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.

1.0.1 Engineering Change Orders

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

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

Affected Item	ECO Number	72.48 Evaluation or Screening Number
MPC-89 Basket	101-6	1021
MPC-37 Basket	102-9	N/A
MPC Enclosure Vessel	101-6	1021
	102-7	1021
HI-STORM FW Overpack	100-5	N/A
HI-TRAC FW	103-7	1021
	103-9	1036
General FSAR Changes	5018-17	1020
	5018-19	1021
	5018-20	N/A
	5018-23	N/A
	5018-24	N/A
	5018-25	N/A

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

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**
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)(2)	2.2.3
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
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2
	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

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
		10CFR72.236(d)	2.3.2, 2.3.5
		10CFR72.236(e)	2.3.2
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

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.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
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

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	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
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

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
6.4.1 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
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

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**
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.1
7.3.1 Fission Gas Products	--	--	7.1
7.3.2 Release of Contents	ISG-18	--	7.1
NA	--	10CFR72.106(b)	7.1
7.4 Supplemental Data	7.V Supplemental Info.	--	--

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. 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

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.d Test Program	10CFR72.162	10.1
	9.III.1.e Appropriate Tests	10CFR72.236(1)	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 (except as described in Subsection 4.5.1) 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). Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. Pursuant to ISG-18, the Holtec MPC is constructed in a manner that 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.

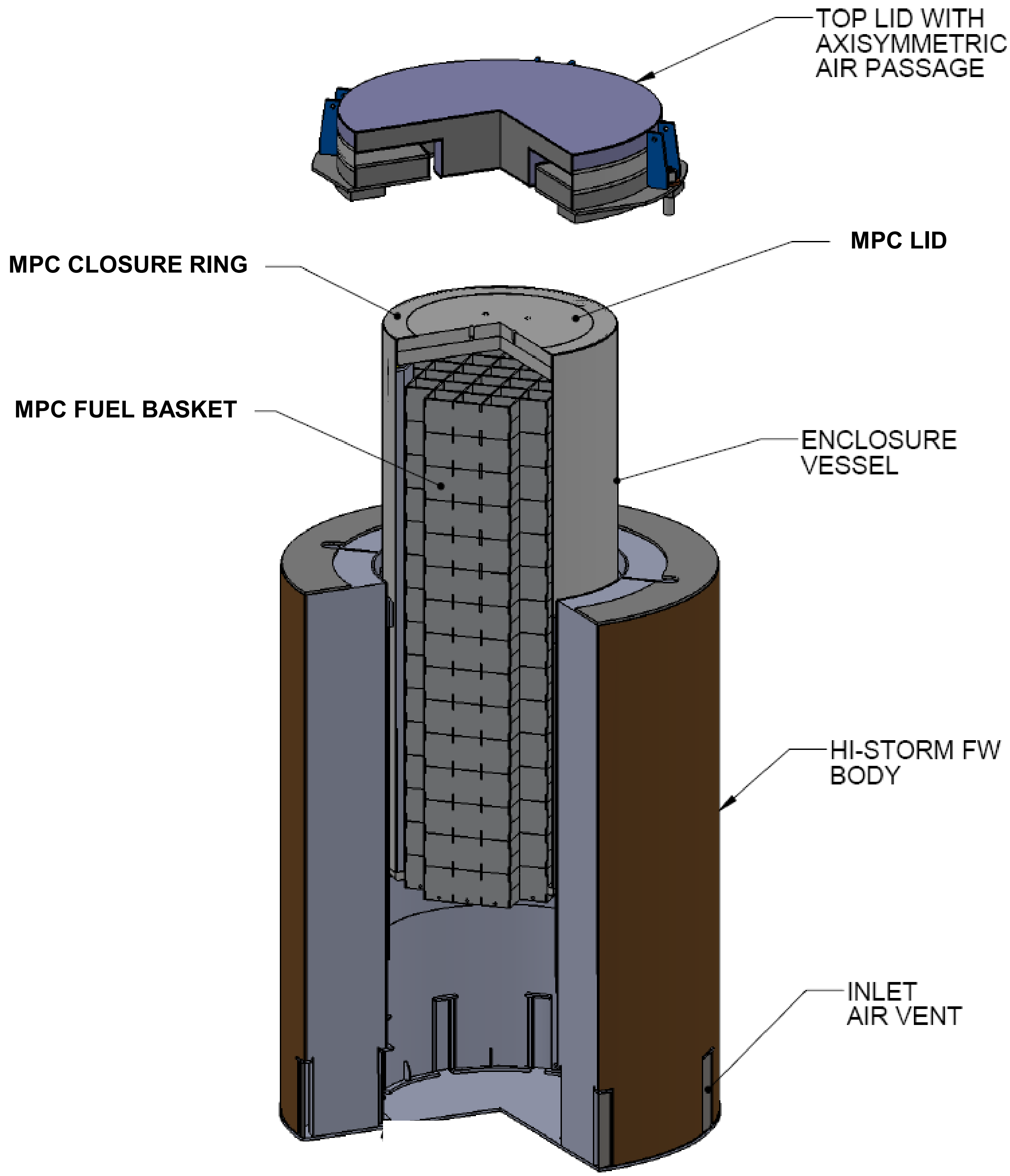


FIGURE 1.1.1: HI-STORM FW VVM AND MPC-37 IN CUTAWAY VIEW

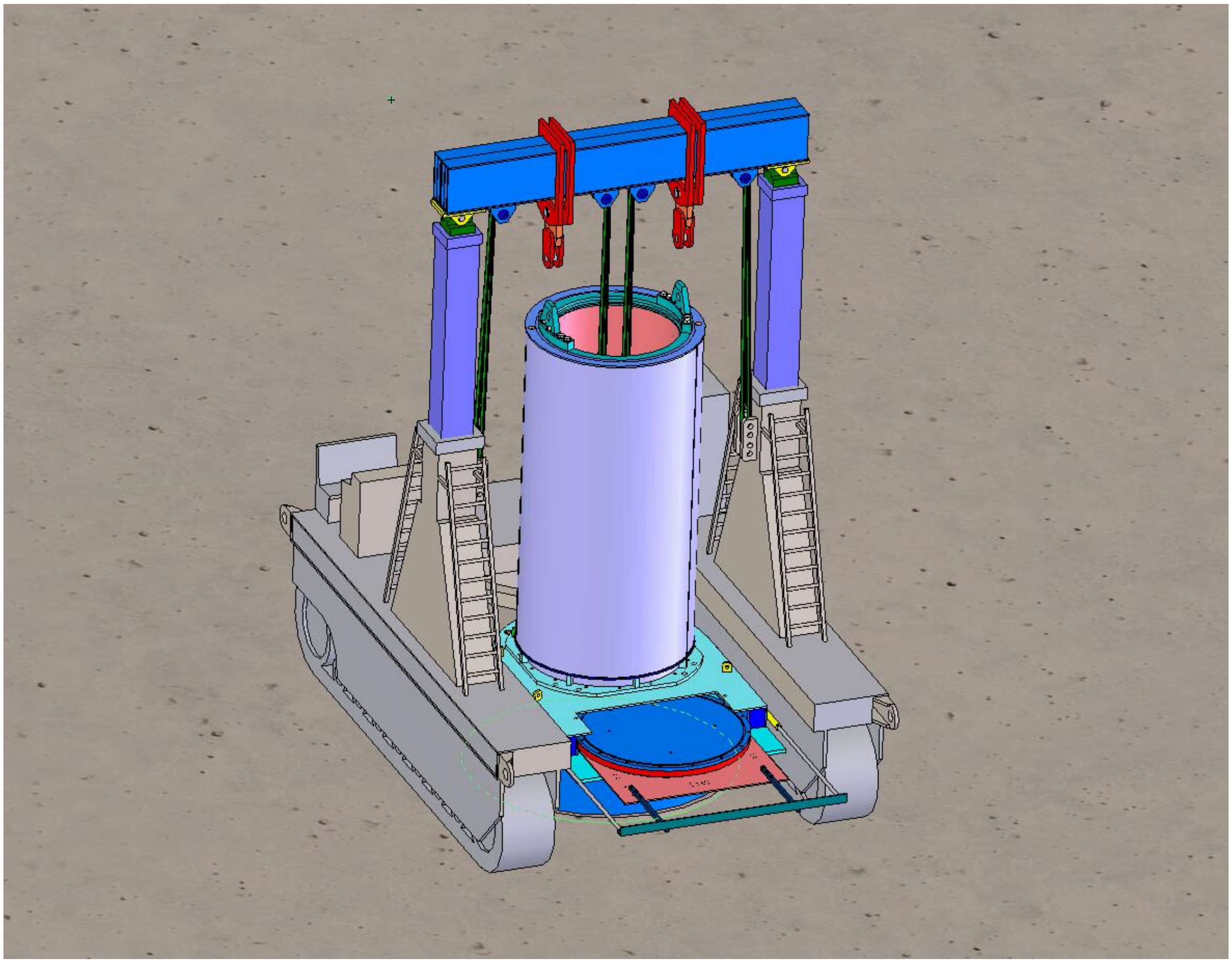


FIGURE 1.1.2: MPC TRANSFER AT THE CANISTER TRANSFER FACILITY (PIT)

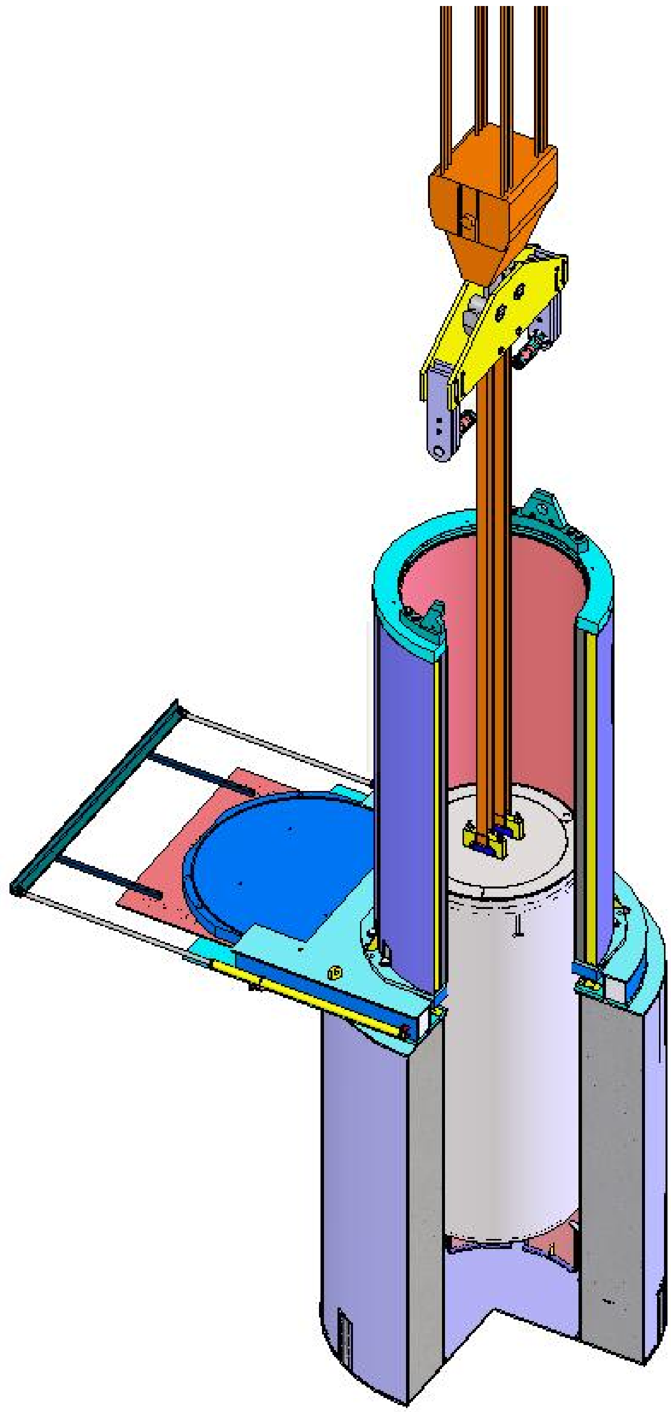


FIGURE 1.1.3: MPC TRANSFER IN THE PLANT'S EGRESS BAY

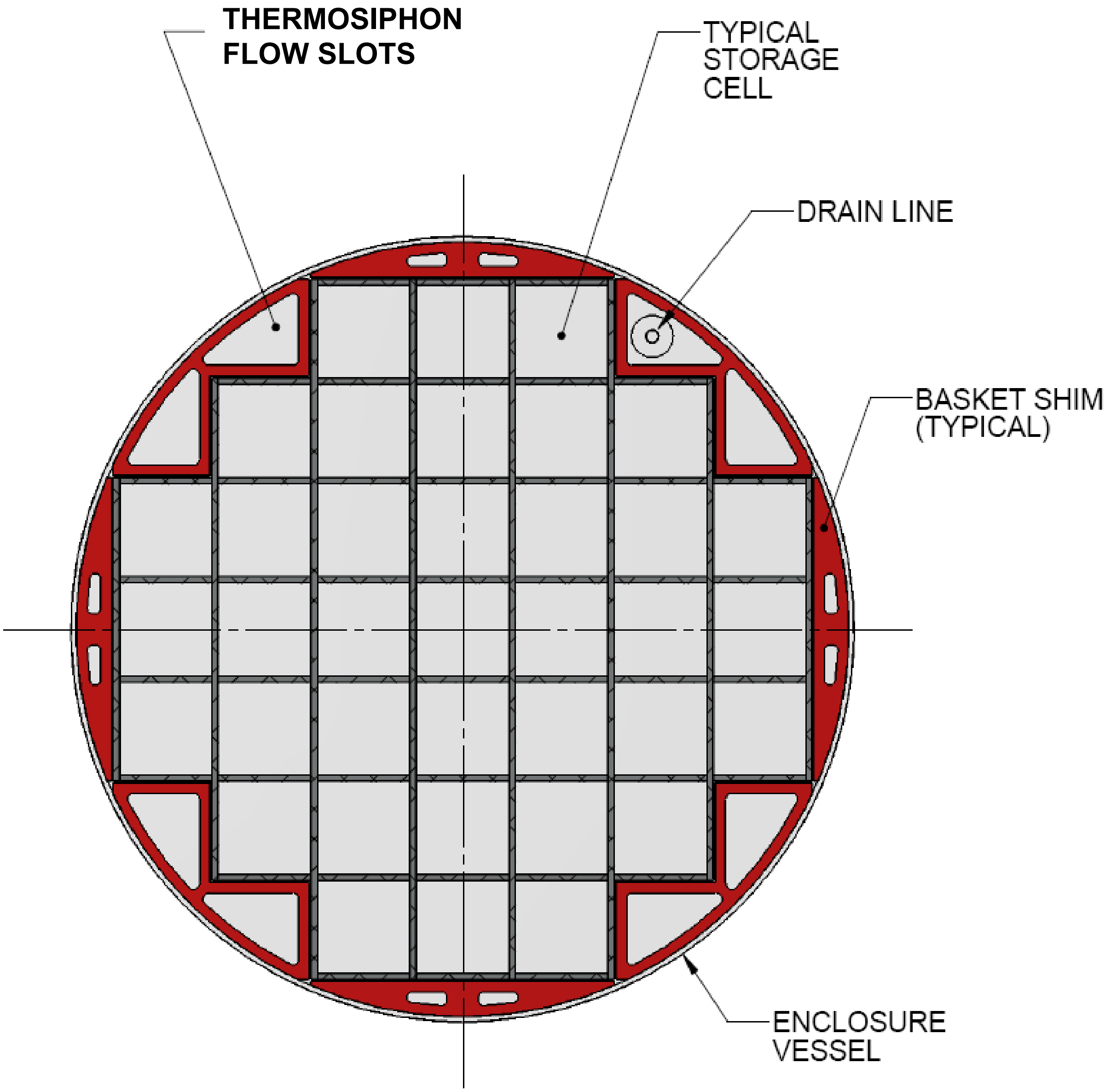


FIGURE 1.1.4: MPC-37 IN CROSS SECTION

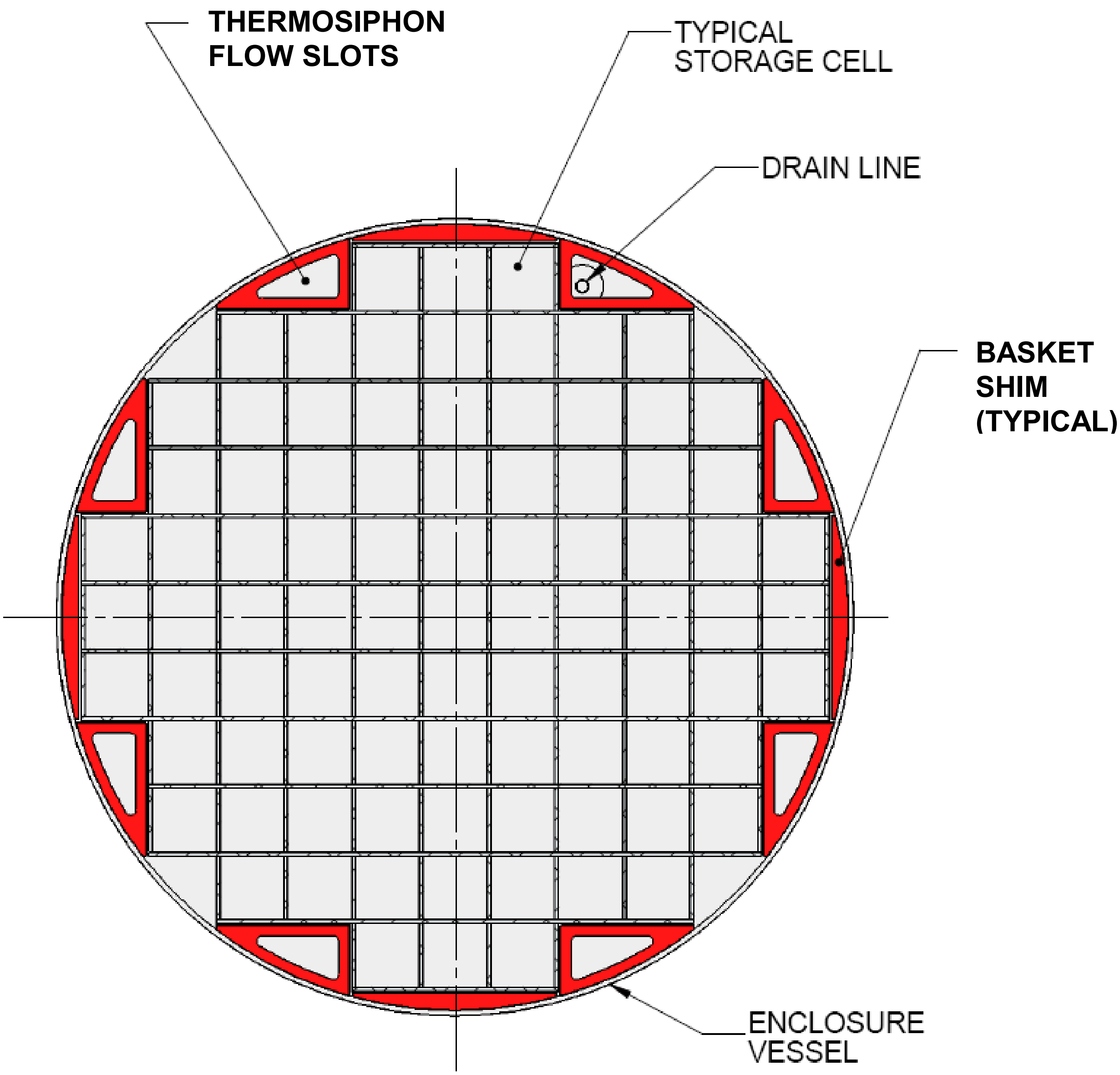


FIGURE 1.1.5: MPC-89 IN CROSS SECTION

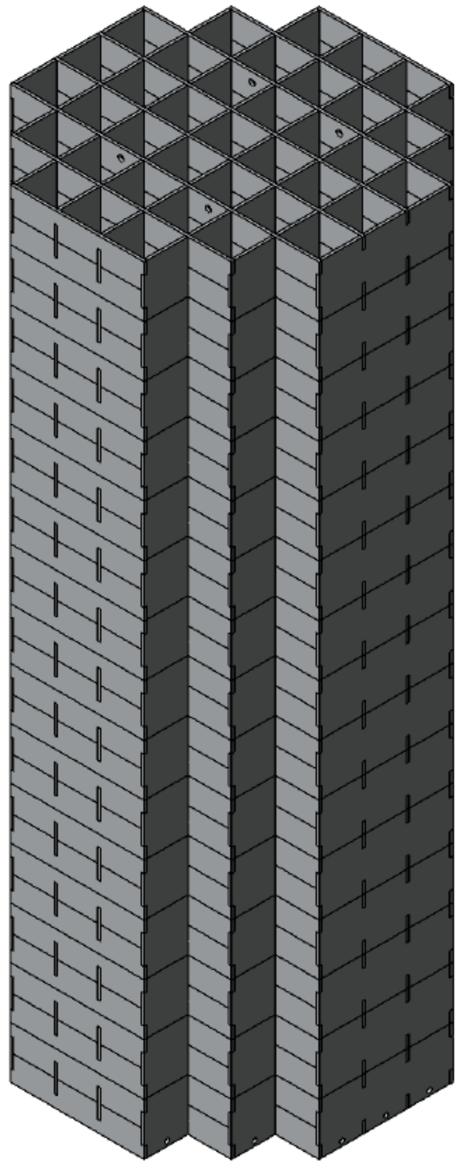


FIGURE 1.1.6: PWR FUEL BASKET (37 STORAGE CELLS) IN PERSPECTIVE VIEW

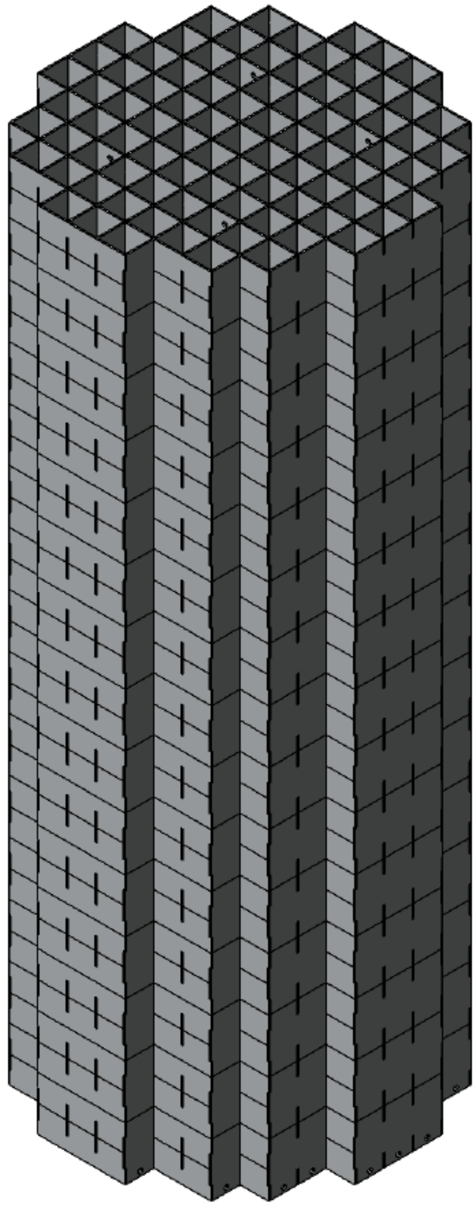


FIGURE 1.1.7: BWR FUEL BASKET (89 STORAGE CELLS) IN PERSPECTIVE VIEW

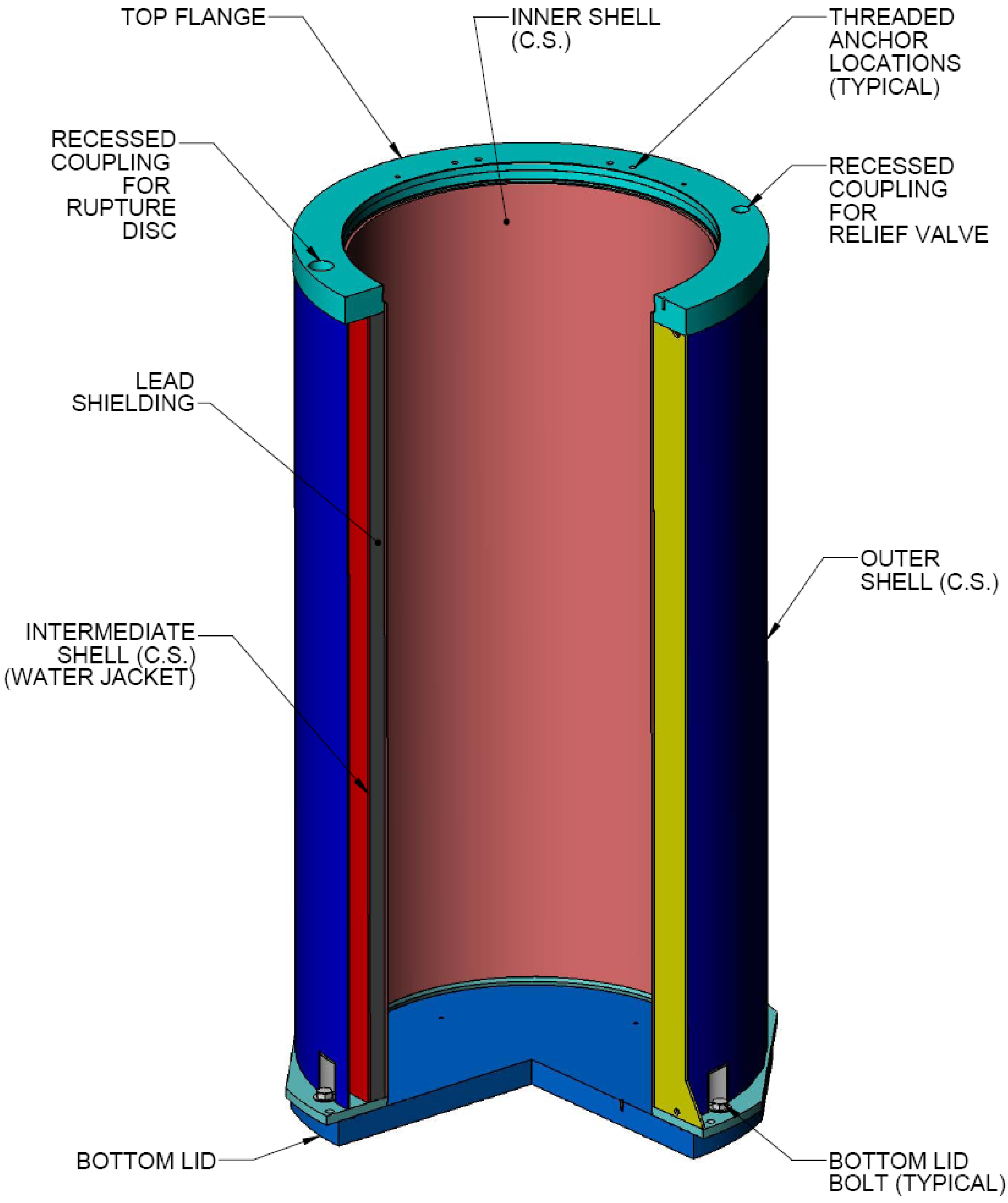


FIGURE 1.1.8: CUTAWAY VIEW OF HI-TRAC VW

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 cannot 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 to approximately 1.5-2.5 inches. A small axial clearance is provided to account for manufacturing tolerances and 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 Table 1.2.7. 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^{o} 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^{o} 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. An optional 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 below.

[**Withheld in Accordance with 10 CFR 2.390]**

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 90º 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 eight 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 (with the rare exception of the transfer cask as described in Subsection 4.5.1).

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. The sub-design heat loads for each cell, region and total canister are in Table 4.4.11.

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%
[Withheld in Accordance with 10 CFR 2.390]	**[Withheld in Accordance with 10 CFR 2.390**]	**[Withheld in Accordance with 10 CFR 2.390**]
[Withheld in Accordance with 10 CFR 2.390]	**[Withheld in Accordance with 10 CFR 2.390**]	**[Withheld in Accordance with 10 CFR 2.390**]
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 Design Basis Heat Load (kW): 44.09 (Pattern A); 45.0 (Pattern B)					
Region No.	Decay Heat Limit per Cell, kW		Number of Cells per Region	Decay Heat Limit per Region, kW	
	Pattern A	Pattern B		Pattern A	Pattern B
1	1.05	1.0	9	9.45	9.0
2	1.70	1.2	12	20.4	14.4
3	0.89	1.35	16	14.24	21.6

Note: See Chapter 4 for decay heat limits per cell when vacuum drying high burnup fuel.

TABLE 1.2.4 MPC-89 HEAT LOAD DATA (See Figure 1.2.2)			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Design Basis 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

TABLE 1.2.8

[Withheld in Accordance with 10 CFR 2.390]

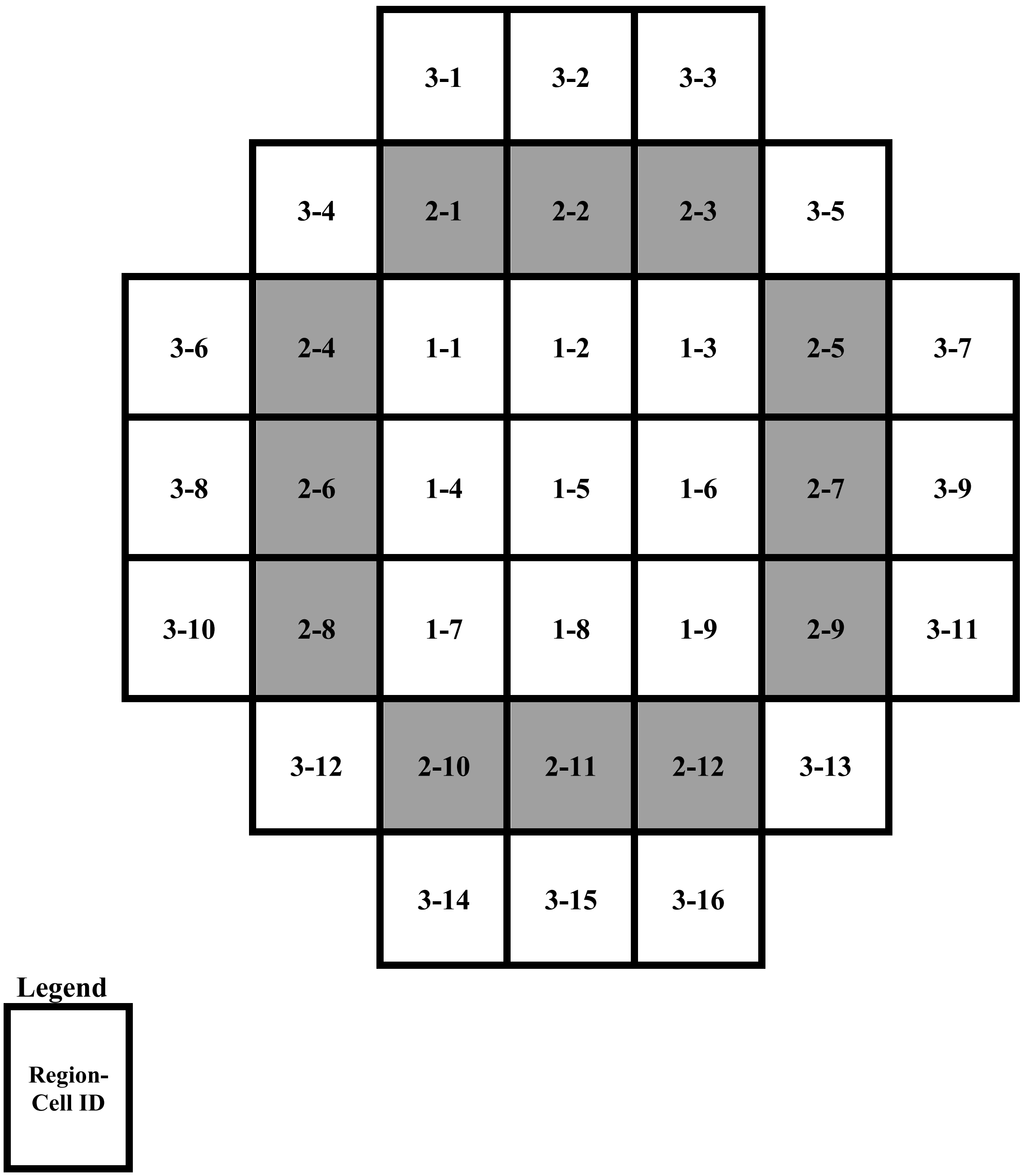


Figure 1.2.1: MPC-37 Basket, Region and Cell Identification

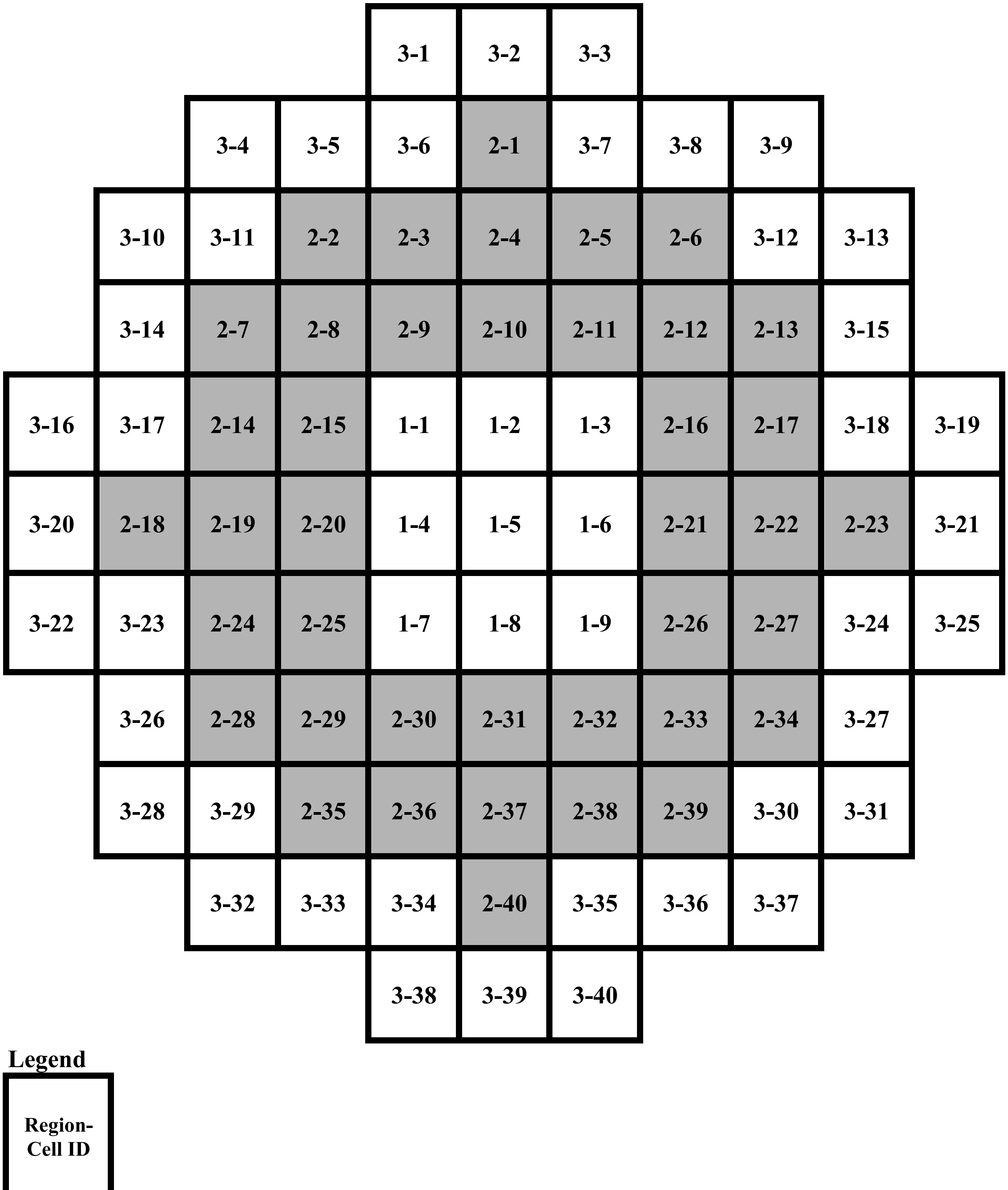


Figure 1.2.2: MPC-89 Basket, Region and Cell Identification

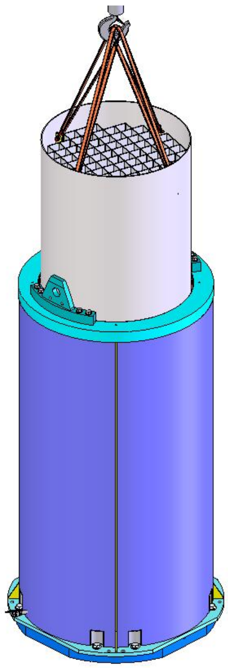
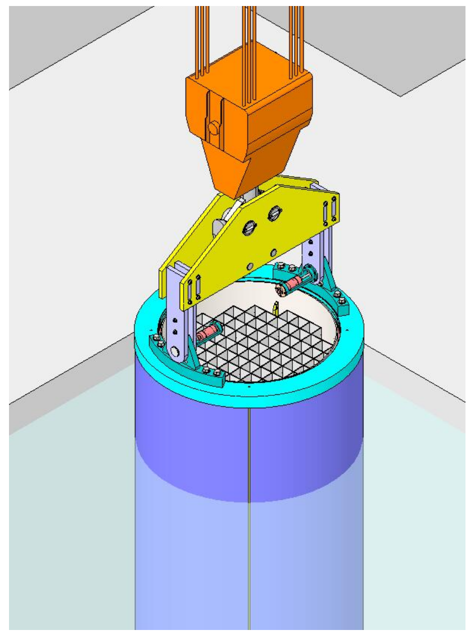
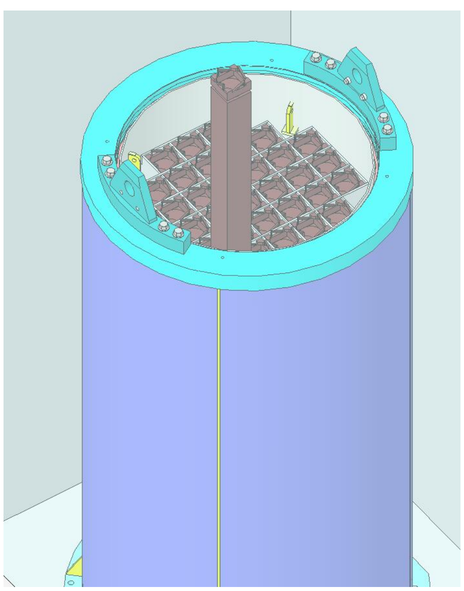
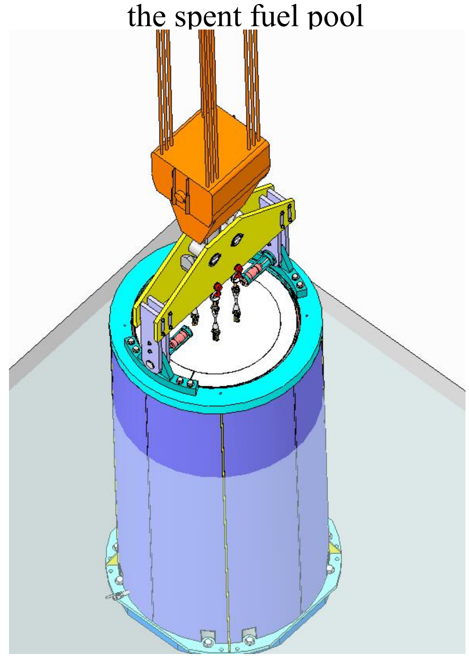
1. MPC Placed in HI-TRAC 	2. HI-TRAC Placement in the spent fuel pool 
3. Fuel Loading into the MPC 	4. HI-TRAC/MPC Removal from the spent fuel pool 

FIGURE 1.2.3: SUMMARY OF TYPICAL LOADING OPERATIONS

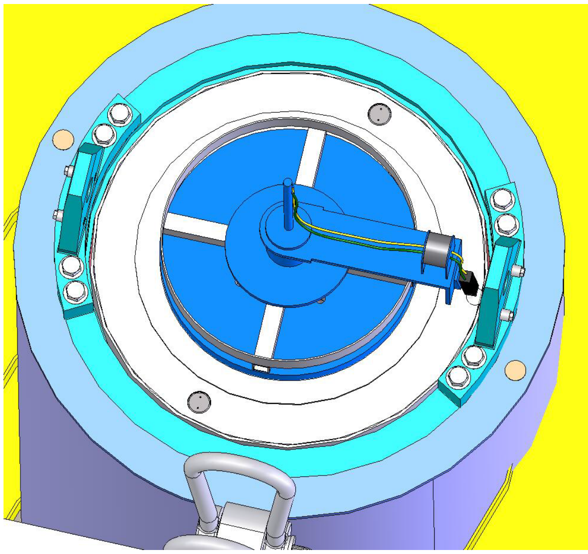
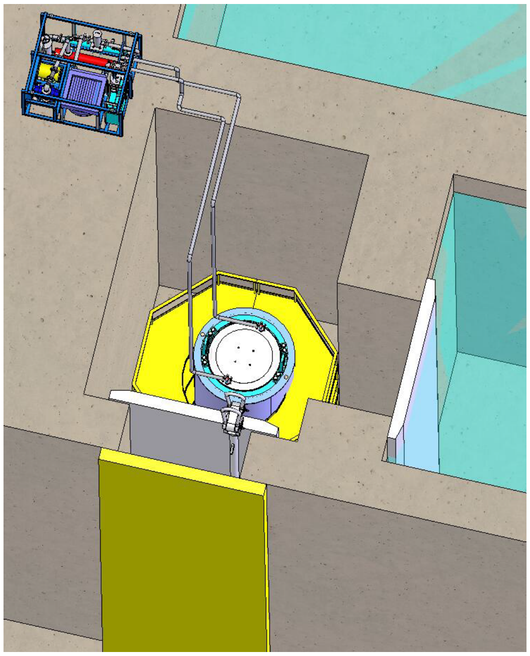
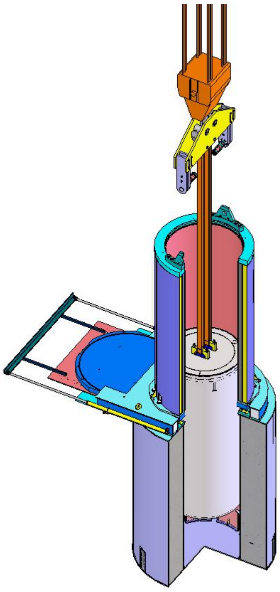
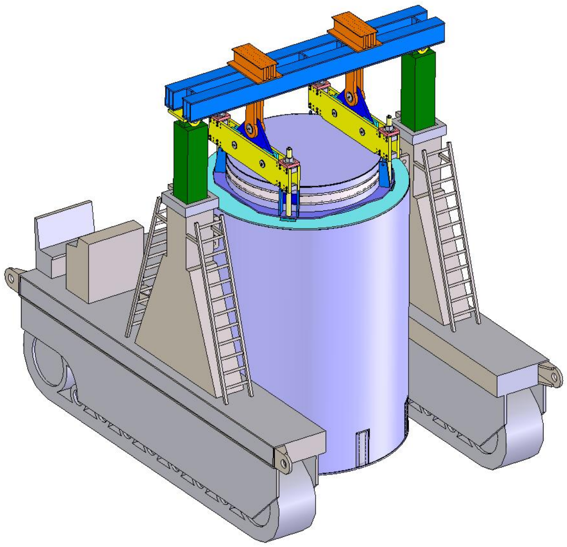
5. MPC Closure Operations (Lid to Shell Welding)	6. MPC Draining, Drying and Backfill
	
7. System Stackup and MPC Transfer Operations	8. HI-STORM Movement to the ISFSI
	

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-STORM Flood/Wind (Storage)	72-1032
HI-STORM UMAX (Storage)	72-1040
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 180D (Transportation)	71-9367
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

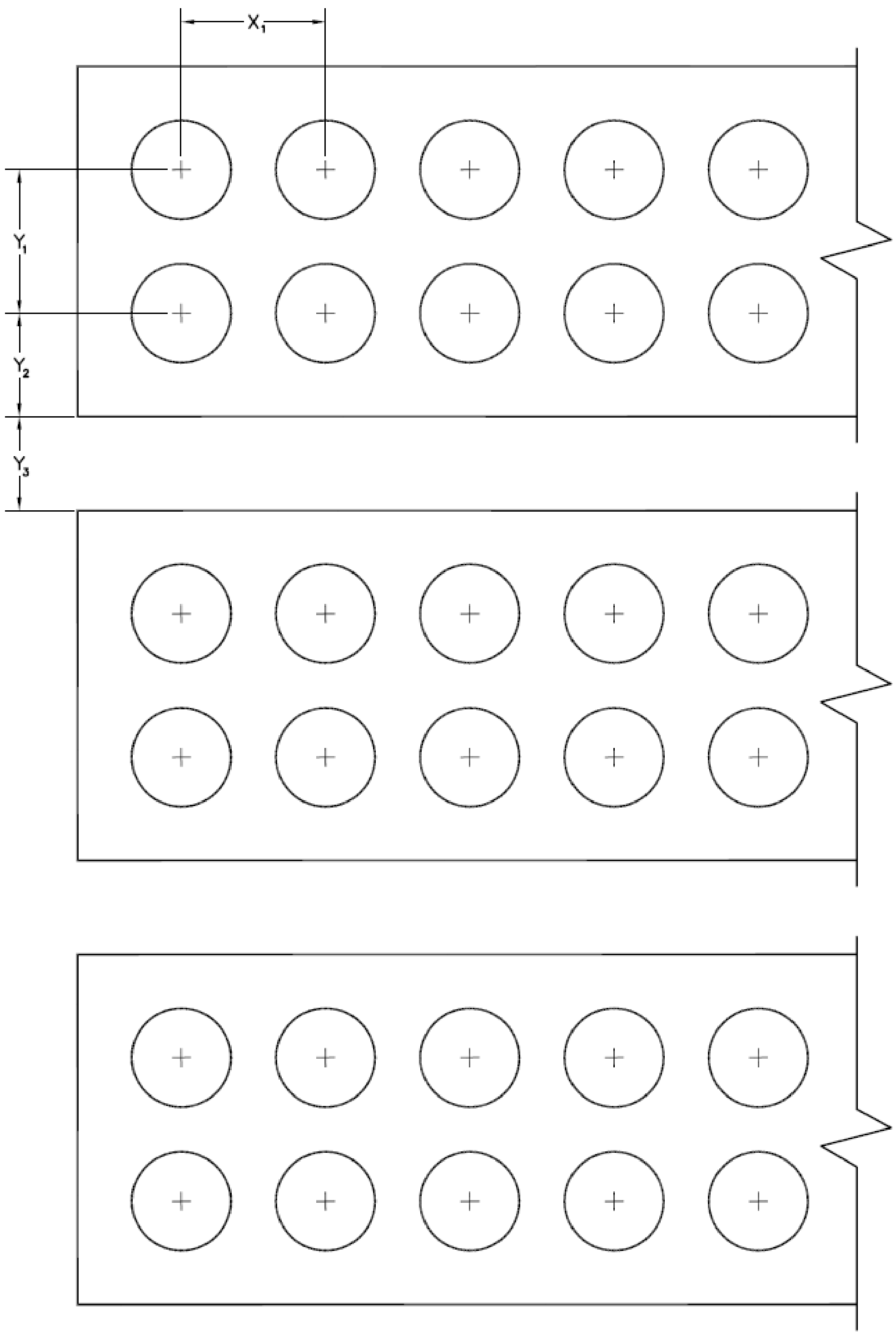


FIGURE 1.4.1: 2xN HI-STORM FW ARRAYS

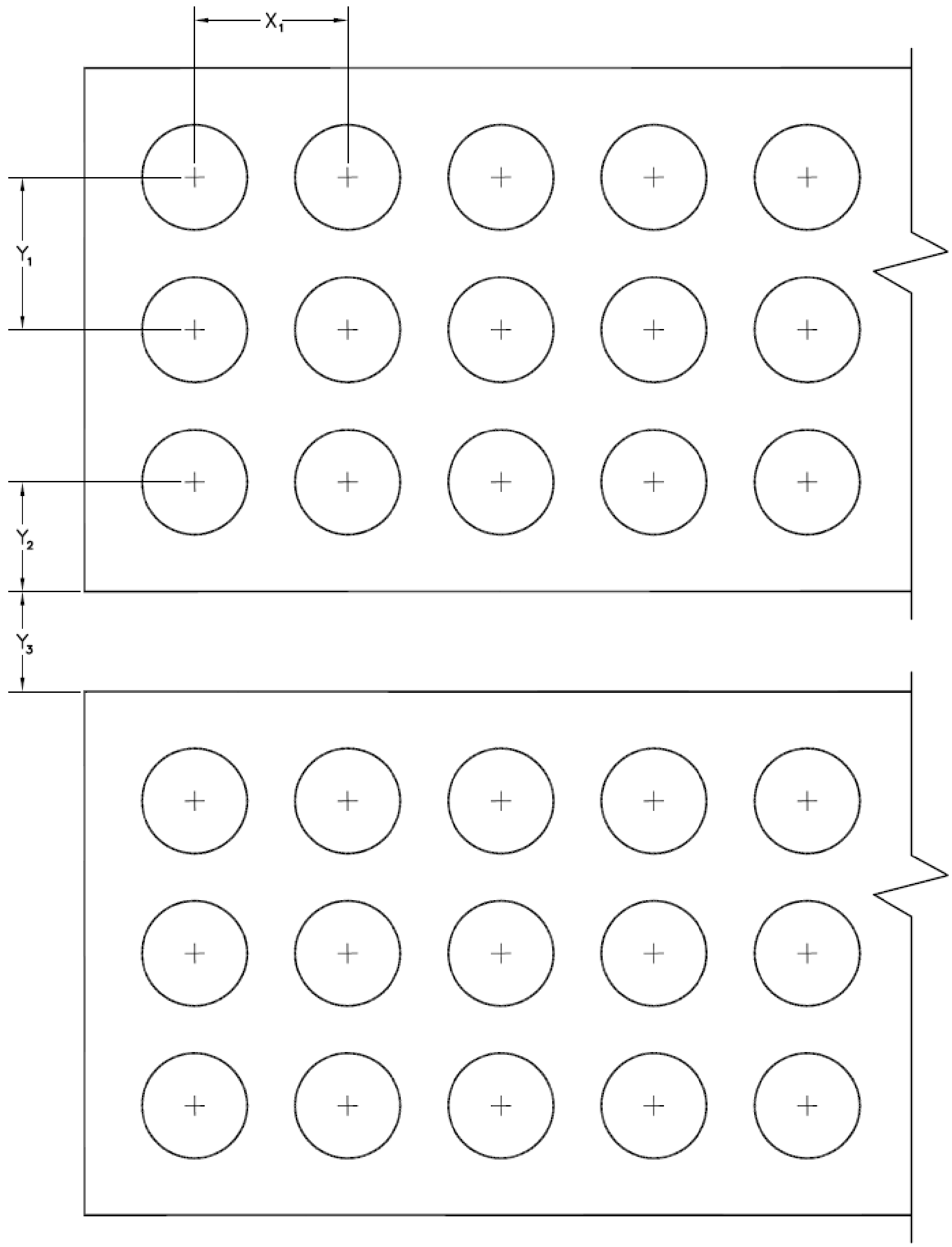


FIGURE 1.4.2: 3xN HI-STORM FW ARRAYS

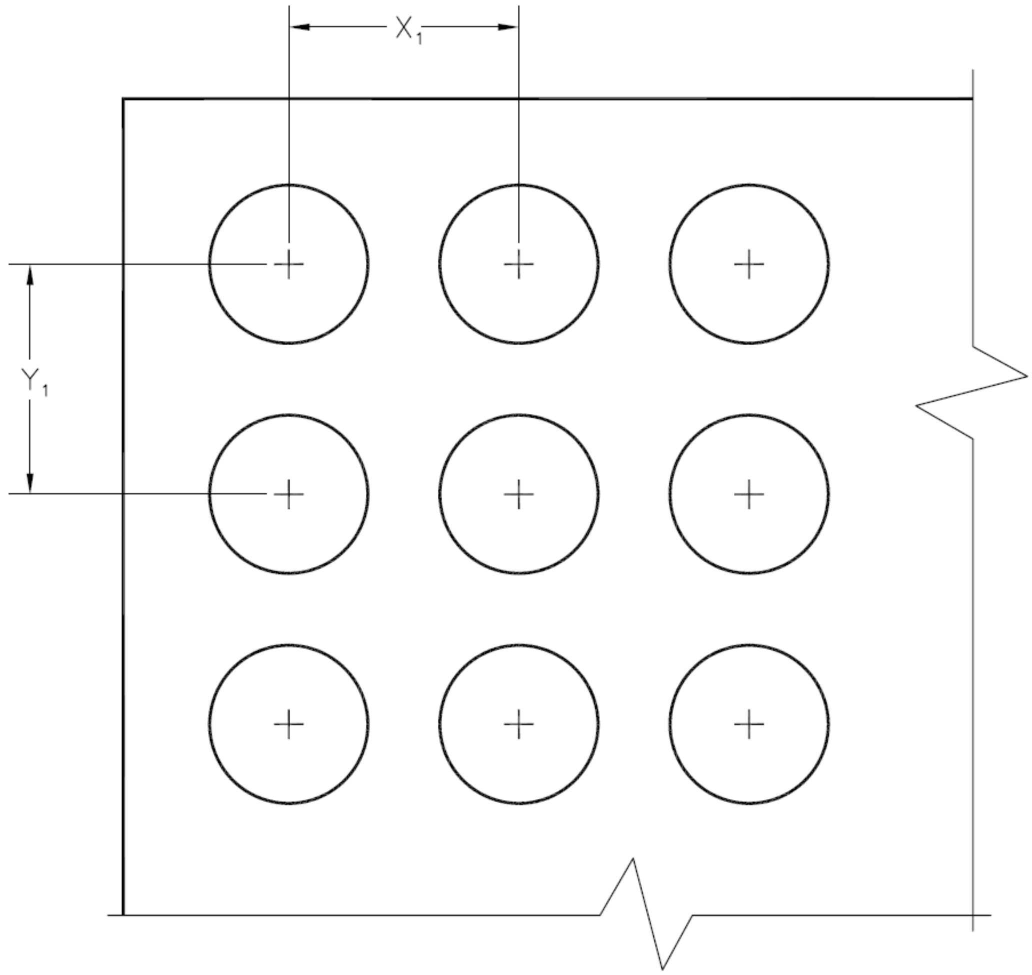


FIGURE 1.4.3: RECTANGULAR HI-STORM FW ARRAY

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	7
6508	HI-STORM LID ASSEMBLY	5
6514	HI-TRAC VW – MPC-37	6
6799	HI-TRAC VW – MPC-89	7
6505	MPC-37 ENCLOSURE VESSEL	7
6506	MPC-37 FUEL BASKET	9
6512	MPC-89 ENCLOSURE VESSEL	8
6507	MPC-89 FUEL BASKET	8

[Withheld in Accordance with 10 CFR 2.390]

1.6 REFERENCES

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[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.

[1.2.6] Metamic-HT Qualification Sourcebook”, Holtec Report No. HI-2084122, (2010) (Holtec Proprietary).

[1.2.7] Metamic-HT Manufacturing Manual”, Nanotec Metals Division, Holtec International, (2009) (Holtec Proprietary).

[1.2.8] Metamic-HT Purchasing Specification”, Holtec Document ID PS-11 (Holtec Proprietary).

[1.2.9] Sampling Procedures and Tables for Inspection by Attributes”, Military Standard MIL-STD-105E, (10/5/1989).

[1.2.10] USNRC Docket No. 72-1004 SER on NUHOMS 61BT (2002).

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

[1.2.12] Dynamic Mechanical Response and Microstructural Evolution of High Strength Aluminum-Scandium (Al-Sc) Alloy, by W.S. Lee and T.H. Chen, Materials Transactions, Vol. 47, No. 2(2006), pp 355-363, Japan Institute for metals.

[1.2.13] Turner, S.E., “Reactivity Effects of Streaming Between Discrete Boron Carbide Particles in Neutron Absorber Panels for Storage or Transport of Spent Nuclear Fuel,” Nuclear Science and Engineering, Vol. 151, Nov. 2005, pp. 344-347.

[1.2.14] Natrella, M.G., “Experimental Statistics”, National Bureau of Standards Handbook 91, National Bureau of Standards, Washington, DC, 1963.

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^{o}F. The lower bound service temperature of the MPC is -40^{o}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^{o}F is linearly extrapolated from the 70^{o}F value using the difference from 70^{o}F to 100^{o}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. (^oF)	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. (^oF)	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. (^oF)	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. (^{o}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./^{o}F x 10^{-6}.

TABLE 1.A.5 THERMAL CONDUCTIVITY vs. TEMPERATURE OF ALLOY-X MATERIALS			
Temp. (^{o}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-^{o}F.

Appendix 1.B (intentionally deleted)

Appendix 1.C (intentionally deleted)