FINAL SAFETY ANALYSIS REPORT

ON

THE HI-STORM FW SYSTEM

By

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- ☐ Calculation Package[3] (Per HQP 3.2)
- ☒ Technical Report (Per HQP 3.2)(Such as a Licensing Report)
- ☐ Design Criterion Document (Per HQP 3.4)
- ☐ Design Specification (Per HQP 3.4)
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GLOSSARY

ALARA is an acronym for As Low As Reasonably Achievable

Ancillary or Ancillary Equipment is the generic name of a device used to carry out short term operations.

Bottom Lid means the removable lid that fastens to the bottom of the HI-TRAC VW transfer cask body to create a gasketed barrier against in-leakage of pool water in the space around the MPC.

BWR is an acronym for Boiling Water Reactor.

CG is an acronym for center of gravity.

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

Confinement Boundary is the outline formed by the all-welded cylindrical enclosure of the MPC shell, MPC baseplate, MPC lid, MPC port cover plates, and the MPC closure ring which provides redundant sealing.

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

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

Cooling Time (or post-irradiation cooling time) for a spent fuel assembly is the time between its final discharge from the reactor to the time it is loaded into the MPC.

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

DAS is the abbreviation for the Decontamination and Assembly Station. It means the location where the Transfer Cask is decontaminated and the MPC is processed (i.e., where all operations culminating in lid and closure ring welding are completed).

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container (or Canister) or DFC means a specially designed enclosure for damaged fuel or fuel debris which permits flow of gaseous and liquid media while minimizing dispersal of gross particulates.

Design Basis Load (DBL) is a loading which bounds one or more events that are applicable to the storage system during its service life.

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

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

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

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

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

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

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

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

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

Fuel Building is the generic term used to denote the building in which the fuel loading and where part of "short-term operations" will occur. The Fuel Building is a Part 50 controlled structure.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, containers or structures that are supporting these loose fuel assembly parts, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

Fuel Spacer or Shim is a metallic part interposed in the space between the fuel and the MPC cavity at either the top or the bottom (or both) ends of the fuel to minimize the axial displacement of the SNF within the MPC due to longitudinal inertia forces.

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

HI-TRAC VW transfer cask or HI-TRAC VW 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 and protects the loaded MPC.

HI-STORM overpack or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel for long term sotrage. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the loaded MPC.

HI-STORM FW System consists of any loaded MPC model placed within the HI-STORM FW overpack.

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.

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

Long-term Storage means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad.

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

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

METAMIC® is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs and in wet storage applications.

METAMIC-HT is the trade name for the metal matrix composite made by imbedding nano-particles of aluminum oxide and fine boron carbide powder on the grain boundaries of aluminum resulting in improved structural strength properties at elevated temperatures.

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

MGDS is an acronym for Mined Geological Disposal System.

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

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

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

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

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

Neutron Absorber is a generic term to indicate any neutron absorber material qualified for use in the HI-STORM FW System.

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

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, Instrument Tube Tie Rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

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

Plain Concrete is concrete that is unreinforced.

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

PWR is an acronym for pressurized water reactor.

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

Regionalized Fuel Storage is a term used to describe an optimized fuel loading strategy wherein the storage locations are ascribed to distinct regions each with its own maximum allowable specific heat generation rate.

Removable Shielding Girdle is an ancillary designed to be installed to provide added shielding to the personnel working in the top region of the transfer cask.

SAR is an acronym for Safety Analysis Report.

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

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

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.

TAL is an acronym for the Threaded Anchor Location. TALs are used in the HI-STORM FW and HI-TRAC VW casks as well as the MPCs.

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

Traveler means the set of sequential instructions used in a controlled manufacturing program to ensure that all required tests and examinations required upon the completion of each significant manufacturing activity are performed and documented for archival reference.

Undamaged Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

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

ZPA is an acronym for zero period acceleration.

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

CHAPTER 1: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This final safety analysis report (FSAR) describes the Holtec International HI-STORM FW System and contains the necessary information and analyses to support a United States Nuclear Regulatory Commission (USNRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under the provisions of 10 CFR 72 [1.0.1]. This report, prepared pursuant to 10 CFR 72.230, describes the basis for NRC approval and issuance of a Certificate of Compliance (CoC) on the HI-STORM FW System under 10 CFR 72, Subpart L to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI) under the general license authorized by 10 CFR 72, Subpart K.

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]. The only deviation in the format from the formatting instruction in Reg. Guide 3.61 is the insertion of a chapter (Chapter 8) on material compatibility pursuant to ISG-15 and renumbering of all subsequent chapters. Rev 1A of NUREG 1536, available only as a draft document at the time of the initial composition of this report (Rev 0), has also been consulted to insure conformance.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM FW System, drawings of the structures, systems, and components (SSCs), designation of their safety classification, and the qualifications of the certificate holder. This report is also suitable for incorporation into a site-specific Safety Analysis Report, which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility that is similar in objective and scope.

Table 1.0.1 provides the principal components of the HI-STORM FW System. An MPC (containing either PWR or BWR fuel) is placed inside the HI-STORM FW overpack for long term storage. The overpack provides shielding, allows for convective cooling, and protects the MPC. The HI-TRAC VW transfer cask is used for MPC transfer and also provides shielding and protection while the MPC is being prepared for storage.

Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10 CFR 72 requirements, and a reference to the applicable report section that addresses each topic.

The HI-STORM FW 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 minor deviations in the method of compliance. Table 1.0.3 lists these deviations, along with a discussion of the approach for compliance, and justification. The justification may be in the form of supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. (The extent of compliance with NUREG-1536 in this docket mirrors that in Docket No. 72-1014.)

The Glossary contains a listing of the terminology and notation used in this FSAR.

The safety evaluations in this FSAR are intended to bound the conditions that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This includes the potential fuel assemblies which will be loaded into the system and the environmental conditions in which the system will be deployed. 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 bases and safety analyses documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM FW System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM FW System FSAR identifies a number of conditions that are 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 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 9 and 10, and 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.

In presenting the bounding generic analyses of this safety report, selected conditions are drawn from

authoritative sources such as Regulatory Guides and NUREGs, where available. For example, the wind and tornado characteristics are excerpted from Reg. Guide 1.76 [1.0.4].

For analyses that do not have a prescribed acceptance limit or bounding condition, illustrative calculations are carried out with a fuel type most commonly used at reactor sites. The Reference SNF for PWR and BWR fuel types are listed in Table 1.0.4. These Reference SNF assemblies are used when fixed limits for compliance are not established by regulations, such as dose rates.

Where the analysis must demonstrate compliance with a fixed limit, such as the reactivity limit of 0.95 in criticality analysis, the most limiting fuel type is used in the analysis. The Design Basis Fuel (Table 2.1.4) may differ depending on the analysis being performed (e.g., thermal, structural, etc...). Thus, broadly speaking, the analyses in this FSAR belong to two categories:

a. Those that are performed to satisfy a specific set of hard limits in the regulations or the Standard Review Plan.
b. Those that are representative in nature and intended to demonstrate the acceptability of the analysis models and capability of the system.

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. Similarly, the following deci-numeric convention is used in the organization of chapters:

a. A chapter is identified by a whole numeral, say *m* (i.e., *m*=3 means Chapter 3).
b. A section is identified by one decimal separating two numerals. Thus, Section 3.1 is a section in Chapter 3.
c. A subsection has three numerals separated by two decimals. Thus, Subsection 3.2.1 is a subsection in Section 3.2.
d. A paragraph is denoted by four numerals separated by three decimals. Thus, Paragraph 3.2.1.1 is a paragraph in Subsection 3.2.1.
e. A subparagraph has five numerals separated by four decimals. Thus, Subparagraph 3.2.1.1.1 is a part of Paragraph 3.2.1.1.

Tables and figures associated with a section are placed after the text narrative. Complete sections are replaced if any material in the section is changed. The specific changes are appropriately annotated. Drawing packages are controlled separately within the Holtec QA program and have individual revision numbers. If a drawing is revised in support of the current FSAR revision, that drawing is included in Section 1.5 at its latest revision level. Upon issuance of the CoC, drawings and text matter in this FSAR may be revised between formal updates under the 10CFR 72.48 process. All changes to the FSAR including the drawings are subject to a rigorous configuration control under the Company's QA program.

TABLE 1.0.1

HI-STORM FW SYSTEM COMPONENTS

Item	Designation (Model Number)
Overpack	HI-STORM FW
PWR Multi-Purpose Canister	MPC-37
BWR Multi-Purpose Canister	MPC-89
Transfer Cask	HI-TRAC VW

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW 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
	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
		10CFR72.122(b)(1)	2.2
		10CFR72.122(b)(2)	2.2.3
		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
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)(2)	2.2.3

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b)(2)	2.2.3
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1
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)	1.5
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a) (4)	2.3.2.2, 4.0
NA	2.III.3.f Operating Procedures	10CFR72.24(f) 10CFR72.128(a)(5)	11.0, 9.0
		10CFR72.236(h)	9.0
		10CFR72.24(1)(2)	1.2.1, 1.2.2
		10CFR72.236(1)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	12.0, 9.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a)(1)	10.0
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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
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a)(2)	2.3.5
		10CFR72.128(a) (3)	2.3.2
		10CFR72.236(d)	2.3.2, 2.3.5
		10CFR72.236(e)	2.3.2

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h) (4) 10CFR72.122(i) 10CFR72.128(a)(1)	2.3.5
2.3.4 Nuclear Criticality Safety	2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	11.4.2
		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
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
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(1)	3.4.4 3.4.7 3.4.10

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4
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
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h)(1)	3.5
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a)(4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria for off-normal and accident conditions	10CFR72.24(d) 10CFR72.122(c)	4.6
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

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
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
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2
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.	--	Appendix 5.A
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
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2, 6.3.3, 6.4.4 to 6.4.9
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FW FSAR
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.	--	Appendix 6.B
	7. Confinement		
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety ISG-18	10CFR72.24(c)(3) 10CFR72.24(1)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h)(l)	7.1, 7.1.1
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.236(1)	7.1
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7. 1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h)(4) 10CFR72.128(a)(l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose ISG-18	10CFR72.104(a)	7.1
7.2.2 Pressurization of Confinement Vessel	--	--	7.1
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.1
7.3.1 Fission Gas Products	--	--	7.1
7.3.2 Release of Contents	ISG-18	--	7.1
NA	--	10CFR72.106(b)	7.1

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FW FSAR
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Material Evaluation			
NA	X.5.1 General Considerations (ISG-15)	10CFR72.24(c)(3) 10CFR72.236(m) 10CFR72.122(a) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.124 10CFR72.128(a)(2)	8.1
	X.5.2 Materials Selection (ISG-15)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.124 10CFR72.128(a)(2) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.236(g) 10CFR72.236(l) 10CFR72.236(h)	8.2, 8.3, 8.4, 8.5, 8.6, 8.7, 8.9, 8.10, 8.11
	X.5.3 Chemical and Galvanic Reactions (ISG-15)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.236(h) 10CFR72.122(h)(1) 10CFR72.236(m)	8.12
	X.5.4 Cladding Integrity (ISG-15) (ISG-11)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.24(c)(3) 10CFR72.236(g) 10CFR72.236(h)	8.13
9. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	9.0 et. seq.
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	9.2

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FW FSAR
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	9.2
	8.III.4 Written Procedures	10CFR72.212(b)(9)	9.2
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	9.2
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	9.2
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	9.2, 9.4
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	9.4
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	9.4
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	9.4
	8.III.4 Written Procedures	10CFR72.212(b) (9)	9.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	9.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	9.0
	8.III.8 Ready Retrieval	10CFR72.122(1)	9.4
8.3 Preparation of the Cask	--	--	9.3.2
8.4 Supplemental Data	--	--	Tables 9.1.1
NA	8.III.9 Design to Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a)(5)	9.2, 9.4
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 9.1.6
10. Acceptance Criteria and Maintenance Program			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	9.1, 10.1
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	10.1
	9.III.1.d Test Program	10CFR72.162	10.1

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FW FSAR
	9.III.1.e Appropriate Tests	10CFR72.236(l)	10.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	10.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	10.1(2)
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	10.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a)(1)	10.2
	9.III.1.h Records of Maintenance	10CFR72.212(b)(8)	10.2
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)	10.1.7, 10.1.1.(12)
	9.III.3 Cask Identification		
11. Radiation Protection			
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	11.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	11.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	11.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	11.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
12. Accident Analyses			

TABLE 1.0.2

REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FW FSAR
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	12.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	12.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	12.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	12.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	12.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	12.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	12.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	9.4
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
13. Operating Controls and Limits			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	13.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	13.0
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	13.0

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 13.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 13.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 13.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 13.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 13.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 13
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 13
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 13.A
NA	12.III.2 SSC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0

TABLE 1.0.2 REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FW FSAR**
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)	9.0, 11.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	9.0
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, 10.0
14. Quality Assurance			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) 10CFR72.140(d)	14.0
	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 10.

(3) Not applicable to HI-STORM FW 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 FW 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 ALTERNATIVES TO NUREG-1536		
NUREG-1536 Guidance	**Alternate Method to Meet NUREG-1536 Intent**	**Justification**
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."	A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM FW 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 FSAR 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.
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..."	The HI-STORM system components are lifted and handled by lifting equipment that meet the applicable provisions in NUREG-0612 and ANSI 14.6 to preclude an uncontrolled lowering of the load.	The HI-STORM FW is a vertically deployed system. All lifting and handling operations occur in the vertical orientation and with symmetrically stressed handling devices. All lifting and handling devices are also required to meet the ANSI provisions to render the potential of a drop event in the part 72 jurisdiction non-credible. The vertical drop analysis is therefore not required.
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..."	HI-STORM FW, like HI-STORM 100, uses plain concrete. The structural function is rendered by a double wall shell of carbon steel. The primary steel shell structure is designed to meet ASME Section III, Subsection NF stress limits for all normal service conditions.	Concrete is provided in the HI-STORM overpack primarily for the purpose of radiation shielding, the reinforcement in the concrete will only serve to create locations of micro-voids that will increase the emitted dose from the cask. Appendix 1.D of the HI-STORM 100 FSAR which provides technical and placement requirements on plain concrete is also invoked for HI-STORM FW concrete.
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.	All free volume calculations use nominal Confinement Boundary dimensions, but the volume occupied by the fuel assemblies is calculated using maximum weights and minimum densities.	Calculating the volume occupied by the fuel assemblies using maximum weights and minimum densities conservatively over predicts the volume occupied by the fuel and correspondingly under predicts the remaining free volume.

TABLE 1.0.3 ALTERNATIVES TO NUREG-1536		
NUREG-1536 Guidance	**Alternate Method to Meet NUREG-1536 Intent**	**Justification**
7.V.4 "Confinement Analysis. Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary."	No confinement leakage analysis is performed and no effluent dose at the controlled area boundary is calculated.	The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the Confinement Boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the Confinement Boundary (e.g., non-destructive examinations and pressure testing). Pursuant to ISG-18, the Holtec MPC is constructed in a manner that precludes leakage from the Confinement Boundary. Therefore, no analysis of leakage from confinement is required.
13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."	Chapter 14 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference.	The NRC has approved the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec will apply this QA program to all important-to-safety dry storage cask activities.

TABLE 1.0.4 REFERENCE SNF DESIGNATIONS	
Fuel Type	**Fuel ID**
PWR	W 17x17
BWR	GE 10x10

1.1 INTRODUCTION TO THE HI-STORM FW SYSTEM

This section and the next section (Section 1.2) provide the necessary information on the HI-STORM FW System pursuant to 10CFR72 paragraphs 72.2(a)(1),(b); 72.122(a),(h)(1); 72.140(c)(2); 72.230(a),(b); and 72.236(a),(c),(h),(m).

HI-STORM (acronym for Holtec International Storage Module) FW System is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10CFR72. The model designation "FW" denotes this as a system which has been specifically engineered to withstand sustained Flood and Wind.

The HI-STORM FW System consists of a sealed metallic multi-purpose canister (MPC) contained within an overpack constructed from a combination of steel and concrete. The design features of the HI-STORM FW components are intended to simplify and reduce the on-site SNF loading and handling work effort, to minimize the burden of in-use monitoring, to provide utmost radiation protection to the plant personnel, and to minimize the site boundary dose.

The HI-STORM FW System can safely store either PWR or BWR fuel assemblies, in the MPC-37 or MPC-89, respectively. The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. The MPC external diameters are identical to allow the use of a single overpack design, however the height of the MPC, as well as the overpack and transfer cask, are variable based on the SNF to be loaded.

Figure 1.1.1 shows the HI-STORM FW System with two of its major constituents, the MPC and the storage overpack, in a cut-away view. The MPC, shown partially withdrawn from the storage overpack, is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the Confinement Boundary for the stored spent nuclear fuel assemblies. The HI-STORM FW storage overpack provides structural protection, cooling, and radiological shielding for the MPC.

The HI-STORM FW overpack is equipped with thru-wall penetrations at the bottom of the overpack and in its lid to permit natural circulation of air to cool the MPC and the contained SNF. The HI-STORM FW 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 at the site. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM FW System since it is completely passive and is composed of proven materials. The HI-STORM FW System can be used either singly or as an array at an ISFSI. The site for an ISFSI can be located either at a nuclear reactor facility or an away-from-a-reactor (AFR) location.

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

The HI-STORM FW overpack is designed to possess certain key elements of flexibility to achieve

ALARA. For example:

- The HI-STORM FW 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 exterior and also in the interior of the MPC.
- The HI-STORM FW overpack handling operations do not require the cask to be downended at any time which eliminates the associated handling risks and facilitates compliance with radiation protection objectives.
- The HI-STORM FW overpack can be loaded with the MPC containing SNF using the HI-TRAC VW transfer cask and prepared for storage while inside the 10CFR50 [1.1.2] facility. From the 10CFR50 facility the loaded overpack is then moved to the ISFSI and stored in a vertical configuration. The overpack can also be directly loaded using the HI-TRAC VW transfer cask adjacent to the ISFSI storage pad. Some examples of MPC transfer between the FW overpack and the HI-TRAC VW transfer cask are illustrated in Figures 1.1.2 (transfer at the cask transfer facility) and 1.1.3 (transfer in the plant's egress (truck/rail) bay).

The HI-STORM FW overpack features an inlet and outlet duct configuration engineered to mitigate the sensitivity of wind direction on the thermal performance of the system. More specifically, the HI-STORM FW overpack features a radially symmetric outlet vent (located in its lid) pursuant to Holtec's Patent Number 7,330,526B2 and inlet ducts arranged at 45-degree intervals in the circumferential direction to approximate an axisymmetric opening configuration, to the extent possible.

A number of design measures are taken in the HI-STORM FW System to limit the fuel cladding temperature rise under a most adverse flood event (i.e., one that is just high enough to block the inlet duct):

a. The overpack's inlet duct is narrow and does not allow a direct pathway through the overpack, therefore the MPC stands directly on the overpack's baseplate. This allows floodwater to come in immediate contact with the bottom of the MPC and assist the ventilation air flow in cooling the MPC.

b. The overpack's inlet duct is tall and the MPC stands directly on the overpack's baseplate, which is welded to the overpack's inner and outer shells. Thus, if the flood water rises high enough to block air flow through the inlet ducts, substantial surface area of the lower region of the MPC will be submerged in the water. Although heat transfer from the exterior of the MPC through air circulation is limited in such a scenario, the reduction is offset by convective cooling through the floodwater itself.

c. The MPCs are equipped with internal thermosiphon capability, which brings the heat emitted by the fuel back to the bottom region of the MPC as the circulating helium flows along the downcomer space around the fuel basket. This thermosiphon action places the heated helium in close thermal communication with the floodwater, further enhancing convective cooling via the floodwater.

The above design features of the HI-STORM FW System are subject to intellectual property protection rights (patent rights) under United States Patent and Trademark Office (USPTO) regulations.

Regardless of the storage cell count, the construction of the MPC is fundamentally the same; the basket is a honeycomb structure comprised of cellular elements. This is positioned within a circumscribing cylindrical canister shell. The egg-crate construction and cell-to-canister shell interface employed in the MPC basket impart the structural stiffness necessary to satisfy the limiting load conditions discussed in Chapter 2. Figures 1.1.4 and 1.1.5 provide cross-sectional views of the PWR and BWR fuel baskets, respectively. Figures 1.1.6 and 1.1.7 provide isometric perspective views of the PWR and BWR fuel baskets, respectively.

The HI-TRAC VW transfer cask is required for shielding and protection of the SNF during loading and closure of the MPC and during movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. Figure 1.1.8 shows a cut away view of the transfer cask. The MPC is placed inside the HI-TRAC VW transfer cask and moved into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC VW/MPC assembly is designed to prevent (contaminated) pool water from entering the narrow annular space between the HI-TRAC VW and the MPC while the assembly is submerged. The HI-TRAC VW transfer cask also allows dry loading (or unloading) of SNF into the MPC in a hot cell.

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

- maximize shielding and physical protection for the MPC;
- maximize resistance to flood and wind;
- minimize the extent of handling of the SNF;
- minimize dose to operators during loading and handling;
- require minimal ongoing surveillance and maintenance by plant staff;
- facilitate SNF transfer of the loaded MPC to a compatible transport overpack for transportation;
- permit rapid and unencumbered decommissioning of the ISFSI;

Finally, design criteria for a forced helium dehydration (FHD) system, as described in Appendix 2.B of the HI-STORM 100 FSAR [1.1.3] is compatible with HI-STORM-FW. Thus, the references to a FHD system in this FSAR imply that its design criteria must comply with the provisions in the latest revision of the HI-STORM 100 FSAR (Docket No. 72-1014).

All HI-STORM FW System components (overpack, transfer cask, and MPC) are designated ITS and their sub-components are categorized in accordance with NUREG/CR-6407 [1.1.4].

The principal ancillaries used in the site implementation of the HI-STORM FW System are summarized in Section 1.2 and referenced in Chapter 9 in the context of loading operations. A listing of common ancillaries needed by the host site is provided in Table 9.2.1. The detailed design of these ancillaries is not specified in this FSAR. In some cases, there are multiple distinct ancillary designs available for a particular application (such as a forced helium dehydrator or a vacuum drying system for drying the MPC) and as such, not every ancillary will be needed by every site. Ancillary designs are typically specific to a site to meet ALARA and personnel safety objectives.

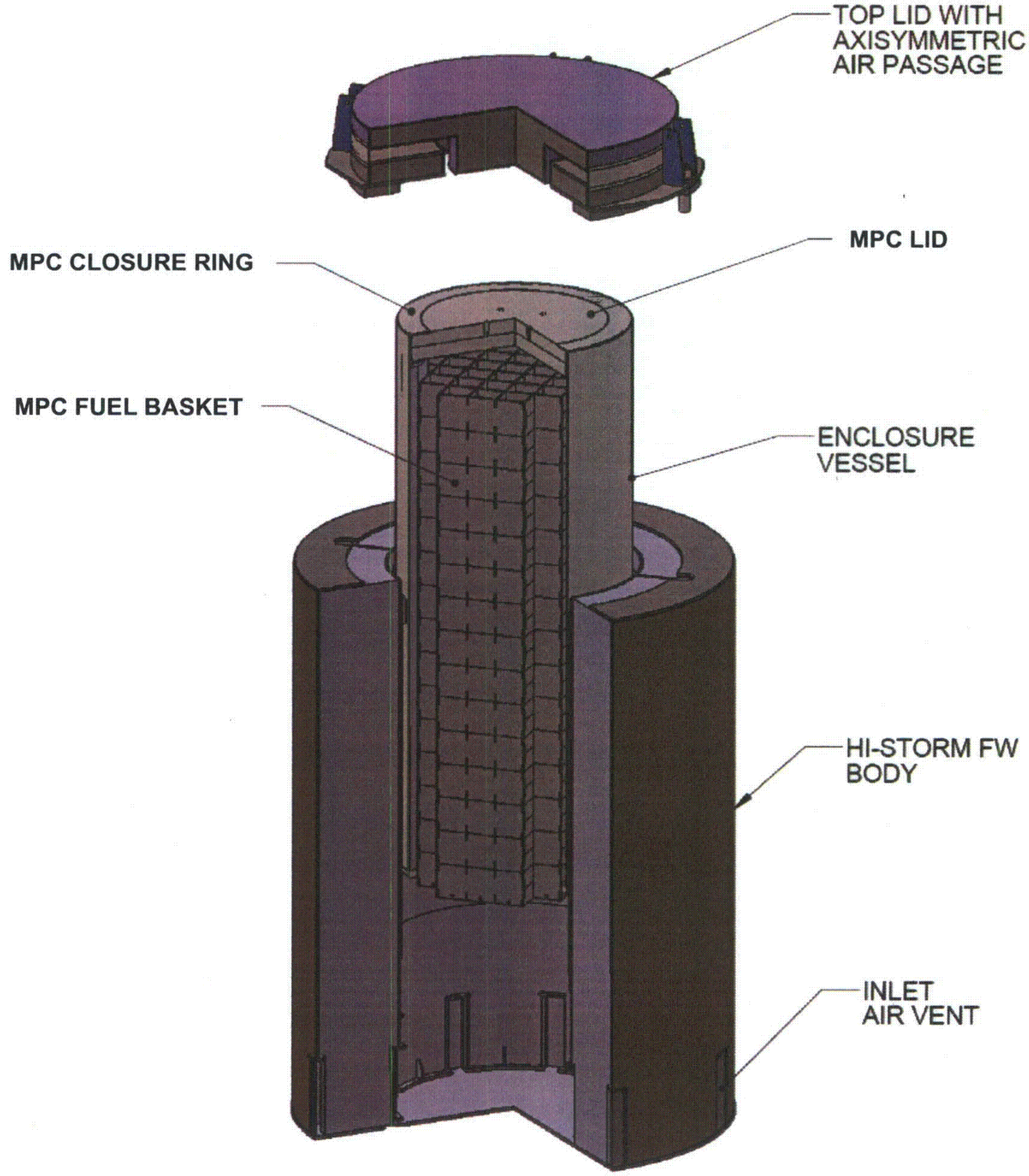


FIGURE 1.1.1: HI-STORM FW VVM AND MPC-37 IN CUTAWAY VIEW

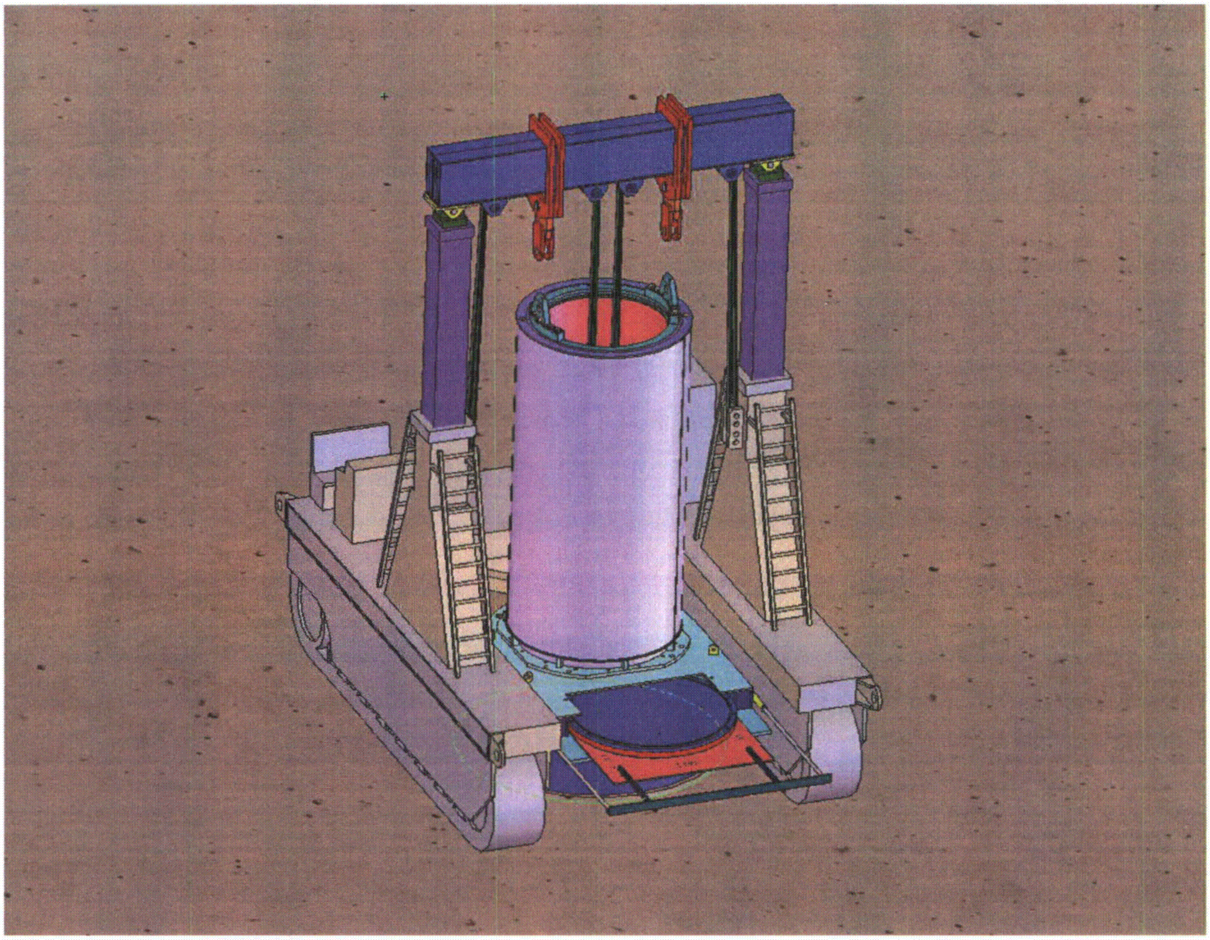


FIGURE 1.1.2: MPC TRANSFER AT THE CANISTER TRANSFER FACILITY (PIT)

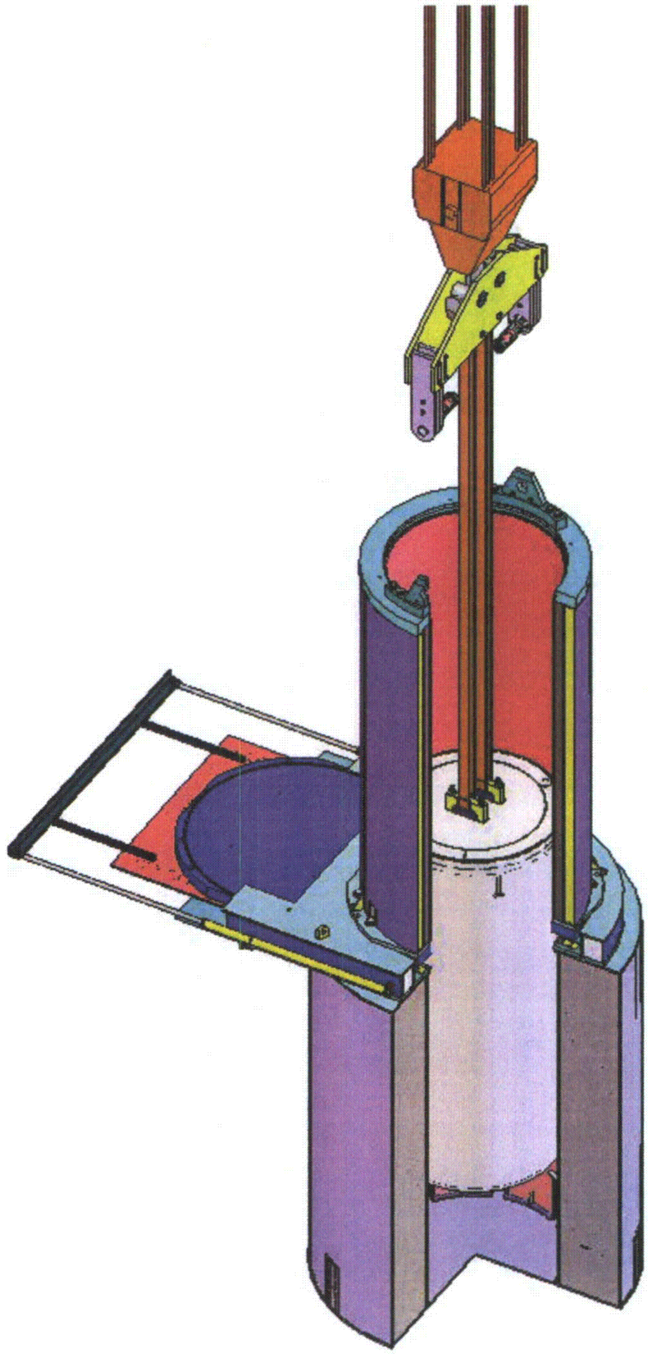


FIGURE 1.1.3: MPC TRANSFER IN THE PLANT'S EGRESS BAY

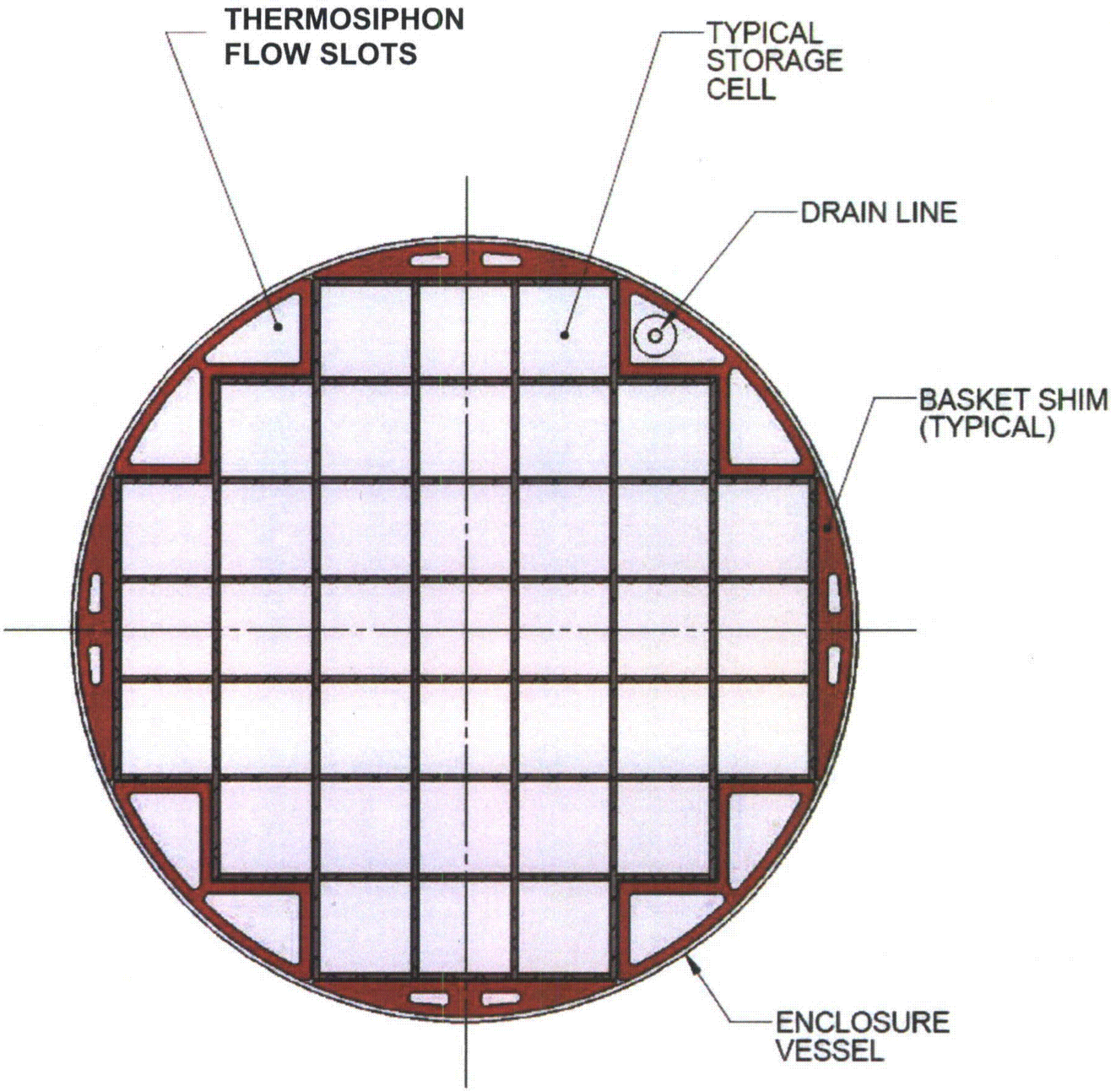


FIGURE 1.1.4: MPC-37 IN CROSS SECTION

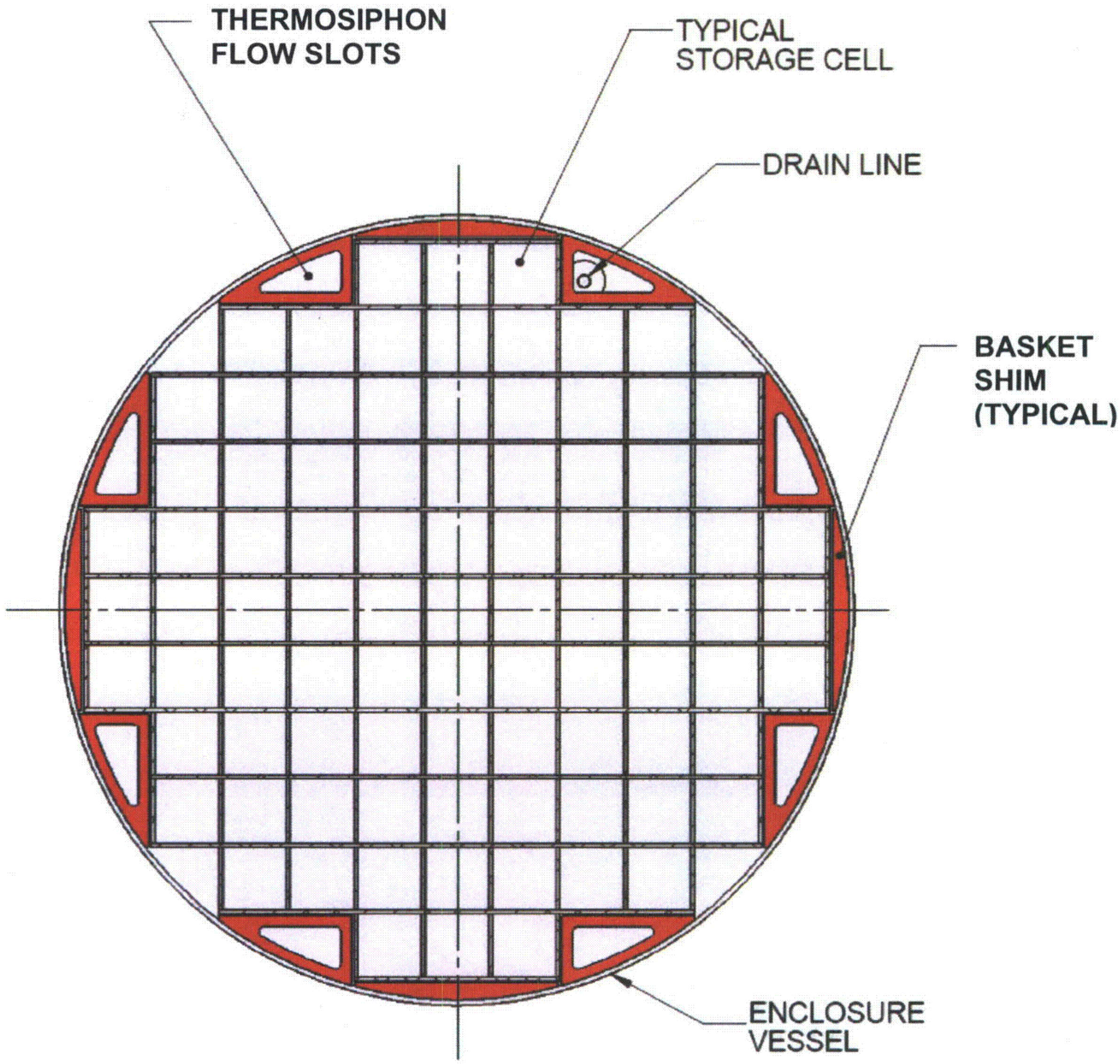


FIGURE 1.1.5: MPC-89 IN CROSS SECTION

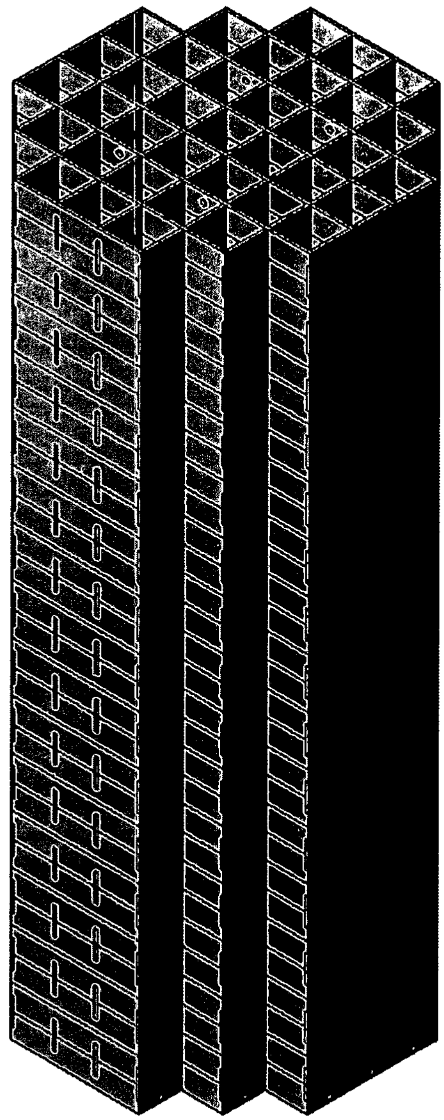


FIGURE 1.1.6: PWR FUEL BASKET (37 STORAGE CELLS) IN PERSPECTIVE VIEW

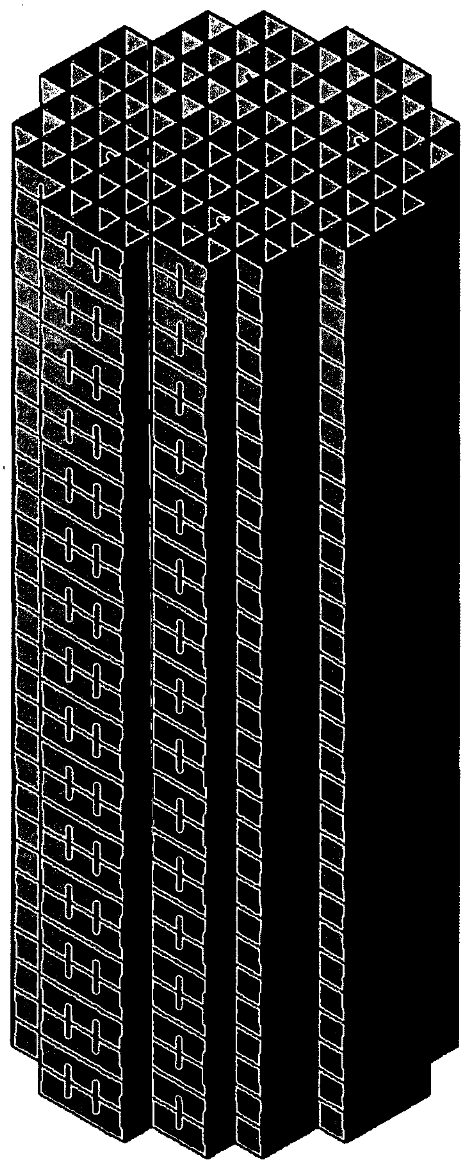


FIGURE 1.1.7: BWR FUEL BASKET (89 STORAGE CELLS) IN PERSPECTIVE VIEW

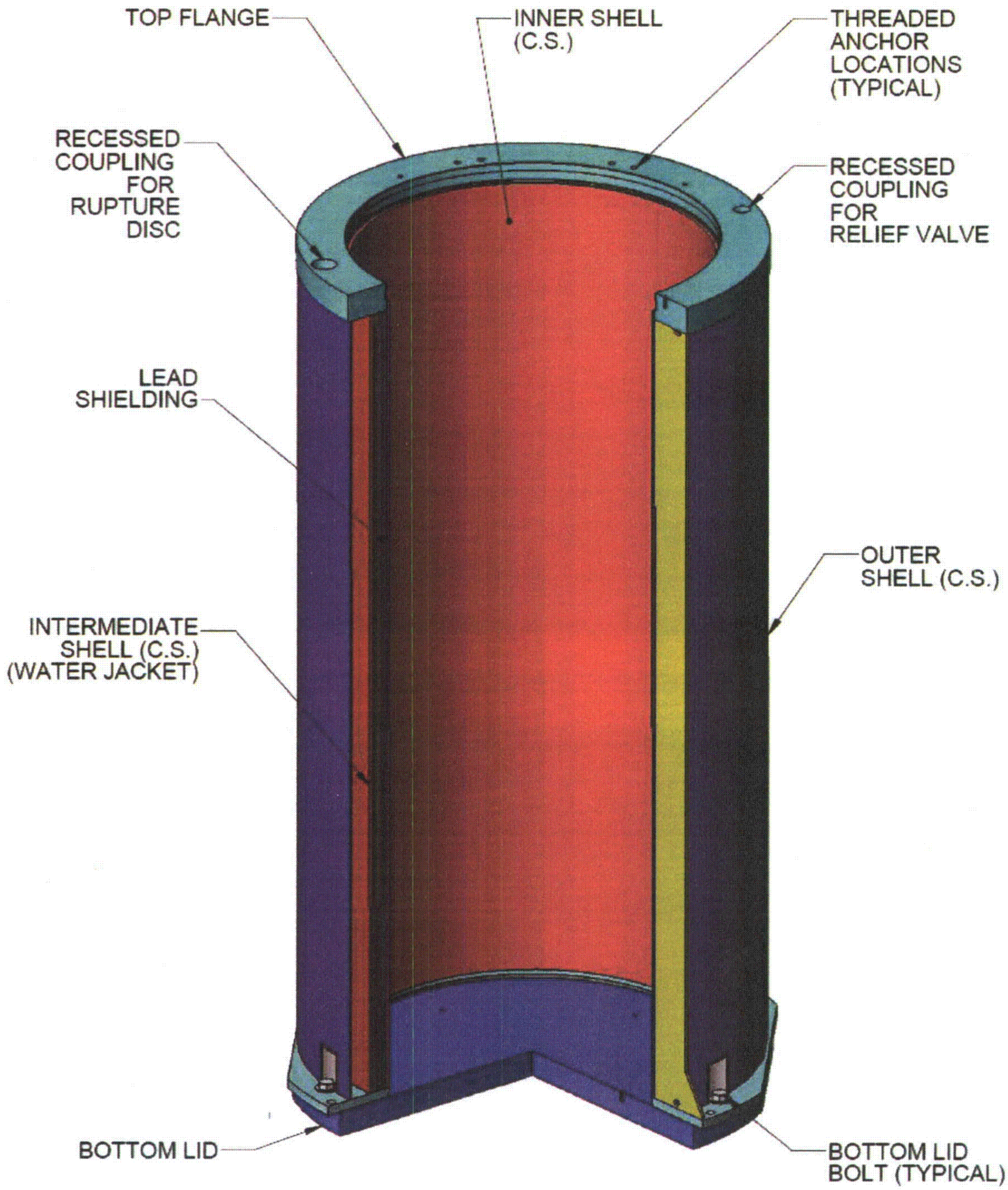


FIGURE 1.1.8: CUTAWAY VIEW OF HI-TRAC VW

1.2 GENERAL DESCRIPTION OF HI-STORM FW SYSTEM

1.2.1 System Characteristics

The HI-STORM FW System consists of interchangeable MPCs, which maintain the configuration of the fuel and is the confinement boundary between the stored spent nuclear fuel and the environment; and a storage overpack that provides structural protection and radiation shielding during long-term storage of the MPC. In addition, a transfer cask that provides the structural and radiation protection of an MPC during its loading, unloading, and transfer to the storage overpack is also subject to certification by the USNRC. Figure 1.1.1 provides a cross sectional view of the HI-STORM FW System with an MPC inserted into HI-STORM FW. Both casks (storage overpack and transfer cask) and the MPC are described below. The description includes information on the design details significant to their functional performance, fabrication techniques and safety features. All structures, systems, and components of the HI-STORM FW System, which are identified as Important-to-Safety (ITS), are specified on the licensing drawings provided in Section 1.5.

There are three types of components subject to certification in the HI-STORM FW docket (see Table 1.0.1).

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

A listing of the common ancillaries not subject to certification but which may be needed by the host site to implement this system is provided in Table 9.2.1.

To ensure compatibility with the HI-STORM FW overpack, MPCs have identical external diameters. Due to the differing storage contents of each MPC, the loaded weight differs among MPCs (see Table 3.2.4 for loaded MPC weight data). Tables 1.2.1 and 1.2.2 contain the key system data and parameters for the MPCs.

The HI-STORM FW System shares certain common attributes with the HI-STORM 100 System, Docket No. 72-1014, namely:

i. the honeycomb design of the MPC fuel basket;
ii. the effective distribution of neutron and gamma shielding materials within the system;
iii. the high heat dissipation capability;.
iv. the engineered features to promote convective heat transfer by passive means;
v. a structurally robust steel-concrete-steel overpack construction.

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flange egg-crate structure where all structural elements (i.e., cell 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 to promote conduction of heat.

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 this section, along with fabrication and safety feature information.

1.2.1.1 Multi-Purpose Canisters

The MPC enclosure vessels are cylindrical weldments with identical and fixed outside diameters. Each MPC is an assembly consisting of a honeycomb fuel basket (Figures 1.1.6 and 1.1.7), a baseplate, a canister shell, a lid, and a closure ring. The number of SNF storage locations in an MPC depends on the type of fuel assembly (PWR or BWR) to be stored in it.

Subsection 1.2.3 and Table 1.2.1 summarize the allowable contents for each MPC model listed in Table 1.0.1. Subsection 2.1.8 provides the detailed specifications for the contents authorized for storage in the HI-STORM FW System. Drawings for the MPCs are provided in Section 1.5.

The MPC enclosure vessel is a fully welded enclosure, which provides the confinement for the stored fuel and radioactive material. The MPC baseplate and shell are made of stainless steel (Alloy X, see Appendix 1.A). The lid is a two piece construction, with the top structural portion made of Alloy X. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring.

The HI-STORM FW System MPCs shares external and internal features with the HI-STORM 100 MPCs certified in the §72-1014 docket, as summarized below.

i. MPC-37 and MPC-89 have an identical enclosure vessel which mimics the enclosure vessel design details used in the HI-STORM 100 counterparts including the shell thickness, the vent and drain port sizes, construction details of the top lid and closure ring, and closure weld details. The baseplate is made slightly thicker to ensure its bending rigidity is

comparable to its counterpart in the HI-STORM 100 system. The material of construction of the pressure retaining components is also identical (options of austenitic stainless steels, denoted as Alloy X, is explained in Appendix 1.A herein as derived from the HI-STORM 100 FSAR with appropriate ASME Code edition updates). There are no gasketed joints in the MPCs.

ii. The top lid of the MPCs contains the same attachment provisions for lifting and handling the loaded canister as the HI-STORM 100 counterparts.

iii. The drain pipe and sump in the bottom baseplate of the MPCs (from which the drain pipe extracts the water during the dewatering operation) are also similar to those in the HI-STORM 100 counterparts.

iv. The fuel basket is assembled from a rectilinear gridwork of plates so that there are no bends or radii at the cell corners. This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls which transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (such as non-mechanistic tipover). This structural feature is shared with the HI-STORM 100 counterparts. Figures 1.1.6 and 1.1.7 show the PWR and BWR fuel baskets, respectively, in perspective view.

v. Precision extruded and/or machined blocks of aluminum alloy with axial holes (basket shims) are installed in the peripheral space between the fuel basket and the enclosure vessel to provide conformal contact surfaces between the basket shims and the fuel basket and between the basket shims and the enclosure vessel shell. The axial holes in the basket shims serve as the passageway for the downward flow of the helium gas under the thermosiphon action. This thermosiphon action is common to all MPCs including those of the HI-STORM 100.

vi. To facilitate an effective convective circulation inside the MPC, the operating pressure is set the same as that in the HI-STORM 100 counterparts.

vii. Like the high capacity baskets in the HI-STORM 100 MPCs, the fuel baskets do not contain flux traps.

Because of the above commonalities, the HI-STORM FW System is loaded in the same manner as the HI-STORM 100 system, and will use similar ancillary equipment, (e.g., lift attachments, lift yokes, lid welding machine, weld removal machine, cask transporter, mating device, low profile transporter or zero profile transporter, drying system, the hydrostatic pressure test system).

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

The MPC incorporates a redundant closure system. The MPC lid is edge-welded (welds are depicted in the licensing drawing in Section 1.5) to the MPC outer shell. The lid is equipped with vent and drain ports that are utilized to remove moisture from the MPC and backfill the MPC with a specified amount of inert gas (helium). The vent and drain ports are closed tight and covered with a port cover (plate) that is seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid; it covers the MPC lid-to shell weld and the vent and drain port cover plates. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the suitably sized threaded anchor locations (TALs) in the MPC lid.

As discussed later in this section, the height of the MPC cavity plays a direct role in setting the amount of shielding available in the transfer cask. To maximize shielding and achieve ALARA within the constraints of a nuclear plant (such as crane capacity), it is necessary to minimize the cavity height of the MPC to the length of the fuel to be stored in it. Accordingly, the height of the MPC cavity is customized for each fuel type listed in Section 2.1. Table 3.2.1 provides the data to set the MPC cavity length as a small adder to the nominal fuel length (with any applicable NFH) to account for manufacturing tolerance, irradiation growth and thermal expansion effects.

For fuel assemblies that are shorter than the MPC cavity length (such as those without a control element in PWR SNF) a fuel shim may be utilized (as appropriate) to reduce the axial gap between the fuel assembly and the MPC cavity. A small axial clearance is provided to account for the irradiation and thermal growth of the fuel assemblies. The actual length of fuel shims (if required) will be determined on a site-specific and fuel assembly-specific basis.

All components of the MPC assembly that may come into contact with spent fuel pool water or the ambient environment are made from stainless steel alloy or aluminum/aluminum alloy materials. Prominent among the aluminum based materials used in the MPC is the Metamic–HT neutron absorber lattice that comprises the fuel basket. As discussed in Chapter 8, concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the HI-STORM FW MPCs. All structural components in an MPC enclosure vessel shall be made of Alloy X, a designation whose origin, as explained in the HI-STORM 100 FSAR [1.1.3], lies in the U.S. DOE's repository program.

As explained in Appendix 1.A, Alloy X (as defined in this FSAR) may be one of the following materials.

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

Any stainless steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed above.

The Alloy X group approach is accomplished by qualifying the MPC for all mechanical, structural, radiological, and thermal conditions using material thermo-physical properties that are the least

favorable for the entire group for the analysis in question. For example, when calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, a material has been defined that is referred to as Alloy X, whose thermo-physical properties, from the MPC design perspective, are the least favorable of the above four candidate materials.

The evaluation of the candidate Alloy X materials to determine the least favorable properties is provided in Appendix 1.A. The Alloy X approach is conservative because no matter which material is ultimately utilized in the MPC construction, it guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

The principal materials used in the manufacturing of the MPC are listed in the licensing drawings (Section 1.5) and the acceptance criteria are provided in Chapter 10. A listing of the fabrication specifications utilized in the manufacturing of HI-STORM FW System components is provided in Tables 1.2.7 and 1.2.8. The specifications, procedures for sizing, forming machining, welding, inspecting, cleaning, and packaging of the completed equipment implemented by the manufacturer on the shop floor are required to conform to the fabrication specification in the above referenced tables.

1.2.1.2 HI-STORM FW Overpack

HI-STORM FW is a vertical ventilated module engineered to be fully compatible with the HI-TRAC VW transfer cask and the MPCs listed in Table 1.0.1. The HI-STORM FW overpack consists of two major parts:

a. A dual wall cylindrical container with a set of inlet ducts near its bottom extremity and an integrally welded baseplate.
b. A removable top lid equipped with a radially symmetric exit vent system.

The HI-STORM FW overpack is a rugged, heavy-walled cylindrical vessel. Figure 1.1.1 provides a pictorial view of the HI-STORM FW overpack with the MPC-37 partially inserted. 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 a steel weldment of cylindrical shells, a thick baseplate, and a top annular plate. A set of four equally spaced radial connectors join the inner and outer shells and define a fixed width annular space for placement of concrete. The overpack lid also has concrete to provide neutron and gamma shielding.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC (Figure 1.1.1) with an annular space between the MPC enclosure vessel and the overpack for ventilation air flow. The upward flowing air in the annular space (drawn from the ambient by a purely passive action), extracts heat from the MPC surface by convective heat transfer. The rate of air flow is governed by the amount of heat in the MPC (i.e., the greater the heat load, the greater the air flow rate).

To maximize the cooling action of the ventilation air stream, the ventilation flow path is optimized

to minimize hydraulic resistance. The HI-STORM FW features eight inlet ducts. Each duct is narrow and tall and of an internally refractive contour which minimizes radiation streaming while optimizing the hydraulic resistance of airflow passages. The inlet air duct design, referred to as the "Radiation Absorbent Duct," is subject to an ongoing action on a provisional Holtec International patent application by the USPTO (ca. March 2009) and is depicted in the licensing drawing in Section 1.5. The Radiation Absorbent Duct also permits the MPC to be placed directly on the baseplate of the overpack instead of on a pedestal that would raise it above the duct.

An array of radial tube-type gussets (MPC guides) welded to the inner shell and the baseplate are shaped to guide the MPC during MPC transfer and ensure it is centered within the overpack. The MPC guides have an insignificant effect on the overall hydraulic resistance of the ventilation air stream. Furthermore, the top array of MPC guides are longitudinally oriented members, sized and aligned to serve as impact attenuators which will crush against the solid MPC lid during an impactive collision, such as a non-mechanistic tip-over scenario.

The height of the storage cavity in the HI-STORM FW overpack is set equal to the height of the MPC plus a fixed amount to allow for thermal growth effects and to provide for adequate ventilation space (low hydraulic resistance) above the MPC (See Table 3.2.1).

The outlet duct is located in the overpack lid (Figure 1.1.1) pursuant to Holtec Patent No. 6,064,710. The outlet duct opening is narrow in height which reduces the radiation streaming path from the contents, however, aside from the minor interference from the support plates, the duct extends circumferentially 360° which significantly increases the flow area and in-turn minimizes hydraulic resistance.

The overpack lid, like the body, is also a steel weldment filled with plain concrete. The lid is equipped with a radial ring welded to its underside which provides additional shielding for the MPC/overpack annulus. The radial ring also serves to center the lid on the overpack body. A third, equally important function of the radial ring is to prevent the lid from sliding across the top surface of the overpack body during a non-mechanistic tip-over event.

Within the ducts, an array of duct photon attenuators (DPAs) may be installed (Holtec Patent No.6,519,307B1) to further decrease the amount of radiation scattered to the environment. These Duct Photo Attenuators (DPAs) are designed to scatter any radiation streaming through the ducts. Scattering the radiation in the ducts reduces the streaming through the overpack penetration resulting in a significant decrease in the local dose rates. The configuration of the DPAs is such that the increase in the resistance to flow in the air inlets and outlets is minimized. The DPAs are not credited in the safety analyses performed in this FSAR, nor are they depicted in the licensing drawings. DPAs can be used at a site if needed to lower site boundary dose rates with an appropriate site-specific engineering evaluation.

Each duct opening is equipped with a heavy duty insect barrier (screen). Routine inspection of the screens or temperature monitoring of the air exiting the outlet ducts is required to ensure that a blockage of the screens is detected and removed in a timely manner. The evaluation of the effects of partial and complete blockage of the air ducts is considered in Chapter 12 of this FSAR.

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 join the overpack inner and outer steel shells. The four anchor blocks are located at 90° angular spacing around the circumference of the top of the overpack body.

Finally, the HI-STORM FW overpack features heat shields engineered to protect the overpack body concrete and the overpack lid concrete from excessive temperature rise due to radiant heat from the MPC. A thin cylindrical steel liner, concentric with the inner shell of the overpack, but slightly smaller in diameter, hangs from the top array of MPC guides. A separate thin steel liner is welded to the underside of the overpack lid. The heat shields are depicted in the licensing drawings in Section 1.5.

The plain concrete between the overpack inner and outer steel shells and the lid is specified to provide the necessary shielding properties (dry density) and compressive strength. The shielding concrete shall be in accordance with the requirements specified in Appendix 1.D of the HI-STORM 100 FSAR [1.1.3] and Table 1.2.5 herein. Commitment to follow the specification of plain concrete in the HI-STORM 100 FSAR in this docket ensures that a common set of concrete placement procedures will be used in both overpack types which will be important for configuration control at sites where both systems may be deployed.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM FW overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. During the postulated fire accident the high thermal inertia characteristics of the HI-STORM FW 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 buttressing the steel shells.

Density and compressive strength are the key parameters that bear upon the performance of concrete in the HI-STORM FW System. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.2.2] are used.

Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM FW overpack concrete.

The principal materials used in the manufacturing of the overpack are listed in the licensing drawings and the acceptance criteria are provided in Chapter 10. Tables 1.2.6 and 1.2.7 provide applicable code paragraphs for manufacturing the HI-STORM FW overpack.

1.2.1.3 HI-TRAC VW Transfer Cask

The HI-TRAC VW transfer cask (Figure 1.1.8) is engineered to be used to perform all short-term loading operations on the MPC beginning with fuel loading and ending with the emplacement of the MPC in the storage overpack. The HI-TRAC VW is also used for short term unloading operations beginning with the removal of the MPC from the storage overpack and ending with fuel unloading.

HI-TRAC VW is designed to meet the following specific performance objectives that are centered on ALARA and physical safety of the plant's operations staff.

a. Provide maximum shielding to the plant personnel engaged in conducting short-term operations.
b. Provide protection of the MPC against extreme environmental phenomena loads, such as tornado-borne missiles, during short-term operations.
c. Serve as the container equipped with the appropriate lifting appurtenances in accordance with ANSI N14.6 [1.2.3] to lift, move, and handle the MPC, as required, to perform the short-term operations.
d. Provide the means to restrain the MPC from sliding and protruding beyond the shielding envelope of the transfer cask under a (postulated) handling accident.
e. Facilitate the transfer of a loaded MPC to or from the HI-STORM FW overpack (or another physically compatible storage or transfer cask) by vertical movement of the MPC without any risk of damage to the canister by friction.

The above performance demands on the HI-TRAC VW are met by its design configuration as summarized below and presented in the licensing drawings in Section 1.5.

HI-TRAC VW is principally made of carbon steel and lead. The cask consists of two major parts, namely (a) a multi-shell cylindrical cask body, and (b) a quick connect/disconnect bottom lid. The cylindrical cask body is made of three concentric shells joined to a solid annular top flange and a solid annular bottom flange by circumferential welds. The innermost and the middle shell are fixed in place by longitudinal ribs which serve as radial connectors between the two shells. The radial connectors provide a continuous path for radial heat transfer and render the dual shell configuration into a stiff beam under flexural loadings. The space between these two shells is occupied by lead, which provides the bulk of the transfer cask's gamma radiation shielding capability and accounts for a major portion of its weight.

Between the middle shell and the outermost shell is the weldment that is referred to as the "water jacket." The water jacket is filled with water and may contain ethylene glycol fortified water, if warranted by the environmental conditions at the time of use. The water jacket provides most of the neutron shielding capability to the cask. The water jacket is outfitted with pressure relief devices to prevent over-pressurization in the case of an off-normal or accident event that causes the water mass inside of it to boil.

The water in the water jacket serves as the neutron shield when required. When the cask is being removed from the pool and the MPC is full of water, the water jacket can be empty. This will minimize weight, if for example, crane capacities are limited, since the water within the MPC cavity is providing the neutron shielding during this time. However, the water jacket must be filled before the MPC is emptied of water. This keeps the load on the crane (i.e., weight of the loaded transfer cask) nearly constant between the lifts before and after MPC processing. Furthermore, the amount of shielding provided by the transfer cask is maximized at all times within crane capacity constraints. The water jacket concept is disclosed in a Holtec Patent [6,587,536 B1].

As the description of loading operations in Chapter 9 of this FSAR indicates, most of the human activities occur near the top of the transfer cask. Therefore, the geometry of the transfer cask is

configured to maximize shielding by eliminating penetrations and discontinuities such as lifting trunnions. Instead, the HI-TRAC VW is lifted using a pair of lift blocks that are anchored into the top forging of the transfer cask using a set of high strength bolts. A device which prevents the MPC from sliding out of the transfer cask is attached to the lift blocks.

The bottom of the transfer cask is equipped with a thick lid. It is provided with a gasket seal against the machined face of the bottom flange creating a watertight (open top) container. A set of bolts that tap into the machined holes in the bottom lid provide the required physical strength to meet the structural imperatives of ANSI N14.6 and as well as bolt pull to maintain joint integrity. The bottom lid can be fastened and released from the cask body by accessing its bolts from above the transfer cask bottom flange, which is an essential design feature to permit MPC transfer operations described in Chapter 9.

To optimize the shielding in the body of HI-TRAC VW, two design strategies have been employed;

1. The height of the HI-TRAC's cavity is set to its optimal value (slightly greater than the MPC height as specified in Table 3.2.1), therefore allowing more shielding to be placed in the radial direction of the transfer cask.
2. The thickness of the lead in the transfer cask shall be customized for the host site. The thickness of the lead cylinder can be varied within the limits given in Table 3.2.2. The nominal radial thickness of the water jacket is fixed and therefore the outside diameter of the HI-TRAC will vary accordingly.

The above design approach permits the quantity of shielding around the body of the transfer cask to be maximized for a given length and weight of fuel in keeping with the practices of ALARA. At some host sites, a lead thickness greater than allowed by Table 3.2.2 may be desirable and may be feasible but will require a site-specific safety evaluation.

The use of the suffix VW in the HI-TRAC's designation is intended to convey this **V**ariable **W**eight feature incorporated by changing the HI-TRAC height and lead thickness to best accord with the MPC height and plant's architecture. Table 3.2.6 provides the operating weight data for a HI-TRAC VW when handling the Reference PWR and BWR fuel in Table 1.0.4.

The principal materials used in the manufacturing of the transfer cask are listed in the licensing drawings and the acceptance criteria are provided in Chapter 10. Tables 1.2.6 and 1.2.7 provide applicable code paragraphs for manufacturing the HI-TRAC VW.

1.2.1.4 Shielding Materials

Steel and concrete are the principal shielding materials in the HI-STORM FW overpack. The steel and concrete shielding materials in the lid provide additional gamma attenuation to reduce both direct and skyshine radiation. The combination of these shielding materials ensures that the radiation and exposure objectives of 10CFR72.104 and 10CFR72.106 are met.

Steel, lead, and water are the principal shielding materials in the HI-TRAC transfer cask. The

combination of these three shielding materials ensures that the radiation and exposure objectives of 10CFR72.106 and ALARA are met. The extent and location of shielding in the transfer cask plays an important role in minimizing the personnel doses during loading, handling, and transfer.

The MPC fuel basket structure provides the initial attenuation of gamma and neutron radiation emitted by the radioactive contents. The MPC shell, baseplate, and thick lid provide additional gamma attenuation to reduce direct radiation.

1.2.1.4.1 Neutron Absorber – Metamic HT

Metamic-HT is the designated neutron absorber in the HI-STORM FW MPC baskets. It is also the structural material of the basket. The properties of Metamic-HT and key characteristics, necessary for ensuring nuclear reactivity control, thermal, and structural performance of the basket, are presented in Appendix 1.B.

1.2.1.4.2 Neutron Shielding

Neutron shielding in the HI-STORM FW overpack is provided by the thick walls of concrete contained inside the steel vessel and the top lid. 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 under long term temperatures required for SNF storage.

The specification of the HI-STORM FW overpack neutron shielding 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 (i.e. under normal condition 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. Final specification of a shield material is a result of optimizing the material properties with respect to the above criteria, along with the design of the shield system, to achieve the desired shielding results.

The HI-TRAC VW transfer cask is equipped with a water jacket providing radial neutron shielding. The water in the water jacket may be fortified with ethylene glycol to prevent freezing under low

temperature operations [1.2.4].

During certain evolutions in the short term handling operations, the MPC may contain water which will supplement neutron shielding.

1.2.1.4.3 Gamma Shielding Material

Gamma shielding in the HI-STORM FW storage overpack is primarily provided by massive concrete sections contained in the robust steel vessel. The carbon steel in the overpack supplements the concrete gamma shielding. To reduce the radiation streaming through the overpack penetrations, duct photon attenuators may be installed (as discussed previously in section 1.2.1.2) to further decrease radiation streaming from the ducts.

In the HI-TRAC VW 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 VW transfer cask.

In the MPC, the gamma shielding is provided by its stainless steel enclosure vessel (including a thick lid); and its aluminum based fuel basket and aluminum alloy basket shims.

1.2.1.5 Lifting Devices

Lifting and handling of the loaded HI-STORM FW overpack is carried out in the vertical upright configuration using the threaded anchor blocks arranged circumferentially at 90° spacing around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment. The overpack anchor blocks are integral to the overpack and designed in accordance with Regulatory Guide 3.61. All lifting appurtenances used with the HI-STORM FW overpack are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

Like the storage overpack, the loaded transfer cask is also lifted using a specially engineered appurtenance denoted as the lift block in Table 9.1.2 and Figure 9.2.1. The top flange of the transfer cask is equipped with threaded holes that allow lifting of the loaded HI-TRAC in the vertical upright configuration. These threaded lifting holes are integral to the transfer cask and are designed in accordance with NUREG 0612. All lifting appurtenances used with the HI-TRAC VW are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

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 and/or lowered through the HI-TRAC VW transfer cask using lifting attachments (functional equivalent of the lift blocks used with HI-TRAC VW). The threaded holes in the MPC lid are integral to the MPC and designed in accordance with NUREG 0612. All lifting appurtenances used with the MPC are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

1.2.1.6 Design Life

The design life of the HI-STORM FW System is 60 years. This is accomplished by using materials 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 (see Chapter 8). A maintenance program, as specified in Chapter 10, is also implemented to ensure the service life of the HI-STORM FW System will exceed its design life of 60 years. The design considerations that assure the HI-STORM FW System performs as designed include the following:

<u>HI-STORM FW Overpack and HI-TRAC VW Transfer Cask</u>

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

<u>MPCs</u>

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

The adequacy of the HI-STORM FW System materials for its design life is discussed in Chapter 8. Transportability considerations pursuant to 10CFR72.236(m) are discussed in Section 2.4.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The design features of the HI-STORM FW System, described in Subsection 1.2.1 in the foregoing, are intended to meet the following principal performance characteristics under all credible modes of operation:

(a) Maintain subcriticality
(b) Prevent unacceptable release of contained radioactive material
(c) Minimize occupational and site boundary dose
(d) Permit retrievability of contents (fuel must be retrievable from the MPC under normal and off-normal conditions in accordance with ISG-2 and the MPC must be recoverable after accident conditions in accordance with ISG-3)

Chapter 11 identifies the many design features built into the HI-STORM FW System to minimize dose and maximize personnel safety. Among the design features intrinsic to the system that facilitate meeting the above objectives are:

i. The loaded HI-STORM FW overpack and loaded HI-TRAC VW transfer cask are

always maintained in a vertical orientation during handling.

ii. The height of the HI-STORM FW overpack and HI-TRAC VW transfer cask is minimized consistent with the length of the SNF. This eliminates the need for major structural modifications at the plant and/or eliminates operational steps that impact ALARA.

iii. The extent of shielding in the transfer cask is maximized at each plant within the crane and architectural limitations of the plant by minimizing the height in accordance with the length of the SNF to permit additional shielding material in the walls of the transfer cask.

iv. The increased number of inlet ducts and the circumferential outlet vents in HI-STORM FW overpack are configured to make the thermal performance less susceptible to wind.

v. Tall and narrow inlet ducts in the HI-STORM FW overpack in conjunction with the thermosiphon action in the MPC design, render the HI-STORM FW System more resistant to a thermally adverse flood condition (Section 2.2).

vi. The design of the HI-STORM FW affords the user the flexibility to utilize higher density concrete than the minimum prescribed value in Table 1.2.5 to further reduce the site boundary dose.

The HI-STORM FW overpack utilizes the same cross-connected dual steel shell configuration used in other HI-STORM models. The dual shell steel weldment with an integrally connected baseplate forms a well defined annulus wherein plain concrete of the desired density is installed. While both steel and concrete in the overpack body are effective in neutron and gamma shielding, the principal role of the radially conjoined steel shell is to provide the structural rigidity to support the mass of the shielding concrete. As calculations in Chapter 3 show, the dual steel shell structure can support the mass of concrete of any available density with ample margin of safety. Consequently, the mass of concrete utilized to shield against the stored fuel is only limited by the density of the available aggregate. Users of HI-STORM 100 systems have used concrete of density approaching 200 lb/ft^3 to realize large dose reductions at ISFSIs to support site specific considerations.

The above comment also applies to the HI-STORM FW overpack lid, which is a massive steel weldment made of plate and shell segments filled with shielding concrete. The steel in the lid, while contributing principally to gamma shielding, provides the needed structural capacity. Concrete performs as a missile barrier and is critical to minimizing skyshine. High density concrete can also used in the HI-STORM FW overpack lid if reducing skyshine is a design objective at a plant.

The site boundary dose from the HI-STORM FW System is minimized by using specially shaped ducts at the bottom of the overpack and in the lid. The ducts and the annular space between the stored MPC and the HI-STORM FW cavity serve to promote ventilation of air to reject the MPC's decay heat to the environment.

The criticality control features of the HI-STORM FW 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.

1.2.2.2 Sequence of Operations

A summary sequence of loading operations necessary to defuel a spent fuel pool using the HI-STORM FW System (shown with MPC Transfer in the plant's Egress Bay) is shown in a series of diagrams in Figure 1.2.3. The loading sequence underscores the inherent simplicity of the loading evolutions and its compliance with ALARA. A more detailed sequence of steps for loading and handling operations is provided in Chapter 9, aided by illustrative figures, to serve as the guidance document for preparing site-specific implementation procedures.

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 entire basket is made of Metamic-HT, a uniform dispersoid of boron carbide and nano-particles of alumina in an aluminum matrix, serves as the neutron absorber. This accrues four major safety and reliability advantages:

(i) The larger B-10 areal density in the Metamic-HT allows higher enriched fuel (i.e., BWR fuel with planar average initial enrichments greater than 4.5 wt% U-235) without relying on gadolinium or burn-up credit.
(ii) The neutron absorber cannot be removed from the basket or displaced within it.
(iii) Axial movement of the fuel with respect to the basket has no reactivity consequence because the entire length of the basket contains the B-10 isotope.
(iv) The larger B-10 areal density in the Metamic-HT reduces the reliance on soluble boron credit during loading/unloading of PWR fuel.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM FW System. A detailed evaluation is provided in Section 3.4.

1.2.2.3.3 Operation Shutdown Modes

The HI-STORM FW System is totally passive and consequently, operation shutdown modes are unnecessary.

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM FW MPC, which is seal welded, non-destructively examined, and pressure tested, confines the radioactive contents. The HI-STORM FW is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any

instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM FW overpack exit vents in lieu of routinely inspecting the vents for blockage.

1.2.2.3.5 Maintenance Technique

Because of its passive nature, the HI-STORM FW System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 10 describes the maintenance program set forth for the HI-STORM FW System.

1.2.3 Cask Contents

This sub-section contains information on the cask contents pursuant to 10 CFR72, paragraphs 72.2(a)(1),(b) and 72.236(a),(c),(h),(m).

The HI-STORM FW System is designed to house both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key system data and parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the Glossary. All fuel assemblies, non-fuel hardware, and neutron sources authorized for packaging in the MPCs must meet the fuel specifications provided in Section 2.1. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers (DFC).

As shown in Figure 1.2.1 (MPC-37) and Figure 1.2.2 (MPC-89), each storage location is assigned to one of three regions, denoted as Region 1, Region 2, and Region 3 with an associated cell identification number. For example, cell identified as 2-4 is Cell 4 in Region 2. A DFC can be stored in the outer peripheral locations of both MPC-37 and MPC-89 as shown in Figures 2.1.1 and 2.1.2, respectively. The permissible heat loads for each cell, region, and the total canister are given in Tables 1.2.3 and 1.2.4 for MPC-37 and MPC-89, respectively.

TABLE 1.2.1

KEY SYSTEM DATA FOR HI-STORM FW SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs	2	1 for PWR 1 for BWR
MPC storage capacity†:	MPC-37	Up to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37.
MPC storage capacity†:	MPC-89	Up to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 89.

† See Chapter 2 for a complete description of authorized cask contents and fuel specifications.

TABLE 1.2.2

KEY PARAMETERS FOR HI-STORM FW MULTI-PURPOSE CANISTERS

Parameter	PWR	BWR
Pre-disposal service life (years)	100	100
Design temperature, max./min. (°F)	752†/-40††	752†/-40††
Design internal pressure (psig) Normal conditions Off-normal conditions Accident Conditions	 100 120 200	 100 120 200
Total heat load, max. (kW)	See Table 1.2.3	See Table 1.2.4
Maximum permissible peak fuel cladding temperature: Long Term Normal (°F) Short Term Operations (°F) Off-normal and Accident (°F)	 752 752 or 1058††† 1058	 752 752 or 1058††† 1058
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
B_4C content (by weight) (min.) in the Metamic-HT Neutron Absorber (storage cell walls)	10%	10%
End closure(s)	Welded	Welded
Fuel handling	Basket cell openings compatible with standard grapples	Basket cell openings 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 and no fuel decay heat load.

††† See Section 4.5 for discussion of the applicability of the 1058°F temperature limit during short-term operations, including MPC drying.

TABLE 1.2.3

MPC-37 HEAT LOAD DATA (See Figure 1.2.1)

Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 47.05			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	1.13	9	10.17
2	1.78	12	21.36
3	0.97	16	15.52

Note: See Chapter 4 for decay heat limits per cell when loading high burnup fuel and using vacuum drying of the MPC.

TABLE 1.2.4

MPC-89 HEAT LOAD DATA (See Figure 1.2.2)

Number of Regions:	3		
Number of Storage Cells:	89		
Maximum Heat Load:	46.36 kW		
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.44	9	3.96
2	0.62	40	24.80
3	0.44	40	17.60

Note: See Chapter 4 for decay heat limits per cell when loading high burnup fuel and using vacuum drying of the MPC.

TABLE 1.2.5 CRITICALITY AND SHIELDING SIGNIFICANT SYSTEM DATA		
Item	**Property**	**Value**
Metamic-HT Neutron Absorber	Nominal Thickness (mm)	10 (MPC-89) 15 (MPC-37)
	Minimum B_4C Weight %	10 (MPC-89) 10 (MPC-37)
Concrete in HI-STORM FW overpack body and lid	Installed Nominal Density (lb/ft^3)	150 (reference) 200 (maximum)

TABLE 1.2.6
REFERENCE ASME CODE PARAGRAPHS FOR HI-STORM FW OVERPACK and HI-TRAC VW TRANSFER CASK, PRIMARY LOAD BEARING PARTS

	Item	Code Paragraph†	Notes, Explanation and Applicability
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The "intervening elements" are termed interfacing SSCs in this FSAR.
3.	Certification of material	NF-2130 (b) and (c)	Materials for ITS components shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2440, NF-4411	-
6.	Welding procedure specification	Section IX	Acceptance Criteria per Subsection NF
7.	Welding material	Section II	-
8.	Definition of Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3124	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 linear structures is applicable for overpack top lid and the overpack and transfer cask shells.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	All cylindrical parts.
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural welds
19.	Backing Strips, Peening	NF-4421 NF-4422	Applies to structural and non-structural welds
20.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
21.	Non-Destructive Examination	NF-5360	Invokes Section V, Applies to Code welds only
22.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	Applies to Code welds only

† All references to the ASME Code refer to applicable sections of the 2007 edition.

	TABLE 1.2.7 SUMMARY REQUIREMENTS FOR MANUFACTURING OF HI-STORM FW SYSTEM COMPONENTS			
	Item	**MPC**	**HI-STORM FW**	**HI-TRAC VW Transfer Cask**
1.	Material Specification	NB-2000 and ASME Section II	ASME Section II	ASME Section II
2.	Pre-welding operations (viz., cutting, forming, and machining)	NB-4000	Holtec Standard Procedures (HSPs)	Holtec Standard Procedures (HSPs)
3.	Weld wire	NB-2000 and ASME Section II	ASME Section II	ASME Section II
4.	Welding Procedure specifications and reference code for acceptance criteria	ASME Section IX and NB-4000	ASME Section IX and ASME Section III, Subsection NF	ASME Section IX
5.	NDE Procedures and reference code for acceptance criteria	ASME Section V, Subsection NB	ASME Section V, Subsection NF	ASME Section V, Subsection NF
6.	Qualification Protocol for Inspection Personnel	SNT-TC-1A	SNT-TC-1A	SNT-TC-1A
7.	Cleaning	ANSI N45.2.1 Section 2	ANSI N45.2.1 Section 2	ANSI N45.2.1 Section 2
8.	Packaging & Shipping	ANSI N45.2.2	ANSI N45.2.2	ANSI N45.2.2
9.	Mix or Plain Concrete	N/A	ACI 318 (2005)	N/A
10.	Inspection and Acceptance	Section 1.5 Drawings and Chapter 10	Section 1.5 Drawings and Chapter 10	Section 1.5 Drawings and Chapter 10
11.	Quality Procedures	Holtec Quality Assurance Procedures Manual	Holtec Quality Assurance Procedures Manual	Holtec Quality Assurance Procedures Manual

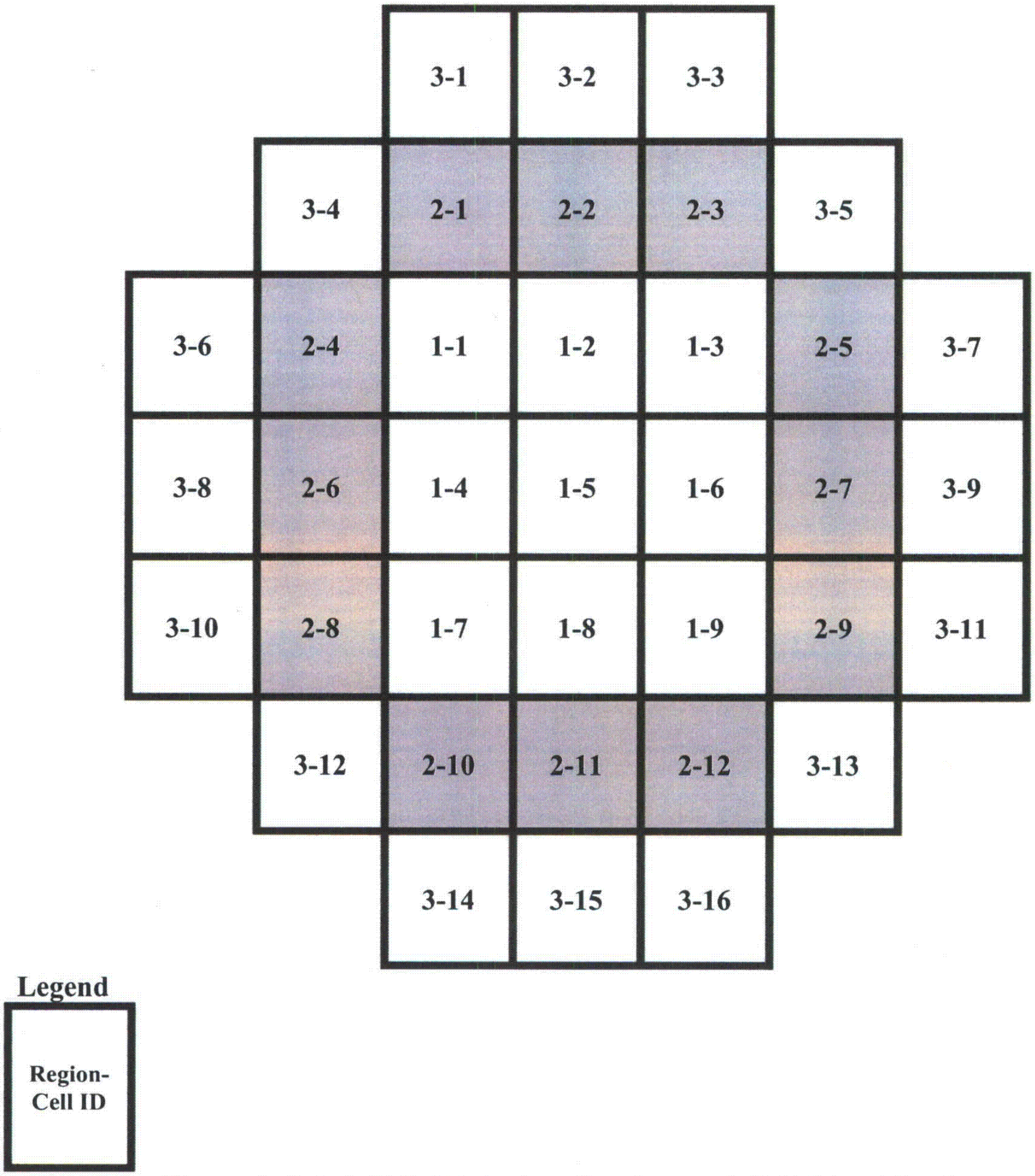


Figure 1.2.1: MPC-37 Basket, Region and Cell Identification

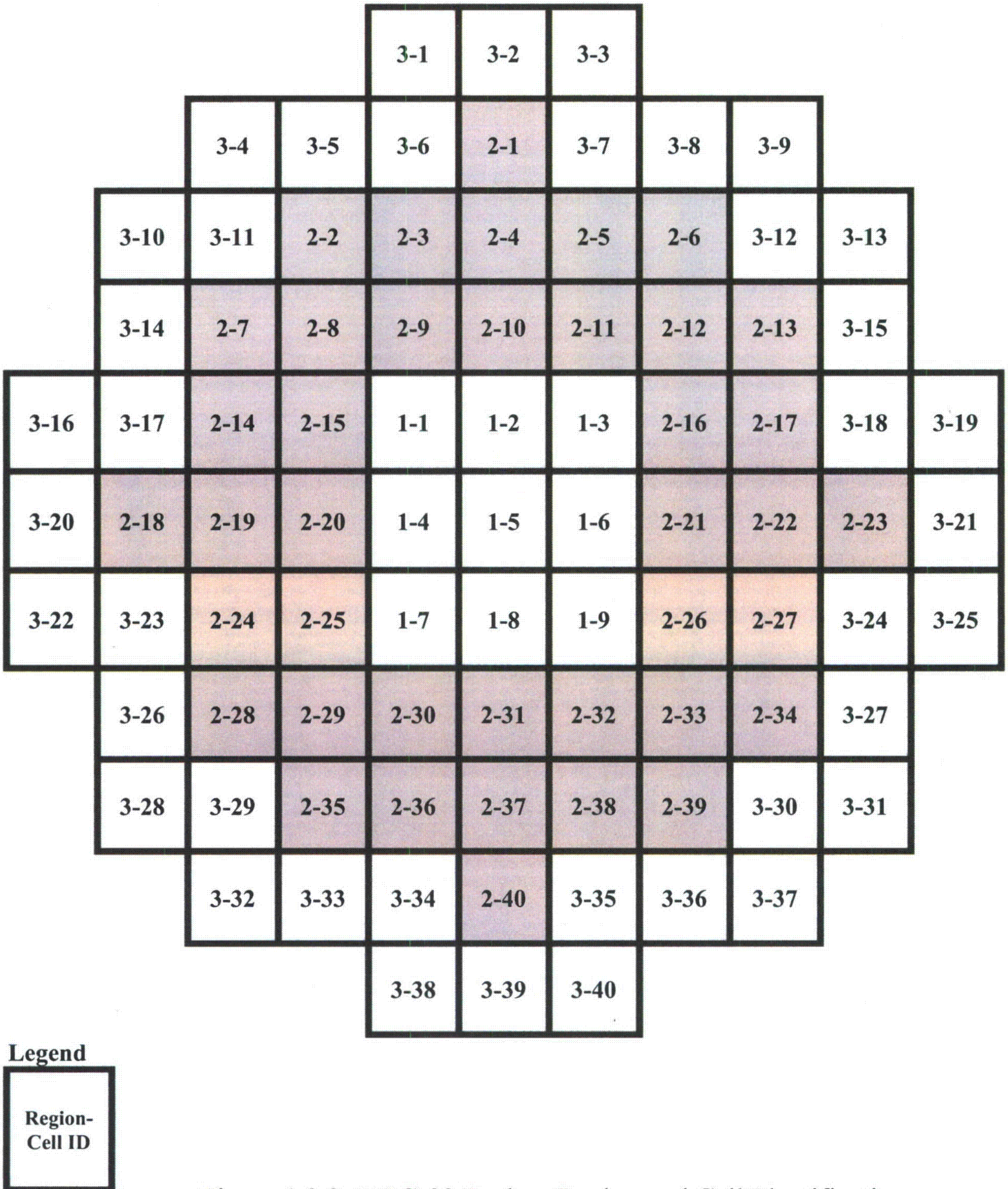
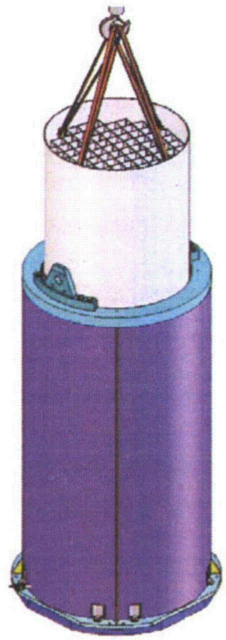
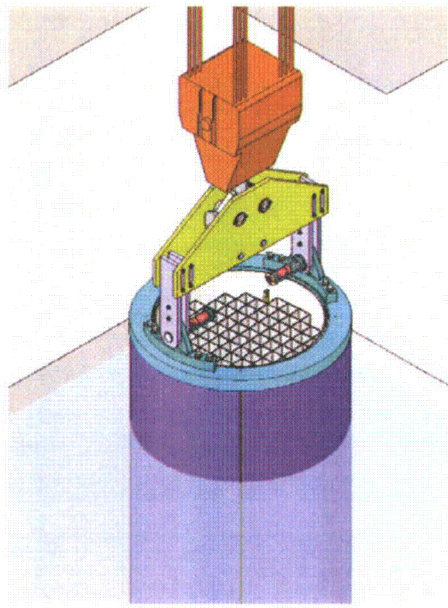


Figure 1.2.2: MPC-89 Basket, Region and Cell Identification

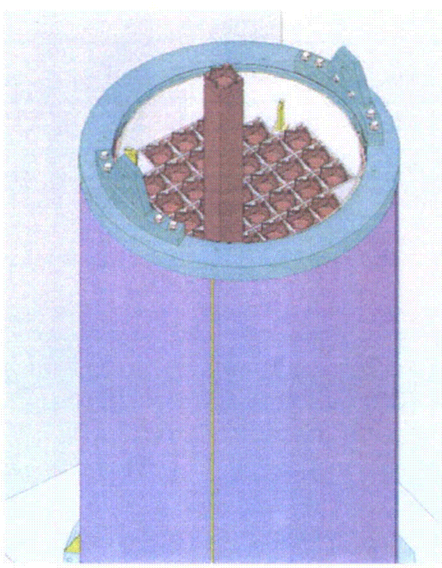
1. MPC Placed in HI-TRAC



2. HI-TRAC Placement in the spent fuel pool



3. Fuel Loading into the MPC



4. HI-TRAC/MPC Removal from the spent fuel pool

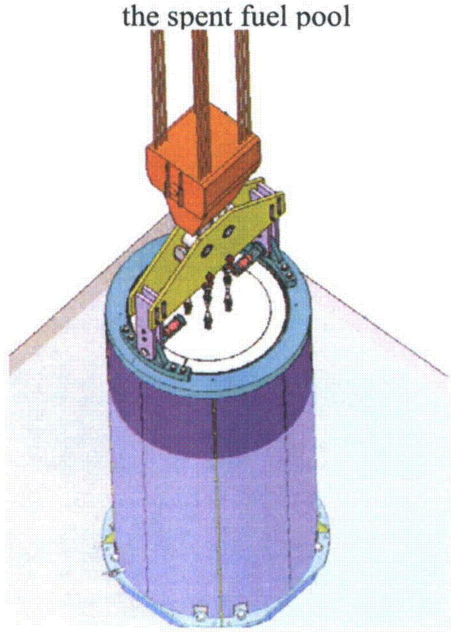


FIGURE 1.2.3: SUMMARY OF TYPICAL LOADING OPERATIONS

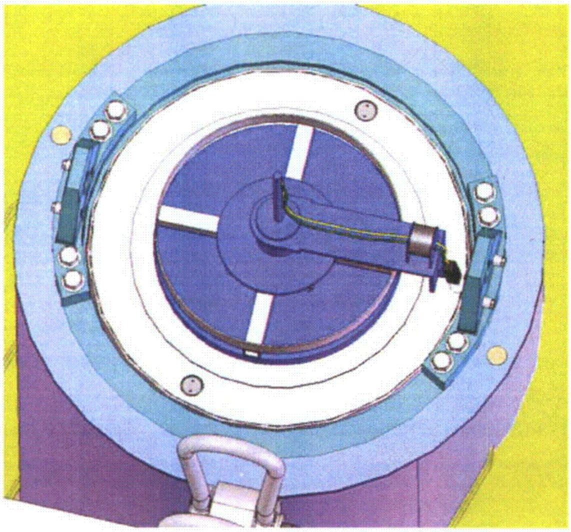
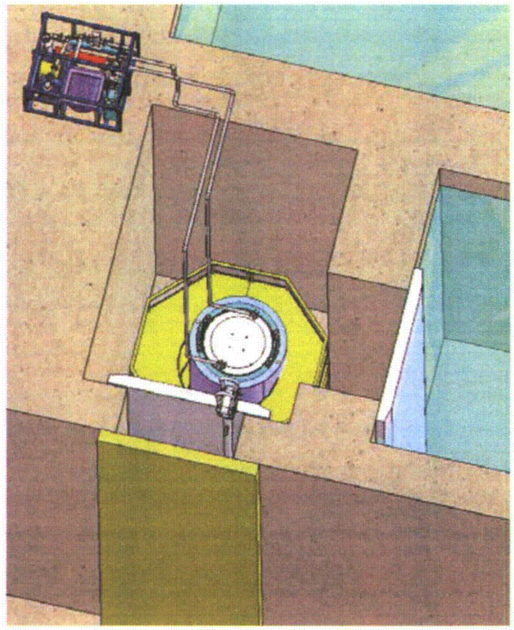
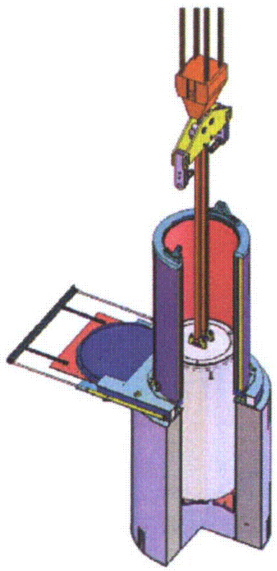
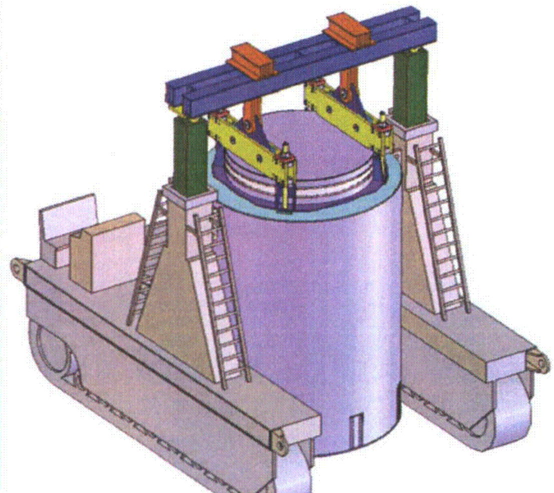
5. MPC Closure Operations (Lid to Shell Welding) 	6. MPC Draining, Drying and Backfill 
7. System Stackup and MPC Transfer Operations 	8. HI-STORM Movement to the ISFSI 

FIGURE 1.2.3 (CONTINUED): SUMMARY OF TYPICAL LOADING OPERATIONS

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section contains the necessary information to fulfill the requirements pertaining to the qualifications of the applicant pursuant to 10 CFR72.2(a)(1),(b) and 72.230(a). Holtec International, headquartered in Marlton, NJ, is the system designer and applicant for certification of the HI-STORM FW system.

Holtec International is an engineering technology company with a principal focus on the power industry. Holtec International Nuclear Power Division (NPD) specializes in spent fuel storage technologies. NPD 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 fuel pool for increased storage capacity) in numerous nuclear plants around the world. Over 80 plants in the U.S., Britain, Brazil, Korea, Mexico and Taiwan have utilized the Company's wet storage technology to extend their in-pool storage capacities.

NPD is also a turnkey provider of dry storage and transportation technologies to nuclear plants around the globe. The company is contracted by over 40 nuclear units in the U.S. to provide the company's vertical ventilated dry storage technology. Utilities in China, Korea, Spain, Ukraine, and Switzerland are also active users of Holtec International's dry storage and transport systems.

Four U.S. commercial plants, namely, Dresden Unit 1, Trojan, Indian Point Unit 1, and Humboldt Bay have thus far been completely defueled using Holtec International's technology. For many of its dry storage clients, Holtec International provides all phases of dry storage including: the required site-specific safety evaluations; ancillary designs; manufacturing of all capital equipment; preparation of site construction procedures; personnel training; dry runs; and fuel loading. The USNRC dockets in parts 71 and 72 currently maintained by the Company are listed in Table 1.3.1

Holtec International's corporate engineering consists of professional engineers and experts with extensive experience in every discipline germane to the fuel storage technologies, namely structural mechanics, heat transfer, computational fluid dynamics, and nuclear physics. Virtually all engineering analyses for Holtec's fuel storage projects (including HI-STORM FW) are carried out by the company's full-time staff. The Company is actively engaged in a continuous improvement program of the state-of-the-art in dry storage and transport of spent nuclear fuel. The active patents and patent applications in the areas of dry storage and transport of SNF held by the Company (ca. January 2009) are listed in Table 1.3.2. Many of these listed patents have been utilized in the design of the HI-STORM FW System.

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

incorporated by reference into this FSAR. Holtec International's QA program has been certified by the USNRC (Certificate No. 71-0784).

The HI-STORM FW System will be fabricated by Holtec International Manufacturing Division (HMD) located in Pittsburgh, Pennsylvania. HMD is a long term N-Stamp holder and fabricator of nuclear components. In particular, HMD has been manufacturing HI-STORM and HI-STAR system components since the inception of Holtec International's dry storage and transportation program in the 1990s. HMD routinely manufactures ASME code components for use in the US and overseas nuclear plants. Both Holtec International's headquarters and the HMD subsidiary have been subject to triennial inspections by the USNRC. If another fabricator is to be used for the fabrication of any part of the HI-STORM FW System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

The Metamic-HT will be fabricated by Holtec International Nanotec Division (Nanotec) located in Lakeland, Florida. Nanotec has been manufacturing classic Metamic for several years for both dry and wet storage applications and in the last few years has been manufacturing and testing Metamic-HT. If another fabricator is to be used for the fabrication of Metamic-HT, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

Holtec International's Nuclear Power Division (NPD) also carries out site services for dry storage deployments at nuclear power plants. Several nuclear plants, such as Trojan (completed) and Waterford (ongoing, ca. 2009) have deployed dry storage at their sites using a turn key contract with Holtec International.

TABLE 1.3.1

USNRC DOCKETS ASSIGNED TO HOLTEC INTERNATIONAL

System Name	Docket Number
HI-STORM 100 (Storage)	72-1014
HI-STAR 100 (Storage)	72-1008
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 60 (Transportation)	71-9336
Holtec Quality Assurance Program	71-0784

TABLE 1.3.2

DRY STORAGE AND TRANSPORT PATENTS
ASSIGNED TO HOLTEC INTERNATIONAL

Colloquial Name of the patent	USPTO Patent Number
Honeycomb Fuel Basket	5,898,747
HI-STORM 100S Overpack	6,064,710
Duct Photon Attenuator	6,519,307B1
HI-TRAC Operation	6,587,536B1
Cask Mating Device (Hermetically Sealable Transfer Cask)	6,625,246B1
Improved Ventilator Overpack	6,718,000B2
Below Grade Transfer Facility	6,793,450B2
HERMIT (Seismic Cask Stabilization Device)	6,848,223B2
Cask Mating Device (operation)	6,853,697
Davit Crane	6,957,942B2
Duct-Fed Underground HI-STORM	7,068,748B2
Forced Helium Dehydrator (design)	7,096,600B2
Below Grade Cask Transfer Facility	7,139,358B2
Forced Gas Flow Canister Dehydration (alternate embodiment)	7,210,247B2
HI-TRAC Operation (Maximizing Radiation Shielding During Cask Transfer Procedures)	7,330,525
HI-STORM 100U	7,330,526B2

1.4 GENERIC CASK ARRAYS

The HI-STORM FW System is stored in a vertical configuration. The required center-to-center spacing between the modules (layout pitch) on the Independent Spent Fuel Storage Installation (ISFSI) pad is guided by operational considerations such as size, accessibility, security, dose, and functionality. Tables 1.4.1, 1.4.2, and 1.4.3 provide the typical layout pitch information for 2 x N (N can be any integer), 3 x N (N can be any integer), and rectangular arrays, respectively.

The following is a generic discussion on the HI-STORM FW ISFSI pad, its suggested arrangement, and supporting infrastructure. The final design of the ISFSI is the responsibility of the user of the HI-STORM FW System.

The HI-STORM FW ISFSI pad is typically 24" to 28" thick, reinforced concrete supported by engineered fill with depth and properties selected to satisfy a site-specific design. The casks are arrayed in the manner of a rectilinear grid such as that shown in Figures 1.4.1, 1.4.2, and 1.4.3. The pitch values in Table 1.4.1 may be varied to suit the user's specific needs. The spacing (X, Y, etc., in the figures) is chosen to satisfy two competing requirements. Typically, the ISFSI owner desires to minimize the spacing in order to produce self-shielding between the storage casks, however the spacing must also be sufficient to allow the transporter access to emplace and remove the overpacks. The HI-STORM FW spacing (pitch) shown in Table 1.4.1 are typical values that meet both competing requirements.

A Canister Transfer Facility (CTF) may be needed in the future (when the Fuel Building is no longer available) to remove the multi-purpose canister from the HI-STORM FW overpack and place it into a HI-STAR transport cask, suitable for offsite shipment. The MPC transfer should be performed in a controlled area. Therefore, the ISFSI facility should preferably be sized to accommodate the CTF; however the construction of the CTF can be performed during a later development phase.

The general area surrounding the HI-STORM FW ISFSI pad will be graded to be compatible with the current drainage features, with additional storm water catch basins and piping added and incorporated into the existing storm water collection system, as necessary. The general area surrounding the ISFSI pad is typically covered with crushed stone or gravel to provide a suitable surface for the transporter and to prevent weeds and other unsuitable foliage from sprouting.

The ISFSI should have an area designated as a HI-STORM FW fabrication pad. This area is used to prepare HI-STORM FW casks for concrete placement, assembly, touch-up painting, storage, and maintenance between the time of initial on-site delivery and actual MPC transfer. An adjacent garage and maintenance shop may also be required for housing the transfer cask and ancillaries, such as the transporter, lifting appurtenances, etc.

If the ISFSI pad is located outside the plant's protected area, a security post building to provide a weather enclosure for temporary security guard support staff may be needed during casks movement and facility access. The building would also provide a common termination point for security equipment wiring and the HI-STORM FW temperature monitoring data acquisition equipment, if used. A backup power diesel generator and associated transformers may be skid mounted on a pad

adjacent to the security post.

The discussion of the security and related systems below presumes that the ISFSI is located outside the plant's protected area. The security requirements are adjusted accordingly if the ISFSI is located inside the plant's protected area.

The requirements on the security system provided below are generic and illustrative of the state-of-the-art practice, i.e., they are not meant to be mandatory provisions. The ISFSI owner bears the ultimate responsibility to comply with all security related regulations and mandates.

1.4.2 Security System and Other Ancillary Requirements

A security system for the ISFSI will be designed to include intrusion detection and camera systems, security fencing, lighting, isolation zones, monitoring systems, and electrical supply. The design must be integrated with the existing plant security system and its components. The system must meet the requirements of 10CFR72 and 10CFR73, and shall be integrated into the existing Plant's Physical Security Plan. The design of the security system shall also take into consideration the guidelines provided by NUREG-1619, NUREG-1497, and NRC Regulatory Guide 5.44.

Electrical design features must also be included for HI-STORM FW temperature monitoring, HI-STORM FW grounding, and the storage/maintenance building, as required. The HI-STORM FW temperature monitoring system (if used) will include thermal detectors mounted directly to the overpacks. These detectors will provide continuous monitoring and data acquisition equipment to collect, process, and transmit data to a central computer system to allow frequent review of data results and to indicate any temperature alerts. The storage building should have sufficient electrical power supply to support lights, outlets, and power equipment associated with maintenance of HI-STORM FW ancillary equipment, such as the transporter. In the event of loss of power to the site, a backup power supply is required.

1.4.2.1 Security System

The ISFSI security system design shall provide the layout for all components and associated power and signal wiring. The security interface building located adjacent to the ISFSI would provide a transition point to connect all of the wiring to the existing plant power and data acquisition systems.

The ISFSI security systems will consist of two separate systems supplementing each other: perimeter intrusion detection system (PIDS) and a closed circuit television (CCTV) system. The PIDS will provide an alarm signal to the existing security system whenever one of the perimeter zones has been accessed without authorization. The CCTV system will provide assessment of the alarming zone. Both of these systems have to work with each other in order to provide proper assessment. All signals generated by the security systems will be transmitted to the Central Alarm Station (CAS) through a robust communication means. The ISFSI security system design will be compatible with the plant's existing design.

The security systems design will include details for PIDS mounting, CCTV system mounting, zone arrangements, fiber optic hardware/cable connections for alarm and tamper, camera and microwave

unit locations, and upgrades to the existing security system to accommodate the new ISFSI systems.

1.4.2.2 Lighting System

The design of the lighting system includes light fixture selection, quantity, mounting, and arrangement throughout ISFSI perimeter and the assessment of illumination levels in foot-candles.

The illumination levels required at the perimeter area and inside the plant's protected area will be maintained at the ISFSI in accordance with plant commitments and regulatory requirements. The design will also include infrared illuminators to be installed, as an option with the CCTV system cameras to provide minimum light level required for IR sensitive cameras.

1.4.2.3 Fence System

The design for ISFSI perimeter fence includes a double fence configuration. The inner fence will be the protected area perimeter and the outer fence will be a nuisance fence to establish the appropriate isolation zone. The typical fence arrangements, including man-gates; vehicle gates; and grounding details; will be based on the existing plant fence specifications and design standards.

1.4.2.4 Electrical System

The conceptual design for the electrical system would entail the following activities and use their results as inputs:

- design for security systems (PIDS and CCTV)
- design for perimeter lighting system (PLS)
- design for temperature monitoring system (TMS) (if used)
- design for storage/support building

The total ISFSI site load will determine what type and size of power source will be used in this application. The existing power distribution facilities must be reviewed to determine a capability of the potential power sources. To be able to add the new ISFSI load to an existing system an analysis will be completed including the evaluation of the existing loads on 4160VAC line, cable sizes, and the approximate cable length. The transformers (4160-480V and 480-208/120V) will be sized accordingly to accommodate a new distribution system. The conceptual design will also include all the aspects of sizing a backup power distribution system based on providing a dedicated diesel generator as a source.

1.4.2.5 Cask Grounding System

The design of the grounding system should be based on NEC requirements and engineering and plant practices. The new grounding system, if required, will surround the ISFSI perimeter and provide a ground path for all ISFSI related equipment and structures including storage casks, microwave equipment and mounting poles, camera and towers, security lighting, perimeter fences, and the security building at the ISFSI site. The grounding system will be connected to the primary source transformer ground.

TABLE 1.4.1 TYPICAL (AND MINIMUM) LAYOUT PITCH AND SPACING DIMENSIONS FOR HI-STORM FW ARRAYS			
Item	**Layout in Figure 1.4.1**	**Layout in Figure 1.4.2**	**Layout in Figure 1.4.3**
X1	16 ft (15 ft)	16 ft (15 ft)	16 ft (15 ft)
Y1	16 ft (15 ft)	16 ft (15 ft)	16 ft (15 ft)
Y2	12 ft	12 ft	N/A
Y3	12 ft	12 ft	N/A

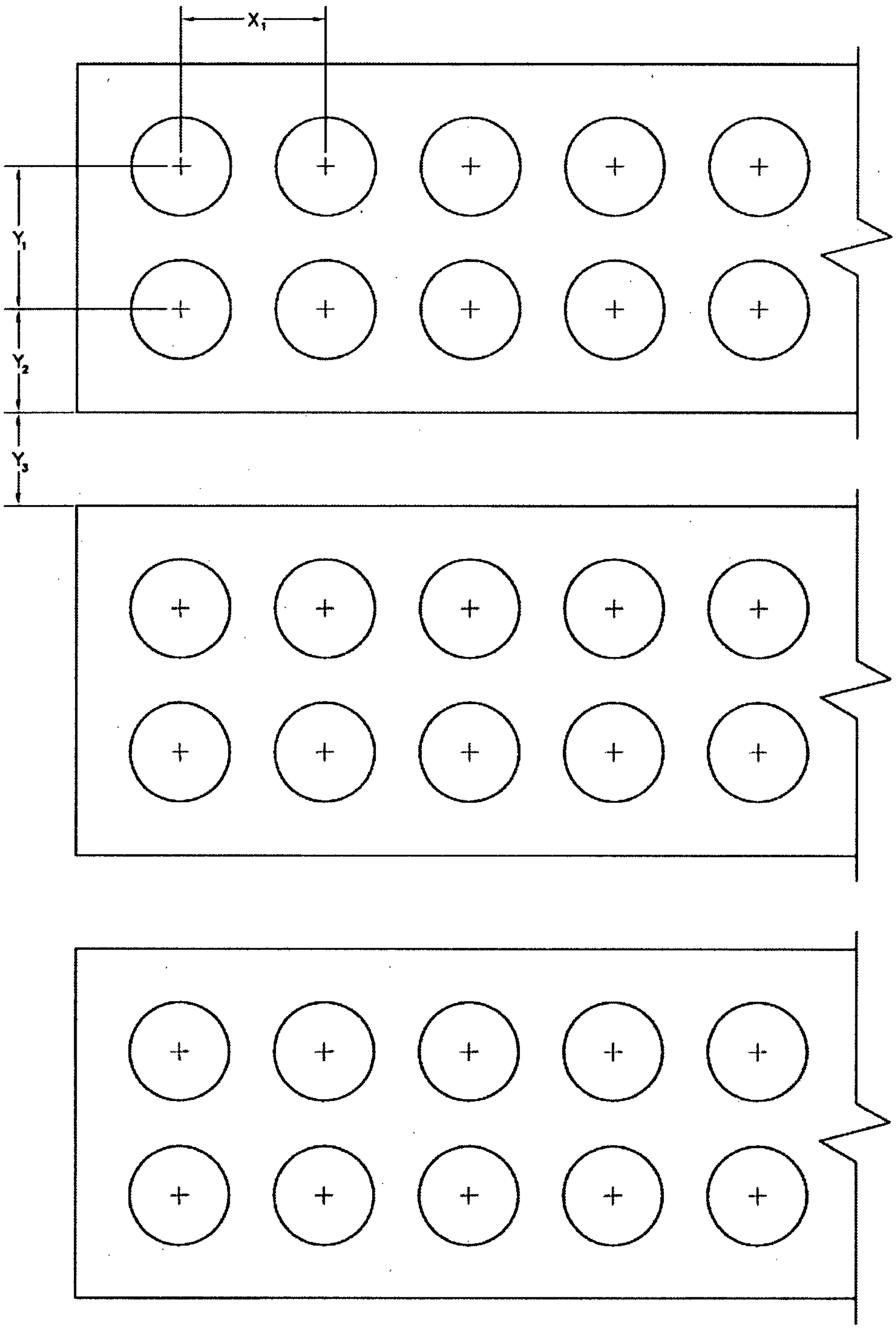


FIGURE 1.4.1: 2xN HI-STORM FW ARRAYS

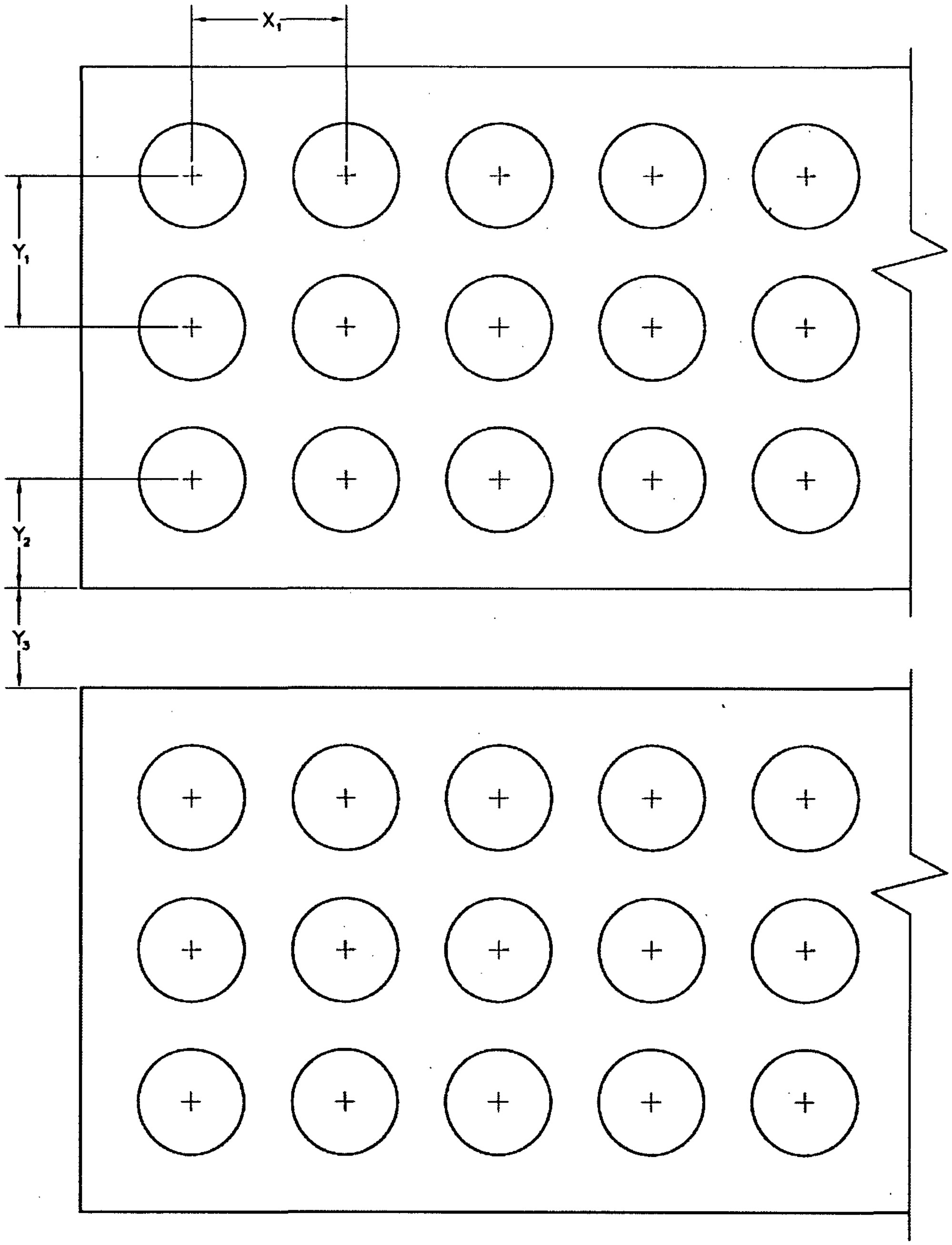


FIGURE 1.4.2: 3xN HI-STORM FW ARRAYS

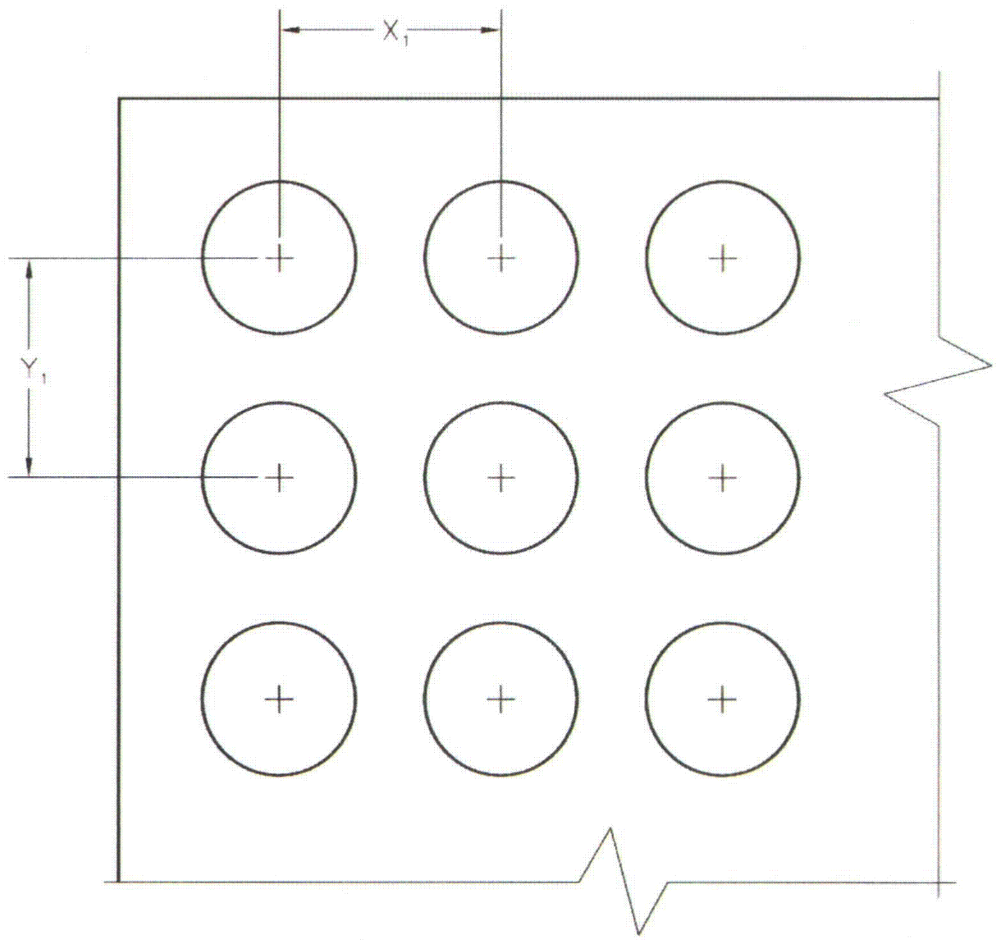


FIGURE 1.4.3: RECTANGULAR HI-STORM FW ARRAY

1.5 DRAWINGS

The following HI-STORM FW System drawings are provided on subsequent pages in this section to fulfill the requirements in 10 CFR 72.2(a)(1),(b) and 72.230(a):

Drawing No.	Title	Revision
6494	HI-STORM FW BODY	2
6508	HI-STORM LID ASSEMBLY	2
6514	HI-TRAC VW – MPC-37	1
6799	HI-TRAC VW – MPC-89	1
6505	MPC-37 ENCLOSURE VESSEL	4
6506	MPC-37 FUEL BASKET	4
6512	MPC-89 ENCLOSURE VESSEL	4
6507	MPC-89 FUEL BASKET	4

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1.6 REFERENCES

[1.0.1] 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Fuel, High-level Radioactive Waste, and Reactor-Related Greater than Class C Waste", Title 10 of the Code of Federal Regulations- Energy, 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] Regulatory Guide 1.76 "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plant", U.S. Nuclear Regulatory Commission, March 2007.

[1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, New York, 2007.

[1.1.2] 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Title 10 of the Code of Federal Regulations, Office of the Federal Register, Washington, D.C.

[1.1.3] USNRC Docket 72-1014, "Final Safety Analysis Report for the HI-STORM 100 System", Holtec Report No. HI-2002444, latest revision.

[1.1.4] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", U.S. Nuclear Regulatory Commission, February 1996.

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

[1.2.3] ANSI N14.6-1993, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More", American National Standards Institute, Inc., Washington D.C., June 1993.

[1.2.4] Companion Guide to the ASME Boiler & Pressure Vessel Code, K.R. Rao (editor), Chapter 56, " Management of Spent Nuclear Fuel", Third Edition, ASME (2009)

[1.2.5] HI-STAR 180 Transportation Package, USNRC Docket No. 71-9325

APPENDIX 1.A: ALLOY X DESCRIPTION

1.A.1 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 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 Least Favorable Material Properties

The following material properties vary between the Alloy X constituents:

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

Each of these material properties are provided in the ASME Code Section II [1.A.1]. Tables 1.A.1 through 1.A.5 provide the ASME Code values for each constituent of Alloy X along with the least

favorable value utilized in this licensing application. The ASME Code only provides values to -20°F. The lower bound service temperature of the MPC is -40°F. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STORM FW 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. 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.

1.A.4 References

[1.A.1] ASME Boiler & Pressure Vessel Code, Section II, Materials (2007).

TABLE 1.A.1

DESIGN STRESS INTENSITY (S_m) vs. TEMPERATURE FOR THE ALLOY-X MATERIALS

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	18.6	18.6	19.3	18.9	18.6
500	17.5	17.5	18.0	17.5	17.5
600	16.6	16.6	17.0	16.5	16.5
650	16.2	16.2	16.6	16.0	16.0
700	15.8	15.8	16.3	15.6	15.6
750	15.5	15.5	16.1	15.2	15.2
800	15.2	15.2	15.9	14.8	14.8

Notes:

1. Source: Table 2A on pages 308, 312, 316, and 320 of [1.A.1].
2. Units of design stress intensity values are ksi.

TABLE 1.A.2 TENSILE STRENGTH (S_u) vs. TEMPERATURE OF ALLOY-X MATERIALS					
Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
100	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
200	71.0 (66.3)	71.0 (66.3)	75.0 (70.0)	75.0 (70.0)	71.0 (66.3)
300	66.2 (61.8)	66.2 (61.8)	72.9 (68.0)	70.7 (66.0)	66.2 (61.8)
400	64.0 (59.7)	64.0 (59.7)	71.9 (67.1)	67.1 (62.6)	64.0 (59.7)
500	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	64.6 (60.3)	63.4 (59.2)
600	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	63.3 (59.0)	63.3 (59.0)
650	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	62.8 (58.6)	62.8 (58.6)
700	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	62.4 (58.3)	62.4 (58.3)
750	63.3 (59.0)	63.3 (59.0)	71.5 (66.7)	62.1 (57.9)	62.1 (57.9)
800	62.8 (58.6)	62.8 (58.6)	70.8 (66.1)	61.7 (57.6)	61.7 (57.6)

Notes:

1. Source: Table U on pages 514, 516, 518, 520, and 522 of [1.A.1].
2. Units of tensile strength are ksi.
3. The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in parentheses are based on SA-336 forged materials (type F304, F304LN, F316, and F316LN), which are used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.

TABLE 1.A.3 YIELD STRESSES (S_y) vs. TEMPERATURE OF ALLOY-X MATERIALS					
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	25.9	25.5	25.0
300	22.4	22.4	23.4	22.9	22.4
400	20.7	20.7	21.4	21.0	20.7
500	19.4	19.4	20.0	19.5	19.4
600	18.4	18.4	18.9	18.3	18.3
650	18.0	18.0	18.5	17.8	17.8
700	17.6	17.6	18.2	17.3	17.3
750	17.2	17.2	17.9	16.9	16.9
800	16.9	16.9	17.7	16.5	16.5

Notes:

1. Source: Table Y-1 on pages 634, 638, 646, and 650 of [1.A.1].
2. Units of yield stress are ksi.

TABLE 1.A.4 COEFFICIENT OF THERMAL EXPANSION vs. TEMPERATURE OF ALLOY-X MATERIALS	
Temp. (°F)	Type 304, 304LN, 316, 316LN
-40	--
100	8.6
150	8.8
200	8.9
250	9.1
300	9.2
350	9.4
400	9.5
450	9.6
500	9.7
550	9.8
600	9.8
650	9.9
700	10.0
750	10.0
800	10.1
850	10.2
900	10.2
950	10.3
1000	10.3
1050	10.4
1100	10.4

Notes:

1. Source: Group 3 alloys from Table TE-1 on pages 749 and 751 of [1.A.1].
2. Units of mean coefficient of thermal expansion are in./in./°F x 10^{-6}.

TABLE 1.A.5 THERMAL CONDUCTIVITY vs. TEMPERATURE OF ALLOY-X MATERIALS			
Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X (minimum of constituent values)
-40	--	--	--
70	8.6	8.2	8.2
100	8.7	8.3	8.3
150	9.0	8.6	8.6
200	9.3	8.8	8.8
250	9.6	9.1	9.1
300	9.8	9.3	9.3
350	10.1	9.5	9.5
400	10.4	9.8	9.8
450	10.6	10.0	10.0
500	10.9	10.2	10.2
550	11.1	10.5	10.5
600	11.3	10.7	10.7
650	11.6	10.9	10.9
700	11.8	11.2	11.2
750	12.0	11.4	11.4
800	12.3	11.6	11.6
850	12.5	11.9	11.9
900	12.7	12.1	12.1
950	12.9	12.3	12.3
1000	13.1	12.5	12.5
1050	13.4	12.8	12.8
1100	13.6	13.0	13.0

Notes:
1. Source: Material groups J and K in Table TCD on page 765, 766, and 775 of [1.A.1].
2. Units of thermal conductivity are Btu/hr-ft-°F.

APPENDIX 1.B: METAMIC-HT

Withheld in Accordance with 10 CFR 2.390

APPENDIX 1.C: METAMIC-HT PROPERTIES SUPPORTING HI-STORM FW ACCIDENT EVALUATIONS

Withheld in Accordance with 10 CFR 2.390

CHAPTER 2†: PRINCIPAL DESIGN CRITERIA

2.0 INTRODUCTION

The design characteristics of the HI-STORM FW System are presented in Chapter 1, Section 1.2. This chapter contains a compilation of loadings and design criteria applicable to the HI-STORM FW System. The loadings and conditions prescribed herein for the MPC, particularly those pertaining to mechanical accidents, are consistent with those required for 10CFR72 compliance. 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 structural design criteria are presented in the subsequent chapters of this FSAR.

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

The design criteria for the MPCs, HI-STORM FW overpack, and HI-TRAC VW transfer cask are summarized in Subsections 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 engineered for a 60 year design life, while satisfying the requirements of 10CFR72. The adequacy of the MPC to meet the above design life is discussed in Section 3.4. The design characteristics of the MPC are described in Section 1.2.

Structural

The MPC is classified as important-to-safety. The MPC structural components include the fuel basket and the enclosure vessel. The fuel basket is designed and fabricated to meet a more stringent displacement limit under mechanical loadings than those implicit in the stress limits of the ASME code (see Section 2.2). The MPC enclosure vessel is designed and fabricated as a Class 1 pressure vessel in accordance with Section III, Subsection NB of the ASME Code, with

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. 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. All terms-of-art used in this chapter are consistent with the terminology of the Glossary.

certain necessary alternatives, as discussed in Section 2.2. The principal exception to the above Code pertains to the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell, as discussed in Section 2.2. In addition, Threaded Anchor Locations (TALs) in the MPC lid are designed in accordance with the requirements of NUREG-0612 for critical lifts to facilitate handling of the loaded MPC.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid-to-shell weld is further verified by performing a progressive liquid penetrant examination of the weld layers, and a Code pressure test.

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

Compliance with the ASME Code, with respect to the design and fabrication of the MPC, and the associated justification are discussed in Section 2.2. The MPC design is analyzed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. The required characteristics of the fuel assemblies to be stored in the MPC are limited in accordance with Section 2.1.

<u>Thermal</u>

The thermal design and operation of the MPC in the HI-STORM FW System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.1]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

ii. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400°C (752°F). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400°C (752°F) for high burnup fuel (HBF) and 570°C (1058°F) for moderate burnup fuel.

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

iv. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65°C (117°F) and the number of excursions to less than 10.

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

i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400°C (752°F). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570°C (1058°F) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.2]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions have been added to ensure these limits are met.

ii. A method of drying, such as forced helium dehydration (FHD) is used if the above temperature limits for short-term operations cannot be met.

iii. The off-normal and accident condition PCT limit remains unchanged at 570 °C (1058°F).

The MPC cavity is dried, either with FHD or vacuum drying, and then it is backfilled with high purity helium to promote heat transfer and prevent cladding degradation.

The normal condition design temperatures for the stainless steel components in the MPC are provided in Table 2.2.3.

Each MPC model allows for regionalized storage where the basket is segregated into three regions as shown in Figures 1.2.1 and 1.2.2. Decay heat limits for regionalized loading are presented in Tables 1.2.3 and 1.2.4 for MPC-37 and MPC-89, respectively. Specific requirements, such as approved locations for DFCs and non-fuel hardware are given in Section 2.1.

Shielding

The dose limits for an ISFSI using the HI-STORM FW System are delineated in 10CFR72.104 and 72.106. Compliance with these regulations for any particular array of casks at an ISFSI is necessarily site-specific and must be demonstrated by the licensee. Dose for a single cask and a representative cask array is illustrated in Chapter 5.

The MPC provides axial shielding at the top and bottom ends to maintain occupational exposures ALARA during canister closure and handling operations. The HI-TRAC VW bottom lid also contains shielding. The occupational doses are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 9).

The dose evaluation is performed for a reference fuel (Table 1.0.4) as described in Section 5.2. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure (ALARA) evaluation, as discussed in Chapter 11.

Criticality

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

Criticality control is maintained by the geometric spacing of the fuel assemblies and the spatially distributed B-10 isotope in the Metamic-HT fuel basket, and for the PWR MPC model, the additional soluble boron in the MPC water. The minimum specified boron concentration in the purchasing specification for Metamic-HT must be met in every lot of the material manufactured. The guaranteed B-10 value in the neutron absorber, assured by the manufacturing process, is further reduced by 10% (90% credit is taken for the Metamic-HT) to accord with NUREG/CR-5661. No credit is taken for fuel burnup or integral poisons such as gadolinia in BWR fuel. The soluble boron concentration requirements (for PWR fuel only) based on the initial enrichment of the fuel assemblies are delineated in Section 2.1 consistent with the criticality analysis described in Chapter 6.

Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions. As discussed in Section 7.1, the HI-STORM FW MPC design meets the guidance in Interim Staff Guidance (ISG)-18 so that leakage of radiological matter from the confinement boundary is non-credible. Therefore, no confinement dose analysis is required or performed. The confinement function of the MPC is verified through pressure testing, helium leak testing, and a rigorous weld examination regimen executed in accordance with the acceptance test program in Chapter 10.

Operations

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

Generic operating procedures for the HI-STORM FW System are provided in Chapter 9. Detailed operating procedures will be developed by the licensee using the information provided in Chapter 9 along with the site-specific requirements that comply with the 10CFR50 Technical Specifications for the plant, and the HI-STORM FW System Certificate of Compliance (CoC).

Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the MPC are described in Chapter 10. The operational controls and limits to be applied to the MPC are discussed in Chapter 13. 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 MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM FW System is addressed in Section 2.4.

2.0.2 HI-STORM FW Overpack Design Criteria

General

The HI-STORM FW overpack is engineered for a 60 year Design Life while satisfying the requirements of 10CFR72. The adequacy of the overpack to meet the required design life is discussed in Subsection 3.4.7. The design characteristics of the HI-STORM FW overpack are summarized in Subsection 1.2.1.

Structural

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

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

The HI-STORM FW overpack plain concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates. As the HI-STORM FW overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete in the analysis to provide an appropriate simulation of the accident conditions postulated in this FSAR. The technical requirements on testing and qualification of the HI-STORM FW overpack plain concrete are in Appendix 1.D of the HI-STORM 100 FSAR. Appendix 1.D is incorporated in this FSAR by reference.

There is no U.S. or international code that is sufficiently comprehensive to provide a completely prescriptive set of requirements for the design, manufacturing, and structural qualification of the overpack. The various sections of the ASME Codes, however, contain a broad range of specifications that can be assembled to provide a complete set of requirements for the design, analysis, shop manufacturing, and final field construction of the overpack. The portions or whole of the Codes and Standards that are invoked for the various elements of the overpack design, analysis, and manufacturing activities (viz., materials, fabrication and inspection) are summarized in Tables 1.2.6, 1.2.7, and 1.2.8.

The ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NF Class 3, [2.0.3], is the applicable code to determine stress limits for the load bearing components of the

overpack when required by the acceptance criteria set down in this chapter. The material types used in the components of the HI-STORM FW System are listed in the licensing drawings.

ACI 318-05 [2.0.4] is the applicable reference code to establish the limits on unreinforced concrete (in the Closure Lid), which is subject to secondary structural loadings. Appendix 1.D contains the design, construction, and testing criteria applicable to the plain concrete in the overpack lid.

As mandated by 10CFR72.24(c)(3) and §72.44(d), Holtec International's quality assurance (QA) program requires all constituent parts of an SSC subject to NRC certification under 10CFR72 to be assigned an ITS category appropriate to its function in the control and confinement of radiation. The ITS designations (ITS or NITS) for the constituent parts of the HI-STORM FW System are provided in the licensing drawings. The QA categorization level (A, B, or C) for ITS parts is provided in Tables 2.0.1 through 2.0.8. A table exists for each licensing drawing and provides the QA level for the parts designated as ITS on the licensing drawings.

The excerpts from the codes, standards, and generally recognized industry publications invoked in this FSAR, supplemented by the commitments in Holtec's QA procedures, provide the necessary technical framework to ensure that the as-installed system would meet the intent of §72.24(c), §72.120(a) and §72.236(b). As required by Holtec's QA Program (discussed in Chapter 14), all operations on ITS components must be performed under QA validated written procedures and specifications that are in compliance with the governing citations of codes, standards, and practices set down in this FSAR.

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2.

Thermal

The thru-thickness temperature limits for the plain concrete in the overpack for long term and short term temperatures are in Table 2.2.3. 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 Subsection 2.2.2. The brittle fracture assessment of structural steel materials used in the storage cask is considered in Section 3.1.

The overpack is designed to dissipate the maximum allowable heat load (shown in Tables 1.2.3 and 1.2.4) from the MPC. The thermal characteristics of the MPC stored inside the overpack are evaluated in Chapter 4.

Shielding

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

The overpack is designed to limit the calculated surface dose rates on the cask for all MPC designs as defined in Subsection 2.3.5. The overpack is also designed to maintain occupational exposures ALARA during MPC processing, 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 operations and a site boundary dose assessment for a typical ISFSI, as described in Chapter 11.

Confinement

The overpack does not perform any confinement function. Confinement during storage is provided by the MPC. The overpack provides physical protection and radiation shielding of the MPC contents during dry storage operations.

Operations

There are no radioactive effluents that result from MPC operations after the MPC is sealed or during storage operations. 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 FW System are provided in Chapter 9. The licensee is required to develop detailed operating procedures based on Chapter 9 with due consideration of site-specific conditions including the applicable 10CFR50 technical specification requirements for the site, and the HI-STORM FW System CoC.

Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the overpack are described in Chapter 10. The operational controls and limits to be applied to the overpack are contained in Chapter 13. 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 FW System, including the overpack, are addressed in Section 2.4.

2.0.3 HI-TRAC VW Transfer Cask Design Criteria

General

The HI-TRAC VW transfer cask is engineered for a 60 year design life. The adequacy of the HI-TRAC VW to meet the above design life commitment is discussed in Section 3.4. The design characteristics of the HI-TRAC VW cask are presented in Subsection 1.2.1.

Structural

The HI-TRAC VW transfer cask includes both structural and non-structural radiation shielding components that are classified as important-to-safety. The structural steel components of the HI-TRAC VW are designed to meet the stress limits of Section III, Subsection NF, of the ASME Code for normal and off-normal storage conditions. The threaded anchor locations for lifting and handling of the transfer cask are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for interfacing lift points.

The HI-TRAC VW transfer cask design is analyzed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. Under accident conditions, the HI-TRAC VW transfer cask must protect the MPC from unacceptable deformation, provide continued shielding, and remain in a condition such that the MPC can be removed from it. The loads applicable to the HI-TRAC VW transfer cask are defined in Tables 2.2.6 and 2.2.13 and Table 3.1.1. The physical characteristics of each MPC for which the HI-TRAC VW is designed are presented in Subsection 1.2.1.

Thermal

The allowable temperatures for the HI-TRAC VW 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 allowable temperatures for the structural steel and shielding components of the HI-TRAC VW are provided in Table 2.2.3. The HI-TRAC VW is designed for off-normal environmental cold conditions, as discussed in Subsection 2.2.2. The evaluation of the potential for brittle fracture in structural steel materials is presented in Section 3.1.

The HI-TRAC VW is designed and evaluated for the maximum heat load analyzed for storage operations. The maximum allowable temperature of water in the HI-TRAC jacket 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 restricted to be less than the saturation temperature at the shell design pressure. Even though the analysis shows that the water jacket will not over-pressurize, a relief device is placed at the top of the water jacket shell. In addition, the water is precluded from freezing during off-normal cold conditions by limiting

the minimum allowable operating temperature and by adding ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1. The working area ambient temperature limit for loading operations is limited in accordance with Table 2.2.2.

Shielding

The HI-TRAC VW 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 the rated capacity of the crane. As discussed in Subsection 1.2.1, the shielding in HI-TRAC VW is maximized within the constraint of the allowable weight at a plant site. The HI-TRAC VW calculated dose rates for a set of reference conditions 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 11. A postulated HI-TRAC VW accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Chapter 5.

The annular area between the MPC outer surface and the HI-TRAC VW inner surface can be isolated to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC VW surfaces expected to require decontamination are coated with a suitable coating. The maximum permissible surface contamination for the HI-TRAC VW is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 11).

Confinement

The HI-TRAC VW transfer cask does not perform any confinement function. The HI-TRAC VW provides physical protection and radiation shielding of the MPC contents during MPC loading, unloading, and transfer operations.

Operations

There are no radioactive effluents that result from MPC transfer operations using HI-TRAC VW. 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 FW System are provided in Chapter 9. The licensee will develop detailed operating procedures based on Chapter 9 along with plant-specific requirements including the Part 50 Technical Specification and SAR, and the HI-STORM FW System CoC.

Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the HI-TRAC VW Transfer Cask are described in Chapter 10. The operational controls and limits to be applied to the HI-

TRAC VW are contained in Chapter 13. Application of these requirements will assure that the HI-TRAC VW is fabricated, operated, and maintained in a manner that satisfies the design criteria given in this chapter.

Decommissioning

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

2.0.4 Principal Design Criteria for the ISFSI Pad

2.0.4.1 Design and Construction Criteria

In compliance with 10CFR72, Subpart F, "General Design Criteria", the HI-STORM FW cask system is classified as "important-to-safety" (ITS). This FSAR explicitly recognizes the HI-STORM FW System as an assemblage of equipment containing numerous ITS components. The reinforced concrete pad, on which the cask is situated, however, is designated as a "not important to safety" (NITS) structure because of a lack of a physical connection between the cask and the pad.

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

- The pad design complies with the structural provisions of this FSAR.
- The material of construction of the pad (viz., the additives used in the pad concrete) are compatible with the ambient environment at the ISFSI site.
- Appropriate structural evaluations are performed pursuant to 10CFR72.212 to demonstrate that the pad is structurally competent to permit the cask to withstand the seismic and other credible inertial loadings at the site.

2.0.4.2 Load Combinations and Applicable Codes

Factored load combinations for ISFSI pad design are provided in NUREG-1536 [1.0.3]. The factored loads applicable to the pad design consist of dead weight of the cask, thermal gradient loads, impact loads arising from handling and accident events, external missiles, and bounding environmental phenomena (such as earthquakes, wind, tornado, and flood).

The factored load combinations presented in Table 3-1 of NUREG 1536 are reduced in number by eliminating loading types that are not germane or controlling in a HI-STORM ISFSI pad

design. The applicable factored load combinations are accordingly adapted from the HI-STORM 100 FSAR and presented below.

a. Definitions

D = Dead load
L = Live load
T = Thermal load
E = DBE seismic load
U_c= Reinforced concrete available strength

b. Load Combinations for the Concrete Pad

Normal Events

$$U_c > 1.4\ D + 1.7\ L$$

Off-Normal Events

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

Accidents

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

As an interfacing structure, the ISFSI pad and its underlying substrate must possess the structural strength to satisfy the above inequalities. As discussed in the HI-STAR 100 FSAR, thermal gradient loads are generally small; therefore, the Off-Normal Event does not generally provide a governing load combination.

Table 2.2.9 provides a reference set of parameters for the ISFSI pad and its foundation that are used solely as input to the non-mechanistic tipover analysis. Analyses in Chapter 3 show that this reference pad design does not violate the design criterion applicable to the non-mechanistic tip-over of the HI-STORM FW storage system. The pad design may be customized to meet the requirements of a particular site, without performing a site-specific tipover analysis, provided that all ISFSI pad strength properties are less than or equal to the values in Table 2.2.9.

Applicable sections of industry codes such as ACI 318-05, "Building Code Requirements for Structural Concrete"; ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R, "Guide for Concrete Floor and Slab Construction"; and ACI 224R-90, "Control of Cracking in Concrete Structures" may be used in the design, structural evaluation, and construction of the concrete pad. However, load combinations in ACI 318-05 are not applicable to the ISFSI pad structural evaluation, and are replaced by the load combinations stated in subparagraph 2.0.4.2.b.

Table 2.0.1 – HI-STORM FW Assembly (Drawing # 6494)		
Item Number*	Part Name	ITS QA Safety Category
1	Assembly, Lid, HI-STORM Ø 113 B.C.	B
2	Lid-Stud	B
3	Heavy Hex Nut, 3 ¼" – 4 UNC	B
5	Plate, HI-STORM FW Heat Shield	B
6	Shielding, HI-STORM FW Body	B
8	Block, HI-STORM FW Cask Anchor	B
11	Plate, HI-STORM FW Body Base	B
15	Shell, HI-STORM FW Outer Shell	B
16	Shell, HI-STORM FW Inner Shell	B
17	Rib, HI-STORM FW Lifting Rib	B
18	Plate, HI-STORM FW Cask Body Top	B
20	Plate, Gamma Shield	C
21	Tube, MPC Guide	C
22	Tube, MPC Guide	C
23	Tube, MPC Guide	C

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.2 – MPC-37 Enclosure Vessel (Drawing # 6505)		
Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
9	Block, Enclosure Vessel Drain Shielding	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
14	Cap, Enclosure Vessel Vent/Drain	C
20	Shim, Enclosure Vessel Type 1 PWR Fuel Basket	C
21	Shim, Enclosure Vessel Type 2 PWR Fuel Basket	C

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.3 – Assembly, MPC-37 Fuel Basket (Drawing # 6506)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A

Table 2.0.4 – Assembly, MPC-89 Fuel Basket (Drawing # 6507)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A
7	Panel, Type 7 Cell Wall	A
8	Panel, Type 8 Cell Wall	A

Table 2.0.5 – Assembly, Lid, HI-STORM Ø113 B.C. (Drawing # 6508)		
Item Number*	Part Name	ITS QA Safety Category
1	Plate, HI-STORM Lid Base	B
2	Plate, HI-STORM Lid Type 1 Round	B
3	Plate, HI-STORM Lid Type 2 Round	B
4	Plate, HI-STORM Lid Type 1 Ring	B
5	Plate, HI-STORM Lid Type 2 Ring	B
6	Plate, HI-STORM Lid Type 3 Ring	B
7	Plate, HI-STORM Lid Type 4 Ring	B
8	Plate, HI-STORM Lid Type 5 Ring	B
9	Plate, HI-STORM Lid Type 6 Ring	B
10	Plate, HI-STORM Lid Upper Shim	B
11	Plate, HI-STORM Lid Lower Shim	B
13	Gusset, HI-STORM Lid	B
16	Shielding, HI-STORM Lid Lower	B
17	Shielding, HI-STORM Lid Upper	B
18	Plate, Heat Shield	B
20	Block, HI-STORM Lid Lifting Anchor	B

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.6 – MPC-89 Enclosure Vessel (Drawing # 6512)		
Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
9	Block, Enclosure Vessel Drain Shielding	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
14	Cap, Enclosure Vessel Vent/Drain	C
20	Shim, Enclosure Vessel Type 1 BWR Fuel Basket	C
21	Shim, Enclosure Vessel Type 2 BWR Fuel Basket	C
22	Shim, Enclosure Vessel Type 3 BWR Fuel Basket	C

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.7 – HI-TRAC VW – MPC-37 (Drawing # 6514)		
Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6" LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.8 – HI-TRAC VW – MPC-89 (Drawing # 6799)		
Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6" LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

2.1 SPENT FUEL TO BE STORED

2.1.1 Determination of the Design Basis Fuel

A central object in the design of the HI-STORM FW System is to ensure that all SNF discharged from the U.S. reactors and not yet loaded into dry storage systems can be stored in a HI-STORM FW MPC. Publications such as references [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors.

The cell openings in the fuel baskets have been sized to accommodate BWR and PWR assemblies. The cavity length of the MPC will be determined for a specific site to accord with the fuel assembly length used at that site, including non-fuel hardware and damaged fuel containers, as applicable.

Table 2.1.1 summarizes the authorized contents for the HI-STORM FW System. Tables 2.1.2 and 2.1.3, which are referenced in Table 2.1.1, provide the fuel characteristics of all groups of fuel assembly types determined to be acceptable for storage in the HI-STORM FW System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.2 and 2.1.3 and meets the other limits specified in Table 2.1.1 is acceptable for storage in the HI-STORM FW System. The groups of fuel assembly types presented in Tables 2.1.2 and 2.1.3 are defined as "array/classes" as described in further detail in Chapter 6. Table 2.1.4 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal, or that are used as reference assembly design is those analyses. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Undamaged SNF Specifications

Undamaged fuel is defined in the Glossary.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in the Glossary.

Damaged fuel assemblies and fuel debris will be loaded into damaged fuel containers (DFCs) (Figure 2.1.6) that have mesh screens on the top and bottom. The DFC will have a removable lid to allow the fuel assembly to be inserted. In storage, the lid will be latched in place. DFC's used to move fuel assemblies will be designed for lifting with either the lid installed or with a separate handling lid. DFC's used to handle fuel and the associated lifting tools will designed in accordance with the requirements of NUREG-0612. The DFC will be fabricated from structural aluminum or stainless steel. The appropriate structural, thermal, shielding, criticality, and confinement evaluations have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for

damaged fuel assemblies and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in this chapter.

2.1.4 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, cross sectional dimensions, and weight. These parameters, which define the mechanical and structural design, are specified in Subsection 2.1.8. An appropriate axial clearance is provided to prevent interference due to the irradiation and thermal growth of the fuel assemblies.

2.1.5 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the fuel's peak cladding temperature (PCT) which is a function of the maximum decay heat per assembly and the decay heat removal capabilities of the HI-STORM FW System.

To ensure the permissible PCT limits are not exceeded, Subsection 1.2 specifies the maximum allowable decay heat per assembly for each MPC model in the three-region configuration (see also Table 1.2.3 and 1.2.4).

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The design basis fuel assembly for thermal calculations for both PWR and BWR fuel is provided in Table 2.1.4.

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 references [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.5 and Figures 2.1.3 and 2.1.4. 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 FW System.

2.1.6 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM FW System are the 10CFR72 §104 and §106 operator-controlled boundary dose rate limits, and the requirement to maintain operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the assembly, which is a function of the assembly type, and the burnup, enrichment and cooling time of the assemblies. Dose rates are further directly affected by the size and arrangement of the ISFSI, and the specifics of the loading operations. All these parameters are site-dependent, and the compliance with the regulatory dose rate requirements are performed in site-specific calculations. The evaluations here are therefore performed with reference fuel assemblies, and with parameters that result in

reasonably conservative dose rates. The reference assemblies given in Table 1.0.4 are the predominant assemblies used in the industry.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Table 2.1.1 provides the acceptable ranges of burnup, enrichment and cooling time for all of the authorized fuel assembly array/classes. Table 2.1.5 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 FW System.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1.

2.1.7 Criticality Parameters for Design Basis SNF

The criticality analyses for the MPC-37 are performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.6 provides the required soluble boron concentrations for this MPC.

2.1.8 Summary of Authorized Contents

Tables 2.1.1 through 2.1.3 specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM FW System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR.

Table 2.1.1

MATERIAL TO BE STORED

PARAMETER	VALUE (Note 1)	
	MPC-37	MPC-89
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the applicable array/class.	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, with or without channels, fuel debris meeting the limits in Table 2.1.3 for the applicable array/class.
Cladding Type	ZR (see Glossary for definition)	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6	≤ 5.0 wt. % U-235
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years Maximum Assembly Average Burnup: 68.2 GWd/mtU	Minimum Cooling Time: 3 years Maximum Assembly Average Burnup: 65 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup	Minimum Cooling Time: 3 years Maximum Burnup†: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable	N/A
Decay heat per fuel storage location	Regionalized Loading: See Table 1.2.3	Regionalized Loading: See Table 1.2.4

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.1

MATERIAL TO BE STORED

PARAMETER	VALUE (Note 1)	
	MPC-37	MPC-89
Fuel Assembly Nominal Length (in.)	Minimum: 157 (with NFH) Reference: 167.2 (with NFH) Maximum: 199.2 (with NFH and DFC)	Minimum: 171 Reference: 176.5 Maximum: 176.5 (with DFC)
Fuel Assembly Width (in.)	≤ 8.54 (nominal design)	≤ 5.95 (nominal design)
Fuel Assembly Weight (lb)	Reference: 1600 (without NFH) 1750 (with NFH), 1850 (with NFH and DFC) Maximum: 2050 (including NFH and DFC)	Reference: 750 (without DFC), 850 (with DFC) Maximum: 850 (including DFC)
Other Limitations	▪ Quantity is limited to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 37. ▪ One NSA. ▪ Up to 30 BPRAs. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, NSAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations specified in Figure 2.1.5.	▪ Quantity is limited to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 89.

Table 2.1.2

PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**14x14 A**	**14x14 B**	**14x14 C**	**15x15 B**	**15x15 C**
No. of Fuel Rod Locations	179	179	176	204	204
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.420	≥ 0.417
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3736	≤ 0.3640
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3671	≤ 0.3570
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.563	≤ 0.563
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	5 (Note 2)	21	21
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.015	≥ 0.0165

Table 2.1.2 (continued)

PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15 D**	**15x15 E**	**15x15 F**	**15x15 H**	**15x15 I**
No. of Fuel Rod Locations	208	208	208	208	216
Fuel Clad O.D. (in.)	≥ 0.430	≥ 0.428	≥ 0.428	≥ 0.414	≥ 0.413
Fuel Clad I.D. (in.)	≤ 0.3800	≤ 0.3790	≤ 0.3820	≤ 0.3700	≤ 0.3670
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3735	≤ 0.3707	≤ 0.3742	≤ 0.3622	≤ 0.3600
Fuel Rod Pitch (in.)	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0150	≥ 0.0140	≥ 0.0140	≥ 0.0140	≥ 0.0140

Table 2.1.2 (continued) PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)						
Fuel Assembly Array and Class	**16x16 A**	**17x17A**	**17x17 B**	**17x17 C**	**17x17 D**	**17x17 E**
No. of Fuel Rod Locations	236	264	264	264	264	265
Fuel Clad O.D. (in.)	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377	≥ 0.372	≥ 0.372
Fuel Clad I.D. (in.)	≤ 0.3350	≤ 0.3150	≤ 0.3310	≤ 0.3330	≤ 0.3310	≤ 0.3310
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252	≤ 0.3232	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502	≤ 0.496	≤ 0.496
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 170	≤ 170
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	≥ 0.0350	≥ 0.016	≥ 0.014	≥ 0.020	≥ 0.014	≥ 0.014

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. Each guide tube replaces four fuel rods.
3. Annular fuel pellets are allowed in the top and bottom 12" of the active fuel length.
4. One Instrument Tube and eight Guide Bars (Solid ZR).

Table 2.1.3

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**7x7 B**	**8x8 B**	**8x8 C**	**8x8 D**	**8x8 E**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	49	63 or 64	62	60 or 61	59
Fuel Clad O.D. (in.)	≥ 0.5630	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930
Fuel Clad I.D. (in.)	≤ 0.4990	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250
Fuel Pellet Dia. (in.)	≤ 0.4910	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160
Fuel Rod Pitch (in.)	≤ 0.738	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	0	1 or 0	2	1 - 4 (Note 6)	5
Water Rod Thickness (in.)	N/A	≥ 0.034	> 0.00	> 0.00	≥ 0.034
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**8x8F**	**9x9 A**	**9x9 B**	**9x9 C**	**9x9 D**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	$\leq$ 4.5 (Note 12)	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	64	74/66 (Note 4)	72	80	79
Fuel Clad O.D. (in.)	$\geq$ 0.4576	$\geq$ 0.4400	$\geq$ 0.4330	$\geq$ 0.4230	$\geq$ 0.4240
Fuel Clad I.D. (in.)	$\leq$ 0.3996	$\leq$ 0.3840	$\leq$ 0.3810	$\leq$ 0.3640	$\leq$ 0.3640
Fuel Pellet Dia. (in.)	$\leq$ 0.3913	$\leq$ 0.3760	$\leq$ 0.3740	$\leq$ 0.3565	$\leq$ 0.3565
Fuel Rod Pitch (in.)	$\leq$ 0.609	$\leq$ 0.566	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	N/A (Note 2)	2	1 (Note 5)	1	2
Water Rod Thickness (in.)	$\geq$ 0.0315	> 0.00	> 0.00	$\geq$ 0.020	$\geq$ 0.0300
Channel Thickness (in.)	$\leq$ 0.055	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.100	$\leq$ 0.100

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**9x9 E (Note 3)**	**9x9 F (Note 3)**	**9x9 G**	**10x10 A**	**10x10 B**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	$\leq$ 4.5 (Note 12)	$\leq$ 4.5 (Note 12)	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	$\geq$ 0.4170	$\geq$ 0.4430	$\geq$ 0.4240	$\geq$ 0.4040	$\geq$ 0.3957
Fuel Clad I.D. (in.)	$\leq$ 0.3640	$\leq$ 0.3860	$\leq$ 0.3640	$\leq$ 0.3520	$\leq$ 0.3480
Fuel Pellet Dia. (in.)	$\leq$ 0.3530	$\leq$ 0.3745	$\leq$ 0.3565	$\leq$ 0.3455	$\leq$ 0.3420
Fuel Rod Pitch (in.)	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.510	$\leq$ 0.510
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	$\geq$ 0.0120	$\geq$ 0.0120	$\geq$ 0.0320	$\geq$ 0.030	> 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120

Table 2.1.3 (continued) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)			
Fuel Assembly Array and Class	**10x10 C**	**10x10 F**	**10x10 G**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	$\leq$ 4.8	$\leq$ 4.7 (Note 13)	$\leq$ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	$\geq$ 0.3780	$\geq$ 0.4035	$\geq$ 0.387
Fuel Clad I.D. (in.)	$\leq$ 0.3294	$\leq$ 0.3570	$\leq$ 0.340
Fuel Pellet Dia. (in.)	$\leq$ 0.3224	$\leq$ 0.3500	$\leq$ 0.334
Fuel Rod Pitch (in.)	$\leq$ 0.488	$\leq$ 0.510	$\leq$ 0.512
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	$\geq$ 0.031	$\geq$ 0.030	$\geq$ 0.031
Channel Thickness (in.)	$\leq$ 0.055	$\leq$ 0.120	$\leq$ 0.060

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

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

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

4. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.

5. Square, replacing nine fuel rods.

6. Variable.

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

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

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

10. These rods may also be sealed at both ends and contain ZR material in lieu of water.

11. Not Used

12. Fuel assemblies classified as damaged fuel assemblies are limited to 4.0 wt.% U-235.

13. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% U-235.

Table 2.1.4

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	BWR	PWR
Reactivity/Criticality	GE-12/14 10x10 (Array/Class 10x10A)	Westinghouse 17x17 OFA (Array/Class 17x17B)
Shielding	GE-12/14 10x10	Westinghouse 17x17 OFA
Thermal-Hydraulic	GE-12/14 10x10	Westinghouse 17x17 OFA

Table 2.1.5
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION[1]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.5485
2	4-1/6% to 8-1/3%	0.8477
3	8-1/3% to 16-2/3%	1.0770
4	16-2/3% to 33-1/3%	1.1050
5	33-1/3% to 50%	1.0980
6	50% to 66-2/3%	1.0790
7	66-2/3% to 83-1/3%	1.0501
8	83-1/3% to 91-2/3%	0.9604
9	91-2/3% to 95-5/6%	0.7338
10	95-5/6% to 100%	0.4670

BWR DISTRIBUTION[2]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.2200
2	4-1/6% to 8-1/3%	0.7600
3	8-1/3% to 16-2/3%	1.0350
4	16-2/3% to 33-1/3%	1.1675
5	33-1/3% to 50%	1.1950
6	50% to 66-2/3%	1.1625
7	66-2/3% to 83-1/3%	1.0725
8	83-1/3% to 91-2/3%	0.8650
9	91-2/3% to 95-5/6%	0.6200
10	95-5/6% to 100%	0.2200

[1] Reference 2.1.7
[2] Reference 2.1.8

Table 2.1.6

Soluble Boron Requirements for MPC-37 Wet Loading and Unloading Operations

Array/Class	**All Undamaged Fuel Assemblies**		**One or More Damaged Fuel Assemblies and/or Fuel Debris**	
	Maximum Initial Enrichment $\leq$ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment $\leq$ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
All 14x14 and 16x16A	1,000	1,500	1,300	1,800
All 15x15 and 17x17	1,500	2,000	1,800	2,300

Note:

1. For maximum initial enrichments between 4.0 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.0 wt% and 5.0 wt% ^{235}U.

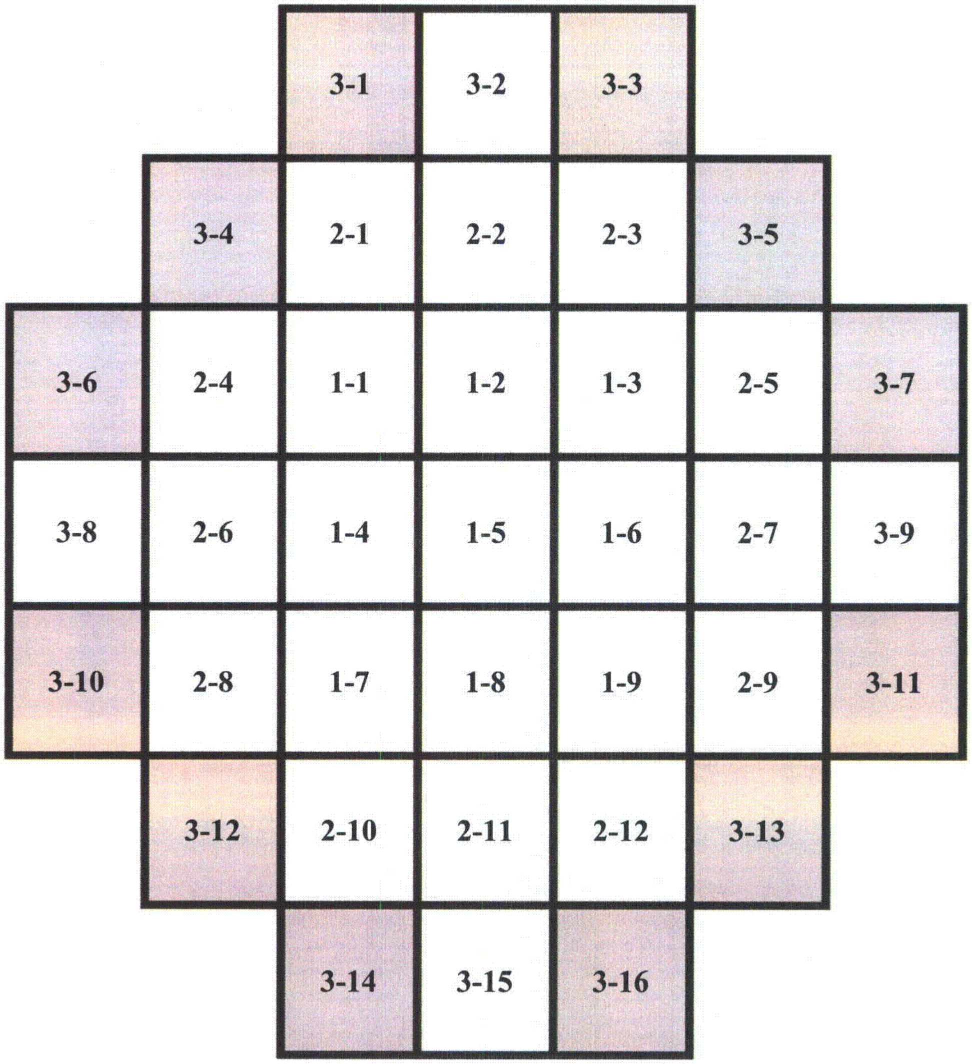

Figure 2.1.1 Location of DFCs for Damaged Fuel or Fuel Debris in the MPC-37(Shaded Cells)

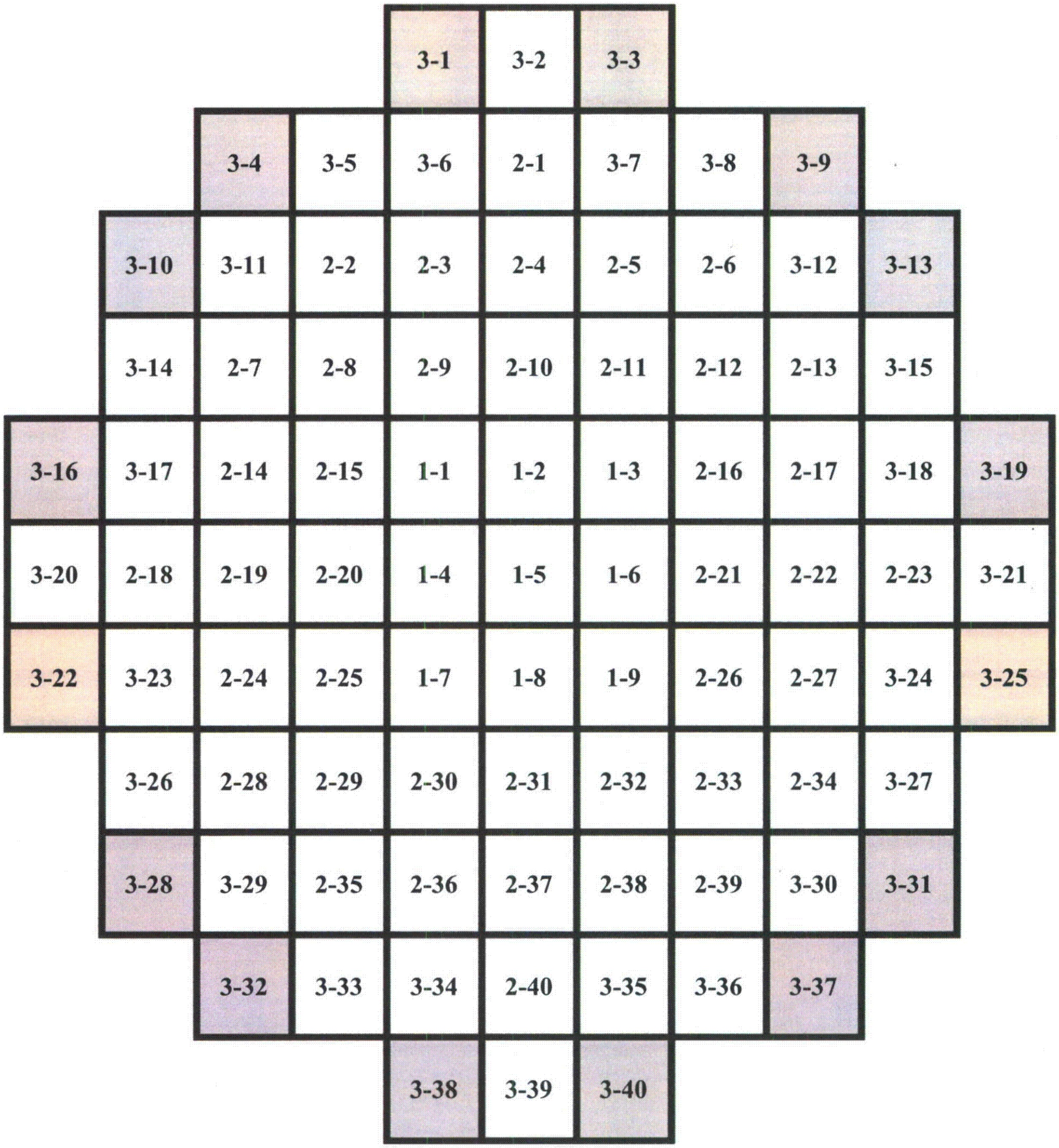

Figure 2.1.2 Location of DFCs for Damaged Fuel or Fuel Debris in the MPC-89 (Shaded Cells)

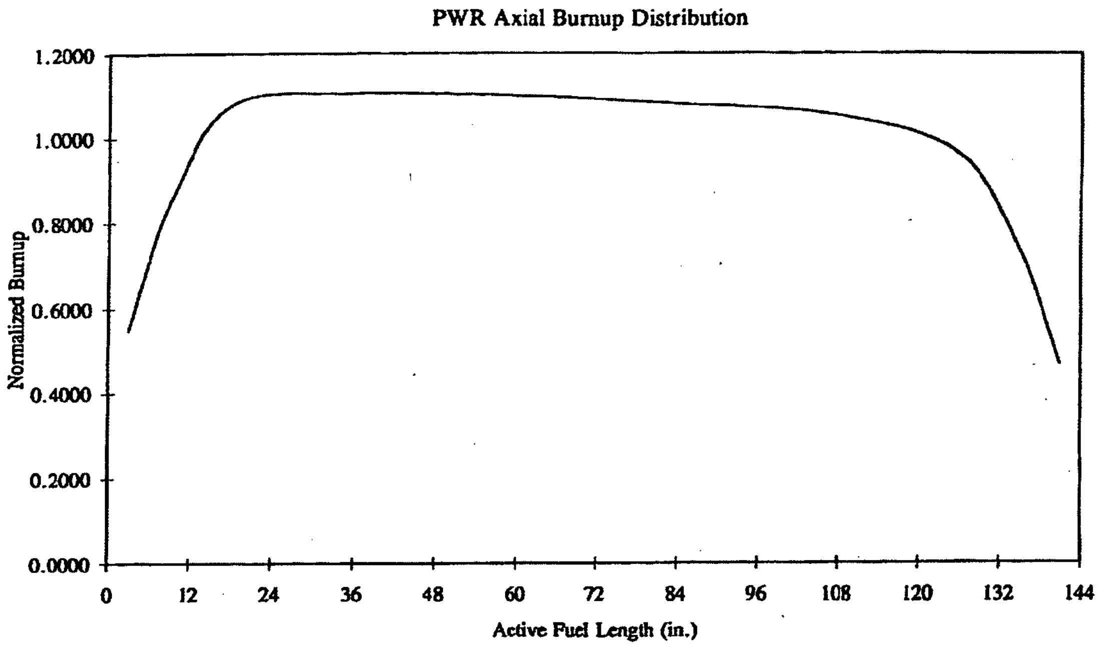

Figure 2.1.3 PWR Axial Burnup Profile with Normalized Distribution

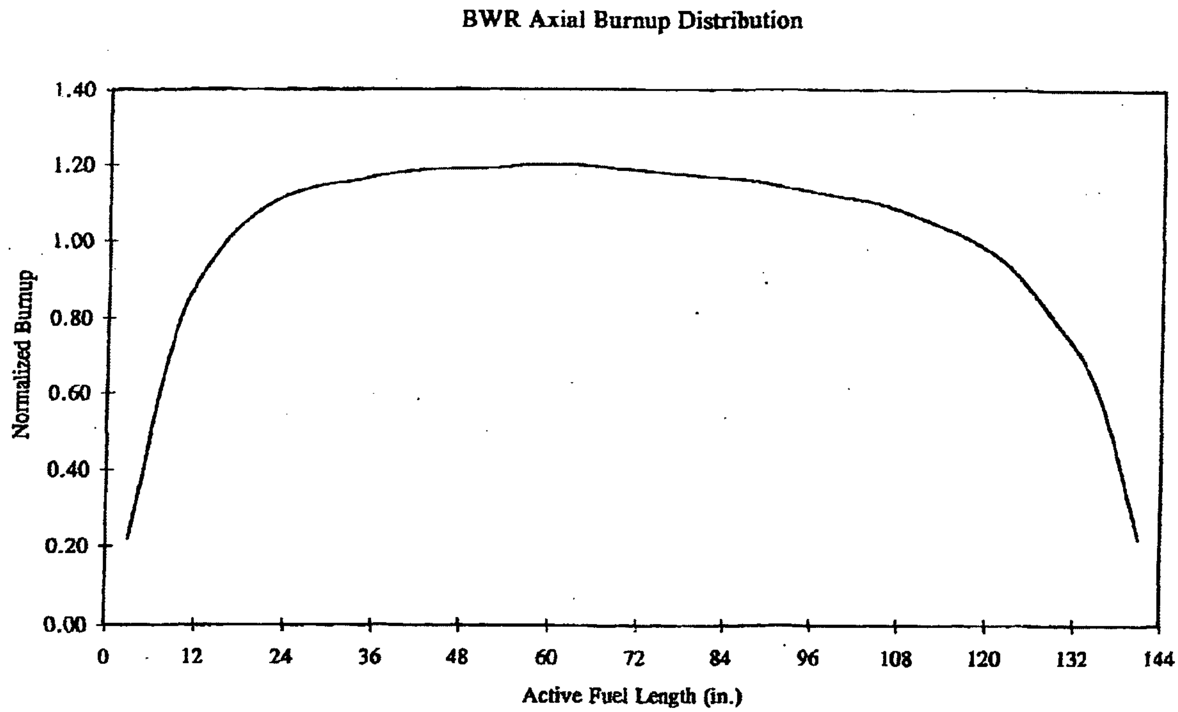

Figure 2.1.4 BWR Axial Burnup Profile with Normalized Distribution

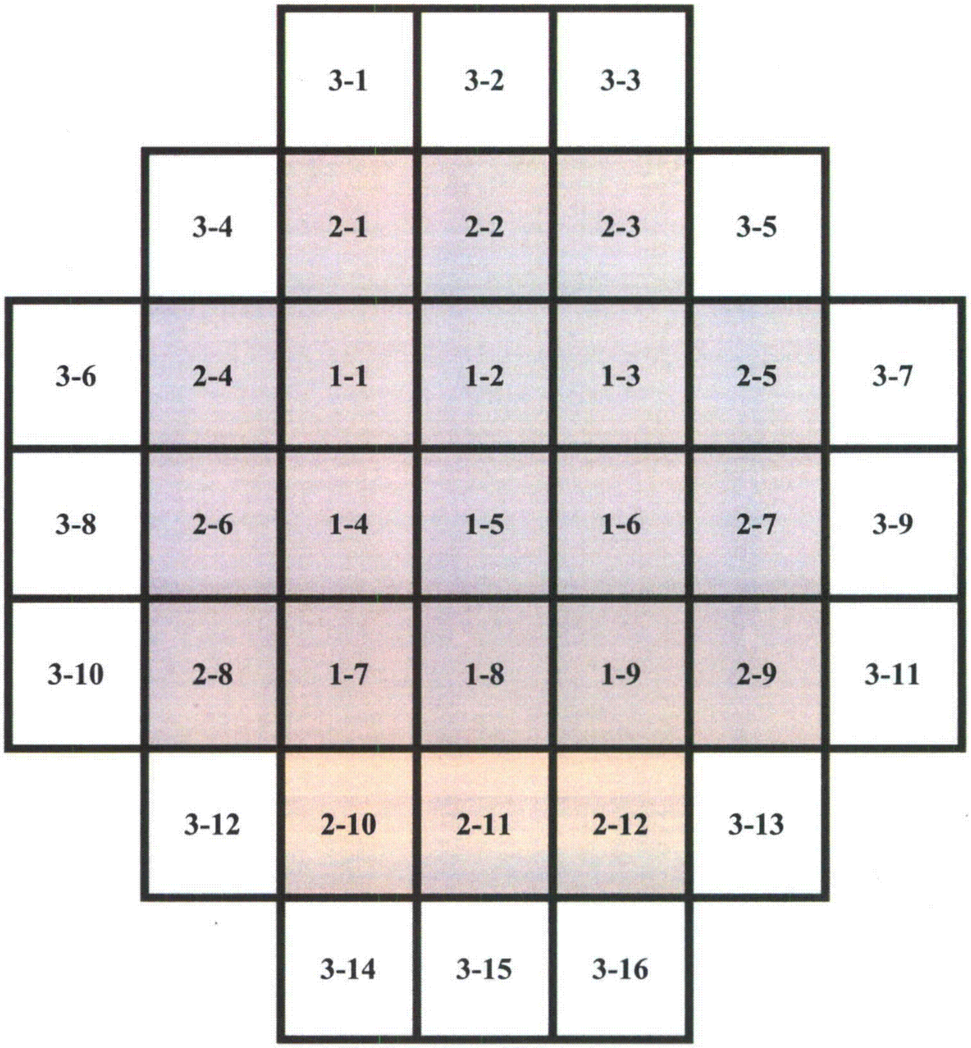

Figure 2.1.5: Location of NSAs, APSRs, RCCAs, CEAs, and CRAs in the MPC-37
(Shaded Cells)

Withheld in Accordance with 10 CFR 2.390

Figure 2.1.6: Damaged Fuel Container (Typical)

2.2 HI-STORM FW DESIGN LOADINGS

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

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

In the following subsections, the design criteria are established for normal, off-normal, and accident conditions for storage. The following conditions of storage and associated loads are identified:

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

ii. Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Partial Blockage of Air Inlets.

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

iv. Short-Term Operations: This loading condition is defined to accord with ISG-11, Revision 3 [2.0.1] guidance. This includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC VW transfer cask.

Each of these conditions and the applicable loads are identified herein with their applicable design criteria. A design criterion is deemed to be satisfied if the allowable limits for the specific loading conditions are not exceeded.

2.2.1 Loadings Applicable to Normal Conditions of Storage

a. Dead Weight

The HI-STORM FW System must withstand the static loads due to the weights of each of its components, including the weight of the HI-TRAC VW with the loaded MPC stacked on top the storage overpack during the MPC transfer.

b. Handling Evolutions

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

i. Vertical lifting and transfer to the ISFSI of the HI-STORM FW overpack containing a loaded MPC.
ii. Vertical lifting and handling of the HI-TRAC VW transfer cask containing a loaded MPC.
iii. Lifting of a loaded MPC.

The dead load of the lifted component is increased by 15% in the stress qualification analyses (to meet ANSI N14.6 guidance) to account for dynamic effects from lifting operations as suggested in CMAA #70 [2.2.1].

Handling operations of the loaded HI-TRAC VW transfer cask or HI-STORM FW overpack are limited to working area ambient temperatures specified in Table 2.2.2. This limitation is specified to ensure a sufficient safety margin against brittle fracture during handling operations.

Table 2.2.6 summarizes the analyses required to qualify all threaded anchor locations in the HI-STORM FW System.

c. 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.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container (DFC), it shall be 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) liberated. For PWR assemblies stored with non-fuel hardware, 100% of the gases in the non-fuel hardware (e.g.,

BPRAs) shall be assumed to be released. The accident condition design pressure shall envelop the case of 100% of the fuel rods ruptured.

The MPC internal pressure under the normal condition of storage must remain below the design pressure specified in Table 2.2.1.

The MPC external pressure is a function of environmental conditions, which may produce a pressure loading. The normal condition external design pressure is specified in Table 2.2.1.

The HI-STORM FW 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 VW transfer cask is not capable of retaining internal pressure due to its open design. Therefore, no analysis is required for the internal pressure loading in HI-TRAC VW transfer cask. However, the HI-TRAC VW transfer cask water jacket may experience an internal vapor pressure due to the heat-up of the water contained in the water jacket. Analysis is performed in Chapter 3 of this report to demonstrate that the water jacket can withstand the design pressure in Table 2.2.1 without a structural failure and that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device is used to ensure that the water jacket design pressure will not be exceeded.

d. Environmental Temperatures and Pressures

To evaluate the long-term effects of ambient temperatures on the HI-STORM FW System, an upper bound value on the annual average ambient temperature for the continental United States is used. The annual average temperature is termed the normal ambient temperature for storage. The normal ambient temperature specified in Table 2.2.2 is bounding for all reactor sites in the contiguous United States. The normal ambient temperature set forth in Table 2.2.2 is intended to ensure that it is greater than the annual average of ambient temperature at any location in the continental United States. In the northern region of the U.S., the design basis normal ambient 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. Insolation based on 10CFR71.71 input averaged over 24 hours shall be used as the additional heat input under the normal and off-normal conditions of storage.

The ambient pressure shall be assumed to be 760mm of Hg coincident with the normal condition temperature, whose bounding value is provided in Table 2.2.2. For sites located substantially above sea level (elevation > 1500 feet), it will be necessary to perform a site specific evaluation of the peak cladding temperature using the site specific ambient temperature (maximum average annual temperature based on 40 year meteorological data for the site). ISG 11, Revision 3 [2.0.1] temperature limits will continue to apply.

All of the above requirements are consistent with those in the HI-STORM 100 FSAR.

e. 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 FW 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, this temperature is referred to as the "Design Temperature for Normal Conditions". Conservative calculations of the steady-state temperature field in the HI-STORM FW System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STORM FW component temperatures at or below the normal condition design temperatures for the HI-STORM FW System defined in Table 2.2.3.

Maintaining fuel rod cladding integrity is also a design consideration. The fuel rod peak cladding temperature (PCT) limits for the long-term storage and short-term operating conditions shall meet the intent of the guidance in ISG-11, Revision 3 [2.0.1]. For moderate burnup fuel the PCT limit for short-term operations is higher than for high burnup fuel [2.0.2].

f. Snow and Ice

The HI-STORM FW System must be capable of withstanding pressure loads due to snow and ice. Section 7.0 of ANSI/ASCE 7-05 [2.2.3] 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 FW System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure load (Table 2.2.8) is set to bound the ANSI/ASCE 7-05 recommendation.

2.2.2 Loadings Applicable to Off-Normal Conditions

As the HI-STORM FW 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 this subsection.

A discussion of the effects of each off-normal condition and the corrective action for each off-normal condition is provided in Section 12.1. Table 2.2.7 contains a list of all normal and off-normal loadings and their applicable acceptance criteria.

a. <u>Pressure</u>

The HI-STORM FW System must withstand loads due to off-normal pressure. The off-normal condition for the MPC internal design pressure, defined herein in Table 2.2.1, 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 as suggested in NUREG-1536.

b. <u>Environmental Temperatures</u>

The HI-STORM FW 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 sufficient duration to allow the system to reach steady-state temperatures.

Limits on the peaks in the time-varying ambient temperature at an ISFSI site are 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 continental 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 FW storage system which essentially flattens the effect of daily temperature variations on the internals of the MPC.

c. <u>Design Temperatures</u>

In addition to the normal condition design temperatures, which apply to long-term storage and short-term normal operating conditions (e.g., MPC drying operations and onsite transport operations), an off-normal/accident condition temperature pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61 is also defined. This is the temperature which may exist during a transient event (examples of such an instance is the blockage of the overpack inlet vents or the fire accident). The off-normal/accident condition temperatures of Table 2.2.3 are given to bound

the maximax (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during the transient event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects, during or immediately after, a transient event.

The off-normal/accident condition temperatures for stainless steel and carbon steel components are chosen such that the material's ultimate tensile strength does not fall below 30% of its room temperature value, based on data in published references [2.2.4 and 2.2.5]. This ensures that the material will not be subject to significant creep rates during these short duration transient events.

d. Leakage of One Seal

The MPC enclosure vessel does not contain gaskets or seals: All confinement boundary closure locations are welded. Because the material of construction (Alloy X, see Appendix 1.A) is known from extensive industrial experience to lend to high integrity, high ductility and high fracture strength welds, the MPC enclosure vessel welds provide a secure barrier against leakage.

The confinement boundary is defined by the MPC shell, MPC baseplate, MPC lid, port cover plates, closure ring, and associated welds. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root (if more than one weld pass is required) and final weld passes. In addition to multi-pass liquid penetrant examination, the MPC lid-to-shell weld is pressure tested. The vent and drain port cover plates are also subject to proven non-destructive evaluations for leak detection such as liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary. Therefore, leakage of one seal is not evaluated for its consequence to the storage system.

e. Partial Blockage of Air Inlets

The loaded HI-STORM FW overpack must withstand the partial blockage of the air inlets. Because the overpack air inlets and outlets are covered by screens and inspected routinely (or alternatively, equipped with temperature monitoring devices), significant blockage of all vents by blowing debris, critters, etc., is very unlikely. Nevertheless, the inherent thermal stability of the HI-STORM FW System shall be demonstrated by assuming all air inlets are partially blocked as an off-normal event.

f. Malfunction of FHD

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

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

Although the FHD System is monitored during its operation, stoppage of FHD operations does not require actions to restore forced cooling for adequate heat dissipation. This is because the condition of natural convection cooling evaluated in Section 4.6 shows that the fuel temperatures remain below off-normal limits. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

2.2.3 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 continues to confine the radioactive material, the MPC fuel basket structure maintains the configuration of the contents, the canister can be recovered from the overpack, and the system continues to provide adequate shielding.

A discussion of the effects of each environmental phenomenon and accident condition is provided in Section 12.2. The consequences of each accident or environmental phenomenon are evaluated against the requirements of 10CFR72.106 and 10CFR20. Section 12.2 also provides the corrective action for each event.

a. Handling Accident

A handling accident in the Part 72 jurisdiction is precluded by the requirements and provisions specified in this FSAR. The loaded HI-STORM FW components will be lifted in the Part 72 operations jurisdiction in accordance with written and Q.A. validated procedures and shall use special lifting devices which comply with ANSI N14.6-1993 [2.2.2]. Also, the lifting and handling equipment (typically the cask transporter) is required to have a built-in redundancy against uncontrolled lowering of the load. Further, the HI-STORM FW is a vertically deployed system, and the handling evolutions in *short term operations*, as discussed in Chapter 9, do not involve downending of the loaded cask to the horizontal configuration (or upending from the horizontal state) at any time. In particular, the loaded MPC shall be lowered into the HI-STORM FW overpack or raised from the overpack using the HI-TRAC VW transfer cask and a MPC lifting system designed in accordance with ANSI N14.6. Therefore, analysis of a handling accident event involving a HI-STORM system component is not required.

b. Non-Mechanistic Tip-Over

The freestanding loaded HI-STORM FW overpack is demonstrated by analysis to remain kinematically stable under all design basis environmental phenomena (tornado, earthquake, etc.)

and postulated accident conditions. The cask tip-over is not an outcome of any environmental phenomenon or accident condition and the cask tip-over is considered a *non-mechanistic* event. Nevertheless, the HI-STORM FW overpack and MPC is analyzed for a hypothetical tip-over event, and the structural integrity of a loaded HI-STORM FW System after a tip-over onto a reinforced concrete pad is demonstrated by analysis to show compliance with 10 CFR 72.236(m) with regards to the future transportability of the MPC.

The following requirements and acceptance criteria apply to the HI-STORM FW overpack under the tipover event:

i. In order to maximize the target stiffness (based on experience with ISFSI pad designs), the ISFSI pad and underlying soil are conservatively modeled using the data in Table 2.2.9.
ii. The tipover is simulated as a gravity-directed rotation of the cask from rest with its CG above its edge on the pad as the system's initial condition. The tipover begins when the cask is given an infinitesimal outward displacement in the radial plane of its tilted configuration.
iii. The MPC will remain in the HI-STORM FW overpack after the tipover event and the overpack will not suffer any ovalization which would preclude the removal of the MPC.
iv. The maximum plastic deformation sustained by the fuel basket panels is limited to the value given in Table 2.2.11.
v. The HI-STORM FW overpack will not suffer a significant loss of shielding.
vi. The confinement boundary will not be breached.

c. Fire

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM FW overpack or loaded HI-TRAC VW transfer cask while it is being moved to the ISFSI.

The HI-STORM FW System must withstand elevated temperatures due to a fire event. The HI-STORM FW overpack and HI-TRAC VW 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 FW overpack and HI-TRAC VW 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 FW overpack and HI-TRAC VW transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475° F to accord with the provisions in 10CFR71.73.

The following acceptance criteria apply to the fire accident:

i. The peak cladding temperature during and after a fire accident shall not exceed the ISG-11 [2.0.1] permissible limit (see Table 2.2.3).

ii. The through-thickness average temperature of concrete at any section shall not exceed its short-term limit in Table 2.2.3.
iii. The steel structure of the overpack shall remain physically stable; i.e., no risk of structural instability such as gross buckling.

d. Partial Blockage of MPC Basket Flow Holes

The HI-STORM FW MPC is designed to prevent reduction of thermosiphon action due to partial blockage of the MPC basket flow holes by fuel cladding failure, fuel debris and crud. The HI-STORM FW 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 of fuel classified as undamaged during storage in the HI-STORM FW. Fuel classified as damaged fuel or fuel debris are placed in damaged fuel containers. The damaged fuel container is equipped with mesh screens which ensure that the damaged fuel and fuel debris will not escape to block the MPC basket flow holes. The MPC is loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities for fuel assemblies reported in an Empire State Electric Energy Research Corporation Report [2.2.6] determines a layer of crud of conservative depth that is assumed to partially block the MPC basket flow holes. The crud depth is listed in Table 2.2.8. The flow holes in the bottom of the fuel basket are designed (as can be seen on the licensing drawings) to ensure that this amount of crud does not block the internal helium circulation.

e. Tornado

The HI-STORM FW System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STORM FW System are consistent with NRC Regulatory Guide 1.76 [2.2.7], ANSI 57.9 [2.2.8], and ASCE 7-05 [2.2.3]. Table 2.2.4 provides the wind speeds and pressure drops the HI-STORM FW overpack can 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 FW overpack, and continued integrity of the MPC confinement boundary, within the storage overpack or HI-TRAC VW transfer cask, must be demonstrated under impact from tornado-generated missiles in conjunction with the wind loadings. Standard Review Plan (SRP) 3.5.1.4 of NUREG-0800 [2.2.9] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile that deforms on impact (large missile); a rigid missile to test penetration resistance (penetrant missile); and a small rigid missile of a size sufficient to pass through any openings in the protective barriers (micro-missile). SRP 3.5.1.4 suggests an automobile for a large missile, a rigid solid steel cylinder for the penetrating 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.

f. Flood

The HI-STORM FW System must withstand pressure and water forces associated with deep and moving flood waters. Resultant loads on the HI-STORM FW 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 FW 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 is 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 moving. The maximum allowable flood water velocity (Table 2.2.8) is established so that the pressure loading from the water is less than the pressure loading which would cause the HI-STORM FW System to slide or tip over. Site-specific safety reviews by the licensee must confirm that flood parameters at the proposed ISFSI site do not exceed the flood depth or water velocity given in Table 2.2.8.

If the flood water depth exceeds the elevation of the top of the HI-STORM FW overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the overpack. Blockage of the air inlets is addressed in 2.2.3 (l).

The hydrological conditions at most reactor sites are 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 FW System for kinematic stability, if necessary. An evaluation for tsunamis† for certain coastal sites should also be performed to demonstrate that the maximum flood depth in Table 2.2.8 will not be exceeded. The factor of safety against sliding or overturning of the cask under the moving flood waters shall be equal to or greater than the value in Table 2.2.8.

The scenario where the flood water raises high enough to block the inlet ducts (and thus cut-off ventilation) and remains stagnant is the most adverse flood condition (thermally) for the storage system. As discussed in Chapter 1, the HI-STORM FW System inlet vent design makes it resistant to such adverse flood scenarios. The results of this analysis are presented in Chapter 4.

g. Earthquakes

The principal effect of an earthquake on the loaded HI-STORM FW overpack is the movement of the MPC inside the overpack cavity causing impact with the cavity inner wall, and, if the earthquake is sufficiently strong, the potential sliding and tilting of the storage system. The

† 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.

acceptance criteria for the storage system under the site's Design Basis Earthquake (DBE) are as follows:

i. The loaded overpacks will not impact each other during the DBE event.
ii. The loaded overpack will not slide off the ISFSI.
iii. The loaded overpack will not tip over.
iv. The confinement boundary will not be breached.

To minimize the need for a seismic analysis at each ISFSI site, the approach utilized in Docket No. 72-1014 is adopted for HI-STORM FW, which divides the DBE into two categories, labeled herein as (i) low intensity and (ii) high intensity. A low intensity earthquake is one whose ZPA is low enough to pass the "static equilibrium test". A high intensity earthquake is one that cannot pass the "static equilibrium test". The limiting value of the static friction coefficient, μ, has been set at 0.53 for freestanding HI-STORM overpack on a reinforced concrete pad in Docket No. 72-1014. The same limit is observed for HI-STORM FW overpack in this report. The criterion for static equilibrium is derived from elementary statics with the simplifying assumption that the cask and its contents are fixed and emulate a rigid body with six degrees-of-freedom. The earthquake is represented by its ZPA in horizontal (the vector sum of the two horizontal ZPAs for a 3-D earthquake site) and vertical directions. The limits on a_H and a_v for HI-STORM FW are readily derived as follows:

i. Prevention of sliding: Assuming the vertical ZPA to be acting to reduce the weight of the cask, horizontal force equilibrium yields:

$$W \cdot a_H \leq \mu \cdot W \cdot (1-a_v)$$

Or $$a_H \leq (1-a_v) \cdot \mu$$

ii. Prevention against "edging" of the cask:

Balancing the moment about the cask's pivot point for edging yields:

$$W \cdot a_H \cdot h \leq W \cdot (1-a_v) \cdot r$$

Or $$a_H \leq (1-a_v) \cdot \frac{r}{h}$$

Where:

r: radius of the footprint of the cask's base
h: height of the CG of the cask
μ: Static friction coefficient between the cask and the ISFSI pad.

The above two inequalities define the limits on a_H and a_v for a site if the earthquake is to be considered of "low intensity." For low intensity earthquake sites, additional analysis to demonstrate integrity of the confinement boundary is not required.

However, if the earthquake's ZPAs do not satisfy either of the above inequalities, then a dynamic analysis using the methodology specified in Chapter 3 shall be performed as a part of the §72.212 safety evaluation.

h. 100% Fuel Rod Rupture

The HI-STORM FW System must withstand loads due to 100% fuel rod rupture. For conservatism, 100% of the fuel rods are assumed to rupture 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. All of the fill gas contained in non-fuel hardware, such as burnable poison rod assemblies (BPRAs), is also assumed to be released concomitantly.

i. Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified will cause failure of the confinement boundary. Section 7.1 provides the rationale to treat leakage of the radiological contents from the MPC as a non-credible event.

j. External Pressure on the MPC Due to Explosion

The loaded HI-STORM FW overpack 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 Subsection 3.4.1, or appropriate preventive measures are taken to preclude internal explosive events (see Subsection 1.2.1). The MPC is composed of non explosive materials and maintains an inert gas environment. Thus explosion during long term storage is not credible. Likewise, the mandatory use of the protective measures at nuclear plants to prevent fires and explosions and the absence of any need for an explosive material during loading and unloading operations eliminates the scenario of an explosion as a credible event. Furthermore, because the MPC is internally pressurized, any short-term external pressure from explosion or even submergence in flood waters will act to reduce the tensile state of stress in the enclosure vessel. Nevertheless, a design basis external pressure (Table 2.2.1) has been defined as a design basis loading event wherein the internal pressure is non-mechanistically assumed to be absent.

k. Lightning

The HI-STORM FW System must withstand loads due to lightning. The effect of lightning on the HI-STORM FW System is evaluated in Chapter 12.

l. Burial Under Debris and Duct Blockage

Debris may collect on the HI-STORM FW overpack vent screens as a result of floods, wind storms, or mud slides. Siting of the ISFSI pad shall ensure that the storage location is not located

over shifting soil. However, if burial under debris is a credible event for an ISFSI, then a thermal analysis to analyze the effect of such an accident condition shall be performed for the site using the analysis methodology presented in Chapter 4. The duration of the burial-under-debris scenario will be based on the ISFSI owner's emergency preparedness program. The following acceptance criteria apply to the burial-under-debris accident event:

i. The fuel cladding temperature shall not exceed the ISG-11, Revision 3 [2.0.1] temperature limits.
ii. The internal pressure in the MPC cavity shall not exceed the accident condition design pressure limit in Table 2.2.1.

The burial-under-debris analysis will be performed if applicable, for the site-specific conditions and heat loads.

m. Extreme Environmental Temperature

The HI-STORM FW System must withstand extreme environmental temperatures. The extreme accident level temperature is specified in Table 2.2.2. The extreme accident level temperature is assumed to occur with steady-state insolation. This temperature is assumed to persist for a sufficient duration to allow the system to reach steady-state temperatures. The HI-STORM FW overpack and MPC have a large thermal inertia; therefore, extreme environmental temperature is a 3-day average for the ISFSI site.

All accident events and extreme environmental phenomena loadings that require analysis are listed in Table 2.2.13 along with the applicable acceptance criteria.

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

Loadings in the second category produce global deformations that must be shown to comply with the applicable acceptance criteria. The relevant loading combinations for the fuel basket, the MPC, the HI-TRAC VW transfer cask and the HI-STORM FW 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.

2.2.4 Applicability of Governing Documents

Section III Subsection NB of the ASME Boiler and Pressure Vessel Code (ASME Code), [2.2.10], is the governing code for the structural design of the MPC. The alternatives to the

ASME Code, Section III Subsection NB, applicable to the MPC in Docket Nos. 72-1008 and 72-1014 are also applicable to the MPC in the HI-STORM FW System, as documented in Table 2.2.14.

The stress limits of ASME Section III Subsection NF [2.0.3] are applied to the HI-STORM FW and HI-TRAC VW structural parts where the applicable loading is designated as a code service condition.

The fuel basket, made of Metamic-HT, is subject to the requirements in Appendix 1.B and is designed to a specific (lateral) deformation limit of its walls under accident conditions of loading (credible and non-mechanistic) (see Table 2.2.11). The basis for the lateral deflection limit in the active fuel region, θ, is provided in [2.2.11].

ACI 318 is the reference code for the plain concrete in the HI-STORM FW overpack. ACI 318.1-85(05) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited in strength analysis.

Each structure, system and component (SSC) of the HI-STORM FW System that is identified as important-to-safety is shown on the licensing drawings.

Tables 1.2.6, 1.2.7, and 1.2.8 provide the information on the applicable Codes and Standards for material procurement, design, fabrication and inspection of the components of the HI-STORM FW System. In particular, the ASME Code is relied on to define allowable stresses for structural analyses of Code materials.

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:

i. Level A Service Limits are used to establish allowables for normal condition load combinations.

ii. Level B Service Limits are used to establish allowables for off-normal conditions.

iii. Level C Service Limits are not used.

iv. Level D Service Limits are used to establish allowables for certain accident conditions.

The ASME Code service limits are used in the structural analyses for definition of allowable stresses and allowable stress intensities, as applicable. 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. Table 2.2.12 lists allowable stress limits for the steel structure of the HI-STORM FW overpack and HI-TRAC VW transfer cask which are analyzed to meet the stress limits of Subsection NF, Class 3 for loadings defined as service levels A, B, and D 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. The loads are listed in Tables 2.2.7 and 2.2.13, along with the applicable acceptance criteria.

2.2.7 Design Basis Loads

Where appropriate, for each loading type, a bounding value is selected in this FSAR to impute an additional margin for the associated loading events. Such bounding loads are referred to as Design Basis Loads (DBL) in this FSAR. For example, the Design Basis External Pressure on the MPC, set down in Table 2.2.1, is a DBL, as it grossly exceeds any credible external pressure that may be postulated for an ISFSI site.

2.2.8 Allowable Limits

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 displacement limit for the MPC fuel basket is expressed as a dimensionless parameter θ defined as [2.2.11]

$$\theta = \frac{\delta}{w}$$

where δ is defined as the maximum total deflection sustained by the basket panels under the loading event and w is the nominal inside (width) dimension of the storage cell. The limiting value of θ is provided in Table 2.2.11. Finally, the steel structure of the overpack and the HI-TRAC VW must meet the stress limits of Subsection NF of ASME Code, Section III for the applicable service conditions.

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

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

S_y: Minimum yield strength at temperature

S_u: Minimum ultimate strength at temperature

Table 2.2.1 DESIGN PRESSURES		
Pressure Location	**Condition**	**Pressure (psig)**
MPC Internal Pressure	Normal	100
	Off-Normal/Short-Term	120
	Accident	200
MPC External Pressure	Normal	(0) Ambient
	Off-Normal/Short-Term	(0) Ambient
	Accident	55
HI-TRAC Water Jacket Internal Pressure	Accident	65
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal/Short-Term	(0) Ambient
	Accident	See Paragraph 3.1.2.1.d

Table 2.2.2

ENVIRONMENTAL TEMPERATURES

HI-STORM FW Overpack		
Condition	Temperature (°F)	Comments
Normal Ambient Temperature	80	Bounding annual average from the contiguous United States
Soil Temperature	77	Bounding annual average from the contiguous United States
Off-Normal Ambient Temperature	-40 (min) 100 (max)	Lower bound does not consider insolation. Upper bound is a 3-day daily average and analysis includes insolation.
Extreme Ambient Temperature	125	3-day daily average and analysis includes insolation
Short-Term Operations	0 (min)	Limit is specified in the technical specifications.
HI-TRAC VW Transfer Cask		
Condition	Temperature (°F)	Comments
Short-Term Operations	0 (min.) 90 (max.)	The lower bound limit is specified in the technical specifications. The upper bound limit is a 3-day daily average with insolation and can be increased for a specific site if justified by the appropriate thermal analysis.

Table 2.2.3 DESIGN TEMPERATURES		
HI-STORM FW Component	**Normal Condition Design Temperature Limits (°F)**	**Off-Normal and Accident Condition Temperature Limits † (°F)**
MPC shell	600	800
MPC basket	752	932
MPC basket shims	752	932
MPC lid	600	800
MPC closure ring	500	800
MPC baseplate	400	800
HI-TRAC VW inner shell	500	700
HI-TRAC VW bottom lid	350	700
HI-TRAC VW top flange	400	650
HI-TRAC VW bottom lid seals	350	N/A
HI-TRAC VW bottom lid bolts	350	800
HI-TRAC VW bottom flange	350	700
HI-TRAC VW radial neutron shield	311	N/A
HI-TRAC VW radial lead gamma shield	350	600
Fuel Cladding	752 (Storage) 752 or 1058 (Short Term Operations)††	1058 (Off-Normal and Accident Conditions)
Overpack concrete	300	350
Overpack Lid Top and Bottom Plate	450	700
Remainder of overpack steel structure	350	700

† For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the fire event, the structure is required to remain physically stable (no specific temperature limits apply)

†† Short term operations include MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

Table 2.2.4

CHARACTERISTICS OF REFERENCE TORNADO

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 LIFTING ANALYSIS CASES					
Loading Case	Item	Location of Threaded Anchor (Material)	Bounding Weight	Dynamic Amplification Factor	Permissible Stress (psi) (Note 1)
HA.	Loaded MPC	Top Lid (stainless steel)	Section 3.2	1.15	Lesser of 0.1 S_u or $S_y/3$
HB.	Loaded HI-TRAC Transfer Cask	Top Flange of the Cask (C.S. forging)	Section 3.2	1.15	Lesser of 0.1 S_u or $S_y/3$
HC.	Loaded HI-STORM 100 Module with Lid	Threaded cylinder embedded and welded to the radial connectors near the top of the cask (carbon steel forging)	Section 3.2	1.15	$S_y/3$
Note 1: The permissible stress applies to the material of the part in which the lift anchor location is tapped. Minimum threaded length of the top shall be used in the analysis. S_u = ultimate strength; S_y = yield strength					

Table 2.2.7

LOADS APPLICABLE TO THE NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

Loading Case	Loading	Affected Item and Part	Magnitude of Loading	Acceptance Criterion
NA.	Snow and Ice	Top lid of HI-STORM FW overpack	Table 2.2.8	The stress in the steel structure must meet NF Class 3 limits for linear structures
NB.	Internal Pressure[1]	MPC Enclosure Vessel	Table 2.2.1	Meet "NB" stress intensity limits
	a. Normal Condition	MPC Enclosure Vessel	Table 2.2.1	Level A condition limit on primary plus secondary stress intensities
	b. Off-Normal Condition	MPC Enclosure Vessel	Table 2.2.1	Level B limits on primary and secondary stress intensities.

1 Normal condition internal pressure is bounded by the Design Internal Pressure in Table 2.2.1. Because the top and bottom extremities of the MPC Enclosure Vessel are each at a uniform temperature due to the recirculating helium, thermal stresses are minimal. Therefore, the Design Internal Pressure envelops the case of the Normal Service condition for the MPC. The same remark applies to the Off-Normal Service condition.

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
Assumed Blockage of MPC Basket Flow Opening by Crud Settling (Depth of Crud, in.)	Accident	1
Cask Environment During the Postulated Fire Event (Deg. F)	Accident	1475
HI-STORM FW Overpack Fire Duration (seconds)	Accident	208
HI-TRAC VW Transfer Cask Fire Duration (minutes)	Accident	4.64
Maximum Submergence Depth due to Flood (ft)	Accident	125
Factor of safety against sliding or overturning from moving flood waters	Accident	1.1

Table 2.2.9 ISFSI PAD DATA FOR NON-MECHANISTIC TIP-OVER ANALYSIS	
Thickness (inch)	36
Concrete Pad Compressive Strength (psi)	6,000
Modulus of elasticity of the subgrade (psi)	28,000

Table 2.2.10

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

Stress Category	Design	Level A	Level D††
Primary Membrane, P_m	S_m	S_m	AMIN ($2.4S_m$, $.7S_u$)
Local Membrane, P_L	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	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.
†††† Governed by NB-3227.2 or F-1331.1(d).

Table 2.2.11 STRUCTURAL DESIGN CRITERIA FOR THE FUEL BASKET	
PARAMETER	VALUE
Minimum service temperature	-40°F
Maximum total (lateral) deflection in the active fuel region - dimensionless	0.005

Table 2.2.12
STRESS AND ACCEPTANCE LIMITS FOR DIFFERENT LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE HI-STORM FW OVERPACK AND HI-TRAC VW

STRESS CATEGORY	DESIGN + NORMAL	OFF-NORMAL	ACCIDENT†
Primary Membrane, P_m	S	$1.33 \cdot S$	See footnote
Primary Membrane, P_m, plus Primary Bending, P_b	$1.5 \cdot S$	$1.995 \cdot S$	See footnote
Shear Stress (Average)	$0.6 \cdot S$	$0.6 \cdot S$	See footnote

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 Stress

† Under accident conditions, the cask must maintain its physical integrity, the loss of solid shielding (lead, concrete, steel, as applicable) shall be minimal and the MPC must remain recoverable.

Table 2.2.13 LOADING EVENTS AND ACCEPTANCE CRITERIA APPLICABLE TO ACCIDENT CONDITIONS AND EXTREME ENVIRONMENTAL PHENOMENA				
Loading Case	Loading or Event	Affected Item or Part	Characteristics of Loading	Notes and Acceptance Criterion
AA.	Non-Mechanistic Tip-Over	HI-STORM FW overpack, Fuel Basket and Enclosure Vessel	Impactive load from the slap-down of the loaded overpack	See Paragraph 2.2.3(b)
AB.	Fire	Fuel Cladding, Shielding Concrete, and FW overpack steel structure	Significant radiant heat input over a short time	See Paragraph 2.2.3(c)
AC.	Tornado-Borne Missile	HI-STORM FW overpack	Impactive loading (Table 2.2.5)	See Paragraph 2.2.3(e)
	a. Large Missile	HI-STORM FW overpack	Acting to tip-over the loaded overpack	Use lower bound cask weight, demonstrate kinematic stability
	b. Medium Missile	HI-STORM FW overpack	May damage shielding concrete	Use lower bound cask weight, demonstrate kinematic stability
	c. Small Missile	HI-STORM FW overpack	Penetration	Prevent penetration of the cask and access to the MPC
AD.	Moving Floodwaters	Loaded Storage Module	Acting to tip-over the loaded overpack (Table 2.2.8)	See Paragraph 2.2.3 (f). Use both lower bound and upper bound cask height and weight to demonstrate kinematic stability.
AE.	Design Basis Earthquake	Loaded Storage Module	Acting to destabilize the cask	See Paragraph 2.2.3(g).
AF.	100% Rod Rupture	MPC confinement boundary	Acts to overpressure the MPC and raise the temperature of the fuel cladding	See Paragraph 2.2.3(h). Demonstrate that the equilibrium pressure in the MPC remains below the Accident Condition Design Pressure (Table 2.2.1) and ISG-11 temperature limits are met by the fuel cladding.

Table 2.2.13 LOADING EVENTS AND ACCEPTANCE CRITERIA APPLICABLE TO ACCIDENT CONDITIONS AND EXTREME ENVIRONMENTAL PHENOMENA				
AG.	Burial Under Debris	Stored SNF	Blocks convection and retards conduction as means for heat dissipation	See Paragraph 2.2.3(l). Determine the permissible time elapsed under debris so that the pressure in the MPC does not exceed the Accident Condition Design Pressure and the fuel cladding temperature remains below the ISG-11 limit.
AH	Design Basis External Pressure	MPC Enclosure Vessel	An assumed non-mechanistic load from deep submergence in flood water or explosion in the vicinity of the ISFSI	Demonstrate that the MPC Enclosure Vessel will not buckle, i.e., become structurally unstable
AJ.	Internal pressure developed in the HI-TRAC water jacket	HI-TRAC Water Jacket	A non-mechanistic (postulated) event	The water jacket will meet Level D stress limits for "NF" components.

TABLE 2.2.14
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

MPC Enclosure Vessel	Subsection NCA	General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.	Because the MPC is not an ASME Code stamped vessel, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the MPCs as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program. Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.
MPC Enclosure Vessel	NB-1100	Statement of requirements for Code stamping of components.	MPC Enclosure Vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC basket supports and lift lugs	NB-1130	NB-1132.2(d) requires that the first connecting weld of a non-pressure retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure retaining portion of the component, where t is the	The lugs that are used exclusively for lifting an empty MPC are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The lug-to-Enclosure Vessel Weld is required to meet the stress limits of Reg. Guide 3.61 in lieu of Subsection NB of the Code.

TABLE 2.2.14
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

		nominal thickness of the pressure retaining material. NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure retaining portion of the component.	
MPC Enclosure Vessel	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 Enclosure Vessel	NB-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are subsumed by the HI-STORM FW FSAR, serving as the Design Specification, which establishes the service conditions and load combinations for the storage system.
MPC Enclosure Vessel	NB-4120	NB-4121.2 and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.	In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, and coating are not, unless explicitly stated by the Code, defined as heat treatment operations.
MPC Enclosure Vessel	NB-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.

TABLE 2.2.14
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

MPC Enclosure Vessel	NB-4122	Implies that with the exception of studs, bolts, nuts and heat exchanger tubes, CMTRs must be traceable to a specific piece of material in a component.	MPCs are built in lots. Material traceability on raw materials to a heat number and corresponding CMTR is maintained by Holtec through markings on the raw material. Where material is cut or processed, markings are transferred accordingly to assure traceability. As materials are assembled into the lot of MPCs being manufactured, documentation is maintained to identify the heat numbers of materials being used for that item in the multiple MPCs being manufactured under that lot. A specific item within a specific MPC will have a number of heat numbers identified as possibly being used for the item in that particular MPC of which one or more of those heat numbers (and corresponding CMTRS) will have actually been used. All of the heat numbers identified will comply with the requirements for the particular item.
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.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only progressive liquid penetrant (PT) examination is permitted. PT examination will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be pressure tested as defined in Chapter 10. Accessibility for leakage inspections preclude a Code compliant pressure test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination. MPC shell and shell to baseplate welds are subject to a fabrication helium leak test prior to loading.

TABLE 2.2.14
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

			The MPC lid-to-shell weld shall be verified by progressive PT examination. PT must include the root and final layers and each approximately 3/8 inch of weld depth. The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, of other means which provide an equivalent record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350.
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM FW System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

The HI-STORM FW System is engineered to provide for the safe long-term storage of spent nuclear fuel (SNF). The HI-STORM FW will withstand all normal, off-normal, and postulated accident conditions without 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 normal and off-normal operating conditions and its retrievability for further processing or ultimate disposal in accordance with 10 CFR 72.122(l) and ISG-2 [2.3.1].

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity which the HI-STORM FW System must confine originates from the spent fuel assemblies and, to a lesser extent, any radioactive particles from contaminated water in the fuel pool which may remain inside the MPC. This radioactivity is confined by multiple engineered barriers.

Contamination on the outside of the MPC 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 VW, and the elastomer seal in the HI-TRAC VW bottom lid (see Chapter 9) prevent the fuel pool water from contacting the exterior of the MPC and interior of the HI-TRAC VW while submerged for fuel loading.

The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, MPC shell, MPC lid, closure ring, port cover plates, and associated welds.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, accident conditions, or external natural phenomena. Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity. Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7. MPC field weld examinations, helium leakage testing of the port cover plate welds, and pressure testing are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 10, to verify the integrity of the confinement boundary.

2.3.2.2 Cask Cooling

To ensure that an effective passive heat removal capability exists for long term satisfactory performance, several thermal design features are incorporated in the storage system. They are as follows:

- The MPC fuel basket is formed by a honeycomb structure of Metamic-HT plates which allows the unimpeded conduction of heat from the center of the basket to the periphery. The MPC cavity is equipped with the capability to circulate helium internally by natural buoyancy effects and transport heat from the interior region of the canister to the peripheral region (Holtec Patent 5,898,747).
- The MPC confinement boundary ensures that the inert gas (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 ISG-11 limits such that fuel cladding does not experience degradation during the long term storage period.
- The HI-STORM FW is optimally designed, with cooling vents and an MPC to overpack annulus, which maximize air flow by ensuring a turbulent flow regime at maximum heat loads.
- Eight inlet ducts located circumferentially around the bottom of the overpack and the outlet vent which circumscribes the entire lid of HI-STORM FW render the ventilation action insensitive to shifting wind conditions.

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Equipment

Design criteria for the HI-STORM FW System are described in Section 2.2. The HI-STORM FW System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized outside of the reactor facility 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 provides guidance for the determination of a component's safety classification [1.1.4].

Users may perform the MPC transfer between the HI-TRAC VW transfer cask and the HI-STORM FW overpack in a location of their choice, depending upon site-specific needs and capabilities. For those users choosing to perform the MPC transfer using devices not integral to structures governed by the regulations of 10 CFR Part 50 (e.g., fuel handling or reactor building), a Canister Transfer Facility (CTF) is required. The CTF is typically a concrete lined cavity of a suitable depth to stage the overpack inside it so that the top of the cask is near grade level (Holtec Patent 7,139,358B2). With the overpack staged inside the cavity, the mating device is installed on top and the HI-TRAC VW is mounted on top of the mating device. The MPC

transfer is carried out by actuating the mating device and moving the MPC vertically to the cylindrical cavity of the recipient cask. The mating device is actuated by removing the bottom lid of the HI-TRAC VW transfer cask (see Figure 1.1.2). The device utilized to lift the HI-TRAC VW transfer cask to place it on the overpack and to vertically transfer the MPC may be of stationary or mobile type, but it must have redundant drop protection features. The cask transporter can be the load handling device at the CTF.

2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STORM FW System, instrumentation, which is important to safety, is not necessary. No instrumentation is required or provided for HI-STORM FW storage operations, other than normal security service instruments and dosimeters.

However, in lieu of performing the periodic inspection of the HI-STORM FW overpack vent screens, temperature elements may be installed in the overpack exit vents to continuously monitor the air temperature. If the temperature elements and associated temperature monitoring instrumentation are used, they shall be designated important to safety.

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:

- Fuel basket constructed of neutron absorbing material with no potential of detachment.
- Favorable geometry provided by the MPC fuel basket.
- A high B-10 concentration (50% greater than the concentration used in the existing state-of-the art designs certified under 10CFR72) leads to a lower reactivity level under all operating scenarios.

Administrative controls shall be used to ensure that fuel placed in the HI-STORM FW 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 security fence surrounded by a physical barrier fence with an appropriate locking and monitoring system is a standard approach to limit access if the ISFSI is located outside the controlled area. 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 FW System.

2.3.5.2 Shielding

The objective of shielding is to assure that radiation dose rates at key locations are as low as practical in order to maintain occupational doses to operating personnel As Low As Reasonably Achievable (ALARA) and to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106 for dose at the controlled area boundary.

The HI-STORM FW is designed to limit dose rates in accordance with 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive doses in excess of the limits given in Table 2.3.1 for normal, off-normal, and accident conditions.

Three locations are of particular interest in the storage mode:

- immediate vicinity of the cask
- restricted area boundary
- controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded overpack are important in consideration of occupational exposure. Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on Reference BWR and PWR fuel (Table 1.0.4).

Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM FW System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask user.

Design objective dose rates for the HI-STORM FW overpack surfaces are presented in Table 2.3.2.

Because of the passive nature of the HI-STORM FW System, human activity related to the system after deployment in storage is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 11, wherein measures to reduce occupational dose are also discussed. The estimated occupational doses for personnel provided in Chapter 11 comply with the requirements of 10CFR20. As discussed in Chapter 11, the HI-STORM FW System has been configured to minimize both the site boundary dose in storage and occupational dose during short term operations to the maximum extent possible.

The analyses and discussions presented in Chapters 5, 9, and 11 demonstrate that the HI-STORM FW System is capable of meeting the radiation dose limits set down in Table 2.3.1.

2.3.5.3 Radiological Alarm System

The HI-STORM FW does not require a radiological alarm system. There are no credible events that could result in release of radioactive materials from the system and direct radiation exposure from the ISFSI is monitored using the plant's existing dose monitoring system.

2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM FW System. Combustible materials will not be stored within an ISFSI. However, for conservatism, a hypothetical fire accident has been analyzed as a bounding condition for HI-STORM FW System. The evaluation of the HI-STORM FW System fire accident is discussed in Chapter 12.

Explosive material will not be stored within an ISFSI. Small overpressures may result from accidents involving explosive materials which are stored or transported in the vicinity of the site. Explosion is an accident loading condition considered in Chapter 12.

Table 2.3.1

RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

MINIMUM DISTANCE TO BOUNDARY OF CONTROLLED AREA (m)	100
NORMAL AND OFF-NORMAL CONDITIONS:	
-Whole Body (mrem/yr)	25
-Thyroid (mrem/yr)	75
-Any Other Critical Organ (mrem/yr)	25
DESIGN BASIS ACCIDENT:	
-TEDE (rem)	5
-DDE + CDE to any individual organ or tissue (other than lens of the eye) (rem)	50
-Lens dose equivalent (rem)	15
-Shallow dose equivalent to skin or any extremity (rem)	50

Table 2.3.2 – Design Objective Dose Rates for HI-STORM FW Overpack Surfaces	
Area of Interest	Dose Rate (mrem/hr)
Radial Surface Excluding Vents	150
Inlet and Outlet Vents	250
Top of the Lid (Horizontal Surface)	60

2.4 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STORM FW System. The HI-STORM FW System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site. As discussed below, Holtec International has taken appropriate steps to ensure that the necessary equipment designs and certifications shall be available to the user of the HI-STORM FW System to expeditiously decommission the ISFSI at the end of the storage facility's required service life.

Towards that end, the MPC has been designed with the objective to transport it in a HI-STAR 190 transportation cask (Figure 2.4.1). Since the loaded MPC is a self-contained "Waste Package", no further handling of the SNF stored in the MPC is required prior to transport to a licensed centralized storage facility or repository.

The MPC which holds the SNF assemblies is engineered to be suitable as a waste package for permanent internment in a deep Mined Geological Disposal System (MGDS). The materials of construction permitted for the MPC are known to be highly resistant to severe environmental conditions. No carbon steel, paint, or coatings are used or permitted in the MPC in areas where they could be exposed to spent fuel pool water or the ambient environment. Therefore, the SNF assemblies stored in the MPC do 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) severing of the MPC closure ring to provide access to the MPC vent and drain. The circumferential welds of the MPC closure lid 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 SNF needs to be removed from the MPC before it is placed into the MGDS, the MPC interior metal surfaces can be decontaminated using existing mechanical or chemical methods to allow for its disposal. This will be facilitated by the 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 FW System may pose is slight activation of the HI-STORM FW materials caused by irradiation over the storage period.

Due to the design of the HI-STORM FW 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.

The long-lived radionuclides produced by the irradiation of the HI-STORM FW System components are listed in Table 2.4.1. The activation of the HI-STORM FW components shall be limited to a cumulative activity of 10 Ci per cubic meter before decommissioning and disposal of the activated item can be carried out.

In any case, the HI-STORM FW System would not impose any additional decommissioning requirements on the licensee of the ISFSI facility per 10CFR72.30, since the HI-STORM FW System could eventually be shipped from the site.

Table 2.4.1 PRINCIPAL LONG-LIVED ISOTOPES PRODUCED DURING IRRADIATION OF THE HI-STORM FW COMPONENTS			
Nuclide	**MPC Stainless Steel**	**HI-STORM Steel**	**HI-STORM Concrete**
^{54}Mn	X	X	X
^{55}Fe	X	X	X
^{59}Ni	X	-	-
^{60}Co	X	-	-
^{63}Ni	X	-	-
^{39}Ar	-	-	X
^{41}Ca	-	-	X

Withheld in Accordance with 10 CFR 2.390

Figure 2.4.1: HI-STAR 190 Transportation Overpack and MPC Shown in Exploded, Cut-Away View

2.5 REGULATORY COMPLIANCE

Chapter 2 provides the principal design criteria and applicable loading related to HI-STORM FW structures, systems, and components designated as 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 detailed evaluation of the design criteria and an assessment of compliance with those criteria are provided in Chapters 3 through 12.

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