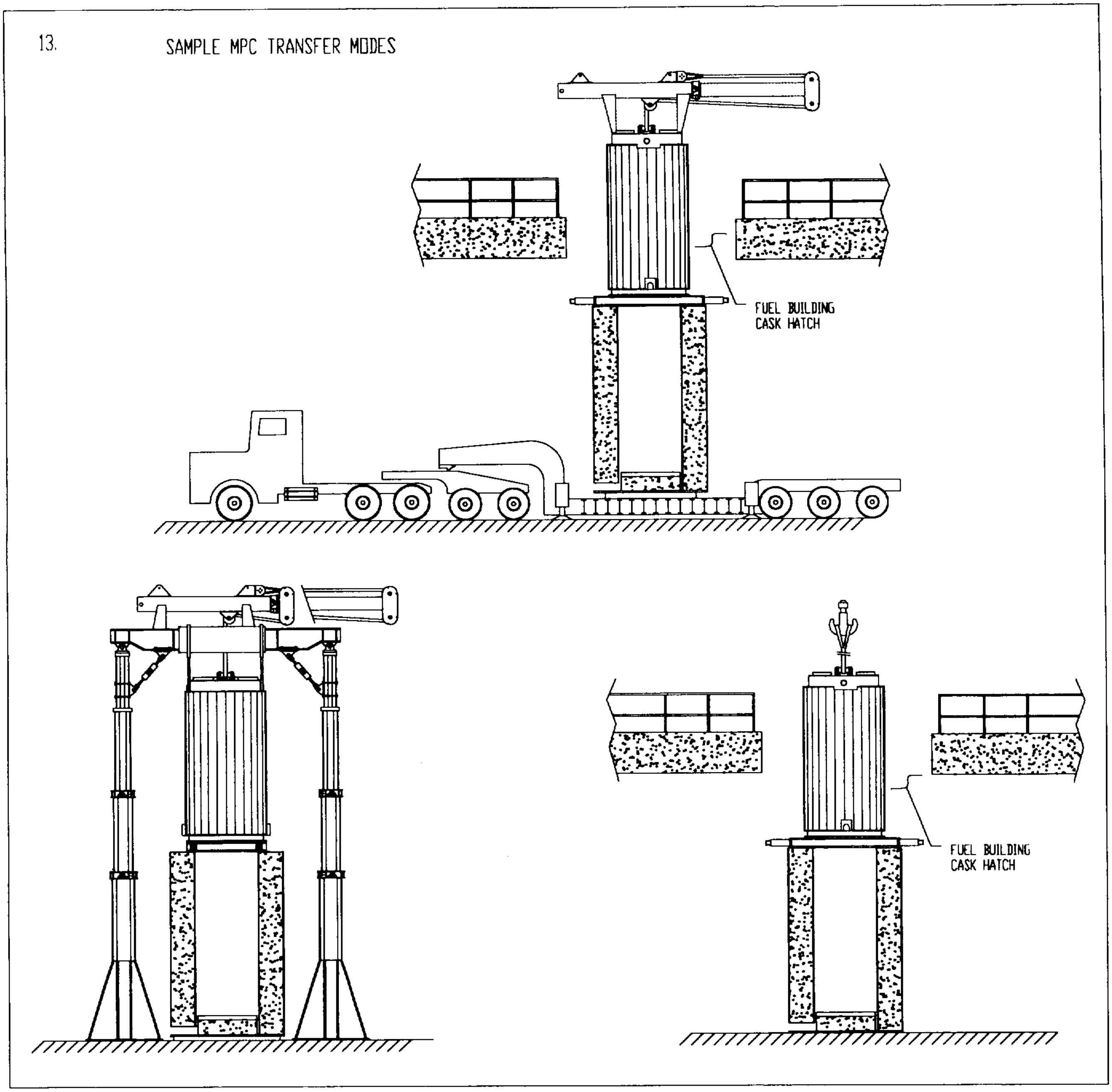


Figure 1.2.16d; Major HI-STORM 100 Loading Operations (Sheet 4 of 6)

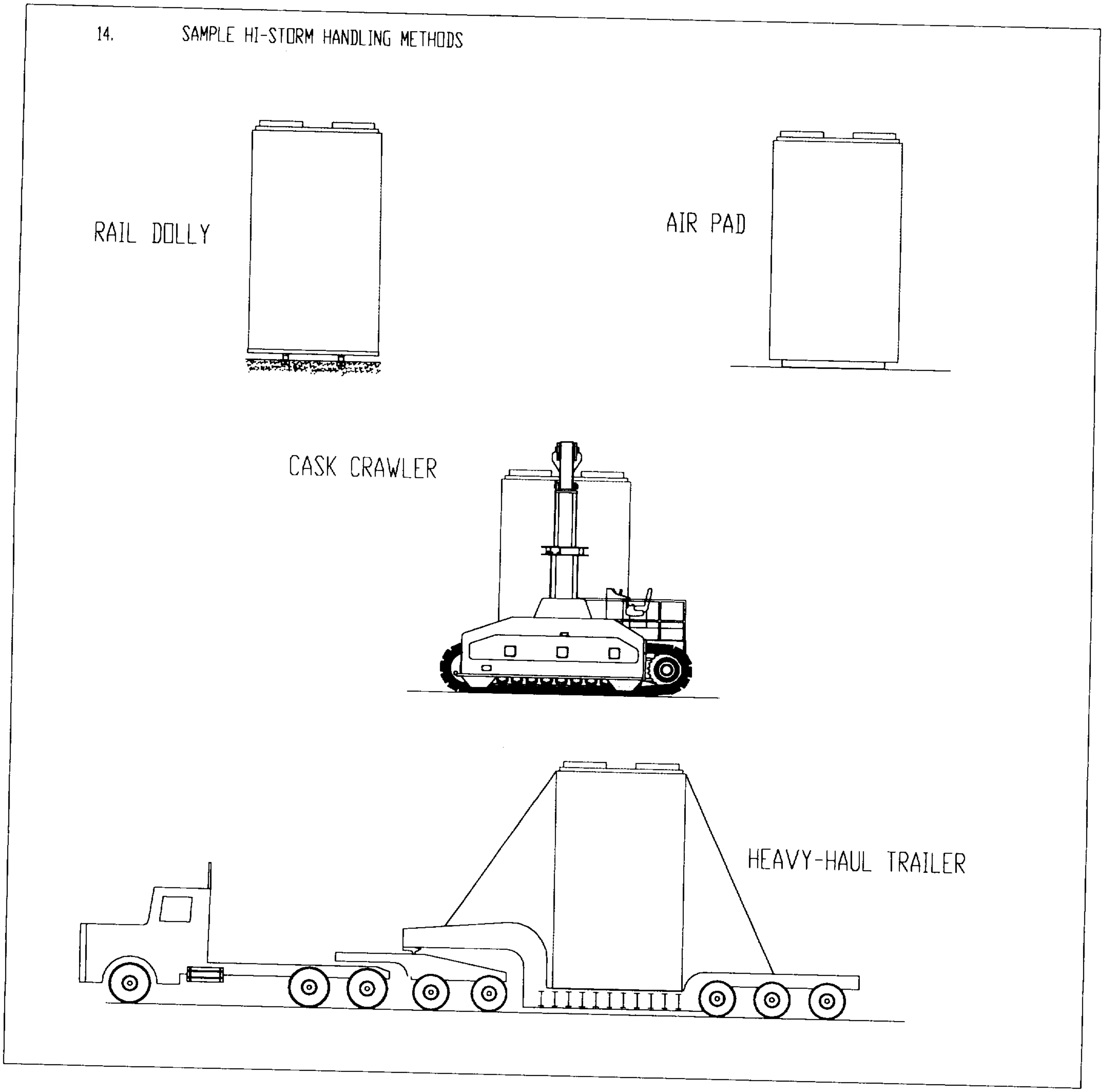


Figure 1.2.16e; Example of HI-STORM 100 Handling Options (Sheet 5 of 6)

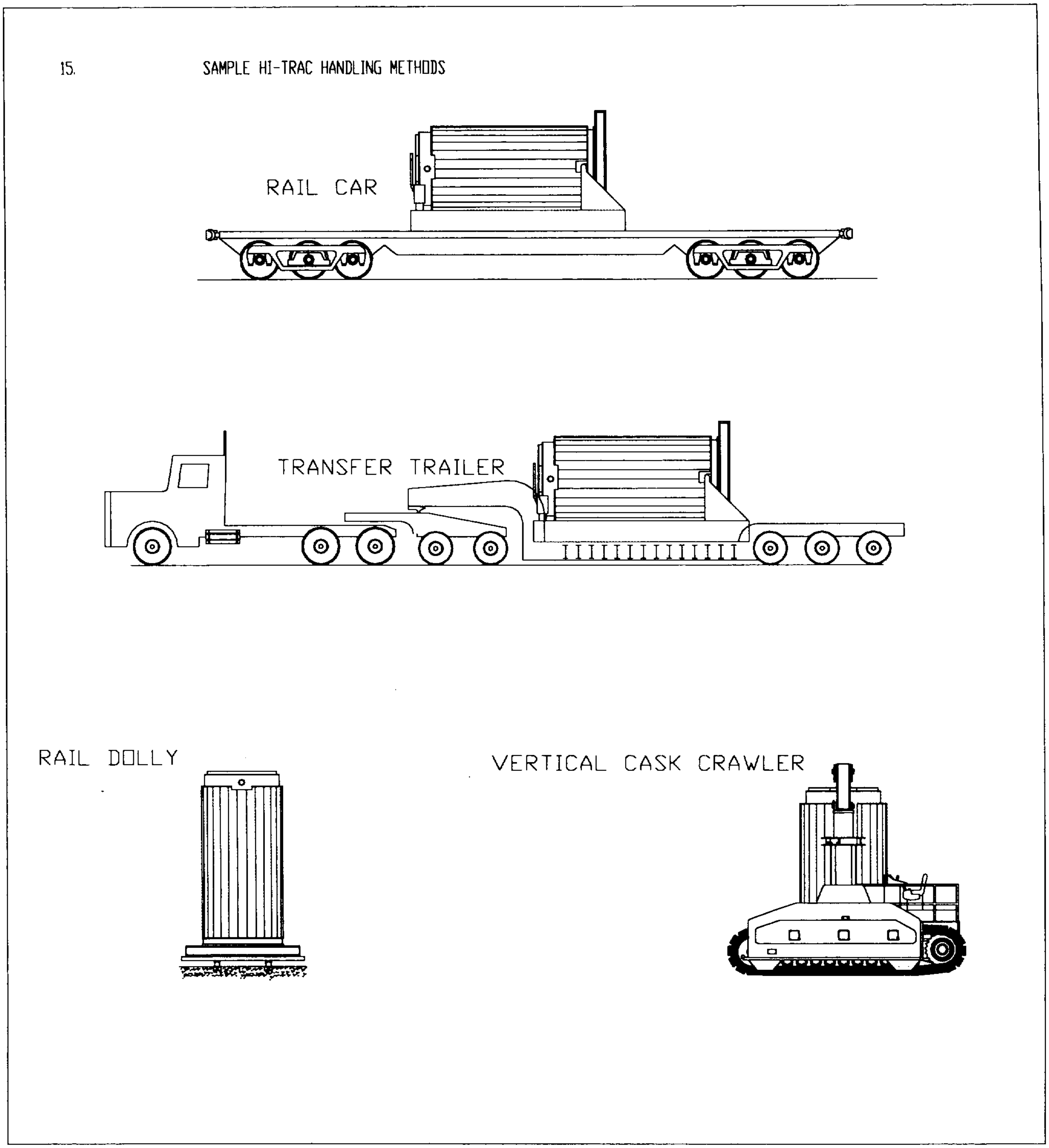


Figure 1.2.16f; Example of HI-TRAC Handling Options (Sheet 6 of 6)

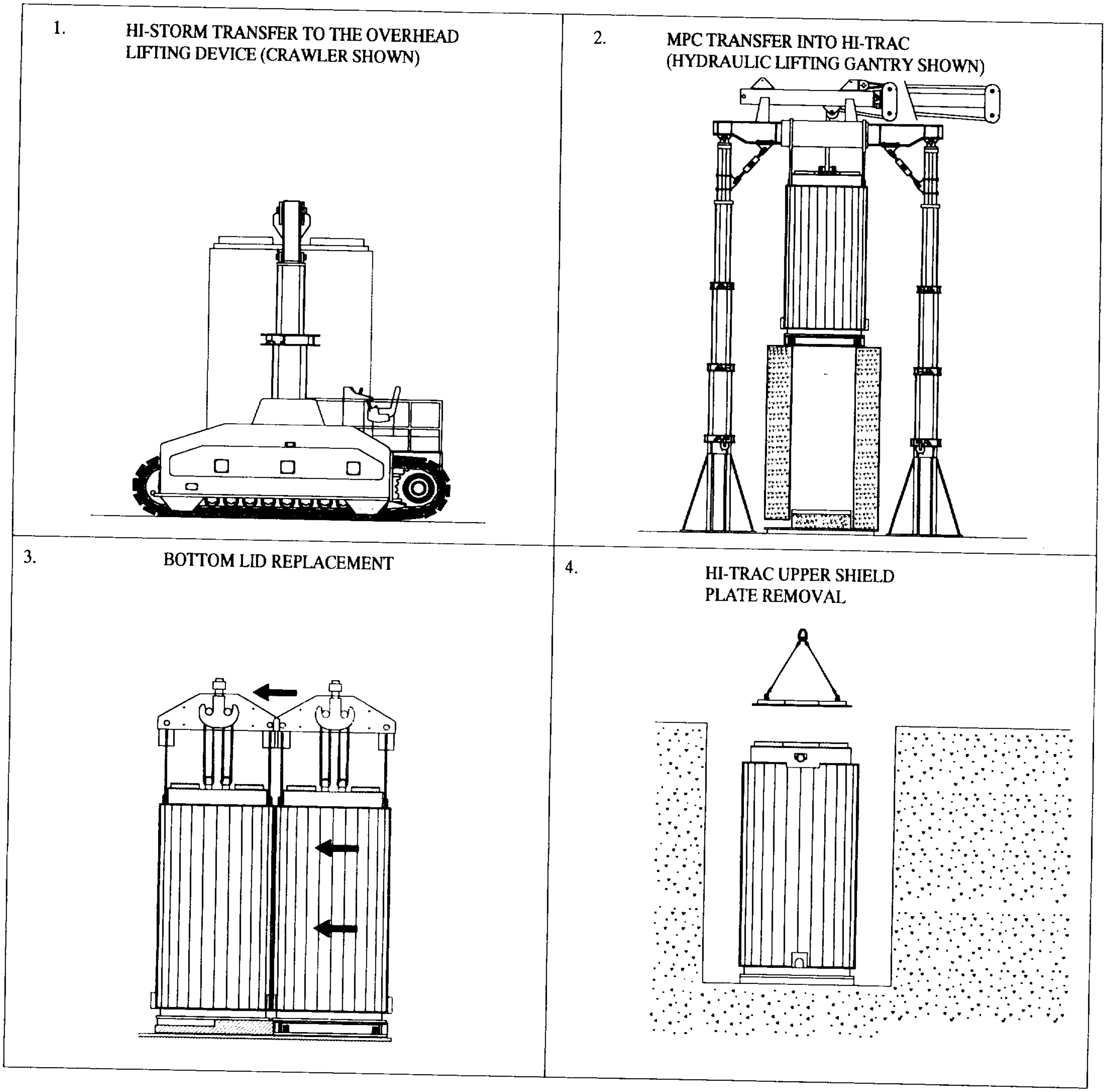


Figure 1.2.17a; Major HI-STORM 100 Unloading Operations (Sheet 1 of 4)

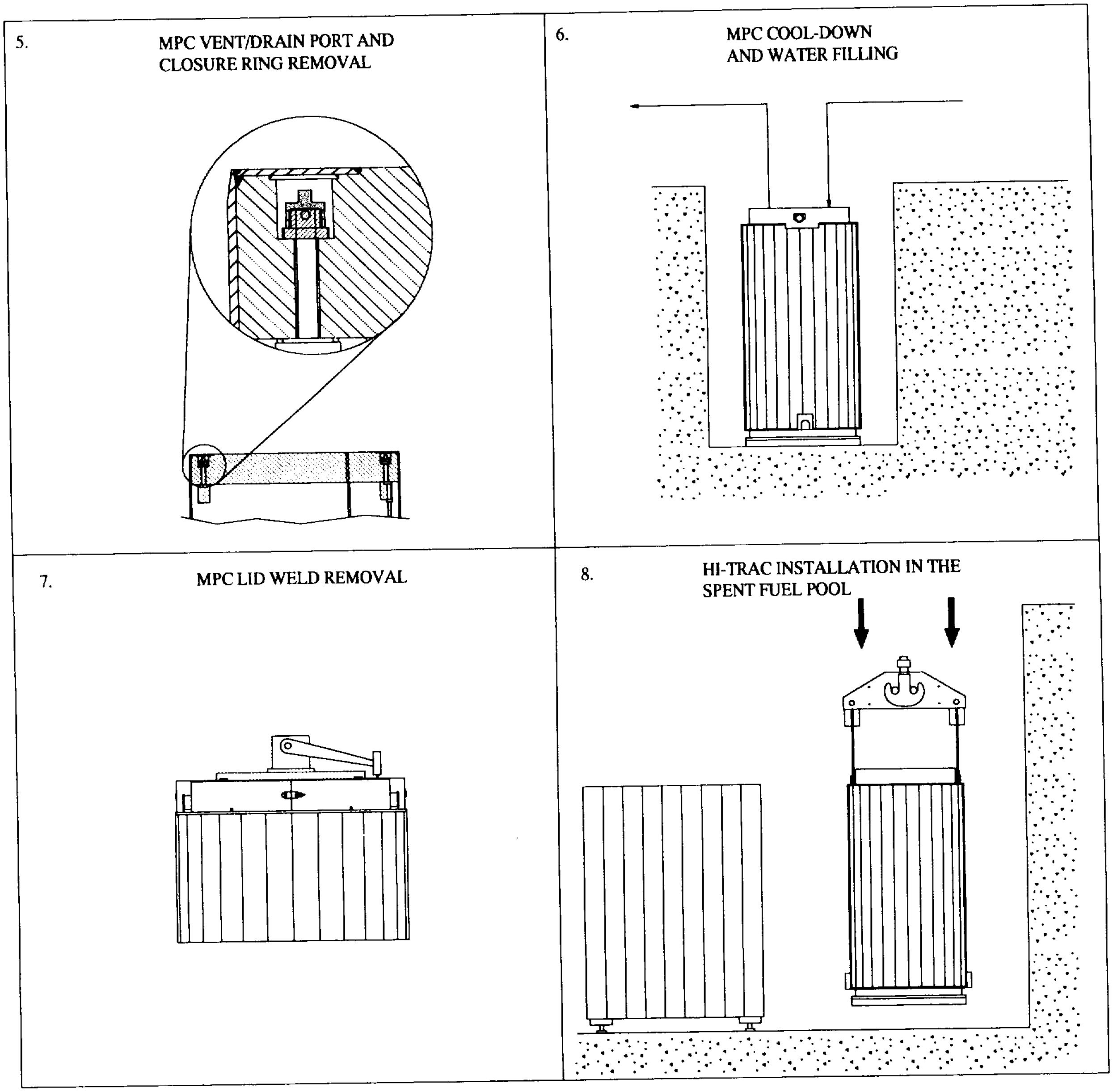


Figure 1.2.17b; Major HI-STORM 100 Unloading Operations (Sheet 2 of 4)

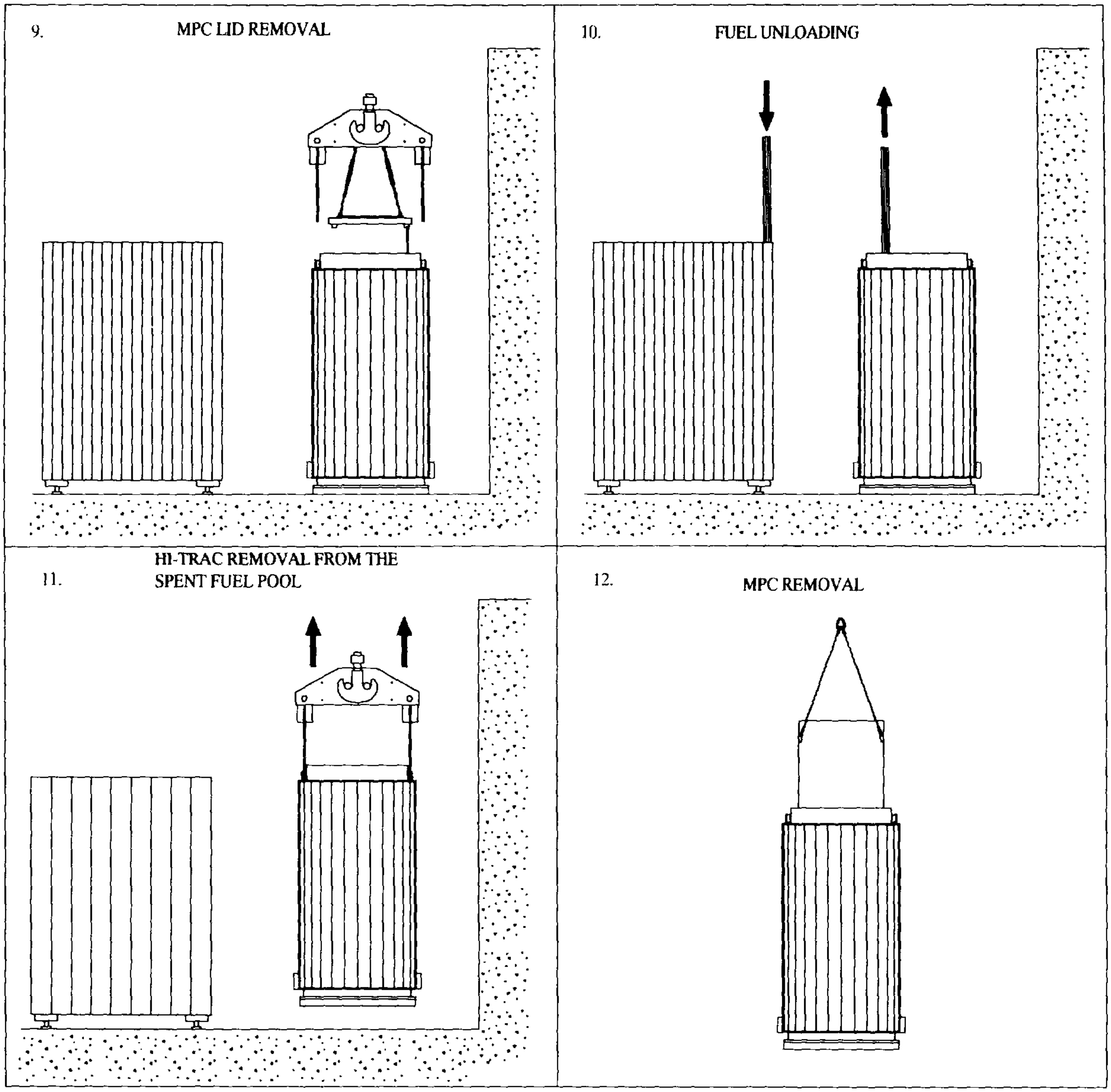


Figure 1.2.17c; Major HI-STORM 100 Unloading Operations (Sheet 3 of 4)

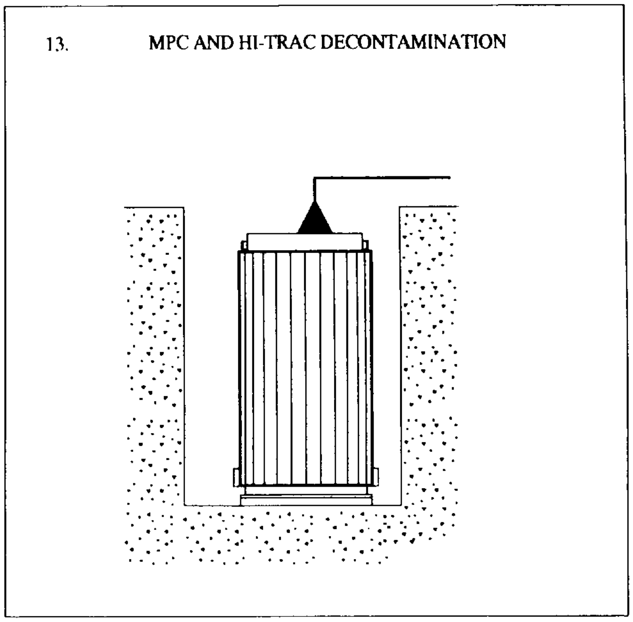


Figure 1.2.17d; Major HI-STORM 100 Unloading Operations (Sheet 4 of 4)

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. A description of the quality assurance program and its method of satisfying 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 provided in Chapter 13.

It is currently planned that the HI-STORM 100 System will be fabricated by U.S. Tool & Die, Inc. (UST&D) of Pittsburgh, Pennsylvania. UST&D is an N-Stamp holder and a highly respected fabricator of nuclear components. UST&D 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 UST&D 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 described in Chapter 13.

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 heat transfer considerations. Tables 1.4.1 and 1.4.2 provide the minimum pitch requirements, determined by heat transfer calculations in Chapter 4. The pitch values are minimums and may be increased to suit the user's specific needs. If MPC transfer operations between the HI-TRAC transfer cask and HI-STORM overpack are to be performed on the ISFSI pad, the minimum cask pitch values may not provide sufficient clearance to perform the transfer operations. An alternative to performing the MPC transfer operations on the ISFSI pad is to provide a separate MPC transfer area at the ISFSI.

For array(s) of two by N casks, the pitch between adjacent rows of casks (P1) and between each adjacent column of casks (P2) shall be in accordance with 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 shall be in accordance with Table 1.4.2. See Figure 1.4.2 for further clarification. The total quantity of rows and columns is unlimited provided the minimum pitch specified in Table 1.4.2 is met.

Table 1.4.1

CASK LAYOUT MINIMUM PITCH DATA FOR 2 BY N ARRAYS

Orientation	**Minimum 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 MINIMUM PITCH DATA FOR SQUARE ARRAYS

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

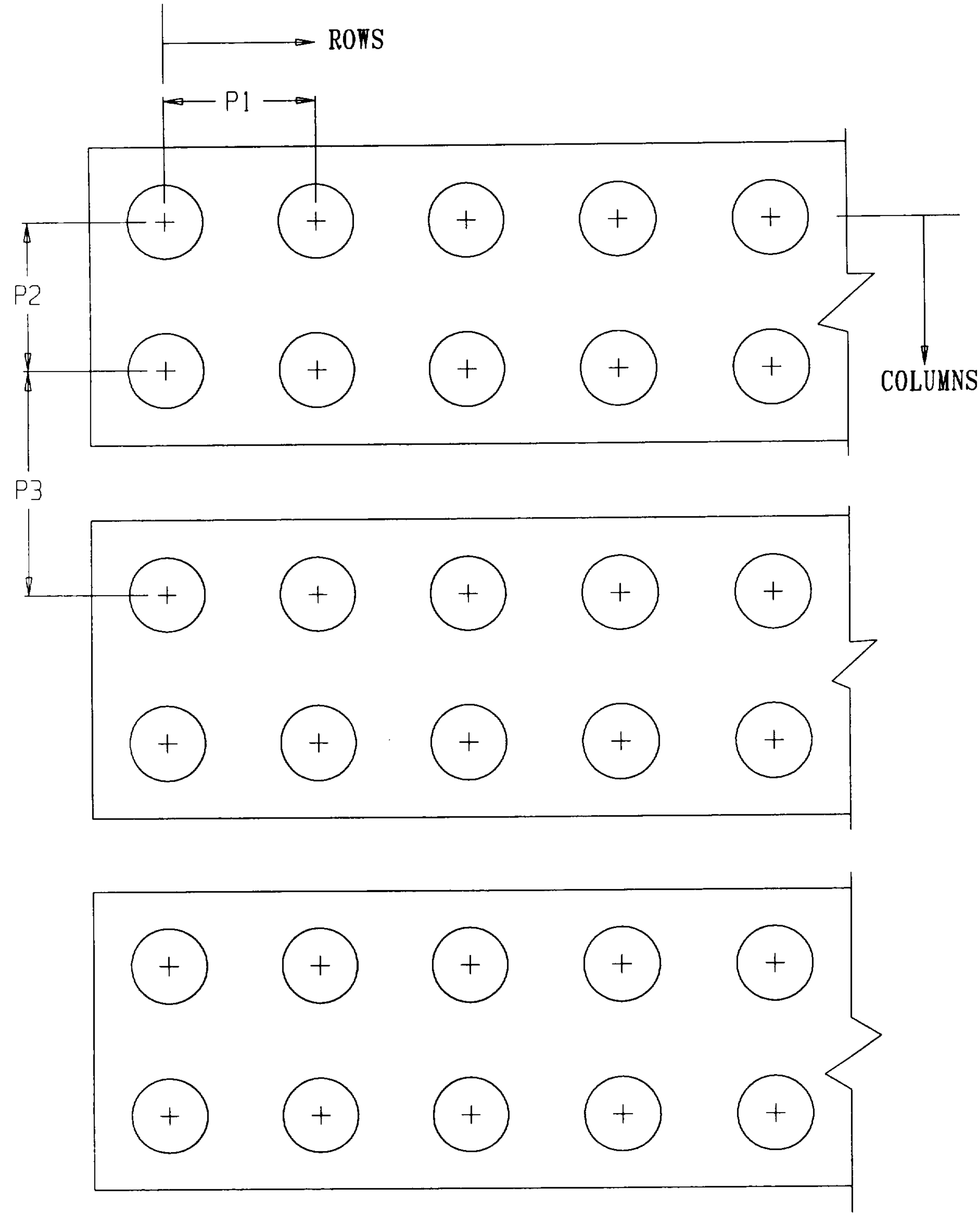


FIGURE 1.4.1; CASK LAYOUT PITCH REQUIREMENTS BASED ON 2 BY N ARRAY(S)

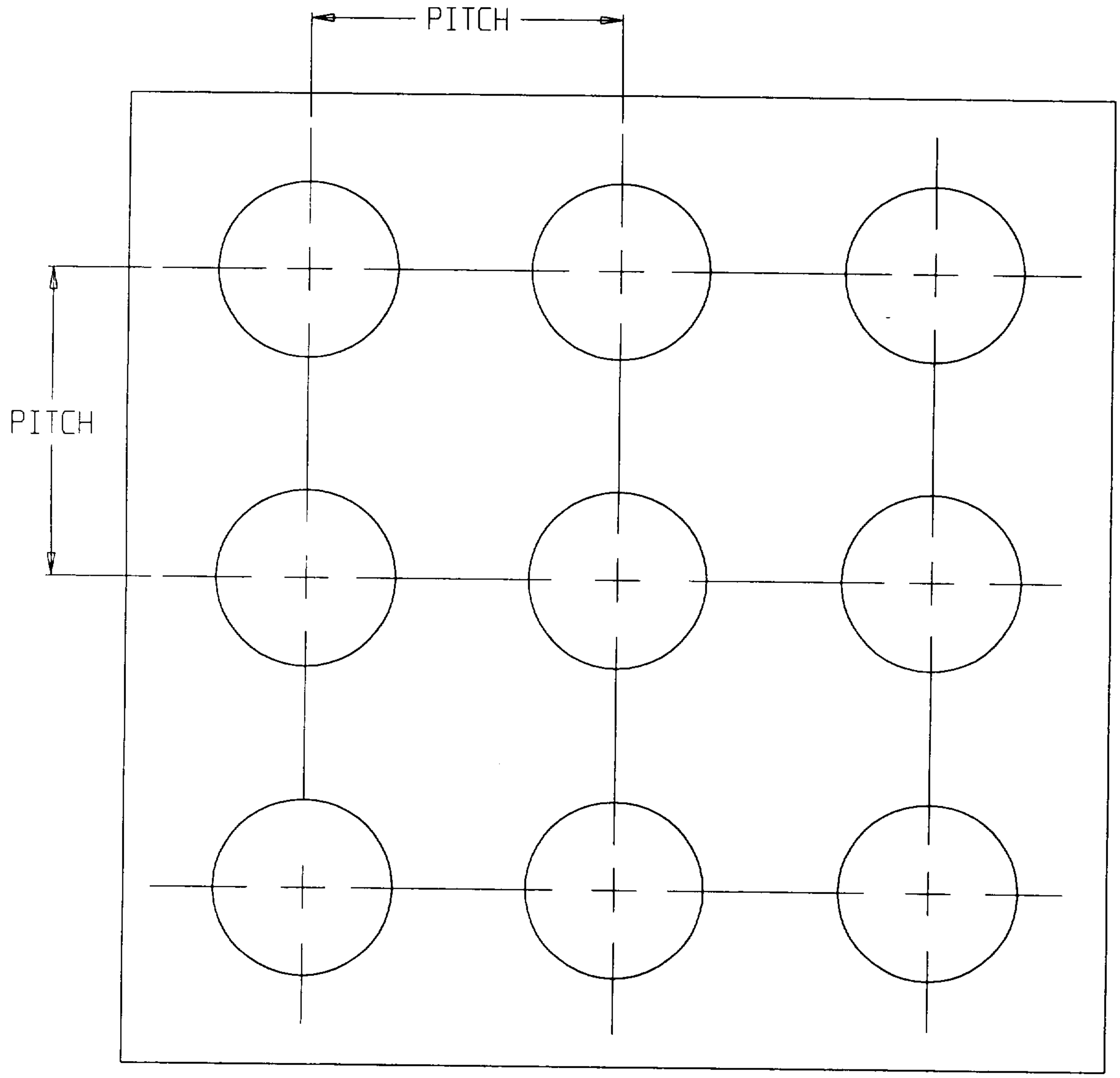


FIGURE 1.4.2; CASK LAYOUT PITCH REQUIREMENTS BASED ON A SQUARE ARRAY

1.5 GENERAL ARRANGEMENT DRAWINGS

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

Drawing Number/Sheet	Description	Rev.
5014-1395 Sht 1/4	HI-STAR 100 MPC-24 Construction	10
5014-1395 Sht 2/4	HI-STAR 100 MPC-24 Construction	9
5014-1395 Sht 3/4	HI-STAR 100 MPC-24 Construction	9
5014-1395 Sht 4/4	HI-STAR 100 MPC-24 Construction	8
5014-1396 Sht 1/6	HI-STAR 100 MPC-24 Construction	12
5014-1396 Sht 2/6	HI-STAR 100 MPC-24 Construction	9
5014-1396 Sht 3/6	HI-STAR 100 MPC-24 Construction	9
5014-1396 Sht 4/6	HI-STAR 100 MPC-24 Construction	8
5014-1396 Sht 5/6	HI-STAR 100 MPC-24 Construction	8
5014-1396 Sht 6/6	HI-STAR 100 MPC-24 Construction	7
5014-1401 Sht 1/4	HI-STAR 100 MPC-68 Construction	11
5014-1401 Sht 2/4	HI-STAR 100 MPC-68 Construction	8
5014-1401 Sht 3/4	HI-STAR 100 MPC-68 Construction	9
5014-1401 Sht 4/4	HI-STAR 100 MPC-68 Construction	8
5014-1402 Sht 1/6	HI-STAR 100 MPC-68 Construction	13
5014-1402 Sht 2/6	HI-STAR 100 MPC-68 Construction	11
5014-1402 Sht 3/6	HI-STAR 100 MPC-68 Construction	11
5014-1402 Sht 4/6	HI-STAR 100 MPC-68 Construction	9
5014-1402 Sht 5/6	HI-STAR 100 MPC-68 Construction	9
5014-1402 Sht 6/6	HI-STAR 100 MPC-68 Construction	7
5014-1495 Sht 1/6	HI-STORM 100 Assembly	8

Drawing Number/Sheet	Description	Rev.
5014-1495 Sht 2/6	Cross Section "Z" - "Z" View of HI-STORM	9
5014-1495 Sht 3/6	Section "Y" - "Y" of HI-STORM	7
5014-1495 Sht 4/6	Section "X" -"X" of HI-STORM	8
5014-1495 Sht 5/6	Section "W" -"W" of HI-STORM	9
5014-1495 Sht 6/6	HI-STORM Outlet Vent Thermocouple Mounting Hardware	3
5014-1561 Sht 1/5	View "A" -"A" of HI-STORM	7
5014-1561 Sht 2/5	Detail "B" of HI-STORM	7
5014-1561 Sht 3/5	Detail of Air Inlet of HI-STORM	7
5014-1561 Sht 4/5	Detail of Air Outlet of HI-STORM	7
5014-1561 Sht 5/5	Miscellaneous Detail of HI-STORM	7
5014-1783 Sht 1/1	General Arrangement Damaged Fuel Container	2
5014-1784 Sht 1/1	Damaged Fuel Container Details	1
5014-1880 Sht 1/10	125 Ton HI-TRAC Outline with Pool Lid	7
5014-1880 Sht 2/10	125 Ton HI-TRAC Body Sectioned Elevation	8
5014-1880 Sht 3/10	125 Ton HI-TRAC Body Sectioned Elevation "B" - "B"	7
5014-1880 Sht 4/10	125 Ton Transfer Cask Detail of Bottom Flange	8
5014-1880 Sht 5/10	125 Ton Transfer Cask Detail of Pool Lid	8
5014-1880 Sht 6/10	125 Ton Transfer Cask Detail of Top Flange	8
5014-1880 Sht 7/10	125 Ton Transfer Cask Detail of Top Lid	7
5014-1880 Sht 8/10	125 Ton Transfer Cask View "Y" - "Y"	7
5014-1880 Sht 9/10	125 Ton Transfer Cask Lifting Trunnion and Locking Pad	5
5014-1880 Sht 10/10	125 Ton Transfer Cask View "Z" - "Z"	7

Drawing Number/Sheet	Description	Rev.
5014-1928 Sht 1/2	125 Ton HI-TRAC Transfer Lid Housing Detail	8
5014-1928 Sht 2/2	125 Ton HI-TRAC Transfer Lid Door Detail	8
5014-2145 Sht 1/10	100 Ton HI-TRAC Outline with Pool Lid	6
5014-2145 Sht 2/10	100 Ton HI-TRAC Body Sectioned Elevation	6
5014-2145 Sht 3/10	100 Ton HI-TRAC Body Sectioned Elevation 'B-B'	6
5014-2145 Sht 4/10	100 Ton HI-TRAC Detail of Bottom Flange	5
5014-2145 Sht 5/10	100 Ton HI-TRAC Detail of Pool Lid	4
5014-2145 Sht 6/10	100 Ton HI-TRAC Detail of Top Flange	6
5014-2145 Sht 7/10	100 Ton HI-TRAC Detail of Top Lid	6
5014-2145 Sht 8/10	100 Ton HI-TRAC View Y-Y	6
5014-2145 Sht 9/10	100 Ton HI-TRAC Lifting Trunnions and Locking Pad	4
5014-2145 Sht 10/10	100 Ton HI-TRAC View Z-Z	5
5014-2152 Sht 1/2	100 Ton HI-TRAC Transfer Lid Housing Detail	6
5014-2152 Sht 2/2	100 Ton HI-TRAC Transfer Lid Door Detail	6
BM-1478, Sht 1/2	Bills-of-Materials for 24-Assembly HI-STAR 100 PWR MPC	9
BM-1478, Sht 2/2	Bills of Material for 24-Assembly HI-STAR 100 PWR MPC	11
BM-1479, Sht 1/2	Bills-of-Material for 68-Assembly HI-STAR 100 BWR MPC	10
BM-1479, Sht 2/2	Bills-of-Material for 68-Assembly HI-STAR 100 BWR MPC	13
BM-1575, Sht 1/2	HI-STORM 100 Storage Overpack Bill of Materials	8
BM-1575, Sht 2/2	HI-STORM 100 Storage Overpack Bill of Materials	8
BM-1819, Sht 1/1	Bills-of-Materials for HI-STAR 100 System Failed Fuel Canister	1

Drawing Number/Sheet	Description	Rev.
BM-1880, Sht 1/2	Bill of Material for 125 Ton HI-TRAC	7
BM-1880, Sht 2/2	Bill of Material for 125 Ton HI-TRAC	5
BM-1928, Sht 1/1	Bill of Material for 125 Ton HI-TRAC Transfer Lid	7
BM-2145 Sht 1/2	Bills-of-Materials for 100 Ton HI-TRAC	4
BM-2145 Sht 2/2	Bills-of-Materials for 100 Ton HI-TRAC	3
BM-2152 Sht 1/1	Bills-of-Materials for 100 Ton HI-TRAC Transfer Lid	5

Notes: 1. The HI-STAR 100 MPCs are identical to the MPCs used in the HI-STORM 100 System.

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 10
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1395 SHT 1 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 9
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1395 SHT 2 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN · ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 9
PROJECT No. 5014	DRAWING No. 1395 SHT 3 OF 4
P.O. No. N/A	

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 8
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1395 SHT 4 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-24 CONSTRUCTION

CLIENT N/A

COMPANION DRAWINGS
1395

REV.
12

PROJECT No. 5014

DRAWING No.

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395	REV. 9
PROJECT No. 5014	DRAWING No.

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN · ANALYSIS · CONSULTING	
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION		
CLIENT N/A		
COMPANION DRAWINGS 1395		REV. 9
PROJECT No. 5014	DRAWING No. 1396 SHT 3 OF 8	
P.O. No. N/A		

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395	REV. 8
PROJECT No. 5014	DRAWING No. 1396 SHT 4 OF 8
P.O. No. N/A	

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395	REV. 8
PROJECT No. 5014	DRAWING No.

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395	REV. 7
PROJECT No. 5014	DRAWING No.

FIGURE WITHHELD UNDER 10 CFR 2.390

BY:	BY:		
REVISION			
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN · ANALYSIS · CONSULTING		
DESCRIPTION	HI-STAR 100 MPC-68 CONSTRUCTION		
CLIENT	N/A		
COMPANION DRAWINGS 1402		REV. 11	
PROJECT No. 5014		DRAWING No.	
P.O. No. N/A		1401 SHT 1 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1402	REV. 8
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1401 SHT 2 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

BY:	BY:		
REVISION			
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING		
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION			
CLIENT N/A			
COMPANION DRAWINGS 1402			REV. 9
PROJECT No. 5014		DRAWING No. 1401 SHT 3 OF 4	
P.O. No. N/A			

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1402	REV. 8
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1401 SHT 4 OF 4

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 13
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1402 SHT 1 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 11
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1402 SHT 2 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 11
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1402 SHT 3 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 9
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1402 SHT 4 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 9
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1402 SHT 5 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 7
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1402 SHT 6 OF 6

FIGURE WITHHELD UNDER 10 CFR 2.390

DESCRIPTION: HI-STORM ASSEMBLY			
CLIENT: N/A			
COMPANION DRAWINGS:			
PROJECT No.: 5014	SCALE: .06 = 1	DRAWING No.: 1495	REV.: 8
P.O. No.: N/A			SHEET: 1 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN · ANALYSIS · CONSULTING

DESCRIPTION: CROSS SECTION "Z" - "Z" VIEW OF HI-STORM			
CLIENT: N/A			
COMPANION DRAWINGS: 1561, B/M 1575			
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 1495	REV.: 9
P.O. No.: N/A			SHEET: 2 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: SECTION "Y" - "Y" OF HI-STORM			
CLIENT: N/A			
COMPANION DRAWINGS: 1561, B/M 1575			
PROJECT No.: 5014	SCALE: .09375 = 1	DRAWING No.: 1495	REV.: 7
P.O. No.: N/A			SHEET: 3 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL		EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: SECTION "X" - "X" OF HI-STORM		
CLIENT: N/A		
COMPANION DRAWINGS: 1561, B/M 1575		
PROJECT No.: 5014	SCALE: .09375 = 1	DRAWING No.: 1495 (E.I.0.2834)	REV.: 8
P.O. No.: N/A			SHEET: 4 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN · ANALYSIS · CONSULTING	
DESCRIPTION: SECTION "W" - "W" OF HI-STORM		
CLIENT: N/A		
COMPANION DRAWINGS: 1561, B/M 1575		
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 1495
P.O. No.: N/A	REV.: 9	SHEET: 5 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: HI-STORM OUTLET VENT TEMPERATURE ELEMENT MOUNTING HARDWARE			
CLIENT: N/A			
COMPANION DRAWINGS: 1561, B/M 1575			
PROJECT No.: 5014	SCALE: N/A	DRAWING No.: 1495 (E.I.U. 2835)	REV.: 3
P.O. No.: N/A			SHEET: 6 of 6

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: VIEW "A" - "A" OF HI-STORM			
CLIENT: N/A			
COMPANION DRAWINGS: 1495, B/M 1575			
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 1561	REV.: 7
P.O. No.: N/A			SHEET: 1 of 5

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: DETAIL "B" OF HI-STORM			
CLIENT: N/A			
COMPANION DRAWINGS: 1495, B/M 1575			
PROJECT No.: 5014	SCALE: .25 = 1	DRAWING No.: 1561	REV.: 7
P.O. No.: N/A			SHEET: 2 of 5

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION:
DETAIL OF AIR INLET
OF HI-STORM

CLIENT:
N/A

COMPANION DRAWINGS:
1495, B/M 1575

PROJECT No.: 5014	SCALE: NONE	DRAWING No.: 1561	REV.: 7
P.O. No.: N/A			SHEET: 3 of 5

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: DETAIL OF AIR OUTLET OF HI-STORM			
CLIENT: N/A			
COMPANION DRAWINGS: 1495, B/M 1575			
PROJECT No.: 5014	SCALE: NONE	DRAWING No.: 1561	REV.: 7
P.O. No.: N/A			SHEET: 4 of 5

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL		EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: MISCELLANEOUS DETAIL OF HI-STORM		
CLIENT: N/A		
COMPANION DRAWINGS: BM 1884		
PROJECT No.: 5014	SCALE: AS NOTED	DRAWING No.: 1561	REV.: 7
P.O. No.: N/A			SHEET: 5 of 5

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: GENERAL ARRANGEMENT DAMAGED FUEL CONTAINER	
CLIENT: LATER	
COMPANION DRAWINGS: 1784	
PROJECT No.: 5014	DRAWING No.: 1783
P.O. No.: LATER	REV.: 2

FIGURE WITHHELD UNDER 10 CFR 2.390

REVISION			
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING		
DESCRIPTION: DAMAGED FUEL CONTAINER DETAILS			
CLIENT: LATER			
COMPANION DRAWINGS: 1783		PREP. BY: P.J.N. 8-12-96	
PROJECT No.: 5014	SCALE: FULL	DRAWING No.: 1784	REV.: 1
P.O. No.: LATER			SHEET: 1 of 1

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: 125TON HI-TRAC OUTLINE WITH POOL LID			
CLIENT: N/A			
COMPANION DRAWINGS: B/M 1880			
PROJECT No.: 5014	SCALE: .04 = 1	DRAWING No.: 1880	REV.: 7
P.O. No.: N/A			SHEET: 1 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION:

125TON HI-TRAC BODY
SECTIONED ELEVATION

CLIENT: N/A

COMPANION DRAWINGS: B/M 1880

PROJECT No.: 5014	SCALE: .10 = 1	DRAWING No.: 1880	REV.: 8
P.O. No.: N/A			SHEET: 2 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: 125TON HI-TRAC BODY SECTIONED ELEVATION "B"-"B"	
CLIENT:	
COMPANION DRAWINGS: B/M 1880	
PROJECT No.: 5014	SCALE: .08 = 1
DRAWING No.: 1880	REV.: 7
P.O. No.:	SHEET: 3 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION:

125TON TRANSFER CASK
DETAIL OF BOTTOM FLANGE

CLIENT:

COMPANION DRAWINGS:
B/M 1880

PROJECT No.: 5014	SCALE: .08 = 1	DRAWING No.:	REV.: 8
P.O. No.:		1880	SHEET: 4 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION:

125TON TRANSFER CASK
DETAIL OF POOL LID

CLIENT:

COMPANION DRAWINGS:
B/M 1880

PROJECT No.:	SCALE:	DRAWING No.:	REV.:
5014	.08 = 1	1880	8
P.O. No.:			SHEET: 5 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL		EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION: 125TON TRANSFER CASK DETAIL OF TOP FLANGE			
CLIENT:			
COMPANION DRAWINGS: B/M 1880			
PROJECT No.: 5014	SCALE: .10 = 1	DRAWING No.: 1880	REV.: 8
P.O. No.:			SHEET: 6 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: 125TON TRANSFER CASK DETAIL OF TOP LID			
CLIENT:			
COMPANION DRAWINGS: B/M 1880			
PROJECT No.: 5014	SCALE: .125 = 1	DRAWING No.: 1880	REV.: 7
P.O. No.:			SHEET: 7 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION:

125TONTRANSFER CASK
VIEW "Y"-"Y"

CLIENT:

COMPANION DRAWINGS:

B/M 1880

PROJECT No.:	SCALE:	DRAWING No.:	REV.:
5014	.1875 = 1	1880	7
P.O. No.:			SHEET: 8 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: 125TON TRANSFER CASK LIFTING TRUNNION AND LOCKING PAD			
CLIENT:			
COMPANION DRAWINGS: B/M 1880			
PROJECT No.: 5014	SCALE: NONE	DRAWING No.: 1880	REV.: 5
P.O. No.:			SHEET: 9 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: 125TON TRANSFER CASK VIEW "Z"-"Z"

CLIENT:

COMPANION DRAWINGS: B/M 1880

PROJECT No.:	SCALE:	DRAWING No.:	REV.:
5014	.20 = 1	1880	7
P.O. No.:			SHEET: 10 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN · ANALYSIS · CONSULTING
DESCRIPTION: 125TON HI-TRAC TRANSFER LID HOUSING DETAIL	
CLIENT: N/A	
COMPANION DRAWINGS: B/M 1884	
PROJECT No.: 5014	SCALE: .0625 = 1
P.O. No.: N/A	DRAWING No.: 1928
REV.: 8	SHEET: 1 of 2

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING		
DESCRIPTION: 125TON HI-TRAC TRANSFER LID DOOR DETAIL			
CLIENT: N/A			
COMPANION DRAWINGS: B/M 1884			
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 1928	REV.: 8
P.O. No.: N/A			SHEET: 2 of 2

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION:
100 TON HI-TRAC OUTLINE
WITH POOL LID

CLIENT: N/A

COMPANION DRAWINGS: BM 2145

PROJECT No.: 5014	SCALE: .04 = 1	DRAWING No.: 2145	REV.: 6
P.O. No.: N/A			SHEET: 1 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING		
DESCRIPTION: 100 TON HI-TRAC BODY SECTIONED ELEVATION			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2145			
PROJECT No.: 5014	SCALE: .10 = 1	DRAWING No.: 2145	REV.: 6
P.O. No.: N/A			SHEET: 2 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: 100 TON HI-TRAC BODY SECTIONED ELEVATION 'B-B'			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2145			
PROJECT No.: 5014	SCALE: .08 = 1	DRAWING No.: 2145	REV.: 6
P.O. No.: N/A			SHEET: 3 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: 100 TON HI-TRAC DETAIL OF BOTTOM FLANGE	
CLIENT: N/A	
COMPANION DRAWINGS: BM 2145	
PROJECT No.: 5014	SCALE: .08 = 1
P.O. No.: N/A	DRAWING No.: 2145
REV.: 5	SHEET: 4 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: 100 TON HI-TRAC DETAIL OF POOL LID			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2145			
PROJECT No.: 5014	SCALE: .08 = 1	DRAWING No.: 2145	REV.: 4
P.O. No.: N/A			SHEET: 5 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: 100 TON HI-TRAC DETAIL OF TOP FLANGE	
CLIENT: N/A	
COMPANION DRAWINGS: BM 2145	
PROJECT No.: 5014 / SCALE: 10 = 1 / P.O. No.: N/A	DRAWING No.: 2145 / REV.: 6 / SHEET: 6 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: 100 TON HI-TRAC DETAIL OF TOP LID			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2145			
PROJECT No.: 5014	SCALE: .125 = 1	DRAWING No.: 2145	REV.: 6
P.O. No.: N/A			SHEET: 7 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: 100 TON HI-TRAC VIEW Y-Y			
CLIENT:			
COMPANION DRAWINGS: N/A			
COMPANION DRAWINGS: BM 2145			
PROJECT No.: 5014	SCALE: .1875 = 1	DRAWING No.: 2145	REV.: 6
P.O. No.: N/A			SHEET: 8 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: 100 TON HI-TRAC LIFTING TRUNNION AND LOCKING PAD	
CLIENT: N/A	
COMPANION DRAWINGS: BM 2145	
PROJECT No.: 5014	SCALE: NONE
P.O. No.: N/A	DRAWING No.: 2145 (E.I.D. 3057)
REV.: 4	SHEET: 9 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION:
100 TON HI-TRAC
VIEW Z-Z

CLIENT:
N/A

COMPANION DRAWINGS:
BM 2145

PROJECT No.:	SCALE:	DRAWING No.:	REV.:
5014	.20 = 1	2145	5
P.O. No.: N/A			SHEET: 10 of 10

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION: 100 TON HI-TRAC TRANSFER LID HOUSING DETAIL			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2152			
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 2152	REV.: 6
P.O. No.: N/A			SHEET: 1 of 2

FIGURE WITHHELD UNDER 10 CFR 2.390

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION: 100 TON HI-TRAC TRANSFER LID DOOR DETAIL			
CLIENT: N/A			
COMPANION DRAWINGS: BM 2152			
PROJECT No.: 5014	SCALE: .0625 = 1	DRAWING No.: 2152	REV.: 6
P.O. No.: N/A			SHEET: 2 of 2

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1478)

REF. DWG. 1395 & 1396.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
9	T.C. 6-4-98 INCORPORATED DCR 5014-116	6/5/98	6/5/98	6-7-98

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	2	ALLOY "X" SEE NOTE 1.	PLATE 5/16" THK.X 63.20" REF W. X 176 1/2" LG	BASKET CELL PLATE
1B	1	↓	PLATE 5/16" THK X 60.57" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1C	2	↓	PLATE 5/16" THK X 43.42" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1D	1	↓	PLATE 5/16" THK X 20.402 " REF W. X 176 1/2"LG	BASKET CELL PLATE
1E	1	↓	PLATE 5/16" THK X 7.7175" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1F	22	↓	PLATE 5/16" THK X10.4625 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
1G	1	↓	PLATE 5/16" THK X 9.7445 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
1H	2	↓	PLATE 5/16" THK X 9.03 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
2	24	↓	PIPE 3"-SCH 80 LGTH AS REQD.	UPPER FUEL SPACER PIPE
3A (3B)	84(12)	BORAL	.075"THK. X 7.5"W.(6 1/4") X 156" LG.PER DET.DWG.1395. SEE NOTE 2.	NEUTRON ABSORBER
4A (4B)	84(12)	ALLOY "X" SEE NOTE 1.	.06" THK. SHEATHING PER DET. DWG. 1395.	SHEATHING
5A	4	↓	PLATE 5/16"THK X 3" W. X 176 1/2"LG.	BASKET CELL PLATE
5B	4	↓	PLATE 5/16"THK X 3 3/4" APPROX. W. X 176 1/2"LG.	BASKET CELL PLATE
5C	4	↓	PLATE 1.5" APP. THK. X 3" W. X 168"LG.	BASKET SUPPORT
5D	4	↓	2 1/2"WIDE X 168"LG. THICKNESS AS REQD	BASKET SUPPORT
5E	4	↓	2 "WIDE X 168"LG. THICKNESS AS REQD.	BASKET SUPPORT
5F	---	↓	DELETED	----
5G	4	↓	1 1/4" WIDE X 168" LG. THICKNESS AS REQD.	BASKET SUPPORT SHIM
5H	---	↓	DELETED	----
6	1	↓	1/2" THK X 68 3/8" O.D. X 187 5/8" LG. CYLINDER.	SHELL
7	1	↓	BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8A	22	↓	9/32"THK. ANGLE X 176 1/2" LG. FROM PLATE PER DET. DWG.1395.	BASKET CELL ANGLE
8B	2	↓	9/32"THK. CHANNEL X 176 1/2" LG. FROM PLATE PER DET.DWG. 1395.	BASKET CELL CHANNEL
9A	1	↓	5/16" THK.X 10" W. APP. X 168" LG. PER DET.	BASKET SUPPORT
9B	2	↓	5/16"THK. X 7 1/2" APP. W. X 168" LG. PER DET.	BASKET SUPPORT
9C	1	↓	5/16"THK. X 5" APP. W. X 168" LG. PER DET.	BASKET SUPPORT
9D	---	↓	DELETED	---
9E	---	↓	DELETED	---
9F	---	↓	DELETED	---
9G	---	↓	DELETED	---
9H	---	↓	DELETED	---
10	4	↓	PLATE 3/4" THK. X 3 1/2" WIDE X 8 3/4" LG.	LIFT LUG

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1478)

REF. DWG. 1395 & 1396.

SHEET 2 OF 2

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
11	S.GEE 8-19-98 INCORPORATED DCP 5014-121	A.S. 8/19/98	8/19/98	8-19-98

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
11	4	ALLOY "X" SEE NOTE 1.	PLATE 3/4" THK. X 4" WIDE X 3" LG.	LIFT LUG BASEPLATE
12	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S. SURFACE HARDENED	BAR 2 11/16" OD X 6.75" LG. DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE
13B	2	304 S/S. SURFACE HARDENED	BAR 2 1/4 OD X 2 1/4 LG. DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	PLATE 9 1/2" THK. X 67 1/4" O.D.	MPC LID
15	1	↓	RING 3/8" THK.X 53" ID. X 67 7/8" O.D.	MPC CLOSURE RING
16	AS REQD.	↓	SHELL SHIM SIZE AS REQD.	SHELL SHIM
17	----	----	DELETED	----
18	----	----	DELETED	----
19	2	ALLOY "X" SEE NOTE 1	PLATE 3/8" THK. X 3 7/8"OD.	PORT COVER PLATE
20	24	A-193-B8 OR SIMILAR	3/4"-10UNC X 1 3/8"LG. HEX BOLT WITH FULL THRD.	UPPER FUEL SPACER BOLT
21	---	---	DELETED	---
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4"LG SOCKET SET SCREW	LID LIFT HOLE PLUG
24	24	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 3 3/4" OD.	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1.	LENGTH, WIDTH AND THICKNESS OF SHIMS AS REQUIRED.	LID SHIM
26	1	S/S	2" FEMALE X 1 1/4" MALE SCH. 40. S/S REDUCER	REDUCER
27	--	-------	DELETED	----
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5.5"LG.	VENT QUICK DISCONN. CPLG.
29	4	ALLOY "X" SEE NOTE 1.	BAR 3/4" OD. X 1/2" LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173 1/2"APPROX. LG.	DRAIN LINE
31	--	-------	DELETED	-----
32	24	↓	6" SQ. TUBING X 1/4" WALL LENGTH AS REQ'D.	LOWER FUEL SPACER COLUMN
33A	24	↓	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPCAER END PLATE
33B	24	↓	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPCAER END PLATE
34	AS REQ'D.	↓	SHIMS 3/8", 3/4" & 2 1/2" WIDE X 168" LG. THICKNESS AS REQ'D.	BASKET SHIMS
35	AS REQ'D.	ALUM. ALLOY 1100	1/8" THICK X 176 1/2" LG. ALUM. SHEET	HEAT CONDUCTION ELEMENTS
36	2	ALUMINUM	0.065" THK 1.494 OD, 0.250" HOLE	SEAL WASHER
37	2	S/S	1/4" DIA X 3/8" LG	SEAL WASHER BOLT
38	2	ALLOY "X" SEE NOTE 1.	1/8" X 10 1/2" X 9 1/2" SHEET	DRAINLINE
39	8	ALLOY "X" SEE NOTE 1.	1/8" X 4" X 4 1/2" APPRX SHEET	DRAINLINE

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. MINIMUM BORAL B-10 LOADING IS 0.0267 g/cm^2 BORAL TO BE PASSIVATED PRIOR TO INSTALLATION.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5C,5D,5E,5G 9A,9B,9C,16,34 AND 35 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-1479)

REF. DWGS. 1401 & 1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
10	S.GIT 2 20 99 INCORPORATED DCR 5014-130	2/20/99	B.G. 2/20/99	2-20-99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	3	ALLOY "X" SEE NOTE 1.	PLATE 1/4" THK. X 65.65"W. X 176" LG PER DET. DWG.1401	BASKET CELL PLATE
1B	4		PLATE 1/4" THK. X 52.67"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1C	2		PLATE 1/4" THK. X 39.69W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1D	2		PLATE 1/4" THK. X 13.73"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1E	78		PLATE 1/4" THK. X 6.24"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
2	68		3"- SCH 80 PIPE LGTH AS REQD	UPPER FUEL SPCER CLUMN
3A	116	BORAL	.101"THK. X 4 3/4"W. X 156" LG.PER DET.DWG.1401. SEE NOTE 2.	NEUTRON ABSORBER
4A	116	ALLOY "X" SEE NOTE 1.	.075" THK. SHEATHING PER DET. DWG.1401.	SHEATHING
5	8		BAR 1" WIDE X 168.5" LG THICKNESS AS REQD.	BASKET SUPPORT SHIM
6	1		1/2" THK X 68 3/8" O.D. X 187 5/8" LG. CYLINDER.	SHELL
7	1		BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8	8		PLATE 5/16"THK. X 10" APPROX.W.X 168 1/2" LG. PER DET. DWG.1401.	BASKET SUPPORT
9A	4		BAR 1" W. X .8" APPROX.THK. X 168 1/2" LG.	BASKET SUPPORT
9B	8		2 1/2" W. X 168 1/2" LG. THICKNESS AS REQD. ROLL TO SHELL I.D.	BASKET SUPPORT
9C	8		2 1/2" WIDE X 168 1/2" LG. THICKNESS AS REQD. ROLL TO SHELL I.D.	BASKET SUPPORT
9D	--		DELETED	--------
10	4		PLATE 3/4" THK. X 3 1/2" WIDE X 8 3/4" LG.	LIFT LUG
11	4		PLATE 3/4" THK. X 2 1/2" WIDE X 4" LG.	LIFT LUG BASEPLATE
12	1		BAR 3.75"OD. X 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S - SURFACE HARDENED	BAR 2 11/16" OD X 6 3/4" REF LG, DIMENSION ON DWG 1402 SHT 4	VENT AND DRAIN TUBE
13B	2	304 S/S - SURFACE HARDENED	BAR 2 1/4" OD X 2 1/4" REF LG, DIMENSION ON DWG 1402 SHT 4	VENT AND DRAIN TUBE CAP
△10 14	1	ALLOY "X" SEE NOTE 1.	PLATE 10" THK. X 67 1/4" O.D. [MPC-68] PLATE 10" THK. X 66 1/4" O.D. [MPC-68F]	MPC LID
△10 15	1	ALLOY "X" SEE NOTE 1.	RING 3/8" THK. X 53" ID. X 67 7/8" O.D. [MPC-68] RING 3/8" THK. X 53" ID. X 67 3/8" O.D. [MPC-68F]	MPC CLOSURE RING
16	AS REQD.	ALLOY "X" SEE NOTE 1.	SHELL SHIMS SIZES AS REQD.	SHELL SHIM

BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-1479)

REF. DWGS. 1401 & 1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
13	S. GEE 2-20-99 INCORPORATED DCR 5014-130	[signature] 2/20/99	[signature] B.G. 2/20/99	[signature] U.O. 2-20-99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
17	1	ALLOY "X" SEE NOTE 1.	1" THK X 68 3/8" OD X 11 5/8" LG. CYLINDER (MPC-68F)	SHELL
18	---	---	DELETED	---
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 3 7/8" OD.	PORT COVER PLATE
20	68	A-193-B8 OR SIMILAR	3/4"-10UNC X 1.375"LG. FULL THRD. HEX. BOLT	UPPER FUEL SPACER BOLT
21	---	---	DELETED	---
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4" LG. SOCKET SET SCREW.	LIFT HOLE PLUG
24	68	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 3 3/4" OD.	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1	LENGTH, WIDTH, THICKNESS AND QUANTITY AS REQD	LID SHIM
26	1	S/S	2" FEMALE X 1 1/4" MALE SCH. 40, S/S REDUCER OR EQUIVALENT	REDUCER
27			DELETED	
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5.5" LG.	VENT SHIELD BLOCK
29	4	ALLOY "X" SEE NOTE 1.	BAR .75"OD X .5"LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173" APPROX.LG.	DRAIN LINE
31	--	--	DELETED	---
32	--	--	DELETD	---
33	68	ALLOY "X" SEE NOTE 1.	4" SQ. TUBE X 1/4" WALL LENGTH AS REQD. (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER COLUMN
34A	68	ALLOY "X" SEE NOTE 1.	3/8" THK. X 5 3/4" SQ. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
34B	68	ALLOY "X" SEE NOTE 1.	3/8" THK. X 5 3/4" SQ. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
35			DELETED	
36	AS REQD	ALLOY "X" SEE NOTE 1.	1" WIDE X 168" LG. THICKNESS AS REQD.	BASKET SHIM
37	AS REQD	ALUM. ALLOY 1100	1/8" THK. X 176" LG. ALUM. SHEET.	HEAT CONDUCTION ELEMENTS
38	2	ALUMINUM	.065" THK X 1.494 OD, .250 HOLE	SEAL WASHER
39	2	S/S	1/4" DIA X 3/8 LG	SEAL WASHER BOLT
40	2	ALLOY "X" SEE NOTE 1	1/8" THK. 6" X 6" APPROX. SHEET	DRAIN LINE
41	8	ALLOY "X" SEE NOTE 1.	1/8" THK. X 6" X 6" APPROX. SHEET	DRAIN LINE

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. MINIMUM BORAL B-10 LOADING IS 0.0372 g/cm^2. BORAL TO BE PASSIVATED PRIOR TO INSTALLATION. FOR MPC-68F, MINIMUM BORAL B-10 LOADING IS 0.01 g/cm^2.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5,8,9A,9B,9C,16,36 AND 37 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

BM-1575 (E.I.D. 2839) BILL OF MATERIAL FOR HI-STORM (DWG. 1495, 1561) SHT 1 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
8	S.GEE 2-1-2000 INCORPORATED ECO 5014-12	[signature] 2/1/00	[signature] 2/1/00	[signature] 2/1/00

	ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
	1	1	SA 516 GR. 70	BASEPLATE	2 THK. X 133 7/8Ø BASEPLATE
	2	1	SA 516 GR. 70	OUTER SHELL	3/4 THK. X 224 1/2 LG. X 132 1/2 O.D. CYLINDER (MAY BE MADE IN SECTIONS, SEE DWG 1495 SHT 5)
	3	1	SA 516 GR. 70	INNER SHELL	1 1/4 THK. X 224 1/2 LG. X 76 O.D. CYLINDER
	4	1	CONCRETE	RADIAL SHIELD	26 3/4 THK. RADIAL SHIELD
	5	1	SA 516 GR. 70	PEDESTAL SHELL	1/4 THK. X 68 3/8 O.D. X 21 5/8 LG. CYLINDER
△8	6	1	SA 516 GR. 70	LID BOTTOM PLATE	1 1/4 THK. X 69" Ø PLATE.
	7	1	SA 516 GR. 70	LID SHELL	1 THK. X 10 1/2 WIDE X 69 O.D.
	8	4	SA 516 GR. 70	EXIT VENT HORIZONTAL PLATE	1 1/4 THK. X 26 WIDE X 30 LG. PLATE (SEE DET. DWG. 1561 SHT. 4)
	9	1	SA 516 GR. 70	TOP PLATE	3/4 THK. X 131 1/2 O.D. X 81 1/2 I.D RING (CUT IN 4 PIECES)
	10	1	SA-516-70	LID TOP PLATE	4 THK. X 126 Ø PLATE (MADE FROM TWO 2" THICK PLATES)
	11	4	SA-516-70	INLET VENT HORIZONTAL PLATE	2 THK. X 16 1/2 WIDE X 30 1/4 LG. PLATE (SEE DET. DWG. 1561 SHT. 3)
	12	8	SA 516 GR. 70	EXIT VENT VERTICAL PLATE	1/2 THK. X 5 1/4 WIDE X 30 APPROX. LG. PLATE
	13	8	SA 516 GR. 70	INLET VENT VERTICAL PLATE	3/4 THK. X 10 WIDE X 29 3/16 APPROX. LG. PLATE
	14	4	SA 516 GR. 70	RADIAL PLATE	3/4 THK. X 27 1/2 WIDE X 224 1/2 LG. PLATE
△8	15	4	SA 193 2H	TOP LID NUT	3 1/4 - 4 UNC HEAVY HEX NUT
△8	16	4	SA 564-630 AGE HARDENED AT 1075°F	LID STUD	3 - 4 UNC X 16 LG. (SEE DWG. 1561, SHT 2)
	17	4	SA 350 LF3 OR SA 203 E	BOLT ANCHOR BLOCK	5 X 5 X 6 ANCHOR BLOCK W/ 3 - 4 UNC X 5 LG HOLE IN CENTER
	18	--	--	DELETED	---
	19	16	SA 516 GR. 70	CHANNEL	3/16 THK. X 6 WIDE X 170 7/8 LG. CHANNEL (SEE DETAIL 1495 SH. 5)
	20	1	SA 516 GR. 70	SHIELD BLOCK RING	1/4 THK. X 64 1/2 I.D. X 85 1/2 O.D. (MAY BE MADE FROM MORE THAN 1 PIECE.)
	21	1	CONCRETE	PEDESTAL SHIELD	17" THK. PLATFORM
	22	1	CONCRETE	LID SHIELD	10 1/2 THK. TOP SHIELD
	23	--	--	DELETED	---
	24	1	SA 516 GR. 70	PEDESTAL PLATFORM	5 THK. X 67 7/8 Ø PLATE (MAY USE MULTIPLE PLATES OF LESSER THICKNESS)
	25	1	CONCRETE	SHIELD BLOCK	8" THK.
	26	1	SA 516 GR. 70	SHIELD BLOCK SHELL	1/2 THK X 86 O.D. CYLINDER X 8" HIGH (MAY MAKE OUT OF MORE THAN 1 PIECE)
	27	1	SA 516 GR. 70	SHIELD BLOCK SHELL	1/2 THK X 64 O.D. CYLINDER X 8" HIGH (MAY MAKE OUT OF MORE THAN 1 PIECE)
*	28	4	SA 516 GR. 70	SHIELD SHELL	3/4 THK. X 58.5 APPROX. WIDE X 205" LG. PLATE
*	29	1	SA 240 304	STORAGE MARKING NAME PLATE	14 GAGE (0.0751 THK.) X 4 WIDE X 10 LG. SHEET
*	30	4	C/S OR S/S	LID PLUGS	1 1/2"-6UNC X 2 1/2" OP BOLT

NOTE:

1) THE CONCRETE MATERIAL IS TO MEET THE REQUIREMENTS SPECIFIED IN APPENDIX 1.D OF THE HI-STORM 100 TSAR DOCKET NUMBER 72-1014 (LATEST REVISION).

2) ALL DIMENSIONS IDENTIFIED ON BM-1575 ARE APPROXIMATE DIMENSIONS EXCEPT THICKNESSES OF STEEL PLATES WHICH IN THE RAW MATERIAL FORM MUST HAVE TOLERANCES MEETING THE APPLICABLE SPECIFICATION.

3) ITEMS WITH A * CONSIDERED NOT TO BE NF CLASS 3 (NON STRUCTURAL)

BM-1575 (E.I.D. 2836) BILL OF MATERIAL FOR HI-STORM (DWG. 1495, 1561) SHT 2 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
8	S. GEE 1-13-2000 INCORPORATED ECO-5014-5	[signature] 1/26/00	[signature] 1/26/00	[signature] 1/26/00

	ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
	31	--	---	DELETED	---
*	32	4	SA 240 304	EXIT VENT SCREEN SHEET	16 GAGE (0.0595 THK.) X 6 1/4 WIDE X 28 LG. SHEET
*	33	4	SA 240 304	EXIT VENT SCREEN FRAME	16 GAGE (0.0595 THK.)
*	34	1	COMMERCIAL	SCREEN	16 WIDE X 212 LG. 6 X 6 MESH 0.020 WIRE Ø 0.147 WIDTH OPEN FROM McMASTER-CARR 101 PAGE# 2521 ITEM# 9220T67 CUT AS NECESSARY OR EQUIVALENT
*	35	4	SA 240 304	INLET VENT SCREEN FRAME	16 GAGE (0.0595 THK.)
△8 *	36	2	COMMERCIAL	THERMOCOUPLE OR RTD	1/8 Ø SHEATH WITH TEMPERATURE ELEMENT (BY USER).
*	37	16	SA240-304	GAMMA SHIELD CROSS PLATE	1/4 THK X 2.75 X 24
*	38	4	SA240-304	GAMMA SHIELD CROSS PLATE	1/4 THK X 24 X 24 5/8
*	39	40	SA240-304	CROSS PLATE TABS	.075 THK X 1/4 X 2 1/2
*	40	8	SA240-304	GAMMA SHIELD CROSS PLATE	1/4 THK X 14 5/8 X 24
*	41	16	SA240-304	GAMMA SHIELD CROSS PLATE	1/4 THK X 3.09 X 24
*	42	2	C/S OR S/S	DRAIN PIPE	1Ø X 1/4THK WALL X 11 1/2 LG
△8 *	43	8	SA240-304	GAMMA SHIELD CROSS PLATE	1/4 THK X 5.09 X 17 1/4
*	44	2	316 SS	COMPRESSION FITTING	1/8" X 1/4 NPT MALE PASS THRU COMPRESSION FITTING (OPTIONAL)
*	45	2	CAST IRON	PROTECTION HEAD	1/2 NPT X 1/2 NPT (OPTIONAL)
*	46	2	304 SS	BUSHING	1/4 X 1/2 NPT (OPTIONAL)
*	47	2	304 SS	COUPLING	1/2 NPT COUPLING W/ MOUNTING STUD 1/2 DIA X 3" LG. (OPTIONAL)
*	48	2	304 SS	HEX NIPPLE	1/2 X 1/2 NPT HEX NIPPLE (OPTIONAL)
*	49	2	304 SS	CONNECTION	1/2 NPT CONDUIT CONNECTION (OPTIONAL)
*	50	3	C/S	SQUARE TUBING	1/2" X 1/2" 16 GAUGE
	51	1	CONCRETE	DUCT SHIELD INSERT	5 3/4 THK X 24 11/16 X 26 LG CONCRETE, ACI 318 (146 LB/CU FT MIN) 21 X 25 GRID OF WWF 4 X 4 -W3.5 X W3.5
	52	1	C/S	EMBEDDED PLATE	1.0 THK X 6 SQ PLATE
	53	2	C/S	NELSON STUD	Ø 1/4 X 2 1/2 LONG

BILL OF MATERIALS HI-STAR 100 SYSTEM FAILED FUEL CANISTER (BM-1819)

REF. DWG. 1783 & 1784

SHEET 1 OF 1

REV. NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	[signature] 12/1/97	[signature] A.S. 12/1/97	[signature] B. GILLIGAN 12/1/97	M. PHIPPS [signature] 12/1/97

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1	1	SA430-WP304	3/4 X 1/2 CONCENTRIC REDUCER SCH. 160	LEAD IN
2	1	SA479-304	ØØ 1 1/2 X 13/16 LG. ROUND BAR	LEAD IN COLLAR
3	1	SA479-304	ØØ 3/8 X 2 LG. ROUND BAR	LOCK PIN
4	1	SA240-304	11 GA. X 1 1/2 X 2 5/8 LG. SHEET	ENGAGEMENT PLATE
5	2	SA479-304	ØØ 3/16 X 13/16 LG. ROUND BAR	ENGAGEMENT PIN
6	4	304 SST.	250 X 250 ØØ.0016 W/.0024 OPENING X ØØ2	WIRE MESH
7	2	SA479-304 (or) SA240-304	3/16 X 1/2 X 5 7/16 LG. BAR	RIM BAR (LONG)
8	2	SA479-304 (or) SA240-304	3/16 X 1/2 X 4 7/16 LG. BAR	RIM BAR (SHORT)
9	2	SA479-304 (or) SA240-304	3/16 X 1 1/4 X 4 7/16 LG. BAR	SIDE (SHORT)
10	2	SA479-304 (or) SA240-304	3/16 X 1 1/4 X 4 13/16 LG. BAR	SIDE (LONG)
11	2	SA479-304 (or) SA240-304	1/2 SQ. X 2 3/16 LG. BAR	LOAD TAB
12	1	SA479-304	ØØ 1 1/2 X 3/4 LG. ROUND BAR	LOAD TAB HUB
13	1	SA479-304	ØØ 1 1/2 X 3 LG. ROUND BAR	LOCKING SHAFT
14	1	SA479-304	ØØ 1/8 X 9/16 LG. ROUND BAR	LOCKING PIN
15	1	SA479-304	ØØ 3/8 X 1 7/8 LG. ROUND BAR	SHEAR PIN
16	1	304 SST. (or) 316 SST.	COMPRESSION SPRING 13/16 I.D. X 1 LG.	LOCKING SPRING
17	1	SA240-304	3/16 X 4 13/16 SQ. PLATE	CLOSURE FRAME BASE PLATE
18	2	304 SST.	1/16 X 3/4 COTTER PIN	COTTER PIN
19	1	SA240-304	3/8 X 4 13/16 SQ. PLATE	FUEL SPACER TOP PLATE
20	1	304 SST.	1/4 WALL X 4 SQ. X 2 1/8 LG. TUBING	FUEL SPACER TUBING
21	1	304 SST.	250 X 250 ØØ.0016 W/.0024 OPENING X ØØ2	WIRE MESH
22	2	SA240-304	11 GA. X 10 3/8 APP. X 138 1/4 LG. SHEET	CANISTER SLEEVE
23	1	SA240-304	11 GA. X 21 3/4 APP. X 2 LG. SHEET	CANISTER COLLAR
24	1	SA240-304	11 GA. X 4 15/16 SQ. SHEET	CANISTER BOTTOM
25	1	SA240-304	11 GA. X ØØ .61 PLUG	LEAD-IN CAP
26	1	SA240-304	11 GA. X 4 1/2 O.D. X 3 1/4 I.D. RING	SCREEN RING
27	4	SA240-304	11 GA. X 1 15/16 O.D. X 1 9/16 I.D. RING	SCREEN RING
28	1	SA312-304	3/4 STD. PIPE SCH. 160 X 1 LG.	LEAD-IN EXTENSION

NOTE: DIMENSIONS ARE APPROXIMATE

BM-1880 BILL OF MATERIAL FOR 125 TON HI-TRAC (DWG. 1880) SHT. 1 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
7	S. GEE 4-30-99 INCORPORATED DCR 5014-131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] 5-4-99

ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
1	1	ASTM B 29	RADIAL LEAD SHIELD	113 CU. FT. COMMON LEAD APPROX.
2	1	SA 516 GR. 70	OUTER SHELL	1 THK. X 81.25 O.D. X 184.75 LG. CYLINDER
3	1	SA 516 GR. 70	INNER SHELL	0.75 THK. X 68.75 I.D. X 184.75 LG. CYLINDER
4	12	SA 516 GR. 70	RADIAL CHANNEL	0.5 THK. X 20 (APPROX) X 168.75 LG.
4A	1	SA 516 GR. 70	RADIAL CHANNEL	0.5 THK. X 20 (APPROX) X 164.625 LG.
4B	1	SA 516 GR. 70	RADIAL CHANNEL	0.5 THK. X 20 (APPROX) X 152.625 LG.
5	10	SA 516 GR. 70	ENCLOSURE SHELL PANEL	0.5 THK. X 11.72 WIDE X 168.75 LG.
5A	4	SA 516 GR. 70	ENCLOSURE SHELL PANELS	0.5 THK. X 11.72 WIDE X 164.625 LG.
6	2	SA 516 GR. 70	WATER JACKET END PLATE	1 THK. X 94.625 O.D. X 81.25 I.D. RING (MAY BE MADE FROM MORE THAN 1 PIECE)
7	1	SA 350 LF3	TOP FLANGE	4.5 THK. X 81.25 O.D. X 68.75 I.D. RING
8	1	SA 516 GR. 70	LOWER WATER JACKET SHELL	0.5 THK. X 86.25 O.D. X 6 LG. CYLINDER
9	1	SA 516 GR. 70 OR SA 350 LF3	BOTTOM FLANGE	2 THK. X 93 O.D. X 68.75 I.D.
10	1	SA 350 LF3	POOL LID OUTER RING	5.5 THK. X 93 O.D. X 75 I.D. RING
11	1	SA 516 GR. 70	POOL LID TOP PLATE	2 THK. X 75 Ø PLATE
12	1	ASTM B 29	POOL LID LEAD SHIELD	6.39 CU. FT. COMMON LEAD APPROX.
13	1	SA 516 GR. 70	TOP LID OUTER RING	0.5 THK. X 75.875 O.D. X 3.75 LG. CYLINDER
14	1	SA 516 GR. 70	TOP LID INNER RING	0.5 THK. X 29 O.D. X 3.75 LG. CYLINDER
15	1	SA 516 GR. 70	TOP LID TOP PLATE	0.5 THK. X 74.875 O.D. X 29 I.D. RING
7 16	1	SA 516 GR. 70	TOP LID BOTTOM PLATE	1.0 THK. X 81.25 O.D. X 27 I.D. RING
17	1	HOLTITE	TOP LID SHIELDING	5.41 CU. FT. APPROX.
18	8	SA 516 GR. 70	FILL PORT CAPS	0.25 THK. X 2.375 Ø PLATE
7 19	18	SA 193 B7	TOP LID STUD	0.5-13 UNC X 2.75 LG. STUDS
7 20	18	SA 193 2H	TOP LID NUT	0.5 - 13 UNC HEAVY HEX
21	1	ELASTOMER	POOL LID GASKET	0.25 THK. X 87.25 O.D. X 86.25 I.D. COMMERCIAL
22	36	SA 193 B7	POOL LID BOLT	1 - 8 UNC X 2.5 LG. HEX. BOLTS

NOTE: 1) ALL SA-350-LF3 MATERIAL MAY BE REPLACED BY SA-203-E.

2) ALL DIMENSIONS ARE FOR REFERENCE ONLY.

BM-1880 BILL OF MATERIAL FOR 125 TON HI-TRAC (DWG. 1880) SHT. 2 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
5	S. GEE 4-30-99 INCORPORATED DCR 5014-131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] 5-4-99

ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
△ 23		---	DELETED	---
△ 24	2	SA 350 LF3	LIFTING TRUNNION BLOCK	7.75 (APPROX) X 8 X 10
25	2	SA 516 GR. 70	LIFTING TRUNNION LOCKING PAD	0.25 THK. X 4.5 X 9.25 LG. PLATE
26	2	SB 637 N07718	LIFTING TRUNNION	6.25 Ø X 7.75 LG. BAR
27	2	SA 516 GR. 70	LIFTING TRUNNION END CAP	0.5 THK. X 6.75 Ø PLATE
28	4	SA 193 B7	END CAP BOLTS	0.5 - 13 UNC X 1.5 LG. WITH 1.25 LG. THREAD
29	2	SA 350 LF3	POCKET TRUNNION	12.375 X 13 X 12.5 BLOCK
30	1	SA 106	DRAIN PIPE	1 SCH. 80 X 7 (APPROX.) LG. PIPE
31		--	DELETED	
32	1	SA 193 B7	DRAIN BOLT	1 - 8UNC X 1.75 LG. SOCKET CAP BOLT
33	4	SA 516 GR. 70	TOP LID TONGUE	0.5 THK. X 0.5 WIDE X 10 (APPROX.) LG. BAR
34	2	SA 516 GR. 70	WATER JACKET END PLATE	1 THK. X 6.6875 WIDE X 30 (APPROX) LG. PLATE
35	4	SA 193 B7	LIFTING TRUNNION PAD BOLT	0.25 - 20 UNC X 1.25 LG. BOLT
36	1	SA 516 GR. 70	POOL LID BOTTOM PLATE	1 THK. X 75 Ø PLATE
37	1	COMMERCIAL	VENT COUPLING	1 1/2-3000 lb. SCREWED HALF COUPLING
38	1	COMMERCIAL	VENT PLUG	1 1/2-3000 lb. SCREWED HEXAGON HEAD PLUG
39	1	COMMERCIAL	PRESSURE RELIEF COUPLING	1-3000 lb. SCREWED HALF COUPLING
40	1	COMMERCIAL	PRESSURE RELIEF VALVE	MEDIUM PRESSURE BRONZE POP VALVE
41	1	SA 106	JACKET DRAIN PIPE	1 1/2 SCH. 40 X 5 LG. PIPE
42	1	COMMERCIAL	JACKET DRAIN VALVE	1 1/2 NONRISING STEM BRONZE GATE VALVE
43	4	C/S OR S/S	HOLE PLUGS	N/A
△ 44	4	SA 516 GR. 70	TOP LID LIFTING BLOCK	1.5 SQ. X 3.25 LG. BLOCK
45		COMMERCIAL	THERMAL EXPANSION FOAM	0.125 THK.
△ 46	4	SA 516 GR 70	GUSSET	0.75 THK X 1.0 WIDE X 3.75 LG PLATE

BM-1928 BILL OF MATERIAL FOR 125 TON HI-TRAC TRANSFER LID (DWG. 1928)

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
7	S.GEE, 5-3-99 INCORPORATED DCR 5014-131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] 5-4-99

ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
1	1	SA 516 GR. 70	LID TOP PLATE	1.5 THK. X 93 WIDE X 128 LG. PLATE
2	1	SA 516 GR. 70	LID BOTTOM PLATE	2 THK. X 93 WIDE X 128 LG. PLATE
3	2	SA 516 GR. 70	LID INTERMEDIATE PLATE	1.5 THK. X 8.375 WIDE X 132 LG. PLATE
4	2	SA 516 GR. 70	LEAD COVER PLATE	1 THK. X 8.375 WIDE X 78 LG. PLATE
5	8	SA 516 GR. 70	LEAD COVER SIDE PLATE	1 THK. X 4.5 WIDE X 8.375 LG. PLATE
6	1	ASTM B 29	SIDE LEAD SHIELD	2.65 (APPROX.) CU. FT.
7	2	SA 36	WHEEL TRACK	0.125 THK. X 0.75 X 0.75 X 128 LG. ANGLE
8	2	SA 516 GR. 70	DOOR TOP PLATE	2 1/4 THK. X 47 WIDE X 65 LG. PLATE (CUT AS NECESSARY)
9	2	ASTM B 29	DOOR LEAD SHIELD	2.9 (APPROX.) CU. FT.
10	2	SA 516 GR. 70	DOOR MIDDLE PLATE	1/2 THK. X 47 WIDE X 65 LG. PLATE (CUT AS NECESSARY)
11	2	HOLTITE	DOOR SHIELDING	3.65 (APPROX.) CU. FT.
12	2	SA 516 GR. 70	DOOR BOTTOM PLATE	3/4 THK. X 47 WIDE X 65 LG. PLATE (CUT AS NECESSARY)
13	4	SA 350 LF3, SA 203E OR SA 516-70	DOOR WHEEL HOUSING	7.5 THK. X 8 WIDE X 25 LG. PLATE (CUT 3.75 X 6)
14	2	SA 516 GR. 70	DOOR INTERFACE PLATE	1 THK. X 3 7/8 WIDE X 80 LG. PLATE
15	2	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 8 WIDE X 65 LG. PLATE
15A	4	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 6 WIDE X 65 LG. PLATE
16	4	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 8 WIDE X 32.625 APPROX. LG. PLATE
17	2	C/S	DOOR HANDLE	0.5 Ø X 17.5 LG. BAR
18	12	COMMERCIAL	DOOR WHEEL	6 X 3 V-GROOVE WHEEL FROM McMASTER-CARR 101 PAGE# 481 ITEM# 2310T19 OR EQUIVALENT
19	12	SA 36	WHEEL SHAFT	1.25 Ø X 6.875 LG. BAR
20	12	SA 516 GR. 70	SHAFT COVER PLATE	0.25 THK. X 2.5 Ø PLATE
21	2	SA 516 GR. 70	LID HOUSING STIFFENER	1 THK. X 3.5 WIDE X 8.375 LG. PLATE
22	4	SA 193 B7	DOOR LOCK BOLT	2 7/8 - 4 UNC X 10.875" LG. HEX. BOLTS W/ 1.5 LG. THREADED AT END
23	4	SA 516 GR. 70	DOOR STOP BLOCK	2 THK. X 2 WIDE X 8 LG. BLOCK
24	8	SA 193 B7	DOOR STOP BLOCK BOLT	1 - 8 UNC X 3 LG. BOLT W/ 2.5 LG. THREADED AT END
25	2	SA 516 GR. 70	DOOR END PLATE	1 THK. X 8 WIDE X 19 LG. PLATE
26	4	SA 516 GR. 70	LIFTING LUG	0.75 THK. X 3 WIDE X 3.5 LG. PLATE
27	4	SA 516 GR. 70	LIFTING LUG PAD	0.5 THK. X 5 SQ. PLATE

NOTE:

1) ALL DIMENSIONS ARE APPROXIMATE.

BM-2145 BILL OF MATERIAL FOR 100 TON HI-TRAC (DWG. 2145) SHT. 1 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
4	S. GEE 4-30-99 INCORPORATED DCR 5014-131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] 5-4-99

	ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
	1	1	ASTM B 29	RADIAL LEAD SHIELD	71.15 CU. FT. COMMON LEAD APPROX.
	2	1	SA 516 GR. 70	OUTER SHELL	1 THK. X 78 O.D. X 184.75 LG. CYLINDER
	3	1	SA 516 GR. 70	INNER SHELL	0.75 THK. X 68.75 I.D. X 184.75 LG. CYLINDER
	4	13	SA 516 GR. 70	RADIAL CHANNEL	.375 THK. X 18.8 (APPROX) X 168.75 LG.
△	4A	2	SA 516 GR. 70	RADIAL CHANNEL	0.375 THK. X 18.8 (APPROX) X 164.625 LG.
△	4B	---	-	DELETED	
	5	11	SA 516 GR. 70	ENCLOSURE SHELL PANELS	0.375 THK. X 9.7125 WIDE X 168.75 LG.
	5A	4	SA 516 GR. 70	ENCLOSURE SHELL PANELS	0.375 THK. X 9.7125 WIDE X 164.625 LG.
△	6	2	SA 516 GR. 70	WATER JACKET END PLATE	1 THK. X 91 O.D. X 78 I.D. RING (MAY BE MADE FROM MORE THAN 1 PIECE)
	7	1	SA 350 LF3	TOP FLANGE	4.5 THK. X 78.00 O.D. X 68.75 I.D. RING
△	8	1	SA 516 GR. 70	LOWER WATER JACKET SHELL	1.25 THK. X 83.00 O.D. X 6 LG. CYLINDER
	9	1	SA 350 LF3, OR SA 516 GR. 70	BOTTOM FLANGE	2 THK. X 89 O.D. X 68.75 I.D.
△	10	1	SA350 LF3	POOL LID OUTER RING	4.0 THK X 89 O.D. X 75 I.D. RING
△	11	1	SA 203-E	POOL LID TOP PLATE	2 THK. X 75 Ø PLATE
△	12	1	ASTM B 29	POOL LID LEAD SHIELD	3.84 CU FT APPROX. COMMON LEAD
△	13	--	---	DELETED	---
△	14		---	DELETED	
△	15	--	---	DELETED	---
△	16	1	SA 516 GR. 70	TOP LID BOTTOM PLATE	1.0 THK. X 78.00 O.D. X 27 I.D. RING
△	17	1	SA 516 GR 70	POOL LID BOTTOM PLATE	.5 THK X 75 Ø PLATE
	18	8	SA 516 GR. 70	FILL PORT CAPS	0.25 THK. X 2.375 Ø PLATE
	19	12	SA 193 B7	TOP LID STUD	0.5-13 UNC X 2.75 LG. STUDS
	20	12	SA 193 2H	TOP LID NUT	0.5 - 13 UNC HEAVY HEX
	21	1	ELASTOMER	POOL LID GASKET	0.25 THK. X 83.625 O.D. X 82.625 I.D. COMMERCIAL
	22	36	SA 193 B7	POOL LID BOLT	1 - 8 UNC X 2.5 LG. HEX. BOLTS
△	23	--	---	DELETED	---
	24	2	SA 350 LF3	LIFTING TRUNNION BLOCK	6 (APPROX) X 8 X 10
	25	2	SA 516 GR. 70	LIFTING TRUNNION LOCKING PAD	0.25 THK X 4.5 X 9.25 LG PLATE

NOTES: 1. ALL SA-350-LF3 MATERIAL MAY BE REPLACED BY SA-203-E.

2. ALL DIMENSIONS ARE FOR REFERENCE ONLY.

BM-2145 BILL OF MATERIAL FOR 100 TON HI-TRAC (DWG. 2145) SHT. 2 OF 2

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
3	S. GEE 4-30-99 INCORPORATED DCR 5014-131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] 5-4-99

	ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
	26	2	SB 637 N07718	LIFTING TRUNNION	6.25 Ø X 7.75 LG. BAR
	27	2	SA 516 GR. 70	LIFTING TRUNNION END CAP	0.5 THK. X 6.75 Ø PLATE
	28	4	SA 193 B7	END CAP BOLTS	0.5 13 UNC X 1.5 LG. WITH 1.25 LG. THREAD
	29	2	SA 350 LF3	REMOVABLE POCKET TRUNNION	3.9375 X 13 X 12.375 BLOCK
	30	6	SA 564-630 (H1100)	DOWEL PINS	1 3/8" Ø BAR
△	31	1	SA 106	DRAIN PIPE	1 SCH 80 X 6 LG APPROX (CUT TO SUIT)
	32	1	SA 193 B7	DRAIN BOLT	1 - 8UNC X 1.75 LG. SOCKET CAP BOLT
	33	4	SA 516 GR. 70	TOP LID TONGUE	0.5 THK. X 0.5 WIDE X 10 (APPROX.) LG. BAR
	34	2	SA 516 GR. 70	WATER JACKET END PLATE	1 THK. X 6.375 WIDE X 30 (APPROX) LG. PLATE
	35	4	SA 193 B7	LIFTING TRUNNION PAD BOLT	0.25 20 UNC X 1.25 LG. BOLT
	36	--	---	---	---
	37	1	COMMERCIAL	VENT COUPLING	1 1/2-3000 lb. SCREWED HALF COUPLING
	38	1	COMMERCIAL	VENT PLUG	1 1/2-3000 lb. SCREWED HEXAGON HEAD PLUG
	39	1	COMMERCIAL	PRESSURE RELIEF COUPLING	1-3000 lb. SCREWED HALF COUPLING
	40	1	COMMERCIAL	PRESSURE RELIEF VALVE	MEDIUM PRESSURE BRONZE POP VALVE
	41	1	SA 106	JACKET DRAIN PIPE	1 1/2 SCH. 40 X 5 LG. PIPE
	42	1	COMMERCIAL	JACKET DRAIN VALVE	1 1/2 NONRISING STEM BRONZE GATE VALVE
	43	4	C/S OR S/S	HOLE PLUGS	N/A
△	44	--	---	DELETED	---
△	45	--	---	DELETED	---
△	46	2	SA 516 GR. 70	SUPPORT GUSSETS	.375 X 1 X 2" LG
	47	2	SA 350 LF3	POCKET TRUNNION BASE	8.03 X 13 X 12.375
△	48	4	SA564-630 (H1100)	POCKET TRUNNION BOLTS	1-8 UNC X 6.25 WITH 2.3125" MIN LG THREAD
△	49	4	SA 516 GR 70	GUSSET	.75 THK X 1.0 WIDE X 3.75 LG PLATE

BM-2152 BILL OF MATERIAL FOR 100 TON HI-TRAC TRANSFER LID (DWG. 2152)

REV. NO.	PREP. BY	CHECKED BY	PROJ. MANAGER	QA. MANAGER
5	S. GEE, 4-30-99 INCORPORATED DCR 5014 131	[signature] 5/4/99	[signature] B. GILLIGAN 5/4/99	[signature] J.G. 5-4-99

	ITEM	QTY.	SPECIFICATION	NOMENCLATURE	DESCRIPTION
	1	1	SA 516 GR. 70	LID TOP PLATE	1.5 THK. X 89 WIDE X 128 LG. PLATE
△5	2	1	SA 516 GR. 70	LID BOTTOM PLATE	1 1/2 THK. X 89 WIDE X 128 LG. PLATE
△5	3	2	SA 516 GR. 70	LID INTERMEDIATE PLATE	1.5 THK. X 8.375 WIDE X 132 LG. PLATE
	4	2	SA 516 GR. 70	LEAD COVER PLATE	1 THK. X 8.375 WIDE X 78 LG. PLATE
△5	5	8	SA 516 GR. 70	LEAD COVER SIDE PLATE	1 THK. X 2.5 WIDE X 8.375 LG. PLATE
	6	1	ASTM B 29	SIDE LEAD SHIELD	1.136 APPROX. CU. FT.
	7	2	SA 36	WHEEL TRACK	0.125 THK. X 0.75 X 0.75 X 128 LG. ANGLE
△5	8	2	SA 516 GR. 70	DOOR TOP PLATE	2.25 THK. X 44.5 WIDE X 65 LG. PLATE (CUT AS NECESSARY)
△5	9	2	ASTM B 29	DOOR LEAD SHIELD	2.04 APPROX CU. FT.
	10		DELETED	- - -	- - -
△5	11	- -	DELETED	- - -	
△5	12	2	SA 516 GR. 70	DOOR BOTTOM PLATE	1/2 THK. X 44.5 WIDE X 65 LG. PLATE (CUT AS NECESSARY)
	13	4	SA 516 GR 70, OR SA 350 LF3, OR SA 203 E	DOOR WHEEL HOUSING	7.5 THK. X 8 WIDE X 25 LG. PLATE (CUT 3.75 X 6)
	14	2	SA 516 GR. 70	DOOR INTERFACE PLATE	1 THK. X 3 7/8 WIDE X 80 LG. PLATE
	15	2	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 8 WIDE X 65 LG. PLATE
△5	15A	4	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 2 WIDE X 65 LG. PLATE
△5	16	4	SA 516 GR. 70	DOOR SIDE PLATE	1 THK. X 4.25 WIDE X 29 APPROX. LG. PLATE
	17	2	C/S	DOOR HANDLE	0.5 Ø X 17.5 LG. BAR
	18	12	COMMERCIAL	DOOR WHEEL	6 X 3 V-GROOVE WHEEL FROM McMASTER-CARR 101 PAGE# 481 ITEM# 2310T19 OR EQUIVALENT
	19	12	SA 36	WHEEL SHAFT	1.25 Ø X 6.875 LG. BAR
	20	12	SA 516 GR. 70	SHAFT COVER PLATE	0.25 THK. X 2.5 Ø PLATE
	21	2	SA 516 GR. 70	LID HOUSING STIFFENER	1 THK. X 1.5 WIDE X 8.375 LG. PLATE
	22	4	SA 193 B7	DOOR LOCK BOLT	2 7/8 - 4 UNC X 10.875 LG. HEX. BOLTS W/ 1.5 LG. THREADED AT END
	23	4	SA 516 GR. 70	DOOR STOP BLOCK	2 THK. X 2 WIDE X 8 LG. BLOCK
	24	8	SA 193 B7	DOOR STOP BLOCK BOLT	1 8 UNC X 3 LG. BOLT W/ 2.5 LG. THREADED AT END
△5	25	2	SA 516 GR. 70	DOOR END PLATE	1 THK. X 6 WIDE X 24 LG. PLATE
	26	4	SA 516 GR. 70	LIFTING LUG	0.75 THK. X 3 WIDE X 3.5 LG. PLATE
	27	4	SA 516 GR. 70	LIFTING LUG PAD	0.5 THK. X 5 SQ. PLATE

NOTES:

1) ALL DIMENSIONS ARE APPROXIMATE.

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 Concrete", ACI 318-95, ACI, Detroit, Michigan.

[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, Topical Safety Analysis Report for the (Holtec International Storage, Transport, and Repository) HI-STAR System, Rev. 9, 1998.

[1.1.3] USNRC Docket No. 71-9261, Safety Analysis Report for Packaging for the (Holtec International Storage, Transport, and Repository) HI-STAR System, Rev. 7, 1998.

[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".

[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] American Concrete Institute, "Building Code Requirements for Structural Concrete", ACI 318-95, ACI, Detroit, Michigan.

[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 N14.6-1978, "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, February, 1978.

[1.2.10] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.

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 (á)
- 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 1.2.3. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STAR 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 References

[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	75.0	75.0	75.0	75.0
100	75.0	75.0	75.0	75.0	75.0
200	71.0	71.0	75.0	75.0	71.0
300	66.0	66.0	73.4	70.9	66.0
400	64.4	64.4	71.8	67.1	64.4
500	63.5	63.5	71.8	64.6	63.5
600	63.5	63.5	71.8	63.1	63.1
650	63.5	63.5	71.8	62.8	62.8
700	63.5	63.5	71.8	62.5	62.5
750	63.1	63.1	71.4	62.2	62.2
800	62.7	62.7	70.9	61.7	61.7

Notes:

1. Source: Table U on pages 437, 439, 441, and 443 of [1.A.1].
2. Units of tensile strength are ksi.

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

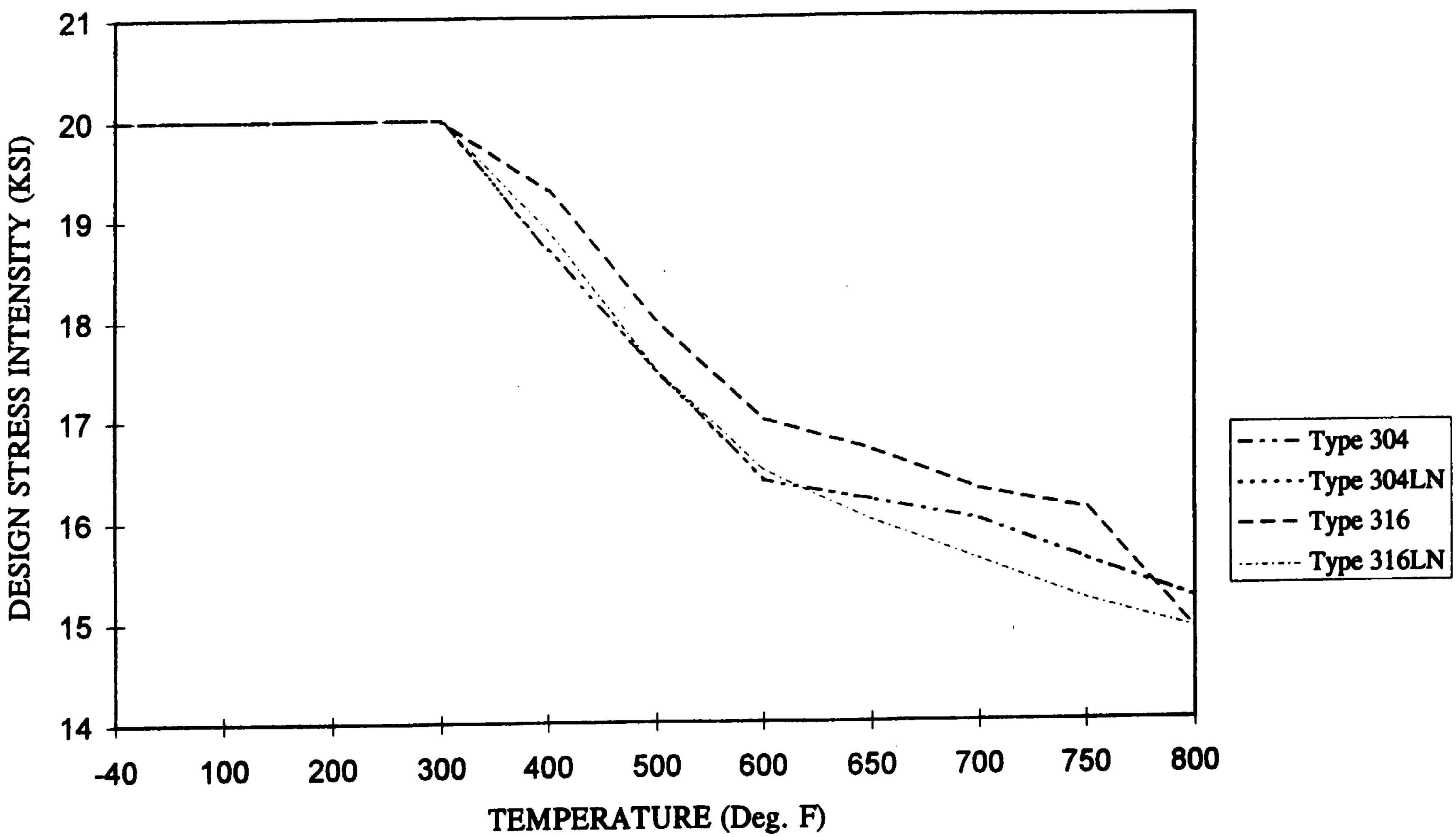
Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X (minimum of constituent values)
-40	8.23	6.96	6.96
70	8.6	7.7	7.7
100	8.7	7.9	7.9
150	9.0	8.2	8.2
200	9.3	8.4	8.4
250	9.6	8.7	8.7
300	9.8	9.0	9.0
350	10.1	9.2	9.2
400	10.4	9.5	9.5
450	10.6	9.8	9.8
500	10.9	10.0	10.0
550	11.1	10.3	10.3
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.

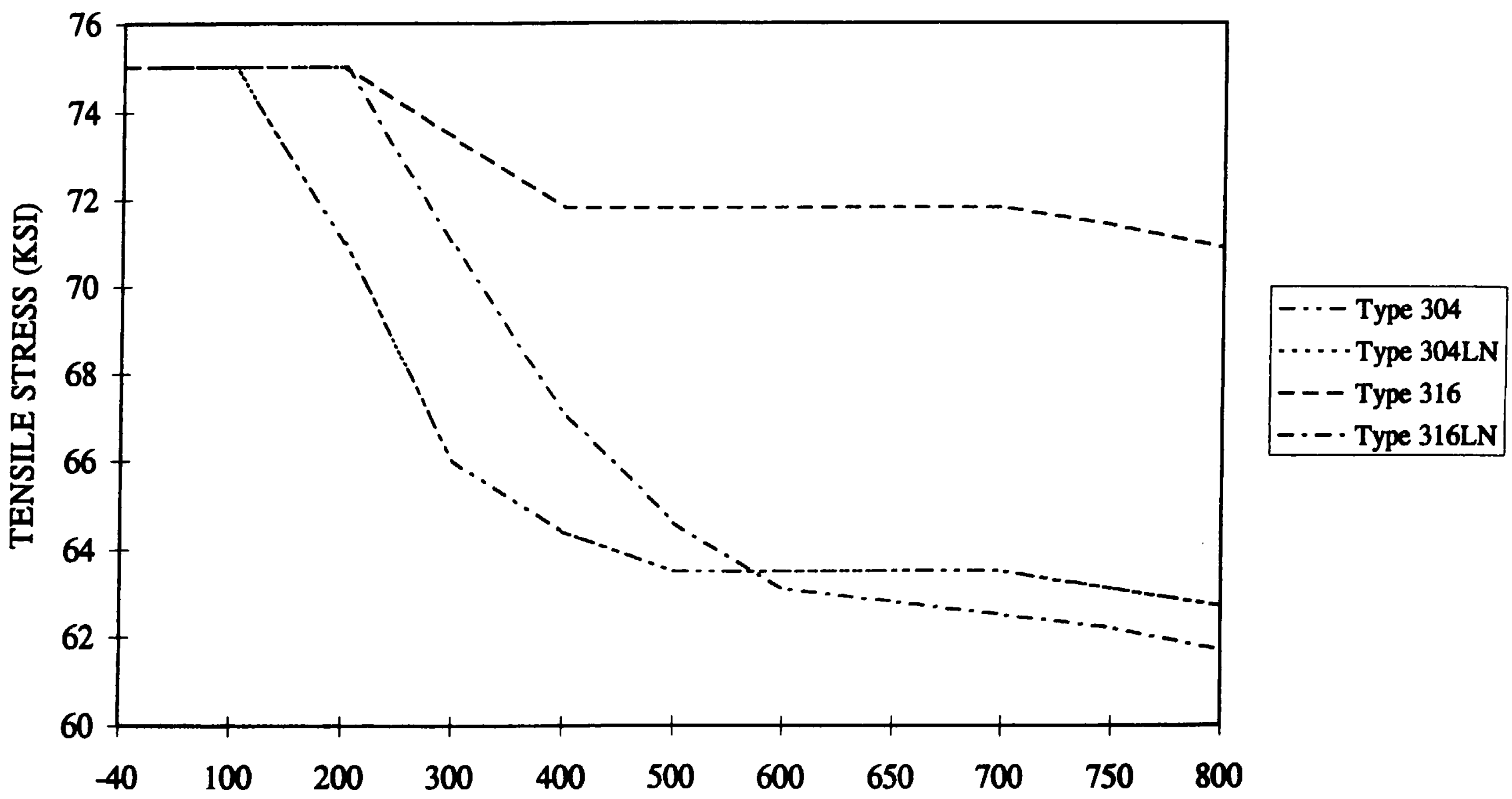
DESIGN STRESS INTENSITY VS. TEMPERATURE



SOURCE: TABLE 1.A.1

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

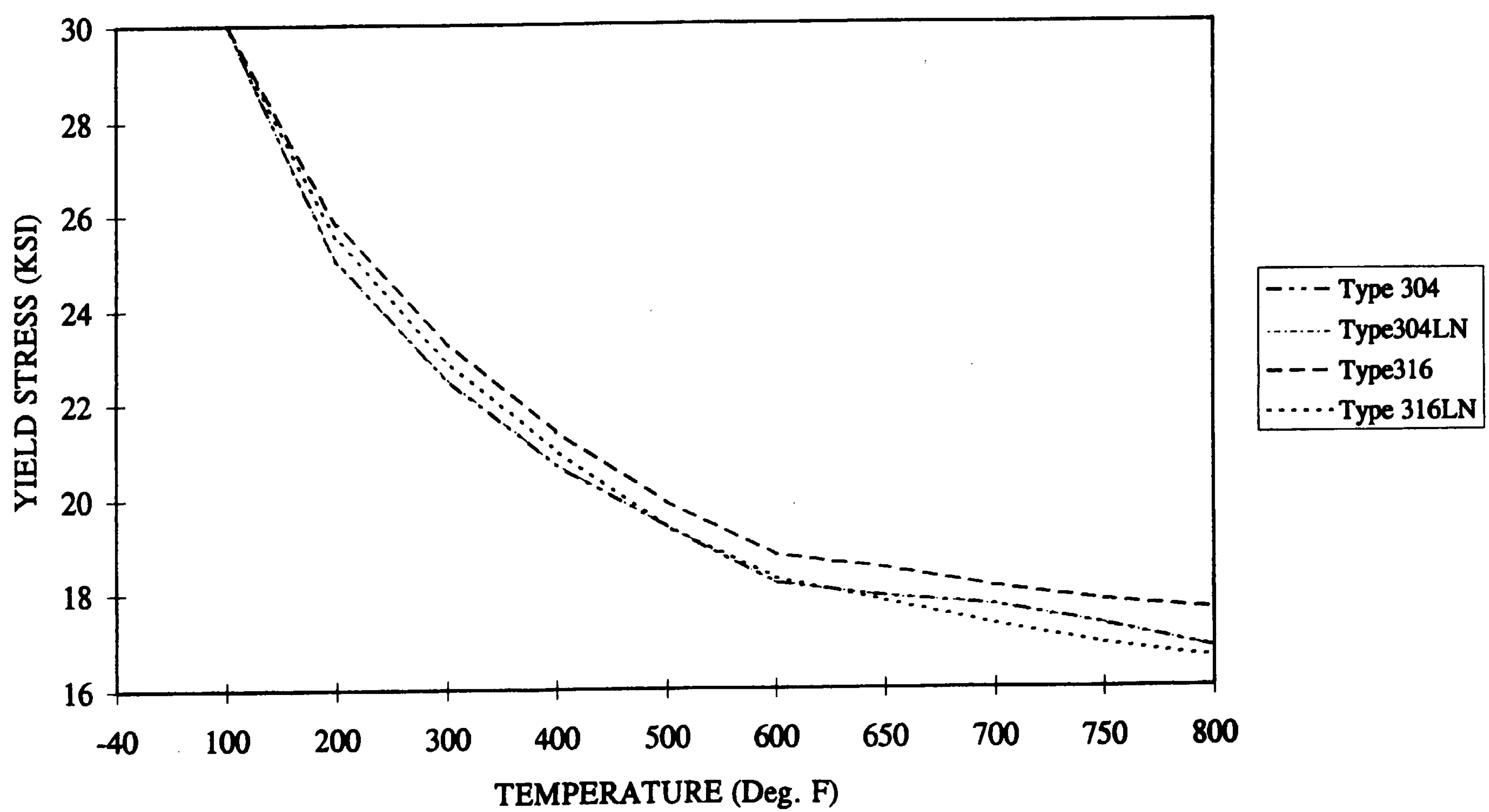
TENSILE STRENGTH VS. TEMPERATURE



SOURCE: TABLE 1.A.2

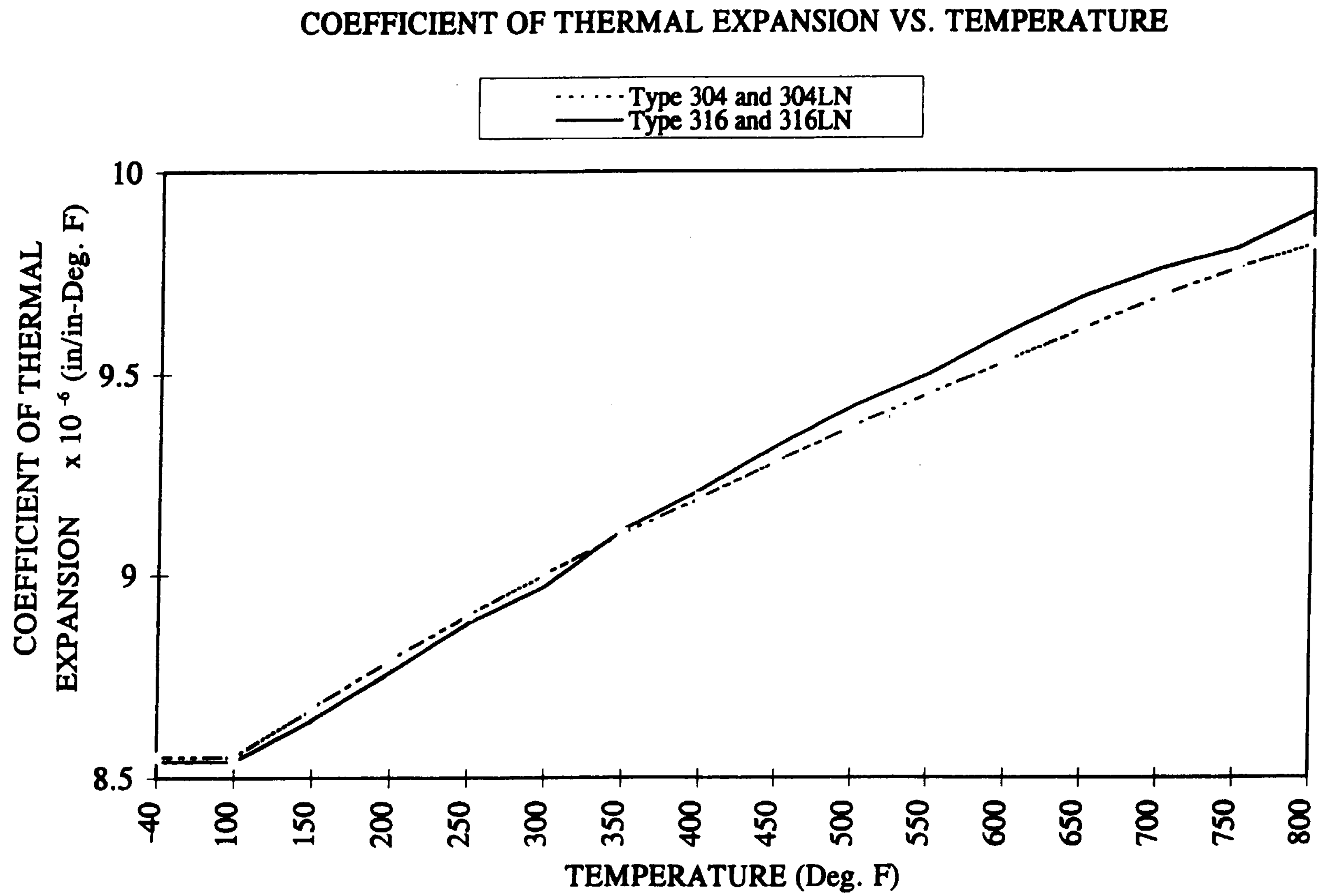
FIGURE 1.A.2; TENSILE STRENGTH VS. TEMPERATURE

YIELD STRESS VS. TEMPERATURE

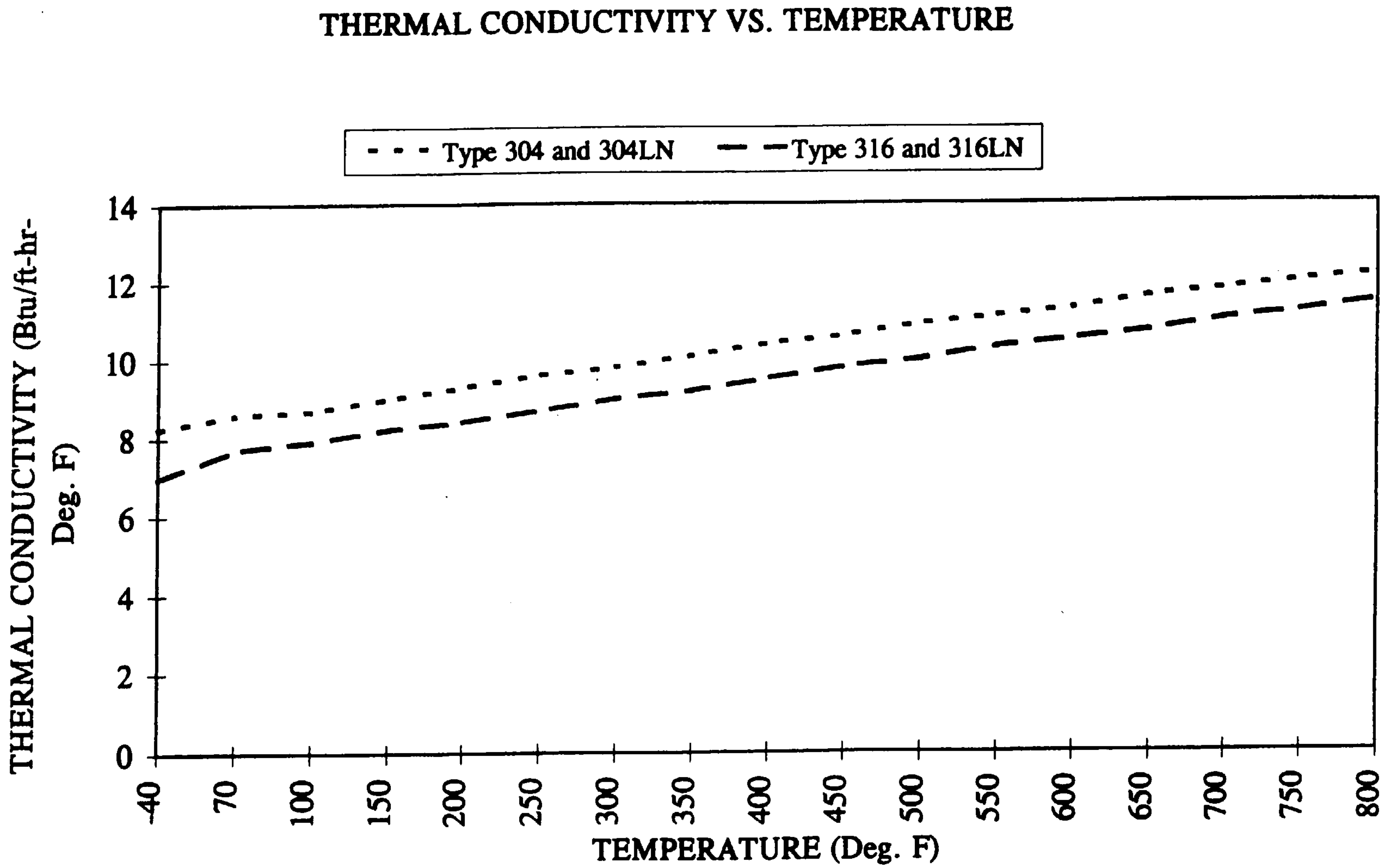


SOURCE: TABLE 1.A.3

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



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



SOURCE: TABLE 1.A.5

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

APPENDIX 1.B: HOLTITE ™ MATERIAL DATA
(Total of 20 Pages Including This Page)

The information provided in this appendix describes the neutron absorber material, Holtite-A (also known commercially as NS-4-FR) for the purpose of confirming its suitability for use as a neutron shield material in spent fuel storage casks.

NS-4-FR 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, hence the "FR" designation in NS-4-FR. NS-4-FR is a generic name for the material which was originally developed by Bisco Inc. and used for many years as a shield material with B_4C or Pb added. NS-4-FR contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A, the Holtec International version of NS-4-FR, contains 1% 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 NS-4-FR mixture. B_4C is added (1% in Holtite-A) 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. Analysis for B_4C throughout a column of Holtite-A has confirmed (Holtec International qualification tests) that the B_4C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B_4C is specified in Holtite-A as a precaution to assure that the B_4C concentration is always adequate throughout the mixture.

NS-4-FR material has been extensively tested for thermal stability, as indicated in the following documents (copies of these documents are attached).

- Letter dated 4/20/87 from Mr. Larry Dietrick, Bisco Products to Mr. Tod Lesser, NAC International, "Weight loss of NS-4-FR under extreme temperature conditions"

- "Experimental Studies On Long-Term Thermal Degradation of Enclosed Neutron Shielding Resin", Asano, Ryoji and Nagao Niomura

- "Thermal Testing of Solid Neutron Shielding Materials", Boonstra, Richard H.

The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity specified in Chapter 1 includes a 4% reduction to conservatively account for potential weight loss at the design temperature of 300°F. However, the BISCO letter dated 4/20/87 provides information stating that samples had been exposed to a continuous temperature of 338°F for 146 days and a maximum of 3.15% weight loss had been experienced. Thus, there is a substantial level of conservatism in the Holtec allowance for weight loss.

Tests on the stability of Holtite-A were also performed by Holtec International. Results of these independent tests at 325°F on 15 samples gave an average weight loss in 97 days of 2.41% and a maximum weight loss of 2.72%. The observations are consistent with published tests on NS-4-FR.

The paper entitled "Experimental Studies on Long-Term Thermal Degradation of Enclosed Neutron Shielding Resin " provides information which corroborates the information provided in the BISCO letter dated 4/20/87. The paper suggests that enclosures of the NS-4-FR material can further decrease the percent weight reduction at elevated temperatures. The NS-4-FR is encapsulated in the HI-STAR 100 overpack and therefore should experience a very small weight reduction during the design life of the HI-STAR 100 System. It should be noted that the shielding analysis conservatively assumes 4% loss in density.

The paper entitled "Thermal Testing of Solid Neutron Shielding Materials" provides information regarding NS-4-FR material stability during a fire accident. Results of the study suggests that NS-4-FR could withstand a fire accident with minimal damage. This data is provided for information only, as the post-accident shielding analysis very conservatively assumes complete degradation of the neutron shield and replaces the neutron shield with a void.

The data and test results presented here confirm that

1. Holtite-A with 1% B_4C has the same thermal stability and characteristics as the previously approved NS-4-FR material,

2. The hydrogen density meets or exceeds minimum NS-4-FR specifications (measured at 0.105 gH_2/cc compared to the NS-4-FR specification of 0.096 gH_2/cc), and

3. The B_4C is 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 in the manner of NS-4-FR, which was licensed previously in Docket No. 71-9235 (NAC-STC).

Table 1.B.1

PROPERTIES OF HOLTITE-A NEUTRON SHIELD

PHYSICAL PROPERTIES (Reference: NAC International Brochure)	
% ATH	62 maximum (confirmed by Holtec in independent analyses)
Specific Gravity	1.68 g/cc maximum
Thermal Conductivity	0.373 Btu/hr/ft-°F
Max. Continuous Operating Temperature	300°F
Specific Heat[†]	0.39 Btu/hr/ft-°F
Hydrogen Density	0.096 g/cc minimum (confirmed by Holtec in independent analyses)
Radiation Resistance	Excellent
Ultimate Tensile Strength	4,250 psi
Tensile elongation	0.65%
Ultimate Compression Strength	10,500 psi
Compression Yield Strength	8,780 psi
Compression Modulus	561,000 psi
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5 (confirmed by Holtec)
wt% Hydrogen	6.0 (confirmed by Holtec)
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	up to 6.5 (Holtite-A uses 1% B_4C)

H BISCO Products Data from Docket M-55, NAC-STC TSAR.

Experimental Studies on Long-term Thermal Degradation of Enclosed Neutron Shielding Resin

Ryoji ASANO[1], *Nagao NIOMURA*[2]

[1]Hitachi Zosen Corporation, Japan
[2]Ocean Cask Lease Co.,Ltd, Japan

INTRODUCTION

Resins which have high Hydrogen atom content are effective for Neutron shielding and are recently used for neutron shielding material of spent fuel shipping casks. As the resins themselves are easily burned at relatively low temperature, which could be the problem during the fire test condition, mixture of resin and fire retardant which main component is a hydroxide compound is usually used as shielding material. The fire retardant prevents resin from burning by decomposing of the hydroxide compound under fire test condition.

When these resins are used for neutron shielding material of cask, their temperature rises during the transportation by decay heat of spent fuel. Therefore, thermal degradation of resin (hereafter called as "heat weight loss") at the operating temperature should be paid attention.

Furthermore, when the resin is used for neutron shielding material, there are two cases. One is to put it on the outside surface of the cask and the other is to enclose it between two layers. In former case, the heat weight loss occurs in air of which study report can be obtained. On the other hand, the latter is the reaction in the enclosed environment which study report can be seldom obtained. Therefore, the study of the heat weight loss in the enclosed environment was carried out for long term period assuming the operating time of the real cask .

TEST MATERIAL

Test material is NS-4-FR supplied by BISCO CO. LTD, U.S.A. Raw materials are epoxi resin, hardener and fire retardant. They are mixed together and hardened according to the manufacturing manual supplied by BISCO. NS-4-FR is the neutron shielding material which contains about 60% of aluminium hydroxide as fire retardant.

TEST

Tests were carried out in order of basic material test, open test, enclosed test and long term cyclic test which simulates the operation term of cask. The test results are explained as follows.

Basic material test
TG tests which can be performed comparatively easily were carried out in order to study basic thermal characteristics of the test material. The test conditions are as follows.

Condition	Case 1	Case 2
Atmospheric gas	Air&N_2	Air&N_2
Gas Flow Rate(cc/min)	150	200
Temp. Rising Rate(°C/min)	3	10
Max.Temperature(°C)	220	530

Heat weight loss could not be detected in the Case 1. The results of Case 2 are as follows.

(1)The weight loss of the test specimen in nitrogen gas was much smaller than that in air between 300°C and 380°C which were shown in Fig 1. It indicates that the test materials are decomposed and loose its weight by oxygen in air and by heat within the temperature range.

(2)Comparing the results between test material and NS-4- FR without fire retardant, the weight loss of latter is less than that of former until 360°C as shown in Fig. 2. It indicates that the weight loss of former is mainly due to the decomposition of aluminium hydroxide as fire retardant. This result means that the decomposition of aluminium hydroxide is important for the weight loss during low temperature. And it is necessary to select a suitable grade of aluminium hydroxide, as the decomposition temperature depends on the purity and grain size of aluminium hydroxide.

TG test results can not be used directly for long term degradation data because the test specimen was pulverized to very small size, and reaction and diffusion is very rapid, but they can be good reference information.

Open test

Open tests were performed varying the shape of test specimen and temperature to study heat weight loss in air. The results are as follows.

(1) Effect of shape of test specimen
To study the effect of the shape, cubic and cylindrical test specimens with nearly equal weight were tested. It was observed that the heat weight loss of latter which had larger surface area was always larger than that of the former. It indicates that effects of surface oxidation and surface diffusion are important factors for the heat weight loss.

(2) Effect of temperature
The heat weight loss at 125°C, 150°C, 175°C and 200°C are shown in Fig.3 as a function of time. The increase of the heat weight loss is observed in 200°C test after 1000 hr. It is supposed that generation of continues crack inside of the test specimen makes it easy to diffuse the decomposed resin component and water.

Enclosed test

Supposing that the neutron shielding material is filled in the enclosures tests were conducted to study the effects of enclosed condition to the heat weight loss. The tests were performed on test specimen in the sealed stainless steel container with Ar atmosphere.

(1) Sealed stainless steel container
Seal container is shown in Fig. 4, of which lid is welded to seal the cavity of the container perfectly. Enough height of container cavity is provided to avoid the effect of welding heat to the test specimen. Ar gas seal hole is seal-welded and cooled immediately by water after replacing air with Ar gas.

(2) Test Condition
Continuous test and cyclic test of 110 hr heating and 58 hr cooling which simulated the actual operating condition of cask were performed at 125°C, 150°C and 175°C. The test duration was from 8 to 16 weeks.

(3) Test Results
Test results are shown also in Fig.3. Main results are as follows.

(a) Test Results at 125°C
The heat weight loss at 1512 hr continuous test and that at 1760 hr cyclic test were negligible. The heat weight loss at this temperature is regarded as insignificant.

(b) Test Results at 150°C
The heat weight losses at both 1224 hr continuous tests, 990 hr cyclic test and 1760 hr cyclic test were almost 1/3 of that of open test.

(c) Test Results at 175°C
The heat weight losses at both 1600 hr continues test, 1210 hr cyclic test and 1760 hr cyclic test were almost half of that of open test.

(d) Due to the few test specimens and short test period, data scattering was observed in the test results. However, heat weight loss of enclosed test is clearly less than that of open test except those at 125°C when no heat weight loss was observed.

Long term cyclic test at 150°C
In order to avoid scatter in test results and to estimate heat weight loss during long term use of cask, long term cyclic tests at 150°C were conducted, where temperature supposed was the maximum working temperature of neutron shielding material during transportation. The results are shown in Fig. 5. Total test specimens were 18 and maximum test period was 56 weeks. One cycle is composed of 110 hr heating and 58 hr cooling which is same as the enclosed test above. The heat weight loss for 56 weeks was about 1.1%.

DISCUSSION

The relation between heat weight loss W(%) and test period D(day) is given from Fig. 5, as follow.

$$W=100.63-0.218\times\log D$$

Using this equation the heat weight loss for 20 years W_{20}(%) can be estimated as follow.

$$W_{20}=100.63-0.218\times\log(20\times365)$$
$$=1.87(\%)$$

From the calculation above it is enough to have the cask design margins of 2.0% heat weight loss even if the scattering in the test results are taken into account.

Water drops were observed on inside surface of sealed container when the lid was cut off to open and to take out the test specimen from the container after test. It is considered that these drops prevented the temperature rise of test specimen by evaporating during the test and reduced the heat weight loss of the specimen.

From the results, it is concluded that NS-4-FR is effective as neutron shielding material of cask, especially when it is used in enclosed condition.

BISCO (150°C HEATED)
TG

<Sample>
9.852mg
(9.852mg)
<Reference>

<Comment>
10°C/min

<Date>
90/04/10 16:51

<Temp.program[C] [C/min] [min]>
1* 20.0-500.0 10.00 0.00
<Gas>
N_2 AND AIR 150.0ml/min
0.0ml/m

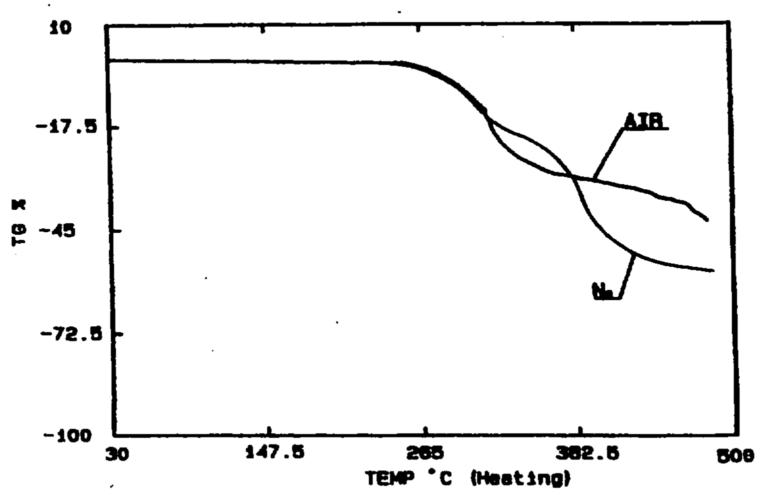


Fig 1 TG OF NS-4-FR IN AIR AND NITROGEN

BISCO
TG

<Sample>
RESIN AND NS-4-FR
7.968mg
(7.968mg)
<Reference>
A1203

<Comment>

<Date>
90/04/26 14:00

<Temp.program[C] [C/min] [min]>
1* 25.0-500.0 10.00 0.00
<Gas>
N_2 200.0ml/min
0.0ml/min

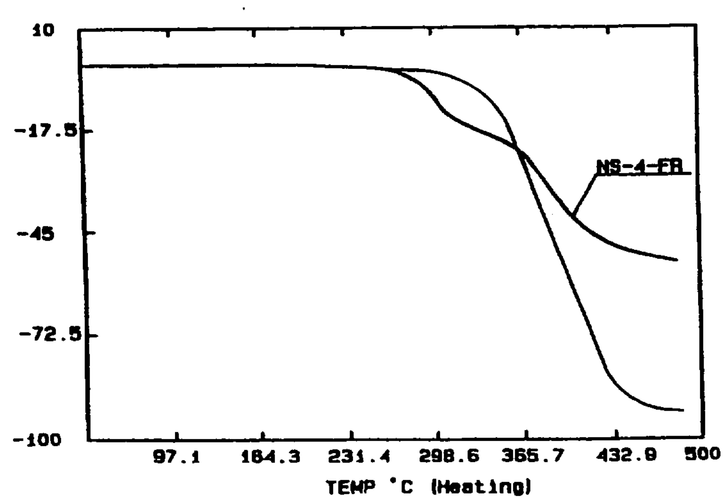


Fig 2 TG OF NS-4-FR W/O FIRE RETARDANT

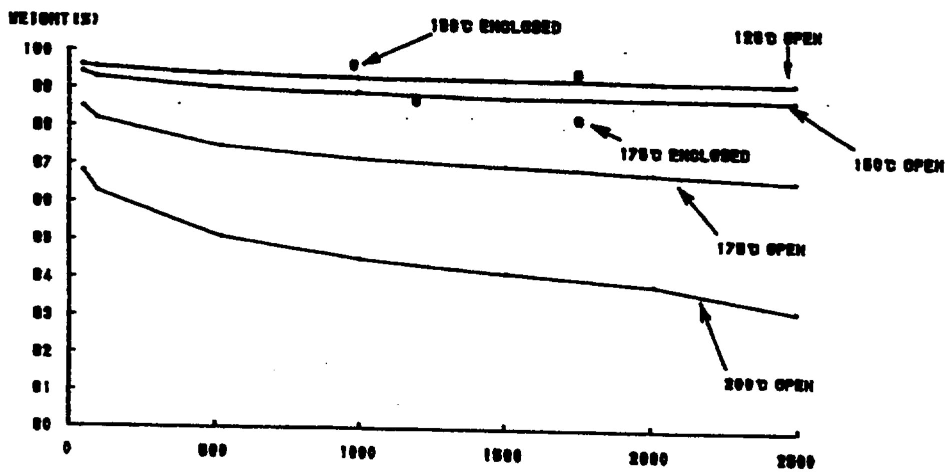


Fig.3 WEIGHT LOSS BY OPEN AND ENCLOSED TEST

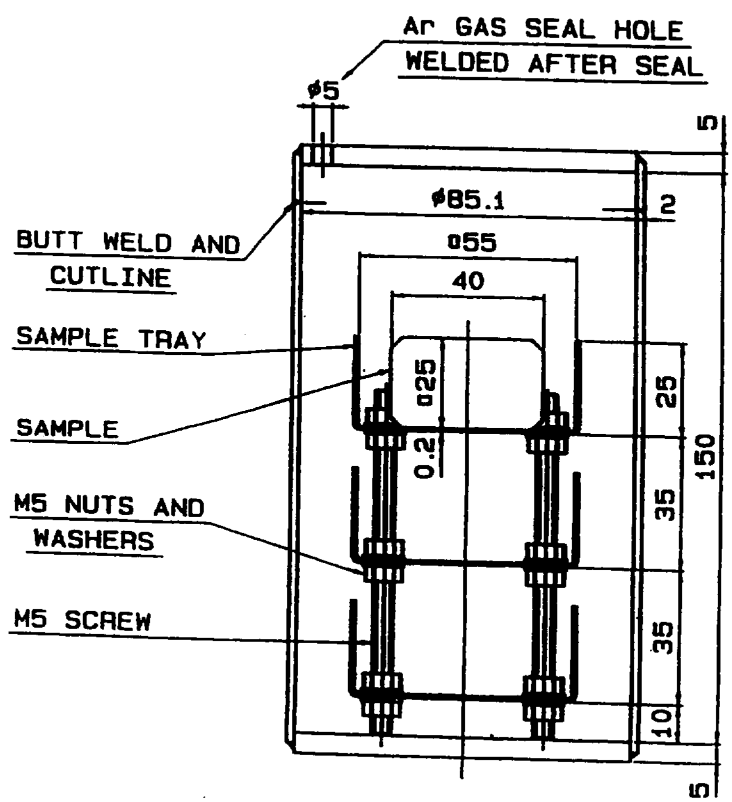


Fig 4 SEALED STAINLESS STEEL CONTAINER

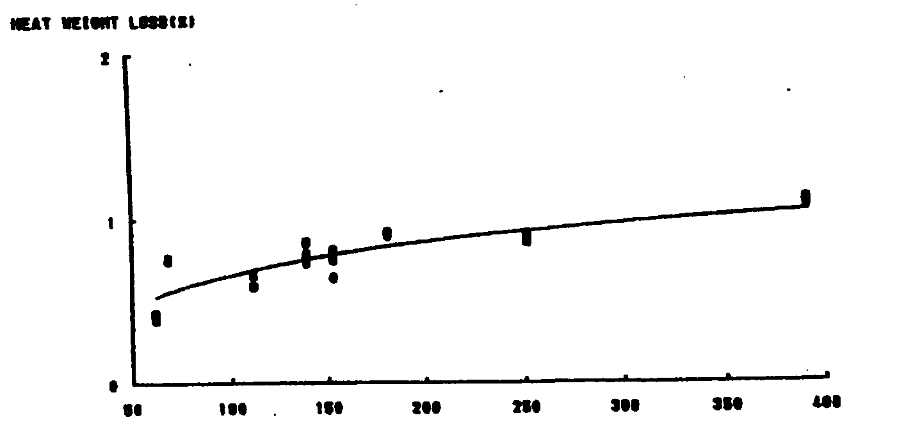


Fig.5 WEIGHT LOSS OF LONG TERM CYCLIC TEST

Thermal Testing of Solid Neutron Shielding Materials*

Richard H. Boonstra

General Atomics, San Diego, California

INTRODUCTION

General Atomics (GA) is currently developing two legal-weight truck casks for the Department of Energy (DOE), Office of Civilian Radioactive Waste Management (OCRWM). These casks, the GA-4 and GA-9, will carry four PWR and nine BWR spent fuel assemblies, respectively. Each cask has a solid neutron shielding material separating the steel body and the outer steel skin. In the thermal accident specified by NRC regulations in 10CFR Part 71 the cask is subjected to an 800°C environment for 30 minutes. The neutron shield need not perform any shielding function during or after the thermal accident, but its behavior must not compromise the ability of the cask to contain the radioactive contents.

In May-June 1989 the first series of full-scale thermal tests was performed on three shielding materials: Bisco Products NS-4-FR, and Reactor Experiments RX-201 and RX-207. The tests are described in *Thermal Testing of Solid Neutron Shielding Materials*, GA-A19897, R. H. Boonstra, General Atomics (1990), and demonstrated the acceptability of these materials in a thermal accident. Subsequent design changes to the cask rendered these materials unattractive in terms of weight or adequate service temperature margin. For the second test series a material specification was developed for a polypropylene based neutron shield with a softening point of at least 280°F. Table 1 lists the neutron shield materials tested. The Envirotech and Bisco materials are not polypropylene, but were tested as potential backup materials in the event that a satisfactory polypropylene could not be found.

TESTING SETUP AND PROCEDURE

Setup

Figure 1 shows a representative test article and thermocouple positions. Each test article consists of blocks of neutron shield contained in a box of dimensions 36 in. x 36 in. x 4.5 in. The box is 11-gage type 304 stainless steel with continuous welds along all seams and a 6-in. x 12-in. hole in the center of a 36-in. x 36-in. face. The hole simulates damage from the hypothetical drop and puncture events and extends 1.5 in. into the material. Six inches of mineral fiber insulation surround the box except for the face with the hole.

Six KNBS (Chromel-Alumel) 20-gage thermocouples (TCs) measure temperatures on the test article. The environment temperatures are determined using five TCs of the same type as on the test article, but shielded to prevent radiant heat loss to the test article surface. These TCs are positioned 6 in. away from the test article surface.

* Work supported by the U.S. Department of Energy, Office of Civilian Radioactive Waste Management, under DOE Field Office, Idaho, Contract DE-AC07-88ID12698.

As shown in Fig. 2, a vertical exposure furnace heats the uninsulated face of the test article. The back wall of the furnace contains a bank of seven natural gas burners. A single globe valve regulates the overall gas pressure to control the test exposure to the temperature range of 800° to 900°C.

TABLE 1
NEUTRON SHIELD MATERIALS IN THERMAL TESTS

Supplier	Material
Kobe Steel, Ltd.	PP-R01 polypropylene, 1% boron
Envirotech Molded Products, Inc.	High-density polyethylene (HDPE), 0.8% boron
Bisco Products, Inc.	Modified NS-4, 4.5% boron
Reactor Experiments, Inc.	High melt index polypropylene (HMPP), 1% boron

Procedure

The completed test article is conditioned at room temperature for at least 24 hr, then moved into position directly in front of the furnace. Recording of the TC data begins 5 minutes prior to ignition of the burners. After ignition, the average furnace temperature (i.e., average of the five furnace TCs) is maintained between 800° and 900°C for 30 minutes. The burners are then shut off and the test article is pulled away to cool in ambient air. When all temperatures have peaked the test article is again conditioned at room temperature for at least 24 hr and then disassembled for inspection.

Material Acceptance Criterion

The neutron shield must not provide a source of thermal input to the cask sufficient to degrade containment integrity. More specifically, the material is acceptable if (1) temperatures on the back surface do not at any time exceed the maximum temperature of the thermal accident environment and (2) it shows no evidence of prolonged combustion (i.e., combustion lasting fo[r a] period of several hours) following the thermal accident.

OBSERVATIONS AND RESULTS

Two of the materials, the modified NS-4 and the HMPP, passed the test and are acceptable fo[r] use as the neutron shielding. The results for the PP-R01 and HDPE materials are inconclusive because the tests were stopped prematurely due to intense smoke combined with inadequate ventilation. Table 2 provides time-related observations for the four tests. Post-test inspection results and temperature plots are discussed below.

Kobe PP-R01

Disassembly of the test article after cooling indicated that despite the observed combustion a relatively small amount of material had been lost. The polypropylene was heavily charred and broken into loose tiles on the front surface. Some individual blocks had fused together but, contrary to expectations, there was little evidence of extensive melting. The total weight lo[ss] was 36.3 lb out of an initial 200.1 lb, or 18%.

Figure 3 gives the average environment temperature and the thermal response on the back surface of the test article, where the peak temperature is about 340°C. The responses exhi[bit] abrupt increases within the first several minutes, a characteristic of all the tests. All temperatures on the back surface meet the acceptance criterion that they not exceed the accident environment temperature, but acceptability of the material cannot be established si[nce] the test was halted.

Envirotech High-Density Polyethylene (HDPE)

The test article was opened to reveal that most of the material remained in the box. Charring was concentrated mostly on material near the hole and in the lower region beneath the hole. Here molten material had collected and resolidified as indicated by a layer of dark material. In the upper portion, above the hole, charring was confined mainly to the surface and virgin, white material in the form of individual blocks was still identifiable. There was very little loose material. The total weight loss was 27.0 lb from an initial weight of 199.6 lb, or about 14%.

The average environment temperature and back surface TC responses are shown in Fig. 4. The responses display the sudden rises seen in the preceding test, although the peak response is less than 150°C. In this respect the HDPE performed better than the PP-R01 and its temperatures also meet the acceptance criterion. However results are indecisive due to termination of the test.

Bisco Modified NS-4

This material is similar to the NS-4-FR tested in 1989 but has a slightly lower hydrogen content and a lower weight. In this test the heating phase successfully proceeded the full 30 minutes, and the material was deemed acceptable.

A post-test inspection indicated immediately that the majority of the material had been retained. This was confirmed by a weight loss of only 6%, 14.5 lb out of 244.6 lb. There was a fairly uniform, black char layer about 1/8 to 3/16 in. thick on the front surface. With the exception of this char layer, which was fragile and separated easily from the remainder of the material, all blocks appeared nearly intact. An average of 2-1/16 in. of undamaged material remained in the front layer of blocks. Some charring was also observed along the sides of blocks that joined at the locations of TCs 1 and 2 and TCs 4 and 5. The back layer of blocks revealed localized discoloration near the top corners and a black, oily material that had condensed on the block surfaces. Apart from this, very little damage was noticed on these blocks.

Figure 5 gives the average environment and back surface temperatures The maximum back surface response is just less than 100°C. The initial peaking of TCs 4 and 6 before 50 minutes is followed by a gradual increase of these temperatures with subsequent maximums between 400 and 500 minutes. The later gradual increase is obviously due to conduction of heat through the material to the back surface; the initial peaks are therefore due to some other phenomenon.

Reactor Experiments High-Melt Index Polypropylene (HMPP)

This material also went through the full 30 minutes of heating and post-test cooldown. It was subsequently judged acceptable for use in the cask.

After complete cooling in ambient air, the total material weight loss was determined to be 109 lb from the initial 194 lb, some 56%. Disassembly of the test article confirmed a significant absence of material. No obvious char layer was noted. The space above the bottom edge of the hole was empty except for a column of partially melted blocks along the left side of the box. In the lower section, beneath the bottom edge of the hole, the material had softened and fused into a solid mass, although gaps between individual blocks could still be seen. The thickness of the fused material at the center of its top edge was approximately 5 in., indicating that the front steel had bowed outward 0.5 in. and allowed additional material to flow down from the top portion. Melted and resolidified material could be seen as distinct layers on the exterior surfaces of the fused, bottom portion.

Temperatures of the average environment and on the back side of the test article are shown in Fig. 6. The maximum back face temperature, 211°C, occurs in TC 6 at about 6.5 minutes. The sudden increase in back face temperatures at the beginning of the test is again noted, particularly in TCs 4 and 6, which were located in the bottom portion of the test article.

CONCLUSIONS

The Bisco modified NS-4 and Reactor Experiments HMPP are both acceptable materials from a thermal accident standpoint for use in the shipping cask. Tests of the Kobe PP-R01 and Envirotech HDPE were stopped for safety reasons, due to inability to deal with the heavy smoke, before completion of the 30-minute heating phase. However these materials may prove satisfactory if they could undergo the complete heating.

Table 3 compares key results for all four materials. The Bisco modified NS-4 is best in terms of survivability and back side temperatures. Despite the more intense combustion of the HMPP during the 30-minute heating phase, the test article did not sustain this combustion when moved away from the furnace. Had the molten material not become entrapped on the test article insulation (which is not a part of the actual cask), the combustion would have expired much earlier. The back side maximum temperature of 211°C is well within the criterion of 800°C. This material therefore is also acceptable. It is preferable to the Bisco modified NS-4 since it is about 20% lighter and at the same time has a hydrogen density 20% higher.

The tests of the PP-R01 and HDPE materials were terminated for safety due to the extremely heavy smoke in the indoor test facility. The combustion of the materials without any external heat input led to the initial belief that they were unacceptable. However the HMPP test showed the same phenomenon, and combustion in the test article eventually ceased when it was moved from the furnace. This fact led to the conclusion that placing the test article against the well-insulated furnace formed an effective heat-retaining environment that kept the temperature high enough to support combustion. Neutron shield material became sufficient fuel in place of the gas burners. Moving the test article to cool in ambient air disrupted this process and the combustion ended. It is thus plausible that the PP-R01 and HDPE would behave like the HMPP and prove acceptable if these tests could proceed the full duration. Note that backside temperatures (see Figs. 3(b) and 4(b)) remained well below the acceptance criterion of 800°C.

TABLE 3
COMPARISON OF NEUTRON SHIELD THERMAL PERFORMANCE

Material	Time to End of Flames (min.)	Weight Loss (%)	Peak back side temps (°C)			
			TC 1	TC 3	TC 4	TC 6
PP-R01	Test terminated	18	341	169	328	336
HDPE	Test terminated	14	83	140	146	78
NS-4	5	6	96	41	47	42
HMPP	11.5 (1), 50 (2)	56	127	203	152	211

(1) Material inside test article
(2) Material outside test article entrapped on insulation

ACKNOWLEDGMENT

The author would like to express his appreciation to Omega Point Laboratories (San Antonio, TX) for performing the tests.

REFERENCES

Boonstra, R.H., *Thermal Testing of Solid Neutron Shielding Materials*, GA-A19897, General Atomics (1990).

FIGURE WITHHELD UNDER 10 CFR 2.390

Fig. 1 Neutron shield test article with thermocouple locations

FIGURE WITHHELD UNDER 10 CFR 2.390

Fig. 2 Neutron shield test furnace

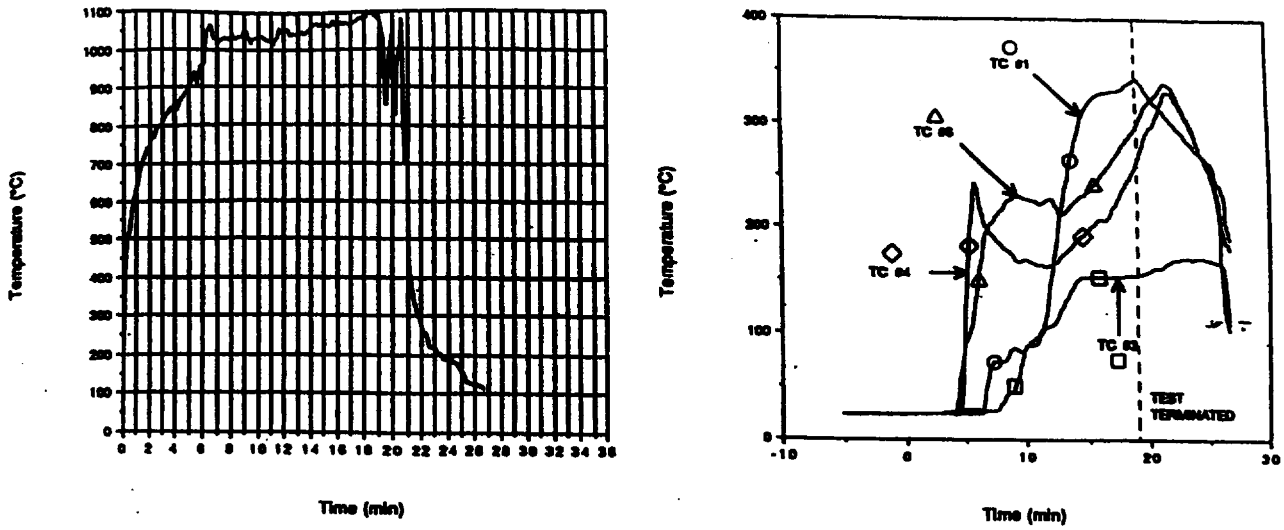


Fig. 3 Kobe PP-R01 test temperatures: (a) average environment, (b) back side response.

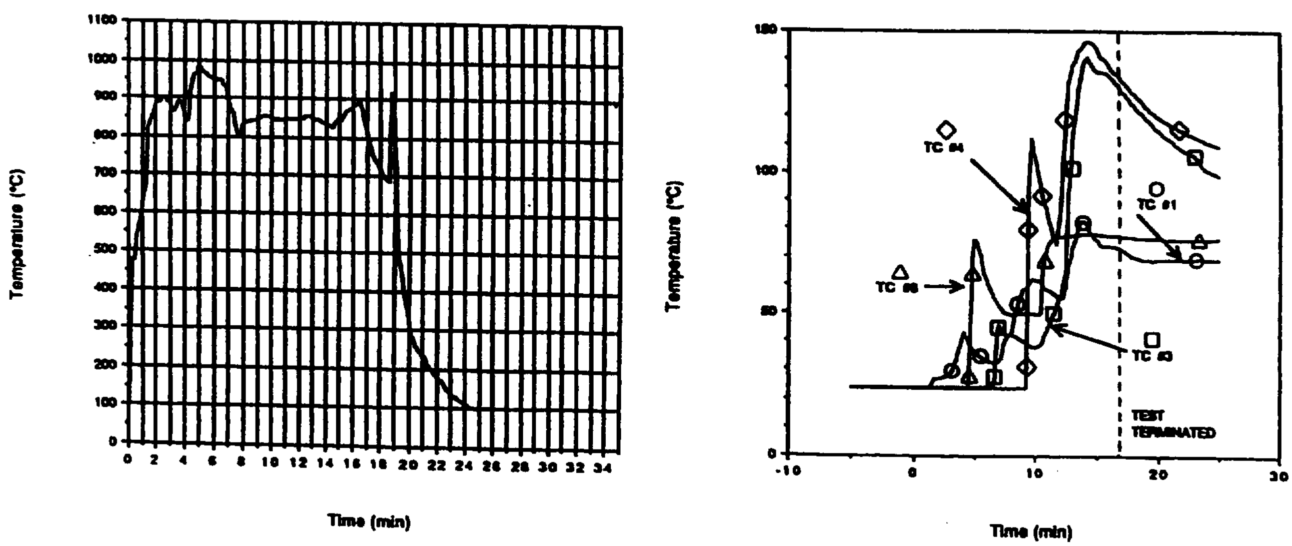


Fig. 4 Envirotech HDPE test temperatures: (a) average environment, (b) back side response.

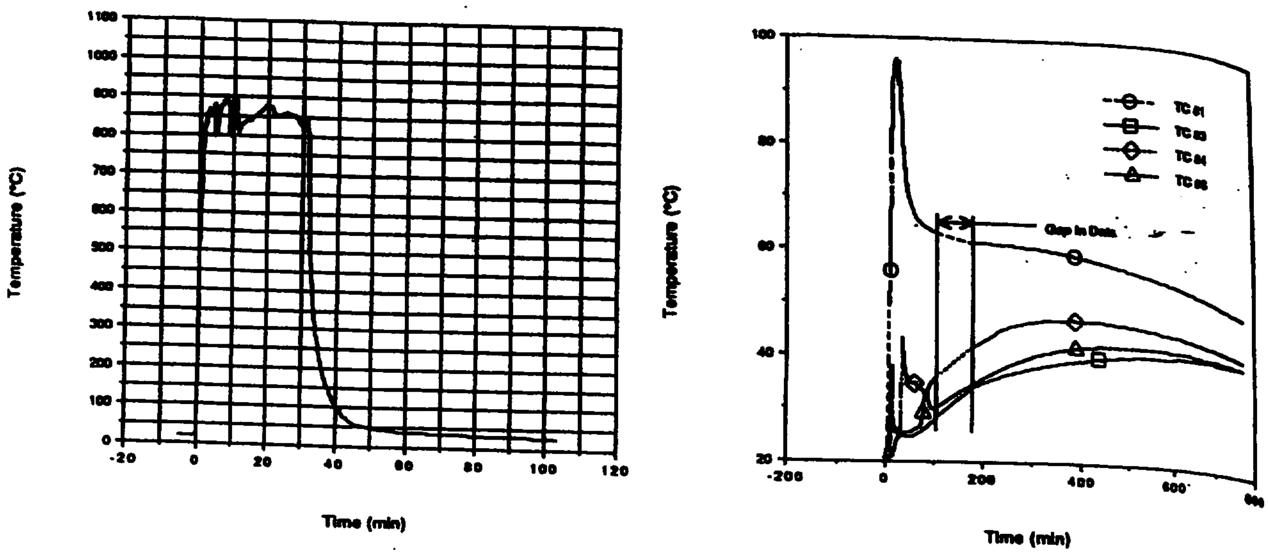


Fig. 5 **Bisco Modified NS-4 test temperatures: (a) average environment, (b) back side response.**

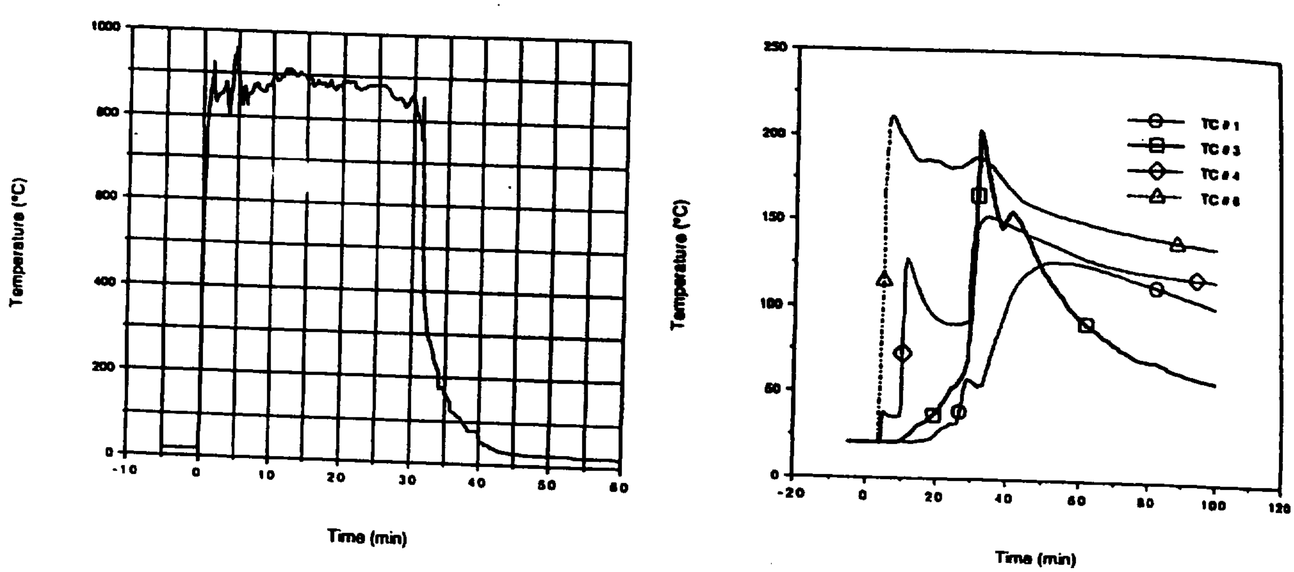


Fig. 6 **Reactor Experiments HMPP test temperatures: (a) average environment, (b) back side response.**

TABLE 2
THERMAL TEST OBSERVATIONS

PP-R01 polypropylene, 1% boron	High-density polyethylene (HDPE), 0.8% boron	Modified NS-4, 4.5% boron	High melt index Polypropylene (HMPP), 1% boron
3 min — Temp. = 800°C, slight smoking and flame from exhaust stack. 5.5 min — Temp. > 900°C, smoke and flames fill furnace interior. Furnace burners turned down, then shut off. 6.5 min — Temp. > 1000°C, extremely heavy smoke. 18.5 min — Temp. = 1100°C, decision made to terminate test. 19 min — Two failed attempts to extinguish flames with CO_2. Test article moved from furnace and flames extinguished with water.	2 min — Temp. = 800°C 2.5 min — material ignited 4.5 min — Temp. > 900°C, material flowing out of box front with vigorous flaming and smoking from exhaust port. Furnace burners turned down, then shut off. 5 min — Temp. ≈ 1000°C, exhaust port damping reduced temp. to within test range. 17 min — Test terminated due to copious smoke production, flames visible along bottom of test article. Test article moved from furnace. Flames observed inside furnace indicating material had been discharged into furnace. Flames extinguished with water.	Additional ventilating capability supplied. Environment achieved desired test range and maintained there for 30 min without difficulty. Some smoke was observed and some flames could be seen from the exhaust port, but intensity of combustion was less than in previous tests. 30 min — Test article moved from furnace. Flames approximately 4 ft high issue from hole in test article. After about 5 min, the flames diminish and self extinguish. White smoke persists another 3.5 min.	3 min — smoke begins issuing from exhaust port. 5 - 6 min — 3-ft column of flame appears, burners adjusted to control temp to specified range. 17 min — Furnace burners shut off. Internal combustion maintains temp. in test range. Molten material observed flowing into water-filled catch pan. 30 min — Test article moved from furnace. Flames issuing from hole and from material that had flowed out of the hole and become trapped in the insulation. Flames from hole gradually diminish and self extinguish after 12 min. Flames from material in insulation self extinguish after another 38 min.

bisco

bisco products, inc.
[illegible]

April 20, 1987

Mr. Todd Lesser
Nuclear Assurance Corporation
5720 Peachtree Plaza
Norcross, GA 30092

SUBJECT: WEIGHT LOSS OF NS-4-FR UNDER EXTREME TEMPERATURE CONDITIONS.

Dear Todd,

As a follow-up to our telephone conversation last week, this letter will confirm the results of the temperature testing that has been performed on NS-4 FR to date.

<u>TEST 1</u>: <u>Weight loss of NS-4-FR at -70°F:</u>

A sample of NS-4-FR was weighed at room temperature, and was then exposed to a low temperature of -170° F for a period of time sufficient to bring the entire sample to a temperature below -70° F. The sample was then weighed and allowed to gradually return to room temperature. A final weight measurement was then taken after the sample had reached room temperature. The beginning and ending weights were identical. The weight at -170° F was slightly higher than the initial weight, probably due to condensation.

CONCLUSION: Since there is no weight loss when exposed to temperatures below -70 F, it can be concluded that there is also no hydrogen loss in the NS-4-FR at that temperature.

<u>TEST 2</u>: <u>Weight loss of NS-4-FR at 338o F (170o C):</u>

This is an update of the thermal aging test begun on November 20, 1986. As of April 17, 1987, the two NS-4-FR samples have been exposed to a continuous temperature of 338°F for 146 straight days. The samples have been periodically pulled out and weighed.

CONCLUSION: The cumulative weight loss as of April 17, 1987 is 3.05% and 3.18% for the two bricks. This test will be continued until such time that the additional weight loss is zero or negligible.

Please let me know if you require any additional information regarding BISCO NS-4-FR material.

Very truly yours,

Larry J. Dietrick
Project Engineer
LJD/hf

APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA
(Total of 6 Pages Including This Page)

The information provided in this appendix specifies the thermal expansion foam (silicone sponge) and paint properties and demonstrates their suitability for use in spent nuclear fuel storage casks. The following is a listing of the information provided.

- HT-800 Series, Silicone Sponge, Bisco Products Technical Data Sheet
- Thermaline 450, Carboline, Product Data Sheet and Application Instructions
- Carboline 890, Carboline, Product Data Sheet and Application Instructions

HT-870 silicone sponge is specified as a thermal expansion foam to be placed in the HI-TRAC transfer cask top lid with the neutron shield. Due to differing thermal expansion of the neutron shield and the carbon steel, the silicone sponge is provided to compress and allow the neutron shield material to expand. The compression-deflection physical properties are provided for the silicone sponge.

Silicone has a long and proven history in the nuclear industry. Silicone is highly resistant to degradation as a result of radiation at the levels required for the HI-TRAC transfer cask. Silicone is inherently inert and stable and will not react with the metal surfaces or neutron shield material. Additionally, typical operating temperatures for silicone sponges range from -50°F to 400°F.

Thermaline 450 is specified to coat the inner cavity of the storage overpack and HI-TRAC transfer cask, and Carboline 890 is specified to coat the external surfaces of the storage overpack and HI-TRAC transfer cask. As can be seen from the product data sheets, the paints are suitable for the design temperatures (see Table 2.2.3) and environment.

HT-800 SERIES

Specification Grade
Silicone Sponge

PHYSICAL PROPERTIES

PROPERTY	SPECIFICATION			TEST METHOD
	HT-870 (Soft)	HT-800 (Medium)	HT-820 (Firm)	
Density	12 - 24 pcf	16 - 28 pcf	20 - 32 pcf	ASTM D-3574
Compression Force @ 25% Deflection	2 - 7 psi	6 - 14 psi	12 - 20 psi	ASTM D-1056
Compression Set (Maximum)	10%	10%	10%	ASTM D-1056 (Compressed 50% for 22 hrs. @ 100°C)
Water Absorption (Maximum)	10%	5%	5%	ASTM D-1056

Available Industry Specifications:

AMS-3195 (HT-800)
AMS-3196 (HT-820)
UL-94 (Limited to specific classes, densities, thicknesses and colors)

Data is based on laboratory tests and should not be used for writing specifications. Each user should run independent tests to confirm material suitability for each specific application.

Bisco Products and Dow Corning neither represent nor [illegible] this material for medical device applications or for pharmaceutical end-use.

[redacted]

2300 East Devon Avenue • Elk Grove Village, Illinois 60007-6120 • Toll Free 800/237-20[illegible]

product data sheet

carboline®

THERMALINE 450

VOC

0928

SELECTION DATA

GENERIC TYPE: A glass flake filled, phenolic modified, amine cured epoxy novalac.

GENERAL PROPERTIES: A dense cross-linked polymer which exhibits outstanding barrier protection against a variety of chemical exposures. Excellent resistance to wet/dry cycling conditions at elevated temperatures. Designed to coat the exterior of insulated piping. It is also suitable for coating non-insulated piping and equipment exposed to chemical attack. The glass flakes help provide excellent abrasion resistance, permeation resistance and internal reinforcement.

- Temperature resistance to 450°F
- Excellent abrasion resistance
- Excellent overall chemical resistance
- Excellent thermal shock resistance

RECOMMENDED USES: Typically used as a one coat system to coat pipes and tanks that will be insulated. May also be used to coat non-insulated pipe, structural steel, equipment or concrete that may be subjected to severe chemical attack, abrasion or other abuse typical of a chemical plant environment.

TYPICAL CHEMICAL RESISTANCE:

Exposure	Splash & Spillage	Fumes
Acids	Excellent	Excellent
Alkalies	Excellent	Excellent
Solvents	Excellent	Excellent
Salt	Excellent	Excellent
Water	Excellent	Excellent

TEMPERATURE RESISTANCE (Under insulation):

Continuous:	425°F (218°C)
Excursions to:	450°F (232°C)

At 200°F (93°C) coating discoloration may be observed without loss of film integrity.

SUBSTRATES: Apply over properly prepared steel.

COMPATIBLE COATINGS: Normally applied directly to substrate. May be applied over epoxies and phenolics as recommended. May be topcoated with epoxies, polyurethanes or other finish coats as recommended.

July 96 Replaces September 95

SPECIFICATION DATA

THEORETICAL SOLIDS CONTENT OF MIXED MATERIAL:

	By Volume
THERMALINE 450	70 ± 2%

VOLATILE ORGANIC CONTENT (VOC):
The following are nominal values:
As supplied: 2.13 lbs./gal. (255 gm./liter).

Thinner	Fluid Ounces/Gal.	Pounds/ Gallon	Grams/ Liter
213	13	2.56	307

RECOMMENDED DRY FILM THICKNESS:
8-10 mils (200-250 microns) to be achieved in 1 or 2 coats.

THEORETICAL COVERAGE PER MIXED GALLON:*
1,117 mil sq. ft. (27.9 sq.m/l at 25 microns)
139 sq. ft at 8 mils (3.5 sq. m/l at 200 microns)
111 sq. ft at 10 mils (2.8 sq.m/l at 250 microns)

*Mixing and application losses will vary and must be taken into consideration when estimating job requirements.

STORAGE CONDITIONS: Store indoors.
Temperature: 40-110°F (4-43°C) Humidity 0-90%

SHELF LIFE: 24 months when stored indoors at 75°F (24°C)

COLOR: Red (0500) and Gray (5742)

GLOSS: Low (Epoxies lose gloss, discolor and eventually chalk in sunlight exposure.)

ORDERING INFORMATION

Prices may be obtained from your Carboline Sales Representative or Carboline Customer Service Department.

APPROXIMATE SHIPPING WEIGHT:

	1's	5's
THERMALINE 450	12 lbs. (5.5 kg)	58 lbs. (26.3 kg)
Thinner 213	8.4 lbs. (3.8 kg)	41 lbs. (18.6 kg)

FLASH POINT: (Setaflash)

THERMALINE 450 Part A:	53°F	(12°C)
THERMALINE 450 Part B:	>200°F	(>93°C)
Thinner 213	22°F	(-6°C)

0928

APPLICATION INSTRUCTIONS

THERMALINE 450

These instructions are not intended to show product recommendations for specific service. They are issued as an aid in determining correct surface preparation, mixing instructions and application procedure. It is assumed that the proper product recommendations have been made. These instructions should be followed closely to obtain the maximum service from the materials.

SURFACE PREPARATION: Remove all oil or grease from surface to be coated with Thinner 2 or Surface Cleaner 3 (refer to Surface Cleaner 3 instructions) in accordance with SSPC-SP 1.

STEEL:

Not Insulated: Abrasive blast to a Commercial Finish in accordance with SSPC-SP 6 and obtain a 2-3 mil (50-75 micron) blast profile.

Under Insulation: Abrasive blast to a Near White Finish in accordance with SSPC-SP 10 and obtain a 2-3 (50-75 micron) blast profile.

MIXING: Power mix each component separately, then combine and power mix in the following proportions.

Allow 30 minutes induction time at 75°F (24°C) prior to use.

	1 Gal. Kit	5 Gal. Kit
THERMALINE 450 Part A:	0.8 gals.	4.0 gals.
THERMALINE 450 Part B:	0.2 gals.	1.0 gals.

THINNING: May be thinned up to 13 oz/gal with Thinner 213.

Use of thinners other than those supplied or approved by Carboline may adversely affect product performance and void product warranty, whether express or implied.

POT LIFE: Three hours at 75°F (24°C) and less at higher temperatures. Pot life ends when coating loses body and begins to sag.

APPLICATION CONDITIONS:

	Material	Surfaces	Ambient	Humidity
Normal	65-85°F (18-29°C)	65-85°F (18-29°C)	65-85°F (18-29°C)	30-60%
Minimum	55°F (13°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	110°F (43°C)	100°F (38°C)	85%

Do not apply when the surface temperature is less than 5°F or 3°C above the dew point.

Special thinning and application techniques may be required above or below normal conditions.

SPRAY: The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

Conventional: Pressure pot equipped with dual regulators, 1/2" I.D. minimum material hose, .110" I.D. fluid tip and appropriate air cap.

July 96 Replaces September 95

Airless:

Pump Ratio:	30:1 (min)*
GPM Output:	3.0 (min)
Material Hose:	1/2" I.D. (min)
Tip Size:	.035"-.041"
Output psi:	2200-2500

*Teflon packings are recommended and are available from the pump manufacturer.

BRUSH: For striping of welds, touch-up of small areas only. Use a natural bristle brush, applying full strokes Avoid rebrushing.

ROLLER: Not recommended.

DRYING TIMES: These times are based on a dry film thickness of 10 mils (250 microns). Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure.

Surface Temperature	Dry To Handle	Dry to Topcoat	Final Cure
50°F (10°C)	18 hours	48 hours	21 days
60°F (16°C)	12 hours	32 hours	14 days
75°F (24°C)	6 hours	16 hours	7 days
90°F (32°C)	3 hours	8 hours	4 days

If the final cure time has been exceeded, the surface must be abraded by sweep blasting prior to the application of any additional coats.

EXCESSIVE HUMIDITY OR CONDENSATION ON THE SURFACE DURING CURING MAY RESULT IN A SURFACE HAZE OR BLUSH; ANY HAZE OR BLUSH MUST BE REMOVED BY WATER WASHING BEFORE RECOATING.

VENTILATION & SAFETY: WARNING: VAPORS MAY CAUSE EXPLOSION. When used in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. In addition to insuring proper ventilation, fresh air respirators or fresh air hoods must be used by all application personnel. Where flammable solvents exist, explosion-proof lighting must be used. Hypersensitive persons should wear clean, protective clothing, gloves and/or protective cream on face, hands and all exposed areas.

CLEANUP: Use Thinner 2.

CAUTION: READ AND FOLLOW ALL CAUTION STATEMENTS ON THIS PRODUCT DATA SHEET AND ON THE MATERIAL SAFETY DATA SHEET FOR THIS PRODUCT.

CAUTION: CONTAINS FLAMMABLE SOLVENTS. KEEP AWAY FROM SPARKS AND OPEN FLAMES. WORKMEN IN CONFINED AREAS MUST WEAR FRESH AIRLINE RESPIRATORS. HYPERSENSITIVE PERSONS SHOULD WEAR GLOVES OR USE PROTECTIVE CREAM. ALL ELECTRICAL EQUIPMENT AND INSTALLATIONS SHOULD BE MADE IN ACCORDANCE WITH THE NATIONAL ELECTRICAL CODE. IN AREAS WHERE EXPLOSION HAZARDS EXIST, WORKMEN SHOULD BE REQUIRED TO USE NONFERROUS TOOLS AND TO WEAR CONDUCTIVE AND NONSPARKING SHOES.

carboline
350 Hanley Industrial Ct • St Louis MO 63144-1599

carboline

CARBOLINE® 890

VOC

SELECTION DATA

GENERIC TYPE: Cross-linked epoxy.

GENERAL PROPERTIES: CARBOLINE 890 is a self priming, high solids, high gloss, high build epoxy mastic. It can be applied by spray, brush, or roller over hand or power tool cleaned steel and is compatible with most existing coatings and tightly adhered rust. The cured film provides a tough, cleanable surface and is available in a wide variety of colors.

- Single coat corrosion protection.
- Excellent chemical resistance.
- Good flexibility and lower stress upon curing than most epoxy coatings.
- Excellent tolerance of damp (not wet) substrates.
- Very good abrasion resistance.
- Suitable replacement for Carbomastic 801.

RECOMMENDED USES: Recommended where a high performance, chemically resistant epoxy coating is desired. Offers outstanding protection for interior floors, walls, piping, equipment and structural steel or as an exterior coating for railcars, structural steel and equipment in various corrosive environments. Industrial environments include Chemical Processing, Offshore Oil and Gas, Food Processing, Pharmaceutical, Water and Waste Water Treatment, Pulp and Paper and Power Generation among others. May be used as a two coat system direct to metal or concrete for Water and Municipal Waste Water immersion. Acceptable for use in incidental food contact areas and as a lining for hopper cars carrying food grade plastic pellets when processed according to FDA criteria (ref: FDA 21 CFR 175.300). Consult Carboline Technical Service Department for other specific uses.

NOT RECOMMENDED FOR: Strong acid or solvent exposures, immersion service other than water, exterior weathering where color retention is desired, such as a finish for tank exteriors or over chlorinated rubber and latex coatings.

TYPICAL CHEMICAL RESISTANCE:

Exposure	Immersion	Splash & Spillage	Fumes
Acids	NR	Very Good	Very Good
Alkalies	NR	Excellent	Excellent
Solvents	NR	Very Good	Excellent
Salt Solutions	Excellent	Excellent	Excellent
Water	Excellent	Excellent	Excellent

TEMPERATURE RESISTANCE: (Non-Immersion)

Continuous:	250°F (121°C)
Non-continuous:	300°F (149°C)

At temperatures above 225°F, coating discoloration and loss of gloss can be observed, without loss of film integrity.

SUBSTRATES: Apply over suitably prepared metal, concrete, or other surfaces as recommended.

COMPATIBLE COATINGS: May be applied directly over inorganic zincs, weathered galvanizing, epoxies, phenolics or other coatings as recommended. A test patch is recommended before use over existing coatings. A mist coat of CARBOLINE 890 is required when applied over inorganic zincs to minimize bubbling. May be topcoated with polyurethanes or acrylics to upgrade weathering resistance. Not recommended over chlorinated rubber or latex coatings. Consult Carboline Technical Service Department for specific recommendations.

SPECIFICATION DATA

THEORETICAL SOLIDS CONTENT OF MIXED MATERIAL:*

	By Volume
CARBOLINE 890	75% ± 2%

VOLATILE ORGANIC CONTENT:*

As Supplied: 1.78 lbs./gal. (214 grams/liter)

Thinned:

Thinner	Fluid Ounces/Gal.	Pounds/ Gallon	Grams/ Liter
2	8	2.08	250
2	13	2.26	271
33	16	2.38	285

*Varies with color

RECOMMENDED DRY FILM THICKNESS PER COAT:
4-6 mils (100-150 microns).
6-8 mils (150-200 microns) DFT for a more uniform gloss over inorganic zincs, or for use over light rust.

In more severe environments a second coat of 4-6 mils (100-150 microns) is recommended.

Dry film thickness in excess of 10 mils (250 microns) per coat is not recommended. Excessive film thickness over inorganic zinc may increase damage during shipping or erection.

THEORETICAL COVERAGE PER MIXED GALLON:
1203 mil sq. ft. (30 sq. m/l at 25 microns)
241 sq. ft. at 5 mils (6.0 sq. m/l at 125 microns)

Mixing and application losses will vary and must be taken into consideration when estimating job requirements.

STORAGE CONDITIONS: Store Indoors
Temperature: 40-110°F (4-43°C) Humidity: 0-100%

SHELF LIFE: 36 months when stored at 75°F (24°C).

COLORS: Available in Carboline Color Chart colors. Some colors may require two coats for adequate hiding.

GLOSS: High gloss (Epoxies lose gloss, discolor and eventually chalk in sunlight exposure).

ORDERING INFORMATION

Prices may be obtained from your Carboline Sales Representative or Carboline Customer Service Department.

APPROXIMATE SHIPPING WEIGHT:

	2 Gal. Kit	10 Gal. Kit
CARBOLINE 890	29 lbs. (13 kg)	145 lbs. (66 kg)

	1's	5's
THINNER #2	8 lbs. (4 kg)	39 lbs. (18 kg)
THINNER #33	9 lbs. (4 kg)	45 lbs. (20 kg)

FLASH POINT: (Setaflash)

CARBOLINE 890 Part A	89°F (32°C)
CARBOLINE 890 Part B	71°F (22°C)
THINNER #2	24°F (-5°C)
THINNER #33	89°F (32°C)

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APPLICATIONS INSTRUCTIONS
CARBOLINE® 890

These instructions are not intended to show product recommendations for specific service. They are issued as an aid in determining correct surface preparation, mixing instructions and application procedure. It is assumed that the proper product recommendations have been made. These instructions should be followed closely to obtain the maximum service from the materials.

SURFACE PREPARATION: Remove all oil or grease from surface to be coated with Thinner #2 or Surface Cleaner #3 (refer to Surface Cleaner #3 instructions) in accordance with SSPC-SP 1.

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Steel: For mild environments Hand Tool or Power Tool Clean in accordance with SSPC-SP 2, SSPC-SP 3 or SSPC-SP 11 to produce a rust-scale free surface.

For more severe environments, abrasive blast to a Commercial Finish in accordance with SSPC-SP 6 and obtain a 1½ - 3 mil (40-75 micron) blast profile.

For immersion service, abrasive blast to a Near White Metal Finish in accordance with SSPC-SP10 and obtain a 1½ - 3 mil (40-75 micron) blast profile.

Concrete: Must be cured at least 28 days at 70°F (21°C) and 50% R.H. or equivalent time. Remove fins and other protrusions by stoning, sanding or grinding. Abrasive blast to open all surface voids and remove all form oils, incompatible curing agents, hardeners, laitance and other foreign matter and produce a surface texture similar to that of a medium grit sandpaper. Voids in the concrete may require surfacing. Blow or vacuum off sand and dust.

MIXING: Power mix separately, then combine and power mix in the following proportions:

	2 Gal. Kit	10 Gal. Kit
CARBOLINE 890 Part A	1 gallon	5 gallons
CARBOLINE 890 Part B	1 gallon	5 gallons

THINNING: For spray applications, may be thinned up to 13 oz./gal. with Thinner #2. For hot and windy conditions, or for brush and roller application, may be thinned up to 16 oz./gal. with Thinner #33.

Use of thinners other than those supplied or approved by Carboline may adversely affect product performance and void product warranty, whether express or implied.

POT LIFE: Three hours at 75°F (24°C) and less at higher temperatures. Pot life ends when material loses film build.

APPLICATION CONDITIONS:

	Material	Surfaces	Ambient	Humidity
Normal	60-85°F (16-29°C)	60-85°F (16-29°C)	60-90°F (16-32°C)	0-80%
Minimum	50°F (10°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	125°F (52°C)	110°F (43°C)	90%

Do not apply or cure the material when the surface temperature is less than 5°F or 3°C above the dew point.

Special thinning and application techniques may be required above or below normal conditions.

SPRAY: This is a high solids coating and may require slight adjustments in spray techniques. Wet film thicknesses are easily and quickly achieved. The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

Conventional: Pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, .070" I.D. fluid tip and appropriate air cap.

Airless:

Pump Ratio:	30:1 (min.)*
GPM Output:	3.0 (min.)
Material Hose:	3/8" I.D. (min.)
Tip Size:	.017-.021"
Output psi:	2100-2300
Filter Size:	60 mesh

*Teflon packings are recommended and are available from the pump manufacturer.

BRUSH OR ROLLER: Use medium bristle brush, or good quality short nap roller. Avoid excessive rebrushing and rerolling. Two coats may be required to obtain desired appearance, hiding and recommended DFT. For best results, tie-in within 10 minutes at 75°F (24°C).

DRYING TIMES: These times are based on a 5 mils (125 microns) dry film thickness. Higher film thicknesses, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure.

Dry to Touch 2 1/2 hours at 75°F (24°C)
Dry to Handle 6 1/2 hours at 75°F (24°C)

Surface Temperature	Recoating With Itself	Dry to Topcoat	Final Cure
50°F (10°C)	12 hours	24 hours	3 days
60°F (16°C)	8 hours	16 hours	2 days
75°F (24°C)	4 hours	8 hours	1 day
90°F (32°C)	2 hours	4 hours	16 hours

Excessive humidity or condensation on the surface during curing can interfere with the cure, can cause discoloration and may result in a surface haze or blush. Any haze or blush must be removed by water washing before recoating. During high humidity conditions, it is recommended that the application be done while temperatures are increasing. For best results over "damp" surfaces, apply by brush or roller.

Maximum Recoat or Topcoat Times at 75°F (24°C):

With Epoxies - 30 days
With Polyurethanes - 90 days

If the maximum recoat time has been exceeded, surface must be abraded by sweep blasting prior to the application of any additional coats.

Minimum cure time before immersion service is 5 days at 75°F (24°C) surface temperature. Cure at temperatures below 60°F (16°C) is not recommended for immersion service.

VENTILATION & SAFETY: WARNING: VAPORS MAY CAUSE EXPLOSION. When used as a tank lining or in enclosed areas thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. In addition to ensuring proper ventilation, fresh air respirators or fresh air hoods must be used by all application personnel. Where flammable solvents exist, explosion-proof lighting must be used. Hypersensitive persons should wear clean, protective clothing, gloves and/or protective cream on face, hands and all exposed areas.

CLEANUP: Use Thinner # 2.

CAUTION: READ AND FOLLOW ALL CAUTION STATEMENTS ON THIS PRODUCT DATA SHEET AND ON THE MATERIAL SAFETY DATA SHEET FOR THIS PRODUCT.

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CAUTION: CONTAINS FLAMMABLE SOLVENTS. KEEP AWAY FROM SPARKS AND OPEN FLAMES. IN CONFINED AREAS, WORKMEN MUST WEAR FRESH AIRLINE RESPIRATORS. HYPERSENSITIVE PERSONS SHOULD WEAR GLOVES OR USE PROTECTIVE CREAM. ALL ELECTRIC EQUIPMENT AND INSTALLATIONS SHOULD BE MADE AND GROUNDED IN ACCORDANCE WITH THE NATIONAL ELECTRICAL CODE. IN AREAS WHERE EXPLOSION HAZARDS EXIST, WORKMEN SHOULD BE REQUIRED TO USE NONFERROUS TOOLS AND TO WEAR CONDUCTIVE AND NONSPARKING SHOES.

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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. 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. ACI 318-95 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 318-95 are used in calculations supporting this FSAR in load cases involving compression or bearing loads on the overpack concrete. However, since ACI 318-95 is intended for commercial applications and the overpack concrete is designated as an ITS Category B material, it is necessary to invoke provisions of ACI 349 (85) (which is sanctioned by NUREG-1536) for all requirements except for the allowable stress formulas (which do not exist in ACI 349) and load combinations. This appendix provides a 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 is included. The formulas for permissible compressive and bearing stresses set forth in ACI 318-95 are used. However, as plain concrete has very limited capabilities in tension, no tensile strength is allotted to the concrete.

The steel structure of the HI-STORM overpack provides the strength to meet all load combinations specified in Chapters 2 and 3. Credit for the structural strength of the plain concrete is limited to 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 applied to the plain concrete.

The shielding performance of the plain concrete is maintained by ensuring that 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.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85) supplemented by the provisions of NUREG 1536 (page 3-21) and standard good practice.

1.D.4 Construction Requirements

The HI-STORM 100 overpack is composed of a steel structure which houses plain concrete. The steel structure acts as the framework for the pouring of the concrete. The steel structure defines the dimensions of the concrete which ensures that the required thickness of concrete is provided. The fabrication sequence for the HI-STORM 100 overpack as it pertains to the concrete is provided below. All item numbers are taken from Drawings 1495 and 1561 provided in Section 1.5. All nomenclature is taken from bill-of-material BM-1575 which is provided in Section 1.5.

The steel structure of the HI-STORM 100 overpack body is assembled at a qualified steel fabrication facility. However, the top plate (Item 9) is not welded to the overpack inner and outer shells (Items 3 and 2, respectively); likewise, the pedestal shell (Item 5) is welded to the baseplate (Item 1), but the pedestal platform (Item 24) is not welded to the pedestal shell. The steel structure of the overpack body is transported to the reactor site or a nearby concrete facility.

Once the steel structure of the body is received, the body will be inspected to ensure the steel structure meets the requirements of Sections 5.1 and 6.1 of ACI 349. The concrete shall be mixed, conveyed, and deposited in accordance with Sections 5.2 through 5.4 of ACI 349. Sufficient rigidity in the steel structure overpack body is provided such that all the concrete may be placed in a single pour into each of the four segments formed by the inner shell (Item 3), outer shell (Item 2), and radial plates (Item 14). If more than one pour is performed, the requirements of Section 6.4 of ACI 349 must be met for construction joints. The pedestal shell may require bracing and support in accordance with Section 6.1 of ACI 349 to maintain the proper position and shape.

Mixing and placing of the concrete shall follow the guidance of Sections 5.6 and 5.7 for cold and hot weather conditions, respectively. Consolidation of the plain concrete shall be performed in accordance with ACI 309-87. As no reinforcement is placed in the concrete, the possibility of voids is greatly diminished. Curing of the concrete shall be in accordance with Section 5.5 of ACI 349, except that accelerated curing in accordance with Section 5.5.3

is not permitted. The water curing method shall be utilized in accordance with ACI 308-92, Standard Practice for Curing Concrete.

After curing, non-shrink grout shall be applied as necessary to account for any shrinkage of the concrete elevation. Then, the top plate (Item 9) is welded to the overpack inner and outer shells (Items 3 and 2, respectively) and the pedestal shell (Item 5) is welded to the pedestal platform (Item 24).

To fabricate the overpack lid an identical process is followed. The lid shell (Item 7) is welded to the lid bottom plate (Item 6) and the inner and outer shield block shells (Items 27 and 26, respectively) are welded to the lid top plate (Item 10). The overpack lid is transported to the reactor site or a nearby concrete facility. The lid will be inspected to ensure the structure meets the requirements of Sections 5.1 and 6.1 of ACI 349. The concrete shall be mixed, conveyed, and deposited in accordance with Sections 5.2 through 5.4 of ACI 349. Sufficient rigidity in the overpack lid is provided such that all the concrete may be placed in a single pour. If bracing and support is required, it shall be provided in accordance with Section 6.1 of ACI 349 to maintain the proper position and shape. If more than one pour is performed, the requirements of Section 6.4 of ACI 349 must be met for construction joints.

Mixing and placing of the concrete for the lid shall follow the guidance of Sections 5.6 and 5.7 for cold and hot weather conditions, respectively. Curing of the concrete shall be in accordance with Section 5.5 of ACI 349, except that accelerated curing in accordance with Section 5.5.3 is not permitted. The water curing method shall be utilized in accordance with ACI 308-92, Standard Practice for Curing Concrete. After curing, the lid shell (Item 7) is welded to the lid top plate (Item 10) and the shield block ring (Item 20) is welded to the inner and outer shield block shells (Items 27 and 26, respectively).

Table 1.D.1 provides the construction limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

1.D.5 Testing Requirements

Table 1.D.2 provides the testing requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

Table 1.D.1: Requirements on Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density (Minimum)	146 (lb/cubic feet)
Specified Compressive Strength	4,000 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318-95
Cement Type and Mill Test Report	Type II; Section 3.2 (ASTM C 150 or ASTM C595)
Aggregate Type	Section 3.3 (including ASTM C33(Note 2))
Nominal Maximum Aggregate Size	3/4 (inch)
Water Quality	Per Section 3.4
Material Testing	Per Section 3.1
Admixtures	Per Section 3.6
Air Content	6%[1] (Table 4.5.1)
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)
Concrete Quality	Per Chapter 4 of ACI 349
Mixing and Placing	Per Chapter 5 of ACI 349
Consolidation	Per ACI 309-87
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Maximum Local Temperature Limit Under Normal and Off-normal Conditions	200°F (See Note 3)
Maximum Local Temperature Limit Under Accident Conditions	350°F (Appendix A, Subsection A.4.2)
Aggregate Maximum Value[2] 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)

Notes:

1. All section and table references are to ACI 349 (85).
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 that has been shown by special tests or actual service to produce concrete of adequate strength and durability meeting the requirements of Tables 1.D.1 and 1.D.2 is acceptable in accordance with ACI 349 Section 3.3.2.
3. The 200°F long term temperature limit is specified in accordance with Paragraph A.4.3 of ACI 349 for normal conditions. The 200°F long term temperature limit is based on (1) the use of Type II cement, specified aggregate criteria, and the specified compressive stress in Table 1.D.1, (2) the relatively small increase in long term temperature limit over the 150EF specified in Paragraph A.4.1, and (3) the very low maximum stresses calculated for normal and off-normal conditions in Section 3.4 of this FSAR.

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1 This limit is specified to accommodate severe exposure to freezing and thawing (Table 4.5.1).

2 The following aggregate types are a priori acceptable: limestone, dolomite, marble, basalt, granite, gabbro, 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 100 overpack concrete.

Table 1.D.1: Requirements on Plain Concrete

TEST	SPECIFICATION
Compression Test	ASTM C31, ASTM C39, ASTM C192
Air Content	ASTM C231
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"

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 requirement, the justification for the exception or clarification, and the Holtec approach to meet the intent of the NUREG-1536 requirement.

2.0 PRINCIPAL DESIGN CRITERIA

The design criteria for the MPC, HI-STORM 100 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, to the maximum extent practicable, 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,

† 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).

Subsection NB of the ASME Code, to the maximum extent practicable, as discussed in Section 2.2.4. The principal exception is the MPC lid, vent and drain cover plates, and closure ring 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.

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 final weld surface, in accordance with the Design Drawings contained in Section 1.5. The integrity of the MPC lid weld is further verified by performing a volumetric examination, a hydrostatic pressure test and a helium leak test, in accordance with the Design Drawings and Technical Specification requirements contained in Chapter 12.

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

Compliance with the ASME Code as it is applied to the design and fabrication of the MPC and the associated justification are discussed in Section 2.2.4. Compliance with the ASME Code is fully consistent with that used by other canister-based dry storage systems previously approved by the NRC.

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 100 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.

Thermal

The allowable zircaloy fuel cladding temperature limits to prevent cladding failure during long-term dry storage conditions for the MPC are based on LLNL Report [2.2.14]. To provide additional conservatism, the permissible fuel cladding temperature limits, which are lower than those calculated with the LLNL methodology, have been calculated based on PNL Report [2.0.3]. Stainless steel cladding is demonstrated to withstand higher temperatures than that of zircaloy cladding in EPRI Report [2.2.13]. However, the zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel fuel cladding. The allowable fuel cladding temperatures which correspond to varying cooling times for the SNF to be stored in the MPCs are provided in Table 2.2.3.

The short-term allowable fuel cladding temperature that is applicable to off-normal and accident conditions, as well as the fuel loading, canister closure, and canister transfer operations in the HI-TRAC transfer cask, is 570EC (1058EF) based on PNL-4835 [2.2.15]. The MPC is backfilled with 99.995% pure helium at a mass specified in Chapter 12 during canister sealing operations to promote heat transfer and prevent cladding degradation.

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

The MPCs are designed for a bounding thermal source term, as described in Section 2.1.6. The maximum allowable fuel assembly heat load for each MPC is limited in accordance with the Technical Specifications contained in Chapter 12.

Shielding

The allowable doses for an ISFSI using the HI-STORM 100 System are delineated in 10CFR72.104 and 72.106. Compliance with this criteria is necessarily site-specific and is to be demonstrated by the licensee, as discussed in Chapters 5 and 12.

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

The MPCs are designed for design basis fuel at the maximum burnup and minimum cooling times, as described in Sections 2.1.7 and 5.2. The radiological source term for the MPCs are limited based on the burnup and cooling times specified in the Technical Specifications contained in Chapter 12. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation in accordance with 10CFR20, as discussed in Chapter 10.

Criticality

The MPCs provide criticality control for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 6.1. The effective neutron multiplication factor is limited to k_{eff} < 0.95 for fresh unirradiated intact fuel with optimum unborated 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 and fixed borated neutron absorbing materials (Boral) incorporated into the fuel basket assembly. The minimum specified boron concentration verified during Boral manufacture is further reduced by 25% for criticality analysis. No credit is taken for burnup. The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC are limited in accordance with the Technical Specifications contained in Chapter 12.

Confinement

The MPC provides for confinement of all radioactive materials for all deign basis normal, off-normal, and postulated accident conditions, as discussed in Section 7.1. A non-mechanistic breach of the canister and subsequent release of available fission products in accordance with specified release fractions is considered, as discussed in Section 7.3. The confinement function of the MPC is verified through hydrostatic testing, helium leak testing and weld examinations performed in accordance with the acceptance test program in Chapter 9 and the Technical Specifications contained in Chapter 12.

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 site-specific requirements that comply with the 10CFR50 Technical Specifications for the plant and the 10CFR72 Technical Specifications for the HI-STORM 100 System contained in Chapter 12.

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 contained 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 100 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 100 Overpack Design Criteria

General

The HI-STORM 100 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 100 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 100 Overpack concrete is shielding of the gamma and neutron radiation emitted by the spent nuclear fuel.

Unlike other concrete storage casks, the HI-STORM 100 Overpack concrete is enclosed in steel inner and outer shells connected to each other by four radial ribs, and top and bottom plates. Where typical concrete storage casks are reinforced by rebar, the HI-STORM 100 Overpack is supported by the inner and outer shells connected by four ribs. As the HI-STORM 100 Overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete. Providing further conservatism, the structural analyses for normal conditions demonstrate that the allowable stress limits of the structural steel are met even with no credit for the strength of the concrete. During accident conditions (e.g., tornado missile, tip-over, end drop, and earthquake), only the compressive strength of the concrete is accounted for in the analysis to provide an appropriate simulation of the accident condition. Where applicable, the compressive strength of the concrete is calculated in accordance with ACI-318-95 [2.0.1].

In recognition of the conservative assessment of the HI-STORM 100 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 100 Overpack concrete as specified in Appendix 1.D.

Steel components of the storage overpack are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for Class 3 plate and shell components. Compliance with the ASME Code is fully consistent with those used by other canister-based dry storage systems previously approved by the NRC.

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 Technical Specification requirements contained in Chapter 12. The load combinations for which the overpack is designed are defined in Section 2.2.7. The physical characteristics of the MPCs for which the overpack is designed are defined in Chapter 1.

Thermal

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

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

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

Shielding

The off-site dose for normal operating conditions at the site 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 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 site 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 typical ISFSI using the HI-STORM 100 System are provided in Chapters 5 and 10. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10CFR72.212.

The overpack is designed to limit the calculated surface dose rate at the cask midplane for all MPCs to 35 mrem/hr or less, as defined in Section 2.3.5. The overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The calculated overpack dose rates are determined in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC transfer operations and a dose assessment for a typical ISFSI, as described in Chapter 10. In addition, overpack dose rates are limited in accordance with the Technical Specification provided in Chapter 12.

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 site-specific conditions and requirements that also comply with the applicable 10CFR50 Technical Specification requirements for the site and the 10CFR72 Technical Specifications for the HI-STORM 100 System contained in Chapter 12.

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, 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 drop from the maximum allowable handling height, consistent with the Technical Specification contained in Chapter 12. The load combinations for which the HI-TRAC is designed are defined in Section 2.2.7. The physical characteristics of each MPC for which the HI-TRAC is designed are defined in Chapter 1.

Thermal

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

The HI-TRAC is designed for the maximum allowable heat load provided in the Technical Specification contained in Chapter 12. 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 corresponding Technical Specifications applicable to the HI-TRAC during hot and cold conditions is contained in Chapter 12. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1.6.

Shielding

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

The 125 ton HI-TRAC provides better shielding than the 100 ton HI-TRAC. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would 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 considerations, or space envelope limitations in the fuel pool or air lock. As with other dose reduction-based plant modifications, individual users who cannot accommodate the 125 ton HI-TRAC due to plant design limitations must perform a cost-benefit analysis of the modifications which would be necessary to use the 125 ton HI-TRAC. The cost of the modification(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.

Operation

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 plant-specific requirements and in accordance with site and HI-STORM 100 System Technical Specification requirements contained in Chapter 12.

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.

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 & Fabrication Codes:			
Enclosure Vessel	ASME Code, Section III, Subsection NB	10CFR72.24(c)(4)	Section 2.0.1
Fuel Basket	ASME Code, Section III, Subsection NG	10CFR72.24(c)(4)	Section 2.0.1
MPC Lifting Points	ANSI N14.6/NUREG-0612	10CFR72.24(c)(4)	Section 1.2.1.4
Design Dead Weights:			
Max. Loaded Canister (dry)	79,987 lb. (MPC-24) 87,241 lb. (MPC-68)	R.G. 3.61	Table 3.2.1
Empty Canister (dry)	39,667 lb. (MPC-24) 39,641 lb. (MPC-68)	R.G. 3.61	Table 3.2.1

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Cavity Pressures:			
Normal:	100 psig	ANSI/ANS 57.9	Section 2.2.1.3
Off-Normal:	100 psig	ANSI/ANS 57.9	Section 2.2.2.1
Accident (Internal)	125 psig	ANSI/ANS 57.9	Section 2.2.3.8
Accident (External)	60 psig	ANSI/ANS 57.9	Sections 2.2.3.6 and 2.2.3.10
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(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	ASME Code Section II, Part D	Table 2.2.3
Neutron Poison:			
Boral (normal)	800°F	See Section 4.3.1	Table 2.2.3
Boral (accident)	950°F	See Section 4.3.1	Table 2.2.3
PWR Fuel Cladding:			

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
5-year cooled	692°F	PNL-6189	Section 4.3
6-year cooled	677°F	PNL-6189	Section 4.3
7-year cooled	636°F	PNL-6189	Section 4.3
10-year cooled	626°F	PNL-6189	Section 4.3
15-year cooled	615°F	PNL-6189	Section 4.3
BWR Fuel Cladding:			
5-year cooled	742°F	PNL-6189	Section 4.3
6-year cooled	714°F	PNL-6189	Section 4.3
7-year cooled	671°F	PNL-6189	Section 4.3
10-year cooled	660°F	PNL-6189	Section 4.3
15-year cooled	648°F	PNL-6189	Section 4.3
Canister Backfill Gas	Helium	-	Section 12.3.3
Canister Backfill Mass	Varies by MPC	-	Section 12.3.3
Short-Term Allowable Fuel Cladding Temperature	1058°F	PNL-4835	Sections 2.0.1 and 4.3
Insolation	Protected by Overpack or HI-TRAC	-	Section 4.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
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	Multi-pass Partial Penetration		
Port Covers	Full 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 and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
Port Covers	Root Pass and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
Leak Testing:			

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Welds Tested	Shell seams, shell-to-baseplate, MPC lid-to-shell, and port covers-to-MPC lid	-	Section 7.1 and Chapters 8, 9, and 12
Medium	Helium	-	Sections 7.2 and Chapter 12
Max. Leak Rate	$5x10^{-6}$ cm^3/sec (helium)	-	Chapter 8
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
Hydrostatic Testing:			
Test Pressure	125 psig (+3, -0 psig)	-	Chapters 8 and 9
Welds Tested	MPC Lid-to-Shell, MPC Shell seams, MPC Shell-to-Baseplate	-	Section 8.1 and 9.1
Medium	Water	-	Section 8.1 and Chapter 9
Retrievability:			
Normal and Off-normal:	No Encroachment on Fuel Assemblies or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.4, 3.5, and 3.1.2
Post (design basis) Accident			

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Criticality:		10CFR72.124 & 10CFR72.236(c)	
Method of Control	Fixed Borated Neutron Absorber & Geometry	-	Section 2.3.4
Min. Boron Loading	0.0267 g/cm^2 (MPC-24) 0.0372 g/cm^2 (MPC-68) 0.01 g/cm^2 (MPC-68F)	-	Section 2.1.8
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)	
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
Storage Mode Position	See Table 2.0.2	10CFR20	Section 5.1.1

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
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) Up to 68 (MPC-68)	-	Table 1.2.1
Type of Cladding	Zircaloy*	-	Table 2.1.6
Fuel Condition	Intact*	-	Section 2.1.2 & Table 2.1.6
* Also designed to accommodate failed fuel, stainless clad fuel, and MOX fuel (Tables 2.1.7 and 2.1.8 and Chapter 12)			
PWR Fuel Assemblies:			
Type/Configuration	Various**	-	Table 2.1.3
** No control components are permitted.			
Max. Burnup	44,700 MWD/MTU (MPC-24)	-	Figure 2.1.6
Max. Enrichment	Varies by fuel design	-	Table 2.1.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Max. Decay Heat/Assembly:			
5-year cooled	870 W (MPC-24)	-	Tables 4.4.20 and 2.2.3
6-year cooled	840.4 W (MPC-24)	-	Tables 4.4.20 and 2.2.3
7-year cooled	757.5 W (MPC-24)	-	Tables 4.4.20 and 2.2.3
10-year cooled	738.3 W (MPC-24)	-	Tables 4.4.20 and 2.2.3
15-year cooled	715.4 W (MPC-24)	-	Tables 4.4.20 and 2.2.3
Minimum Cooling Time:	5 years (Intact Zr Clad Fuel) 8 years (Intact SS Clad Fuel)		
Max. Fuel Assembly Weight:	1,680 lb.	-	Table 2.1.6
Max. Fuel Assembly Length: (Unirradiated Nominal)	176.8 in.	-	Table 2.1.6
Max. Fuel Assembly Width (Unirradiated Nominal)	8.54 in.	-	Table 2.1.6
Fuel Rod Fill Gas:			
Pressure (max.)	500 psig	-	Section 4.3 & Table 4.3.2

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
BWR Fuel Assemblies:			
Type	Various	-	Table 2.1.4
Max. Burnup	41,700 MWD/MTU	-	Figure 2.1.6
Max. Enrichment	Varies by fuel design	-	Section 6.1 and Chapter 12
Max. Decay Heat/Assy.:			
5-year cooled	314.7 W (MPC-68)	-	Tables 4.4.21 and 2.2.3
6-year cooled	298.7 W (MPC-68)	-	Tables 4.4.21 and 2.2.3
7-year cooled	270.7 W (MPC-68)	-	Tables 4.4.21 and 2.2.3
10-year cooled	264.0 W (MPC-68)	-	Tables 4.4.21 and 2.2.3
15-year cooled	256.6 W (MPC-68)	-	Tables 4.4.21 and 2.2.3
Minimum Cooling Time:	5 yrs (Intact Zr Clad Fuel) 18 yrs. (Damaged Zr Clad Fuel) 18 yrs. (Zr Clad Fuel Debris) 10 yrs. (Intact SS Clad Fuel)		
Max. Fuel Assembly Weight:			
w/channels	700 lb.	-	Table 2.1.6
Max. Fuel Assembly Length (Unirradiated Nominal)	176.2 in.	-	Table 2.1.6

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Max. Fuel Assembly Width (Unirradiated Nominal)	5.85 in.	-	Table 2.1.6
Fuel Rod Fill Gas:			
End-of-Life Hot Standby Pressure (max.)	147 psig	-	Table 4.3.5

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
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
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 or Horizontal	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
Fuel Rod Rupture Releases:			
Fuel Rod Failures	1%	NUREG-1536	Section 2.2.1.3
Fill Gases	100%	NUREG-1536	Section 2.2.1.3
Fission Gases	30%	NUREG-1536	Section 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)	

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Ambient Temperature	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.2.2
Leakage of One Seal	No Loss of Confinement	ANSI/ANS 57.9	Section 2.2.2.4
Partial Blockage of Overpack Air Inlets	Two Air Inlets Blocked	-	Section 2.2.2.5
Fuel Rod Rupture Releases:			
Fuel Rod Failures	10%	NUREG-1536	Section 2.2.2.1
Fill Gases	100%	NUREG-1536	Section 2.2.2.1
Fission Gases	30%	NUREG-1536	Section 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	100%	NUREG-1536	Section 2.2.3.8
Fill Gases	100%	NUREG-1536	Section 2.2.3.8
Fission Gases	30%	NUREG-1536	Section 2.2.3.8
Particulates & Volatiles	See Table 7.3.1	-	Sections 2.2.3.9 and 7.3

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Confinement Boundary Leakage	7.5x10^{-6} cm^3/sec (helium)	-	Sections 2.2.3.9 and 7.3
Explosive Overpressure	60 psig (external)	10CFR72.122(c)	Section 2.2.3.10
Airflow Blockage:			
Vent Blockage	100% of Overpack Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
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
Partial Blockage of MPC Basket Vent Holes	Crud Depth (Table 2.2.8)	ESEERCO Project EP91-29	Section 2.2.3.4
Extreme Environmental Temperature	See Table 2.0.2	-	Section 2.2.3.14

Table 2.0.2
HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Section 2.0.2
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design & Fabrication Codes:			
Concrete			
Design	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Fabrication	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Compressive Strength	ACI 318-95 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Structural Steel			
Design	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Fabrication	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2

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

Type	Criteria	Basis	FSAR Reference
Max. Empty Overpack:			
Assembled with Top Cover	267,190 lb.	R.G. 3.61	Table 3.2.1
Max. MPC/Overpack	354,431 lb.	R.G. 3.61	Table 3.2.1
Design Cavity Pressures	N/A	-	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
Thermal:			
Maximum Design Temperatures:			
Concrete			
Local Maximum (Normal)	200°F	ACI 349 Appendix A	Table 2.2.3
Local Maximum (Accident)	350°F	ACI 349 Appendix A	Table 2.2.3
Steel Structure	350°F	ASME Code Section II, Part D	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.4.1.1.8
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A

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

Type	Criteria	Basis	FSAR Reference
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
Retrievability:			
Normal and Off-normal	No damage which precludes Retrieval of MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 and 3.4
Accident			Sections 3.5 and 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:		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
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 organ	10CFR72.104	Sections 5.1.1, 7.2, and 10.1
On Controlled Area Boundary from Design Basis Accident	5 rem to whole body or to any organ	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

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

Type	Criteria	Basis	FSAR Reference
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.)	239,877 lb. (125-ton HI-TRAC w/transfer lid)	R.G. 3.61	Table 3.2.2 Section 2.2.1.2
Dry Loaded MPC (max.)	87,241 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 ANSI N14.6	NUREG-0612 ANSI N14.6	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature During Handling Operations	0°F	ANSI/ANS 57.9	Section 2.2.1.2
Snow and Ice Load	100 lb./ft^2	ASCE 7-88	Section 2.2.1.6

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

Type	Criteria	Basis	FSAR Reference
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	Two Air Inlet Ducts Blocked	-	Section 2.2.2.5
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.94	
Drop Cases:			
End	11 in.	-	Section 2.2.3.1
Tip-Over	Assumed (Non-mechanistic)	-	Section 2.2.3.2
Fire:			
Duration	217 seconds	10CFR72.122(c)	Section 2.2.3.3
Temperature	1475°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

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

Type	Criteria	Basis	FSAR Reference
Ambient Temperature	80°F	10CFR72.128(a)(4)	Section 2.2.3.13
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			
Resultant Max. ZPA Horizontal Ground (Max. ZPA Vertical Ground)	$G_H + 0.53G_V = 0.53$	10CFR72.102(f)	Section 3.4.7.1
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			

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

Type	Criteria	Basis	FSAR Reference
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 & Fabrication 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
Design Weights:			
Max. Empty Cask:			
W/Pool Lid & No Top Lid	140,258 lb. (125-ton HI-TRAC) 99,758 lb. (100-ton HI-TRAC)	R.G. 3.61	Table 3.2.2
W/Top Lid & Transfer Lid	152,636 lb. (125-ton HI-TRAC) 109,470 lb. (100-ton HI-TRAC)	R.G. 3.61	Table 3.2.2

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

Type	Criteria	Basis	FSAR Reference
Max. MPC/HI-TRAC with Yoke (in-pool lift):			
Water Jacket Empty	199,394 lb. (100-ton HI-TRAC)	R.G. 3.61	Table 3.2.4
Water Jacket Full	248,105 lb. (125-ton HI-TRAC)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3
Water Jacket Cavity	60 psig (internal)	ANSI/ANS 57.9	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Section 2.0.3
	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			

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

Type	Criteria	Basis	FSAR Reference
Lead	350°F (max.)	-	Table 2.2.3
Liquid Neutron Shield	307°F (max.)	-	Table 2.2.3
Solid Neutron Shield	300°F (max.)	Manufacturer Data	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.5.1.1.3
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
Retrievability:			
Normal and Off-normal	No encroachment on MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 & 3.4
After Design-basis (Postulated) Accident			Section 3.5 & 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:		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
Design Bases:			

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

Type	Criteria	Basis	FSAR Reference
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperatures:			
Lifetime Average	100°F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load			
Max. Loaded Canister			
Dry	87,241 lb.	R.G. 3.61	Table 3.2.1
Wet	103,898 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 or Horizontal	-	Section 1.2.2.2

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

Type	Criteria	Basis	FSAR Reference
Test Loads:			
Trunnions	300% of vertical design load	NUREG-0612 & ANSI N14.6	Section 9.1.2.1
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	0°F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100°F	ANSI/ANS 57.9	Section 2.2.2.2
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1
Fire			
Duration	4.8 minutes	10CFR72.122(c)	Section 2.2.3.3
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

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

Type	Criteria	Basis	FSAR Reference
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

2.1 SPENT FUEL TO BE STORED

2.1.1 Determination of The Design Basis Fuel

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

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

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

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

In summary, the geometric compatibility of the SNF with the MPC designs does not require the definition of a design basis fuel assembly. This, however, is not the case for structural, confinement, shielding, thermal-hydraulic, and criticality criteria. In fact, a particular fuel type in a category (PWR or BWR) may not control the cask design in all of the above-mentioned criteria. To ensure that no SNF listed in Refs. [2.1.1] and [2.1.2] which is geometrically admissible in the MPC is precluded, it is necessary to determine the governing fuel specification for each analysis criteria. To make the necessary determinations, potential candidate fuel assemblies for each qualification criteria were considered. Table 2.1.1 lists the PWR fuel assemblies which were evaluated. These fuel assemblies were evaluated to define the governing design criteria for PWR fuel. The BWR fuel assembly designs evaluated are listed in Table 2.1.2. Tables 2.1.3 and 2.1.4 provide the fuel characteristics determined to be acceptable for storage in the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 is acceptable for storage in the HI-STORM 100 System. Table 2.1.5 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and decay heat generation. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Intact SNF Specifications

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

Intact fuel assemblies with missing pins cannot be loaded into the HI-STORM 100 unless dummy fuel pins, which occupy a volume greater than or equal to the original fuel pins, replace the missing pins prior to loading. Any intact fuel assembly which falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly, as defined in the Technical Specifications of Chapter 12 can be safely stored in the HI-STORM 100 System.

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

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel assemblies are defined as fuel assemblies with known or suspected cladding defects greater than pinhole leaks and hairline cracks or missing fuel rods that are not replaced with dummy fuel rods, and which may have mechanical damage which would not allow it to be handled by normal means; however, there shall be no loose components. No loose fuel debris is allowed with the damaged fuel assembly.

Fuel debris is defined as fuel assemblies with known or suspected defects greater than pinhole leaks or hairline cracks such as ruptured fuel rods, severed fuel rods, or loose fuel pellets, and which cannot be handled by normal means.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with 250 x 250 fine mesh screens, prior to placement in the HI-STORM 100 System. This application requests approval of Dresden Unit 1 (UO_2 rods and MOX fuel rods) and Humboldt Bay fuel arrays (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) as damaged fuel assembly contents for storage in the MPC-68 and fuel debris as contents for storage in the MPC-68F. The design characteristics bounding Dresden Unit 1 and Humboldt Bay SNF are given in Table 2.1.7. The placement of a single damaged fuel assembly in an MPC-68 or a single fuel debris damaged fuel container in an MPC-68F necessitates that all fuel assemblies (intact, damaged, or debris) stored in that MPC meet the maximum heat generation requirements specified in Table 2.1.7. The fuel characteristics specified in Table 2.1.4 for Dresden 1 and Humboldt Bay fuel arrays have been evaluated in this FSAR and are acceptable for storage as damaged fuel or fuel debris in the HI-STORM 100 System. The DFC design is illustrated in Figure 2.1.1 and the Design Drawings are provided in Section 1.5. Because of the long cooling time, small size, and low weight of spent fuel assemblies qualified as damaged fuel or fuel debris, the DFC and its contents are bounded by the structural, thermal, and shielding analyses performed for the intact BWR design basis fuel. Separate criticality analysis of the bounding fuel assembly for the damaged fuel and fuel debris has been performed in Chapter 6.

2.1.4 Deleted

2.1.5 Structural Parameters for Design Basis SNF

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

2.1.6 Thermal Parameters for Design Basis SNF

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

To ensure the allowable fuel cladding temperature limits are not exceeded, Table 2.0.1 specifies the allowable decay heat per assembly versus cooling time for zircaloy clad fuel in each MPC type. Tables 2.1.7 and 2.1.8 provide the maximum heat generation for damaged zircaloy clad fuel assemblies and stainless steel clad fuel assemblies, respectively. Due to the conservative thermal assessment and the long cooling time of the damaged and stainless steel clad fuel, a reduction in decay heat load is not required as the cooling time increases beyond the minimum specified.

The specified decay heat load can be attained by varying burnups and cooling times. Figure 2.1.6 provides the burnup and cooling time characteristics for intact zircaloy clad fuel to meet the thermal requirements for the MPC-24 and MPC-68. Any intact zircaloy clad fuel assembly with a burnup and cooling time which lies on or below the curve of Figure 2.1.6 is thermally acceptable for loading into the HI-STORM 100 System. Each point on the curve produces a decay heat equal to or below the value specified in Table 2.0.1.

Figure 2.1.6 does not extend beyond 15 years of cooling time. For fuel assemblies with cooling times greater than 15 years, the maximum allowed burnup will be limited to maximum burnup value specified for 15 years. As shown in Figure 2.1.6 the allowable burnup increases as the cooling time increases due to the decay of radioactivity over time. Therefore, limiting the maximum burnup for fuel assemblies with more than 15 years of cooling time to the corresponding burnup value for 15 years ensures that the decay heat load from these older fuel assemblies will be less than the values analyzed in this FSAR.

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

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

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 the maximum burnup, minimum cooling time, and a conservative enrichment corresponding to the burnup. 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. Table 2.1.7 provides the burnup and cooling time values which meet the radiological source term requirements for damaged BWR fuel in the MPC-68 and fuel debris in the MPC-68F. Table 2.1.8 provides the burnup and cooling time values which meet the radiological source term requirements for intact stainless steel clad fuel. Figure 2.1.6 provides illustrative burnup and cooling time values which meet the radiological source term requirements for zircaloy clad fuel in each MPC type.

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.

2.1.8 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68 features a basket without flux traps. In the MPC-68 basket, there is one panel of neutron absorber between two adjacent fuel assemblies. The MPC-24 employs 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 MPC-24 Boral ^{10}B areal density is specified at a minimum loading of 0.0267 g/cm^2. The MPC-68 Boral ^{10}B areal density is specified at a minimum loading of 0.0372 g/cm^2. The MPC-68F Boral ^{10}B areal density is specified at a minimum loading of 0.01 g/cm^2.

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

2.1.9 Summary of SNF Design Criteria

An intact zircaloy clad fuel assembly is acceptable for storage in a HI-STORM 100 System if it fulfills the following criteria:

a. It satisfies the physical characteristics listed in Tables 2.1.3 or 2.1.4, and 2.1.6.

b. Its initial enrichment is less than that indicated by Table 2.1.6 for the fuel assembly and MPC type.

c. The period from discharge is greater than or equal to the minimum cooling time listed in Table 2.1.6, and the decay heat is equal to or less than the maximum value stated in Table 2.0.1 for a given cooling time.

d. The average burnup of the fuel assembly is less than or equal to the burnup specified in Figure 2.1.6 for a given cooling time.

A damaged fuel assembly shall meet the characteristics specified in Table 2.1.7 for storage in the MPC-68. A fuel assembly classified as fuel debris shall meet the characteristics specified in Table 2.1.7 for storage in the MPC-68F.

Stainless steel clad fuel assemblies shall meet the characteristics specified in Table 2.1.8 for storage in the MPC-24 and MPC-68.

MOX BWR fuel assemblies shall meet the requirements of Tables 2.1.6 and 2.1.7 for intact and damaged fuel/fuel debris, respectively.

No PWR control components are to be included with the fuel assembly.

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
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

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 (except 8x8 WE (QUAD+))	All 9x9	All 10x10
Humboldt Bay	All 6x6	All 7x7 (Zircaloy 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 and Class	**14x14 A**	**14x14 B**	**14x14 C**	**14x14 D**	**15x15 A**
Clad Material (Note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.) (Note 4)	≤ 402	≤ 402	≤ 410	≤ 400	≤ 420
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3840	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3770	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	0.556	0.556	0.580	0.556	0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5 (see Note 3)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.040	≥ 0.0145	≥ 0.0165

Notes:
1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. Each guide tube replaces 4 fuel rods.
4. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
5. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array and Class	**15x15 B**	**15x15 C**	**15x15 D**	**15x15 E**	**15x15 F**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (wt % ^{235}U)	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1
No. of Fuel Rods	204	204	208	208	208
Clad O.D. (in.)	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Clad I.D. (in.)	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Pellet Dia. (in.)	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	0.563	0.563	0.568	0.568	0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	21	17	17	17
Guide Tube Thickness (in.)	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
4. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array and Class	**15x15 G**	**16x16 A**	**17x17A**	**17x17 B**	**17x17 C**
Clad Material (Note 2)	SS	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 4)	≤ 420	≤ 430	≤ 450	≤ 464	≤ 460
Initial Enrichment (wt % ^{235}U)	≤ 4.0	≤ 4.6	≤ 4.0	≤ 4.0	≤ 4.0
No. of Fuel Rods	204	236	264	264	264
Clad O.D. (in.)	≥ 0.422	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Clad I.D. (in.)	≤ 0.3890	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Pellet Dia. (in.)	≤ 0.3825	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	0.563	0.506	0.496	0.496	0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	5 (see note 3)	25	25	25
Guide Tube Thickness (in.)	≥ 0.0145	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Notes:
1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. Each guide tube replaces 4 fuel rods.
4. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
5. Description of the fuel assembly class designation is provided in Chapter 6.

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 4)	≤ 108	≤ 108	≤ 108	≤ 100	≤ 195	≤ 120
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for UO_2 rods. See Note 3 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.0	≤ 4.0
No. of Fuel Rods	36	36 (up to 9 MOX rods)	36	49	49	64
Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Clad I.D. (in.)	≤ 0.4945	≤ 0.4945	≤ 0.4990	≤ 0.4200	≤ 0.4990	≤ 0.3620
Pellet Dia. (in.)	≤ 0.4940	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4880	≤ 0.3580
Fuel Rod Pitch (in.)	0.694	0.694	0.740	0.631	0.738	0.523
Active Fuel Length (in.)	≤ 110	≤ 110	≤ 77.5	≤ 79	≤ 150	≤ 110
No. of Water Rods	0	0	0	0	0	0
Water Rod Thickness (in.)	N/A	N/A	N/A	N/A	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Notes:
1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. ≤ 0.612 wt. % ^{235}U and ≤ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu).
4. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
5. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array and Class	8x8 B	8x8 C	8x8 D	8x8 E	9x9 A	9x9 B
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 6)	≤ 185	≤ 185	≤ 185	≤ 180	≤ 173	≤ 173
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	63	62	60	59	74/66 (Note 3)	72
Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4400	≥ 0.4330
Clad I.D. (in.)	≤ 0.4250	≤ 0.4250	≤ 0.4190	≤ 0.4250	≤ 0.3840	≤ 0.3810
Pellet Dia. (in.)	≤ 0.4160	≤ 0.4160	≤ 0.4110	≤ 0.4160	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	0.636 - 0.641	0.636 - 0.641	0.640	0.640	0.566	0.569
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods	1	2	1 - 4 (see note 5)	5	2	1 (see note 4)
Water Rod Thickness (in.)	≥ 0.034	≥ 0.00	≥ 0.00	≥ 0.034	≥ 0.00	≥ 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.120	≤ 0.120

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
4. Square, replacing 9 fuel rods.
5. Variable.
6. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
7. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array and Class	**9x9 C**	**9x9 D**	**9x9 E**	**9x9 F**	**10x10 A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 4)	≤ 173	≤ 170	≤ 170	≤ 170	≤ 182
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note 3)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3590	≤ 0.3810	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3525	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	0.572	0.572	0.572	0.572	0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods	1	2	5	5	2
Water Rod Thickness (in.)	0.020	≥ 0.0305	≥ 0.0305	≥ 0.0305	≥ 0.030
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.100	≤ 0.120

Notes:
1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
4. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
5. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array and Class	10x10 B	10x10 C	10x10 D	10x10 E
Clad Material (Note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 6)	≤ 182	≤ 180	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	91/83 (Note 3)	96	100	96
Clad O.D. (in.)	≥ 0.3957	≥ 0.3790	≥ 0.3960	≥ 0.3940
Clad I.D. (in.)	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Pellet Dia. (in.)	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	0.510	0.488	0.565	0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods	1 (Note 4)	5 (Note 5)	0	4
Water Rod Thickness (in.)	≥ 0.00	≥ 0.034	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made of Zirconium or Zirconium alloys.
3. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
4. Square, replacing nine fuel rods.
5. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
6. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user.
7. Description of the fuel assembly class designation is provided in Chapter 6.

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

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

Table 2.1.6

CHARACTERISTICS FOR DESIGN BASIS INTACT ZIRCALOY CLAD FUEL ASSEMBLIES

	MPC-68	MPC-24
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	5.85	8.54
Max. assembly length† (in.)	176.2	176.8
Max. assembly weight†† (lb.)	700	1680
Max. active fuel length† (in.)	150	150
Fuel rod clad material	zircaloy	zircaloy
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
	MPC-68	**MPC-24**
Max. initial enrichment (wt% ^{235}U)	See Table 2.1.4	See Table 2.1.3
Max. heat generation (W)	Table 2.0.1 115 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A)	Table 2.0.1
Max. average burnup (MWD/MTU)	See Figure 2.1.6 30,000 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A)	See Figure 2.1.6
Min. cooling time (years)	See Figure 2.1.6 18 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A)	See Figure 2.1.6

† Unirradiated nominal design dimensions are shown.

†† Fuel assembly weight including hardware based on DOE MPC DPS [2.1.6].

Table 2.1.7
DESIGN CHARACTERISTICS FOR DAMAGED ZIRCALOY CLAD FUEL ASSEMBLIES

	MPC-68 (Damaged Fuel)	MPC-68F (Fuel Debris)
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	4.7	4.7
Max. assembly length† (in.)	135	135
Max. assembly weight†† (lb.)	400	400
Max. active fuel length† (in.)	110	110
Fuel rod clad material	zircaloy	zircaloy
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
Max. heat generation (W)	115	115
Min. cooling time (yr)	18	18
Max. initial enrichment (wt.% ^{235}U) for UO_2 rods	2.7	2.7
Max. initial enrichment for MOX rods	0.612 wt.% ^{235}U 1.578 wt. % Total Fissile Plutonium	0.612 wt.% ^{235}U 1.578 wt. % Total Fissile Plutonium
Max. average burnup (MWD/MTU)	30,000	30,000

Note:

1. A maximum of four (4) damaged fuel containers with BWR zircaloy clad fuel debris may be stored in the MPC-68F with the remaining locations filled with undamaged or damaged fuel assemblies meeting the maximum heat generation specifications of this table.

† Dimensions envelop unirradiated nominal dimensions of Array/Class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A (Dresden Unit 1 and Humboldt Bay SNF)

†† Fuel assembly weight including hardware based on DOE MPC DPS [2.1.6]. Weight does not include damaged fuel container.

Table 2.1.8

DESIGN CHARACTERISTICS FOR INTACT STAINLESS STEEL CLAD FUEL ASSEMBLIES

	BWR MPC-68	PWR MPC-24
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	5.62	8.42
Max. assembly length† (in.)	102.5	138.8
Max. assembly weight†† (lb.)	400	1421
Max. active fuel length† (in.)	83	122
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
Max. heat generation (W)	95	710
Min. cooling time (yr)	10	8
Max. initial enrichment (wt.% ^{235}U)	4.0	4.0
Max. average burnup (MWD/MTU)	22,500	40,000

† Unirradiated nominal dimensions are shown.

†† Fuel assembly weight including hardware based on DOE MPC DPS [2.1.6].

Table 2.1.9

PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	**Assembly Length w/o C.C.† (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

† C.C. is an abbreviation for Control Components.

Table 2.1.10

BWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel for Reactor 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	23.6
Humboldt Bay	95.0	8.0	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	144.5†	11.2	110	17.0	14.5
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

† Fuel assembly length includes the damaged fuel container.

Table 2.1.11
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION†		
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††		
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

† Reference 2.1.7

†† Reference 2.1.8

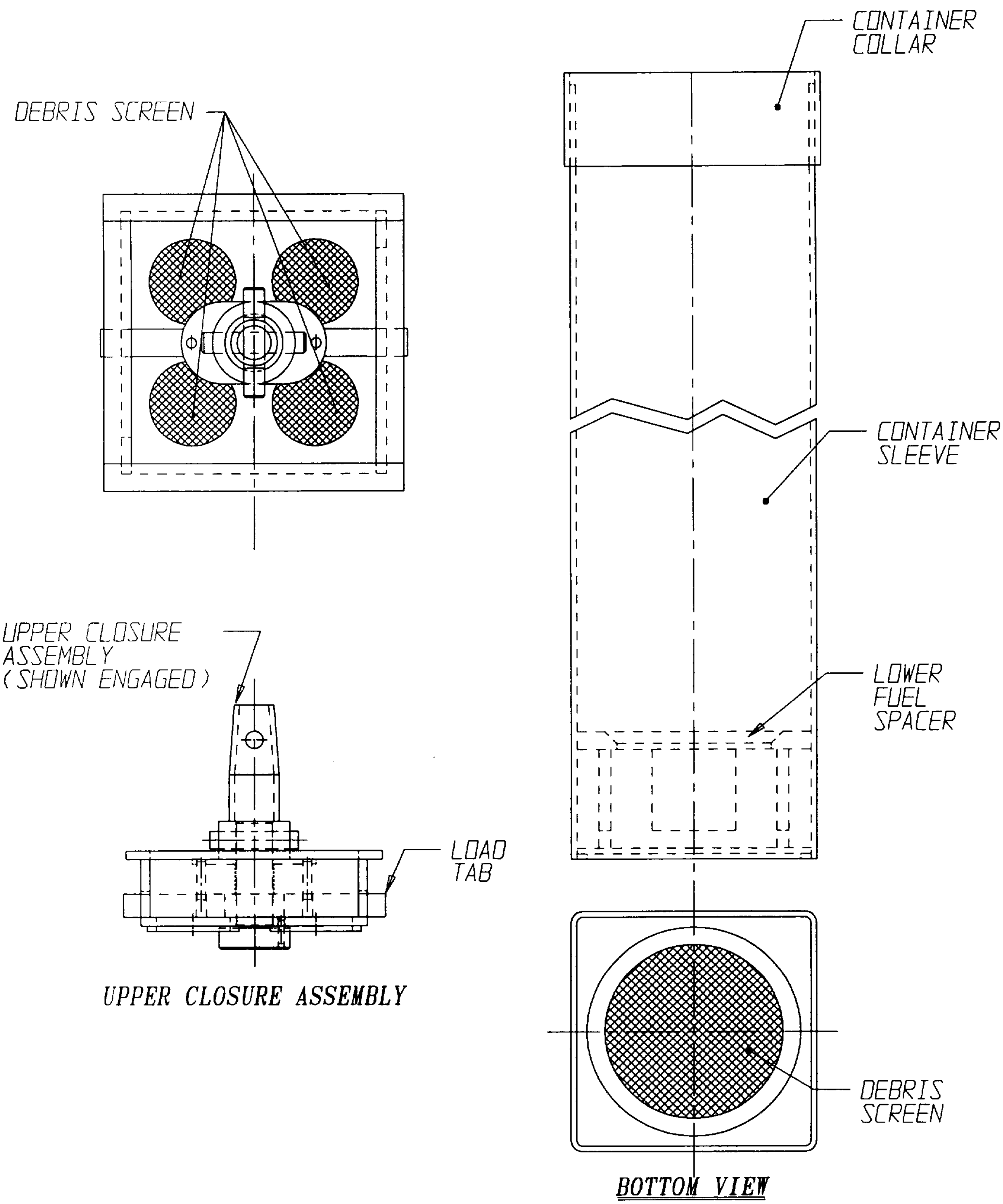

FIGURE 2.1.1; DAMAGED FUEL CONTAINER FOR DRESDEN UNIT-1/ HUMBOLT BAY SNF

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FIGURE 2.1.2; DELETED

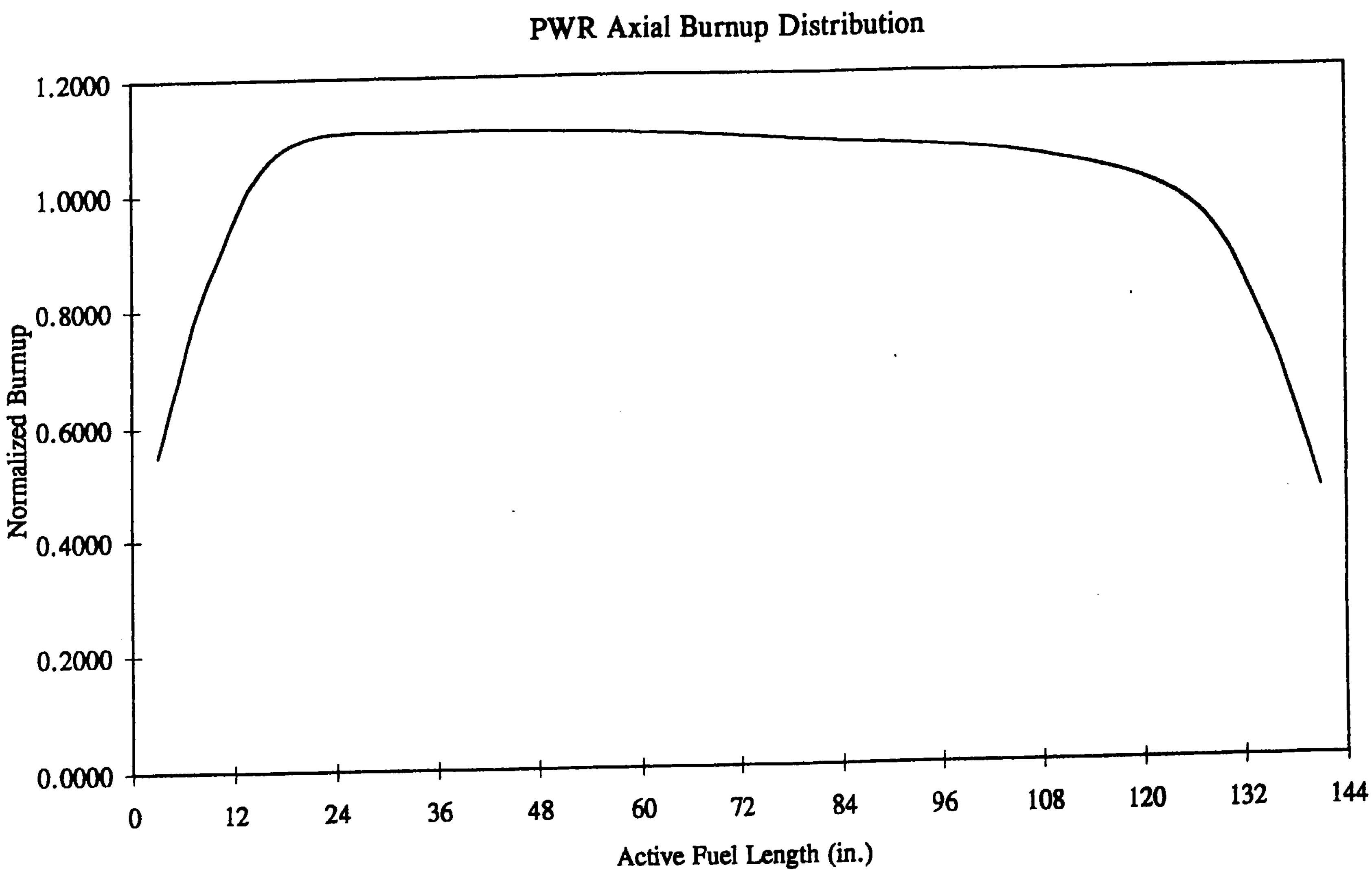

Figure 2.1.3; PWR Axial Burnup Profile with Normalized Distribution

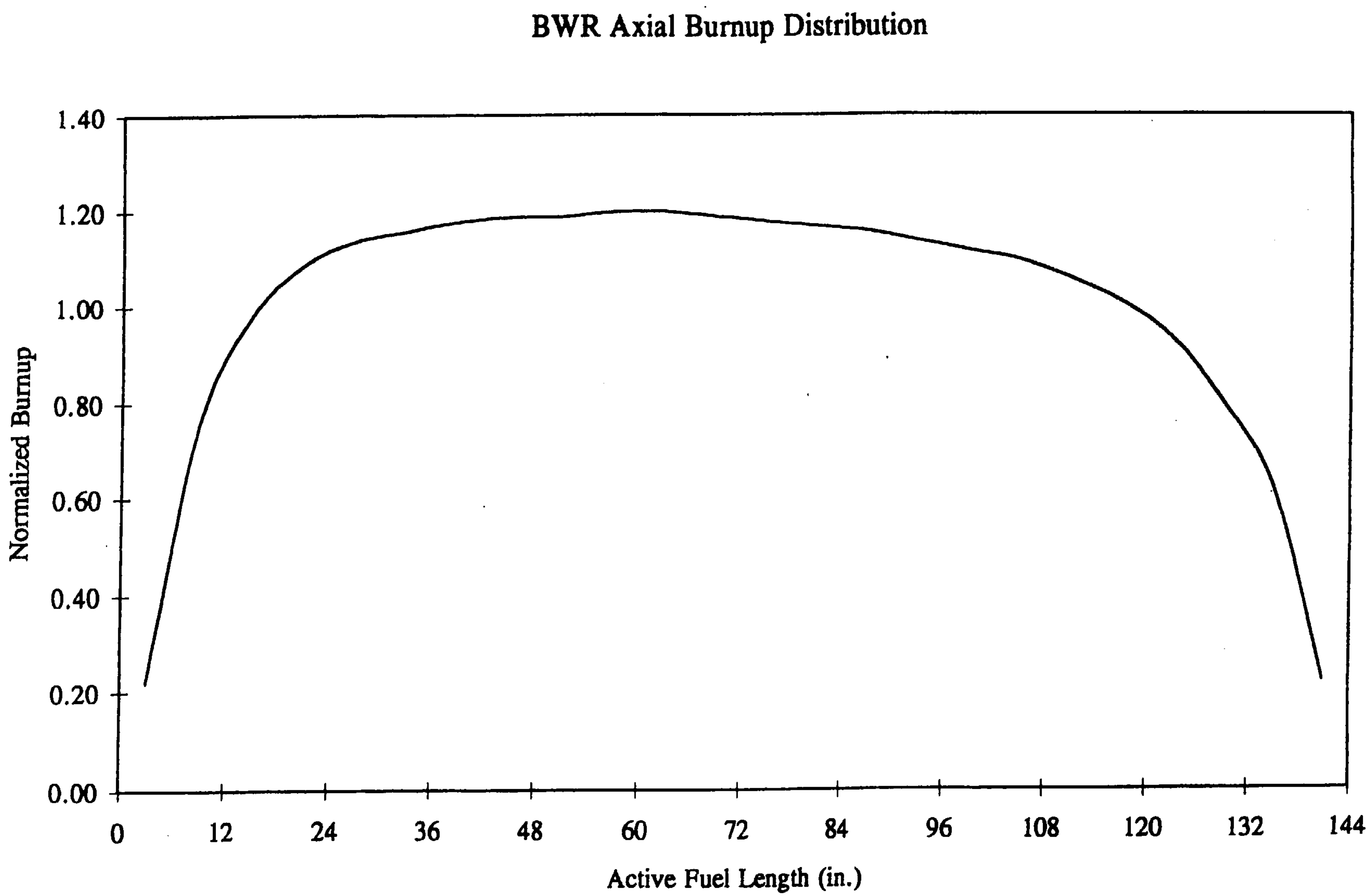

Figure 2.1.4; BWR Axial Burnup Profile with Normalized Distribution

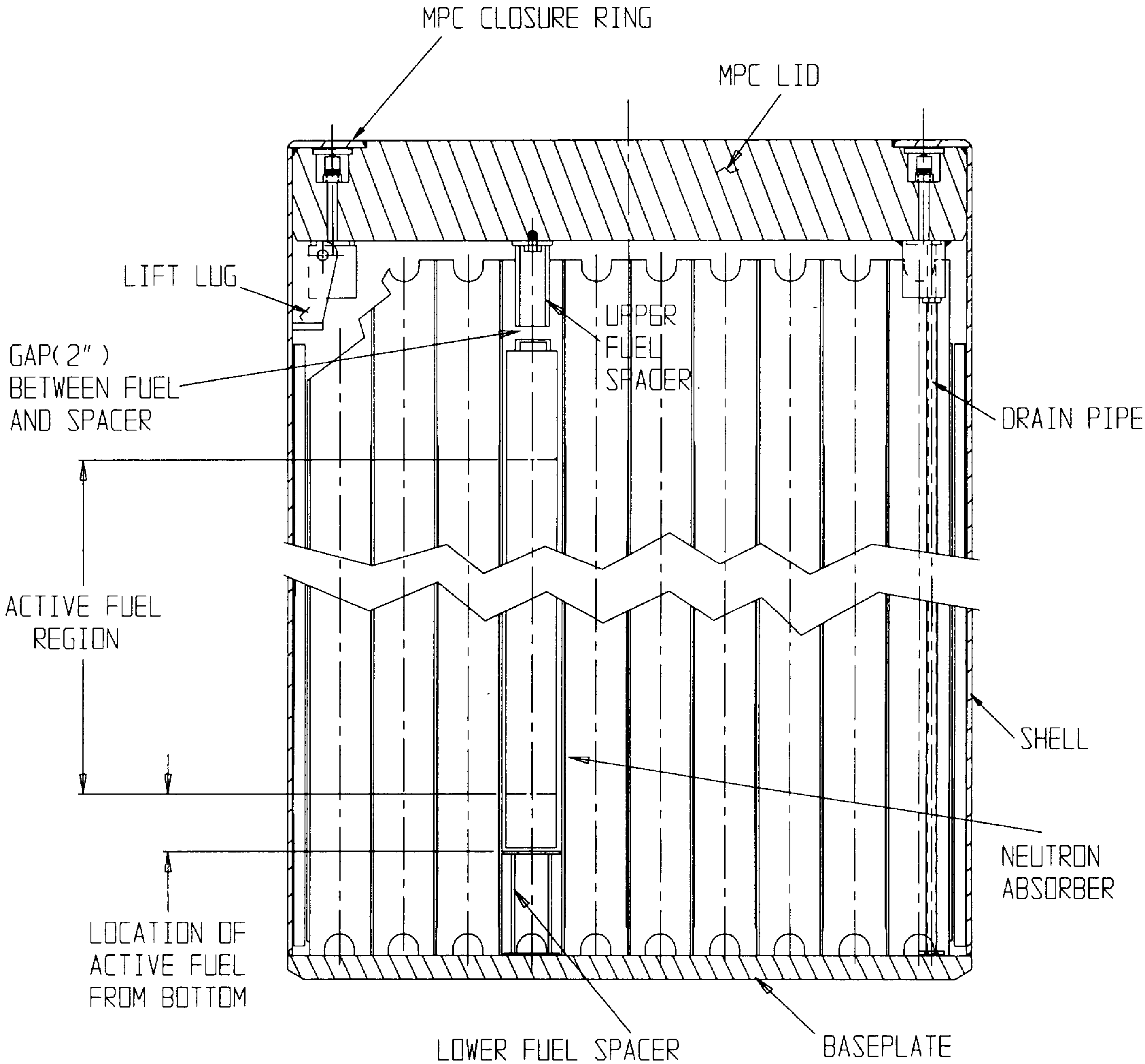

FIGURE 2.1.5; MPC WITH UPPER AND LOWER FUEL SPACERS

ILLUSTRATIVE BURNUP AND COOLING TIME FOR DECAY HEAT AND RADIATION SOURCE TERMS

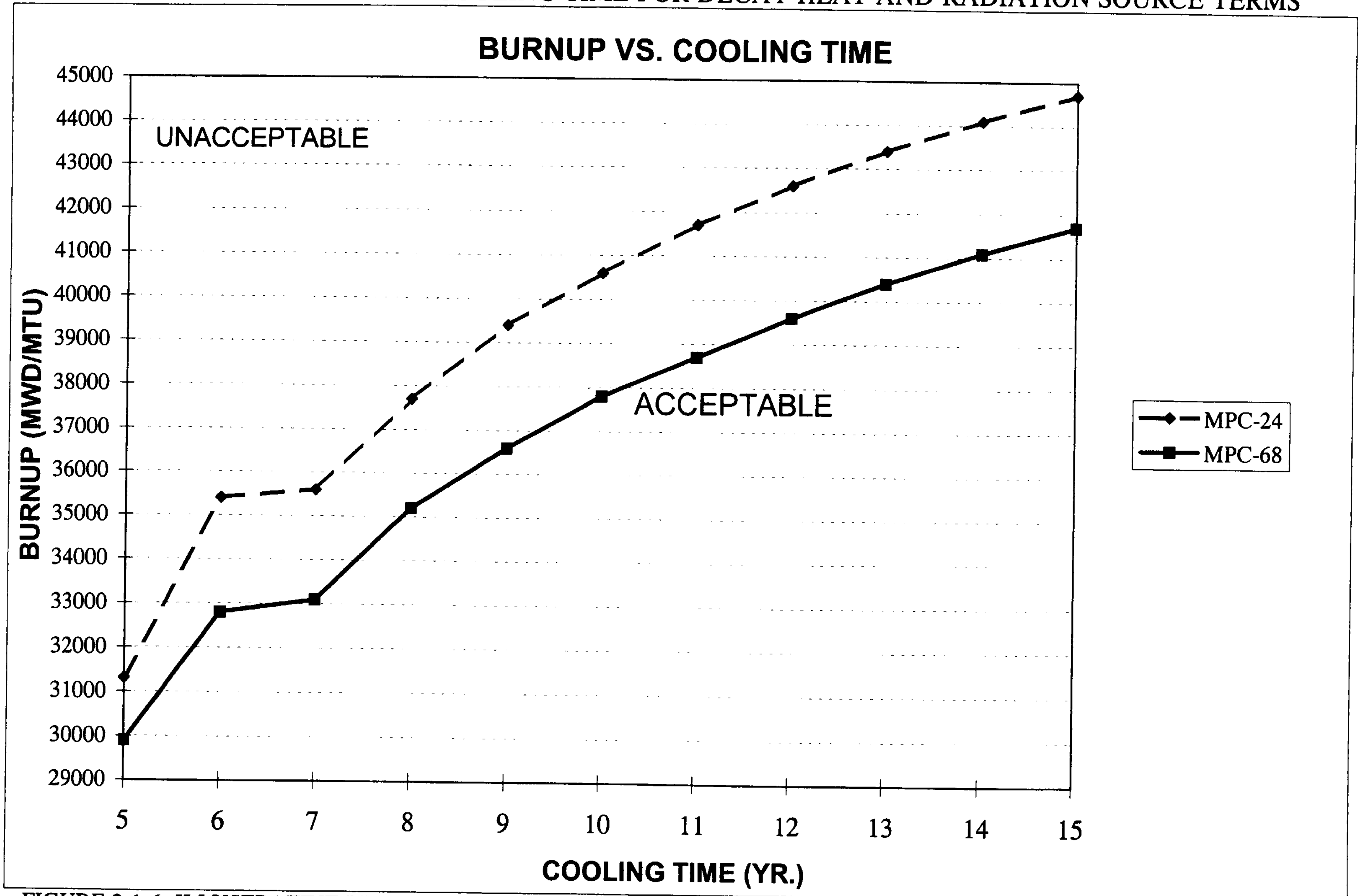

FIGURE 2.1.6: ILLUSTRATIVE BURNUP AND COOLING TIME FOR DECAY HEAT AND RADIATION SOURCE TERMS

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FIGURE 2.1.7; DELETED

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FIGURE 2.1.8; DELETED

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 which 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 Condition: Dead Weight, Handling, Pressure, Temperature, Snow

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

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

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 100 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 100 Overpack is limited to ambient temperatures above 0EF. 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 100 Overpack for use during handling operations at a minimum service temperature of 0° F.

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

2.2.1.3 Pressure

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

The normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 1% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) released in accordance with NUREG-1536.

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

† 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.

For the storage of damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies or fuel debris (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) 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. This condition is bounded by the pressure calculation for design basis intact fuel with 100% of the fuel rods ruptured in all 68 of the BWR fuel assemblies. It is shown in Chapter 4 that the normal condition design pressure is not exceeded with 100% of the fuel rods ruptured in all 68 of the design basis BWR 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 normal design pressure.

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

The HI-STORM 100 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 100 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 which demonstrates that the design pressure in Table 2.2.1 can be withstood by the water jacket and Chapter 4 demonstrates by analysis that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device set at the design pressure is provided, which ensures the pressure will not be exceeded.

2.2.1.4 Environmental Temperatures

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

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

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

2.2.1.5 Design Temperatures

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

Maintaining fuel rod cladding integrity is also a design consideration. The maximum fuel rod cladding temperature limits for normal conditions are calculated by the DCCG (Diffusion Controlled Cavity Growth) methodology outlined in the LLNL report [2.2.14] in accordance with NUREG-1536. However, for conservatism, the PNL methodology outlined in PNL report [2.0.3] produces a lower fuel cladding temperature, which is used to establish the *permissible* fuel cladding temperature limits, which are used to determine the allowable fuel decay heat load. Maximum fuel rod stainless steel cladding temperature limits recommended in EPRI report [2.2.13] are greater than the long-term allowable zircaloy fuel cladding temperature limits. However, in this FSAR the long-term zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel clad fuel. The short term temperature limits for zircaloy and stainless steel cladding are taken from references [2.2.15] and [2.2.13], respectively. A detailed description of the maximum fuel rod cladding temperature limits determination is provided in Section 4.3.

2.2.1.6 Snow and Ice

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

2.2.2 Off-Normal Conditions Design Criteria

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

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

2.2.2.1 Pressure

The HI-STORM 100 System must withstand loads due to off-normal pressure. The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H^3, Kr, and Xe) released in accordance with NUREG-1536. For conservatism, the MPC normal internal design pressure bounds both normal and off-normal conditions. Therefore, the normal and off-normal condition MPC internal pressures are set equal for analysis purposes.

2.2.2.2 Environmental Temperatures

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

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

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

2.2.2.3 Design Temperatures

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

2.2.2.4 Leakage of One Seal

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

The confinement boundary is defined by the MPC shell, baseplate, MPC lid, port cover plates, and closure ring. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root and final pass. In addition to liquid penetrant examination, the MPC lid-to-shell weld is leakage tested, hydrostatic tested, and volumetrically examined or multi-pass liquid penetrant examined. The vent and drain port cover plates are leakage tested in addition to the liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary. Although leakage of one seal is not a credible accident, it is analyzed in Chapter 11.

2.2.2.5 Partial Blockage of Air Inlets

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

2.2.2.6 Off-Normal HI-TRAC Handling

During upending and/or downending of the HI-TRAC 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.

If the lifting device is allowed to "go slack", the total weight would be applied to the lower pocket trunnions only. Under this off-normal condition, the pocket trunnions would each be required to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to demonstrate that the pocket trunnions possess sufficient strength to support the increased load under this off-normal condition.

2.2.3 Environmental Phenomena and Accident Condition Design Criteria

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

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

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

2.2.3.1 Handling Accident

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

The loaded HI-STORM 100 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 MPC must be maintained below 60g's under axial loading to ensure the analysis performed in the HI-STAR Safety Analysis Reports [2.2.4 and 2.2.5] bounds the HI-STORM 100 Overpack vertical handling accident. Additionally, the overpack must continue to suitably shield the radiation emitted from the loaded MPC. The use of lifting equipment with redundant drop protection and lifting devices designed in accordance with the requirements specified in Section 2.3.3.1 to lift the loaded overpack will eliminate the lift height limit. The lift height limit is dependent on the characteristics of the impacting surface which are specified in Table 2.2.9. For site-specific conditions, which are not encompassed by Table 2.2.9, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the analyses in Appendix 3.A and shall be reviewed by the Certificate Holder.

The loaded HI-TRAC will be lifted so that the side of the cask 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 which 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) remain attached. Furthermore, the HI-TRAC inner shell is demonstrated by analysis to not deform sufficiently to affect retrieval of the MPC. The horizontal lift height limit is dependent on the characteristics of the impacting surface which are specified in Table 2.2.9. For site-specific conditions, which are not encompassed by Table 2.2.9, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the analyses in Appendix 3.AN and shall be reviewed by the Certificate Holder. The use, during horizontal lifting of the loaded HI-TRAC outside of the reactor facilities, of lifting equipment with redundant drop protection and lifting devices designed in accordance with the requirements specified in Section 2.3.3.1, will eliminate the need for a horizontal lift height limit.

The loaded HI-TRAC, when lifted in the vertical position outside of the reactor facility shall be lifted by lifting equipment with redundant drop protection features and lifting devices designed in accordance with ANSI N14.6. Therefore, a vertical drop or tip-over is not a credible accident for the HI-TRAC transfer cask and no vertical lift height limit is provided. Likewise, while the loaded HI-TRAC is positioned atop the HI-STORM 100 Overpack for transfer of the MPC into the overpack, 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. This condition of use for the MPC transfer operations from the HI-TRAC transfer cask to the HI-STORM 100 Overpack is specified in the Technical Specifications in Chapter 12 and Subsection 2.3.3.1, and is included in the operating procedures of Chapter 8.

The loaded MPC is lowered into the HI-STORM or HI-STAR Overpacks or raised from the overpacks using the HI-TRAC transfer cask and a MPC lifting system designed to be single failure proof and lifting devices designed in accordance with ANSI N14.6. 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. The concrete pad design for storing the HI-STORM 100 System shall comply with Table 2.2.9 and shall be 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 those calculated by the dynamic models used in the structural qualifications.

2.2.3.2 Tip-Over

The 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 requirements are specified in Table 2.2.9.

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 100 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 100 Overpack and HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Table 2.2.8 provides the fire durations for the HI-STORM 100 Overpack and HI-TRAC transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475°F in accordance with 10CFR71.73.

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

2.2.3.4 Partial Blockage of MPC Basket Vent Holes

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

2.2.3.5 Tornado

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

The kinematic stability of the HI-STORM 100 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 which 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, tornado missile shields shall be placed at either end of the HI-TRAC to prevent tornado missiles from impacting either end of the HI-TRAC. The tornado missile shield 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 shield positioned to protect 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 shields 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 and the resultant differential pressure on the overpack is stated in Table 2.2.1. 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 100 Overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the HI-STORM 100 Overpack. Blockage of the air inlets is addressed in Subsection 2.2.3.12.

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

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

2.2.3.7 Seismic Design Loadings

The HI-STORM 100 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

† 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.

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 occur simultaneously. The maximum horizontal ZPA provided in Table 2.2.8 is the vector sum of two horizontal earthquakes.

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.

2.2.3.9 Confinement Boundary Leakage

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

2.2.3.10 Explosion

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

2.2.3.11 Lightning

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

2.2.3.12 Burial Under Debris

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

2.2.3.13 100% Blockage of Air Inlets

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

2.2.3.14 Extreme Environmental Temperature

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

2.2.4 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. The MPC enclosure vessel and fuel basket are designed in accordance with Section III, Subsections NB Class 1 and NG Class 1, respectively. The metal structure of the overpack and the HI-TRAC transfer cask are designed in accordance with Section III, Subsection NF Class 3. The ASME Code is applied to each component consistent with the function of the component.

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

Table 2.2.6 provides a summary of each structure, system and component (SSC) of the HI-STORM 100 System which is identified as important to safety, along with its function and governing Code. Some components perform multiple functions and in those cases, the most restrictive Code is applied. In accordance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components", and according to importance to safety, components of the HI-STORM 100 System are classified as A, B, C, or NITS (not important to safety) in Table 2.2.6. Section 13.1 provides the criteria used to classify each item. The classification of necessary auxiliary equipment is provided in Table 8.1.6.

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

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

The MPC utilized in the HI-STORM 100 System is identical to the MPC described in the applications for the HI-STAR 100 System for storage (Docket 72-1008) and transport (Docket 71-9261) certification. To avoid unnecessary repetition of the large numbers of stress analyses, attention is directed in this document to establish that the MPC loadings for storage in the HI-STORM 100 System do not exceed those computed in the referenced applications. 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 .

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 100 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 which primarily affect kinematic stability, and (ii) those which produce significant stresses. The loadings in the former category are principally applicable to the overpack. Wind (W), earthquake (E), tornado (W'), 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 which demonstrate that the overpack structure remains unbreached by the postulated missiles are provided in Chapter 3.

Loadings in the second category produce global stresses which 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 100 Overpack are different because of differences in their function. For example, the fuel basket does not experience a pressure loading because it is not a pressure vessel. The specific load combination for each component is specified in Subsection 2.2.7.

2.2.7 Load Combinations

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

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

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

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

Finally, it should be noted that the load combinations identified in NUREG-1536 are considered as applicable to the HI-STORM 100 System. The majority of load combinations in NUREG-1536 are directed towards reinforced concrete structures. Those load combinations applicable to steel structures are directed towards 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 detach. The following definitions of terms apply to the tables on stress intensity limits; these definitions are the same as those used throughout the ASME Code:

S_m: Value of Design Stress Intensity listed in ASME Code Section II, Part D, Tables 2A, 2B and 4

S_y: Minimum yield strength at temperature

S_u: Minimum ultimate strength at temperature

Table 2.2.1

DESIGN PRESSURES

Pressure Location	Condition	Pressure (psig)
MPC Internal Pressure	Normal	100
	Off-Normal	100
	Accident	125
MPC External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	60
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	10 Differential Pressure for 1 second maximum or 5 Differential Pressure steady state
HI-TRAC Water Jacket	Normal	60
	Off-normal	60
	Accident	N/A (Under accident conditions, the water jacket is assumed to have lost all water thru the pressure relief valves)

Table 2.2.2

ENVIRONMENTAL TEMPERATURES

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

Note:

1. Handling operations with the loaded HI-STORM 100 Overpack and HI-TRAC transfer cask are limited to environmental temperatures greater than 0° F as specified in Subsection 2.2.1.2 and the Technical Specifications in Chapter 12.

Table 2.2.3

DESIGN TEMPERATURES

HI-STORM 100 Component	Normal Condition Design Temp. (Long-Term Events) (°F)	Off-Normal and Accident Condition Temp.[1] Limits (Short-Term Events) (°F)
MPC shell	450	775
MPC basket	725	950
MPC Boral	800	950
MPC lid	550	775
MPC closure ring	400	775
MPC baseplate	400	775
MPC Heat Conduction Elements	725	950
HI-TRAC inner shell	400	600
HI-TRAC pool lid/transfer lid	350	700
HI-TRAC top lid	400	700
HI-TRAC top flange	400	700
HI-TRAC pool lid seals	350	N/A
HI-TRAC bottom lid bolts	350	700
HI-TRAC bottom flange	350	700
HI-TRAC top lid neutron shielding	300	300
HI-TRAC radial neutron shield	307	N/A
HI-TRAC radial lead gamma shield	350	600
Remainder of HI-TRAC	350	700
Zircaloy fuel cladding (5-year cooled)	692 (PWR) 742 (BWR)	1058
Zircaloy fuel cladding (6-year cooled)	677 (PWR) 714 (BWR)	1058

Table 2.2.3 (continued)

DESIGN TEMPERATURES

HI-STORM 100 Component	Normal Condition Design Temp. (Long-Term Events) (°F)	Off-Normal and Accident Condition Temp.[1] Limits (Short-Term Events) (°F)
Zircaloy fuel cladding (7-year cooled)	636 (PWR) 671 (BWR)	1058
Zircaloy fuel cladding (10-year cooled)	626 (PWR) 660(BWR)	1058
Zircaloy fuel cladding (15-year cooled)	615(PWR) 648 (BWR)	1058
Overpack outer shell	350	600
Overpack concrete	200	350
Overpack inner shell	350	400
Overpack Lid Top Plate	350	550
Remainder of overpack steel structure	350	400

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)
Helium Retention	Shell	A	ASME Section III; Subsection NB	Alloy X [5]	See Appendix 1.A	NA	NA
Helium Retention	Baseplate	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Helium Retention	Lid	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Helium Retention	Closure Ring	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Helium Retention	Port Cover Plates	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Basket Cell Plates	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Boral	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	Alloy X	See Appendix 1.A	NA	NA
Heat Transfer	Heat Conduction Elements	B	Non-code	Aluminum; Alloy 1100	NA	Sandblast Specified Surfaces	Aluminum/SS

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in Chapter 13. NITS stands for Not Important To Safety.
5) For details on Alloy X material, see Appendix 1.A.

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 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	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Angled Plates)	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Flat Plates)	B	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug Baseplate	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Bolt	NITS	Non-code	A193-B8	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)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer End Plate	B	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 Chapter 13. NITS stands for Not Important To Safety.
5) For details on Alloy X material, see Appendix 1.A.

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	Vent Shield Block Spacer	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Vent and Drain Tube	C	Non-code	304 S/S	Per ASME Section II	Thread area surface hardened	NA
Operations	Vent & Drain Cap	C	Non-code	304 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
Operations	Damaged Fuel Container	C	ASME Section III; Subsection NG	Primarily 304 S/S	See Appendix 1.A	NA	NA

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in Chapter 13. NITS stands for Not Important To Safety.
5) For details on Alloy X material, see Appendix 1.A.

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	4	NA	NA
Shielding	Shield Block Ring & Shell	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Shielding	Pedestal Shield	B	ACI 349, App. 1-D	Concrete	4	NA	NA
Shielding	Lid Shield	B	ACI 349, App. 1-D	Concrete	4	NA	NA
Shielding	Shield Shell	B	See Note 6	SA516-70	See Table 3.3.2	NA	NA
Shielding	Shield Block	B	ACI 349, App. 1-D	Concrete	4	NA	NA
Shielding	Gamma Shield Cross Plates & Tabs	C	Non-code	SA240-304	NA	NA	NA
Structural Integrity	Baseplate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.3	See Note 5	NA
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Concrete Form	Pedestal Shell	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
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 &	B	ASME Section III;	SA516-70	See Table 3.3.2	See Note 5	NA

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in Chapter 13. NITS stands for Not Important To Safety.
5) All exposed steel surfaces (except threaded holes) to be painted with Carboline 890.
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)
	Horizontal Plates		Subsection NF				
Thermal	Exit Vent Vertical & Horizontal Plates	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	
Structural Integrity	Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	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	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 III; Subsection NF	SA564-630	See Table 3.3.4	Threads to have cadmium coating (or similar)	NA
Structural Integrity	Bolt Anchor Block	A	ASME Section III; Subsection NF ANSI N14.6	SA350-LF3 Or SA203E	See Table 3.3.3	See Note 5	NA
Structural Integrity	Channel	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Pedestal Platform	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Storage Marking	NITS	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 Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in Chapter 13. NITS stands for Not Important To Safety.
5) All exposed steel surfaces (except threaded holes) to be painted with Carboline 890.
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)
	Nameplate						
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	Thermocouple & Associated Temperature Monitoring Equipment	B	Non-code	NA	NA	NA	NA
Operations	Screen	NITS	Non-code	Mesh Wire	NA	NA	NA
Operations	Paint	C	Non-code	Carboline 890	NA	NA	NA

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in Chapter 13. NITS stands for Not Important To Safety.
5) All exposed steel surfaces (except threaded holes) to be painted with Carboline 890.
6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

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

Primary Function	Component (3)	Safety Class (4)	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Shielding	Radial Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Pool Lid Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Top Lid Shielding	B	Non-code	Holtite	NA	NA	NA
Shielding	Plugs for Lifting Holes	NITS	Non-code	C/S	NA	NA	
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial Channels	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Enclosure Shell Panels	B	ASME Section III; Subsection NF	SA516-70	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 (SA203E)	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
Structural Integrity	Bottom Flange	B	ASME Section III; Subsection NF	SA350-LF3 (SA516-70))	See Table 3.3.3 (Table 3.3.2)	See Note 5	NA
Structural Integrity	Pool Lid Outer Ring	B	ASME Section III; Subsection NF	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Pool Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA

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

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK [1,2]

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

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

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Pool Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Lifting Block	C	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Thermal Expansion Foam	NITS	Non-code	Silicone or similar	NA	NA	NA
Operations	Top Lid Stud	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	SA193-2H	NA	NA	NA
Operations	Pool Lid Gasket	NITS	Non-code	Elastomer	NA	NA	NA
Operations	Pool Lid & Top Lid Tongues	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	NA	NA
Operations	Lifting Trunnion Locking Pad	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Lifting Trunnion End Cap	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	End Cap Bolts	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	Lifting Trunnion Pad Bolt	NITS	Non-code	SA193-B7	See Table 3.3.4	NA	NA
Operations	Couplings, Valves and Vent Plug	NITS	Non-code	Steel or Bronze	NA	NA	Bronze/Steel

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

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM

HI-TRAC TRANSFER LID [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 Chapter 13. NITS stands for Not Important To Safety.
5) All exterior surfaces to be painted with Carboline 890. Top surface of doors to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM

HI-TRAC TRANSFER LID [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-36	36 (yield)	See Note 5	NA
Structural Integrity	Shaft Cover Plate	C	Non-code	SA516-70	See Table 3.3.2	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	NA	See Note 5	NA
Operations	Door Wheels	NITS	Non-code	Forged Steel	NA	NA	NA
Operations	Door Stop Block	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Door Stop Block Bolt	C	Non-code	SA193-B7	See Table 3.3.4	NA	NA

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

Table 2.2.7

HI-STORM 100 ASME BOILER AND PRESSURE VESSEL CODE APPLICABILITY

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

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)
Cask Environment During the Postulated Fire Event (°F)	Accident	1475
HI-STORM 100 Overpack Fire Duration (seconds)	Accident	217
HI-TRAC Transfer Cask Fire Duration (minutes)	Accident	4.8
Maximum submergence depth due to flood (ft)	Accident	125
Flood water velocity (ft/s)	Accident	15
Interaction Relation for Horizontal & Vertical ZPA (Zero Period Acceleration) for HI-STORM†	Accident	$G_H + 0.53G_V = 0.53$††
HI-STORM 100 Overpack Vertical Lift Height Limit (in.)	Accident	11
HI-TRAC Transfer Cask Horizontal Lift Height Limit (in.)	Accident	42

† The maximum horizontal ZPA is specified as the vector sum of the ZPA g-loading in two orthogonal directions.

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

Table 2.2.9

CHARACTERISTICS OF REFERENCE ISFSI PAD†

Concrete thickness	≤ 36 inches
Concrete Compressive Strength	≤ 4,200 psi at 28 days
Reinforcement Top and Bottom (both directions)	Reinforcing bar shall be 60 ksi Yield Strength ASTM Material
Subgrade Soil Effective Modulus of Elasticity††	≤ 28,000 psi (measured prior to ISFSI pad installation)
Top Concrete Surface	A static coefficient of friction of ≥ 0.53 between the ISFSI pad and the bottom of the overpack shall be verified by test. The test procedure shall follow the guidelines included in the Sliding Analysis in Subsection 3.4.7.1

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

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

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

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

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

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

††† No specific stress intensity 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	TN
Off-Normal HI-TRAC Handling	HN

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

Table 2.2.13 (continued)

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

ACCIDENT CONDITIONS	
LOADING	**NOTATION**
Handling Accident	HN
Earthquake	E
Fire	T^*
Tornado Missile	M
Tornado Wind	WN
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††† (HN pocket trunnion)
	2	D , T' , H , P_o	N/A	N/A
Accident (Level D)	1	D , T , P_i , HN	D , T , HN	D , T , HN
	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 Pi (water jacket) is 60 psig and, therefore, there is no off-normal temperature or load combination because Load Case 1, Normal (Level A), is identical to Load Case 1 Off-normal (Level B). Only the off-normal handling load on the pocket trunnion is analyzed separately.

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

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

Table 2.2.15

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

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

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Lid Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT examination alone is used, at a minimum, it will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be hydrostatically tested as defined in Chapter 8. Accessibility for leakage inspections preclude a Code compliant hydrostatic test. All MPC vessel welds (except closure ring and vent/drain cover plate) are inspected by RT or UT. The vent/drain cover plate welds are confirmed by helium leakage testing and liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant.

Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	HI-STORM 100 System to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
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 EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System will be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package will be in conformance with Holtec's QA program.
Overpack 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.

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
Overpack Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member thickness of 1" or greater is loaded to transmit loads in the through thickness direction.	The large margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large and warrant an exemption. The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have a minimum safety factor of greater than 12 during lifting. The top lid plate to lid shell weld has a safety factor greater than 6 under 45g's.
Overpack Steel Structure	NF-3256	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 prequalified 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.

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.
Handling Load (accident)	HI-STORM vertical and horizontal deceleration loadings are less than those in HI-STAR, but the HI-STORM cavity inner diameter is different and therefore the horizontal loading on the MPC is analyzed in Chapter 3.

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

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

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

1. The MPC confinement barrier is an enclosure vessel designed in accordance with the ASME Code, Subsection NB with confinement welds inspected by radiography (RT) or ultrasonic testing (UT). Where RT or UT is not possible, a redundant closure system is provided with field welds which are hydrostatically tested, helium leakage tested and inspected by the liquid penetrant method.

2. The MPC confinement barrier is surrounded by the HI-STORM overpack which provides for the physical protection of the MPC.

3. The HI-STORM 100 System is designed to meet the requirements of storage while maintaining the safety of the SNF.

4. The SNF once initially loaded in the MPC does not require opening of the canister for repackaging to transport the SNF.

5. The decay heat emitted by the SNF is rejected from the HI-STORM 100 System through passive means. No active cooling systems are employed.

It is recognized that a rugged design with large safety margins is essential, but that is not sufficient to ensure acceptable performance over the service life of any system. A carefully planned oversight and surveillance plan which does not diminish system integrity but provides reliable information on the effect of passage of time on the performance of the system is essential. Such a surveillance and performance assay program will be developed to be compatible with the specific conditions of the licensee's facility where the HI-STORM 100 System is installed. The general requirements for the acceptance testing and maintenance programs are provided in Chapter 9. Surveillance requirements are specified in Chapter 12.

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.

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 to the maximum extent practical, the MPC confinement boundary provides assurance that there will be no release of radioactive materials from the cask under all postulated loading conditions. Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity. Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7. MPC field weld examinations, hydrostatic testing, and helium leak testing are performed to verify the confinement function in accordance with the Technical Specifications contained in Chapter 12. Fabrication inspections and tests are also performed, as discussed in Chapter 9, to verify the confinement boundary.

2.3.2.2 Cask Cooling

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

- The heat rejection capacity of the HI-STORM 100 System is deliberately understated by conservatively determining the design basis fuel. The decay heat value in Table 2.1.6 is developed by computing the decay heat from the design basis fuel assembly which produces the highest heat generation rate for a given burnup. Additional margin is built into the calculated cask cooling rate by using a design basis fuel assembly which offers maximum resistance to the transmission of heat (minimum thermal conductivity).

- The MPC fuel basket is formed by a honeycomb structure of stainless steel plates with full-length edge-welded intersections, which allows the unimpaired conduction of heat.

- The MPC confinement boundary ensures that the helium atmosphere inside the MPC is maintained during normal, off-normal, and accident conditions of storage and transfer. The MPC confinement boundary maintains the helium confinement atmosphere below the design temperatures and pressures stated in Table 2.2.3 and Table 2.2.1, respectively.

- The MPC thermal design maintains the fuel rod cladding temperatures below the values stated in Chapter 4 such that fuel cladding is not degraded during the long term storage period.

- The HI-STORM is optimally designed with cooling vents and a MPC to overpack annulus which maximize air flow, while providing superior radiation shielding. The vents and annulus allow cooling air to circulate past the MPC removing the decay heat.

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Equipment

Design criteria for the HI-STORM 100 System are described in Section 2.2. The HI-STORM 100 System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized outside of the reactor facility's 10CFR Part 50 structures may be broken down into two broad categories, namely Important to Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", provides guidance for the determination of a component's safety classification. Certain ancillary equipment (such as trailers, rail cars, skids, portable cranes, transporters, or air pads) are not required to be designated as ITS for most ISFSI implementations, if the HI-STORM 100 is designed to withstand the failure of these components.

The listing and ITS designation of ancillary equipment in Table 8.1.6 follows NUREG/CR-6407. ITS ancillary equipment utilized in activities which occur outside the 10CFR Part 50 structure shall be engineered using a written specification which describes 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 specification document shall contain the following information, as applicable:

1. Functions and boundaries of the ancillary equipment

2. The environmental conditions of the ISFSI site, including tornado-borne missile, tornado wind, seismic, fire, lightning, explosion, ambient humidity limits, flood, tsunami and any other environmental hazards unique to the site.

3. Material requirements including impact testing requirements

4. Applicable codes and standards

5. Acceptance testing requirements

6. Quality assurance requirements

7. Foundation type and permissible loading

8. Applicable loads and load combinations

9. Pre-service examination requirements

10. In-use inspection and maintenance requirements

11. Number and magnitude of repetitive loading significant to fatigue

12. Insulation and enclosure requirements (on electrical motors and machinery)

13. Applicable Reg. Guides and NUREGs.

14. Welding requirements

15. Painting, marking, and identification requirements

16. Design Report documentation requirements

17. Operational and Maintenance (O&M) Manual information requirements

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

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

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

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

In both scenarios, HI-TRAC is mounted on top of the overpack (HI-STAR 100 or HI-STORM 100) 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. 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.

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).

b. Upending and downending of a HI-TRAC transfer cask on a heavy-haul transfer trailer or other transporter (see Figure 2.3.2 for an example)

c. Raising and placement of a HI-TRAC transfer cask on top of a HI-STORM 100 overpack for MPC transfer operations (see Figure 2.3.3).

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).

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

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

ii. Other Functional Requirements:

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

a. Devices and areas to support installation and removal of the HI-STORM 100 lid.

b. Devices and areas to support installation and removal of the HI-STORM 100 shield block inserts.

c. Devices and areas to support installation and removal of the HI-STAR 100 closure plate.

d. Devices and areas to support installation and removal of the HI-STAR 100 transfer collar.

e. Features to support positioning and alignment of the HI-STORM 100 overpack and the HI-TRAC transfer cask.

f. Features to support positioning and alignment of the HI-STAR 100 overpack and the HI-TRAC transfer cask.

g. Areas to support jacking of a loaded HI-STORM 100 overpack for insertion of a translocation device underneath.

h. Devices and areas to support placement of an empty MPC in the HI-TRAC transfer cask or HI-STAR 100 overpack

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

iii. Definitions:

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

- Connector Brackets: The mechanical part used in the load path which connects to the cask trunnions. A fabricated weldment, slings, and turnbuckles are typical examples of connector brackets.

- CTF structure: The CTF structure is the stationary, anchored portion of the CTF which provides the required structural function to support MPC transfer operations, including lateral stabilization of the HI-TRAC transfer cask and, if required, the overpack, to protect against seismic events. The MPC lifter, if used in the CTF design, is integrated into the CTF structure (see Lifter Mount).

- HI-TRAC lifter(s): The HI-TRAC lifter is the mechanical lifting device, typically consisting of jacks or hoists, that is utilized to lift a loaded or unloaded HI-TRAC to the required elevation in the CTF so that it can be mounted on the overpack.†

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

- Lift Platform: The lift platform is the intermediate structure that transfers the vertical load of the HI-TRAC transfer cask to the HI-TRAC lifters.

- Mobile crane: A mobile crane is a device defined in ASME B30.5-1994, Mobile and Locomotive Cranes. A mobile crane may be used in lieu of the HI-TRAC lifter and/or an MPC lifter provided all requirements set forth in this subsection are satisfied.

- MPC lifter: The MPC lifter is a mechanical lifting device, typically consisting of jacks or hoists, that is utilized to vertically transfer the MPC between the HI-TRAC transfer cask and the overpack.

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

- Single-Failure-Proof (SFP): A single-failure-proof handling device is one wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria given in of NUREG-0612.

- Translocation Device: A low vertical profile device used to laterally position an overpack such that the bottom surface of the overpack is fully supported by the top surface of the device. Typical translocation devices are air pads and Hillman rollers.

iv. Important to Safety Designation:

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

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

B. Environmental and Design Conditions

i. Lowest Service Temperature (LST): The LST for the CTF is 100°F (consistent with the specification for the HI-TRAC transfer cask in Subsection 3.1.2.3).

ii. Snow and Ice Load, S: The CTF structure shall be designed to withstand the dead weight of snow and ice for unheated structures as set forth in ASCE 7-88 [2.2.2] for the specific ISFSI site.

iii. Tornado Missile, M, and Wind,W': The tornado and tornado-generated missile data applicable to the HI-STORM 100 System (Tables 2.2.4 and 2.2.5) will be used in the design of the CTF structure unless existing site design basis data or a probabilistic risk assessment (PRA) for the CTF site with due consideration of short operation durations indicates that a less severe tornado missile impact or wind loading on the CTF structure can be postulated. The PRA analysis can be performed in the manner of the EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology Computer Code Manual". USNRC Reg. Guide 1.117 and Section 2.2.3 of NUREG-800 may be used for guidance in establishing the appropriate tornado missile and wind loading for the CTF structure.

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

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

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

The wind speed specified in this FSAR (Tables 2.2.4 and 2.2.5), likewise, shall be evaluated for their applicability to the site. Lower or higher site-specific wind velocity, compared to the design basis values cited in this

FSAR shall be used if justified by appropriate analysis, which may include PRA.

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

iv. Flood: The CTF will be assumed to be flooded to the highest elevation for the CTF facility determined from the local meteorological data. The flood velocity shall be taken as the largest value defined for the ISFSI site.

v. Lightning: Meteorological data for the region surrounding the ISFSI site shall be used to specify the applicable lightning input to the CTF structure for personnel safety evaluation purposes.

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

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

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

C. Heavy Load Handling:

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

ii. NUREG-0612 Conformance:

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

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

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 cranes, the guidance of Section 5.1.6(2) of NUREG-0612 does not apply.

iii. Defense-in-Depth Measures:

a. The lift platform and the lifter mount shall be designed to ensure that the stresses produced under the apparent dead load, D*, are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.

b. The CTF structure shall be designed to ensure that the stresses produced in it under the apparent dead load, D*, are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.

c. Maximum deflection of the lift platform and the lifter mount under the apparent dead load shall comply with the limits set forth in CMAA-70.

d. When the HI-TRAC transfer cask is stacked on the overpack, HI-TRAC shall be either held by the lifting device or laterally restrained by the CTF structure. Furthermore, when the HI-TRAC transfer cask is placed atop the overpack, the overpack shall be laterally restrained from uncontrolled movement, if required by the analysis specified in Subsection 2.3.3.1.N.

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

f. In order to minimize occupational radiation exposure to ISFSI personnel, design of the MPC lifting attachment (viz., sling) should not require any human activity inside the HI-TRAC cylindrical space.

g. The HI-TRAC lifter and MPC lifter shall possess design features to avoid side-sway of the payload during lifting operations.

h. The lifter (HI-TRAC and MPC) design shall ensure that any electrical malfunction in the motor or the power supply will not lead to an uncontrolled lowering of the load.

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

iv. Shielding Surety:

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

v. Specific Requirements for Mobile Cranes:

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

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

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

vi. Lift Height Limitation: The HI-TRAC lift heights shall be governed by the Technical Specifications.

vii. Control of Side Sway: Procedures shall provide provisions to ensure that the load is lifted essentially vertically with positive control of the load. Key cask lifting and transfer procedures, as determined by the user, should be reviewed by the Certificate Holder before their use.

D. Loads and Load Combinations for the CTF Structure

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

E. Materials and Failure Modes

i. Acceptable Materials and Material Properties: All materials used in the design of the CTF shall be ASTM approved or equal, consistent with the ITS category of the part. Reinforced concrete, if used, shall comply with the provisions of ACI 318 (89). The material property and allowable stress values for all steel structurals shall be taken from the ASME and B&PV Code, Section II, wherever such data is available; otherwise, the data provided in the ASTM standards shall be used.

ii. Brittle Fracture: All structural components in the CTF structure and the lift platform designated as primary load bearing shall have an NDTT equal to 0EF or lower (consistent with the ductile fracture requirements for ASME Section III, Subsection NF, Class 3 structures).

iii. Fatigue: Fatigue failure modes of primary structural members in the CTF structure whose failure may result in uncontrolled lowering of the HI-TRAC transfer cask or the MPC (critical members) shall be evaluated. A minimum factor of safety of 2 on the number of permissible loading cycles on the critical members shall apply.

iv. Buckling: For all critical members in the CTF structure (defined above), potential failure modes through buckling under axial compression shall be considered. The margin of safety against buckling shall comply with the provisions of ASME Section III, Subsection NF, for Class 3 linear structures.

F. CTF Pad

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

G. Miscellaneous Components

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

H. Structural Welds

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

I. Foundation

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

J. Rail Access

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

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

i. The cask handling bay in the CTF shall allow access of a vertical cask crawler or translocation device carrying a transfer cask or overpack. The building floor shall be equipped with a smooth transition to the cask travel route such that the vertical cask crawler tracks do not have to negotiate sharp lips or slope transitions and the translocation devices have a smooth transition. Grading of exterior aprons shall be no more than necessary to allow water drainage.

ii. If roll-up doors are used, the roll up doors shall have no raised threshold that could damage the vertical cask crawler tracks (if a crawler is used).

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

L. Facility Floor

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

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

iii. The rails, in areas where translocation device operations may occur shall be below the finish level of the floor. Flush inserts, if necessary, shall be sized for installation by hand.

M. Cask Connector Brackets

i. Primary lifting attachments between the cask and the lifting platform are the cask connector brackets. The cask connector brackets may be lengthened or shortened to allow for differences in the vehicle deck height of the cask delivery vehicle and the various lifting operations. The connector brackets shall be designed to perform cask lifting, upending and downending functions. The brackets shall be designed in accordance with ANSI N14.6 [Reference 2.2.3] and load tested at 300% of the load applied to them during normal handling.

ii. The connector brackets shall be equipped with a positive engagement to ensure that the cask lifting attachments do not become inadvertently disconnected during a seismic event and during normal cask handling operations.

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

N. Cask Restraint System

A time-history analysis of the stacked overpack/HI-TRAC transfer cask assemblage under the postulated ISFSI Level D events in Table 2.3.2 shall be performed to demonstrate that a minimum margin of safety of 1.1 against overturning or kinematic instability exists and that the CTF structure complies with the applicable stress limits

(Table 2.3.2) and that the maximum permissible deceleration loading specified in the FSAR is not exceeded. If required to meet the minimum margin of safety of 1.1, a cask restraining system shall be incorporated into the design of the Cask Transfer Facility to provide lateral restraint to the overpack (HI-STORM 100 or HI-STAR 100).

O. Design Life

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

P. Testing Requirements

In addition to testing recommended in NUREG-0612 (1980), a structural adequacy test of the CTF structure at 125% of its operating load prior to its first use in a cask loading campaign shall be performed. This test should be performed in accordance with the guidance provided in the CMAA Specification 70 [2.2.16].

Q. Quality Assurance Requirements

All components of the CTF shall be manufactured in full compliance with the quality assurance requirements applicable to the ITS category of the component as set forth in Chapter 13 of this FSAR.

R. Documentation Requirements

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

- Maintenance Drawings
- Operating Procedures

ii. Design Report: A QA-validated design report documenting full compliance with the provisions of this specification shall be prepared and archived for future reference in accordance with the provisions of Chapter 13 of this FSAR.

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, thermocouples may be installed in two of the overpack exit vents to continuously monitor the air temperature. If the thermocouples and associated temperature monitoring instrumentation are used, they shall be designated important to safety as specified in Table 2.2.6.

The thermocouples 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 thermocouple 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 (Boral™) in the MPC fuel basket walls.

b. Favorable geometry provided by the MPC fuel basket

Administrative controls specified as Technical Specifications are provided in Chapter 12 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 an annual dose equivalent greater than the values stated below for normal and off-normal conditions. Further, an individual located at the site boundary must not receive a dose to the whole body or any organ from any design basis accident greater than the values listed in Table 2.3.1.

The objective of shielding is to assure that radiation dose rates at key locations are below acceptable levels for those locations. Three locations are of particular interest in the storage mode:

- immediate vicinity of the cask
- restricted area boundary
- controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded overpack are important in consideration of occupational exposure. A design objective for the maximum average radial surface dose rate has been established as 40 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 60 mrem/hr. The average dose rate at the top of the overpack is limited to below 10 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, 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 two transfer cask designs are provided (i.e.,125 ton HI-TRAC and 100 ton HI-TRAC). The 125 ton HI-TRAC provides better shielding than the 100 ton HI-TRAC due to the increased shielding thickness and corresponding greater weight. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would 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 considerations, or space envelope limitations in the fuel pool or air lock. As with other dose reduction-based plant modifications, individual users who cannot accommodate the 125 ton HI-TRAC due to plant design limitations must perform a cost-benefit analysis of the modifications which would be necessary to use the 125 ton HI-TRAC. The cost of the modification(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 meets the above radiation dose limits and design objectives.

2.3.5.3 Radiological Alarm System

There are no credible events which could result in release of radioactive materials or increases in direct radiation above the requirements of 10CFR72.106. In addition, the non-mechanistic release as the result of a hypothetical accident is described in Chapter 7, and results in a dose to an individual at the controlled area boundary of a very small magnitude. Therefore, radiological alarm systems are not necessary.

2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM 100 System. No such materials would be stored within an ISFSI. However, for conservatism we have analyzed a hypothetical fire accident as a bounding condition for HI-STORM 100. An evaluation of the HI-STORM 100 System in a fire accident is discussed in Chapter 11.

Small overpressures may result from accidents involving explosive materials which are stored or transported near the site. Explosion is an accident loading condition considered in Chapter 11.

Table 2.3.1

RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

BOUNDARY OF CONTROLLED AREA (m) (minimum)	100
NORMAL AND OFF-NORMAL CONDITIONS:	
Whole Body (mrem/yr)	25
Thyroid (mrem/yr)	75
Any Other Organ (mrem/yr)	25
DESIGN BASIS ACCIDENT:	
Whole Body (rem)	5
Any Organ (rem)	5

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	Level D	Factor of safety against overturning shall be ≥1.1
D+E or D+Y	Level D	

[†]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.

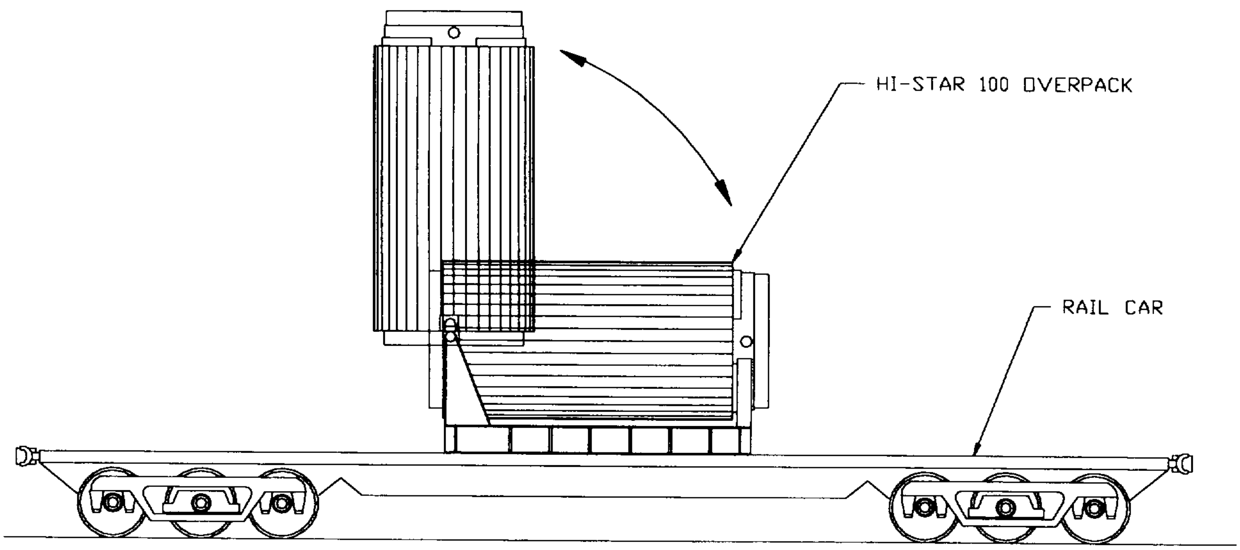

FIGURE 2.3.1; HI-STAR 100 UPENDING AND DOWNENDING ON A RAIL CAR

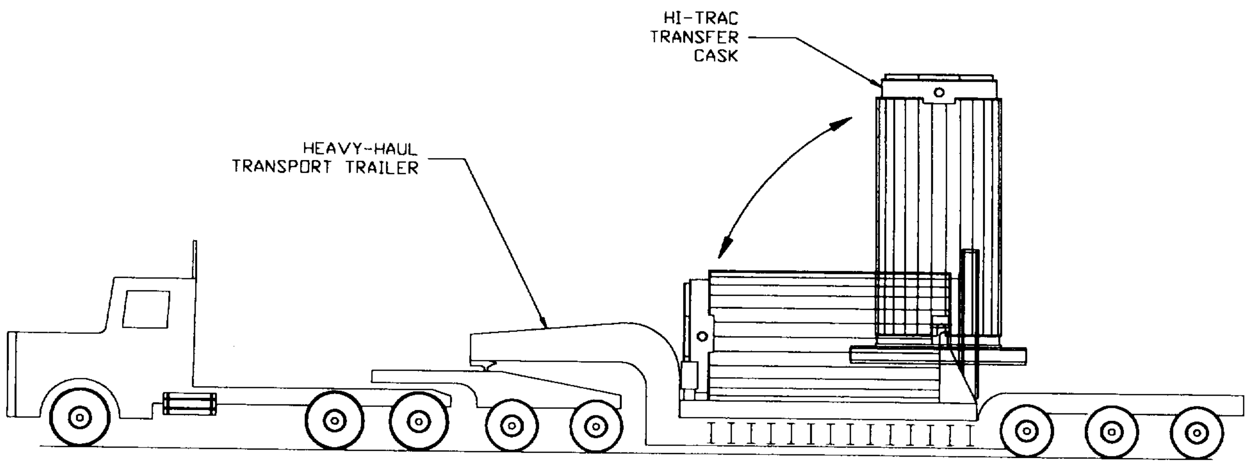

FIGURE 2.3.2; HI-TRAC UPENDING AND DOWNENDING ON A HEAVY-HAUL TRANSPORT TRAILER

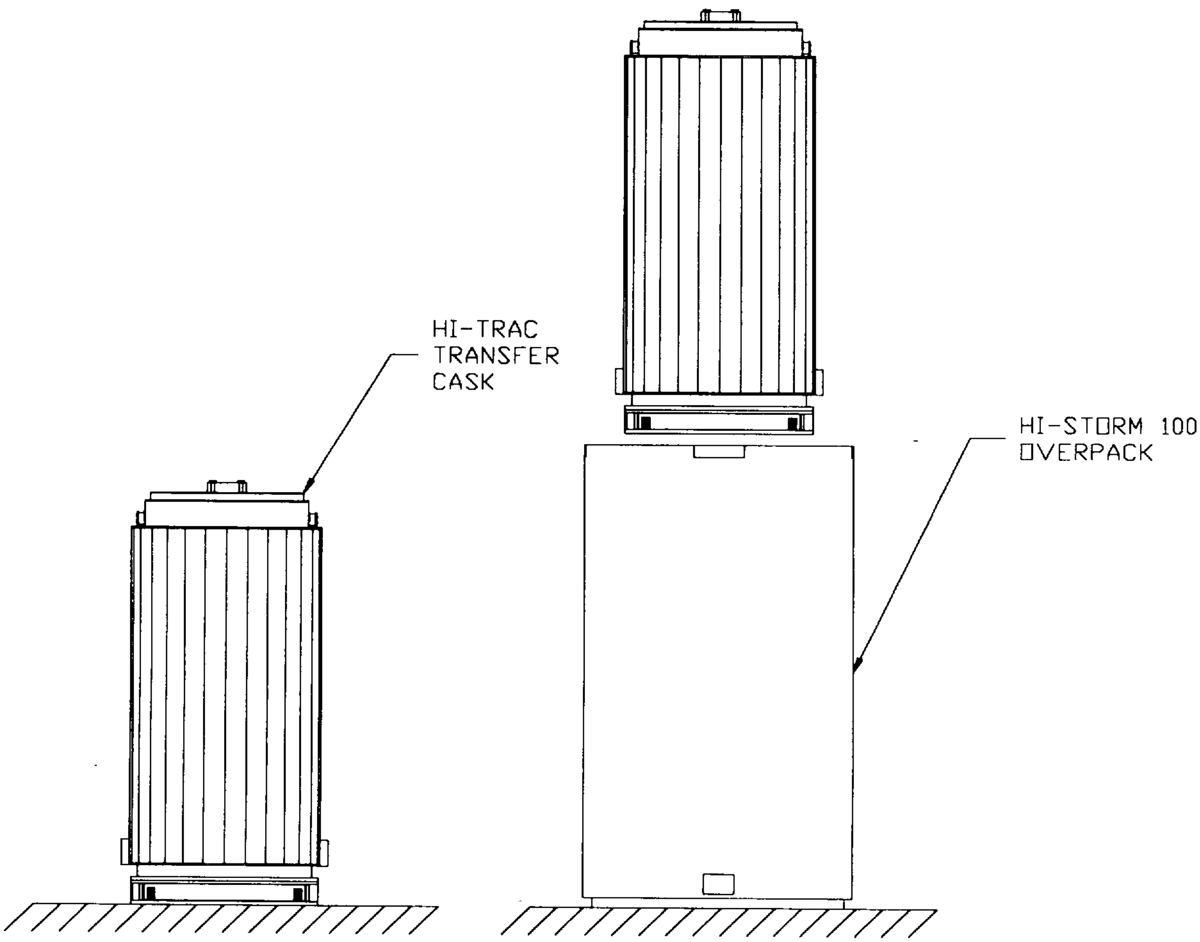

FIGURE 2.3.3; HI-TRAC PLACEMENT ON HI-STORM 100 FOR MPC TRANSFER OPERATIONS

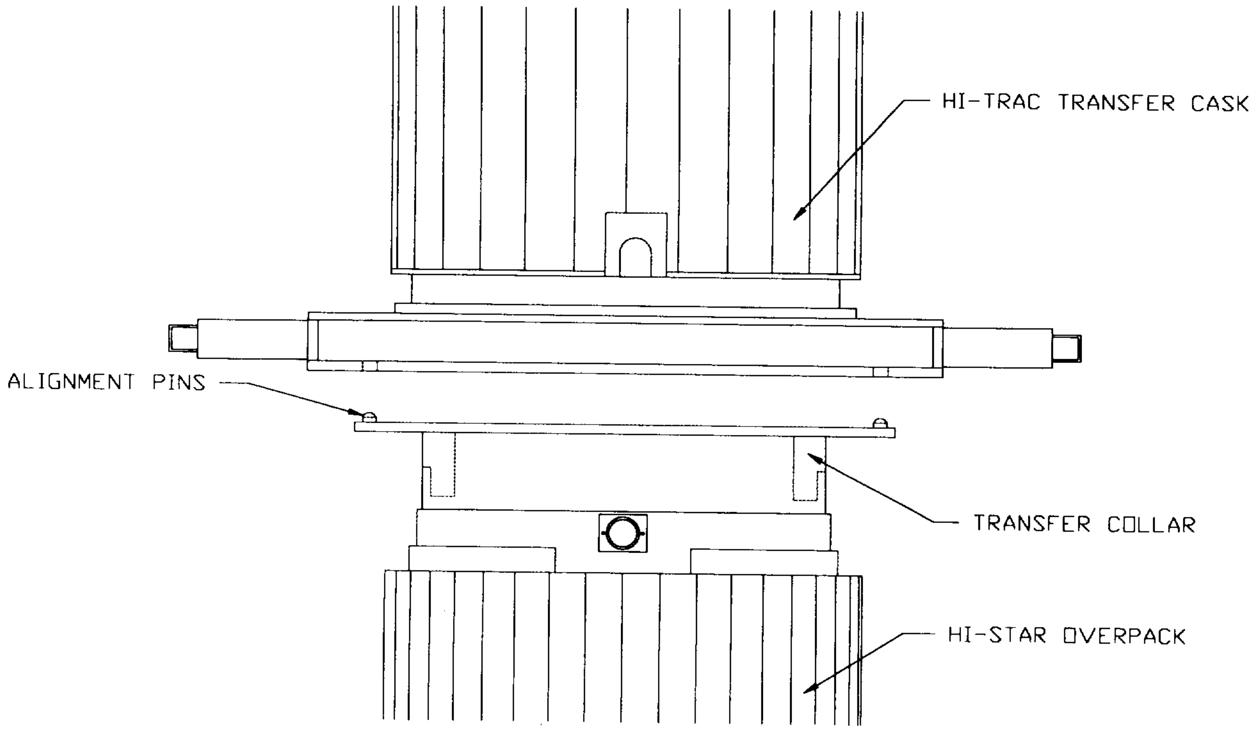

FIGURE 2.3.4; HI-TRAC PLACEMENT ON HI-STAR 100 FOR MPC TRANSFER OPERATIONS

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. Therefore, the SNF assemblies stored in the MPC should not need to be removed. However, to ensure a practical, feasible method to defuel the MPC, the top of the MPC is equipped with sufficient gamma shielding and markings locating the drain and vent locations to enable semiautomatic (or remotely actuated) boring of the MPC lid to provide access to the MPC vent and drain. The circumferential welds of the MPC lid closure ring can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the overpack consists of steel and concrete rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the overpack, and the latter reused for storage of other MPCs.

In either case, the overpack would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

If the MPC needs to be opened and separated from the SNF before the fuel is placed into the MGDS, the MPC interior metal surfaces will be decontaminated using existing mechanical or chemical methods. This will be facilitated by the MPC fuel basket and interior structures' smooth metal surfaces designed to minimize crud traps. After the surface contamination is removed, the MPC radioactivity will be diminished significantly, allowing near-surface burial or secondary applications at the licensee's facility.

It is also likely that both the overpack and MPC, or extensive portions of both, can be further decontaminated to allow recycle or reuse options. After decontamination, the only radiological hazard the HI-STORM 100 System may pose is slight activation of the HI-STORM 100 materials caused by irradiation over a 40-year storage period.

Due to the design of the HI-STORM 100 System, no residual contamination is expected to be left behind on the concrete ISFSI pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last overpack is removed.

To evaluate the effects on the MPC and HI-STORM overpack caused by irradiation over a 40-year storage period, the following analysis is provided. Table 2.4.1 provides the conservatively determined quantities of the major nuclides after 40 years of irradiation. The calculation of the material activation is based on the following:

- Beyond design basis fuel assemblies (B&W 15x15, 3.7% enrichment, 47,500 MWD/MTU, and eight-year cooling time) stored for 40 years.
- Material quantities based on the Design 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	6.65e-4
^{55}Fe	1.07e-3
^{59}Ni	8.79e-7
^{60}Co	9.39e-5
^{63}Ni	2.98e-5
Total	1.86e-3

HI-STORM OVERPACK ACTIVATION

Nuclide	Activity After 40-Year Storage (Ci/m^3)
Overpack Steel	
^{54}Mn	1.09e-4
^{55}Fe	2.06e-3
Total	2.17e-3
Overpack Concrete	
^{39}Ar	9.11e-7
^{41}Ca	7.36e-8
^{54}Mn	4.79e-7
^{55}Fe	8.90e-6
Total	1.04e-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.

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