

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

by

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## DOCUMENT ISSUANCE AND REVISION STATUS

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4.	4	IR	4/18/19	800055	JD	5/2/19	332505	N/A	N/A	N/A
5.	5	RST	4/18/19	639459	RT	5/2/19	722464	N/A	N/A	N/A
6.	6	TH	4/18/19	734553	N/A	N/A	N/A	N/A	N/A	N/A
7.	7	N/A	N/A	N/A	JD	5/2/19	269258	N/A	N/A	N/A
8.	8	RM	4/18/19	620165	JD	5/2/19	158312	N/A	N/A	N/A
9.	9	RM	4/18/19	539350	RM	5/3/19	941433	N/A	N/A	N/A
10.	10	RST	4/18/19	558196	N/A	N/A	N/A	N/A	N/A	N/A
11.	11	IR	4/18/19	692352	AM	5/2/19	448101	N/A	N/A	N/A
12.	12	CO	4/18/19	732243	JD	5/2/19	591968	N/A	N/A	N/A
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2.	2	RN	3/31/16	199310	KM	11/18/16	556375	JT	4/27/18	372130
3.	3	CB	3/31/16	775623	AB	11/18/16	146179	CB	4/27/18	553397
4.	4	AM	3/31/16	826335	AM	11/18/16	398445	MH	4/27/18	101980
5.	5	BK	3/31/16	54720	BK	11/18/16	670240	N/A	N/A	N/A
6.	6	VM	3/31/16	218915	VM	11/18/16	734131	JT	4/27/18	215805
7.	7	RN	3/31/16	87952	N/A	N/A	N/A	N/A	N/A	N/A
8.	8	JG	3/31/16	157492	RM	11/18/16	435637	N/A	N/A	N/A
9.	9	JG	3/31/16	326160	KM	11/18/16	644870	N/A	N/A	N/A
10.	10	BK	3/31/16	23104	N/A	N/A	N/A	N/A	N/A	N/A
11.	11	AM	3/31/16	833282	N/A	N/A	N/A	N/A	N/A	N/A
12.	12	RN	3/31/16	211239	N/A	N/A	N/A	JT	4/27/18	563527
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2.	2	IR	4/24/2012	133944	TH	7/26/2013	292559	RN	3/12/2014	10988
3.	3	CWB	4/15/2012	253466	CB	7/26/2013	305363	AB	3/12/2014	154830
4.	4	IR	4/24/2012	314945	IR	7/26/2013	734752	AM	3/12/2014	203620
5.	5	KB	4/15/2012	685233	PS	7/26/2013	655200	PS	3/12/2014	493600
6.	6	VG	4/15/2012	547279	TH	7/27/2013	874739	RN	3/12/2014	48837
7.	7	VG	4/15/2012	574727	BK	7/27/2013	199808	RN	3/12/2014	398001
8.	8	VG	4/15/2012	227587	VG	7/27/2013	517656	AM	3/12/2014	128836
9.	9	VG	4/15/2012	951243	VG	7/27/2013	166560	RN	3/12/2014	276825
10.	10	VG	4/15/2012	514322	BK	7/27/2013	947187	AB	3/12/2014	720054
11.	11	VG	4/15/2012	577089	IR	7/27/2013	730705	AM	3/12/2014	123376
12.	12	VG	4/24/2012	514303	IR	7/27/2013	419931	AM	3/12/2014	458752
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2.	2	TSM	8/8/08	352556	TSM	1/18/10	430085	TSM	2/13/10	657114
3.	3	CWB	8/8/08	223114	CWB	1/18/10	606298	AB	2/13/10	204726
4.	4	IR	8/8/08	808688	IR	1/18/10	792928	IR	2/13/10	745213
5.	5	SPA	8/8/08	410785	SPA	1/18/10	798514	SPA	2/13/10	446703
6.	6	SPA	8/7/08	372258	SPA	1/18/10	809344	SPA	2/13/10	597586
7.	7	KC	8/7/08	951285	TSM	1/18/10	156383	TSM	2/13/10	517668
8.	8	TSM	8/8/08	85811	TSM	1/18/10	604039	TSM	2/13/10	155974
9.	9	TSM	8/6/08	782515	TSM	1/18/10	711656	TSM	2/13/10	132301
10.	10	SPA	8/8/08	218691	TSM	1/18/10	547146	TSM	2/13/10	218054
11.	11	ER	7/10/08	955026	TSM	1/18/10	205144	TSM	2/13/10	518926
12.	12	TSM	8/6/08	841345	TSM	1/18/10	262082	TSM	2/13/10	399215
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2.	2	LEH	4/10/06	352962	TSM	6/19/07	815387	TSM	2/7/08	808432
3.	3	CWB	4/10/06	926484	CWB	6/19/07	405257	CWB	2/7/08	370539
4.	4	DMM	4/10/06	53264	DMM	6/20/07	839967	DMM	2/7/08	846793
5.	5	ERD	4/10/06	91451	SPA	6/20/07	654660	SPA	2/7/08	906070
6.	6	SPA	4/10/06	142478	SPA	6/20/07	658909	SPA	2/7/08	794294
7.	7	KC	4/10/06	930273	KC	6/20/07	69449	KC	2/7/08	334538
8.	8	JG	4/10/06	728141	JDG	6/19/07	540392	JDG	2/7/08	121986
9.	9	LEH	4/10/06	590912	KC	6/20/07	965280	TSM	2/7/08	835761
10.	10	ERD	4/10/06	604641	SPA	6/20/07	565174	SPA	2/7/08	928175
11.	11	DMM	4/10/06	870251	ER	6/20/07	582705	ER	2/7/08	31423
12.	12	SPA	4/10/06	911918	ER	6/20/07	438028	TSM	2/7/08	101336
13.	13	SPA	4/10/06	515528	TSM	6/19/07	837745	TSM	2/7/08	367403

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2.	2	BGU	9/17/02	181394	BGU	1/12/04	635397	SPA	5/26/05	306544
3.	3	CWB	9/13/02	553330	CWB	2/17/04	322728	CWB	5/26/05	118516
4.	4	IR	8/22/02	829317	IR	1/30/04	42783	ER	5/26/05	619640
5.	5	ERD	9/13/02	136063	ERD	2/17/04	27030	ERD	5/26/05	531811
6.	6	SPA	8/22/02	751052	SPA	3/3/04	235143	SPA	5/26/05	804742
7.	7	KC	8/22/02	181406	KC	3/3/04	538268	KC	5/26/05	618709
8.	8	JG	8/22/02	786745	JG	3/5/04	705490	JG	5/26/05	672421
9.	9	BGU	8/22/02	118193	BGU	1/12/04	305051	LEH	5/26/05	896141
10.	10	JG	8/22/02	504833	JG	2/17/04	384984	JG	5/26/05	511173
11.	11	ER	8/22/02	687910	ER	1/30/04	189574	ER	5/26/05	268920
12.	12	BGU	8/22/02	415966	BGU	2/2/04	988602	SPA	5/26/05	822043
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**DOCUMENT CATEGORIZATION**

In accordance with the Holtec Quality Assurance Manual and associated Holtec Quality Procedures (HQPs), this document is categorized as a:

- Calculation Package<sup>3</sup> (Per HQP 3.2)       Technical Report (Per HQP 3.2)(Such as a Licensing Report)
- Design Criterion Document (Per HQP 3.4)       Design Specification (Per HQP 3.4)
- Other (Specify):

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3. Revisions to this document may be made by adding supplements to the document and replacing the "Table of Contents", this page and the "Revision Log".

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

HI-STORM 100 Cask Storage System Final Safety Analysis Report (FSAR) Revision #	NRC Certificate of Compliance (CoC) 1014 Amendment No.
0	0
1	1
2*	1
3	2
4*	2
5	3
6	4
7	5
8	6
9	7
10*	7
11	8
11.1	8 Rev. 1
12	9
13	9 Rev. 1
14	10
15*	10
16	11
17	12
18	13

\* - These Revisions incorporated changes made via ECO/72.48 process only.

<b>FSAR SECTION REVISION STATUS, LIST OF AFFECTED SECTIONS AND REVISION SUMMARY</b>		
<b>FSAR Report No.:</b> HI-2002444		<b>FSAR Revision Number:</b> 18
<b>FSAR Title:</b>	Final Safety Analysis Report on the HI-STORM 100 System	
<p>This FSAR is submitted to the USNRC in support of Holtec International's application to secure a CoC under 10CFR Part 72.</p> <p>FSAR review and verification are controlled at the chapter level and changes are annotated at the chapter level.</p> <p>A section in a chapter is identified by two numerals separated by a decimal. Each section begins on a fresh page. Unless indicated as a "complete revision" in the summary description of change below, if any change in the content is made, then the change is indicated by a "bar" in the right page margin and the revision number of the entire chapter including applicable figures (annotated in the footer) is changed.</p> <p>A summary description of change is provided below for each FSAR chapter. Minor editorial changes to this FSAR may not be summarized in the description of change.</p>		
<b>Chapter 1</b>		
<b>Affected Section or Table No.</b>	<b>Current Revision No.</b>	<b>Summary Description of Change</b>
Subsection 1.0.1 Table 1.0.3	18	Updated per ECO. Updated per ECO.



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PERIOD COVERED – May 3, 2019 through May 14, 2019

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.

Affected Item	ECO Number	72.48 Evaluation or Screening Number
MPC-68/68F/68FF/68M Basket		
MPC-24/24E/24EF Basket		
MPC-32 Basket		
MPC Enclosure Vessel		
HI-STORM Overpack		
HI-STORM 100U VVM		
HI-TRAC 100 and 100D Transfer Cask		
HI-TRAC 125 and 125D Transfer Cask		
General FSAR Changes	ECO-5014-293R0	N/A

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## GLOSSARY

**ALARA** is an acronym for As Low As Reasonably Achievable.

**All Metamic Fuel Baskets or AMFBs** refers to those baskets in which both the structural and neutron absorption functions are rendered by panels made from Metamic-HT. There is no separate neutron absorber or stainless steel in AMFBs.

**Alloy X Fuel Baskets** refers to those baskets that employ the Metamic neutron absorber panels affixed to the Alloy X grid- plate in an irremovable configuration.

**Ancillary or Ancillary Equipment** is the generic name of a device used to carry out Short -term Operations.

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

**Boral™** means Boral manufactured by AAR Advanced Structures.

**BWR** is an acronym for boiling water reactor.

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

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

**Confinement Boundary** means the outline formed by the all-welded cylindrical enclosure of the Multi-Purpose Canister (MPC) shell, MPC baseplate, MPC lid, MPC port cover plates, and the MPC closure ring which provides redundant sealing.

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

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

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

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

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

**DBE** means Design Basis Earthquake.

**DCSS** is an acronym for Dry Cask Storage System.

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

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

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

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

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

**Design Report** is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. The FSAR serves as the Design Report for the HI-STORM 100 System.

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

**Enclosure Vessel (or MPC Enclosure Vessel)** means the pressure vessel defined by the

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

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

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

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

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

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

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

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

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

**HI-TRAC** is an acronym for **Holtec International Transfer Cask**.

**HI-TRAC transfer cask or HI-TRAC** means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields and protects the loaded MPC. In this FSAR there are two HI-TRAC transfer casks, the 125-ton dual-purpose lid design (HI-TRAC 125D) and the 100-ton dual purpose lid design (HI-TRAC 100D). The 125-ton HI-TRAC is provided for use at sites with a maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to both HI-TRAC transfer cask designs, unless the discussion requires distinguishing among the designs..

**HI-STORM overpack** or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel for long term storage. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. The term “overpack” as used in this FSAR refers to the HI-STORM 100S and HI-STORM 100S Version B.



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

**Holtite<sup>TM</sup>** is the trade name for all present and future neutron shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A and Holtite B are candidate neutron shield materials qualified under the Holtite R&D program.

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

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

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

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

**Lifting Device** means a component of a lifting and handling system that is not classified as a *special lifting device*. ANSI N14.6 is not applicable to a lifting *device*. These include non-active structural components (components that bear the primary load but are not a constituent of a moving part, e.g., gear train, hydraulic cylinder) of the system. Examples of lifting devices used with Holtec's systems include: A Low Profile Transporter, a VCT overhead beam or the main girder of a gantry crane.

**Licensing Basis Heat Load (LBHL)** of a canister/cask is the aggregate of the permissible heat load for all fuel storage locations that constitute the fuel basket.

**Location Specific Heat Load** is the permissible heat load associated with a storage location; also known as the Specific heat load for that location.

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

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

service condition.

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

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

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

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

**MGDS** is an acronym for Mined Geological Disposal System.

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

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

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

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

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

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

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

**Non-Fuel Hardware** is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug

Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, instrument tube tie-rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

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

**Plain Concrete** is concrete that is unreinforced.

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

**PWR** is an acronym for pressurized water reactor.

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

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

**SAR** is an acronym for Safety Analysis Report.

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

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

**Single Failure Proof** in order for a lifting device or special lifting device to be considered single failure proof, the design must follow the guidance in NUREG-0612, which requires that a single failure proof device have twice the normal safety margin. This designation can be achieved by either providing redundant devices (load paths) or providing twice the design factor required by the applicable code.

**SNF** is an acronym for spent nuclear fuel.

**Special Lifting Device** refers to a components to which ANSI N14.6 applies. As stated in ANSI N14.6 (both 1978 and 1993 versions), “This standard shall apply to *special lifting devices* that transmit the load from lifting attachments, which are structural parts of a container to the hook(s) of an overhead hoisting system.”

Examples of special lifting devices used with Holtec's systems include MPC lift cleats, lift brackets, and lift yokes.

**SSC** is an acronym for Structures, Systems and Components.

**Stainless Steel Fuel Baskets or SSFBs** refer to fuel baskets that employ stainless steel for structural function, and neutron absorption capability is provided by a discrete panel of Metamic or an equivalent neutron absorber. MPC-68 contains an SSFB. ( See also "Alloy X Fuel Basket ")

**STP** is Standard Temperature and Pressure conditions.

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

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

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

**Undamaged Fuel Assembly** is: a) a fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial of 3.3 wt% U-235, without known or suspected grossly breached spent fuel rods, and which can be handled by normal means. An undamaged fuel assembly may be a repaired/reconstituted fuel assembly.

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

**ZPA** is an acronym for zero period acceleration.

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

## CHAPTER 1<sup>†</sup>: GENERAL DESCRIPTION

### 1.0 GENERAL INFORMATION

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

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

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.1 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable FSAR section that addresses each topic.

The HI-STORM 100 FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all guidance. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.2 herein. Table 1.0.2 also contains the justification for the alternative method for compliance adopted in this FSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

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

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

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

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

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

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

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

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

### 1.0.1 Design Compatibility of Licensed HI-STORM 100 System Components

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

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

HI-TRAC Design Compatibility			
HI-TRAC Model	Program Number	Serial Number	Applicable CoC Amendments <sup>Note 1</sup>
125	1025	001	0,1,2,3,4,5,6,7,8,9,10,11,12,13
125	1025	002	0,1,2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	003	1,2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	004	1,2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	005	2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	007	1,2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	008	1,2,3,4,5,6,7,8,9,10,11,12,13
125D	1025	009	2,3,4,5,6,7,8,9,10,11, 12,13
125D	1025	010	3,4,5,6,7,8,9,10,11,12,13
125D	1025	011	5,6,7,8,9,10,11,12,13
125D	1025	012	5,6,7,8,9,10,11,12,13
125D	1025	013	5,6,7,8,9,10,11,12,13
125D	1025	014	3,4,5,6,7,8,9,10,11,12,13
125D	1025	015	3,4,5,6,7,8,9,10,11,12,13
125D	1025	016	5,6,7,8,9,10,11,12,13
125D	1025	017	7,8,9,10,11,12,13
125D	1025	019	7,8,9,10,11,12,13
100	1026	001	0,1,2,3,4,5,6,7,8,9,10,11,12,13
100D	1026	003	2,3,4,5,6,7,8,9,10,11,12,13
100D	1026	004	2,3,4,5,6,7,8,9,10,11,12,13
100D	1026	005	2,3,4,5,6,7,8,9,10,11,12,13
100D	1026	006	2,3,4,5,6,7,8,9,10,11,12,13
100D Version IP1	1026	008	4
100D	1026	009	6,7,8,9,10,11,12,13

Note 1: The changes approved in Revision 1 to CoC Amendments 8 and 9 do not have any impact on the HI-TRAC designs and compatibility. Therefore, any HI-TRAC identified as compatible with Amendments 8 or 9 is by extension compatible with Amendment 8 Rev. 1 or Amendment 9 Rev. 1 respectively. Amendment 8 Rev 1 is incorporated into HI-STORM 100 FSAR Rev. 11.1.



## 1.0.2 Summary of Changes to the HI-STORM 100 CoC

The HI-STORM 100 System, USNRC Docket No. 72-1014, has undergone several changes throughout the CoC Amendment revision process. A summary of the changes made in each amendment revision can be found in Table 1.0.3. Strictly editorial changes that do not alter the content of the CoC have been excluded below. Full details of all changes may be found in each respective CoC Amendment and Safety Evaluation Report (SER).

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
<b>1. General Description</b>			
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
<b>2. Principal Design Criteria</b>			
2.1 Spent Fuel To Be Stored	2.III.2.a Spent Fuel Specifications	10CFR72.2(a)(1) 10CFR72.236(a)	2.1
2.2 Design Criteria for Environmental Conditions and Natural Phenomena	2.III.2.b External Conditions, 2.III.3.b Structural, 2.III.3.c Thermal	10CFR72.122(b)	2.2
		10CFR72.122(c)	2.2.3.3, 2.2.3.10
		10CFR72.122(b)(1)	2.2
		10CFR72.122(b)(2)	2.2.3.11
		10CFR72.122(h)(1)	2.0
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.5

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b) (2)	2.2.3.6
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b) (2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b) (2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a) (4)	2.3.2.2, 4.0
NA	2.III.3f Operating Procedures	10CFR72.24(f) 10CFR72.128(a) (5) 10CFR72.236(h) 10CFR72.24(1)(2) 10CFR72.236(1) 10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0 8.0 1.2.1, 1.2.2 2.3.2.1 10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a) (1)	9.0
2.3 Safety Protection Systems	--	--	2.3

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a) (2)	2.3.5.2
		10CFR72.128(a) (3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
10CFR72.236(e)	2.3.2.1		
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h) (4) 10CFR72.122(i) 10CFR72.128(a) (1)	2.3.5
2.3.4 Nuclear Criticality Safety		2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236(h)	2.4
	14.III.1 Design	10CFR72.130	2.4
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4

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

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
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
<b>4. Thermal Evaluation</b>			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a) (4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.122(h) (1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
<b>5. Shielding Evaluation</b>			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configurations	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a) (2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
<b>6. Criticality Evaluation</b>			
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 Computational Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
6.3.2 Cask Regional  Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2, 6.3.3
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6 Supplemental Data	6.V.5 Supplemental Info.	--	Appendices 6.B,6.C, and 6.D
<b>7. Confinement</b>			
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



<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.236(1)	7.1
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7.1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h) (4) 10CFR72.128(a) (l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose ISG-18	10CFR72.104(a)	7.1
7.2.2 Pressurization of Confinement Vessel	--	--	7.1
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(1)	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.	--	--
<b>8. Operating Procedures</b>			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5

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

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	9.1
	9.III.1.d Test Program	10CFR72.162	9.1
	9.III.1.e Appropriate Tests	10CFR72.236(1)	9.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1 <sup>(2)</sup>
9.2	Maintenance Program	9.III.1.b Maintenance	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a) (1)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b) (8)	9.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	<sup>(3)</sup>
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	<sup>(4)</sup>
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	9.1.7, 9.1.1.(12)
	9.III.3 Cask Identification		
<b>10. Radiation Protection</b>			
10.1	Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA 10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	10.1
10.2	Radiation Protection Design Features	10.V.1.b Design Features 10CFR72.126(a)(6)	10.2

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
<b>11. Accident Analyses</b>			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off- Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
<b>12. Operating Controls and Limits</b>			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart E 10CFR72 Subpart F	12.0
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 12.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 12.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SSC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	8.0
NA	12.III.2 Confinement Effectiveness	10CFR72.236(j)	9.0
NA	12.III.2 Evaluation for Confinement	10CFR72.236(l)	7.1, 7.2, 9.0
<b>13. Quality Assurance</b>			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) 10CFR72.140(d)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Notes:

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

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



NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."</p>	<p><u>Clarification:</u> As stated in NUREG-1536, 3.V.(d), page 3-11, "Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted outside the spent fuel building, or the maximum deceleration that the cask could experience in a drop." The maximum deceleration for a corner drop is specified as 45g's for the HI-STORM overpack. No carry height limit is specified for the corner drop.</p>	<p>In Chapter 3, the MPC and HI-STORM overpack are evaluated under a 45g radial loading. A 45g axial loading on the MPC is bounded by the analysis presented in the HI-STAR FSAR, Docket 72-1008, under a 60g loading, and is not repeated in this FSAR. In Chapter 3, the HI-STORM overpack is evaluated under a 45g axial loading. Therefore, the HI-STORM overpack and MPC are qualified for a 45g loading as a result of a corner drop. Depending on the design of the lifting device, the type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses may be required to demonstrate that the deceleration limit of 45g's is not exceeded.</p>
<p>3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced..."</p> <p>3.V.2.b.i.(2)(b), Page 3-20, Para. 1, "The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359".</p> <p>3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and construction of reinforced concrete structures important to safety but not within the scope of</p>	<p><u>Exception:</u> The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used as guidance for the material selection and specification, and placement of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack, including clarifications on implementation of this code. ACI 318 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.</p>	<p>Concrete is provided in the HI-STORM overpack primarily for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose.</p> <p>To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. The temperature limits for normal conditions are per Paragraph A.4.3 of Appendix A to ACI 349 and temperature limits for</p>

<b>NUREG-1536 Guidance</b>	<b>Alternate Method to Meet NUREG-1536 Intent</b>	<b>Justification</b>
<p>ACI 359 should comply with the requirements of ACI 349".</p>		<p>off-normal and accident conditions are per Paragraph A.4.2 of Appendix A to ACI 349.</p> <p>Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.</p>
<p>3.V.3.b.i.(2), Page 3-29, Para. 1, "The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with Section III of the ASME B&amp;PV Code."</p>	<p><u>Clarification:</u> The HI-STORM overpack steel structure is designed in accordance with the ASME B&amp;PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.</p>	<p>The overpack structure is a steel weldment consisting of "plate and shell type" members. As such, it is appropriate to design the structure to Section III, Class 3 of Subsection NF. The very same approach has been used in the structural evaluation of the "intermediate shells" in the HI-STAR 100 overpack (Docket Number 72-1008) previously reviewed and approved by the USNRC.</p>
<p>4.IV.5, Page 4-2 "for each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage."</p> <p>4.IV.1, Page 4-3, Para. 1 "the staff should verify that cladding temperatures for each fuel type proposed for storage will be below the expected damage thresholds for normal conditions of storage."</p> <p>4.IV.1, Page 4-3, Para. 2 "fuel cladding limits for each fuel type should be defined in the SAR with thermal restrictions in the DCSS technical specifications."</p>	<p><u>Clarification:</u> As described in Section 4.3, all fuel array types authorized for storage are assigned a single peak fuel cladding temperature limit.</p>	<p>As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak normal fuel cladding temperature limit of 400°C.</p>

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
4.V.1, Page 4-3, Para. 4 "the applicant should verify that these cladding temperature limits are appropriate for all fuel types proposed for storage, and that the fuel cladding temperatures will remain below the limit for facility operations (e.g., fuel transfer) and the worst-case credible accident."		
4.V.4.a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."	<u>Clarification:</u> As discussed in Section 4.4, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based model.	The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature difference conservatively maximizes the basket wall temperature.
4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."	<u>Exception:</u> All calculations of fuel assembly effective thermal conductivities use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.	The calculated effective thermal conductivities based on nominal design fuel dimensions have been compared with available literature values [1.0.6] and are demonstrated to be conservative.
4.V.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented."	<u>Clarification:</u> No additional heat balance is performed or provided.	The FLUENT computational fluid dynamics program used to perform evaluations of the HI-STORM Overpack and HI-TRAC transfer cask, which uses a discretized numerical solution algorithm, enforces an energy balance on all discretized volumes throughout the computational domain. This solution method,

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
		therefore, ensures a heat balance at the surface of the cask.
4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."	<u>Exception:</u> No input or output file listings are provided in Chapter 4.	A complete set of computer program input and output files would be in excess of three hundred pages. All computer files are considered proprietary because they provide details of the design and analysis methods. In order to minimize the amount of proprietary information in the FSAR, computer files are provided in the proprietary calculation packages.
4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures.	<u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the fuel assemblies is calculated using maximum weights and minimum densities.	Calculating the volume occupied by the fuel assemblies using maximum weights and minimum densities conservatively overpredicts the volume occupied by the fuel and correspondingly underpredicts the remaining free volume.

NUREG-1536 Guidance	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>7.V.4 “Confinement Analysis. Review the applicant’s confinement analysis and the resulting annual dose at the controlled area boundary.”</p>	<p><u>Exception:</u> No confinement analysis is performed and no effluent dose at the controlled area boundary is calculated.</p>	<p>The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g., non-destructive examinations in accordance with Chapter 9).</p> <p>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.</p> <p>Pursuant to ISG-18, the Holtec MPC is constructed in a manner that supports leakage from the confinement boundary being non-credible. Therefore, no confinement analysis is required.</p>
<p>9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."</p>	<p><u>Clarification:</u> Section 9.1.1.1 and the Design Drawings specify that the ASME Code, Section III, Subsection NB, Article NB-5332 will be used for the acceptance criteria for the volumetric examination of the MPC lid-to-shell weld.</p>	<p>In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria as defined by either the ASME code, or an alternative approach..." The ASME Code, Section III, Subsection NB, Paragraph NB-5332 is appropriate acceptance criteria for pre-service examination.</p>

<b>NUREG-1536 Guidance</b>	<b>Alternate Method to Meet NUREG-1536 Intent</b>	<b>Justification</b>
9.V.1.d, Para. 1 "Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron absorbing material."	<u>Exception:</u> Subsection 9.1.5 describes the control of special processes, such as neutron shield material installation, to be performed in lieu of scanning or probing with neutron sources.	<p>The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation.</p> <p>The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures.</p> <p>The composition of the Holtite-A is confirmed by inspection and tests prior to first use.</p> <p>Following the first loading for the HI-TRAC transfer cask and each HI-STORM overpack, a shield effectiveness test is performed in accordance with written approved procedures, as specified in Section 9.1.</p>
13.III, "the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."	<u>Exception:</u> Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.	The NRC has approved Revision 13 of the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec will apply this QA program to all important-to-safety dry storage cask activities. Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation.

Amendment Revision	Summary of Changes
0	Initial Issue
1	<ul style="list-style-type: none"> <li>• Addition of four new MPCs. The new MPCs include the MPC-24E, MPC-24EF, MPC-32, and MPC-68FF</li> <li>• Addition of the following new Damaged Fuel Containers (DFC) <ul style="list-style-type: none"> <li>○ TN/D-1 DFC</li> <li>○ Holtec Generic PWR DFC</li> <li>○ Holtec Generic BWR DFC</li> </ul> </li> <li>• Addition of the HI-STORM 100S, 100A, and 100SA overpacks</li> <li>• Allow for Overpack to be anchored to the ISFSI</li> <li>• Allow for storage of high-burnup fuel</li> <li>• Revision of thermal analysis to include natural convection</li> <li>• Allow helium backfill requirement to use helium density measurement</li> <li>• Allow for the use of a helium drying system (FHD)</li> <li>• Addition of soluble boron requirement for higher enriched fuels</li> <li>• Update surveillance criteria in LCO 3.1.2 (SFSC Heat Removal System)</li> <li>• Update HI-TRAC and Overpack surface dose rate acceptance criteria (LCO 3.2.1 &amp; 3.2.3)</li> <li>• Updated Cask Transport Evaluation Program (Appendix A, Section 5.5)</li> <li>• Editorial change to relocate special requirements for first systems and pre-operational testing and training requirements to the main body of CoC</li> <li>• Allow for storage of these non-fuel hardware: burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), axial power shaping rods (APSRs), control element assemblies (CEAs), wet annular burnable absorbers (WABAs), rod cluster control assemblies (RCCAs), water displacement guide tube plugs and orifice rod assemblies</li> <li>• Update Fuel Specifications and Loading Conditions to allow for regionalized fuel loading (Appendix B, Section 2.1)</li> </ul>
2	<ul style="list-style-type: none"> <li>• Addition of METAMIC as an alternate neutron poison material</li> <li>• Addition of new MPC-32F as new MPC type</li> <li>• Allow storage of damaged fuel in MPC-32. Allow storage of damaged fuel and damaged fuel debris in MPC-32F <ul style="list-style-type: none"> <li>○ Addition of soluble boron requirement for MPC-32 and MPC-32F</li> </ul> </li> <li>• Revision of MPC cavity drying limits</li> <li>• Removal of Aluminum Heat Conduction Elements (AHCE) from system</li> </ul>

Amendment Revision	Summary of Changes
2 (cont'd)	<ul style="list-style-type: none"> <li>• Allow for helium leak test for either (1) lid-to-shell weld or (2) vent and drain port cover plate welds to satisfy ISG-18 requirements</li> <li>• Update CoC to account for changes in ISG-11 Revision 3</li> <li>• Addition of cavity bulk helium temperature requirement during reflooding operations of the MPC</li> <li>• Addition of Tech Spec Program 5.7 (Radiation Protection) &amp; modified associated LCO</li> <li>• Allow for storage of vibration suppressor inserts (non-fuel hardware)</li> <li>• Increase maximum authorized initial enrichment for PWR damaged fuel and fuel debris to 5.0 wt. %</li> <li>• Revise burn-up as a function of cooling time and fuel array/class</li> <li>• Revise completion times for required action to reflect blocked duct accident analysis and surveillance requirement acceptance criteria for temperature measurement</li> <li>• Revise Appendix B tables for new fuel assembly burnup limits</li> <li>• Revise maximum allowable uranium masses for certain fuel assemblies</li> <li>• Revise maximum allowable burn-up for non-fuel hardware inserts</li> <li>• Updated ASME code edition of record. Added new and revised ASME code alternatives. Incorporated new language related to Code alternatives</li> <li>• Increase off-normal design pressure from 100psig to 110psig</li> <li>• Increase normal temperature limit for overpack lid top plate from 350°F to 450°F</li> <li>• Modify CoC Condition 11 language to address component certification</li> <li>• Allow for maximum boron carbide content in METAMIC to 33.0 wt. %</li> <li>• Incorporated Neutron Absorber Tests (FSAR section 9.1.5.3) to the CoC by reference</li> <li>• Modification of design temperatures of MPC shell, overpack concrete, and Holtite neutron shield material</li> <li>• Addition of Supplemental Cooling system (SCS) as well as Condition 11 in the CoC for the SCS</li> <li>• Addition of FHD and SCS malfunctions as new off normal events</li> <li>• Addition of SCS failure as new accident event</li> <li>• Addition of degraded cask/pad interface friction requirement</li> </ul>
3	<ul style="list-style-type: none"> <li>• Remove requirement for MPC cavity cooling prior to reflow with water <ul style="list-style-type: none"> <li>○ Replace helium exit temperature requirement with MPC cavity pressure</li> </ul> </li> </ul>



Amendment Revision	Summary of Changes
3 (cont'd)	<ul style="list-style-type: none"> <li>• Allow for linear interpolation between minimum soluble boron concentrations of 4.1 wt% and 5.0 wt% fuel enrichment in MPC-32/32F</li> <li>• Modify definition of: <ul style="list-style-type: none"> <li>○ Non-fuel hardware to include primary and secondary neutron sources</li> <li>○ Fuel debris to include containers or structures that provide support to loose fuel assembly parts</li> <li>○ Damaged fuel to a performance based definition</li> </ul> </li> <li>• Allow for storage of neutron source assemblies in MPC-24, 24E, 24EF, 32, 32F</li> <li>• Allow for storage of PWR fuel assemblies with annular fuel pellets in the top and bottom 12 inches of the active fuel length</li> <li>• Addition of water level requirement for cask unloading</li> </ul>
4	<ul style="list-style-type: none"> <li>• Addition of site-specific options to permit use of a modified HI-STORM 100S Version B, MPC-32/32F, and HI-TRAC 100D at Indian Point Unit 1 (IP1). Changes include: <ul style="list-style-type: none"> <li>○ Modification of soluble boron requirements for array/class 14x14E, IP1 fuel</li> <li>○ Modification of helium gas backfill requirements for array/class 14x14E IP1 fuel</li> <li>○ Addition of 5<sup>th</sup> damaged fuel container design (Holtec IP1 DFC)</li> <li>○ Addition of separate burnup, cooling time, and decay heat limits for array/class 14x14E IP1 fuel for loading in MPC-32 and -32F</li> <li>○ Addition of antimony-beryllium secondary sources as approved contents</li> <li>○ Allow for loading of all IP1 fuel assemblies in DFCs</li> <li>○ Reduction of maximum enrichment for array/class 14x14E IP1 fuel from 5.0 to 4.5 wt.% <sup>235</sup>U</li> </ul> </li> </ul>
5	<ul style="list-style-type: none"> <li>• Removal of requirement to perform subsequent thermal validations tests on thermal systems (a condition 9 requirement). Condition 9 now requires validation test for overpacks with heat loads equal to or greater than 20 kW</li> <li>• Increase in design basis maximum decay heat loads to: <ul style="list-style-type: none"> <li>○ 34 kW for uniform loading</li> <li>○ 36.9 kW for regionalized loading</li> <li>○ Update to LCO 3.1.1 and 3.1.2 to reflect new heat loads</li> <li>○ Addition of new decay heat regionalized scheme</li> </ul> </li> <li>• Increase in maximum fuel assembly weight for BWR fuel in MPC-68 from 700 to 730 lbs.</li> </ul>

Amendment Revision	Summary of Changes
5 (cont'd)	<ul style="list-style-type: none"> <li>• Changes to assembly characteristics of PWR 16x16 fuel assemblies to be qualified for storage. Characteristic changes include: <ul style="list-style-type: none"> <li>○ Maximum fuel assembly weight for assemblies that do not require upper and lower fuel spacers</li> <li>○ Fuel Rod Clad Inner Diameter</li> <li>○ Minimum guide/instrument tube thickness</li> <li>○ Minimum soluble boron concentration for all intact fuel assemblies</li> </ul> </li> <li>• Change in the fuel storage locations in MPC-32 for fuel with Axial Power Shaping Rod Assemblies (APSRs)</li> <li>• Change in fuel storage locations in MPC-24, -24E, and -32 for fuel with Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCAAs), and Control Element Assemblies (CEAs)</li> <li>• Elimination of restriction that fuel debris can only be loaded into the MPC-24EF, -32F, -68F, and 68FF</li> <li>• Requirement for MPC components exposed to spent fuel pool water, or the ambient environment, to be made out of stainless steel or, for MPC internals, neutron absorber, or aluminum</li> <li>• Addition of a threshold heat load below which operation of the SCS is not required <ul style="list-style-type: none"> <li>○ Modification to SCS design criteria to simplify the system</li> </ul> </li> </ul>
6	<ul style="list-style-type: none"> <li>• Allow for the use of instrument tube tie rods (ITTRs) to the approved contents of the MPC-24 and MPC-32 models</li> </ul>
7	<ul style="list-style-type: none"> <li>• Addition of the HI-STORM 100U and associated systems <ul style="list-style-type: none"> <li>○ Addition of restriction on support foundation</li> </ul> </li> <li>• Upgrade thermal simulations to utilize 3-D model of VVMs</li> <li>• Incorporation of mandatory radiation protection perimeter around loaded VVMs</li> <li>• Reinstates decay heat limits for damaged fuel and fuel debris in aboveground systems</li> <li>• Include provisions based on previously approved changes in Amendment 6 to include ITTRs to the approved contents of MPC-24 and MPC-32 models</li> </ul>
8	<ul style="list-style-type: none"> <li>• Addition of MPC-68M to approved models along with two new BWR fuel assembly/array classes</li> <li>• Addition of new PWR fuel assembly array/class for loading in MPC-32</li> <li>• Revised Condition 3 to include leak testing on confinement boundary base material as well <ul style="list-style-type: none"> <li>○ Revised of LCO 3.1.1.3 to be consistent with this change</li> </ul> </li> </ul>

Amendment Revision	Summary of Changes
8 Rev 1	<ul style="list-style-type: none"> <li>• Change burnup/cooling time limits for thimble plug devices (TPDs)</li> <li>• Change Metamic-HT testing requirements</li> <li>• Change Metamic-HT minimum guaranteed values (MGVs)</li> <li>• Allow for BWR fuel affected by certain corrosion mechanisms within specific guidelines to be classified as undamaged fuel</li> </ul>
9	<ul style="list-style-type: none"> <li>• Broaden the subgrade requirements for HI-STORM 100U system. Changes include: <ul style="list-style-type: none"> <li>○ Remove restriction that required the ISFSI Support Foundation Pad (SFP) to rest on a subgrade material with a shear wave velocity <math>\geq</math> 3,500 ft/s or bedrock</li> <li>○ Remove restriction that required any excavation, near an operating 100U ISFSI, to be a distance of ten times the depth of excavation away from the ISFSI</li> <li>○ Remove requirement that accounts for amplification in the seismic analysis</li> </ul> </li> <li>• Update thermal model and methodology for HI-TRAC from 2-D to 3-D thermal-hydraulic model. Specific changes include: <ul style="list-style-type: none"> <li>○ Removal of requirement for SCS for any heat load less than 90% of maximum allowable heat load limits (when MPC contains at least one fuel assembly with an average burnup greater than 45,000 MWd/MTU) to maintain spent nuclear fuel peak cladding temperatures below ISG-11 guidance limits</li> <li>○ Increase decay heat thresholds for MPC vacuum drying for unlimited and time restricted vacuum drying</li> </ul> </li> <li>• Adding time-to-boil limits for various decay heat loads and initial spent fuel pool temperatures</li> </ul>
9 Rev 1	<ul style="list-style-type: none"> <li>• Change burnup/cooling time limits for thimble plug devices (TPDs)</li> <li>• Change Metamic-HT testing requirements</li> <li>• Change Metamic-HT minimum guaranteed values (MGVs)</li> <li>• Allow for BWR fuel affected by certain corrosion mechanisms within specific guidelines to be classified as undamaged fuel</li> </ul>
10	<ul style="list-style-type: none"> <li>• Addition of new 16x16B and 16x16C fuel classes to approved contents</li> <li>• Change ASME code alternative table to allow use of more recent code versions of material SA-516/516A grade 70</li> <li>• Revision of CoC Condition 9 for additional clarity and guidance</li> </ul>

Amendment Revision	Summary of Changes
11	<ul style="list-style-type: none"> <li>• Increase the per storage location weight limit for cells authorized for DFCs in the MPC-68, -68FF, and 68M.</li> <li>• Modify the surveillance requirements and actions regarding blockage of the overpack vents.</li> <li>• Allow for the storage of low enriched CILC fuel as undamaged fuel assemblies, thereby eliminating the need to place these assemblies into DFCs.</li> <li>• Increase the allowable enrichment limit for BWR 10x10G fuel assemblies to include assemblies currently in use.</li> <li>• Lower the minimum soluble boron concentration limits for PWR 17x17A fuel assemblies.</li> <li>• Increase the allowable burnup limit to accommodate non-fuel hardware consisting of neutron source assemblies (NSAs) in combination with other control components.</li> <li>• Expand the allowable contents of the MPC-68M to include thoria rods/canister. <ul style="list-style-type: none"> <li>○ Add a second permissible composition of thoria rods allowed for storage in all MPC-68 models, including the MPC-68M.</li> </ul> </li> <li>• Clarify that lifting devices and lifting attachments shall be designed in accordance with the applicable stress limits of ANSI N14.6 and/or NUREG-0612.</li> </ul>
12	<ul style="list-style-type: none"> <li>• Add new loading patterns for the MPC-68M.</li> <li>• Add an exception to the ASME Code to allow the use of certain duplex stainless steels in the HI-STORM 100 system.</li> <li>• Permit the storage of damaged fuel/ fuel debris in DFCs under the regionalized loading pattern.</li> <li>• Add cyclic vacuum drying for all MPCs.</li> </ul>
13	<ul style="list-style-type: none"> <li>• Updated initial uranium weights (kg/assembly) for the 16x16B and 16x16C assembly classes to match the value used for the 16x16A assembly</li> </ul>

## 1.1 INTRODUCTION

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

The HI-STORM 100S and HI-STORM 100S Version B overpack designs are variants of the HI-STORM 100 overpack design and have their own drawings in Section 1.5. The "S" suffix indicates an enhanced overpack design, as described later in this section. "Version B" indicates an enhanced HI-STORM 100S overpack design. The HI-STORM 100S and 100S Version B accept the same MPCs and fuel types as the HI-STORM 100 overpack and the basic structural, shielding, and thermal-hydraulic characteristics remain unchanged. Hereafter in this FSAR reference to HI-STORM 100 System or the HI-STORM overpack is construed to apply to the various models, viz., HI-STORM 100, the HI-STORM 100S, and the HI-STORM 100S Version B. Where necessary, the text distinguishes among the different overpack designs. See Figures 1.1.1A and 1.1.3A for pictorial views of the HI-STORM 100S overpack design. See Figures 1.1.1B and 1.1.3B for pictorial views of the HI-STORM 100S Version B design.

The HI-STORM 100A overpack is a variant of two of the three HI-STORM 100 System overpack designs and is specially outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the ISFSI pad in high seismic applications. In the following, the modified structure of the HI-STORM 100A, in each of four quadrants, is denoted as a "sector lug." The HI-STORM 100A anchor design is applicable to the HI-STORM 100S overpack design, in which case the assembly would be named HI-STORM 100SA. The HI-STORM 100A anchor design is not applicable to the HI-STORM 100S Version B overpack design. Therefore, the HI-STORM 100S Version B overpack cannot be deployed in the anchored configuration at this time. Hereafter in the text, discussion of HI-STORM 100A applies to both the standard (HI-STORM 100A) and HI-STORM 100SA overpacks, unless otherwise clarified.

The axial channels welded to the HI-STORM 100S and HI-STORM 100 S Version B body may be optionally replaced with partial length tubular members aligned with the top lid of the MPC essentially identical to those used in the HI-STORM FW overpack. The tubular members provide significantly lower friction loss on the ventilation air in comparison to the full length channels. The necessary updates to this FSAR to incorporate the above improvement are made to Chapters 2, 3, and 4.

The HI-STORM 100 System is designed to accommodate a wide variety of spent nuclear fuel assemblies in a single basic overpack design by utilizing different MPCs. The external diameters of all MPCs are identical to allow the use of a single overpack. Each of the MPCs has different internals

(baskets) to accommodate distinct fuel characteristics. Each MPC is identified by the maximum quantity of fuel assemblies it is capable of receiving. The MPC-24, MPC-24E, and MPC-24EF contain a maximum of 24 PWR fuel assemblies; the MPC-32 and MPC-32F contain a maximum of 32 PWR fuel assemblies; and the MPC-68, MPC-68F, and MPC-68FF contain a maximum of 68 BWR fuel assemblies.

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

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

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

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

The HI-STORM 100 System is autonomous inasmuch as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any

other facility, structures, or components. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM 100 System since it is completely passive and is composed of materials with long proven histories in the nuclear industry. The HI-STORM 100 System can be used either singly or as the basic storage module in an ISFSI. The site for an ISFSI can be located either at a reactor or away from a reactor.

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

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

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

For example:

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

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

STORM 100A configuration is provided in Subsection 1.2.1.2.1. The anchored design is applicable to the HI-STORM 100 and the HI-STORM 100S overpack designs only.

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

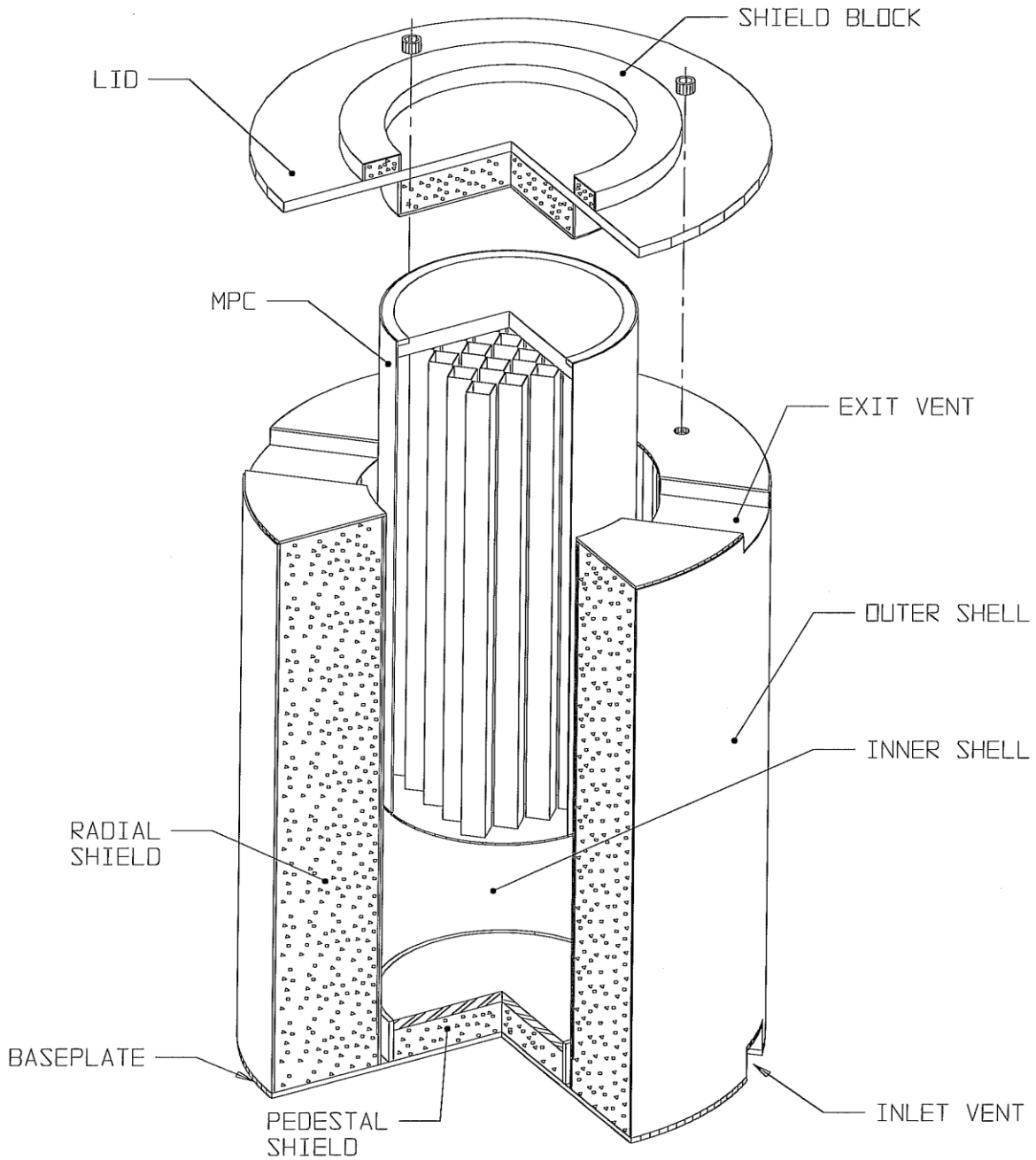
The MPC enclosure vessel is identical in external diameter to those presented in References [1.1.2] and [1.1.3]. However, certain fuel basket models may not be certified for storage or transportation in the HI-STAR 100 System. The Part 71 and 72 CoCs for HI-STAR 100 should be consulted for the MPC models that are certified for that system. Referencing these documents, as applicable, avoids repetition of information on the MPCs which is comprehensively set forth in the above-mentioned Holtec International documents docketed with the NRC. However, sufficient information and drawings are presented in this report to maintain clarity of exposition of technical data.

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

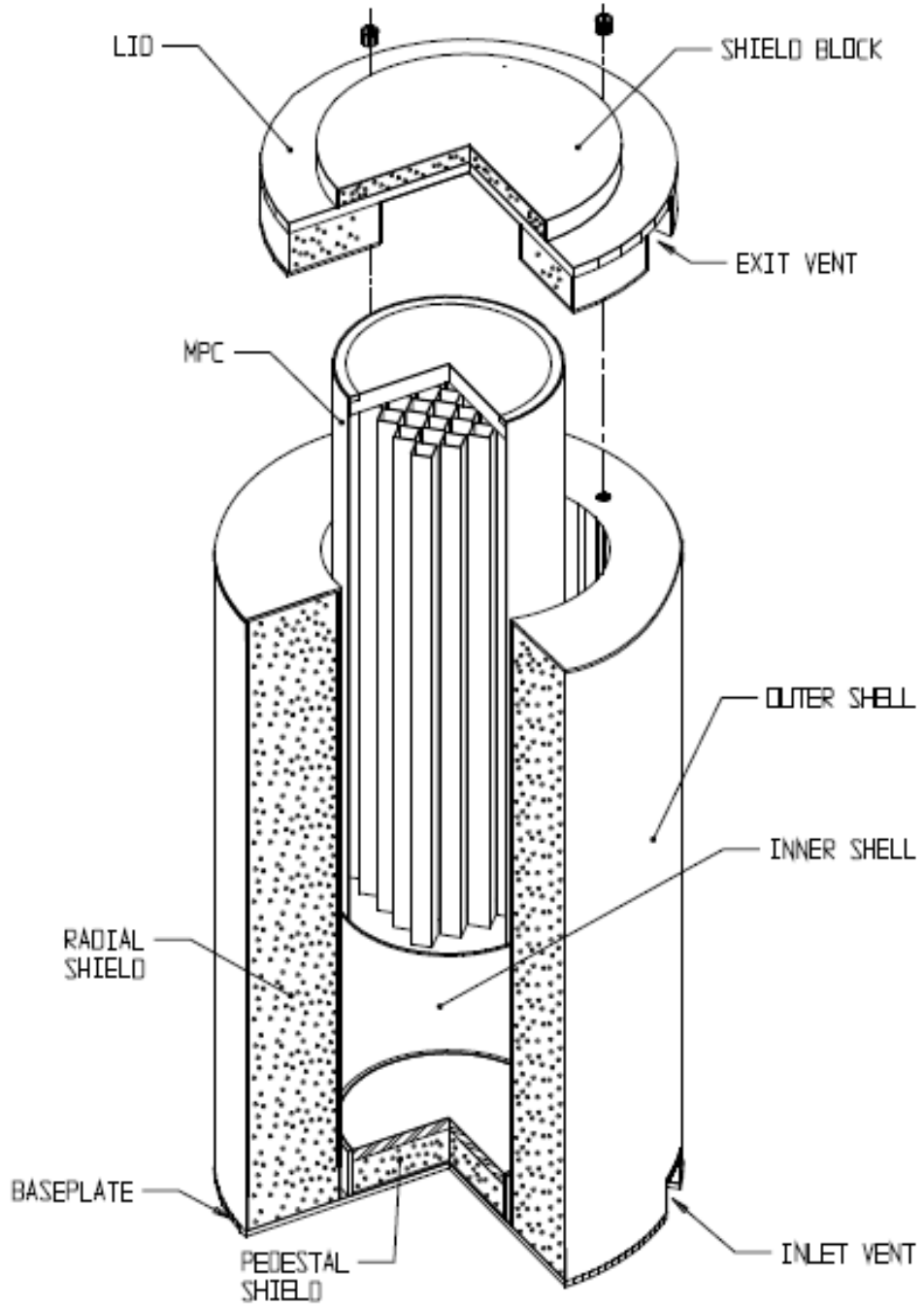
To summarize, the HI-STORM 100 System has been engineered to:

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

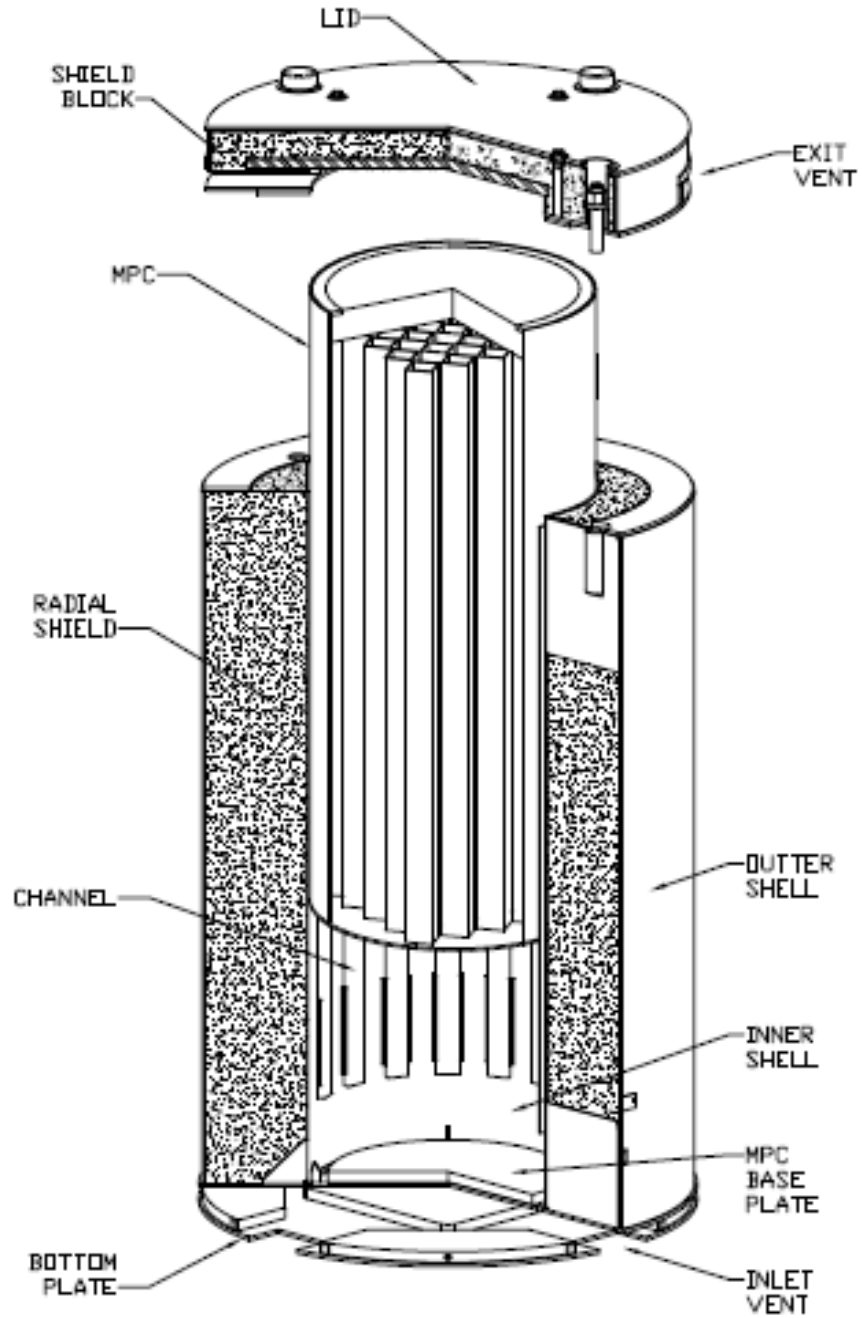




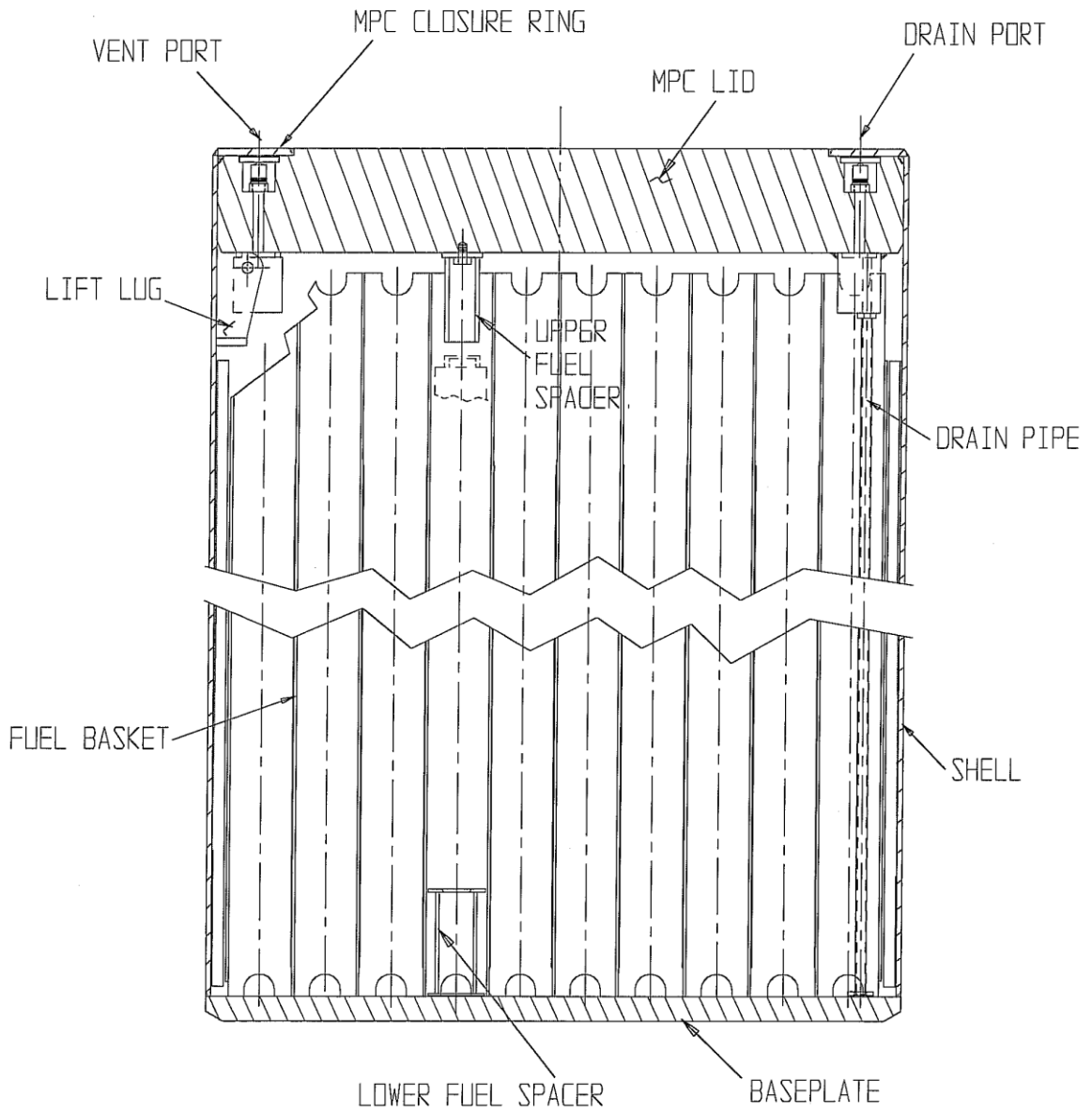
**FIGURE 1.1.1: HI-STORM 100 OVERPACK WITH MPC PARTIALLY INSERTED**



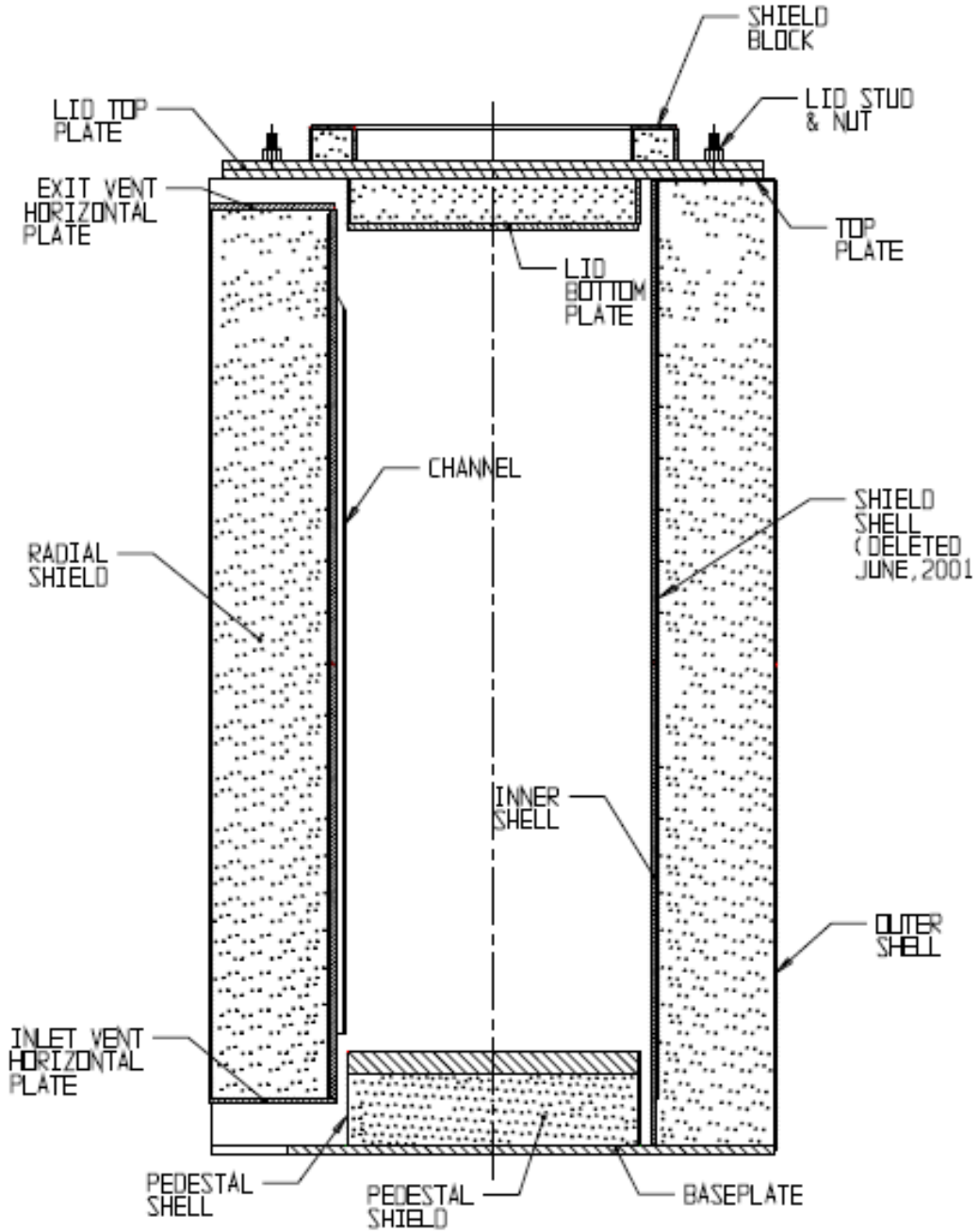
**FIGURE 1.1.1A: HI-STORM 100S OVERPACK WITH MPC PARTIALLY INSERTED**



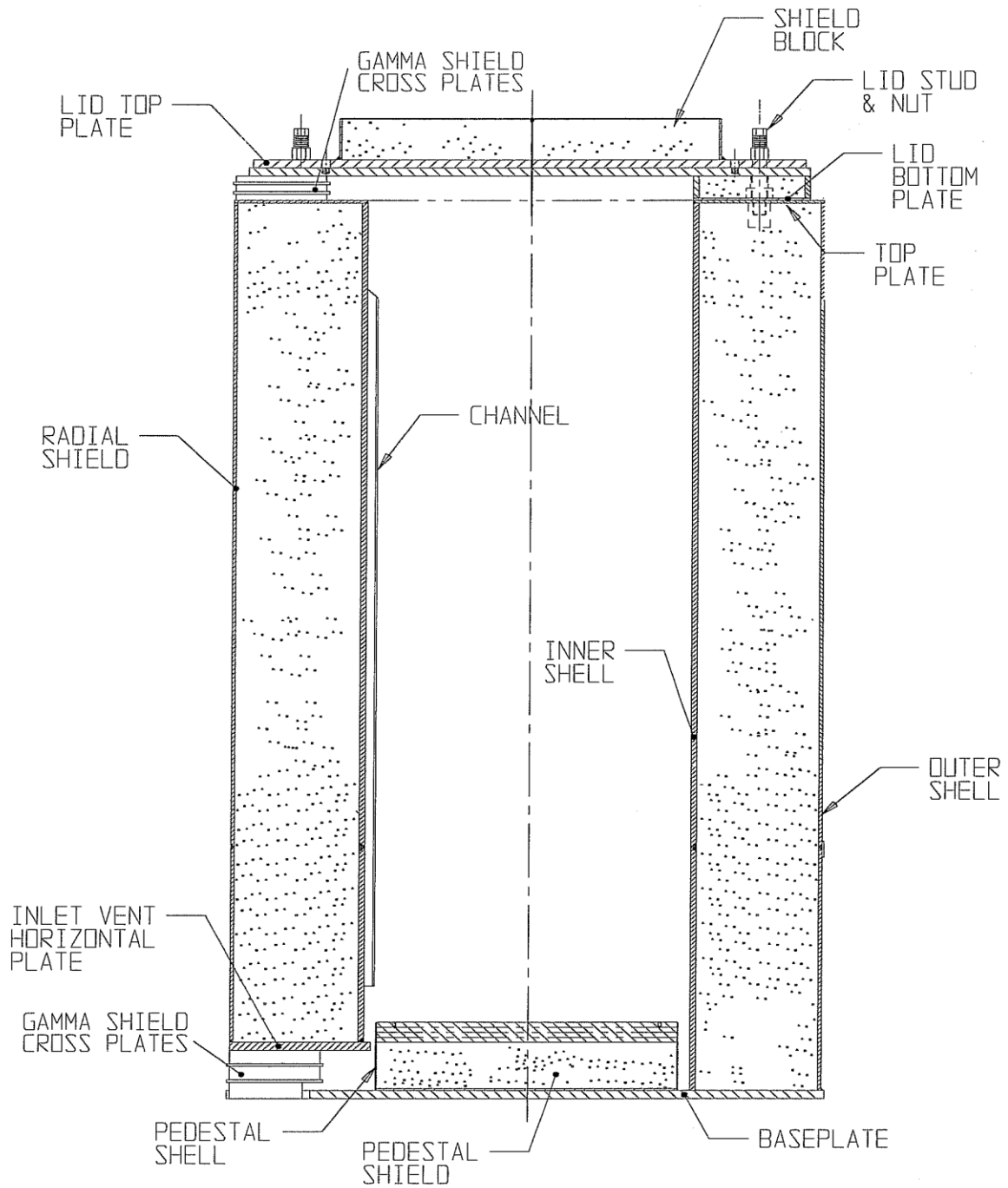
**FIGURE 1.1.1B: HI-STORM 100S VERSION B OVERPACK WITH MPC PARTIALLY INSERTED**



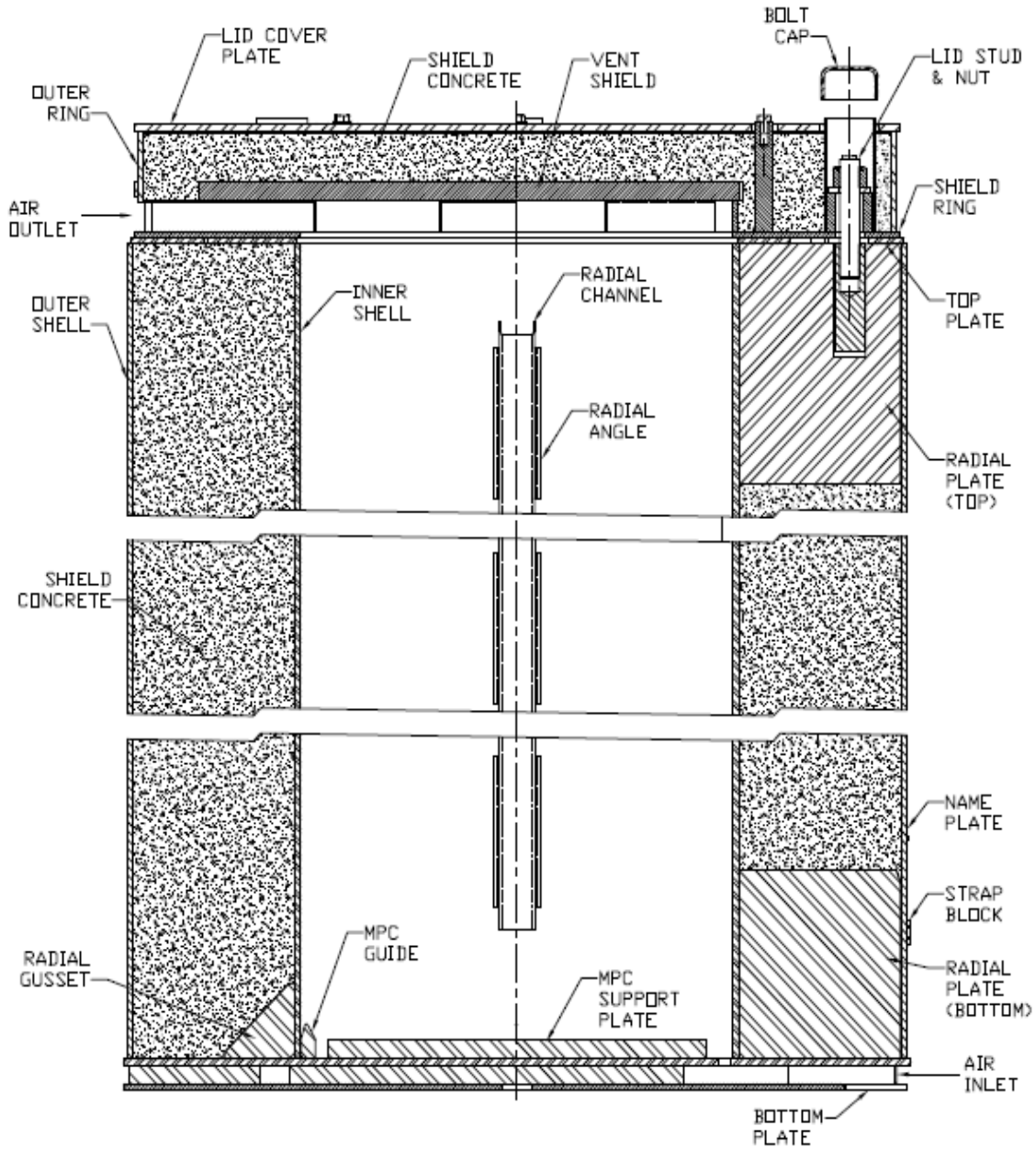
**FIGURE 1.1.2: CROSS SECTION ELEVATION VIEW OF MPC**



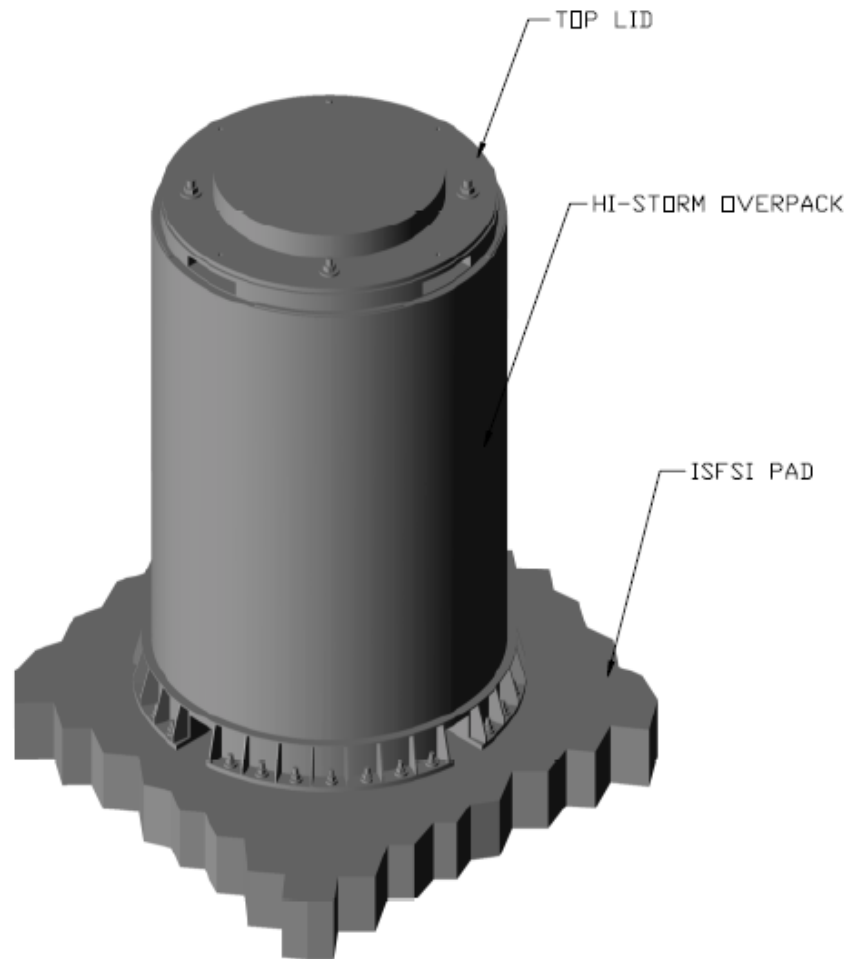
**FIGURE 1.1.3: HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW**



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

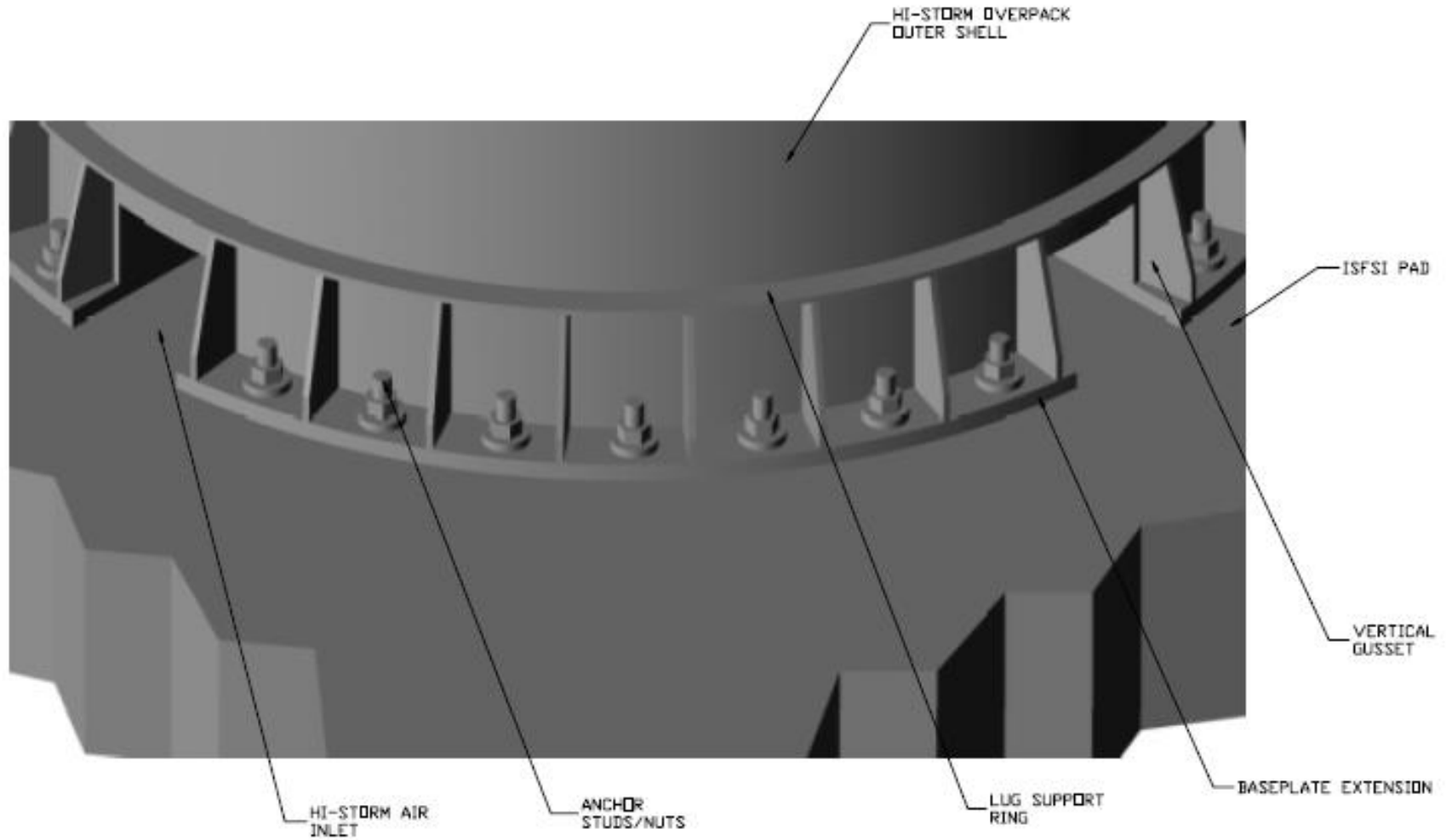


**FIGURE 1.1.3B: HI-STORM 100S VERSION B OVERPACK CROSS SECTIONAL ELEVATION VIEW**



**FIGURE 1.1.4: A PICTORIAL VIEW OF THE HI-STORM 100A OVERPACK  
(100SA MODEL SHOWN)**





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

## 1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

### 1.2.1 System Characteristics

The basic HI-STORM 100 System consists of interchangeable MPCs providing a confinement boundary for BWR or PWR spent nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the storage overpack. Figures 1.2.1 and 1.2.1A provide example cross sectional views of the HI-STORM 100 System with an MPC inserted into HI-STORM 100 and HI-STORM 100S storage overpacks, respectively. Figure 1.1.1B provides similar information for the HI-STORM 100 System using a HI-STORM 100S Version B overpack. Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. All structures, systems, and components of the HI-STORM 100 System, which are identified as Important to Safety are specified in Table 2.2.6. This discussion is supplemented with a full set of drawings in Section 1.5.

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

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

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

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

All MPCs have identical external diameters. The outer diameter of the MPC is 68-3/8 inches<sup>†</sup> and the maximum overall length is approximately 190-1/2 inches. See Section 1.5 for the MPC drawings. Due to the differing storage contents of each MPC, the maximum loaded weight differs among MPCs. See Table 3.2.1 for each MPC weight. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons. Tables 1.2.1 and 1.2.2 contain the key system data and parameters for the MPCs.

<sup>†</sup> Dimensions discussed in this section are considered nominal values.

A single, base HI-STORM overpack design is provided which is capable of storing each type of MPC. The overpack inner cavity is sized to accommodate the MPCs. The inner diameter of the overpack inner shell is 73-1/2 inches and the height of the cavity is 191-1/2 inches. The overpack inner shell is provided with channels (rectangular or cylindrical) distributed around the inner cavity to present an inside diameter of 69-1/2 inches. The channels are intended to offer a flexible medium to absorb some of the impact during a non-mechanistic tip-over, while still allowing the cooling air flow through the ventilated overpack. The outer diameter of the overpack is 132-1/2 inches. The overall height of the HI-STORM 100 overpack is 239-1/2 inches.

There are two variants of the HI-STORM 100S overpack, differing from each other only in height and weight. The HI-STORM 100S(232) is 232 inches high, and the HI-STORM 100S(243) is 243 inches high. The HI-STORM 100S(243) is approximately 10,100 lbs heavier assuming standard density concrete. Hereafter in the text, these two versions of the HI-STORM 100S overpack will only be referred to as HI-STORM 100S and will be discussed separately only if the design feature being discussed is different between the two overpacks. See Section 1.5 for drawings.

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

The weight of the overpack without an MPC varies from approximately 135 tons to 160 tons. See Table 3.2.1 for the detailed weights.

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

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

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

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

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

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

#### 1.2.1.1 Multi-Purpose Canisters

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

There are eight MPC models, distinguished by the type and number of fuel assemblies authorized for loading. Section 1.2.3 and Table 1.2.1 summarize the allowable contents for each MPC model. Section 2.1.9 provides the detailed specifications for the contents authorized for storage in the HI-STORM 100 System. Drawings for the MPCs are provided in Section 1.5.

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

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

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

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

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

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

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

The top end of the MPC incorporates a redundant closure system. Figure 1.2.6 shows the MPC

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

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

The MPC confinement boundary is constructed entirely from stainless steel alloy materials. All MPC components that may come into contact with spent fuel pool water or the ambient environment (with the exception of neutron absorber, aluminum seals on vent and drain port caps, and optional aluminum heat conduction elements) must be constructed from stainless steel alloy materials. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the MPC. All structural components in a MPC shall be made of Alloy X, a designation which warrants further explanation.

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

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

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

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

For the MPC design and analysis, Alloy X (as defined in this FSAR) may be one of the following materials. Any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed above, except that the steel pieces comprising the MPC shell (i.e., the 1/2" thick cylinder) must be fabricated from the same Alloy X stainless steel type.

The Fuel Baskets types used in the HI-STORM MPCs can be divided into two discrete categories:

- A. Alloy X Fuel baskets (ASME Code material)
- B. Metamic-HT baskets (Non-Code material)

#### A. Alloy X Baskets

An alloy X basket is one in which the structural function is rendered by Alloy X (defined below) and the neutron absorption function is rendered by Metamic (classic). 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 acceptable Alloy X materials listed above. Additional material grades (viz. duplex stainless steel, UNS S31803) are adopted in the Alloy X roster in Appendix 1.A. These duplex steels shall not be used for the fabrication of MPC baskets and internal components. Additional basket material types may be adopted in the Alloy X roster if determined to be acceptable through the §72.48 process.

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

The evaluation of the 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.

In Alloy X baskets, the Metamic neutron absorber panels are completely enclosed in stainless steel sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld. Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is

an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient evolutions. Finally, the pocket joint detail ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.

## B. Metamic HT Baskets

In Metamic HT baskets, both the structural and criticality prevention function is rendered by Metamic HT.

The MPC-68M in Supplement 1.III is a Metamic HT basket. MPC-68M contains a 68 storage cell BWR fuel basket made of co-planar slotted plates of Metamic-HT. The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. The fuel basket is assembled from a rectilinear gridwork of plates so that there are no bends or radii at the cell corners. This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls that must transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (e.g., non-mechanistic tipover, uncontrolled lowering of a cask during on-site transfer, or off-site transport accident events, etc.).

**Basket Shims:** The HI-STORM 100 MPCs utilize two types of shims that constrain lateral rattling of the basket structures. These are:

1. Alloy X Shims which are characterized by being held in place by welding them to either the MPC Enclosure Vessel or the MPC fuel basket. The Alloy X shims are shaped to provide maximum support to the basket under lateral loading scenarios. They have been primarily used in Alloy X baskets.
2. Extruded Aluminum Shims: These shims, primarily used in all Metamic HT Fuel Baskets, are installed in the peripheral space between the fuel basket and the Enclosure Vessel to provide conformal contact surfaces between the Basket Shims and the fuel basket and between the Basket Shims and the Enclosure Vessel shell. The axial holes in the corner basket shims serve as the passageway for the downwards flow of the helium gas under the thermosiphon action, which is intrinsic to the thermal performance of all MPCs in the HI-STORM 100 system. Extruded Aluminum Shims have been allowed in the MPC-68M since the initial licensing by the NRC. Extruded Aluminum Shims may be used in other Alloy X and Metamic Fuel Baskets after the necessary safety evaluation pursuant to §72.48 is successfully performed.

### 1.2.1.2 Overpacks

#### 1.2.1.2.1 HI-STORM Overpack

The HI-STORM overpacks are rugged, heavy-walled cylindrical vessels. Figures 1.1.3B, 1.2.7, 1.2.8,



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

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

The storage system has air ducts to allow for passive natural convection cooling of the contained MPC. A minimum of four air inlets and four air outlets are located at the lower and upper extremities of the storage system, respectively. The location of the air outlets in the HI-STORM 100 and the HI-STORM 100S (including Version B) design differ in that the outlet ducts for the HI-STORM 100 overpack are located in the overpack body and are aligned vertically with the inlet ducts at the bottom of the overpack body. The air outlet ducts in the HI-STORM 100S and -100S Version B are integral to the lid assembly and are not in vertical alignment with the inlet ducts. See the drawings in Section 1.5 for details of the overpack air inlet and outlet duct designs. The air inlets and outlets are covered by a screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, described in Chapter 11 of this FSAR, evaluates the effects of partial and complete blockage of the air ducts.

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

Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates, which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM 100) or the inlet air duct horizontal plates (HI-STORM 100S) (see Figure 1.2.7). The HI-STORM 100S Version B overpack design incorporates partial-length radial plates at the top of the overpack to secure the anchor blocks and uses both gussets and partial-length radial plates at the bottom of the overpack for structural stability. Details of this arrangement are shown in the drawings in Section 1.5.

The four anchor blocks are located on 90° arcs around the circumference of the top of the overpack lid. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, Hillman rollers, or other design based on site-specific needs and capabilities. Slings or other suitable devices mate with lifting lugs that are inserted into threaded holes in the top surface of the overpack lid to allow lifting of the overpack lid. After the lid is bolted to the storage overpack main body, these lifting bolts shall be removed and replaced with flush plugs.

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

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

Density and compressive strength are the key parameters that delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal (HI-STORM 100 and -100S overpacks only), and overpack lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.0.5] are used.

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

There are three base HI-STORM overpack designs - HI-STORM 100, HI-STORM 100S, and HI-STORM 100S Version B. The significant differences among the three are overpack height, MPC

pedestal height, location of the air outlet ducts, and the vertical alignment of the inlet and outlet air ducts. The HI-STORM 100 overpack is approximately 240 inches high from the bottom of the baseplate to the top of the lid bolts and 227 inches high without the lid installed. There are two variants of the HI-STORM 100S overpack design, differing only in height and weight. The HI-STORM 100S(232) is approximately 232 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 211 inches high without the lid installed. The HI-STORM 100S(243) is approximately 243 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 222 inches high without the lid installed. There are also variants of the HI-STORM 100S Version B overpack design, differing only in height and weight. The HI-STORM 100S-218 is approximately 218 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and approximately 199 inches high without the lid installed. The HI-STORM 100S-229 is approximately 229 inches high from the bottom of the baseplate to the top of the lid in its final storage configuration and 210 inches high without the lid installed.

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

The anchored embodiment of the HI-STORM overpack is referred to as HI-STORM 100A or HI-STORM 100SA. The HI-STORM 100S version B overpack design may not be deployed in the anchored configuration at this time. As explained in the foregoing, the HI-STORM overpack is a steel weldment, which makes it a relatively simple matter to extend the overpack baseplate, form lugs, and then anchor the cask to the reinforced concrete structure of the ISFSI. In HI-STORM terminology, these lugs are referred to as “sector lugs.” The sector lugs, as shown in Figure 1.1.5 and the drawing in Section 1.5, are formed by extending the HI-STORM overpack baseplate, welding vertical gussets to the baseplate extension and to the overpack outer shell and, finally, welding a horizontal lug support ring in the form of an annular sector to the vertical gussets and to the outer shell. The baseplate is equipped with regularly spaced clearance holes (round or slotted) through which the anchor studs can pass. The sector lugs are bolted to the ISFSI pad using anchor studs that are made of a creep-resistant, high-ductility, environmentally compatible material. The bolts are pre-loaded to a precise axial stress using a “stud tensioner” rather than a torque wrench. Pre-tensioning the anchors using a stud tensioner eliminates any shear stress in the bolt, which is unavoidable if a torquing device is employed (Chapter 3 of the text “Mechanical Design of Heat Exchangers and Pressure Vessel Components”, by Arcturus Publishers, 1984, K.P. Singh and A.I. Soler, provides additional information on stud tensioners). The axial stress in the anchors induced by pre-tensioning is kept below 75% of the material yield stress, such that during the seismic event the maximum bolt axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF (for Level D conditions). Figures 1.1.4 and 1.1.5 provide visual depictions of the anchored HI-STORM 100A configuration. This configuration also applies to the HI-STORM 100SA.

The anchor studs pass through liberal clearance holes (circular or slotted) in the sector lugs (0.75” minimum clearance) such that the fastening of the studs to the ISFSI pad can be carried out without

mechanical interference from the body of the sector lug. The two clearance hole configurations give the ISFSI pad designer flexibility in the design of the anchor embedment in the ISFSI concrete. The axial force in the anchors produces a compressive load at the overpack/pad interface. This compressive force,  $F$ , imparts a lateral load bearing capacity to the cask/pad interface that is equal to  $\mu F$  ( $\mu \leq 0.53$  per Table 2.2.8). As is shown in Chapter 3 of this FSAR, the lateral load-bearing capacity of the HI-STORM/pad interface ( $\mu F$ ) is many times greater than the horizontal (sliding) force exerted on the cask under the postulated DBE seismic event. Thus, the potential for lateral sliding of the HI-STORM 100A System during a seismic event is precluded, as is the potential for any bending action on the anchor studs.

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

The sector lugs in HI-STORM 100A are made of the same steel material as the baseplate and the shell (SA516- Gr. 70) which helps ensure high quality fillet welds used to join the lugs to the body of the overpack. The material for the anchor studs can be selected from a family of allowable stud materials listed in the ASME Code (Section II). A representative sampling of permitted materials is listed in Table 1.2.7. The menu of materials will enable the ISFSI owner to select a fastener material that is resistant to corrosion in the local ISFSI environment. For example, for ISFSIs located in marine environments (e.g., coastal reactor sites), carbon steel studs would not be recommended without concomitant periodic inspection and coating maintenance programs. Table 1.2.7 provides the chemical composition of several acceptable fastener materials to help the ISFSI owner select the most appropriate material for his site. The two mechanical properties, ultimate strength  $\sigma_u$  and yield strength  $\sigma_y$  are also listed. For purposes of structural evaluations, the lower bound values of  $\sigma_u$  and  $\sigma_y$  from the menu of materials listed in Table 1.2.7 are used (see Table 3.4.10).

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

#### 1.2.1.2.2 HI-TRAC (Transfer Cask) - Standard Design

Like the storage overpack, the HI-TRAC transfer cask is a rugged, heavy-walled cylindrical vessel. The main structural function of the transfer cask is provided by carbon steel, and the main neutron and gamma shielding functions are provided by water and lead, respectively. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. Figure 1.2.9 provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. See Section 1.2.1.2.3 for discussion of the optional HI-TRAC 100D and 125D designs.

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

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

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

#### 1.2.1.2.3 HI-TRAC 100D and 125D Transfer Casks

As an option to using either of the standard HI-TRAC transfer cask designs, users may choose to use the optional HI-TRAC 100D or 125D designs. Figure 1.2.9A provides a typical cross section of the HI-TRAC-125D with the pool lid installed. The HI-TRAC 100D (figure not shown) is similar to the HI-TRAC 125D except for the top lid (which contains no Holtite). Like the standard designs, the optional designs are designed and constructed in accordance with ASME III, Subsection NF, with certain NRC-approved alternatives, as discussed in Section 2.2.4. Functionally equivalent, the major differences between the HI-TRAC 100D and 125D designs and the standard designs are as follows:

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

The interface between the MPC and the transfer cask is the same between the standard designs and the optional designs. The optional designs are capable of withstanding all loads defined in the design basis for the transfer cask during normal, off-normal, and accident modes of operation with adequate safety margins. In lieu of swapping the pool lid for the transfer lid to facilitate MPC transfer, the pool lids remain on the HI-TRAC 100D and 125D until MPC transfer is required. The HI-STORM mating device is located between, and secured with bolting (if required by seismic analysis), to the top of the HI-STORM overpack and the HI-TRAC 100D or 125D transfer cask. The mating device is used to remove the pool lid to provide a pathway for MPC transfer between the overpack and the transfer cask. Section 1.2.2.2 provides additional detail on the differences between the standard transfer cask designs and the optional HI-TRAC 100D or 125D designs during operations.

### 1.2.1.3 Shielding Materials

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

In the HI-STORM storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the lid, and the HI-STORM 100 and –100S have a thick circular concrete pedestal upon which the MPC rests. This concrete pedestal is not necessary in the HI-STORM 100S Version B overpack design. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shielding integral to the lid provide additional gamma attenuation in the upward direction,

reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness. Gamma shield cross plates, as depicted in Figure 5.3.19, provide attenuation of scattered gamma radiation as it exits the inlet and outlet air ducts.

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

#### 1.2.1.3.1 Fixed Neutron Absorbers

##### 1.2.1.3.1.1 Boral<sup>TM</sup>

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

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

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

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

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

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

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

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

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

Operating experience in nuclear plants with fuel loading of Boral equipped MPCs as well as laboratory test data indicate that the aluminium used in the manufacture of the Boral may react with water, resulting in the generation of hydrogen. The numerous variables (i.e., aluminium particle size, pool temperature, pool chemistry, etc.) that influence the extent of the hydrogen produced make it impossible to predict the amount of hydrogen that may be generated during MPC loading or



unloading at a particular plant. Therefore, due to the variability in hydrogen generation from the Boral-water reaction, the operating procedures in Chapter 8 require monitoring for combustible gases and purging the space beneath the MPC lid during loading and unloading operations when an ignition event could occur (i.e., when the space beneath the MPC lid is open to the welding or cutting operation).

#### 1.2.1.3.1.2 METAMIC®

METAMIC® is a neutron absorber material developed by the Reynolds Aluminum Company in the mid-1990s for spent fuel reactivity control in dry and wet storage applications. Metallurgically, METAMIC® is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy reinforced with Type 1 ASTM C-750 boron carbide. METAMIC® is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. As described in the U.S. patents held by METAMIC, Inc.<sup>12</sup>, the high performance and reliability of METAMIC® derives from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields excellent and uniform homogeneity.

The powders are carefully blended without binders or other additives that could potentially adversely influence performance. The maximum percentage of B<sub>4</sub>C that can be dispersed in the aluminum alloy 6061 matrix is approximately 40 wt.%, although extensive manufacturing and testing experience is limited to approximately 31 wt.%. The blend of powders is isostatically compacted into a green billet under high pressure and vacuum sintered to near theoretical density.

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

METAMIC® has been subjected to an extensive array of tests sponsored by the Electric Power Research Institute (EPRI) that evaluated the functional performance of the material at elevated temperatures (up to 900°F) and radiation levels (1E+11 rads gamma). The results of the tests documented in an EPRI report (Ref. [1.2.11]) indicate that METAMIC® maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state. The main conclusions provided in the above-referenced EPRI report are summarized below:

- The metal matrix configuration produced by the powder metallurgy process with a complete absence of open porosity in METAMIC® ensures that its density is essentially equal to the theoretical density.
- The physical and neutronic properties of METAMIC® are essentially unaltered under exposure to elevated temperatures (750° F - 900° F).

<sup>1</sup> U.S. Patent No. 5,965,829, “Radiation Absorbing Refractory Composition”.

<sup>2</sup> U.S. Patent No. 6,042,779, “Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super, Hypereutectic Al/Si.”

- No detectable change in the neutron attenuation characteristics under accelerated corrosion test conditions has been observed.

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

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

Consistent with its role in reactivity control, all METAMIC<sup>®</sup> material procured for use in the Holtec MPCs will be qualified as important-to-safety (ITS) Category A item. ITS category A manufactured items, as required by Holtec's NRC-approved Quality Assurance program, must be produced to essentially preclude the potential of an error in the procurement of constituent materials and the manufacturing processes. Accordingly, material and manufacturing control processes must be established to eliminate the incidence of errors, and inspection steps must be implemented to serve as an independent set of barriers to ensure that all critical characteristics defined for the material by the cask designer are met in the manufactured product.

All manufacturing and in-process steps in the production of METAMIC<sup>®</sup> shall be carried out using written procedures. As required by the company's quality program, the material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances as set forth in the applicable Holtec QA procedures to ensure that all METAMIC<sup>®</sup> panels procured meet with the requirements appropriate for the quality genre of the MPCs. Additional details pertaining to the qualification and production tests for METAMIC<sup>®</sup> are summarized in Subsection 9.1.5.3.

Because of the absence of interconnected porosities, the time required to dehydrate a METAMIC<sup>®</sup>-equipped MPC is expected to be less compared to an MPC containing Boral.

NUREG/CR-5661 (Ref. [1.2.14]) recommends limiting poison material credit to 75% of the minimum <sup>10</sup>B loading because of concerns for potential "streaming" of neutrons, and allows for greater percentage credit in criticality analysis "if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented". The value of 75% is characterized in NUREG/CR-5661 as a very conservative value, based on experiments with neutron poison containing relatively large B<sub>4</sub>C particles, such as BORAL with an average particle size in excess of 100 microns. METAMIC<sup>®</sup>, however, has a much smaller particle size of typically

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

less than 25 microns on average. Any streaming concerns would therefore be drastically reduced.

Analyses performed by Holtec International show that the streaming due to particle size is practically non-existent in METAMIC<sup>®</sup>. Further, EPRI's neutron attenuation measurements on 31 and 15 B<sub>4</sub>C weight percent METAMIC<sup>®</sup> showed that METAMIC<sup>®</sup> exhibits very uniform <sup>10</sup>B areal density. This makes it easy to reliably establish and verify the presence and microscopic and macroscopic uniformity of the <sup>10</sup>B in the material. Therefore, 90% credit is applied to the minimum <sup>10</sup>B areal density in the criticality calculations, i.e. a 10% penalty is applied. This 10% penalty is considered conservative since there are no significant remaining uncertainties in the <sup>10</sup>B areal density. In Chapter 9 the qualification and on production tests for METAMIC<sup>®</sup> to support 90% <sup>10</sup>B credit are specified. With 90% credit, the target weight percent of boron carbide in METAMIC<sup>®</sup> is 31 for all MPCs, as summarized in Table 1.2.8, consistent with the test coupons used in the EPRI evaluations [1.2.11]. The maximum permitted value is 33.0 wt% to allow for necessary fabrication flexibility.

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

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

Mechanical Properties of 31wt.% B <sub>4</sub> C METAMIC		
Property	As-Fabricated	After 48 hours of 900°F Temperature Soak
Yield Strength (psi)	32937 ± 3132	28744 ± 3246
Ultimate Strength (psi)	40141 ± 1860	34608 ± 1513
Elongation (%)	1.8 ± 0.8	5.7 ± 3.1

The required flexural strain of the neutron absorber to ensure that it will not fracture when the supporting basket wall flexes due to the worst case lateral loading is 0.2%, which is the flexural strain of the Alloy X basket panel material. The 1% minimum elongation of 31wt.% B<sub>4</sub>C METAMIC<sup>®</sup> indicated by the above table means that a large margin of safety against cracking exists, so there is no need to perform testing of the METAMIC<sup>®</sup> for mechanical properties.

EPRI's extensive characterization effort [1.2.11], which was focused on 15 and 31 wt.% B<sub>4</sub>C METAMIC<sup>®</sup> served as the principal basis for a recent USNRC SER for 31wt.% B<sub>4</sub>C METAMIC for used in wet storage [1.2.12]. Additional studies on METAMIC<sup>®</sup> [1.2.13], EPRI's and others work

provide the confidence that 31wt.% B<sub>4</sub>C METAMIC<sup>®</sup> will perform its intended function in the MPCs.

#### 1.2.1.3.1.3 Locational Fixity of Neutron Absorbers

Both Boral and METAMIC<sup>®</sup> neutron absorber panels are completely enclosed in Alloy X (stainless steel) sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld. Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient evolutions. Finally, the pocket joint detail ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.

#### 1.2.1.3.2 Neutron Shielding

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

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

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

Neutron attenuation in the HI-STORM overpack is provided by the thick walls of concrete contained in the steel vessel, lid, and pedestal (only for the HI-STORM 100 and –100S overpack designs). Concrete is a shielding material with a long proven history in the nuclear industry. The concrete composition has been specified to ensure its continued integrity at the long term temperatures required for SNF storage.

The HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding.

Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) will be added to reduce the freezing point for low temperature operations (e.g., below 32°F) [1.2.7].

Neutron shielding in the HI-TRAC 125 and 125D transfer casks in the axial direction is provided by Holtite-A within the top lid. HI-TRAC 125 also contains Holtite-A in the transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal B<sub>4</sub>C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

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

### Density

The specific gravity of Holtite-A is 1.68 g/cm<sup>3</sup> as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm<sup>3</sup>. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm<sup>3</sup> to underestimate the shielding capabilities of the neutron shield.

### Hydrogen

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

### Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (nominal) weight concentration. Holtite-A may be specified with a B<sub>4</sub>C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal B<sub>4</sub>C weight percent of 1%.

### Design Temperature

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

### Thermal Conductivity

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

### 1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7)<sup>†</sup>. Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the pedestal (HI-STORM 100 and –100S overpack designs only) and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are installed in the ducts (Figures 1.2.8 and 1.2.8A) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets. See Figure 5.3.19 and the drawings in Section 1.5 for more details of the gamma shield cross plate designs for each overpack design.

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

### 1.2.1.4 Lifting Devices

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

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment.

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

The standard design HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket trunnions. The HI-TRAC 100D and 125D are equipped with only lifting trunnions. The lifting trunnions are positioned just below the top forging. The two pocket trunnions are located above the bottom forging and attached to the outer shell. The pocket trunnions are designed to allow rotation of the HI-TRAC. All trunnions are built from a high strength alloy with proven corrosion and non-galling characteristics. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The lifting trunnions are installed by threading into tapped holes just below the top forging.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC.

<sup>†</sup> The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated.

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

#### 1.2.1.5 Design Life

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

#### HI-STORM Overpack and HI-TRAC Transfer Cask

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

#### MPC

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

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

#### 1.2.2 Operational Characteristics

##### 1.2.2.1 Design Features

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

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

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

primarily provided for radiation shielding and the steel is primarily provided for structural functions.

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

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

#### 1.2.2.2 Sequence of Operations

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

#### Loading Operations

At the start of loading operations, the HI-TRAC transfer cask is configured with the pool lid installed. The HI-TRAC water jacket is filled with demineralized water or a 25% ethylene glycol solution depending on the ambient temperature conditions. The lift yoke is used to position HI-TRAC in the designated preparation area or setdown area for HI-TRAC inspection and MPC insertion. The annulus is filled with plant demineralized water, and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with water (borated if necessary). Based on the MPC model and fuel enrichment, this may be borated water or plant demineralized water (see Section 2.1). HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

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

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. The Automated Welding System baseplate shield (if used) is installed to reduce dose rates



around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS) or other approved welding process. Liquid penetrant examinations are performed on the root and final passes. A multi-layer liquid penetrant or volumetric examination is also performed on the MPC lid-to-shell weld. The MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line. At the appropriate time in the sequence of activities, appropriate non-destructive tests on the MPC Enclosure Vessel as set down in Chapter 9 are performed.

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

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

Following moisture removal, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage and provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal-welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes. The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.

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

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

and rigging support the MPC during the transfer operations.

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

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

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

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

The HI-TRAC 100D and 125D do not include pocket trunnions in their designs. Therefore, the user must downend the transfer cask onto the transport frame using appropriately designed rigging in

accordance with the site's heavy load control program.

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

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

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

After the HI-TRAC is positioned atop the HI-STORM or the mating device, the MPC is raised slightly. In the standard design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 100D and 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and HI-TRAC is removed from on top of HI-STORM or disconnected from the mating device, as applicable.

For the HI-STORM 100S and the HI-STORM 100S Version B, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 100D or 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs and nuts are installed.

After the HI-STORM has been loaded either within the fuel building or at a dedicated cask transfer facility, the HI-STORM is then moved to its designated position on the ISFSI pad. The HI-STORM overpack may be moved using a number of methods as long as the handling limitations listed in the technical specifications are not exceeded. The loaded HI-STORM must be handled in the vertical orientation, and may be lifted from the top by the anchor blocks or from the bottom by the inlet vents. After the loaded HI-STORM is lifted, it may be placed on a transport mechanism or continue to be lifted by the lid studs and transported to the storage location. The transport mechanism may be

an air pad, crawler, railcar, heavy-haul trailer, dolly, etc. During the transport of the loaded HI-STORM, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms. Once in position at the storage pad, vent operability testing is performed to ensure that the system is functioning within its design parameters.

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

### Unloading Operations

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

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

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

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

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established around the MPC ports. The RVOAs are attached to the vent and drain port. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is flooded with borated or unborated water, as required. The MPC lid-to-MPC shell weld is removed. Then, all

weld removal equipment is removed with the MPC lid left in place.

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

### 1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

#### 1.2.2.3.1 Criticality Prevention

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

Each MPC model is equipped with neutron absorber plates affixed to the fuel cell walls as shown on the drawings in Section 1.5. The minimum  $^{10}\text{B}$  areal density specified for the neutron absorber in each MPC model is shown in Table 1.2.2. These values are chosen to be consistent with the assumptions made in the criticality analyses.

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

#### 1.2.2.3.2 Chemical Safety

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

#### 1.2.2.3.3 Operation Shutdown Modes

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

#### 1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is a high integrity pressure vessel designed and constructed to be *leak tight* as discussed in Chapter 7. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM overpack exit vents in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 for additional details.

#### 1.2.2.3.5 Maintenance Technique

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

#### 1.2.3 Cask Contents

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

#### MPC-24

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

#### MPC 24E and MPC-24EF

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

#### MPC-32 and MPC-32F

The MPC-32 and MPC-32F are designed to store up to thirty two (32) PWR fuel assemblies with or

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

### MPC-68F

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

### MPC-68 and MPC-68FF

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

Table 1.2.1

## KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs	8	5 for PWR 3 for BWR
MPC storage capacity <sup>†</sup> :	MPC-24 MPC-24E MPC-24EF	Up to 24 intact ZR or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to four damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24E or MPC-24EF  OR  Up to 32 intact ZR or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to 8 damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-32 or MPC-32F.
	MPC-32 MPC-32F	

<sup>†</sup> See Section 2.1 for a complete description of authorized cask contents and fuel specifications.





Table 1.2.2

## KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725 <sup>o†</sup> /-40 <sup>o††</sup>	725 <sup>o†</sup> /-40 <sup>o††</sup>
Design internal pressure (psig)		
Normal conditions	100	100
Off-normal/Short-term conditions	110	110
Accident Conditions	200	200
Total heat load, max. (kW) <sup>††††</sup>	36.9	36.9
Maximum permissible peak fuel cladding temperature:		
Long Term Normal (°F)	752	752
Short Term Operations (°F)	752 or 1058 <sup>†††</sup>	752 or 1058 <sup>†††</sup>
Off-normal and Accident (°F)	1058	1058

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

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

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

†††† Maximum heat load shown is for regionalized loading.

Table 1.2.2 (cont'd)

## KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
MPC internal environment Helium fill (99.995% fill helium purity) (See Note 2)	(all pressure ranges are at a reference temperature of 70°F)	(all pressure ranges are at a reference temperature of 70°F)
MPC-24 (heat load $\leq$ 27.77 kW, see Table 2.1.31)	$\geq$ 29.3 psig and $\leq$ 48.5 psig OR 0.1212 +/-10% g-moles/liter	
(heat load > 27.77 kW)	$\geq$ 45.5 psig and $\leq$ 48.5 psig	
MPC-24E/24EF (heat load $\leq$ 28.17 kW, see Table 2.1.31)	$\geq$ 29.3 psig and $\leq$ 48.5 psig OR 0.1212 +/-10% g-moles/liter	
(heat load > 28.17 kW)	$\geq$ 45.5 psig and $\leq$ 48.5 psig	
MPC-68/68F/68FF (heat load $\leq$ 28.19 kW, see Table 2.1.31)		$\geq$ 29.3 psig and $\leq$ 48.5 psig OR 0.1218 +/-10% g-moles/liter
(heat load >28.19 kW)		$\geq$ 45.5 psig and $\leq$ 48.5 psig
MPC-32/32F (heat load $\leq$ 28.74 kW, see Table 2.1.31)	$\geq$ 29.3 psig and $\leq$ 48.5 psig	
(heat load >28.74 kW)	$\geq$ 45.5 psig and $\leq$ 48.5 psig	
Maximum permissible multiplication factor ( $k_{eff}$ ) including all uncertainties and biases	< 0.95	< 0.95
Fixed Neutron Absorber <sup>10</sup> B Areal Density (g/cm <sup>2</sup> )	0.0267/0.0223 (MPC-24)	0.0372/0.0310 (MPC-68 & MPC-68FF)
Boral/Metamic	0.0372/0.0310 (MPC-24E, MPC-24EF MPC-32 & MPC-32F)	0.01/NA (MPC-68F) (See Note 1)
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

NOTE:

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

Table 1.2.3

INTENTIONALLY DELETED

Table 1.2.4

INTENTIONALLY DELETED

Table 1.2.5

INTENTIONALLY DELETED

Table 1.2.6

## HI-STORM 100 OPERATIONS SEQUENCE

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



Table 1.2.7

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

**ASME MATERIALS FOR BOLTING**

Composition	I.D.	Type Grade or UNC No.	Ultimate Strength (ksi)	Yield Strength (ksi)	Code Permitted Size Range <sup>†</sup>
C	SA-354	BC K04100	125	109	$t \leq 2.5''$
$\frac{3}{4}$ Cr	SA-574	51B37M	170	135	$t \geq 5/8''$
1 Cr – 1/5 Mo	SA-574	4142	170	135	$t \geq 5/8''$
1 Cr-1/2 Mo-V	SA-540	B21 (K 14073)	165	150	$t \leq 4''$
5 Cr – 1/2 Mo	SA-193	B7	125	105	$t \leq 2.5''$
2Ni – 3/4 Cr – 1/4 Mo	SA-540	B23 (H-43400)	135	120	
2Ni – 3/4 Cr – 1/3 Mo	SA-540	B-24 (K-24064)	135	120	
17Cr-4Ni-4Cu	SA-564	630 (H-1100)	140	115	
17Cr-4Ni-4Cu	SA-564	630 (H-1075)	145	125	
25Ni-15Cr-2Ti	SA-638	660	130	85	
22CR-13Ni-5Mn	SA-479	XM-19 (S20910)	135	105	

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

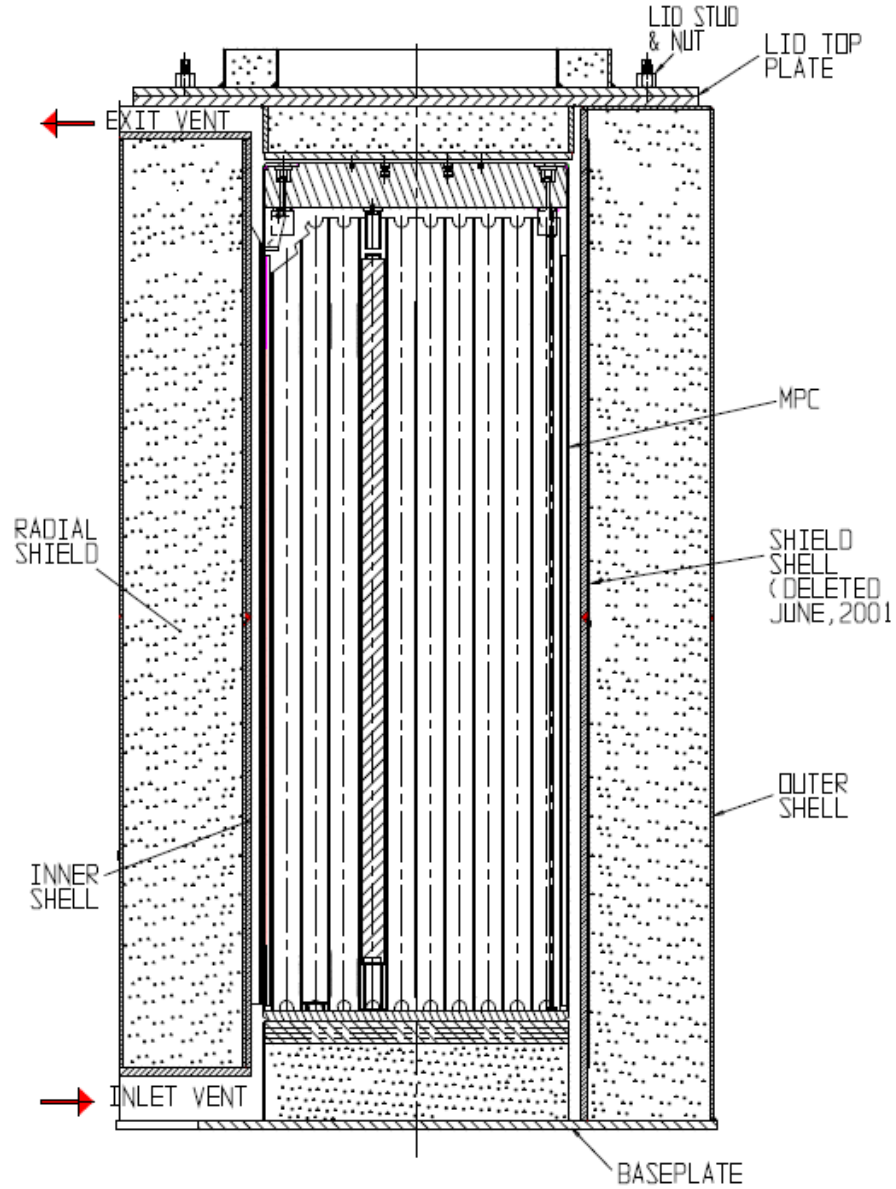
<sup>†</sup> Nominal diameter of the bolt (or rod) as listed in the Code tables. Two-inch diameter studs/rods are specified for the HI-STORM 100A.

Table 1.2.8

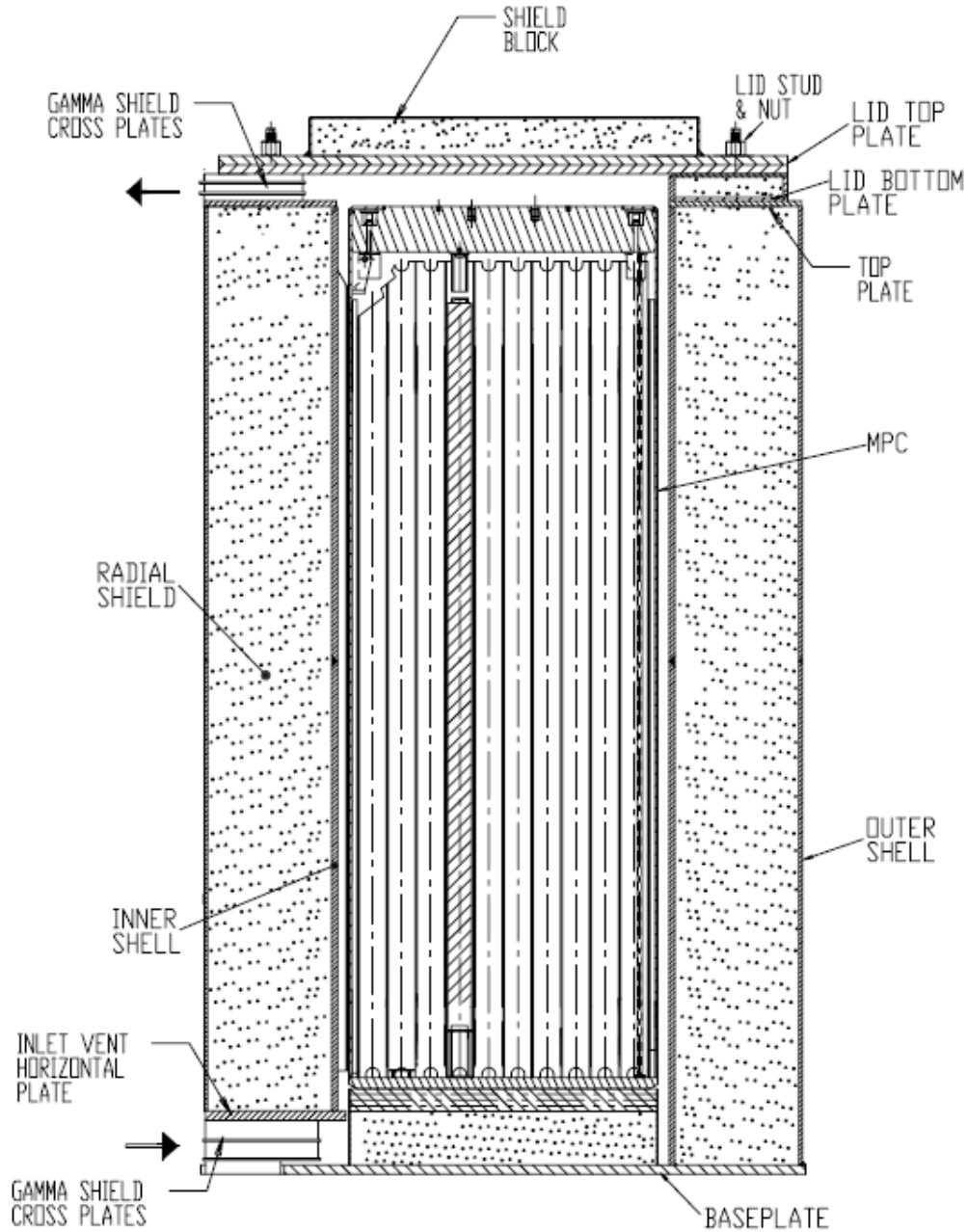
METAMIC<sup>®</sup> DATA FOR HOLTEC MPCs

MPC Type	Min. B-10 areal density required by criticality analysis (g/cm <sup>2</sup> )	Nominal Weight Percent of B <sub>4</sub> C and Reference <i>METAMIC</i> <sup>®</sup> Panel Thickness			
		100% Credit	90% Credit	75% Credit	Ref. Thickness (inch) (see note)
MPC-24	0.020	27.6	31	37.2	0.075
MPC-68, -68FF, -32, -32F, -24E, and -24EF	0.0279	27.8	31	37.4	0.104

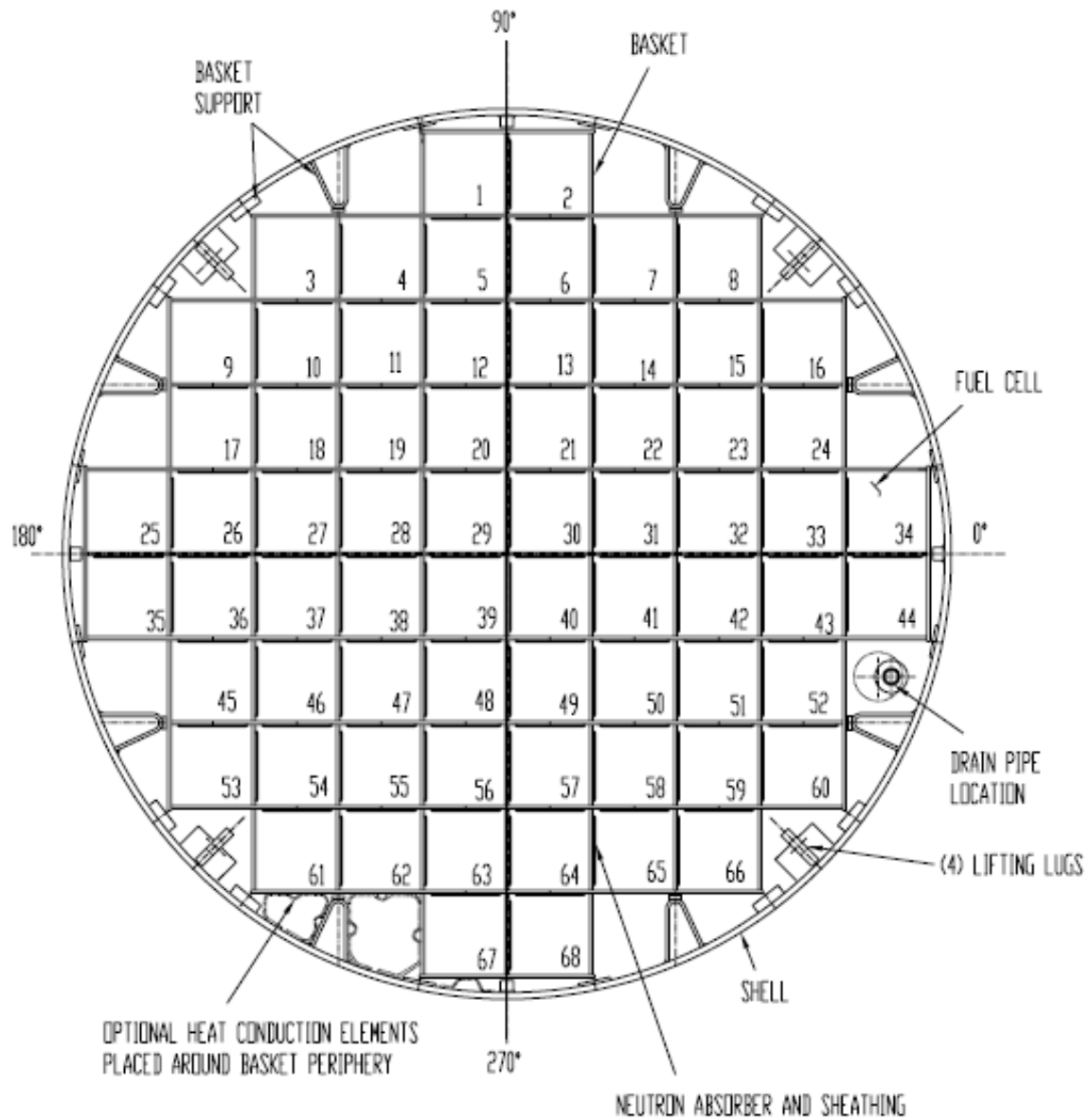
Note: The drawings in Section 1.5 show slightly larger thickness to ensure that the minimum B-10 areal density is conservative under all conditions.



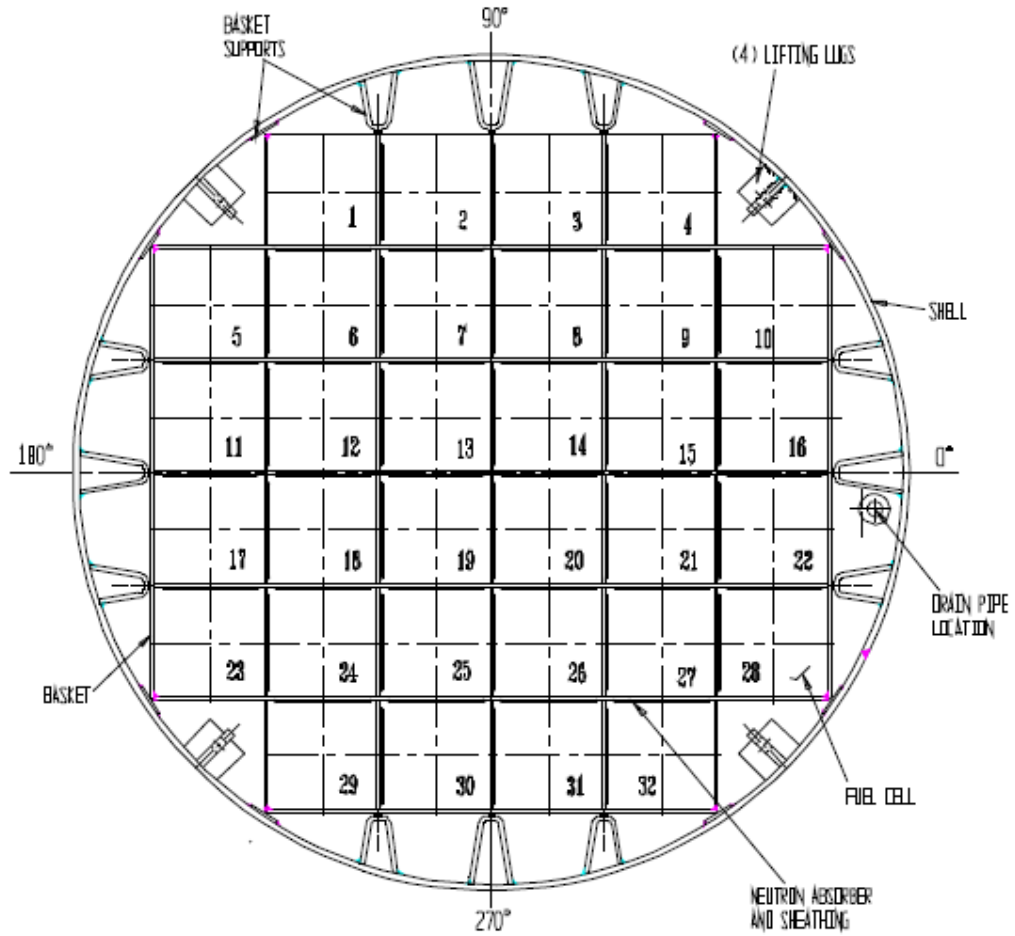
**FIGURE 1.2.1: CROSS SECTION VIEW OF THE HI-STORM 100 SYSTEM**



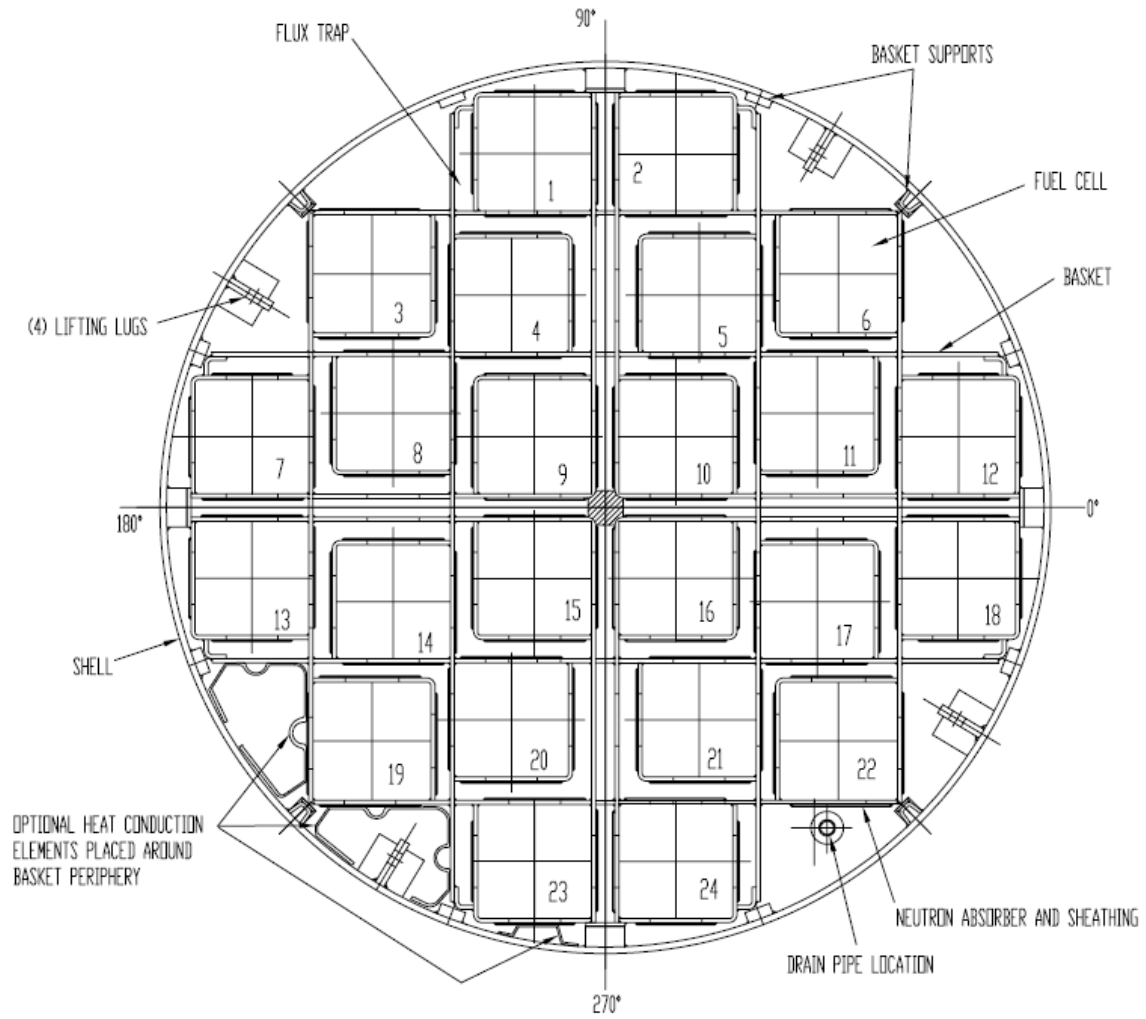
**FIGURE 1.2.1A: CROSS SECTION VIEW OF THE HI-STORM 100S SYSTEM**



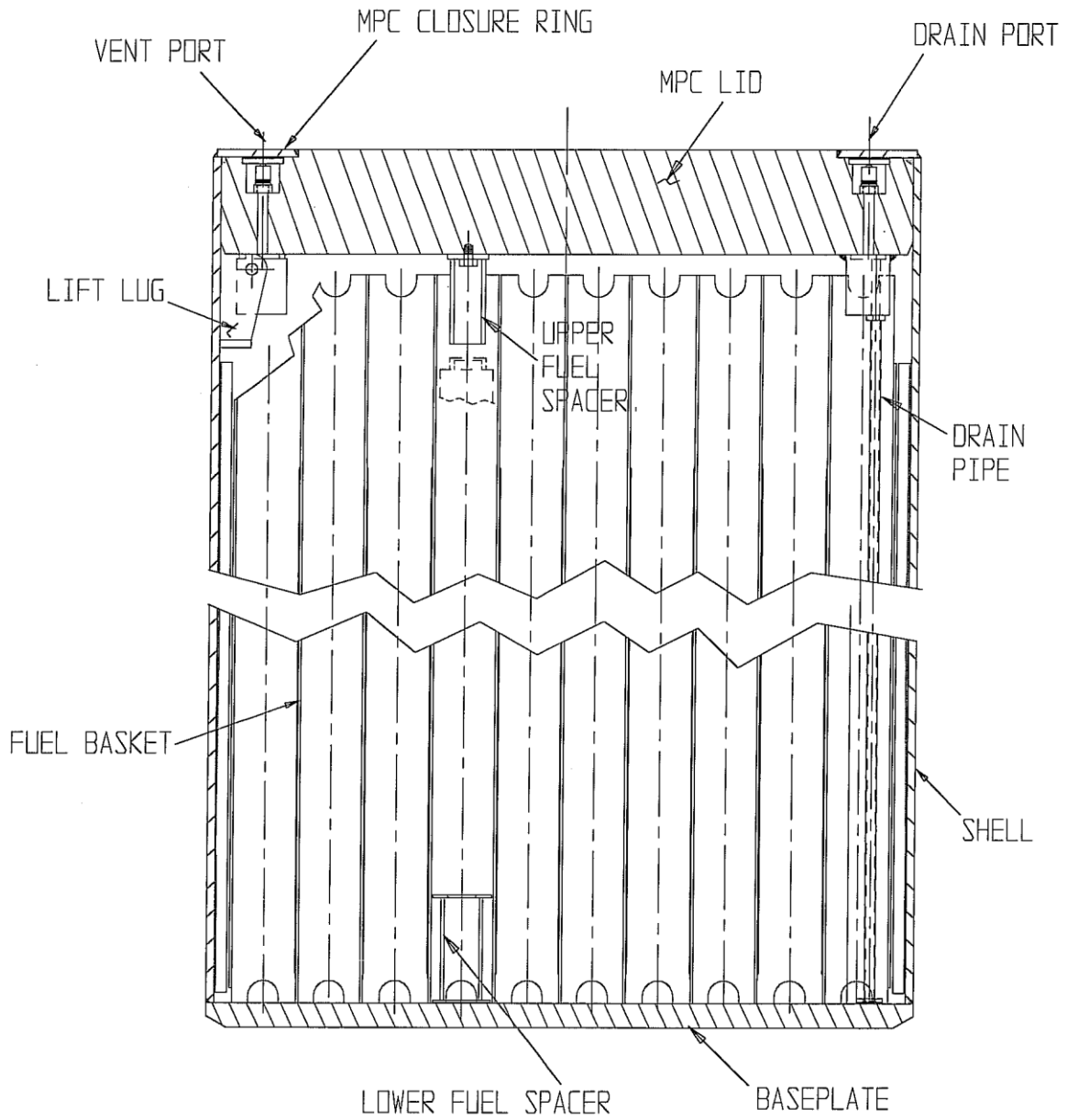
**FIGURE 1.2.2: MPC-68/68F/68FF CROSS SECTION VIEW**



**FIGURE 1.2.3: MPC-32/32F CROSS SECTION**

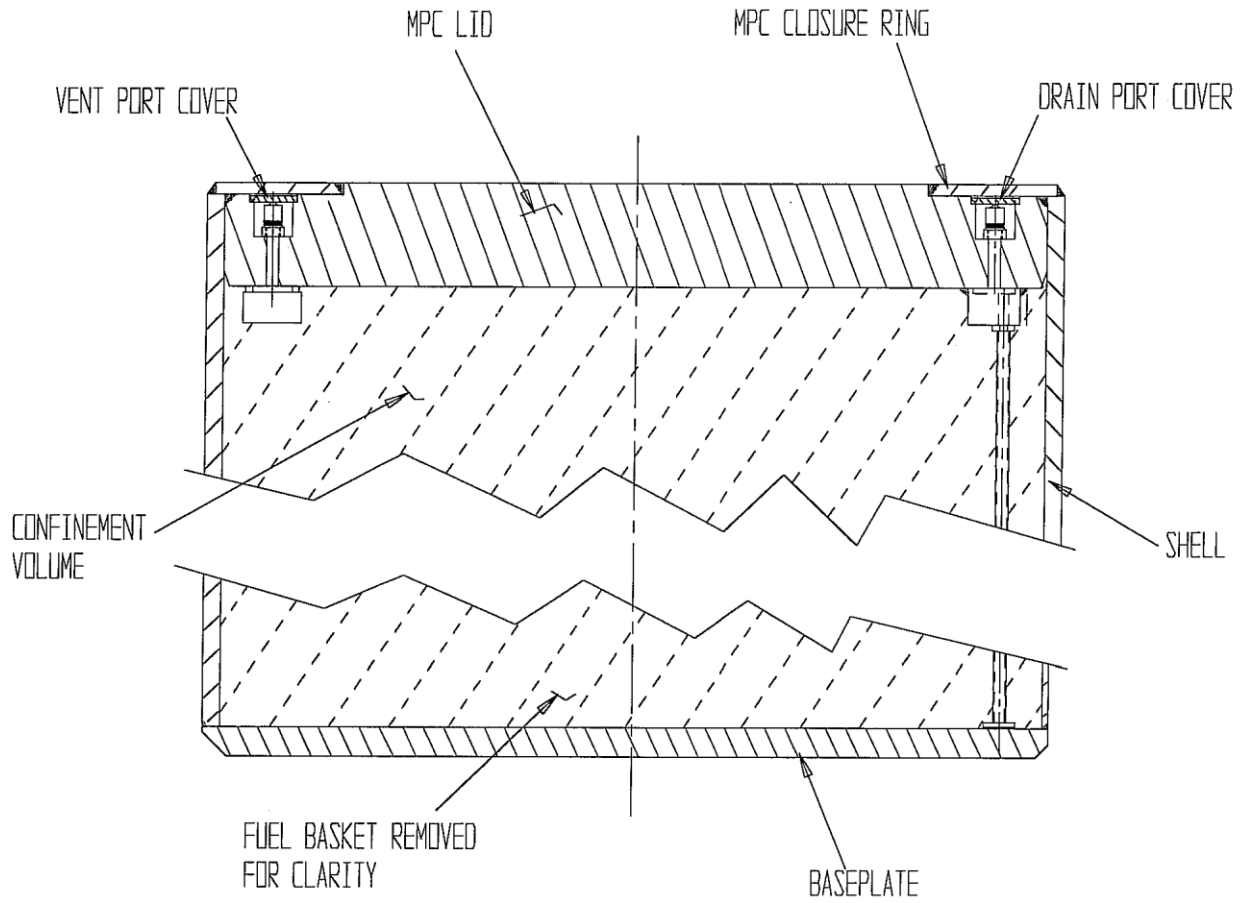


**FIGURE 1.2.4: MPC-24/24E/24EF CROSS SECTION VIEW**

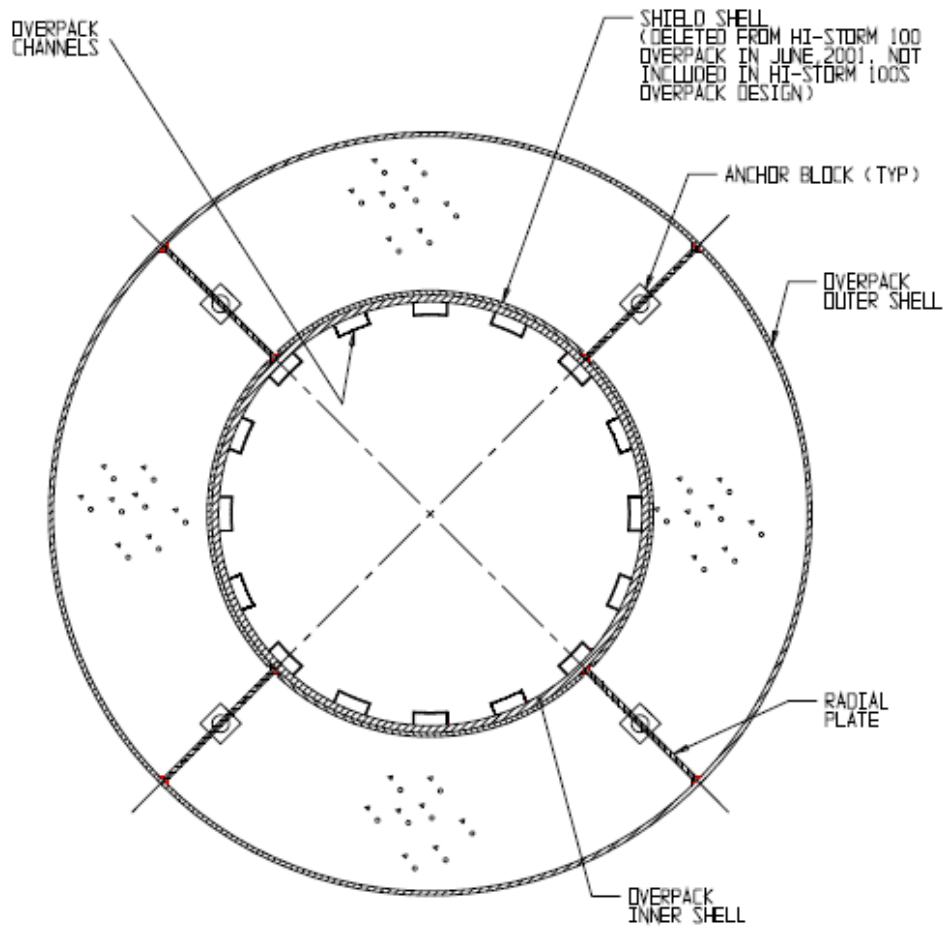


**FIGURE 1.2.5: CROSS SECTION ELEVATION VIEW OF MPC**

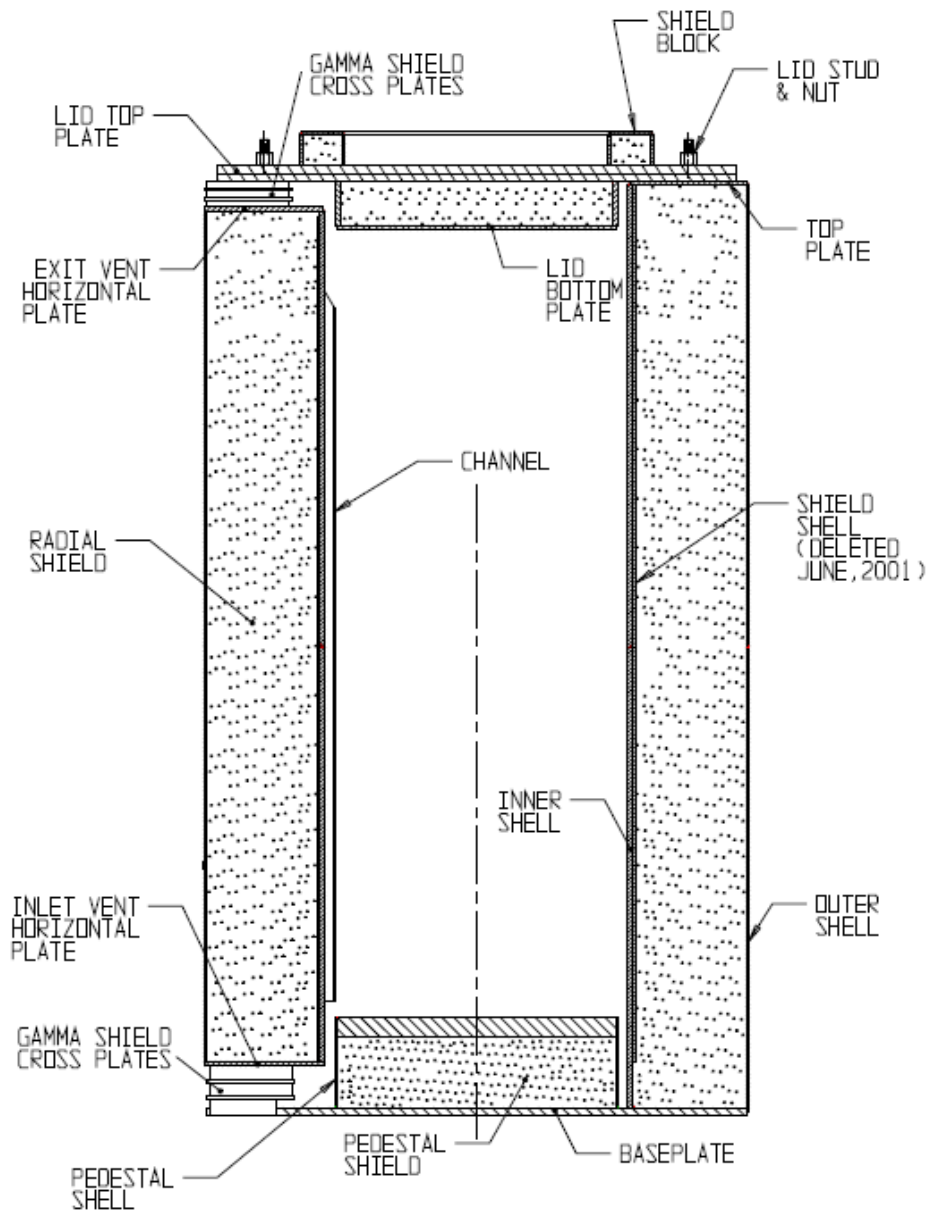




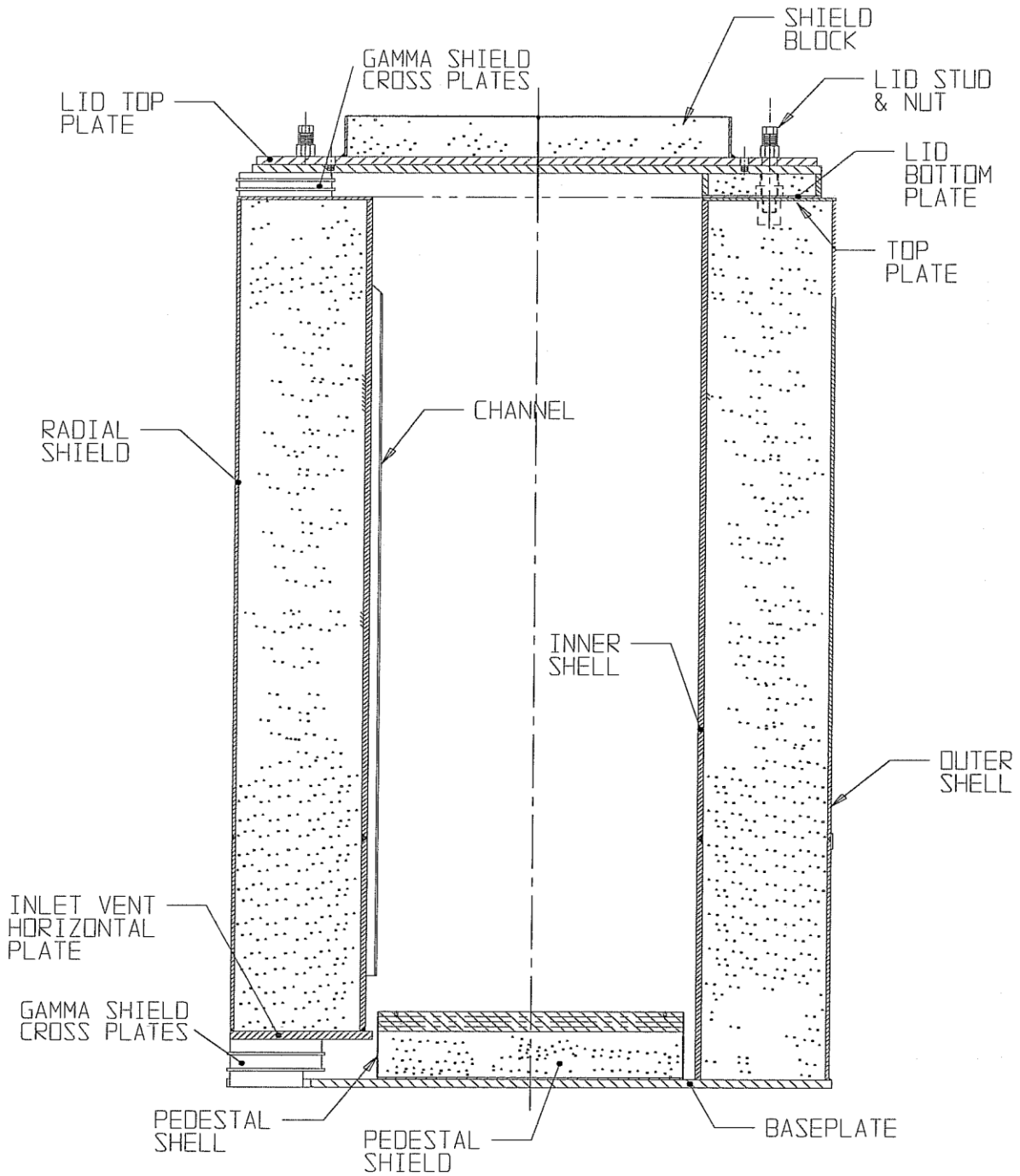
**FIGURE 1.2.6: MPC CONFINEMENT BOUNDARY**



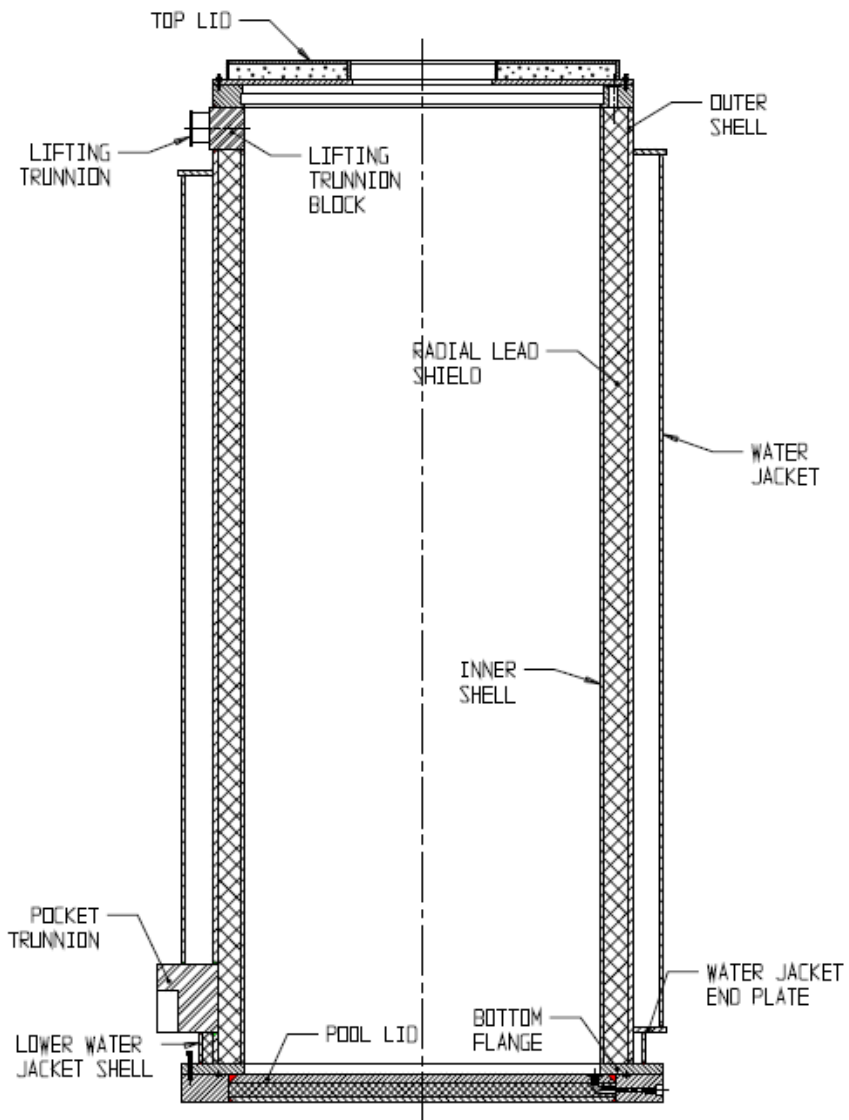
**FIGURE 1.2.7: CROSS SECTION OF HI-STORM OVERPACK**



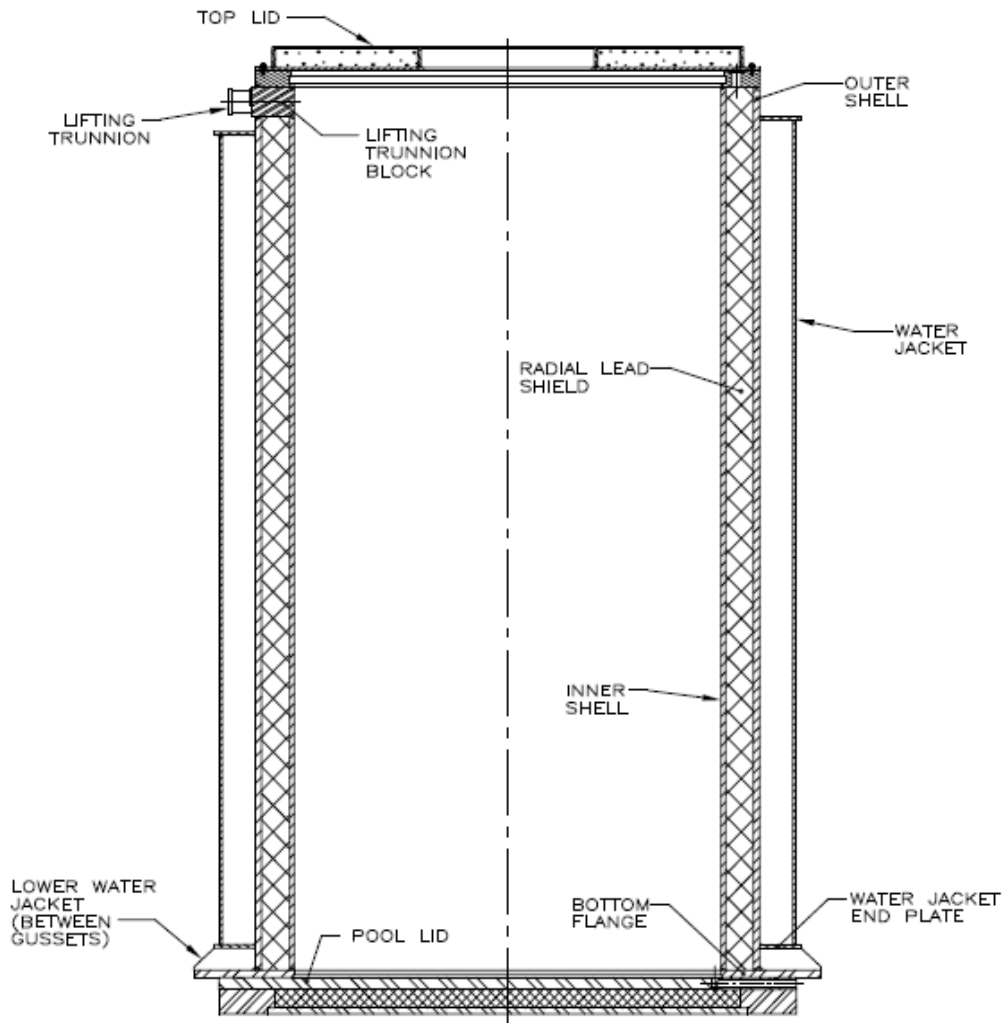
**FIGURE 1.2.8: HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW**



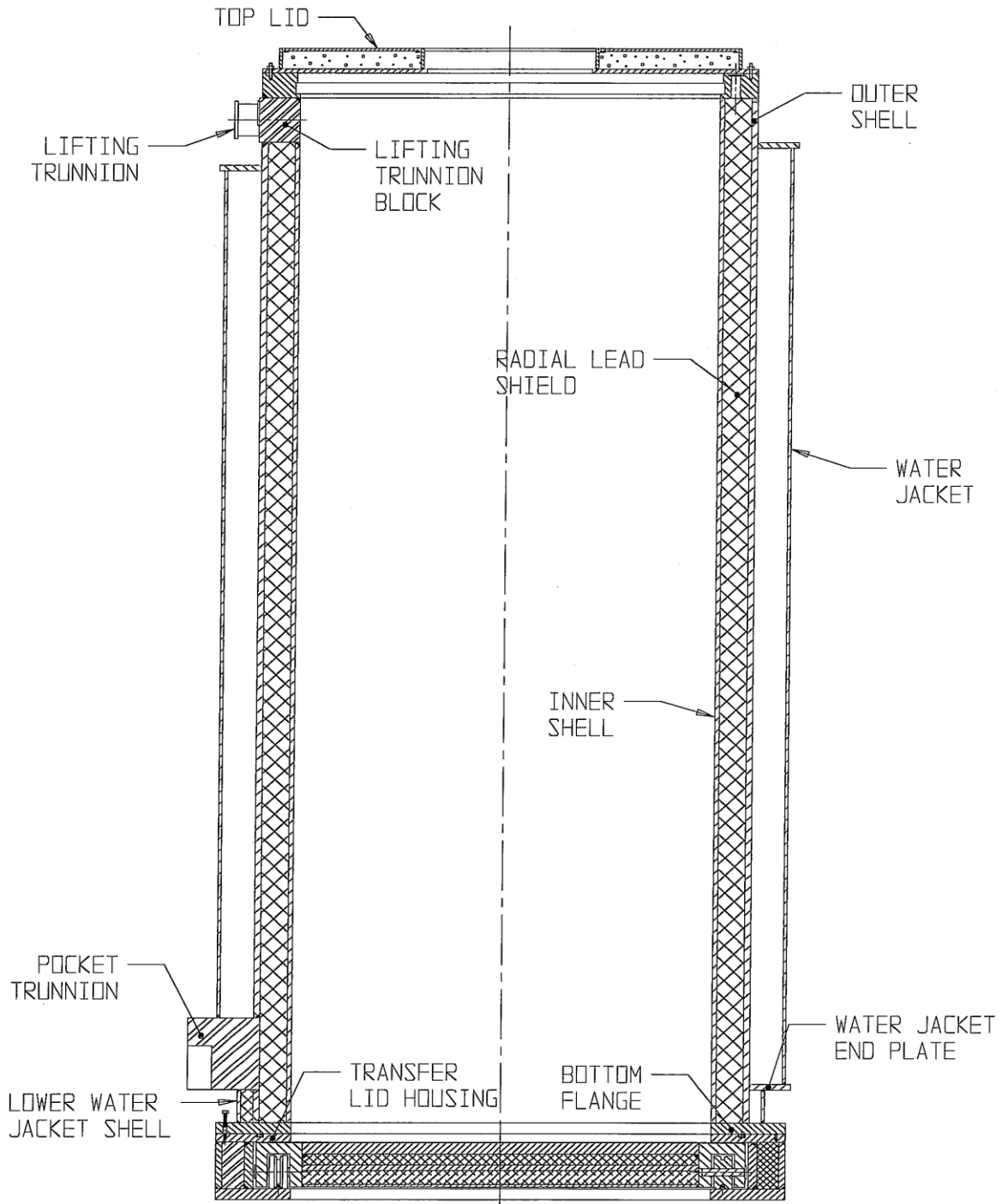
**FIGURE 1.2.8A: HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW**



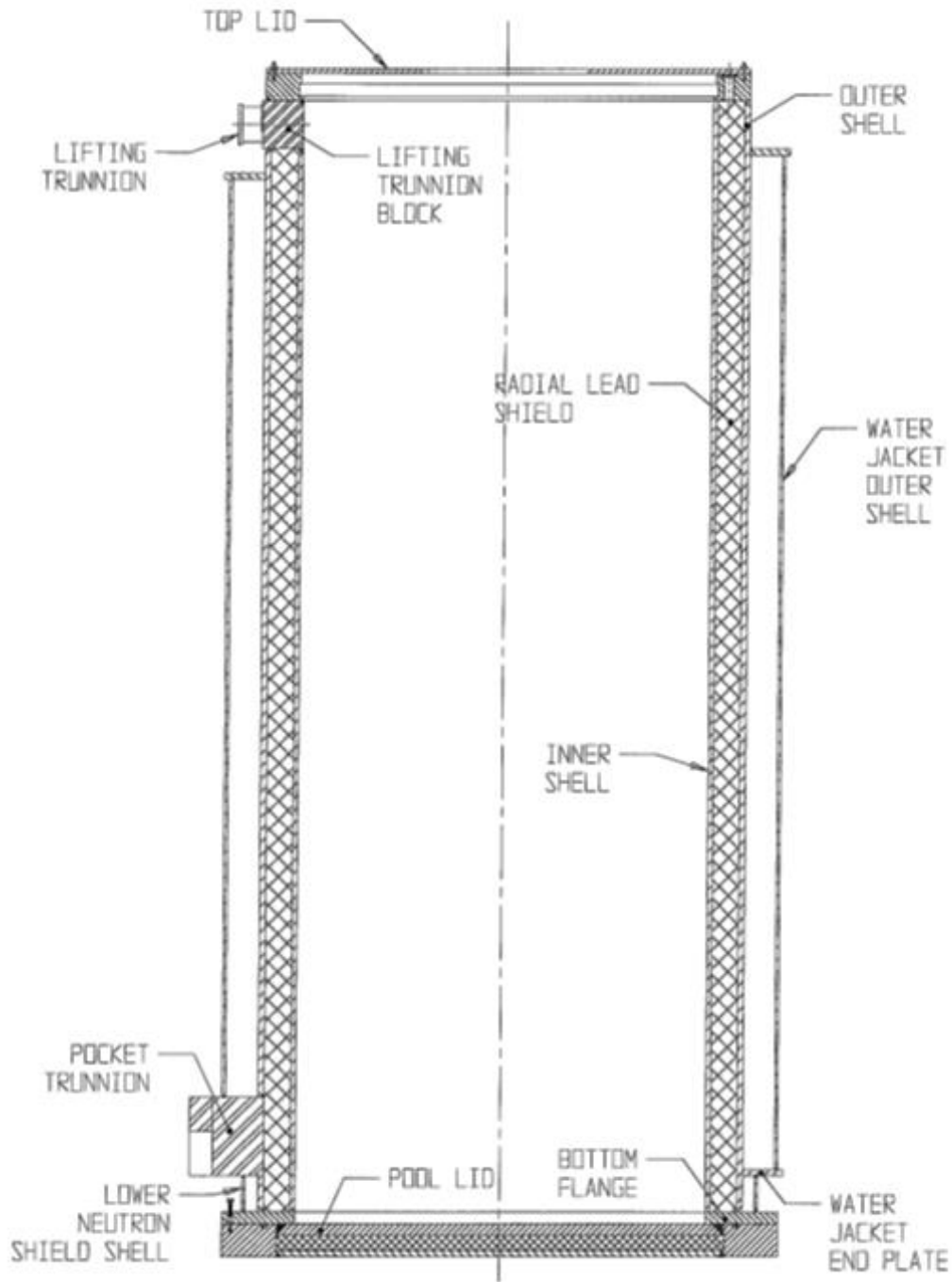
**FIGURE 1.2.9: 125 TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW**



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

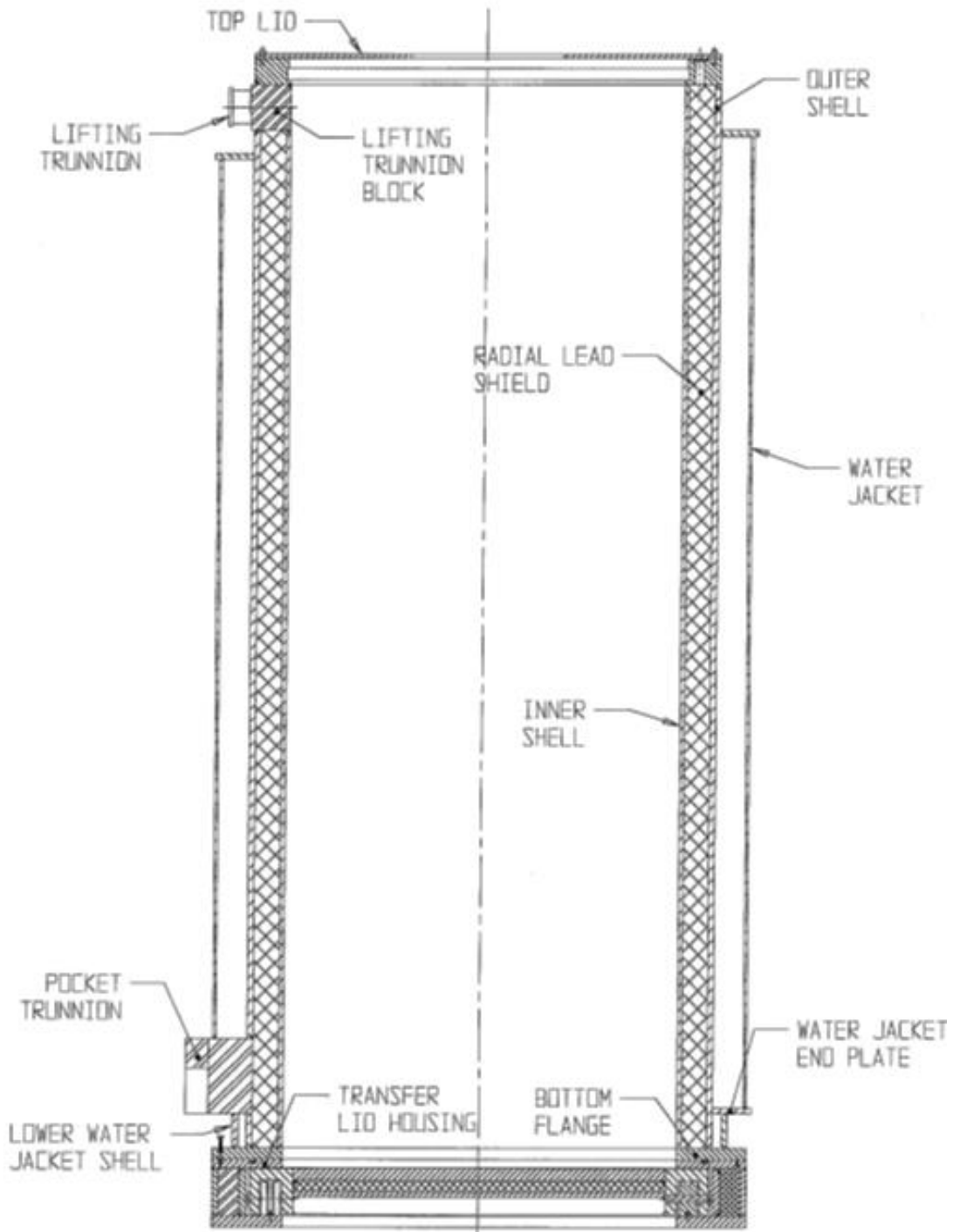


**FIGURE 1.2.10: 125 TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW**



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



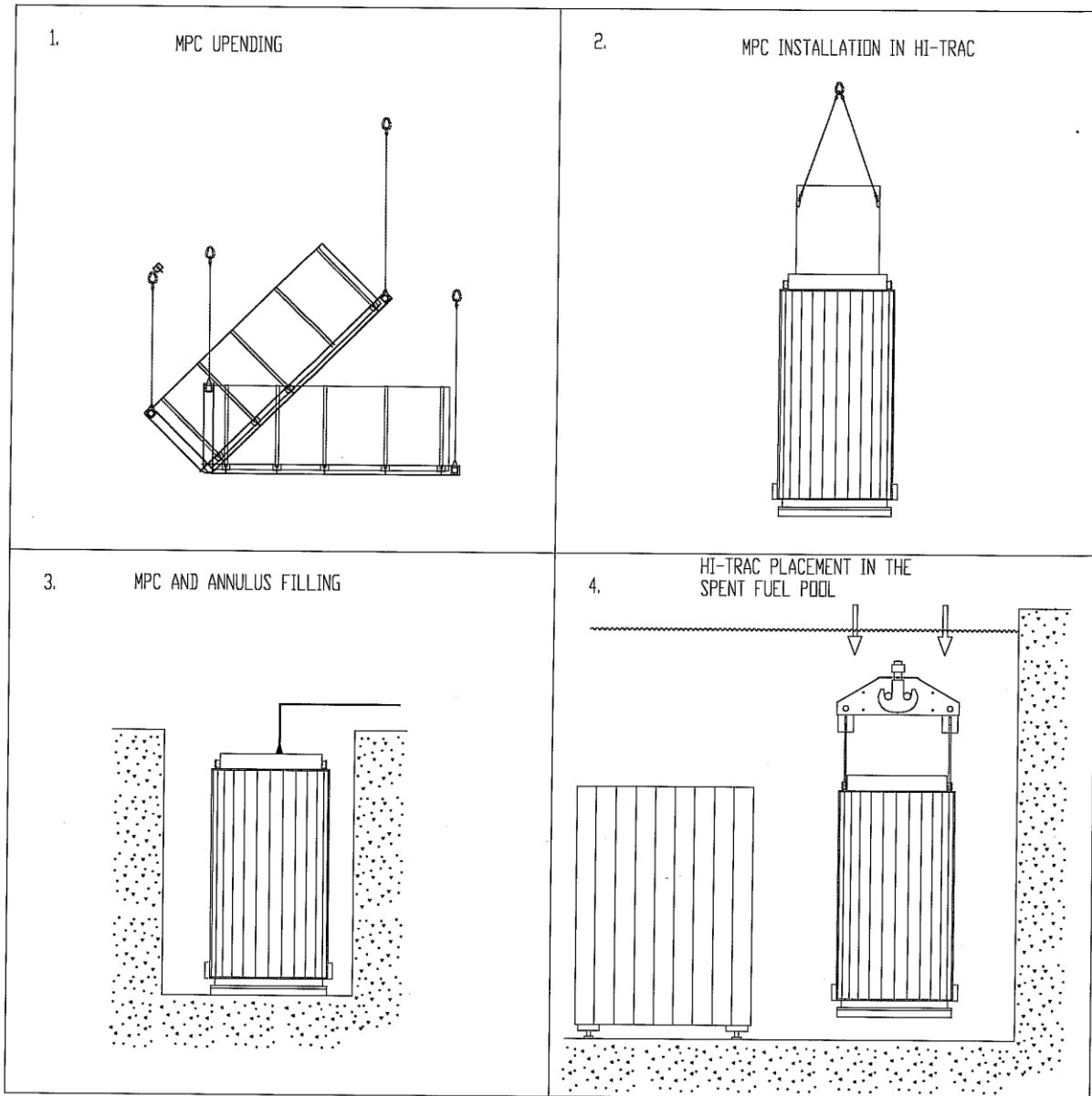


**FIGURE 1.2.12: 100 TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW**

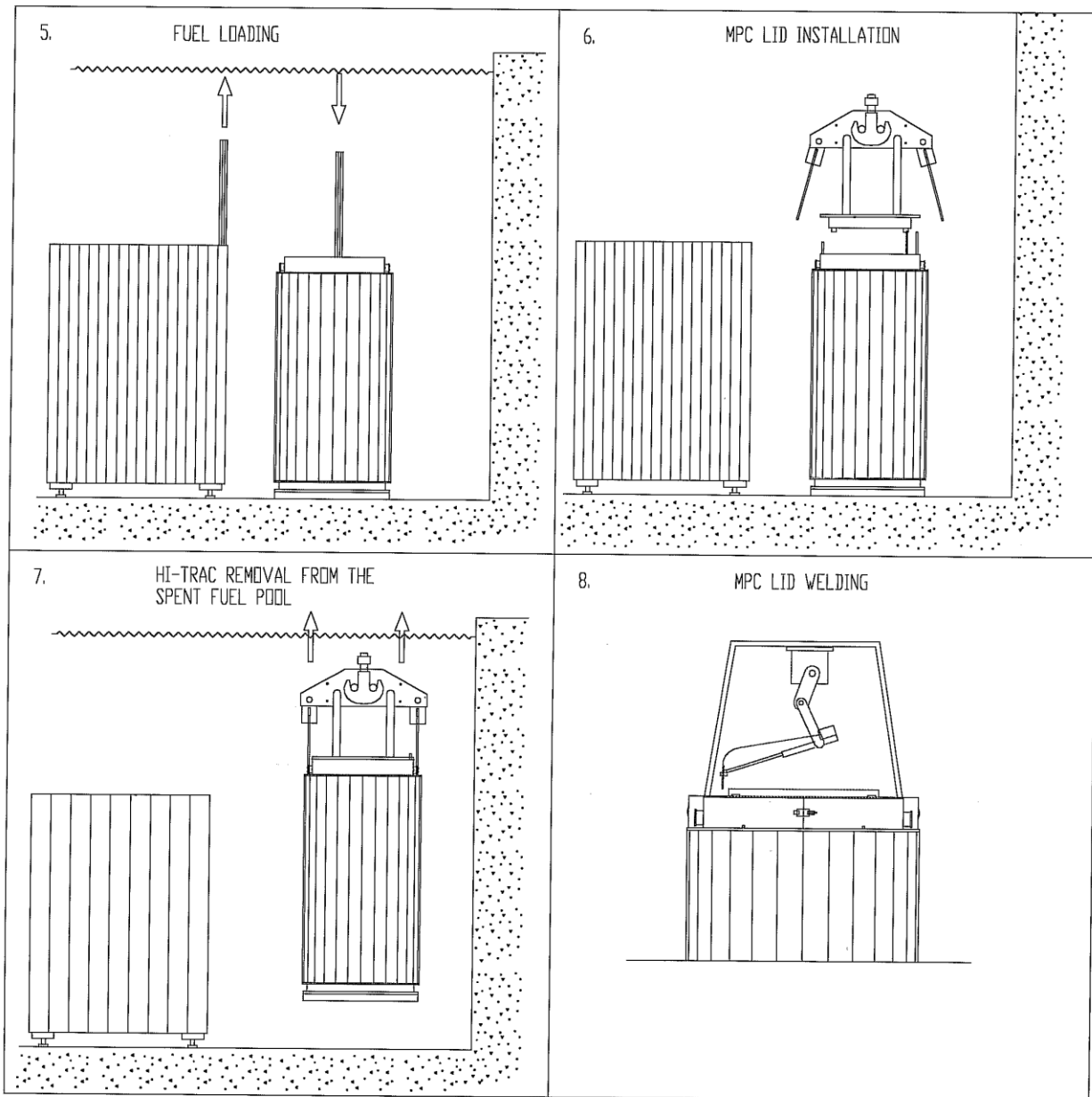
**FIGURE 1.2.13: DELETED**

**FIGURE 1.2.14: DELETED**

**FIGURE 1.2.15: DELETED**



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



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

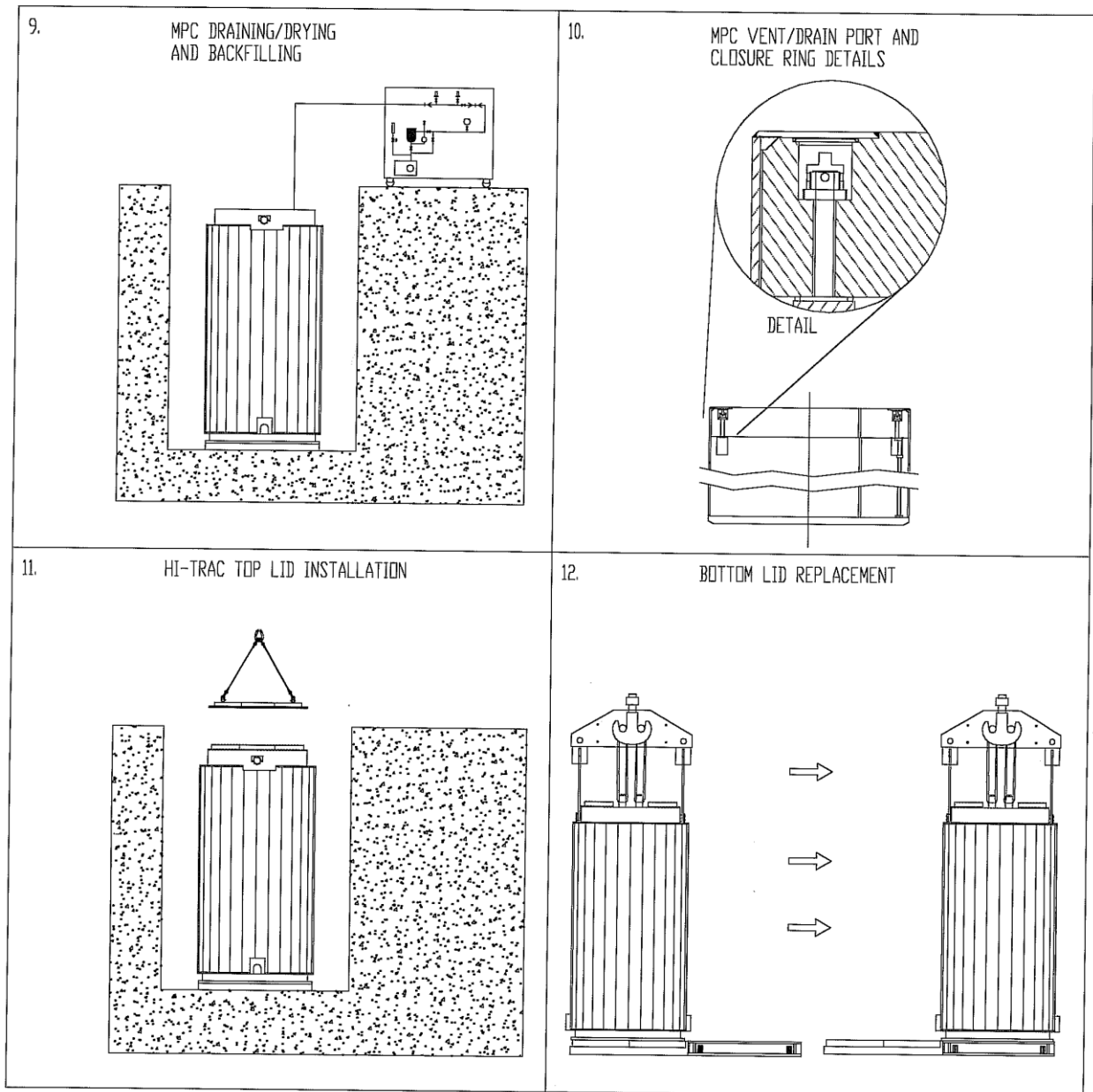
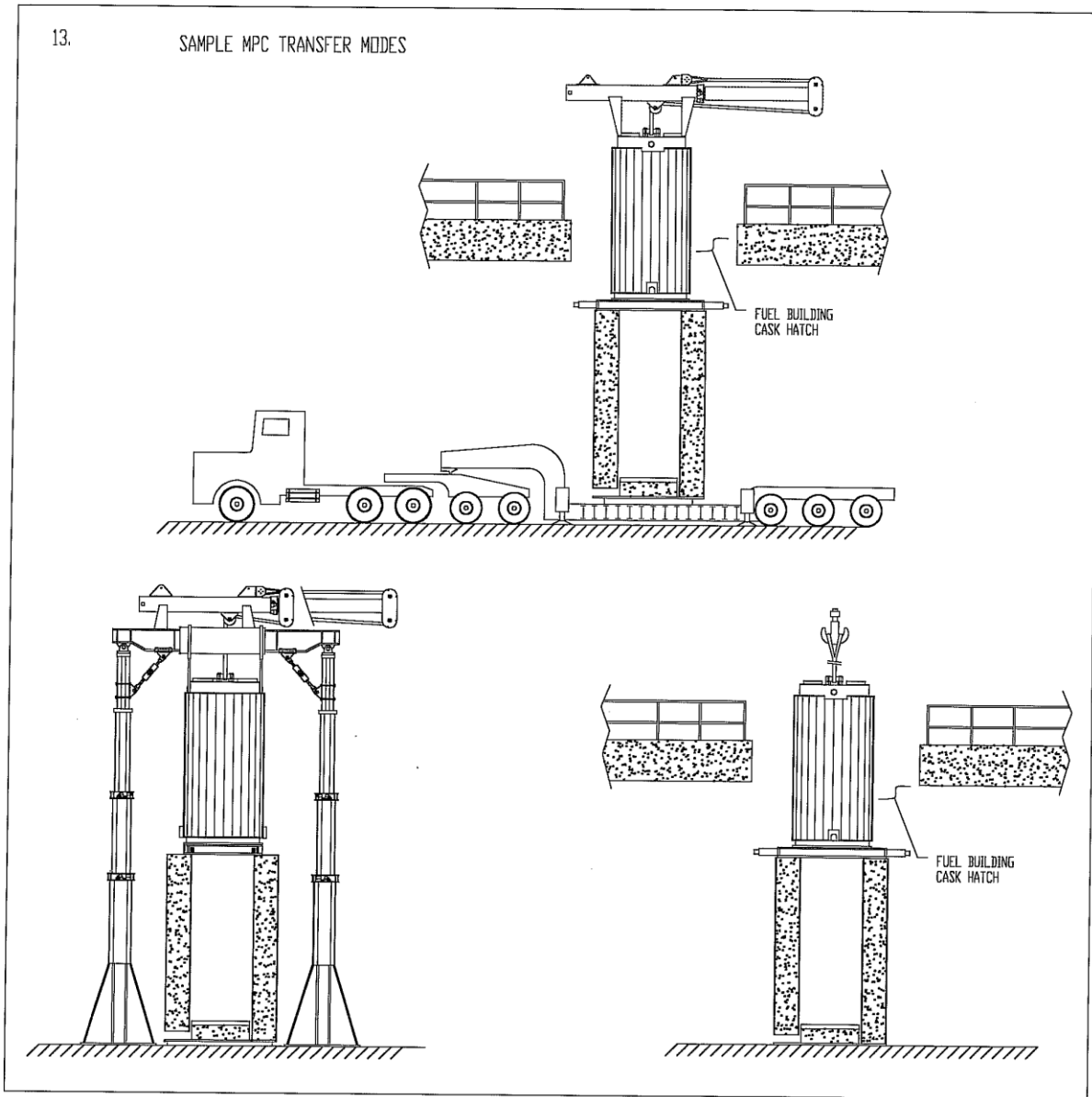
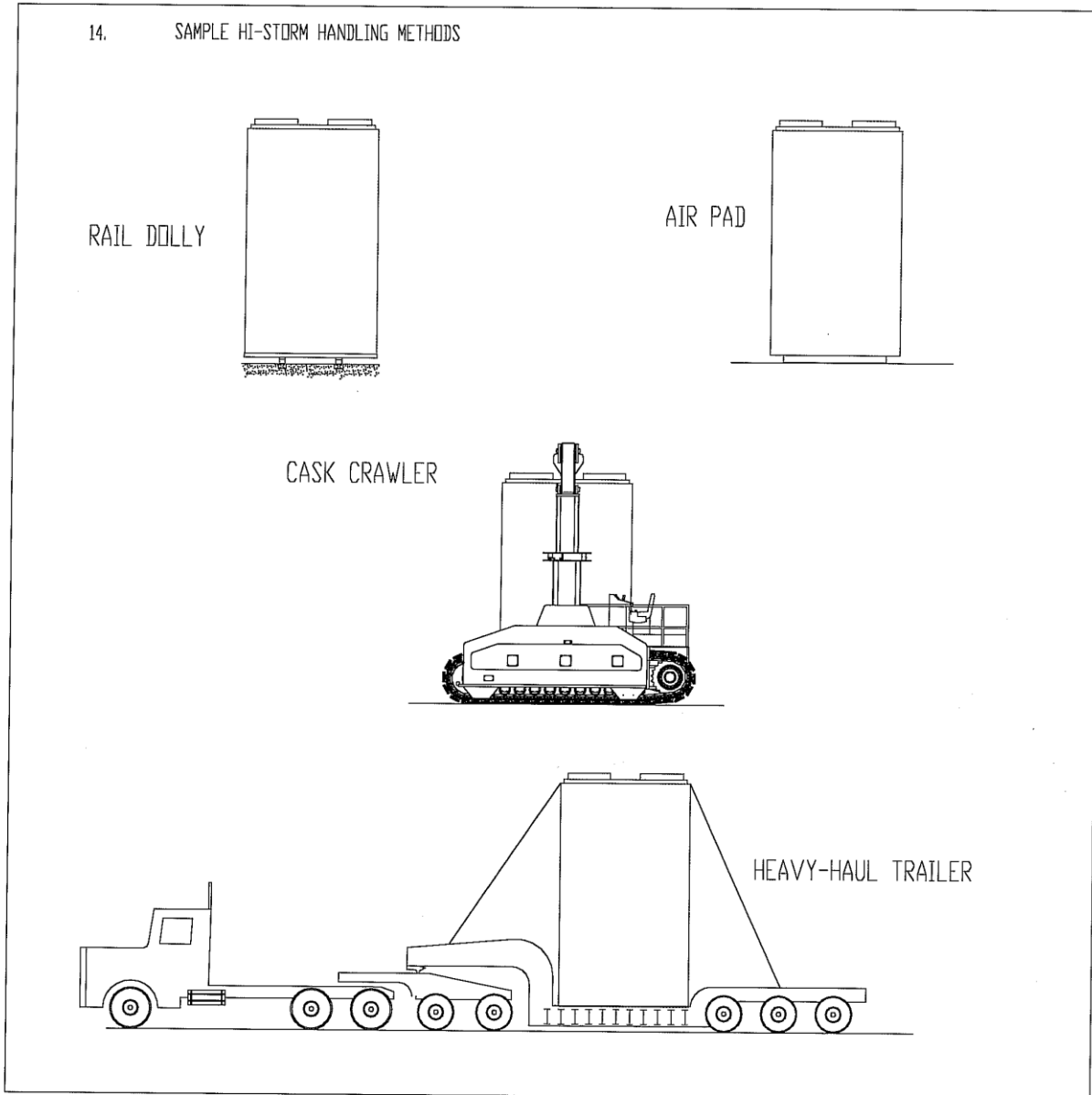


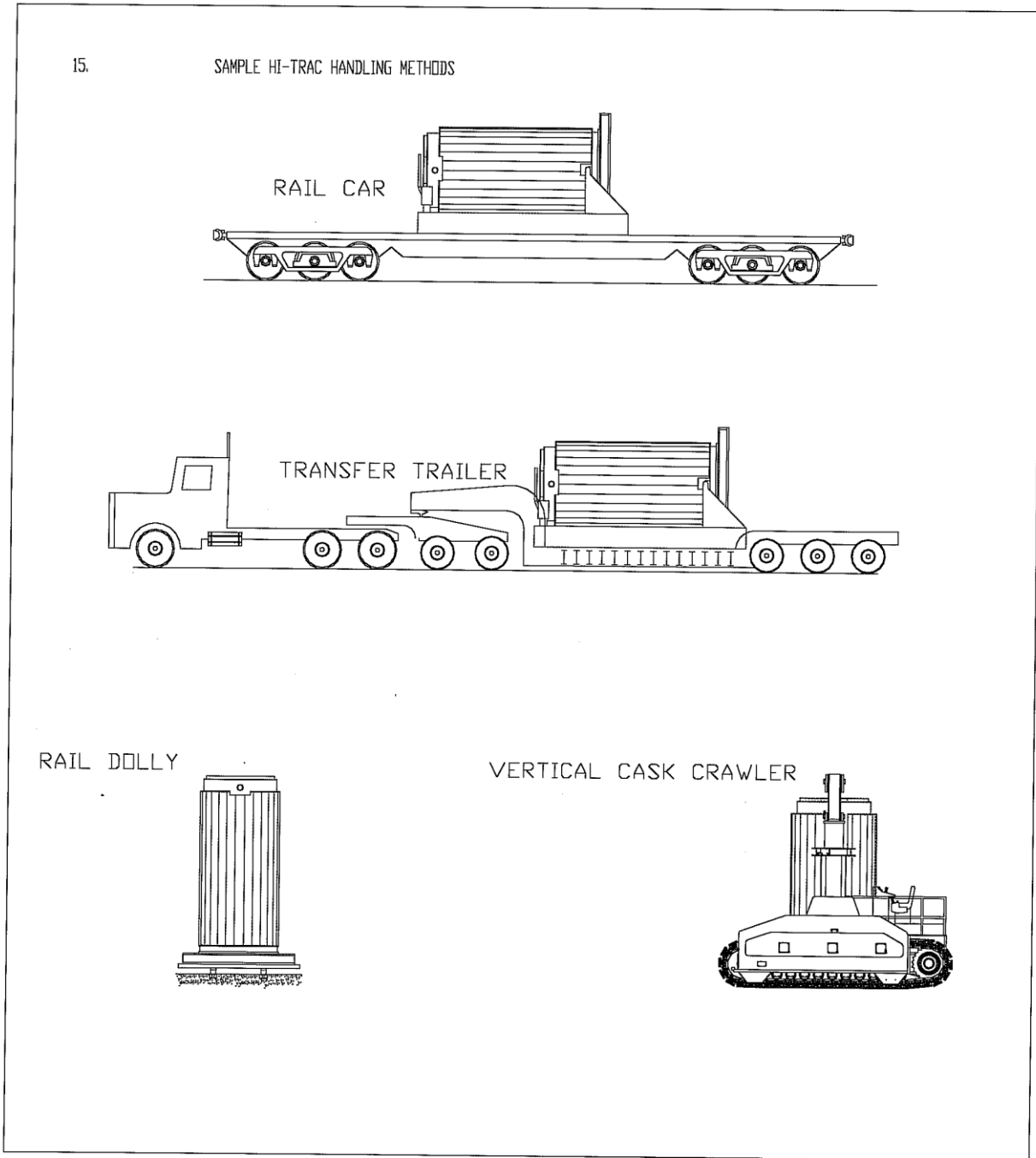
Figure 1.2.16c: Major HI-STORM 100 Loading Operations (Sheet 3 of 6)



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

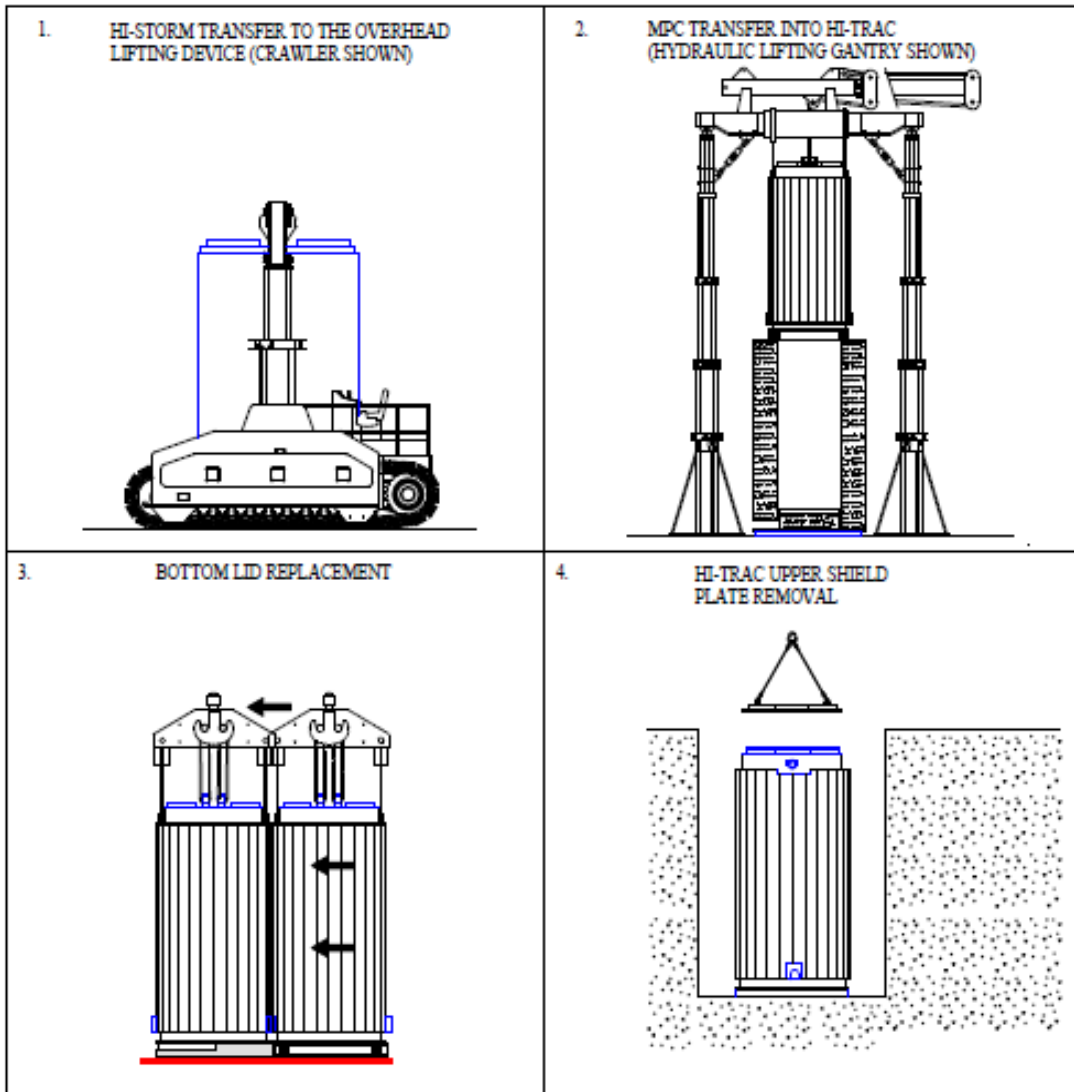


**Figure 1.2.16e: Example of HI-STORM 100 Handling Options (Sheet 5 of 6)**

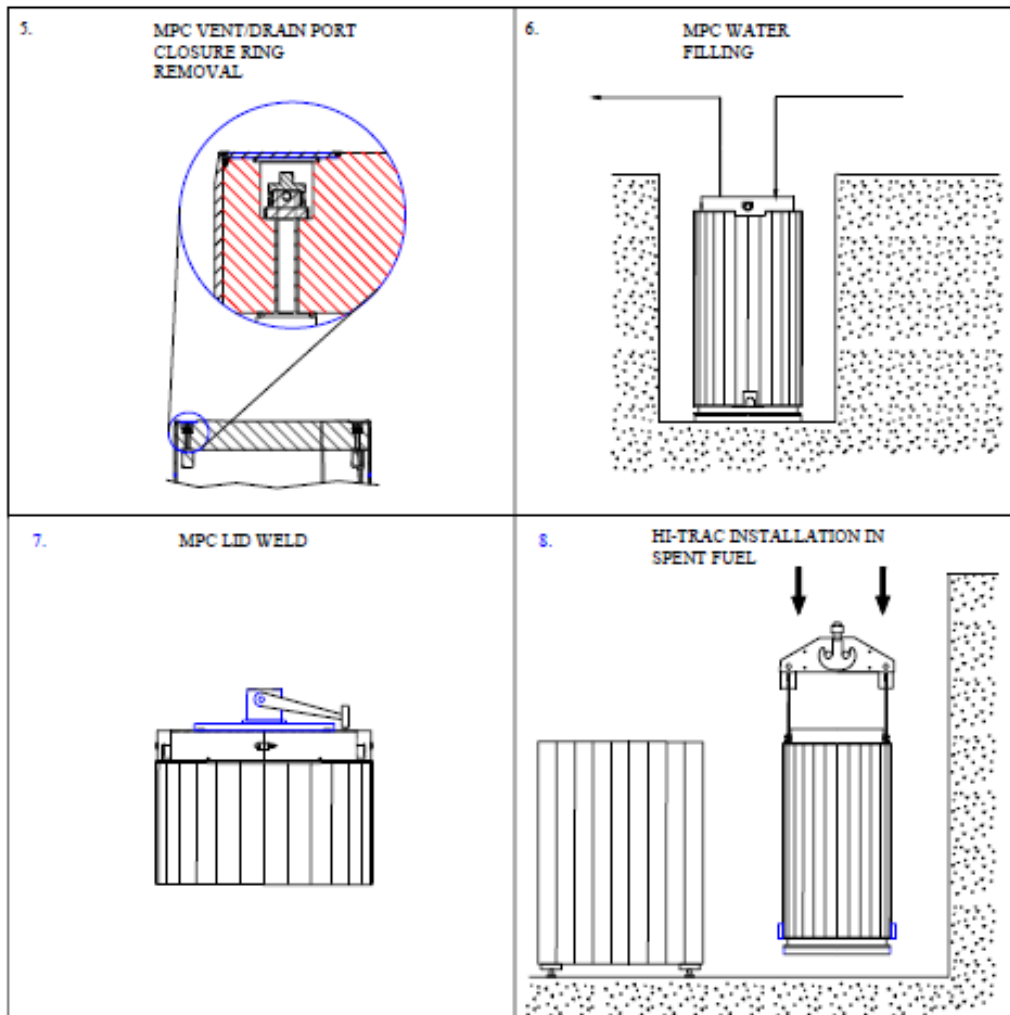


**Figure 1.2.16f: Example of HI-TRAC Handling Options (Sheet 6 of 6)**

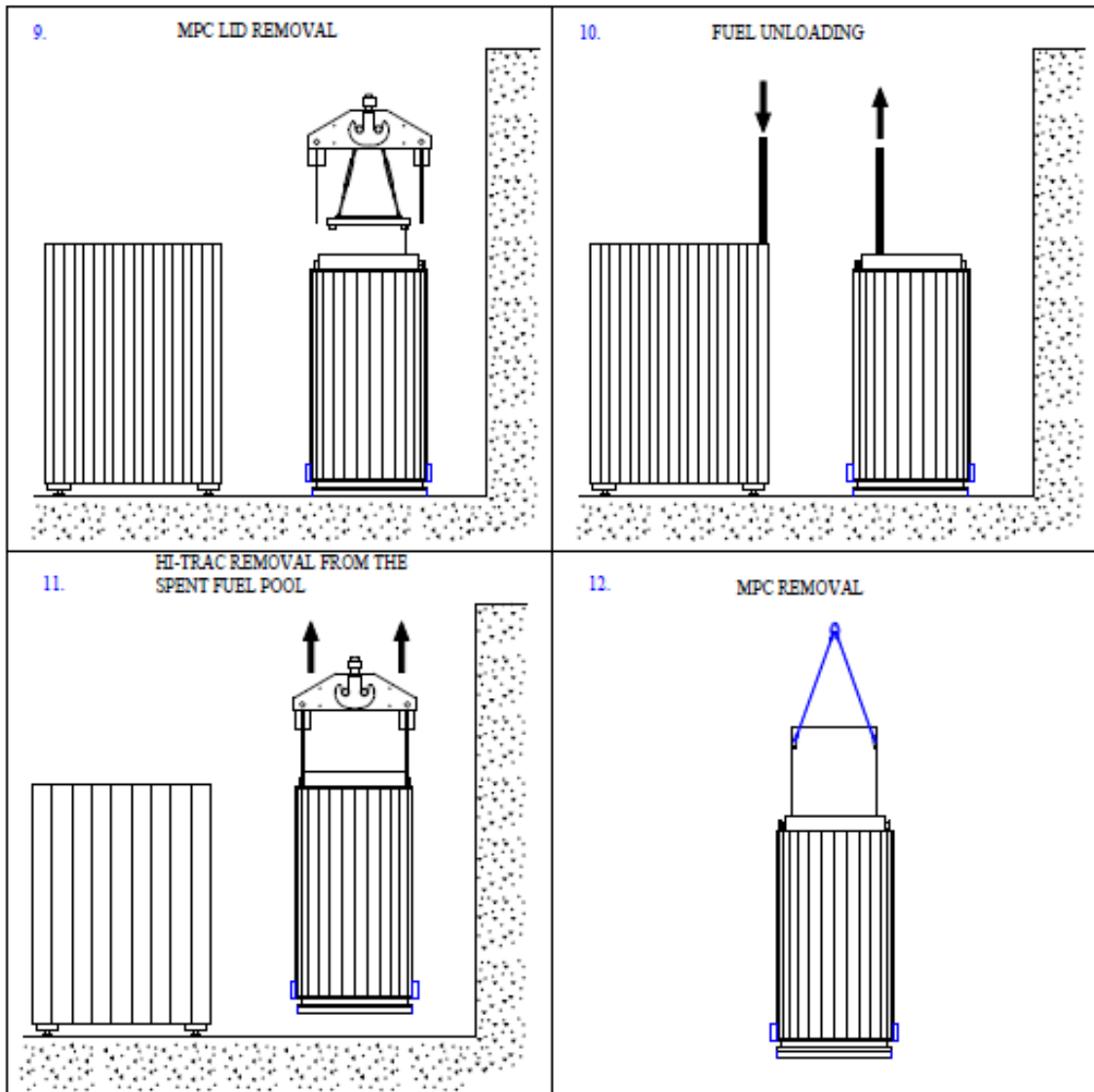




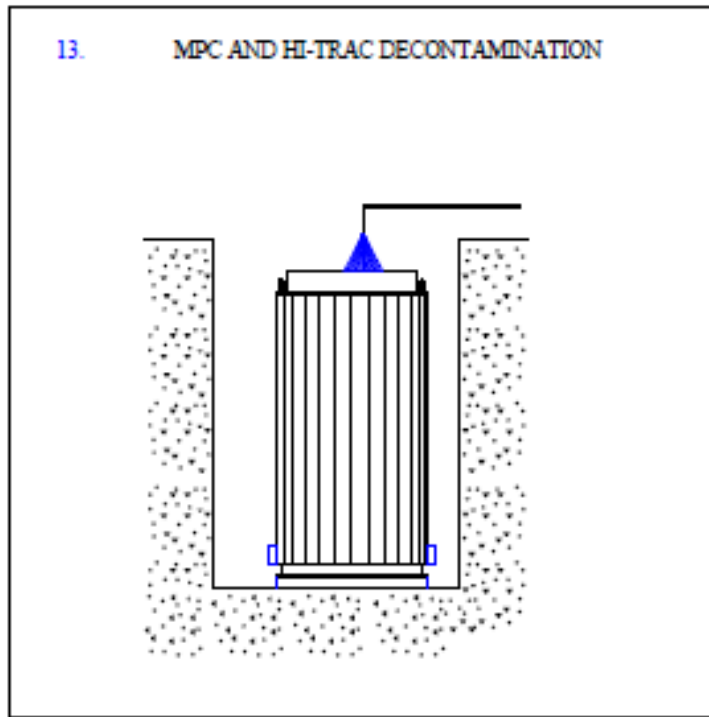
**Figure 1.2.17a: Major HI-STORM 100 Unloading Operations (Sheet 1 of 4)**



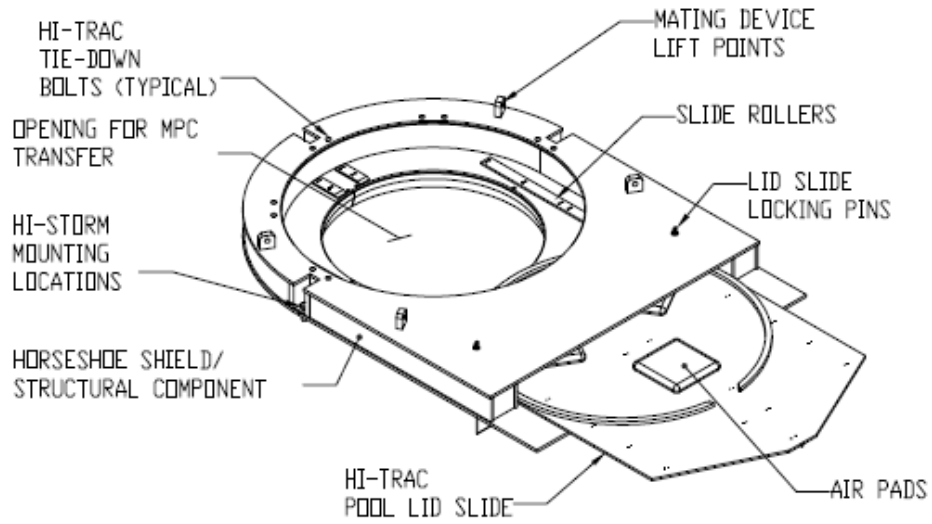
**Figure 1.2.17b: Major HI-STORM 100 Unloading Operations (Sheet 2 of 4)**



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



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



**Figure 1.2.18: HI-STORM MATING DEVICE**

### 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, based in Marlton, NJ and Jupiter Florida is the system designer and applicant for certification of the HI-STORM 100 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 90 plants in the U.S., Britain, Brazil, Korea, Mexico, China and Taiwan have utilized the Company's wet storage technology to establish their state-of-the-art in-pool storage capacities.

Holtec's NPD is also a turnkey provider of dry storage and transportation technologies to nuclear plants around the globe. The company is contracted by over 50 nuclear plants in the U.S. to provide the company's vertical ventilated dry storage technology. Utilities in Belgium, China, Korea, Mexico, Spain, Sweden, Ukraine, the United Kingdom 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 the HI-STORM UMAX and HI-STORM FW systems) 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. October 2015) are listed in Table 1.3.2.

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 System will be fabricated by the Holtec International Manufacturing Division (HMD) located in Pittsburgh, Pennsylvania with subcontract services from the NPD- affiliated plant, Orrvilon in Orrville, Ohio. HMD is a long-term ASME 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 U.S. and overseas nuclear plants. Holtec International's engineering organization, based in Marlton, NJ and the HMD subsidiary in Pittsburgh, PA 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 100 System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program approved by the USNRC.

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, Waterford 3, Pilgrim and Comanche Peak have deployed dry storage at their sites using a turnkey contract with Holtec International.

Table 1.3.1

## USNRC DOCKETS ASSIGNED TO HOLTEC INTERNATIONAL

<b>System Name</b>	<b>Docket Number</b>
HI-STORM 100 (Storage)	72-1014
HI-STAR 100 (Storage)	72-1008
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 60 (Transportation)	71-9336
Holtec Quality Assurance Program	71-0784
HI-STORM FW (Storage)	72-1032
HI-STORM UMAX	72-1040
HI-STAR 80 (Transportation)	71-9374
HI-STAR 330 (ATB-1T) (Transportation)	71-9375



Item No.	Colloquial Name of the Patent	USPTO Patent Number
1.	Honeycomb Fuel Basket	5,898,747
2.	Radiation Absorbing Refractory Composition (METAMIC)	5,965,829
3.	HI-STORM 100S Overpack	6,064,710
4.	Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super Hypereutectic A1/S1(METAMIC-CLASSIC)	6,042,779
5.	Duct Photon Attenuator	6,519,307B1
6.	HI-TRAC Operation	6,587,536B1
7.	Cask Mating Device (Hermetically Sealable Transfer Cask)	6,625,246B1
8.	Improved Ventilator Overpack	6,718,000B2
9.	Below Grade Transfer Facility	6,793,450B2
10.	HERMIT (Seismic Cask Stabilization Device)	6,848,223B2
11.	Cask Mating Device ( operation)	6,853,697
12.	Davit Crane	6,957,942B2
13.	Duct-Fed Underground HI-STORM	7,068,748B2
14.	Forced Helium Dehydrator (design)	7,096,600B2
15.	Below Grade Cask Transfer Facility	7,139,358B2
16.	Forced Gas Flow Canister Dehydration (alternate embodiment)	7,210,247B2
17.	HI-TRAC Operation (Maximizing Radiation Shielding During Cask Transfer Procedures)	7,330,525
18.	HI-STORM 100U	7,330,526B2
19.	Flood Resistant HI-STORM	7,590,213B1
20.	HI-STORM 100M (Underground Manifoldded module assembly)	7,676,016B2
21.	Dew Point Temperature Based Canister Dehydration	7,707,741B2
22.	Optimized Weight Transfer Cask with Detachable Shielding	7,786,456B2
23.	VESCAP (Apparatus, System, and Method for Facilitating Transfer of High Level Radioactive Waste to and/or From a Pool)	7,820,870B2
24.	HI-STORM 100F (Counter-flow Underground Vertical Ventilated Module)	7,933,374B2
25.	Apparatus for Transporting and/or Storing Radioactive Materials Having Jacket Adapted to Facilitate Thermo-siphon Fluid Flow	7,994,380B2

Table 1.3.2 (Cont'd)		
DRY STORAGE AND TRANSPORT PATENTS HELD BY HOLTEC INTERNATIONAL		
Item No.	Colloquial Name of the Patent	USPTO Patent Number
26.	Method of Removing Radioactive Materials from Submerged State and/or Preparing Spent Nuclear Fuel for Dry Storage	8,067,659B2
27.	HI-STORM 100US	8,098,790B1
28.	Canister Apparatus and Basket for Transporting, Storing and/or Supporting Spent Nuclear Fuel(Double Wall Canister)	8,135,107B2
29.	Method of Storing High Level Waste (HI-STORM 100F)	8351562B2
30.	Apparatus for Providing Additional Radiation Shielding to a Container Holding Radioactive Materials, and Method of Using the Same to Handle and/or Process Radioactive Materials	8415521B2
31.	Spent Fuel Basket, Apparatus and Method Using the Same for Storing High Level Radioactive Waste	8548112B2
32.	System and Method for Preparing a Container Loaded with Wet Radioactive Elements for Dry Storage	8561318B2
33.	Systems And Methods For Storing Spent Nuclear Fuel(100U)	8625732
34.	System and method for the ventilated storage of high level radioactive waste in a clustered arrangement	8660230B2
35.	Fuel Basket Spacer Apparatus and Method Using the Same for Storing High Level Radioactive Waste(HI-STAR 180)	8712001B2
36.	Manifold system for the ventilated storage of high level waste and a method of using the same to store high level waste in a below-grade environment	8718220B2
37.	Method and Apparatus for Preparing Spent Nuclear Fuel for Dry Storage	8737559B2
38.	Apparatus for Storing and/or Transporting High Level Radioactive Waste and Method for Manufacturing the Same	8798224B2
39.	Method and Apparatus for Preparing Spent Nuclear Fuel for Dry Storage	20140317952A1
40.	Ventilated System for Storing High Level Nuclear Waste	8905259B2
41.	Method for Controlling Temperature of a Portion of a Radioactive Waste Storage System and for Implementing the Same	9105365B2

<b>Table 1.3.3 HOLTEC INTERNATIONAL PENDING PATENTS ON FUEL STORAGE</b>			
	<b>Title</b>	<b>Submittal Date</b>	<b>Patent File No.</b>
1.	System and Method for the Ventilated Storage of High Level Radioactive Waste in a Clustered Arrangement	22-Dec-08	US20090159550
2.	System and Method for Storing Spent Nuclear Fuel Having Manifoldd Underground Vertical Ventilated Module	19-Feb-10	US20100150297
3.	Cask Apparatus, System and Method for Transporting and/or Storing High Level Waste	28-Apr-10	US20100272225
4.	Container System for Radioactive Waste	15-Apr-13	W02013155520
5.	System and Method of Storing and/or Transferring High Level Radioactive Waste	18-Apr-13	W02013158914
6.	Container and System for Handling Damaged Nuclear Fuel, and Method of Making the Same	19-Feb-14	W0201355445
7.	Storage System for Nuclear Fuel	24-Jun-14	W02013096966
8.	A Radioactive Material Storage Canister and Method for Sealing Same	03-Jul-14	W02014105977
9.	Method of Storing High Level Radioactive Waste	07-Jul-14	US20140192946A1
10.	System and Method for Storing and Leak Testing a Radioactive Materials Storage Canister	26-Feb-15	W02014036561
11.	Neutron Shielding Ring Apparatus and Method Using the Same for Storing High Level Radioactive Waste	08-Aug-15	US20080084958

## 1.4 GENERIC CASK ARRAYS

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

Table 1.4.1 provides recommended cask spacing data for array(s) of two by N casks. The pitch between adjacent rows of casks and between each adjacent column of casks are denoted by  $P_1$  and  $P_2$  in Table 1.4.1. There may be an unlimited number of rows. The distance between adjacent arrays of two by N casks ( $P_3$ ) shall be as specified in Table 1.4.1. See Figure 1.4.1 for further clarification. The pattern of required pitches and distances may be repeated for an unlimited number of columns.

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

Table 1.4.1

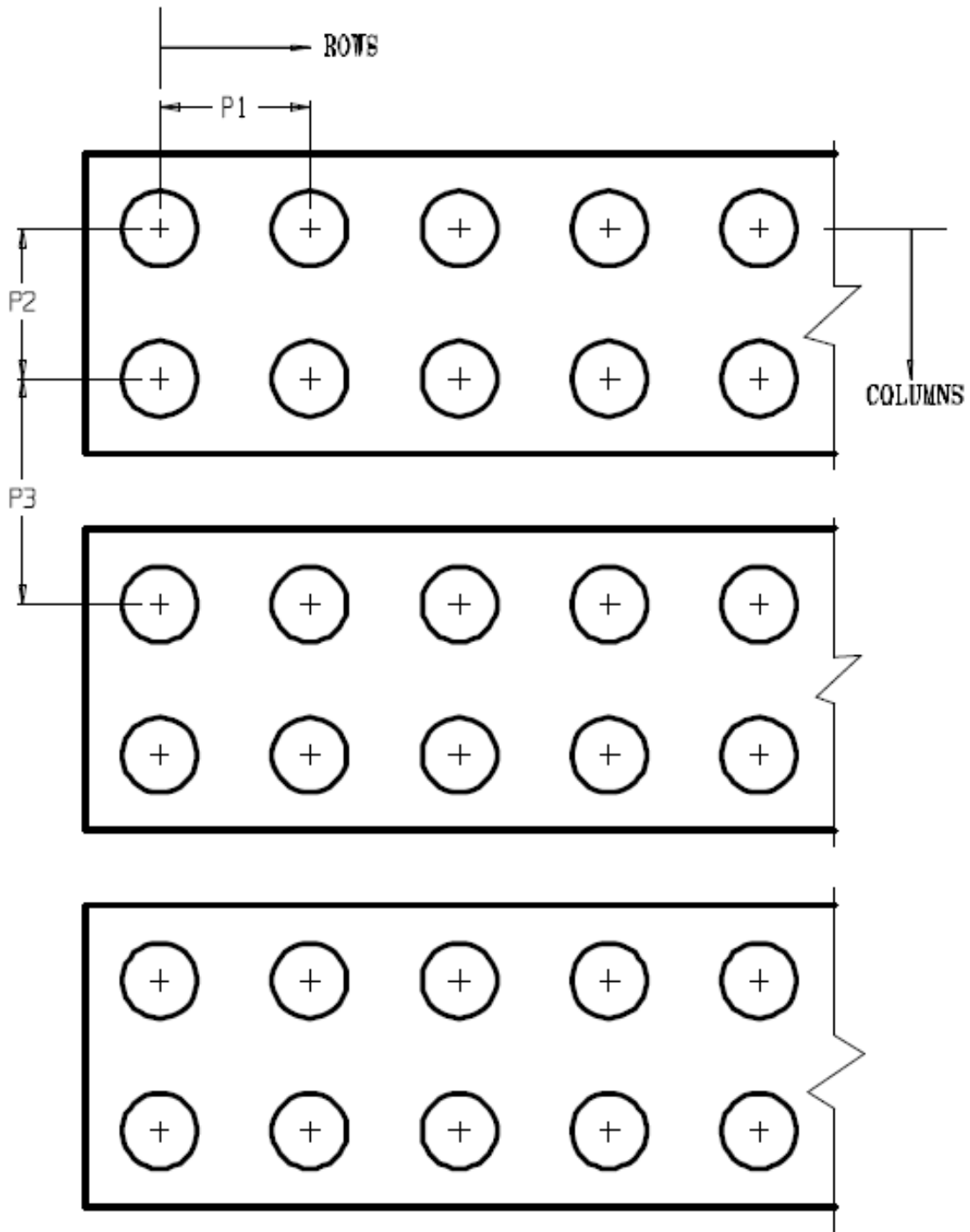
## CASK LAYOUT PITCH DATA FOR 2 BY N ARRAYS

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

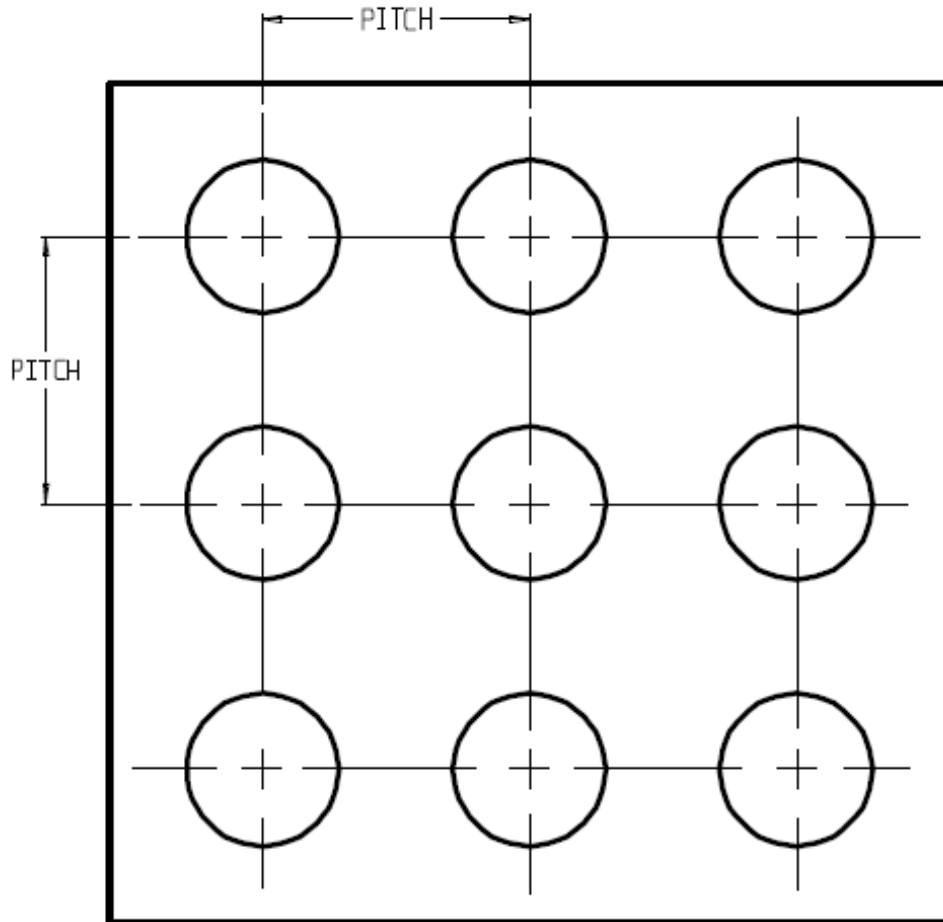
Table 1.4.2

CASK LAYOUT PITCH DATA FOR SQUARE ARRAYS

<b>Orientation</b>	<b>Nominal Cask Pitch (ft.)</b>
Between adjacent casks	18' - 8"



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



**FIGURE 1.4.2: CASK LAYOUT PITCH REQUIREMENTS BASED ON A SQUARE ARRAY**



## 1.5 DRAWINGS

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

<b>Drawing Number/Sheet</b>	<b>Description</b>	<b>Rev.</b>
3923	MPC Enclosure Vessel	40
3925	MPC-24E/EF Fuel Basket Assembly	10
3926	MPC-24 Fuel Basket Assembly	16
3927	MPC-32 Fuel Basket Assembly	20
3928	MPC-68/68F/68FF Basket Assembly	20
7195	MPC-68M Basket Assembly	15
1495 Sht 1/6	HI-STORM 100 Assembly	13
1495 Sht 2/6	Cross Section "Z" - "Z" View of HI-STORM	18
1495 Sht 3/6	Section "Y" - "Y" of HI-STORM	12
1495 Sht 4/6	Section "X" - "X" of HI-STORM	13
1495 Sht 5/6	Section "W" - "W" of HI-STORM	15
1561 Sht 1/6	View "A" - "A" of HI-STORM	11
1561 Sht 2/6	Detail "B" of HI-STORM	15
1561 Sht 3/6	Detail of Air Inlet of HI-STORM	11
1561 Sht 4/6	Detail of Air Outlet of HI-STORM	12
3669	HI-STORM 100S Assembly	23
1880 Sht 1/10	125 Ton HI-TRAC Outline with Pool Lid	9
1880 Sht 2/10	125 Ton HI-TRAC Body Sectioned Elevation	10
1880 Sht 3/10	125 Ton HI-TRAC Body Sectioned Elevation "B" - "B"	9
1880 Sht 4/10	125 Ton Transfer Cask Detail of Bottom Flange	10
1880 Sht 5/10	125 Ton Transfer Cask Detail of Pool Lid	10
1880 Sht 6/10	125 Ton Transfer Cask Detail of Top Flange	10
1880 Sht 7/10	125 Ton Transfer Cask Detail of Top Lid	9
1880 Sht 8/10	125 Ton Transfer Cask View "Y" - "Y"	9
1880 Sht 9/10	125 Ton Transfer Cask Lifting Trunnion and Locking Pad	7
1880 Sht 10/10	125 Ton Transfer Cask View "Z" - "Z"	9
1928 Sht 1/2	125 Ton HI-TRAC Transfer Lid Housing Detail	11
1928 Sht 2/2	125 Ton HI-TRAC Transfer Lid Door Detail	10
2145 Sht 1/10	100 Ton HI-TRAC Outline with Pool Lid	8
2145 Sht 2/10	100 Ton HI-TRAC Body Sectioned Elevation	8
2145 Sht 3/10	100 Ton HI-TRAC Body Sectioned Elevation 'B-B'	8
2145 Sht 4/10	100 Ton HI-TRAC Detail of Bottom Flange	7
2145 Sht 5/10	100 Ton HI-TRAC Detail of Pool Lid	6
2145 Sht 6/10	100 Ton HI-TRAC Detail of Top Flange	8
2145 Sht 7/10	100 Ton HI-TRAC Detail of Top Lid	8
2145 Sht 8/10	100 Ton HI-TRAC View Y-Y	8
2145 Sht 9/10	100 Ton HI-TRAC Lifting Trunnions and Locking Pad	5
2145 Sht 10/10	100 Ton HI-TRAC View Z-Z	7

<b>Drawing Number/Sheet</b>	<b>Description</b>	<b>Rev.</b>
2152 Sht 1/2	100 Ton HI-TRAC Transfer Lid Housing Detail	10
2152 Sht 2/2	100 Ton HI-TRAC Transfer Lid Door Detail	8
3187	Lug and Anchoring Detail for HI-STORM 100A	2
BM-1575, Sht 1/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1575, Sht 2/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1880, Sht 1/2	Bill-of-Material for 125 Ton HI-TRAC	9
BM-1880, Sht 2/2	Bill-of-Material for 125 Ton HI-TRAC	7
BM-1928, Sht 1/1	Bill-of-Material for 125 Ton HI-TRAC Transfer Lid	10
BM-2145 Sht 1/2	Bill-of-Material for 100 Ton HI-TRAC	6
BM-2145 Sht 2/2	Bill-of-Material for 100 Ton HI-TRAC	5
BM-2152 Sht 1/1	Bill-of-Material for 100 Ton HI-TRAC Transfer Lid	8
3768	125 Ton HI-TRAC 125D Assembly	12
4116	HI-STORM 100S Version B	30
4128	100 Ton HI-TRAC 100D Assembly	9

[PROPRIETARY DRAWINGS WITHHELD PER 10 CFR 2.390]

## 1.6 REFERENCES

- [1.0.1] 10CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.
- [1.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989.
- [1.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.
- [1.0.4] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan.<sup>†</sup>
- [1.0.5] American Concrete Institute, "Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)".
- [1.0.6] "Spent Nuclear Fuel Effective Thermal Conductivity Report," U.S. Department of Energy Document Identifier BBA000000-01717-5705-00010, Rev. 00, Tables S-1 through S-4.
- [1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, 1995 with Addenda through 1997.
- [1.1.2] USNRC Docket No. 72-1008, Final Safety Analysis Report for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.
- [1.1.3] USNRC Docket No. 71-9261, Safety Analysis Report for Packaging for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.
- [1.1.4] 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.
- [1.1.5] Deleted.
- [1.2.1] U.S. NRC Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket".

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

- [1.2.2] Directory of Nuclear Reactors, Vol. II, Research, Test & Experimental Reactors, International Atomic Energy Agency, Vienna, 1959.
- [1.2.3] V.L. McKinney and T. Rockwell III, "Boral: A New Thermal-Neutron Shield", USAEC Report AECD-3625, August 29, 1949.
- [1.2.4] Reactor Shielding Design Manual, USAEC Report TID-7004, March 1956.
- [1.2.5] "Safety Analysis Report for the NAC Storable Transport Cask", Revision 8, September 1994, Nuclear Assurance Corporation (USNRC Docket No. 71-9235).
- [1.2.6] Deleted.
- [1.2.7] Materials Handbook, 13<sup>th</sup> Edition, Brady, G.S. and H.R. Clauser, McGraw-Hill, 1991, Page 310.
- [1.2.8] Deleted.
- [1.2.9] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, June, 1993.
- [1.2.10] Deleted.
- [1.2.11] "Qualification of METAMIC<sup>®</sup> for Spent Fuel Storage Application," EPRI, 1003137, Final Report, October 2001.
- [1.2.12] "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications," Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., Docket No. 50-313 and 50-368, USNRC, June 2003.
- [1.2.13] "Metamic 6061+40% Boron Carbide Metal Matrix Composite Test", California Consolidated Tech. Inc. Report dated August 21, 2001 to NAC International.
- [1.2.14] "Recommendations for Preparing the Criticality Safety Evaluation for Transportation Packages," NUREG/CR-5661, USNRC, Dyer and Parks, ORNL.

## APPENDIX 1.A: ALLOY X DESCRIPTION

### 1.A ALLOY X DESCRIPTION

#### 1.A.1 Alloy X Introduction

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

- Type 316
- Type 316LN
- Type 304
- Type 304LN
- Duplex Stainless Alloy S31803 [1.A.3]

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.

Duplex stainless steels (DSS) are sensitive to the manufacturing processes employed in welding operations. Control of microstructure stability plays a vital role. The intermetallic microstructure is a complex function of the attendant parameters. For example, Cr and Mo promote ferrite and intermetallic phases, whereas N and Ni promote austenite.

During welding the balance between the ferritic and austenitic phases may be disturbed due to ferritization at high temperatures associated with welding operations. Ferrite content over 70% will lead to lower ductility and reduced corrosion resistance. Coarse ferritic grains are harmful for DSS toughness besides of impairing the austenite reformation at the heat affected zone (HAZ) [1.A.5]. The best metallurgical condition for welding is achieved by the most rapid quenching from the annealing temperature that produces a fine grained DSS structure with the required ferrite content (less than 70%).

Besides the austenite-ferrite phase balance, the second major concern with duplex steels and their chemical composition is the formation of detrimental intermetallic phases, precipitating preferentially in the ferrite, at elevated temperatures in the range of approximately 600 - 1750°F reaching an uncertain state of fragility at 887°F [1.A.4] and above. The mechanical (toughness) and corrosion properties of the weld and HAZ are deteriorated due to the presence of intermetallic phases.

Welding of DSS is associated with problems in HAZ which can be loss of corrosion, toughness, or post-weld cracking. The heat input and cooling rates in welding are important as they control ferrite to austenite transformation. Exceedingly low heat input may result in fusion zones and

HAZ which are excessively ferritic (above 70%) [1.A.6]. Exceedingly high heat input increases the danger of forming intermetallic phases [1.A.6]. In both cases the impact toughness and corrosion resistance of the DSS will be seriously affected. Hence, heat input must be 0.6 – 2.6 kJ/mm to retain the phase balance, limit the width of the HAZ, and obtain a sigma phase free product [1.A.5]. Further, cooling rate from the solution annealing temperature must exceed 0.3°C/s to avoid sigma phase and satisfy the generally accepted toughness requirements [1.A.8]. The maximum interpass temperature is limited to 150°C (302°F) [1.A.6].

DSS have chloride stress corrosion cracking (CSCC) resistance significantly greater than that of the austenitic stainless steels, but they are not completely immune. Experimental results indicate that DSS is prone to stress corrosion cracking at temperatures above 100°C [1.A.9]. Poor welding practice, a low pH, presence of Hydrogen in welds, and/or high ferrite (>70%) can contribute to failures at temperatures below 100°C.

Holtec will make sure that this material shall be used *only* if the metal temperature of the MPC shell can be assured to remain below the limit in Table 1.A.6 under all *normal operating* modes [1.A.3]. Likewise, under short term and accident conditions, such as the “inlet duct blockage” scenario, the maximum metal temperature of duplex stainless steel must be held below the limit in Table 1.A.6.

To confirm that the required properties are achieved in production, Holtec will implement a test program to insure that the weldments are tested for the absence of detrimental intermetallic phases. The test program will comply with ASTM A923 and will use metallographic examination, impact testing and corrosion testing to demonstrate the absence of such detrimental phases. The test will be intended to determine the presence or absence of intermetallic phase to the extent that it is detrimental to the toughness and corrosion resistance of the material. The test *shall* be implemented to products during weld procedure qualification as well as during fabrication which will provide the assurance that the weldments are *free* from detrimental intermetallic phases, and *provide* the required corrosion resistance and fracture toughness [1.A.7].

It is noted that DSS material shall not be used for the fabrication of MPC baskets and MPC internal components. For other stainless steels listed as members of Alloy X above, the design temperature limits in Table 2.2.3 remain unmodified. It is also noted that DSS shall be used only in aboveground systems.

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

### 1.A.2 Alloy X Common Material Properties

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

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

- Poisson's Ratio

The comparative values for Modulus of Elasticity at different temperatures are provided in Table 1.A.7. The values utilized for this licensing application are provided in their appropriate chapters.

### 1.A.3 Alloy X Least Favorable Material Properties

The following material properties vary between the Alloy X constituents:

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

Each of these material properties are provided in the ASME Code Section II [1.A.10]. 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 from -20°F. The design temperature of the MPC is -40°F, below -20°F, as stated in Table 1.2.2. Most of the above-mentioned properties improve as the temperature drops. For this reason, the values at the lowest design temperature for the HI-STORM 100 System have been assumed to be equal to the lowest value stated in the ASME Code. The lone exceptions are the coefficient of thermal expansion and thermal conductivity. As they decrease with decreasing temperature, its value for -40°F is linearly extrapolated from the 70°F value with the slope based on data from 70°F to 100°F.

The Alloy X material properties are the minimum values of the group for the design stress intensity, tensile strength, yield strength, and coefficient of thermal conductivity. Using minimum values of design stress intensity is conservative because lower design stress intensities lead to lower allowables that are based on design stress intensity. Similarly, using minimum values of tensile strength and yield strength is conservative because lower values of tensile strength and yield strength lead to lower allowables that are based on tensile strength and yield strength. When compared to calculated values, these lower allowables result in factors of safety that are conservative for any of the constituent materials of Alloy X. Using the minimum value of thermal conductivity has the effect of reducing the heat rejection rate from the canister which is conservative. 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 to support a conservative safety evaluation. However, for any internal interference assessment the actual values of coefficients of thermal expansion from the ASME Code or Table 1.A.4 will be used.

### 1.A.4 References

- [1.A.1] ASME Boiler & Pressure Vessel Code Section II, 1995 ed. with Addenda through 1997.
- [1.A.2] ASME Boiler & Pressure Vessel Code Section II, 2013 ed. with Addenda through 2014
- [1.A.3] ASME Code Case N-635-1 (2013)

- [1.A.4] C. Örnek, D. Engelberg, S. Lyon and T. Ladwein, "Effect of "475°C Embrittlement" on the Corrosion Behaviour of Grade 2205 Duplex Stainless Steel Investigated Using Local Probing Techniques," *Corrosion Management Magazine*, no. 115, pp. 9-11, 2013.
- [1.A.5] C.R. Xavier, H.G. Delgado Jr., J.A de Castro, "An Experimental and Numerical Approach for the Welding Effects on the Duplex Stainless Steel Microstructure" – *Materials Research Vol. 18(3)* pp. 489-502, 2015.
- [1.A.6] "Practical guidelines for Fabrication of Duplex Stainless Steels" – International Molybdenum Association, 2014.
- [1.A.7] ASTM A923-14, "Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels" – W Conshohocken, PA, ASTM International 2014.
- [1.A.8] J. Charles, "Duplex Stainless Steels, A Review After DSS '07 held in GRADO" – *Steel Research International Vol. 79(6)* pp. 455-465, 2008.
- [1.A.9] A. Leonard, "Review of external stress corrosion cracking of 22%Cr duplex stainless steel, Phase 1 – Operational data acquisition," – HSE RR 129, Her Majesty's Stationery Office, Norwich, UK, 2003.
- [1.A.10] ASME Boiler & Pressure Vessel Code Section II, Part D, 2015.



Table 1.A.1

ALLOY X DESIGN STRESS INTENSITY ( $S_m$ ) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Duplex Stainless Steel S31803 [Notes 3]	Alloy X (minimum of constituent values)
-40	20.0	20.0	20.0	20.0	30.0	20.0
100	20.0	20.0	20.0	20.0	30.0	20.0
200	20.0	20.0	20.0	20.0	30.0	20.0
300	20.0	20.0	20.0	20.0	28.9	20.0
400	18.7	18.7	19.3	18.9	27.8	18.7
500	17.5	17.5	18.0	17.5	27.2	17.5
600	16.4	16.4	17.0	16.5	26.9	16.4
650	16.2	16.2	16.7	16.0	-	16.0
700	16.0	16.0	16.3	15.6	-	15.6
750	15.6	15.6	16.1	15.2	-	15.2
800	15.2	15.2	15.9	14.9	-	14.9

## Notes:

1. Source: Table 2A on pages 314, 318, 326, and 330 of [1.A.1] for Type 316/316LN/304/304LN.
2. Units of design stress intensity values are ksi.
3. Design stress intensity values have been derived based on the basis established in Mandatory Appendix 2 page 924 and 925 which essentially states that the stress intensity value at temperature is the minimum of one-third of the tensile strength or two-thirds of the yield strength at temperature.
4. Maximum temperature of use for duplex stainless steel under both long term storage and short term / accident conditions is noted in Table 1.A.6.

Table 1.A.2

ALLOY X TENSILE STRENGTH ( $S_u$ ) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Duplex Stainless Steel S31803 [Notes 4]	Alloy X (minimum of constituent values)
-40	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	90 (90)	75.0 (70.0)
100	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	90 (90)	75.0 (70.0)
200	71.0 (66.2)	71.0 (66.2)	75.0 (70.0)	75.0 (70.0)	90 (90)	71.0 (66.2)
300	66.0 (61.5)	66.0 (61.5)	73.4 (68.5)	70.9 (66.0)	86.8 (86.8)	66.0 (61.5)
400	64.4 (60.0)	64.4 (60.0)	71.8 (67.0)	67.1 (62.6)	83.5 (83.5)	64.4 (60.0)
500	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	64.6 (60.3)	81.6 (81.6)	63.5 (59.3)
600	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	63.1 (58.9)	80.7 (80.7)	63.1 (58.9)
650	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	62.8 (58.6)	-	62.8 (58.6)
700	63.5 (59.3)	63.5 (59.3)	71.8 (67.0)	62.5 (58.4)	-	62.5 (58.4)
750	63.1 (58.9)	63.1 (58.9)	71.4 (66.5)	62.2 (58.1)	-	62.2 (58.1)
800	62.7 (58.5)	62.7 (58.5)	70.9 (66.2)	61.7 (57.6)	-	61.7 (57.6)

## Notes:

1. Source: Table U on pages 437, 439, 441, and 443 of [1.A.1] for Type 304/304LN/316/316LN.
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) or SA-182 forged material (S31803), which are used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.
4. Table U on page 521 of [1.A.10] for DSS UNS S31803
5. Maximum temperature of use for duplex stainless steel under both long term storage and short term / accident conditions is noted in Table 1.A.6.

Table 1.A.3

ALLOY X YIELD STRESSES ( $S_y$ ) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Duplex Stainless Steel S31803 [Notes 3]	Alloy X (minimum of constituent values)
-40	30.0	30.0	30.0	30.0	65.0 (65.0)	30.0
100	30.0	30.0	30.0	30.0	65.0 (65.0)	30.0
200	25.0	25.0	25.8	25.5	57.8 (57.8)	25.0
300	22.5	22.5	23.3	22.9	53.7 (53.7)	22.5
400	20.7	20.7	21.4	21.0	51.2 (51.2)	20.7
500	19.4	19.4	19.9	19.4	49.6 (49.6)	19.4
600	18.2	18.2	18.8	18.3	47.9 (47.9)	18.2
650	17.9	17.9	18.5	17.8	-	17.8
700	17.7	17.7	18.1	17.3	-	17.3
750	17.3	17.3	17.8	16.9	-	16.9
800	16.8	16.8	17.6	16.6	-	16.6

## Notes:

1. Source: Table Y-1 on pages 518, 519, 522, 523, 530, 531, 534, and 535 of [1.A.1] for Type 304/304LN/316/316LN.
2. Units of yield stress are ksi.
3. Table Y-1 on page 672 and 673 of [1.A.10] for DSS UNS S31803. Values in parentheses are based on SA-182 forged material (S31803) which is used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.
4. Maximum temperature of use for duplex stainless steel under both long term storage and short term / accident conditions is noted in Table 1.A.6.

Table 1.A.4

ALLOY X COEFFICIENT OF THERMAL EXPANSION  
vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Duplex Stainless Steel S31803 [Notes 3]	Alloy X Maximum	Alloy X Minimum
-40	8.55	8.54	6.63	8.55	6.63
100	8.55	8.54	7.1	8.55	7.1
150	8.67	8.64	7.3	8.67	7.3
200	8.79	8.76	7.5	8.79	7.5
250	8.90	8.88	7.6	8.90	7.6
300	9.00	8.97	7.8	9.00	7.8
350	9.10	9.11	7.9	9.11	7.9
400	9.19	9.21	8.0	9.21	8.0
450	9.28	9.32	8.1	9.32	8.1
500	9.37	9.42	8.3	9.42	8.3
550	9.45	9.50	8.4	9.50	8.4
600	9.53	9.60	8.4	9.60	8.4
650	9.61	9.69	-	9.69	9.61
700	9.69	9.76	-	9.76	9.69
750	9.76	9.81	-	9.81	9.76
800	9.82	9.90	-	9.90	9.82

## Notes:

1. Source: Table TE-1 on pages 590 and 591 of [1.A.1], for Type 304/304LN/316/316LN.
2. Units of coefficient of thermal expansion are in./in.-°F x 10<sup>-6</sup>.
3. Table TE-1 on page 753 of [1.A.10] for DSS UNS S31803.
4. Maximum temperature of use for duplex stainless steel under both long term storage and short term / accident conditions is noted in Table 1.A.6.

Table 1.A.5

## ALLOY X THERMAL CONDUCTIVITY vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Duplex Stainless Steel S31803 [Notes 3]	Alloy X (minimum of constituent values)
-40	8.23	6.96	7.83	6.96
70	8.6	7.7	8.2	7.7
100	8.7	7.9	8.3	7.9
150	9.0	8.2	8.6	8.2
200	9.3	8.4	8.8	8.4
250	9.6	8.7	9.1	8.7
300	9.8	9.0	9.3	9.0
350	10.1	9.2	9.5	9.2
400	10.4	9.5	9.8	9.5
450	10.6	9.8	10.0	9.8
500	10.9	10.0	10.2	10.0
550	11.1	10.3	10.5	10.3
600	11.3	10.5	10.7	10.5
650	11.6	10.7	-	10.7
700	11.8	11.0	-	11.0
750	12.0	11.2	-	11.2
800	12.2	11.5	-	11.5

## Notes:

1. Source: Table TCD on page 606 of [1.A.1] for Type 304/304LN/316/316LN.
2. Units of thermal conductivity are Btu/hr-ft-°F.
3. Table TCD on page 773 of [1.A.10] for DSS UNS S31803.
4. Maximum temperature of use for duplex stainless steel under both long term storage and short term / accident conditions is noted in Table 1.A.6.

Table 1.A.6

## DUPLEX STAINLESS STEEL TEMPERATURE LIMITS†

Parameter	Value
Long Term, Normal Condition Design Temperature Limits (Long-Term Events) (° F)	550
Short-Term Events, Off-Normal, and Accident Condition Temperature Limits (° F)	600

† These temperature limits take precedence over those in Table 2.2.3

Table 1.A.7

## ALLOY X MODULI OF ELASTICITY (E) vs. TEMPERATURE

Temp. (Deg. F)	Moduli of Elasticity (E)	
	Austenitic stainless steels (304, 304LN, 316, 316LN)	Duplex stainless steel (UNS S31803)
-40	28.82	29.78
100	28.14	28.82
150	27.87	28.51
200	27.6	28.2
250	27.3	27.85
300	27.0	27.5
350	26.75	27.25
400	26.5	27.0
450	26.15	26.7
500	25.8	26.4
550	25.55	26.2
600	25.3	26.0
650	25.05	-
700	24.8	-
750	24.45	-
800	24.1	-

## Definitions:

E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for E values of austenitic stainless steels is material group G in Table TM-1 of [3.3.1].
2. Source for E values of duplex stainless steel is material group H in Table TM-1 of [1.A.10].

**FIGURES 1.A.1 through 1.A.5  
INTENTIONALLY DELETED**



## APPENDIX 1.B: HOLTITE™ NEUTRON SHIELD MATERIAL

The shielding against neutron fluence in Holtec's dry storage casks is provided by Holtite™, which is a boron carbide impregnated, hydrogen rich, radiation resistant, neutron shield material initially developed by Holtec International in the 1990s as a part of the company's HI-STAR 100 design development program. The classical Holtite product, labeled Holtite-A, utilizes Aluminum Tri-hydrate (ATH) in a hydrogen rich epoxy compound. A mixing machine, developed by Holtec, is used in the cask manufacturing facility to mix and then pour the Holtite-A slurry into the cask's cavities. The shielding material cures *in-situ*, ensuring that no gaps or voids will exist. Holtite-A was subjected to extensive studies of its critical characteristics (viz., radiation resistance, physical stability at service temperature and homogeneity) during its evaluation and validation program [1B.1, 1.B.2] which led to its regulatory approval in the HI-STAR 100 dockets by the USNRC (dockets 71-9261 and 72-1008) in the 1990s and subsequent use in the manufactured HI-STAR 100 casks, many of which have been deployed around the world. The key physical properties including the Design Temperature of Holtite-A is provided in Table 1.B.1.

In addition to Holtite-A, Holtec also utilizes Holtite-B [1.B.3.]. Holtite-B is the combination of a hydrogen rich polymer, Nylon plus a neutron capture agent, B<sub>4</sub>C and an optional thermal conductivity enhancing agent, aluminum. Holtite-B's physical properties are provided in Table 1.B.2. Holtite B is particularly suitable for use in elevated temperature applications (Table 1.B.3). As Table 1.B.4 indicates, both Holtite A and Holtite B are NRC-reviewed neutron shield materials, having been licensed and used in several cask models by different regulatory agencies.

### 1.B.1 References

- [1.B.1] Holtite-A: Development History and Thermal Performance Data, Holtec Report HI-2002396, Rev. 3(Holtec Proprietary)
- [1.B.2] Holtite-A: Results of Pre-and Post- Irradiation Tests and Measurements, Holtec Report HI-2002420, Rev. 1(Holtec Proprietary)
- [1.B.3] Holtite B Sourcebook, Holtec Report number 2073684, Latest Revision (Holtec Proprietary)

Table 1.B.1: Reference Physical Properties of as-manufactured Holtite-A	
% ATH	62 nominal
Specific Gravity	1.68 g/cc nominal
Design Temperature	150 °C
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent

Table 1.B.2: Primary Characteristics of Holtite-B	
Characteristic	Holtite-B
Minimum Bulk Density	1.13 g/cc
Minimum Hydrogen Density	0.1065 g/cc
Minimum Boron Carbide Content	2.0 wt. %
Reference Thermal Conductivity	0.25 W/m-K– 0.4 W/m-K
Note: Primary characteristics of Holtite-B may be adjusted for site-specific applications.	

Holtite B Version	Temperature, Deg. C
Type 6 polymer	185
Type 66 polymer	204

Holtite-A		Holtite-B	
Cask model name & country of the regulatory authority	Cask's function (Transport/storage)	Cask model name & country of the regulatory authority	Cask's function (Transport/storage)
HI-STAR 100 (USA)	Storage and transport	HI-STAR 180(USA)	Transport
HI-TRAC 100 & HI-TRAC 100D (USA)	Canister transfer(Vertical)	HI-STAR 180D (USA)	Transport
HI-STORM MIC (UK)	Canister storage	HI-TRAC 150D (UK)	Canister transfer (Vertical)
HI-TRAC-H (Ukraine)	Horizontal canister transfer for NUHOMS (RBMK fuel)		

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

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

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

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

### 1.D.1 Introduction

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

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

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

### 1.D.2 Design Requirements

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

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

The shielding performance of the plain concrete is maintained by ensuring that the minimum concrete density is met during its placement and the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions utilize the temperature limits provided in Table 1.D.1. The temperature limits for transient conditions (such as under partial or full duct blockage and fire) are set down in reference [1.D.4] in conformance with the guidelines set forth in the ACI code, supplemented by data from the published permanent literature. A state-of-the-art review paper [1.D.5] provides extensive test data on the temperature dependence of plain concrete. In reference [1.D.4] the guidance from ACI and the published archival data on plain concrete has been used to establish temperature limits for normal and transient conditions listed in Table 1.D.1 herein. The literature citation summarized in Section 1.D.2.1 below are retained because they date back to the original issue of this appendix even though the additional test data in [1.D.5] subsumes them.

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

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

With respect to concrete compressive strength at bulk temperatures up to 300°F, test studies for elevated temperatures were performed by Carette and Malhotra [1.D.1] that examined conditions very similar to those of the HI-STORM concrete. Their tests were performed on 4” diameter by 8” long test cylinders. The test conditions most closely match the HI-STORM concrete with respect to water-to-cement ratio, limestone aggregate and temperature. While the HI-STORM storage period is much greater than 4 months (testing duration per [1.D.1]), the investigators state “any major strength loss is found to occur within the first month of exposure.” The four-month compressive strength for these conditions was actually determined to be greater than the nominal concrete strengths despite the elevated temperatures. This is attributable to the increase in compressive strength that accompanies concrete aging, which more than offsets the temperature effects. Per [1.D.1], approximately 45% reduction in compressive strength and about 5% reduction in weight is observed in limestone concrete (water to cement ratio, w/c = 0.33) exposed for one month at 300°C (572°F).

Exposure of concrete to elevated temperatures can affect its properties due to the dehydration or loss

of absorbed and chemically combined water. With respect to concrete shielding performance at local temperatures above 300°F, [1.D.4] examined weight loss and thermal degradation mechanisms of concrete at elevated temperatures.

To address a postulated accident that may occur during the 30-day vent blockage condition (i.e. a tornado borne missile impact), the compressive strength of the concrete is conservatively reduced by 50% even though the maximum temperature experienced by concrete during 30-day vent blockage accident is less than 450°F and is about 300°F during long term normal condition. The evaluations (Supplements 15 and 25 of HI-2012769) conclude that the concrete in overpack, post 50% strength reduction, is acceptable during and after the 30-day vent blockage accident, and during the long term normal condition.

To evaluate the effect of hydrogen loss on the shielding performance of the HI-STORM 100, it was assumed that entire hydrogen is lost from the concrete. This is an excessively conservative estimate of an upper bound dose rate effect. Water and hydrogen is present in concrete in two forms, chemically bound water/hydrogen, and physically bound water. The material properties for the concrete in the HI-STORM 100 assume hydrogen content that is less than or maybe equal to that in the chemically bound water. As the entire weight loss is attributed to water loss, the total amount of hydrogen in concrete is expected to be no less than 0.8 wt%. The assumed hydrogen content in concrete composition listed in Table 5.3.2 of the FSAR is only 0.6 wt%, which is used for shielding analyses. The analyses results show that shielding performance with such reductions in hydrogen content is negligible. Additionally, the results demonstrate that the hypothetical HI-STORM 100% duct blockage accident condition is bounded by the HI-TRAC accident condition discussed in Section 5.1.2 of the FSAR.

### 1.D.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements, drawn from ACI 349-85 and supplemented by the provisions of NUREG 1536 (page 3-21), are intended to ensure that the “*critical characteristics*” of the concrete placed in the HI-STORM overpack comply with the requirements of this Appendix and standard good practice.

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

In accordance with the requirement in Section 3.3 of Appendix B of the HI-STORM 100 CoC, Section 1.D.4, Table 1.D.1 and Table 1.D.2 were developed using the guidance of ACI 349-85, to the extent it needs to be applied to the unique application of placing unreinforced concrete inside the

steel enclosure of the HI-STORM overpack. Other concrete standards were used, as appropriate, to provide the controls necessary to assure that the *critical characteristics* of the overpack concrete will be achieved and that the concrete will perform its design function.

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

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

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

#### 1.D.3.2 Concrete Mix Design and Material Requirements

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

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

For the standard use local course and fine aggregates supplied by the local batch, a high level of confidence based on continued use in area concrete obviates the need for many of the aggregate testing recommended by ASTM C33. However, certain testing relevant to confirming the acceptability of the aggregate is required by this specification. For both the local fine and course aggregate, laboratory testing shall be carried out to confirm grading per ASTM C33 as well as the test per ASTM C117 to determine materials finer than 200 sieve. A laboratory technician shall also



visually inspect the source pile to evaluate the aggregates for any deleterious substances or organic impurities. If this visual inspection reveals any evidence of deleterious substances or organic impurities, additional aggregate testing that addresses deleterious substances per ASTM C33 for both fine and course aggregates as well as organic impurities testing per ASTM C40 for the local fine aggregate shall be conducted.

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

#### 1.D.3.3 Shielding Enhancer Additives

Shielding Enhancer Additives may be used to enhance the shielding effectiveness or neutron capture effectiveness of the plain concrete. To enhance the gamma shielding effectiveness, it is permissible to use a higher density conventional aggregate and/or to incorporate suitable high specific gravity non-organic material as additives. To improve the neutron capture effectiveness, granules of material with high neutron absorption cross-sections may also be dispersed in the plain concrete. The Shielding Enhancer Additives shall meet the product performance criteria set forth in Table 1.D.3. The use of Shielding Enhancer Additives requires consideration be given to the effects on Chapter 4 thermal evaluations and Chapter 5 shielding evaluations concerning the plain concrete on a site-specific basis. If the use of Shielding Enhancer Additives causes the overpack to exceed the bounding weight used in Chapter 2 structural qualifications, the HI-STORM shell and/or lid “steel structure”, as applicable, shall be evaluated to insure that all structural requirements set forth in Chapter 2 are satisfied. It is noted that any specific material selected for use as a Shielding Enhancer Additive may require additional consideration under the §72.48 process.

#### 1.D.4 Construction Requirements

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

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

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

#### 1.D.5 Testing Requirements

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

#### 1.D.6 References

- [1.D.1] Carette and Malhotra, "Performance of Dolostone and Limestone Concretes at Sustained High Temperatures," Temperature Effects on Concrete, ASTM STP 858.
- [1.D.2] Deleted
- [1.D.3] Concrete Manual, 8<sup>th</sup> Edition, US Bureau of Proclamation, Denver, Colorado, 1975.
- [1.D.4] Holtec Position Paper, DS-289, "Maximum Permissible Temperature in Plain Concrete in HI-STORM System Components Under Off-Normal and Accident Conditions", Revision 5.
- [1.D.5] Kodur, V., "Properties of Concrete at Elevated Temperatures", ISRN Civil Engineering, Volume 2014 (2014), Article ID 468510, 15 pages.

Table 1.D.1  
Requirements for Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density in overpack body (Minimum) (see Table 3.2.1 for information on maximum concrete density)	140 lb/ft <sup>3</sup>
Density in lid and pedestal (Minimum) (See Table 3.2.1 for information on maximum concrete density)	140 lb/ft <sup>3</sup> (HI-STORM 100S Version B does not have a concrete-filled pedestal)
Specified Compressive Strength	3,300 psi (Note 5)
Compressive and Bearing Stress Limit	Per ACI 318.1-89(92)
Cement Type and Mill Test Report	Type II; (ASTM C 150 or ASTM C595)
Aggregate Type	Fine and coarse aggregate as required (Note 2)
Nominal Maximum Aggregate Size	1-1/2 (inch)
Water Quality	Deleted
Material Testing	See Note 4.
Admixtures	Deleted
Maximum Water to Cement Ratio	0.5 (Table 4.5.2)
Maximum Water Soluble Chloride Ion Cl in Concrete	1.00 percent by weight of cement (Table 4.5.4) (See Table 1.D.2, Note 1)
Concrete Quality	Deleted
Mixing and Placing	See Note 6.
Consolidation	Deleted
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Through-Thickness Section Average <sup>†</sup> Temperature Limit Under Long Term Conditions	300°F (See Note 3)
Local Temperature Limit Under 30-Day 100% Vent Blockage Accident Condition	See Table 2.2.3
Local Temperature Limit Under Short Term, Off Normal, and Accident Conditions	See Table 2.2.3
Aggregate Maximum Value <sup>††</sup> of Coefficient of Thermal Expansion (tangent in the range of 70°F to 100°F)	6E-06 inch/inch/°F (NUREG-1536, 3.V.2.b.i.(2)(c)2.b)

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

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

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

Notes:

1. Deleted
2. The coarse aggregate shall meet the requirements of ASTM C33 for class designation 1S from Table 3. However, if the requirements of ASTM C33 cannot be met, concrete aggregates that have been shown by special tests or actual service to produce concrete of adequate strength, unit weight, and durability meeting the requirements of Tables 1.D.1 and 1.D.2 are acceptable in accordance with ACI 349 Section 3.3.2. The high-density coarse aggregate percentage of Material Finer than No. 200 Sieve may be increased to 10 % if the material is essentially free of clay or shale.
3. The 300°F long term temperature limit is specified in accordance with Paragraph A.4.3 of Appendix A to ACI 349 for normal conditions considering the very low maximum stresses calculated and discussed in Section 3.4 of this FSAR for normal conditions. In accordance with this paragraph of the governing code, the specified concrete compressive strength is supported by test data and the concrete is shown not to deteriorate, as evidenced by a lack of reduction in concrete density or durability.
4. Tests of materials and concrete, as required, shall be made in accordance with standards of the American Society for Testing and Materials (ASTM) as specified here, to ensure that the *critical characteristics* for the HI-STORM concrete are achieved. ASTM Standards to be used include: C 31-96, C 33-82, C 39-96, C 88-76, C 131-81, C 138-92, C 143-98, C 150-97, C 172-90, C 192-95, C 494-92, C 637-73. More recent approved editions of the referenced standards may be used.
5. The compressive strength of HI-STORM concrete is used in determining the extent of penetration into the cask by a medium or small Design Basis Missile treated in this FSAR. For sites subject to more severe missiles, the minimum concrete strength may be increased, as necessary, to meet site specific requirements. Lower than the reference strength is seldom required to improve safety margins at a site; however, if required, a lower concrete compressive strength than the nominal value in the table may be employed with Holtec structural engineering approval.
6. Water and admixtures may be added at the job site to bring both the slump and wet unit weight of the concrete within the mix design limits. Water or admixtures shall not be added to the concrete after placement activities have started. The tolerance for individual and combined aggregate weights in the concrete batch may be outside of tolerances specified in ASTM C94, provided that the wet unit weight of the concrete is tested prior to placement and confirmed to be within the approved range.

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

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

## Notes:

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

Table 1.D.3: Required Properties of Shielding Enhancer Additives (“Aggregate”)

<b>ITEM</b>	<b>REQUIRED PROPERTY</b>
1	The Aggregate does not chemically react extensively with the other constituents in the concrete.
2	The concrete meets the minimum compressive strength requirement specified for the specific HI-STORM application.
3	The Aggregate remains physically stable at the maximum permissible normal and short term temperatures set down in Table 1.D.1.
4	The maximum Aggregate size is bounded by the limit in Table 1.D.1.
5	The concrete mixing process is qualified to ensure the additive aggregate is evenly distributed with a maximum permissible distribution difference of 10%.

## SUPPLEMENT 1.I

### GENERAL DESCRIPTION OF HI-STORM 100U SYSTEM

#### 1.I.0 GENERAL INFORMATION

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

#### 1.I.1 INTRODUCTION

HI-STORM 100U, like HI-STORM 100<sup>3</sup> and HI-STORM 100S<sup>4</sup>, is a vertical, ventilated dry spent fuel storage system engineered to be fully compatible with the presently certified HI-TRAC transfer casks and MPCs. HI-STORM 100U is an underground vertical ventilated module (VVM) designed to accept all MPC models for storage at an ISFSI (see Figure 1.I.1). ISFSIs employing the VVM may be designed for any number of MPCs and expanded to add additional storage modules as the need arises. Each VVM stores one MPC.

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

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

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

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

3 U.S. Patent No. 6,064,710 dated May 16, 2000.

4 U.S. Patent No. 6,718,000 dated April 6, 2004.

with the surrounding subgrade for most of its height except for the top region where it is girdled by the ISFSI pad. The ISFSI pad serves several purposes in the HI-STORM 100U storage system, such as:

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

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

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

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

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

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

With the Closure Lid removed, the CEC is a closed bottom, open top, thick walled cylindrical vessel that has no penetrations or openings. Thus, groundwater has no path for intrusion into the interior space of the MPC storage cavity. Likewise, any water that may be introduced into the MPC storage cavity through the air passages in the top lid will not drain out on its own. The Bottom Plate of the CEC is round and



slightly larger in diameter than the Container Shell to accommodate an all around weld between the plate and the shell.

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

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

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

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

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

- i. Both the inlet and outlet air passages are located in the Closure Lid, so there are no lateral radiation leakage paths during the MPC lowering or raising operation. The need for shield blocks (necessary to close off vents in some aboveground HI-STORM 100 overpacks) is eliminated.
- ii. Both inlet and outlet passages are radially symmetric so that the air cooling action in the system is not affected by the change in the horizontal direction of the wind.
- iii. By locating the air inlet at the periphery of the Closure Lid and the air outlet at its top central axis, mixing of entering and exiting air streams is essentially eliminated.
- iv. The inlet and outlet air passages are made of “formed and flued” heads (i.e., surfaces of revolution) that serve three major design objectives as noted below.
  - a. The curved passages eliminate any direct line of sight to the MPC storage space and serve as an effective means to scatter the photons streaming from the stored fuel.
  - b. The curved steel plates significantly increase the load bearing capacity of the Closure Lid much in the manner as a curved beam exhibits considerably greater lateral load bearing capacity in comparison to its straight counterpart. This design feature is a valuable attribute if a “beyond-the-design basis” impact scenario involving a large and energetic missile needs to be evaluated for a particular ISFSI site.
  - c. The curved passages, as is well known in classical hydraulics, provide for minimum loss of pressure in the coolant air stream, resulting in a more vigorous ventilation action.
- v. The Closure Lid rests on the Container Flange and is gasketed to minimize foreign material intrusion.
- vi. The top surface of the Closure Lid is also curved and extended beyond the air inlet perimeter to efficiently drain off rainwater.
- vii. The Container Flange restrains the Closure Lid against horizontal movement, during a Design Basis Earthquake event or a tornado missile strike.
- viii. The radially symmetric air inlet passage in the lid is geometrically aligned with the annular opening formed between by the Divider Shell and the CEC Shell.
- ix. Because the inlet opening extends around the circumference of the Closure Lid, the hydraulic resistance to the incoming airflow, a common limitation in ventilated modules, is minimized. A similar airflow resistance minimization facility is built into the pathway for the exiting air. A circumferentially circumscribing vent opening is also quite obviously

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

- x. To minimize the VVM's height, a portion of the Closure Lid extends into the cylindrical space above the MPC. This cylindrical below-surface extension of the Closure Lid is also made of steel filled with shielding concrete to maximize the blockage of skyward radiation issuing from the MPC.
- xi. All inlet and outlet air passages are equipped with screens, as in the aboveground HI-STORM overpacks, to prevent debris, insects, and small animals from entering the VVM. Although the screen is a non-structural member, it is designed for long-term durability and easy maintainability to ensure that its installation, removal, and maintenance are ALARA.

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

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

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

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

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

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

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

- i. The MPC is supported on MPC Bearing Pads to provide an inlet air plenum at the bottom of the storage cavity (Figure 1.I.2). The bottom of the MPC, however, will be in contact with water if the cutouts at the bottom of the Divider Shell were to be filled with water cutting off feed air. As long as the MPC is wetted with water, the peak cladding temperature of the stored spent fuel will not exceed the regulatory off-normal condition temperature limit. Thus, the VVM configuration provides a built-in protection against flood events.
- ii. Like the HI-STORM 100A and 100SA models, tipover of the canister in storage is not possible.
- iii. Although the modules may be closely spaced, as illustrated in Figure 1.I.5, the design permits any MPC located in any cavity to be independently accessed and retrieved using a HI-TRAC transfer cask.
- iv. A cask transporter typical of those used in numerous Holtec ISFSI projects for on-site transport of loaded HI-TRACs and HI-STORMs can provide the means to deliver the loaded HI-TRAC to the HI-STORM 100U VVM and to carry out the MPC lowering operation (Figure 1.I.7). The same cask transporter can also be used to remove an MPC from storage and place it in a recipient HI-TRAC transfer cask.
- v. To exploit the biological shielding provided by the surrounding soil subgrade, the MPC is entirely situated well below the top-of-grade level. The open plenum above the MPC also acts to boost the ventilation action of the coolant air.

- vi. Because the VVM is rendered into an integral part of the subgrade, it cannot be translocated to another ISFSI site. It also cannot be lifted and, therefore, is not subject to the potential for a handling accident.
- vii. Removal of water from the bottom of the storage cavity can be carried out by the simple expedient use of a flexible hose inserted through either the inlet or the outlet passageway.
- viii. As discussed in Supplement 3.I.4, all exposed surfaces of the VVM are coated with proven surface preservatives that meet the toxicological and extraction test requirements of ANSI/NSF Standard 61.
- ix. The VVM is a formed metallic welded structure with a removable Closure Lid. The Closure Lid is also a formed metallic welded structure but filled with shielding concrete. The requirements on the shielding concrete are specified in Appendix 1.D.

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

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

The VVM implements seals or gaskets at the Closure Lid. The outer seal is a weather seal (between the Closure Lid and the top of the Divider Shell), which facilitates maintenance by minimizing foreign material intrusion into the MPC storage cavity. The inner seal (between the Closure Lid skirt and the Divider Shell (not shown on the licensing drawing 4501)) provides an enhanced barrier against mixing of inlet and outlet air in the annular space between the Divider Shell and the cylindrical protrusion in the Closure Lid (even though the pressure differential between the two sides is extremely low – less than a few inches of water). The outer seal relies on the weight of the Closure Lid to insure sealing. A polymeric gasket made from EPDM<sup>5</sup> is preferred for this purpose. The inner seal is made of a durable radiation and heat resistant material and designed to have no credible mechanism for significant degradation or

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

detachment from its sealing location. The seals do not provide a safety function because their loss during operation would not have an effect on safe operation of the system.

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

#### 1.1.2.2 HI-STORM 100U System Sequence of Operations

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

#### 1.1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Same as in Section 1.3.

#### 1.1.4 GENERIC CASK ARRAYS

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

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

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

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

#### 1.I.5 FIGURES AND DRAWINGS

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

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

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

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

[PROPRIETARY DRAWINGS WITHHELD PER 10 CFR 2.390]

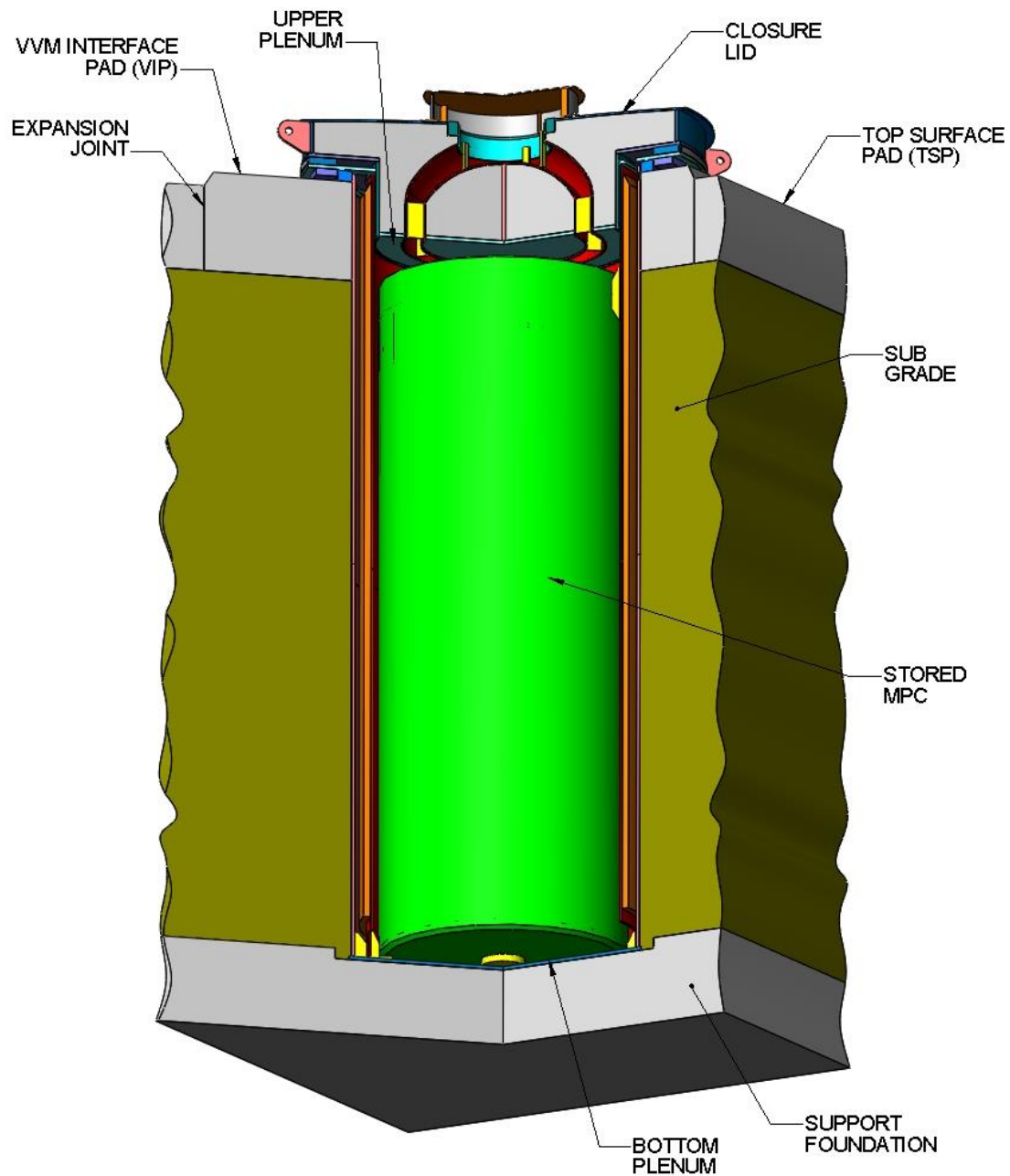


FIGURE 1.1.1: CUT-AWAY VIEW OF HI-STORM 100U SYSTEM)

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.1.5.



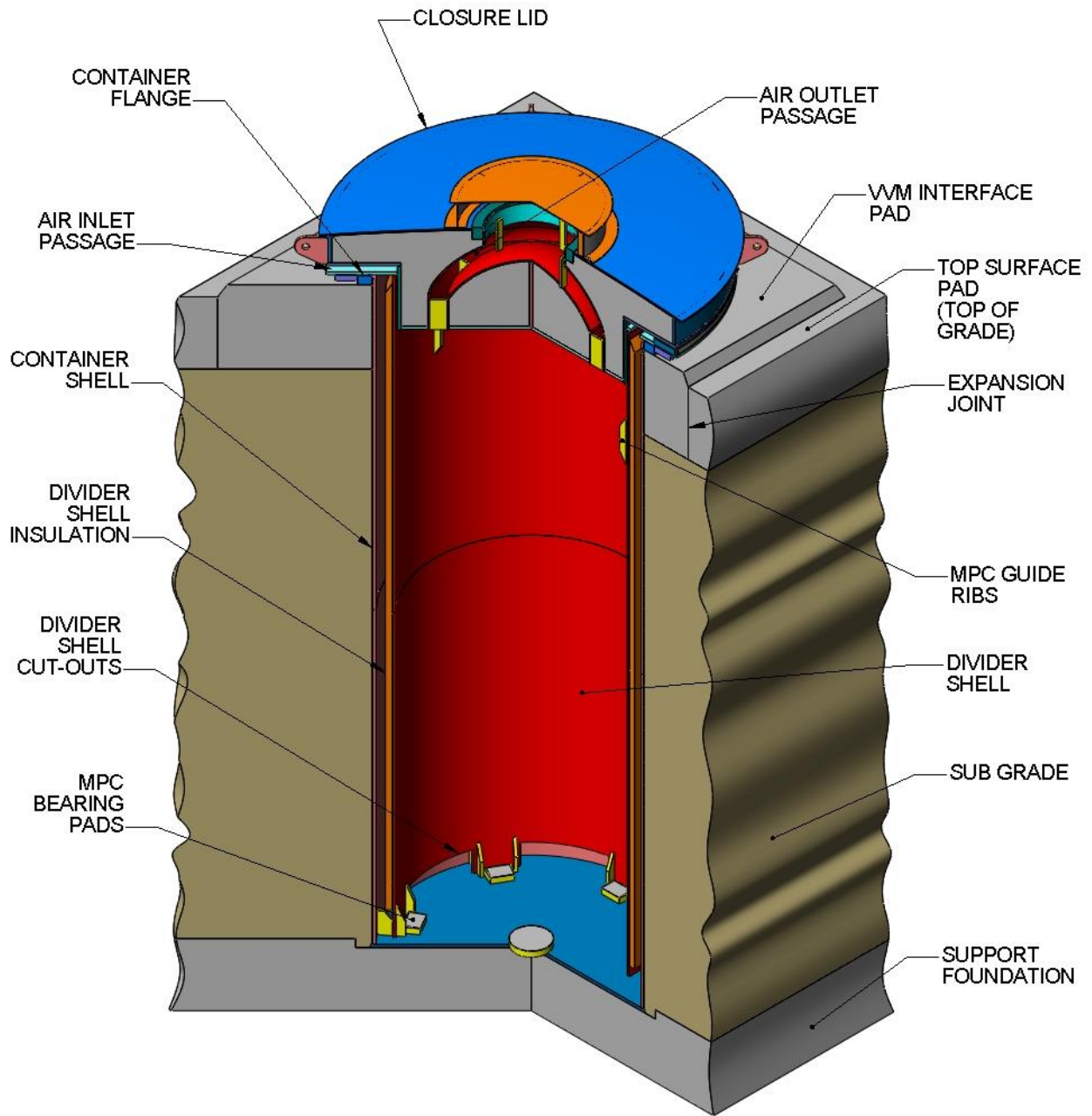


FIGURE 1.I.2: CUT-AWAY VIEW OF THE HI-STORM 100U VVM

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

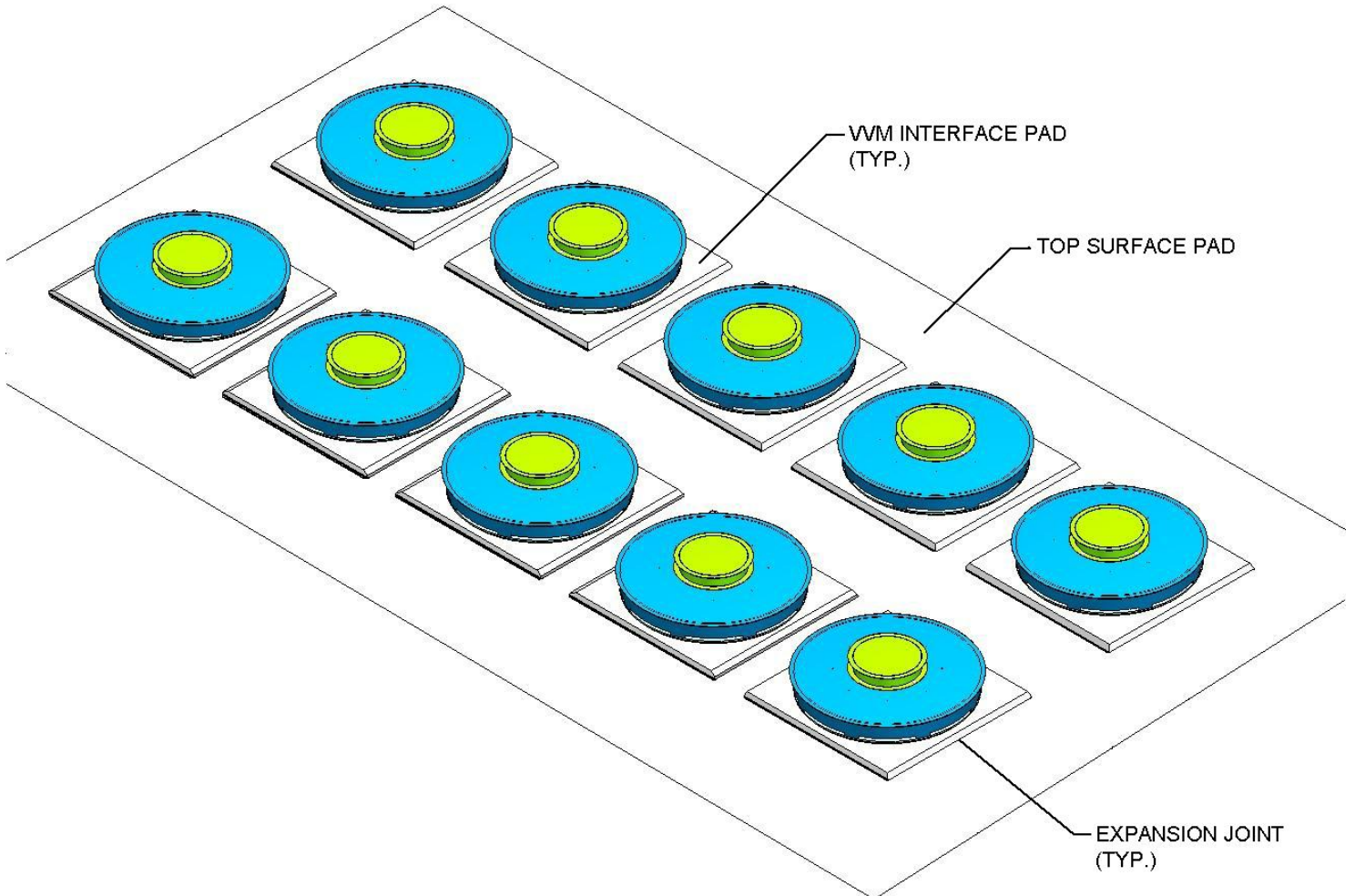


FIGURE 1.1.3: TYPICAL HI-STORM 100U SYSTEM ISFSI 2 x 5 ARRAY

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.1.5.

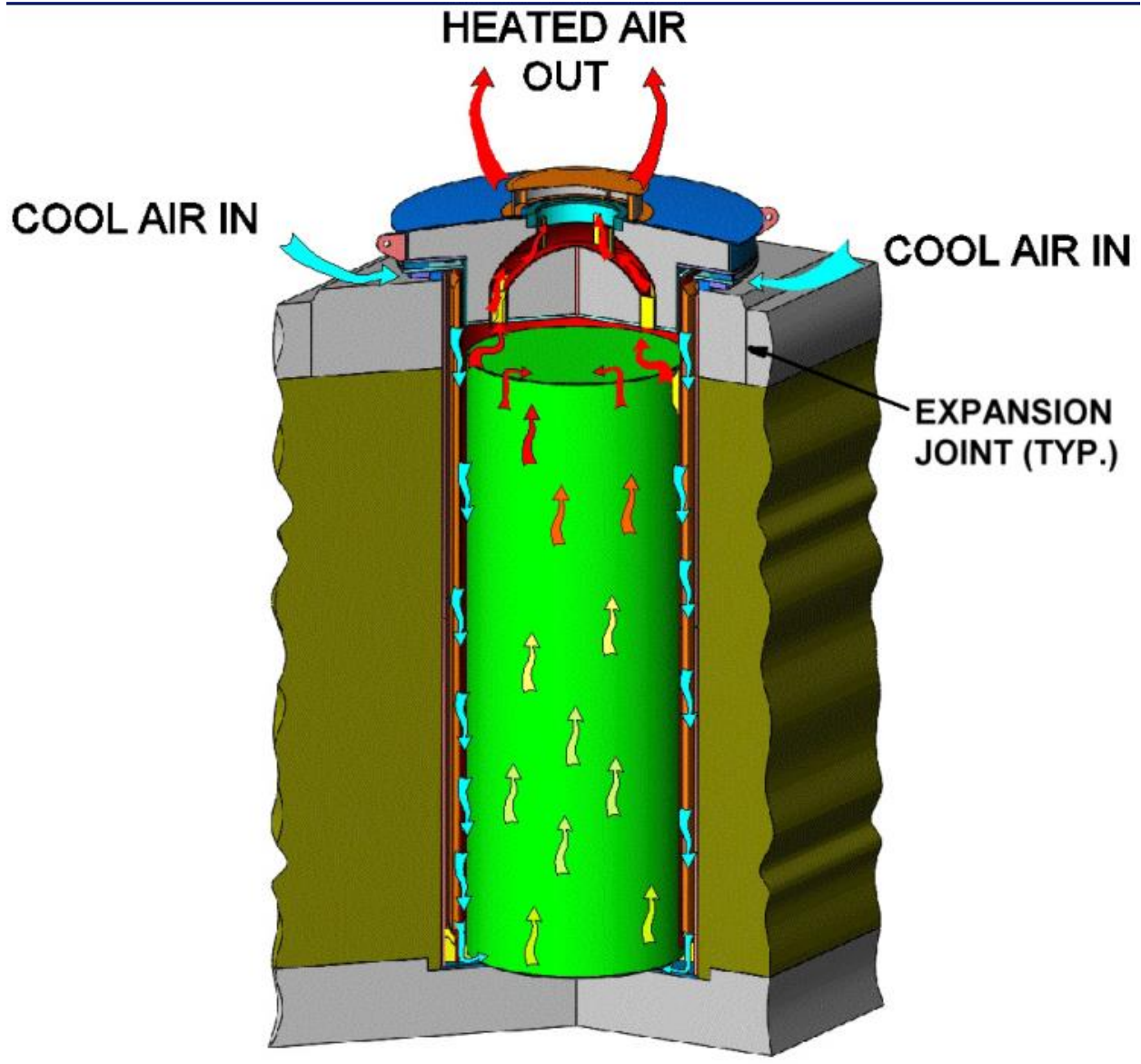


FIGURE 1.I.4: HI-STORM 100U SYSTEM AIR FLOW PATTERN

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

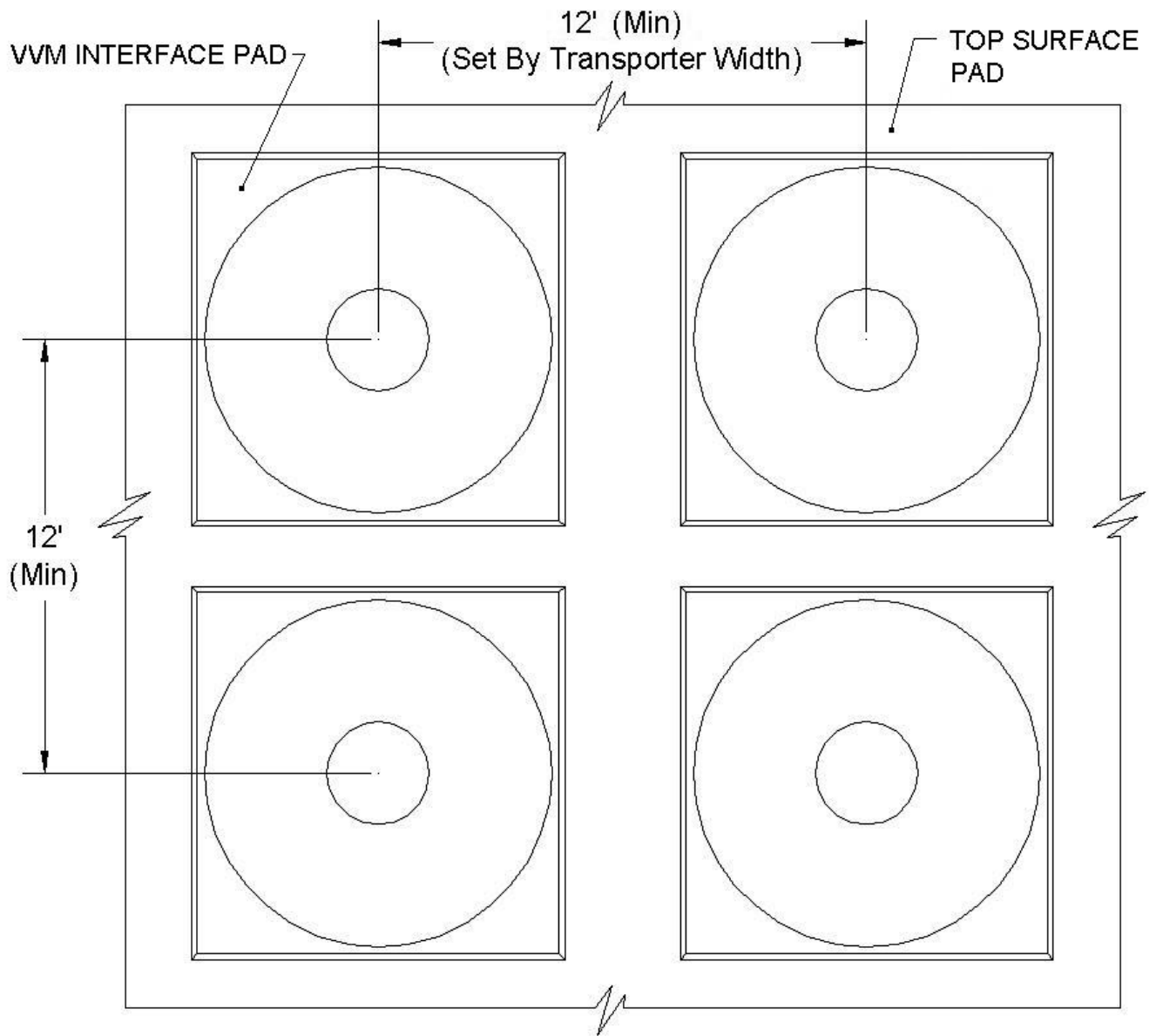


FIGURE 1.1.5: PLAN VIEW OF A 2X2 HI-STORM 100U SYSTEM STORAGE ARRAY

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.1.5.

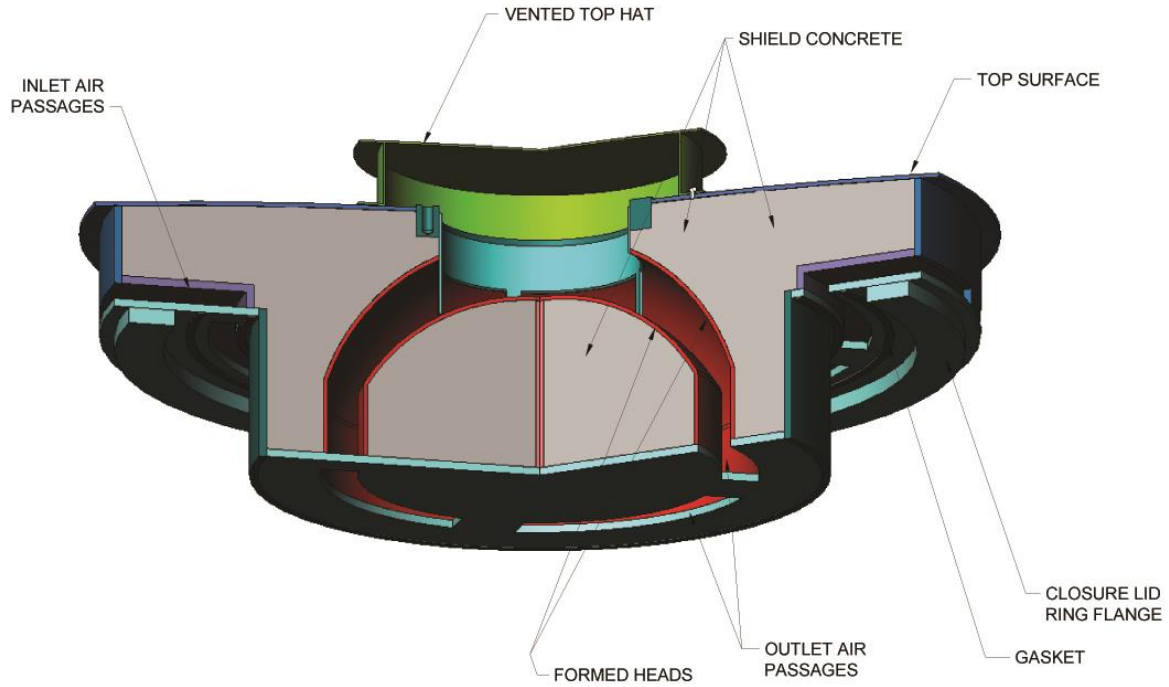


FIGURE 1.I.6; HI-STORM 100U VVM CLOSURE LID GENERAL ARRANGEMENT (SHOWN IN CUT-AWAY VIEW)

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



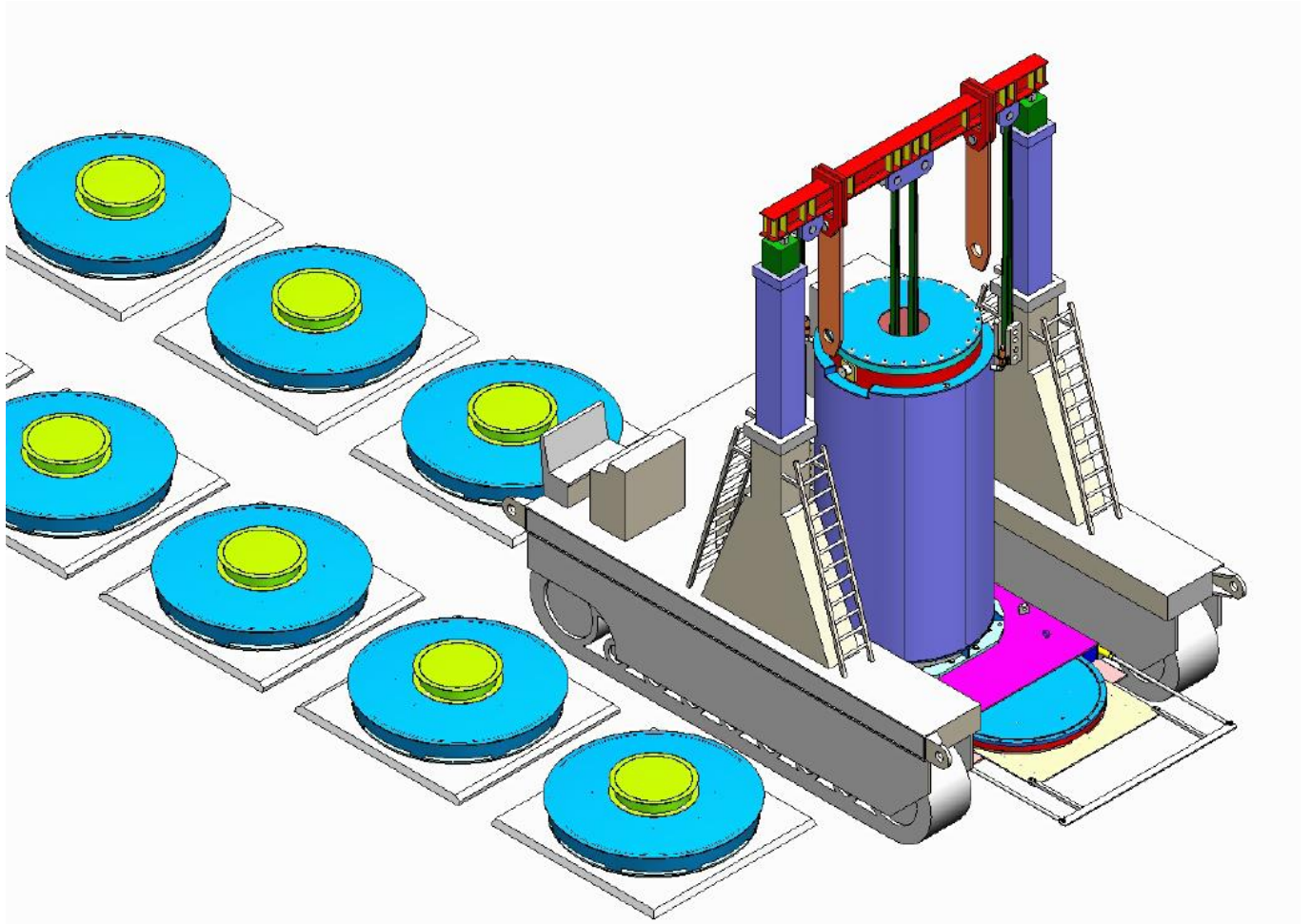


FIGURE 1.1.7; MPC TRANSFER IN A HI-STORM 100U VVM USING A VERTICAL CASK TRANSPORTER

Note: The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Minor details of the HI-STORM 100U depicted here may vary slightly from the licensing drawings in Subsection 1.1.5.

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

### 1.III.0      GENERAL INFORMATION

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

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

### 1.III.1      INTRODUCTION

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

NUREG-1536 Revision 1 Section 8.4.17 provides guidance on fuel classification which is focused on the functional and performance based requirements of the fuel assembly. The following definitions, consistent with this guidance, are added and are specific to the “III” Supplements:

**Undamaged Fuel Assembly** is defined as: a) a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial enrichment of 3.3 wt% U-235, without known or suspected **grossly breached spent fuel rods**, and which can be handled by normal means. An **undamaged fuel assembly** may be a **repaired/reconstituted fuel assembly**.

**Grossly breached spent fuel rod** is a spent nuclear fuel rod with a cladding defect that could lead to the release of fuel particulate greater than the average size fuel fragment for that particular assembly. A gross cladding breach may be confirmed by visual examination, through a review of reactor operating records indicating the presence of heavy metal isotopes, or other acceptable inspection means.

**Repaired/reconstituted fuel assembly** is a spent nuclear fuel assembly which contains dummy fuel rod(s) that displaces an amount of water greater than or equal to the original fuel rod(s) and/or which contains structural repairs so it can be handled by normal means.

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

#### 1.III.2.1      MPC-68M Characteristics

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

- i. The fuel basket is assembled from a rectilinear gridwork of plates so that there are no bends or radii at the cell corners. This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls that must transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (e.g., non-mechanistic tipover, uncontrolled lowering of a cask during on-site transfer, or off-site transport accident events, etc.).
- ii. Extruded aluminum shims (the so-called “*Basket Shims*”) are installed in the peripheral space between the fuel basket and the Enclosure Vessel to provide conformal contact surfaces between the Basket Shims and the fuel basket and between the Basket Shims and the Enclosure Vessel shell. The axial holes in the corner basket shims serve as the passageway for the downwards flow of the helium gas under the thermosiphon action, which is intrinsic to the thermal performance of all MPCs in the HI-STORM 100 system.
- iii. To facilitate an effective convective circulation inside the Canister, the Enclosure Vessel is pressurized to the same pressure level specified for the MPC-68 model. See Table 1.III.1.
- iv. The fuel basket consists of adjacent square openings (cells) separated by one wall of neutron absorber.

The mechanical and thermophysical properties of Metamic-HT presented in Section 1.III.2.4 are supported by the Metamic-HT Sourcebook[1.III.3]. The basket shims are made from creep resistant aluminum alloy similar to those used in HI-STAR 180 (Docket No. 71-9325) [2.III.6.2]. All design aspects of the MPC enclosure vessel (confinement boundary), including the vent and drain port arrangements in the MPC lid, are unchanged. The MPC-68M may be loaded in all licensed aboveground HI-STORM 100 overpacks.

#### 1.III.2.2 Operational Characteristics

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

All short-term operations, including draining of the MPC, welding of the lid, drying and filling of the MPC cavity with inert gas, and handling of the MPC remain unchanged from the existing practice, except



that the dried and helium filled MPC-68M may reside indefinitely in the HI-TRAC transfer cask without the aid of the supplemental cooling system. The increased thermal conductivity of the Metamic-HT basket maintains the steady state fuel cladding temperatures below the ISG-11, Revision 3 [2.0.8] limits; thus eliminating the need for the supplemental cooling system. All other loading and unloading procedures and operations described in Chapter 8 apply to MPC-68M with the clarifications and limitations presented in Supplement 8.III.

#### 1.III.2.2.1 Criticality Prevention

Metamic-HT is the designated neutron absorber and structural material in the MPC-68M. The properties of Metamic-HT and key characteristics, necessary for ensuring nuclear reactivity control, thermal, and structural performance of the basket, are presented in Section 1.III.2.4.

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

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

During storage, criticality control measures beyond the integral neutron poison in the storage cell walls are not necessary because of the low reactivity of the fuel in the dry helium-filled canister and the design features that prevent water from intruding into the canister fuel storage space.

#### 1.III.2.2.2 Structural Considerations

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

#### 1.III.2.2.3 Thermal Considerations

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

#### 1.III.2.2.4 Shielding Considerations

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

### 1.III.2.3 Cask Contents

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

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## (b) Thermo-Physical Properties of Metamic-HT

Metamic-HT was subject to an exhaustive test program prior to its certification by the USNRC in the HI-STAR 180 transport package. The results of the tests provide the authoritative information on the characteristics of Metamic-HT and are archived in the Metamic-HT Sourcebook [1.III.3] which is the source of much of the information summarized in this sub-section. As defined in this SAR, a *critical characteristic* of a material is one that is significant to its safety function. For purposes of manufacturing control, those *critical characteristics* that play a direct role in the ability of the fuel basket to render an essential safety function and are subject to minor variation in production are defined as Primary properties. Those properties that are not used in any of the safety analyses in this SAR are termed “ancillary” properties. Finally, those properties that bear upon the safety analysis but are essentially non-variant (such as density, heat capacity etc.) are denoted as “Invariant” properties. Primary properties are differentiated from the ancillary and invariant properties by the following two key characteristics which require them to be subjected to an extensive statistical in-production testing protocol (set down in Chapter 9).

- (i) They are used in the safety analysis simulations, and
- (ii) They are subject to lot-to-lot variability due to variation in the manufacturing parameters.

Thermal and inertia properties of Metamic-HT, such as density, heat capacity, thermal expansion coefficient, thermal conductivity and emissivity are stable properties that are not subject to significant variations among different lots in the production cycle. Therefore, following the practice similarly used in the ASME codes, such properties are treated as “invariant” and their reference values are excerpted from the Sourcebook and provided in Table 1.III.3 for analysis purposes. Properties that are not used in this safety analysis are elongation, charpy impact strength and lateral expansion. In accordance with the definition above these are ancillary properties.

#### **Primary Properties**

Tensile strength, yield strength, young’s modulus and percent area reduction are properties that directly bear upon the structural strength of the basket. Therefore, their minimum guaranteed values (MGVs) specified in this report must be used in the strength and deflection analyses performed on the basket and reported in Supplement 3.III herein. The specified MGVs are subject to a programmatic confirmatory test regimen during production as described in Supplement 9.III.

Young’s modulus is the slope of the stress/strain curve in the elastic range of the material. The deflection of a structure, such as a laterally loaded beam, is inversely proportional to the material’s Young’s modulus (Classical Kirchoff’s Equation). Because the structural acceptance criterion for the Metamic-HT panels is deflection based, the lower bound value of E (i.e. MGV) should be used in the analysis models.

Area reduction measured in the tensile test coupons enables the “true strain” to be correlated to the “engineering strain”. This property is needed for defining the stress-strain curve of the material in the structural analysis model.

The MGVs of all primary properties identified in the foregoing are provided in Table 1.III.2 and are either equal to or less than their corresponding values derived from the test program [1.III.3].

Finally, as is well known in the literature [1.III.10], the yield and ultimate strength of aluminum alloys, like all metals, may be substantially greater than the static values (reported as MGVs) because of dynamic effects that arise from rapid strain rates associated with impactive or impulsive loads. For conservatism, contrary to the standard practice for stainless steel baskets, *credit for strain rate effects on Metamic-HT's strength properties has not been taken in this SAR*. This additional conservatism has been deliberately incorporated to amplify the computed strains and deflections in the basket panels, which has the direct effect of reducing the computed safety margins.

#### **Ancillary Properties**

Ancillary properties are those that are not required in the safety analysis in this FSAR. Properties that are not used in this safety analysis are elongation, charpy impact strength and lateral expansion. Elongation is not utilized in the structural model because it is a redundant property, subsumed in the mechanical properties discussed above. Therefore, it is an ancillary property.

Creep of the basket panel under extended lateral loading from the basket under normal service condition was another subject of investigation in the Metamic-HT qualification program. If the basket is horizontal (which is the case under transport) then the lateral deflection of the basket panels due to creep would impact the criticality control characteristics. Therefore, it is necessary to ensure that the total deformation due to creep will remain within a specified (tolerable) limit. To make the generic safety determination for Metamic-HT panels in the fuel basket, creep tests were carried out for a significant length of time (approximately 20,000 hours) at different temperatures and stress levels to permit a correlation to be developed. As shown in the Metamic-HT sourcebook, the design basis creep equation developed therein bounds all creep data from the test program with large margins. Because the fuel basket is always in the vertical orientation in HI-STORM, creep is not an operative mechanism for basket deformation. Therefore, creep strength qualifies as an ancillary property.

The data on charpy strength and lateral expansion were compiled in the Metamic-HT evaluation program (2005-2008) to study crack propagation in material containing internal flaws or pre-existing defects to ensure that any flaw in the Metamic-HT panels will not grow and propagate due to stress encountered during service. A bounding crack propagation study in the Metamic-HT Sourcebook using lowerbound properties from the test program confirmed that the Metamic-HT panels in the baskets are not vulnerable to crack propagation under transport service conditions (which entails repetitive impactive loadings). In storage, where there are no significant impactive loads under any of the operating modes,  $C_v$  and  $\lambda$  are not germane to any safety analysis in this FSAR making them as ancillary properties (i.e. not essential to

safety analysis). Authoritative test based values of the above ancillary properties can be obtained from the Metamic-HT sourcebook [1.III.3], if needed.

### **Invariant Properties**

Thermal and inertia properties of Metamic-HT, such as density, heat capacity, thermal expansion coefficient, thermal conductivity and emissivity are stable properties that are not subject to significant variations among different lots in the production cycle. Therefore, following the practice similarly used in the ASME codes, such properties are treated as “invariant” and their reference values are excerpted from the Sourcebook and provided in Table 1.III.3 for analysis purposes. The significance of invariant properties in the thermal design of Metamic-HT equipped casks is evaluated below.

#### **Thermal Conductivity and Emissivity**

The thermal performance of the cask is a direct function of emissivity and thermal conductivity properties. Therefore, to obtain a lowerbound thermal performance, it is necessary to define a lowerbound value for analysis. To this end suitable lowerbound emissivity and conductivity values supported by Metamic-HT test data are adopted in Table 1.III.3. The use of lowerbound thermal conductivity and emissivity in the thermal analysis ensures that for any given heat load, the computed metal temperatures will be upperbound values. Because the mechanical strength properties of metals decrease with increasing temperature, the over prediction of temperatures in the thermal analysis also ensures conservatism in the structural analysis by requiring the use of smaller-than-actual strength values.

#### **Specific Gravity, Coefficient of Thermal Expansion and Specific Heat**

As justified below these properties have a second order effect on the thermal design of Metamic-HT equipped casks.

Specific gravity is used to establish the weight of the cask. Because the fuel basket is constructed of a light weight material (90% is aluminum) its weight is a small fraction of the host cask. Minor variations in specific gravity do not affect the weight of the cask. Therefore, the use of nominal (average) specific gravity of Metamic-HT is appropriate in cask design and analysis. Accordingly, suitable mean specific gravity properties are computed from test data and tabulated in Table 1.III.3.

Coefficient of Thermal Expansion is used to compute the growth of the basket and to ensure that a condition of “restraint of free expansion” would not exist. Because the clearances engineered in the cask design are quite liberal, as a practical matter, restraint of free expansion is not an issue. Therefore, the use of a mean value in cask design work is appropriate. Accordingly, suitable mean thermal expansion properties are computed from test data and tabulated in Table 1.III.3.

Specific Heat is used in the thermal transient analysis under a fire event. Even in this case, variation in  $C_p$  will have a minor effect on the results because of the small thermal inertia of the fuel basket relative to other items such as the stored fuel in the cask. Therefore the Specific Heat value of Metamic-HT should be pegged to the average measured value from test data. Accordingly, suitable mean heat capacity properties are computed from test data and tabulated in Table 1.III.3.

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(e) Welding of Metamic-HT

Welding of Metamic-HT is conducted in accordance with Holtec approved procedures. As Metamic-HT is a non-code material, weld requirements are specified to meet the required safety and structural function of the basket. Weld acceptance criteria is stated in the licensing drawings provided in Section 1.5. In accordance with practice of joining metal-matrix composites, Friction Stir Welding (FSW) technique was adopted and qualified by Holtec for welding Metamic-HT. This process yields stronger joint strength on a repeatable basis compared to classical welding methods such as metal inert gas or tungsten inert gas welding. The FSW process implemented in the manufacturing of Metamic-HT basket require that the welding procedure and welders are appropriately qualified. This is accomplished by tensile testing of weld coupons (emulating ASME Section IX) to meet the weld strength requirements and conducting radiography to verify soundness of the weld. The above procedure qualification protocol, provided in the Metamic-HT Manufacturing Manual [1.III.4], has been established to accord with the unique bonding characteristics of Metamic-HT and to ensure that the required minimum joint strength is realized with full assurance in the production of the fuel baskets.

1.III.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Same as in Section 1.3.

1.III.4 GENERIC CASK ARRAYS

Same as in Section 1.4.

1.III.5 DRAWINGS

[PROPRIETARY DRAWINGS WITHHELD PER 10 CFR 2.390]

1.III.6 REFERENCES

[PROPRIETARY TEXT REMOVED PER 10 CFR 2.390]

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PROPRIETARY TEXT REMOVED PER 10 CFR 2.390

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

	BWR
MPC internal environment Helium fill (99.995% fill helium purity)  (heat load $\leq$ 28.19 kW)  (heat load $>$ 28.19 kW) Quarter Symmetric Heat Load (QSHL)	(all pressure ranges are at a reference temperature of 70°F)  > 29.3 psig and < 48.5 psig OR 0.1218 +/-10% g-moles/liter  > 45.5 psig and < 48.5 psig $\geq$ 43.5 psig and $\leq$ 46.5 psig
B <sub>4</sub> C content in Metamic-HT (wt. %)	As specified on drawing in Section 1.5

Table 1.III.2

[PROPRIETARY TABLE WITHHELD PER 10 CFR 2.390]

**Table 1.III.3**  
Invariant Properties of Metamic-HT Used in Safety Analysis

	Property	Temperature, °C	Property Value (Note 1)	Property Type
1.	Thermal conductivity, k (W/m <sup>°k</sup> )	Ambient 200/300/350 450/500	180/180/180/180 180/180	Invariant
2.	Emissivity (dimensionless), e	150≤T≤500	See Note 2	Invariant
3.	Nominal specific gravity, (dimensionless) s	Ambient	2.705	Invariant
4.	Average thermal expansion, coefficient, Γ (°C <sup>-1</sup> )	30≤T≤500	21x10 <sup>-6</sup>	Invariant
5.	Specific Heat, C <sub>p</sub> (J/kg-°C)	100/200/350 350≤T≤500	823.3/914.1/1024.2 Note 3	Invariant

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

#### Note 2: Emissivity Equation

Either dark gray or ebony coat anodizing are acceptable options. The thermal evaluations in Chapter 4 of this SAR use the lowerbound Dark Grey specification.

#### Dark Gray Anodized Metamic-HT

T: Temperature (°F)

$$e = 0.2 + 0.6 \sin[\pi(T-100)/1304] \quad (100^\circ\text{F} \leq T \leq 752^\circ\text{F})$$

$$e = 0.8 \quad (T > 752^\circ\text{F})$$

#### Ebony Coat Anodized Metamic-HT

T: Temperature (°F)

$$e = 0.35 + 1.541\text{E-}03 * (T - 100) \quad (100^\circ\text{F} \leq T \leq 392^\circ\text{F})$$

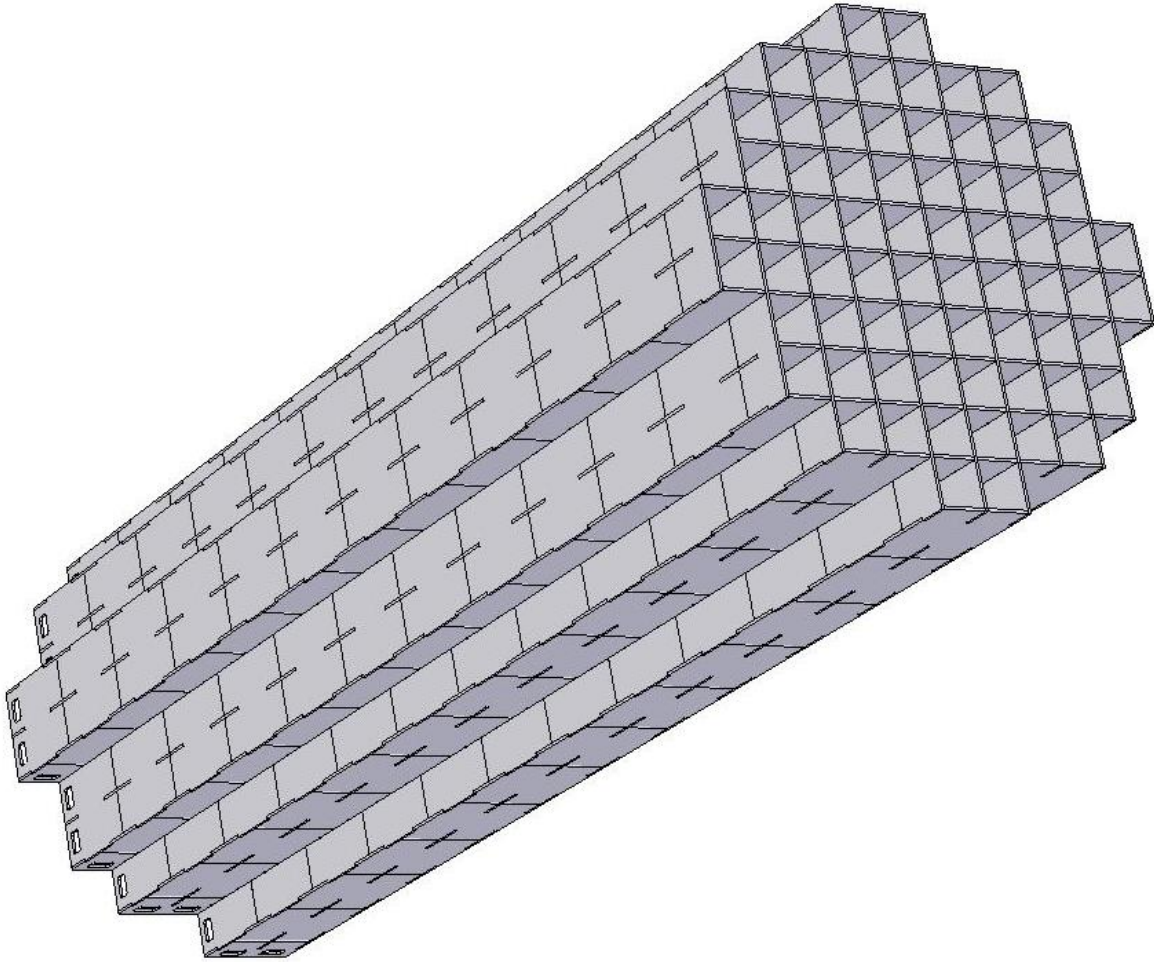
$$e = 0.8 + 2.777\text{E-}04 * (T - 392) \quad (392^\circ\text{F} < T \leq 752^\circ\text{F})$$

$$e = 0.9 \quad (T > 752^\circ\text{F})$$

#### Note 3: Heat Capacity Function

T: Temperature (°C)

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



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

				<b>1</b>	<b>2</b>				
		<b>3</b>	<b>4</b>	<b>5</b>	<b>6</b>	<b>7</b>	<b>8</b>		
<b>9</b>	<b>10</b>	<b>11</b>	<b>12</b>	<b>13</b>	<b>14</b>	<b>15</b>	<b>16</b>		
<b>17</b>	<b>18</b>	<b>19</b>	<b>20</b>	<b>21</b>	<b>22</b>	<b>23</b>	<b>24</b>		
<b>25</b>	<b>26</b>	<b>27</b>	<b>28</b>	<b>29</b>	<b>30</b>	<b>31</b>	<b>32</b>	<b>33</b>	<b>34</b>
<b>35</b>	<b>36</b>	<b>37</b>	<b>38</b>	<b>39</b>	<b>40</b>	<b>41</b>	<b>42</b>	<b>43</b>	<b>44</b>
	<b>45</b>	<b>46</b>	<b>47</b>	<b>48</b>	<b>49</b>	<b>50</b>	<b>51</b>	<b>52</b>	
	<b>53</b>	<b>54</b>	<b>55</b>	<b>56</b>	<b>57</b>	<b>58</b>	<b>59</b>	<b>60</b>	
		<b>61</b>	<b>62</b>	<b>63</b>	<b>64</b>	<b>65</b>	<b>66</b>		
				<b>67</b>	<b>68</b>				

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

## **APPENDIX 1.III.A: Intentionally Deleted**

## **APPENDIX 1.III.B: Intentionally Deleted**

## CHAPTER 2<sup>†</sup>: PRINCIPAL DESIGN CRITERIA

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

This chapter is in full compliance with NUREG-1536, except for the exceptions and clarifications provided in Table 1.0.2. Table 1.0.2 provides the NUREG-1536 review guidance, the justification for the exception or clarification, and the Holtec approach to meet the intent of the NUREG-1536 guidance.

### 2.0 PRINCIPAL DESIGN CRITERIA

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

#### 2.0.1 MPC Design Criteria

##### General

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

##### Structural

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

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



welds to the MPC lid and shell, as discussed in Section 2.2.4. In addition, the threaded holes in the MPC lid are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for critical lifts to facilitate vertical MPC transfer.

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

The MPC closure welds are partial penetration welds that are structurally qualified by analysis, as presented in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination of the root pass and/or final weld surface (if more than one weld pass was required), in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid weld is further insured by the non-destructive tests set forth in Chapter 9 and by a rigorous stress analysis described in Chapter 3.

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

### Thermal

The design and operation of the HI-STORM 100 System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.8]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

- i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.
- ii. The maximum value of the *calculated* temperature for all CSF (including ZR and stainless steel fuel cladding materials) under long-term normal conditions of storage must remain below 400°C (752°F). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400°C (752°F) for high burnup fuel and 570°C (1058°F) for moderate burnup fuel.
- iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570°C (1058°F).
- iv. For High Burnup Fuel (HBF), operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65°C (117°F).

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

- i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400°C (752°F). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570°C (1058°F) because fuel cladding stress is shown to be less than approximately 90 MPa per Reference [2.0.9]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions added to ensure these limits are met (see Section 4.5).
- ii. For MPCs containing at least one high burnup fuel (HBF) assembly or if the MPC heat load is greater than the threshold heat load defined in Table 4.5.1, the forced helium dehydration (FHD) method of MPC cavity drying must be used to meet the normal operations PCT limit and satisfy the 65°C temperature excursion criterion for HBF.
- iii. The off-normal and accident condition PCT limit remains unchanged (1058°F).
- iv. For MPCs loaded with one or more high burnup fuel assemblies and the MPC heat load is greater than threshold heat load defined in Table 4.5.4, the Supplemental Cooling System (SCS) is required to ensure fuel cladding temperatures remain below the applicable temperature limit (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C.

The MPC cavity is dried using either a vacuum drying system, or a forced helium dehydration system (see Appendix 2.B). The MPC is backfilled with 99.995% pure helium in accordance with the limits in Table 1.2.2 during canister sealing operations to promote heat transfer and prevent cladding degradation.

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

The MPCs are designed for a bounding thermal source term, as described in Section 2.1.6. The maximum allowable fuel assembly heat load for each MPC is limited as specified in Section 2.1.9.

Each MPC model, except MPC-68F, allows for two fuel loading strategies. The first is uniform fuel loading, wherein any authorized fuel assembly may be stored in any fuel storage location up to a maximum specific heat emission rate, subject to other restrictions, such as location requirements for damaged fuel containers (DFCs) and fuel with integral non-fuel hardware (e.g., APSR). The second is regionalized fuel loading, wherein the basket is segregated into two regions. Regionalized loading allows for storage of fuel assemblies with higher heat emission rates than would otherwise be authorized for uniform loading. Regionalized loading strategies must also comply with other

requirements, such as those for DFCs and non-fuel hardware. Specific fuel assembly cooling time, burnup, and decay heat limits for regionalized loading are presented in Section 2.1.9. The two fuel loading regions are defined by fuel storage location number in Table 2.1.27 (refer to Figures 1.2.2 through 1.2.4). For MPC-68F, only uniform loading is permitted.

### Shielding

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

Compliance with these regulations for a single cask and several representative cask arrays is demonstrated in Chapters 5 and 10.

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

The MPCs are designed for design basis fuel as described in Sections 2.1.7 and 5.2. The radiological source term for the MPCs is limited based on the burnup and cooling times specified in Section 2.1.9. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation, as discussed in Chapter 10.

### Criticality

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

Criticality control is maintained by the geometric spacing of the fuel assemblies, fixed borated neutron absorbing materials incorporated into the fuel basket assembly, and, for certain MPC models, soluble boron in the MPC water. The minimum specified boron concentration verified during neutron absorber manufacture is further reduced by 25% for criticality analysis for Boral-equipped MPCs and by 10% for METAMIC<sup>®</sup>-equipped MPCs. No credit is taken for burnup. The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC is limited. Enrichment limits and soluble boron concentration requirements are delineated in Section 2.1.9 consistent with the criticality analysis described in Chapter 6.

### Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions. As discussed in Section 7.1, the Holtec MPC design meets the guidance in Interim Staff Guidance 18 to classify confinement boundary leakage as non-credible. Therefore, no confinement dose analysis is performed. The confinement function of the

MPC is verified through appropriate non-destructive examinations performed in accordance with the acceptance test program in Chapter 9.

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

### Operations

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

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

### Acceptance Tests and Maintenance

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

### Decommissioning

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

## 2.0.2 HI-STORM Overpack Design Criteria

### General

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

### Structural

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

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

Unlike other concrete storage casks, the HI-STORM overpack concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates. Where typical concrete storage casks are reinforced by rebar, the HI-STORM overpack is supported by the inner and outer shells connected by radial ribs. As the HI-STORM overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete. Providing further conservatism, the structural analyses for normal conditions demonstrate that the allowable stress limits of the structural steel are met even with no credit for the strength of the concrete. During accident conditions (e.g., tornado missile, tip-over, end drop, and earthquake), only the compressive strength of the concrete is accounted for in the analysis to provide an appropriate simulation of the accident condition. Where applicable, the compressive strength of the concrete is calculated in accordance with ACI-318.1-89 (92) [2.0.1].

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

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

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the overpack must protect the MPC from deformation, provide continued adequate performance, and allow the retrieval of the MPC under all conditions. These design loadings include a postulated drop accident from the maximum allowable handling height, consistent with the analysis described in Section 3.4.10. The load combinations for which the overpack is designed are defined in Section 2.2.7. The physical characteristics of the MPCs for which the overpack is designed are defined in Chapter 1.

### Thermal

The allowable long-term through-thickness section average temperature limit for the overpack concrete is established in accordance with Paragraph A.4.3 of Appendix A to ACI 349, which allows the use of elevated temperature limits if test data supporting the compressive strength is available and an evaluation to show no concrete deterioration provided. Appendix 1.D specifies the cement and aggregate requirements to allow the utilization of the temperature limits in Table 2.2.3. The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in Table 2.2.3.

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

### Section 3.1.2.3.

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

#### Shielding

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

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

#### Confinement

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

#### Operations

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

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

#### Acceptance Tests and Maintenance

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

### Decommissioning

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

#### 2.0.3 HI-TRAC Transfer Cask Design Criteria

##### General

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

##### Structural

The HI-TRAC transfer cask includes both structural and non-structural biological shielding components that are classified as important to safety. The structural steel components of the HI-TRAC, with the exception of the lifting trunnions, are designed and fabricated in accordance with the applicable requirements of Section III, Subsection NF, of the ASME Code with certain NRC-approved alternatives, as discussed in Section 2.2.4. The lifting trunnions and associated attachments are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for non-redundant lifting devices.

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

##### Thermal

The allowable temperatures for the HI-TRAC transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The top lids of the HI-TRAC 125 and HI-TRAC 125D incorporate Holtite-A shielding material. This material has a maximum allowable temperature in accordance with the manufacturer's test data. The specific allowable temperatures for the structural steel and shielding

components of the HI-TRAC are provided in Table 2.2.3. The HI-TRAC is designed for off-normal environmental cold conditions, as discussed in Section 2.2.2.2. The structural steel materials susceptible to brittle fracture are discussed in Section 3.1.2.3.

The HI-TRAC is designed for the maximum heat load analyzed for storage operations. When the MPC contains any high burnup fuel assemblies and the MPC decay heat is greater than the threshold heat load defined in Table 4.5.4, the Supplemental Cooling System (SCS) will be required for certain time periods while the MPC is inside the HI-TRAC transfer cask (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C. The HI-TRAC water jacket maximum allowable temperature is a function of the internal pressure. To preclude over pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is limited to less than the saturation temperature at the shell design pressure. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the minimum allowable temperature and adding ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1.6. The working area ambient temperature limit for loading operations is limited in accordance with the design criteria established for the transfer cask.

### Shielding

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

HI-TRAC dose rates are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

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

The HI-TRAC provides a means to isolate the annular area between the MPC outer surface and the HI-TRAC inner surface to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC surfaces expected to require



decontamination are coated. The maximum permissible surface contamination for the HI-TRAC is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

### Confinement

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

### Operations

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

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

### Acceptance Tests and Maintenance

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

### Decommissioning

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

#### 2.0.4 Principal Design Criteria for the ISFSI Pad

##### 2.0.4.1 Design and Construction Criteria

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

safety.

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

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

Because tipover of a cask installed in an anchored configuration is not feasible, the pad does not need to be engineered to accommodate this non-mechanistic event. However, the permissible carry height for a loaded HI-STORM 100A overpack must be established for the specific ISFSI pad using the methodology described in this FSAR, if the load handling device is not designed in accordance with ANSIN 14.6 and does not have redundant drop protection design features. These requirements are specified in the CoC. However, to serve as an effective and reliable anchor, the pad must be made appropriately stiff and suitably secured to preclude pad uplift during a seismic event.

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

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

A complete listing of design and construction requirements for an ISFSI pad on which an anchored

HI-STORM 100A will be deployed is provided in Appendix 2.A. A sample embedment design is depicted in Figure 2.A.1.

#### 2.0.4.2 Applicable Codes

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

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

##### a. Definitions

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

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

b. Load Combinations for the Concrete Pad

The notation and acceptance criteria of NUREG-1536 apply.

Normal Events

$$U_c > 1.4D + 1.7L$$

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

Off-Normal Events

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

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

Accident-Level Events

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

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

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

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

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

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

Normal Events

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

Off-Normal Events

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

Accident-Level Events

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

c. Load Combination for the Anchor Studs

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

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

### 2.0.4.3 Limiting Design Parameters

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

In summary, the requirements for the ISFSI pad for free-standing and anchored HI-STORM deployment are similar with a few differences. Table 2.0.5 summarizes their commonality and differences in a succinct manner with the basis for the difference fully explained.

#### 2.0.4.4 Anchored Cask/ISFSI Interface

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

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Table 1.2.2
License	20 yrs.	10CFR72.42(a) and 10CFR72.236(g)	-
<b>Structural:</b>			
Design Codes:			
Enclosure Vessel	ASME Code, Section III, Subsection NB	10CFR72.24(c)(4)	Section 2.0.1
Fuel Basket	ASME Code, Section III, Subsection NG for core supports (NG-1121)	10CFR72.24(c)(4)	Section 2.0.1
MPC Fuel Basket Supports (Angled Plates)	ASME Code, Section III, Subsection NG for internal structures (NG-1122)	10CFR72.24(c)(4)	Section 2.0.1
Basket Shims (for MPC-32 only)	ASME Code, Section III, Subsection NG for internal structures (NG-1122)	10CFR72.24(c)(4)	Section 2.0.1
MPC Lifting Points	NUREG-0612 & Regulatory Guide 3.61	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights <sup>†</sup> :			
Max. Loaded Canister (dry)	90,000 lb.	R.G. 3.61	Table 3.2.1
Empty Canister (dry)	42,000 lb. (MPC-24) 45,000 lb. (MPC-24E/EF) 39,000 lb. (MPC-68/68F/68FF) 36,000 lb. (MPC-32)	R.G. 3.61	Table 3.2.1
Design Cavity Pressures:			
Long-Term Normal:	100 psig	ANSI/ANS 57.9	Section 2.2.1.3

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



Type	Criteria	Basis	FSAR Reference
Short-Term Normal:	110 psig	ANSI/ANS 57.9	Section 2.2.1.3
Off-Normal:	110 psig	ANSI/ANS 57.9	Section 2.2.2.1
Accident (Internal)	200 psig	ANSI/ANS 57.9	Section 2.2.3.8
Accident (External)	60 psig	ANSI/ANS 57.9	Sections 2.2.3.6 and 2.2.3.10
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(l)	Section 2.0.1
<b>Thermal:</b>			
Maximum Design Temperatures:			
Structural Materials:			
Stainless Steel (Normal)	725° F	ASME Code Section II, Part D	Table 2.2.3
Stainless Steel (Accident)	1058° F	See Subsection 2.2.2.3	Table 2.2.3
Neutron Poison:			
Neutron Absorber (normal)	800° F	See Table 4.3.1 and Subsection 1.2.1.3.1	Table 2.2.3
Neutron Absorber (accident)	1058° F	See Table 4.3.1 and Subsection 1.2.1.3.1	Table 2.2.3
Canister Drying	$\leq 3$ torr for $\geq 30$ minutes (VDS)  $\leq 21^{\circ}\text{F}$ exiting the demoisturizer for $\geq 30$ minutes or a dew point of the MPC exit gas $\leq 22.9^{\circ}\text{F}$ for $\geq 30$ minutes(FHD)	NUREG-1536, ISG-11, Rev. 3	Section 4.5, Appendix 2.B
Canister Backfill Gas	Helium	-	Section 4.4
Canister Backfill	Varies (see Table 1.2.2)	Thermal Analysis	Section 4. 4
Fuel cladding temperature limit for long term storage conditions	752 °F (400 °C)	ISG-11, Rev. 3	Section 4.3

Type	Criteria	Basis	FSAR Reference
Fuel cladding temperature limit for normal short-term operating conditions (e.g., MPC drying and onsite transport)	752 °F (400 °C), except certain MPCs containing all moderate burnup fuel (MBF) may use 1058°F (570°C) for normal short-term operating conditions	ISG-11, Rev. 3	Sections 4.3 and 4.5
Fuel cladding temperature limit for Off-Normal and Accident Events	1058° F (570 °C)	ISG-11, Rev. 3	Sections 2.0.1 and 4.3
Insulation	Protected by overpack or HI-TRAC	-	Section 4.3
<b>Confinement:</b>		10CFR72.128(a)(3) and 10CFR72.236(d) and (e)	
Closure Welds:			
Shell Seams and Shell-to-Baseplate	Full Penetration	-	Section 1.5 and Table 9.1.4
MPC Lid	Multi-pass Partial Penetration	10CFR72.236(e)	Section 1.5 and Table 9.1.4
MPC Closure Ring	Partial Penetration		
Port Covers	Partial Penetration		
NDE:			
Shell Seams and Shell-to-Baseplate	100% RT or UT	-	Table 9.1.4
MPC Lid	Root Pass and Final Surface 100% PT; Volumetric Inspection or 100% Surface PT each 3/8" of weld depth	-	Chapter 8 and Table 9.1.4
Closure Ring	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4

Type	Criteria	Basis	FSAR Reference
Port Covers	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
Leak Testing:			
Welds Tested	MPC shell to shell and MPC shell to baseplate welds (Fabrication). Port covers-to-MPC lid (field)	ISG-25  ISG-18	Section 9.1
Base Metals Tested	MPC shell, MPC baseplate, MPC lid (Fabrication). MPC vent and drain port cover plates (Field).	ISG-25	Section 9.1
Medium	Helium	ANSI N14.5	Section 9.1
Max. Leak Rate	Leaktight	ANSI N14.5	Section 9.1
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
Pressure Testing (if specified):			
Minimum Test Pressure	125 psig (hydrostatic) 120 psig (pneumatic)	-	Governing requirements are specified in Sections 8.1 and 9.1
Welds Tested	MPC Lid-to-Shell, MPC Shell seams, MPC Shell-to-Baseplate	-	Sections 8.1 and 9.1-
Medium	Water or helium	-	Section 8.1 and Chapter 9-
<b>Retrievability:</b>			
Normal and Off-normal:	No Encroachment on Fuel Assemblies	10CFR72.122(f) & (l)	Sections 3.4 and 3.1.2
Post (design basis) Accident			
<b>Criticality:</b>		10CFR72.124 & 10CFR72.236(c)	
Method of Control	Fixed Borated Neutron Absorber, Geometry, and Soluble Boron	-	Section 2.3.4

Type	Criteria	Basis	FSAR Reference
Min. <sup>10</sup> B Loading (Boral/METAMIC®)	0.0267/0.0223 g/cm <sup>2</sup> (MPC-24) 0.0372/0.0310 g/cm <sup>2</sup> (MPC-68, MPC-68FF, MPC-24E, MPC- 24EF, MPC-32 and MPC-32F) 0.01 g/cm <sup>2</sup> (MPC-68F)	-	Sections 2.1.8 and 6.1
Minimum Soluble Boron	Varies (see Tables 2.1.14 and 2.1.16)	Criticality Analysis	Sections 2.1.9 and 6.1
Max. k <sub>eff</sub>	0.95	-	Sections 6.1 and 2.3.4
Min. Burnup	0.0 GWd/MTU (fresh fuel)	-	Section 6.1
<b>Radiation Protection/Shielding:</b>		10CFR72.126, & 10CFR72.128(a)(2)	
MPC: (normal/off-normal/accident)			
MPC Closure	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
MPC Transfer	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
Exterior of Shielding: (normal/off-normal/accident)			
Transfer Mode Position	See Table 2.0.3	10CFR20	Section 5.1.1
ISFSI Controlled Area Boundary	See Table 2.0.2	10CFR72.104 & 10CFR72.106	Section 5.1.1 and Chapter 10
<b>Design Bases:</b>		10CFR72.236(a)	
Spent Fuel Specification:			
Assemblies/Canister	Up to 24 (MPC-24, MPC-24E & MPC-24EF) Up to 32 (MPC-32 and MPC-32F) Up to 68 (MPC-68, MPC-68F, & MPC-68FF)	-	Table 1.2.1 and Section 2.1.9
Type of Cladding	ZR and Stainless Steel	-	Section 2.1.9
Fuel Condition	Intact, Damaged, and Debris	-	Sections 2.1.2, 2.1.3, and 2.1.9
PWR Fuel Assemblies:			

Type	Criteria	Basis	FSAR Reference
Type/Configuration	Various	-	Section 2.1.9
Max. Burnup	68,200 MWD/MTU	-	Sections 2.1.9 and 6.2
Max. Enrichment	Varies by fuel design	-	Table 2.1.3 and Section 2.1.9
Max. Decay Heat/ MPC <sup>†</sup> :	36.9 kW	-	Section 4.4
Minimum Cooling Time:	3 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)	-	Section 2.1.9
Max. Fuel Assembly Weight: (including non-fuel hardware and DFC, as applicable)	1,720 lb. for fuel assemblies that do not require fuel spacers, otherwise 1,680 lb.	-	Section 2.1.9
Max. Fuel Assembly Length: (Unirradiated Nominal)	176.8 in.	-	Section 2.1.9
Max. Fuel Assembly Width (Unirradiated Nominal)	8.54 in.	-	Section 2.1.9
<b>BWR Fuel Assemblies:</b>			
Type	Various	-	Sections 2.1.9 and 6.2
Max. Burnup	65,000 MWD/MTU	-	Section 2.1.9
Max. Enrichment	Varies by fuel design	-	Section 2.1.9, Table 2.1.4
Max. Decay Heat/ MPC <sup>†</sup> .	36.9 kW 42.8 kW (MPC-68M)	-	Section 4.4
Minimum Cooling Time:	2 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)	-	Section 2.1.9
Max. Fuel Assembly Weight:			
w/channels and DFC, as applicable	730 lb. (intact fuel)	-	Section 2.1.9
Max. Fuel Assembly Length (Unirradiated Nominal)	176.5in.	-	Section 2.1.9

<sup>†</sup> Section 2.1.9.1 describes the decay heat limits per assembly. Maximum decay heat shown is for regionalized loading.

Type	Criteria	Basis	FSAR Reference
Max. Fuel Assembly Width (Unirradiated Nominal)	5.85 in.	-	Section 2.1.9
<b>Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperatures	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.1.4
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/3 Yield	NUREG-0612 Regulatory Guide 3.61	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
Fuel Rod Rupture Releases:			
Source Term Release Fraction	1%	NUREG-1536	Sections 2.2.1.3
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3
Snow and Ice	Protected by Overpack	ASCE 7-88	Section 2.2.1.6
<b>Off-Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperature	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.2.2
Leakage of One Seal	N/A	ISG-18	Sections 2.2.2.4 and 7.1
Partial Blockage of Overpack Air Inlets	50% of Air Inlets Blocked	-	Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10%	NUREG-1536	Sections 2.2.2.1
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1

Type	Criteria	Basis	FSAR Reference
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>		10CFR72.24(d)(2) & 10CFR72.94	
Tip Over	See Table 2.0.2	-	Section 2.2.3.2
End Drop	See Table 2.0.2	-	Section 2.2.3.1
Side Drop	See Table 2.0.3	-	Section 2.2.3.1
Fire	See Tables 2.0.2 and 2.0.3	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture Releases:			
Fuel Rod Failures (including non-fuel hardware)	100%	NUREG-1536	Sections 2.2.3.8
Fill Gases	100%	NUREG-1536	Sections 2.2.3.8
Fission Gases	30%	NUREG-1536	Sections 2.2.3.8
Particulates & Volatiles	See Table 7.3.1	-	Sections 2.2.3.9
Confinement Boundary Leakage	None	ISG-18 / ANSI N 14.5	Sections 2.2.3.9 and 7.1
Explosive Overpressure	60 psig (external)	10CFR72.122(c)	Section 2.2.3.10
Airflow Blockage:			
Vent Blockage	100% of Overpack Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Partial Blockage of MPC Basket Vent Holes	Crud Depth (Table 2.2.8)	ESEERCO Project EP91-29	Section 2.2.3.4
<b>Design Basis Natural Phenomenon Design Events and Conditions:</b>		10CFR72.92 & 10CFR72.122(b)(2)	
Flood Water Depth	125 ft.	ANSI/ANS 57.9	Section 2.2.3.6
Seismic	See Table 2.0.2	10CFR72.102(f)	Section 2.2.3.7
Wind	Protected by Overpack	ASCE-7-88	Section 2.2.3.5
Tornado & Missiles	Protected by Overpack	RG 1.76 & NUREG-0800	Section 2.2.3.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	See Table 2.0.2	NFPA 78	Section 2.2.3.11
Extreme Environmental Temperature	See Table 2.0.2	-	Section 2.2.3.14

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Section 2.0.2
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
<b>Structural:</b>			
Design & Fabrication Codes:			
Concrete			
Design	ACI 349 as clarified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Fabrication	ACI 349 as clarified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Compressive Strength	ACI 318.1-89 (92)as clarified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Structural Steel			
Design	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Fabrication	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Dead Weights <sup>†</sup> :			
Max. Loaded MPC (Dry)	90,000 lb. (MPC- 32)	R.G. 3.61	Table 3.2.1
Max. Empty Overpack Assembled with Top Lid (150 pcf concrete/200pcf concrete)	270,000/320,000 lb.	R.G. 3.61	Table 3.2.1
Max. MPC/Overpack (150 pcf concrete/200pcf concrete)	360,000/410,000 lb.	R.G. 3.61	Table 3.2.1
Design Cavity Pressures	N/A	-	Section 2.2.1.3

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



<b>Type</b>	<b>Criteria</b>	<b>Basis</b>	<b>FSAR Reference</b>
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
<b>Thermal:</b>			
Maximum Design Temperatures:			
Concrete			
Through-Thickness Section Average (Normal)	Table 2.2.3	ACI 349, Appendix A (Paragraph A.4.3)	Section 2.0.2, and Tables 1.D.1 and 2.2.3
Local Maximum (Off-Normal and Accident)	Table 2.2.3	Holtec Position Paper DS-289	Section 2.0.2, and Tables 1.D.1 and 2.2.3
Steel Structure		ASME Code Section II, Part D	Table 2.2.3
	See Table 2.2.3		
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.4.1.1.8
<b>Confinement:</b>	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
<b>Retrievability:</b>			
Normal and Off-Normal	No damage that precludes Retrieval of MPC	10CFR72.122(f) & (l)	Section 3.4
Accident			Section 3.4
<b>Criticality:</b>	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
<b>Radiation Protection/Shielding:</b>		10CFR72.126 & 10CFR72.128(a)(2)	
Overpack (Normal/Off-Normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10

Type	Criteria	Basis	FSAR Reference
Beyond Controlled Area During Normal Operation and Anticipated Occurrences	25 mrem/yr. to whole body 75 mrem/yr. to thyroid 25 mrem/yr. to any critical organ	10CFR72.104	Sections 5.1.1, 7.2, and 10.1
At Controlled Area Boundary from Design Basis Accident	5 rem TEDE or sum of DDE and CDE to any individual organ or tissue (other than lens of eye) $\leq$ 50 rem. 15 rem lens dose. 50 rem shallow dose to skin or extremity.	10CFR72.106	Sections 5.1.2, 7.3, and 10.1
<b>Design Bases:</b>			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1.9
<b>Normal Design Event Conditions:</b>			
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load <sup>†</sup> :		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	250,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.4 Section 2.2.1.2
Dry Loaded MPC (max.)	90,000 lb.	R.G. 3.61	Table 3.2.1 and Section 2.2.1.2
<b>Handling:</b>			
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3.5
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature During Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2

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

Type	Criteria	Basis	FSAR Reference
Snow and Ice Load	100 lb./ft <sup>2</sup>	ASCE 7-88	Section 2.2.1.6
Wet/Dry Loading	Dry	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
<b>Off-Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	-40° F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100° F	ANSI/ANS 57.9	Section 2.2.2.2
Partial Blockage of Air Inlets	50% of Air Inlets Blocked	-	Section 2.2.2.5
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>		10CFR72.94	
Drop Cases:			
End	11 in.	-	Section 2.2.3.1
Tip-Over (Not applicable for HI-STORM 100A)	Assumed (Non-mechanistic)	-	Section 2.2.3.2
Fire:			
Duration	217 seconds	10CFR72.122(c)	Section 2.2.3.3
Temperature	1,475° F	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture	See Table 2.0.1	-	Section 2.2.3.8
Air Flow Blockage:			
Vent Blockage	100% of Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Ambient Temperature	80° F	10CFR72.128(a)(4)	Section 2.2.3.13
Explosive Overpressure External Differential Pressure	10 psid instantaneous, 5 psid steady state	10 CFR 72.128(a)(4)	Table 2.2.1
<b>Design-Basis Natural Phenomenon Design Events and Conditions:</b>		10CFR72.92 & 10CFR72.122(b)(2)	
Flood			
Height	125 ft.	RG 1.59	Section 2.2.3.6
Velocity	15 ft/sec.	RG 1.59	Section 2.2.3.6
Seismic			
Max. acceleration at top of ISFSI pad	Free Standing: $G_H + 0.53G_V \leq 0.53$ Anchored:	10CFR72.102(f)	Section 3.4.7.1 Section 3.4.7.3

Type	Criteria	Basis	FSAR Reference
	$G_H \leq 2.12, G_V \leq 1.5$		
Tornado			
Wind			
Max. Wind Speed	360 mph	RG 1.76	Section 2.2.3.5
Pressure Drop	3.0 psi	RG 1.76	Section 2.2.3.5
Missiles			Section 2.2.3.5
Automobile			
Weight	1,800 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder			
Weight	125 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	8 in.	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	0.22 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	1 in.	NUREG-0800	Table 2.2.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	Resistance Heat-Up	NFPA 70 & 78	Section 2.2.3.11
Extreme Environmental Temperature	125° F	-	Section 2.2.3.14
<b>Load Combinations:</b>	See Table 2.2.14 and Table 3.1.5	ANSI/ANS 57.9 and NUREG-1536	Section 2.2.7

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Section 2.0.3
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
<b>Structural:</b>			
Design Codes:			
Structural Steel	ASME Code, Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.3
Lifting Trunnions	NUREG-0612 & Regulatory Guide 3.61	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights <sup>†</sup> :			
Max. Empty Cask:			
w/top lid and pool lid installed and water jacket filled	143,500 lb. (HI-TRAC 125) 102,000 lb. (HI-TRAC 100) 102,000 lb. (HI-TRAC 100D) 146,000 lb. (HI-TRAC 125D)	R.G. 3.61	Table 3.2.2
w/top lid and transfer lid installed and water jacket filled (N/A for HI-TRAC 100D and 125D)	155,000 lb. (HI-TRAC 125) 111,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.2
Max. MPC/HI-TRAC with Yoke (in-pool lift):	250,000 lb. (HI-TRAC 125 and 125D) 200,000 lb. (HI-TRAC 100 and 100D)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3

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

Type	Criteria	Basis	FSAR Reference
Water Jacket Cavity	60 psig (internal)	ANSI/ANS 57.9	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Section 2.0.3
	Continued adequate performance of HI-TRAC transfer cask	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
<b>Thermal:</b>			
Maximum Design Temperature			
Structural Materials	400° F	ASME Code Section II, Part D	Table 2.2.3
Shielding Materials			
Lead	350° F (max.)		Table 2.2.3
Liquid Neutron Shield	307° F (max.)	-	Table 2.2.3
Solid Neutron Shield	300° F (max.) (long term) 350°F (max.) (short term)	Test Data	Appendix 1.B and Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.5.1.1.3
<b>Confinement:</b>	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
<b>Retrievability:</b>			
Normal and Off-Normal	No encroachment on MPC	10CFR72.122(f) & (l)	Section 3.4
After Design-basis (Postulated Accident)			Section 3.4
<b>Criticality:</b>	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
<b>Radiation Protection/Shielding:</b>		10CFR72.126 & 10CFR72.128(a)(2)	
Transfer Cask (Normal/Off-Normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10

Type	Criteria	Basis	FSAR Reference
<b>Design Bases:</b>			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
<b>Normal Design Event Conditions:</b>			
Ambient Temperature:	80 ° F	ANSI/ANS 57.9	Section 2.2.1.4
<b>Live Load†</b>			
Max. Loaded Canister			
Dry	90,000 lb.	R.G. 3.61	Table 3.2.1
Wet (including water in HI-TRAC annulus)	106,570 lb.	R.G. 3.61	Table 3.2.4
<b>Handling:</b>			
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/3 Yield	NUREG-0612 Regulatory Guide 3.61	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature for Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
<b>Test Loads:</b>			
Trunnions	300% of vertical design load	NUREG-0612 & Regulatory Guide 3.61	Section 9.1.2.1
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>			
		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1
<b>Fire</b>			
Duration	4.8 minutes	10CFR72.122(c)	Section 2.2.3.3

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

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



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

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

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

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

Item	Free-Standing	Anchored	Comments
5. Maximum seismic input on the pad/cask contact surface. $G_H$ is the vectorial sum of the two horizontal ZPAs and $G_V$ is the vertical ZPA	$G_H + \mu G_V \leq \mu$ (see note 1 below)	$G_H \leq 2.12$  AND  $G_V \leq 1.5$	
6. Required minimum value of cask to pad static coefficient of friction ( $\mu$ , must be confirmed by testing if a value greater than 0.53 is used).	Greater than or equal to 0.53 (per Table 2.2.9).	Not applicable	
7. Applicable Wind and Large Missile Loads	Per Table 2.2.4, missile and wind loading different from the tabulated values, require 10CFR 72.48 evaluation	The maximum overturning moment at the base of the cask due to lateral missile and/or wind action must be less than $1 \times 10^7$ ft-lb.	The bases are provided in Section 3.4.8 for free-standing casks; the limit for anchored casks ensures that the anchorage system will have the same structural margins established for seismic loading.
8. Small and medium missiles (penetrant missile)	Per Table 2.2.5, missiles and wind loading different from the tabulated value, require 10CFR 72.48 evaluation.	Same as for free-standing cask construction.	
9. Design Loadings for the ISFSI Pad	Per load combinations in Section 2.0.4 using site-specific load.	Same as for free-standing cask.	

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

## 2.1 SPENT FUEL TO BE STORED

### 2.1.1 Determination of The Design Basis Fuel

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

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

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

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

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

determined to be acceptable for storage in the HI-STORM 100 System. Section 2.1.9 summarizes the authorized contents for the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 and meets the other limits specified in Section 2.1.9 is acceptable for storage in the HI-STORM 100 System. Tables 2.1.3 and 2.1.4 present the groups of fuel assembly types defined as “array/classes” as described in further detail in Chapter 6. Table 2.1.5 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal. Additional information on the design basis fuel definition is presented in the following subsections.

### 2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STORM 100 System is intact ZR or stainless steel (SS) clad fuel assemblies with the characteristics listed in Tables 2.1.17 through 2.1.24.

Intact fuel assemblies without fuel rods in fuel rod locations cannot be loaded into the HI-STORM 100 unless dummy fuel rods, which occupy a volume greater than or equal to the original fuel rods, replace the missing rods prior to loading. Any intact fuel assembly that falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly, as defined in Section 2.1.9 can be safely stored in the HI-STORM 100 System. If irradiated dummy stainless steel rods are present in the fuel assembly, the dummy/replacement rods will be considered in the site specific dose calculations.

The range of fuel characteristics specified in Tables 2.1.3 and 2.1.4 have been evaluated in this FSAR and are acceptable for storage in the HI-STORM 100 System within the decay heat, burnup, and cooling time limits specified in Section 2.1.9 for intact fuel assemblies.

### 2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in the glossary.

Damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with mesh screens having between 30x30 and 250x250 openings per inch, for storage in the HI-STORM 100 System (see Figures 2.1.1 and 2.1.2B, C, and D. Note: Figures 2.1.2B, C, and D are for illustrative purposes only). The MPC-24, MPC-24EF, MPC-32 and MPC-32F are designed to accommodate PWR damaged fuel and fuel debris. The MPC-68, MPC-68F and MPC-68FF are designed to accommodate BWR damaged fuel and fuel debris. The appropriate structural, thermal, shielding, criticality, and confinement analyses have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for damaged fuel assemblies and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in Section 2.1.9. Dresden Unit 1 fuel assemblies contained in Transnuclear-designed damaged fuel canisters and one Dresden Unit 1 thoria rod canister have been approved for storage directly in the HI-STORM 100

System without re-packaging (see Figures 2.1.2 and 2.1.2A).

MPC contents classified as fuel debris are required to be stored in DFCs. The basket designs for the standard and “F” model MPCs are identical. The lid and shell designs of the “F” models are unique in that the upper shell portion of the canister is thickened for additional strength needed to qualify as a secondary containment, which used to be required under hypothetical accident conditions of transportation under 10 CFR 71. Figure 2.1.9 shows the details of the differences between the standard and “F” model MPC shells. These details are common for both the PWR and BWR series MPC models.

#### 2.1.4 Deleted

#### 2.1.5 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are specified in Section 2.1.9. The centers of gravity reported in Section 3.2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 to 2-1/2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10. In order to qualify for storage in the MPC, the SNF must satisfy the physical parameters listed in Section 2.1.9.

#### 2.1.6 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly and the decay heat removal capabilities of the HI-STORM 100 System. No attempt is made to link the maximum allowable decay heat per fuel assembly with burnup, enrichment, or cooling time. Rather, the decay heat per fuel assembly is adjusted to yield peak fuel cladding temperatures with an allowance for margin to the temperature limit.

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

Section 2.1.9 also includes separate cooling time, burnup, and decay heat limits for uniform fuel loading and regionalized fuel loading. Regionalized loading allows higher heat emitting fuel assemblies to be stored in the center fuel storage locations than would otherwise be authorized for storage under uniform loading conditions.

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The bounding fuel assembly design for thermal calculations for each fuel type is provided in Table 2.1.5.

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

Except for MPC-68F, fuel may be stored in the MPC using one of two storage strategies, namely, uniform loading and regionalized loading. Uniform loading allows storage of any fuel assembly in any fuel storage location, subject to additional restrictions, such as those for loading of fuel assemblies containing non-fuel hardware as defined in the glossary. Regionalized fuel loading allows for higher heat emitting fuel assemblies to be stored in some storage locations with lower heat emitting fuel assemblies in the remaining fuel storage locations. Regionalized loading allows storage of higher heat emitting fuel assemblies than would otherwise be permitted using the uniform loading strategy. The definition of the regions for each MPC model is provided in Table 2.1.27. Regionalized fuel loading is not permitted in MPC-68F.

#### 2.1.7 Radiological Parameters for Design Basis SNF

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

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

The design basis dose rates can be met by a variety of burnup levels and cooling times. Section 2.1.9 provides the procedure for determining burnup and cooling time limits for all of the authorized fuel assembly array/classes for both uniform fuel loading and regionalized loading. Table 2.1.11 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Thoria Rod Canister is authorized for storage in combination with other intact and damaged fuel, and fuel debris as specified in Section 2.1.9.

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

### 2.1.8 Criticality Parameters for Design Basis SNF

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

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

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

The criticality analyses for the MPC-24, MPC-24E and MPC-24EF (all with higher enriched fuel) and for the MPC-32 and MPC-32F were performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.14 and 2.1.16 provide the required soluble boron concentrations for these MPCs.

### 2.1.9 Summary of Authorized Contents

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

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



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

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

- A decay heat less than or equal to the maximum allowable value.
- A burnup less than or equal to the maximum allowable value.
- A cooling time greater than or equal to the minimum allowable value.

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

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

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

#### 2.1.9.1.2 Design Heat Load for ZR-Clad Fuel

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

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

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

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

$$Q_d = n q \quad \text{Equation a}$$

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

$$Q_t = \sum_i^n r_i \quad \text{Equation b}$$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- (i) Choose a value of  $X$  in the permissible range ( $0.5 \leq X \leq 3$ )
- (ii) Calculate  $q_2$  using the following equation:

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

where:

$$y = 0.23/X^{0.1}$$

$q_2$  = Maximum allowable decay heat per fuel storage location in Region 2 (kW)

$Q_d$  = Maximum uniform storage MPC decay heat (34 kW)

$X$  = Ratio of  $q_1$  to  $q_2$  chosen in Step (i)

$n_1$  = Number of fuel storage locations in Region 1 from Table 2.1.27

$n_2$  = Number of fuel storage locations in Region 2 from Table 2.1.27

- (iii) Calculate  $q_1$  using the following equation:

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

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

It should be emphasized that the variable two-region scheme of storage does not introduce any new

complication in the dry storage implementation. As compared to uniform loading in MPC-32, where  $q = 1.0625$  kW for all cells, the regionalized loading gives the user the flexibility to load the MPC with more varying heat loads. It is noted that for  $X < 1$   $Q_{CoC}$  is greater than  $Q_d$ , for  $X = 1$   $Q_{CoC}$  equals  $Q_d$ , and for  $X > 1$   $Q_{CoC}$  is less than  $Q_d$ . For ALARA and regardless of which loading pattern is used, a plant should always seek to preferentially locate the fuel with the higher heat loads toward the center of the MPC. If the need arises to place younger fuel into dry storage a regionalized pattern with  $X < 1$  may be more appropriate.

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

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

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

The calculation described in this section is used to determine the maximum allowable fuel assembly burnup for minimum cooling times between 2 and 40 years, using maximum decay heat and minimum enrichment as input values. This calculation may be used to create multiple burnup versus cooling time tables for a particular fuel assembly array/class and different minimum enrichments. The allowable maximum burnup for a specific fuel assembly may be calculated based on the assembly's particular enrichment and cooling time.

- (i) Choose a fuel assembly minimum enrichment,  $E_{235}$ .
- (ii) Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 2 and 40 years using the equation below:

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

Equation j

Where:

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

$q$  = Maximum allowable decay heat per fuel storage location determined in Section 2.1.9.1.1 or 2.1.9.1.2 (kW)

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

A through G = Coefficients from Tables 2.1.28 or 2.1.29 for the applicable fuel assembly

array/class and minimum cooling time.

#### 2.1.9.1.4 Other Considerations

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

- Calculated burnup limits shall be rounded down to the nearest integer
- Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR fuel must be reduced to be equal to these values.
- Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a minimum cooling time of 4.5 years may be interpolated between those burnups calculated for 4 and 5 years.
- ZR-clad fuel assemblies must have a minimum enrichment, as defined in the glossary, greater than or equal to the value used in determining the maximum allowable burnup per Section 2.1.9.1.3 to be authorized for storage in the MPC.
- When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any PWR non-fuel hardware, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.
- There are two options for helium backfill range (shown in Table 1.2.2). The lower helium backfill range has different per cell heat load limits given in Table 2.1.31.

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

#### 2.1.9.1.5 Supplemental Cooling Threshold Heat Loads

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

Table 2.1.1

## PWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

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

Table 2.1.2

## BWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

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

Table 2.1.3  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	14x14 A	14x14 B	14x14 C	14x14 D	14x14E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	$\leq 365$	$\leq 412$	$\leq 438$	$\leq 400$	$\leq 206$
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % $^{235}\text{U}$ ) (Note 7)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.0$ (24) $\leq 5.0$ (24E/24EF)	$\leq 5.0$ (24) $\leq 5.0$ (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit - see Note 5) (wt % $^{235}\text{U}$ )	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	$\geq 0.400$	$\geq 0.417$	$\geq 0.440$	$\geq 0.422$	$\geq 0.3415$
Fuel Clad I.D. (in.)	$\leq 0.3514$	$\leq 0.3734$	$\leq 0.3880$	$\leq 0.3890$	$\leq 0.3175$
Fuel Pellet Dia. (in.) (Note 8)	$\leq 0.3444$	$\leq 0.3659$	$\leq 0.3805$	$\leq 0.3835$	$\leq 0.3130$
Fuel Rod Pitch (in.)	$\leq 0.556$	$\leq 0.556$	$\leq 0.580$	$\leq 0.556$	Note 6
Active Fuel Length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 144$	$\leq 102$
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	$\geq 0.017$	$\geq 0.017$	$\geq 0.038$	$\geq 0.0145$	N/A



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

Fuel Assembly Array/Class	15x15 A	15x15 B	15x15 C	15x15 D	15x15 E	15x15 F
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 473	≤ 473	≤ 473	≤ 495	≤ 495	≤ 495
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit – see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.) (Note 8)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

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

Fuel Assembly Array and Class	15x15 G	15x15H	15x15I	16x16 A	16x16B	16x16C
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 495	≤ 495	≤ 448	≤ 448	≤ 448
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	N/A (Note 9)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit – see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0 (Note 9)	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	216	236	236	235
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.413	≥ 0.382	≥ 0.374	≥ 0.374
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.367	≤ 0.3350	≤ 0.3290	≤ 0.3290
Fuel Pellet Dia. (in.) (Note 8)	≤ 0.3825	≤ 0.3622	≤ 0.360	≤ 0.3255	≤ 0.3225	≤ 0.3225
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.550	≤ 0.506	≤ 0.506	≤ 0.485
Active Fuel length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	9 (Note 10)	5 (Note 4)	5 (Note 4)	21
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0140	≥ 0.0350	≥ 0.0400	≥ 0.0157

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

Fuel Assembly Array and Class	17x17A	17x17 B	17x17 C
Clad Material (Note 2)	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 433	≤ 474	≤ 480
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit – see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	264	264	264
Fuel Clad O.D. (in.)	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.) (Note 8)	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1.3 (continued)  
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. See glossary for the definition of “ZR.”
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users’ fuel records to account for manufacturer’s tolerances.
4. Each guide tube replaces four fuel rods.
5. Soluble boron concentration per Tables 2.1.14 and 2.1.16, as applicable.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum initial enrichment of the intact fuel assemblies, damaged fuel assemblies and fuel debris is 4.0 wt.% <sup>235</sup>U.
8. Annular fuel pellets are allowed in the top and bottom 12” of the active fuel length.
9. This fuel assembly array/class can only be loaded in MPC-32.
10. One Instrument Tube and eight Guide Bars (Solid Zr).

Table 2.1.4  
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	6x6 A	6x6 B	6x6 C	7x7 A	7x7 B	8x8 A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 198	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 2.7	≤ 2.7 for UO <sub>2</sub> rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

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

<b>Fuel Assembly Array and Class</b>	<b>8x8 B</b>	<b>8x8 C</b>	<b>8x8 D</b>	<b>8x8 E</b>	<b>8x8F</b>	<b>9x9 A</b>
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 190	≤ 190	≤ 190	≤ 191	≤ 180
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

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

Fuel Assembly Array and Class	9x9 B	9x9 C	9x9 D	9x9 E (Note 13)	9x9 F (Note 13)	9x9 G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 180	≤ 182	≤ 182	≤ 183	≤ 183	≤ 164
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

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

<b>Fuel Assembly Array and Class</b>	<b>10x10 A</b>	<b>10x10 B</b>	<b>10x10 C</b>	<b>10x10 D</b>	<b>10x10 E</b>
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U)(Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080



Table 2.1.4 (continued)  
BWR FUEL ASSEMBLY CHARACTERISTICS

## NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. See glossary for the definition of “ZR.”
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users’ fuel records to account for manufacturer tolerances.
4.  $\leq 0.635$  wt. %  $^{235}\text{U}$  and  $\leq 1.578$  wt. % total fissile plutonium ( $^{239}\text{Pu}$  and  $^{241}\text{Pu}$ ), (wt. % of total fuel weight, i.e.,  $\text{UO}_2$  plus  $\text{PuO}_2$ )
5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
6. Square, replacing nine fuel rods.
7. Variable.
8. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
11. These rods may also be sealed at both ends and contain Zr material in lieu of water.
12. This assembly is known as “QUAD+.” It has four rectangular water cross segments dividing the assembly into four quadrants.
13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.
14. For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum planar average initial enrichment for the intact fuel assemblies is limited to 3.7 wt.%  $^{235}\text{U}$ , as applicable.

Table 2.1.5

## DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

<b>Criterion</b>	<b>BWR Fuel</b>	<b>PWR Fuel</b>
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Array/Class 10x10A)	B&W 15x15 (Array/Class 15x15F)
Shielding	GE 7x7	B&W 15x15
Thermal-Hydraulic	GE-12/14 10x10	<u>W</u> 17x17 OFA
Structural	730 lbs. for intact fuel 830 lbs. for canisterized fuel (intact and canisterized fuel include channels)	1680 lbs. including any control components

Tables 2.1.6 through 2.1.8

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Table 2.1.9

## SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length w/o NFH <sup>1</sup> (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
CE 14x14	157	4.1	137	9.5	10.0
CE 16x16	176.8	4.7	150	0	0
BW 15x15	165.7	8.4	141.8	6.7	4.1
W 17x17 OFA	159.8	3.7	144	8.2	8.5
W 17x17 Std	159.8	3.7	144	8.2	8.5
W 17x17 V5H	160.1	3.7	144	7.9	8.5
W 15x15	159.8	3.7	144	8.2	8.5
W 14x14 Std	159.8	3.7	145.2	9.2	7.5
W 14x14 OFA	159.8	3.7	144	8.2	8.5
Ft. Calhoun	146	6.6	128	10.25	20.25
St. Lucie 2	158.2	5.2	136.7	10.25	8.05
B&W 15x15 SS	137.1	3.873	120.5	19.25	19.25
W 15x15 SS	137.1	3.7	122	19.25	19.25
W 14x14 SS	137.1	3.7	120	19.25	19.25
Indian Point 1	137.2	17.705	101.5	18.75	20.0

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two to 2-1/2 inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.

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

Table 2.1.10

## SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

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

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two to 2-1/2 inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.

<sup>†</sup> Fuel assembly length includes the damaged fuel container.

Table 2.1.11  
 NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

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

<sup>1</sup> Reference 2.1.7

<sup>2</sup> Reference 2.1.8

Table 2.1.12

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

<b>PARAMETER</b>	<b>MPC-68 or MPC-68F</b>
Cladding Type	Zircaloy
Composition	98.2 wt.% ThO <sub>2</sub> , 1.8 wt.% UO <sub>2</sub> with an enrichment of 93.5 wt. % <sup>235</sup> U or 98.5 wt. % ThO <sub>2</sub> , 1.5 wt. % UO <sub>2</sub> with an enrichment of 93.5 wt. % <sup>235</sup> U
Number of Rods Per Thoria Canister	≤ 18
Decay Heat Per Thoria Canister	≤ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time ≥ 18 years and average burnup ≤ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	≤ 27 kg/canister
Fuel Cladding O.D.	≥ 0.412 inches
Fuel Cladding I.D.	≤ 0.362 inches
Fuel Pellet O.D.	≤ 0.358 inches
Active Fuel Length	≤ 111 inches
Canister Weight	≤ 550 lbs., including Thoria Rods
Canister Material	Type 304 SS

Table 2.1.13

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Table 2.1.14

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

<b>MPC MODEL</b>	<b>FUEL ASSEMBLY MAXIMUM AVERAGE ENRICHMENT (wt % <sup>235</sup>U)</b>	<b>MINIMUM SOLUBLE BORON CONCENTRATION (ppmb)</b>
MPC-24	All fuel assemblies with initial enrichment <sup>1</sup> less than the prescribed value for soluble boron credit	0
MPC-24	One or more fuel assemblies with an initial enrichment <sup>1</sup> greater than or equal to the prescribed value for no soluble boron credit and $\leq 5.0$ wt. %	$\geq 400$
MPC-24E/24EF	All fuel assemblies with initial enrichment <sup>1</sup> less than the prescribed value for soluble boron credit	0
MPC-24E/24EF	All fuel assemblies classified as intact fuel assemblies and one or more fuel assemblies with an initial enrichment <sup>1</sup> greater than or equal to the prescribed value for no soluble boron credit and $\leq 5.0$ wt. %	$\geq 300$
MPC-24E/24EF	One or more fuel assemblies classified as damaged fuel or fuel debris and one or more fuel assemblies with initial enrichment $> 4.0$ wt.% and $\leq 5.0$ wt.%	$\geq 600$

<sup>1</sup>Refer to Table 2.1.3 for these enrichments.



Table 2.1.15

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

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

Notes:

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

Table 2.1.16

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

Fuel Assembly Array/Class	All Intact Fuel Assemblies		One or More Damaged Fuel Assemblies or Fuel Debris	
	Max. Initial Enrichment $\leq 4.1$ wt.% $^{235}\text{U}$ (ppmb)	Max. Initial Enrichment 5.0 wt.% $^{235}\text{U}$ (ppmb)	Max. Initial Enrichment $\leq 4.1$ wt.% $^{235}\text{U}$ (ppmb)	Max. Initial Enrichment 5.0 wt.% $^{235}\text{U}$ (ppmb)
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G/I	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A	1,400	2,000	1,500	2,300
17x17A	1,600	2,200	1,800	2,600
17x17B/C	1,900	2,600	2,100	2,900

## Note:

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

Table 2.1.17

## LIMITS FOR MATERIAL TO BE STORED IN MPC-24

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq 8$ years and $\leq 40,000$ MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq 710$ Watts
Non-Fuel Hardware Burnup and Cooling Time	As specified in Table 2.1.25
Fuel Assembly Length	$\leq 176.8$ in. (nominal design)
Fuel Assembly Width	$\leq 8.54$ in. (nominal design)
Fuel Assembly Weight	$\leq 1,720$ lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise $\leq 1,680$ lbs (including non-fuel hardware)
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to 24 PWR intact fuel assemblies.</li> <li>▪ Damaged fuel assemblies and fuel debris are not permitted for loading in MPC-24.</li> <li>▪ One NSA is authorized to be loaded with a fuel assembly in fuel storage location 9, 10, 15, or 16.</li> <li>▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location.</li> <li>▪ APSRs may be loaded with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16</li> <li>▪ CRAs, RCCAs and/or CEAs may be stored with fuel assemblies in fuel cell locations 4, 5, 8 through 11, 14 through 17, 20, and/or 21.</li> <li>▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.</li> </ul>

Table 2.1.18

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Table 2.1.19

## LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

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

Table 2.1.19 (cont'd)

## LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	VALUE
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to four (4) DFCs containing Dresden Unit 1 or Humboldt Bay uranium oxide or MOX fuel debris. The remaining fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable: <ul style="list-style-type: none"> <li>- uranium oxide BWR intact fuel assemblies</li> <li>- MOX BWR intact fuel assemblies</li> <li>- uranium oxide BWR damaged fuel assemblies in DFCs</li> <li>- MOX BWR damaged fuel assemblies in DFCs</li> <li>- up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12.</li> </ul> </li> <li>▪ Stainless steel channels are not permitted.</li> <li>▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.</li> </ul>

## Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.
2. Only fuel from the Dresden Unit 1 and Humboldt Bay plants are permitted for storage in the MPC-68F.

Table 2.1.20

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

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

Table 2.1.20 (cont'd)

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

PARAMETER	VALUE
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies.</li> <li>▪ One NSA is permitted for loading with a fuel assembly in fuel storage location 9, 10, 15, or 16.</li> <li>▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location.</li> <li>▪ APSRs may be loaded with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16.</li> <li>▪ CRAs, RCCAs and/or CEAs may be stored with fuel assemblies in fuel cell locations 4, 5, 8 through 11, 14 through 17, 20, and/or 21.</li> <li>▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.</li> </ul>

## Notes:

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



Table 2.1.21

[INTENTIONALLY DELETED]

Table 2.1.22

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

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

Table 2.1.22 (cont'd)  
LIMITS FOR MATERIAL TO BE STORED IN MPC-68 AND MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Assembly Width	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq 4.7$ in. (nominal design)  All Other array/classes: $\leq 5.85$ in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq 4.7$ in. (nominal design)  All Other array/classes: $\leq 5.85$ in. (nominal design)
Fuel Assembly Weight	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq 400$ lbs. (including channels)  All Other array/classes: $\leq 730$ lbs. (including channels)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq 550$ lbs. (including channels and DFC)  All Other array/classes: $\leq 830$ lbs. (including channels and DFC)
Other Limitations	<ul style="list-style-type: none"> <li>▪ For assembly/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A, up to 68 intact fuel assemblies or damaged fuel assemblies in DFCs may be stored. Fuel debris in DFCs may be stored in up to 8 locations. A Dresden Unit 1 Thoria Rod Container may be stored in one location.</li> <li>▪ For all other array/classes, up to 16 DFCs containing damaged fuel assemblies and/or up to eight (8) DFCs containing fuel assemblies classified as fuel debris may be stored. DFCs shall be located only in fuel cell locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68, with the balance comprised of intact fuel assemblies meeting the above specifications, up to a total of 68.</li> <li>▪ SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50.</li> <li>▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.</li> </ul>	

## NOTES:

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

Table 2.1.23

[INTENTIONALLY DELETED]

Table 2.1.24

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

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

Table 2.1.24 (cont'd)

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

PARAMETER	VALUE
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32.</li> <li>▪ One NSA is permitted for loading with a fuel assembly in fuel storage location 13, 14, 19, or 20.</li> <li>▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location.</li> <li>▪ CRAs, RCCAs, CEAs, or APSRs may only be loaded with fuel assemblies in fuel cell locations 7, 8, 12-15, 18-21, 25 and/or 26.</li> <li>▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.</li> </ul>

## NOTES:

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

Table 2.1.25  
NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, 3, and 7)

Post-irradiation Cooling Time (yrs)	NSA with NFH, Inserts (Note 4) Maximum Burnup (MWD/MTU)	NSA without NFH, Guide Tube Hardware, or Control Component (Note 5) Maximum Burnup (MWD/MTU)	APSR Maximum Burnup (MWD/MTU)
≥ 3	≤ 24,635	N/A (Note 6)	N/A
≥ 4	≤ 30,000	N/A	N/A
≥ 5	≤ 36,748	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	-	≤ 54,500
≥ 7	≤ 52,900	-	≤ 68,000
≥ 8	≤ 60,000	-	≤ 83,000
≥ 9	≤ 78,784	-	≤ 111,000
≥ 10	≤ 101,826	-	≤ 180,000
≥ 11	≤ 141,982	-	≤ 630,000
≥ 12	≤ 360,000	-	-

## NOTES:

- Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.
- Linear interpolation between points is permitted, except that APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 11 years.
- Applicable to uniform loading and regionalized loading.
- Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), vibration suppressor inserts, and Neutron Source Assemblies (NSAs) in combination with other control components (i.e., BPRAs, TPDs, and/or RCCAs).
- Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, orifice rod assemblies, Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), Rod Cluster Control Assemblies (RCCAs), and NSAs without other forms of control components.
- N/A means not authorized for loading at this cooling time.
- Non-fuel hardware burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post-irradiation.

Table 2.1.26  
DESIGN HEAT EMISSION RATES  
(UNIFORM LOADING, ZR-CLAD, ABOVEGROUND STORAGE<sup>1</sup>)

MPC	Decay Heat (kW)	
	Per Intact Fuel Assembly <sup>2</sup>	Per MPC
MPC-24/24E/24EF	1.416	34
MPC-32/32F	1.062	34
MPC-68/68FF	0.5	34
	Per Damaged Fuel Assembly or Fuel Debris	Per MPC with Damaged Fuel Assembly or Fuel Debris
MPC-24E/24EF	$\leq 1.114$	$\leq 26.7$
MPC-32/32F	$\leq 0.718$	$\leq 23$
MPC-68/68FF	$\leq 0.393$	$\leq 26.7$

<sup>1</sup> Maximum allowable heat loads in 100U underground storage are defined in Supplement 2.I

<sup>2</sup> This limit applies to each storage cell and should include decay heat from any NFH



Table 2.1.27

## MPC FUEL STORAGE REGIONS

MPC	Number of Storage Cells		Storage Cell IDs**	
	Inner Region (n <sub>1</sub> )	Outer Region (n <sub>2</sub> )	Inner Region	Outer Region
MPC-24/24E/24EF	12	12	4, 5 8 through 11 14 through 17 20 and 21	All other locations
MPC-32/32F	12	20	7, 8, 12 through 15, 18 through 21, 25 and 26	All other locations
MPC-68/68FF	32	36	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All other locations
** See Figures 1.2.2 through 1.2.4 for storage cell numbering				

Table 2.1.28  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
2.0	8716.89	1454.67	-91.96	-168.45	2047.50	-209.91	-738.51
2.25	10917.50	1441.49	-112.76	-162.14	2274.96	-266.46	-788.45
2.5	13452.90	1258.44	-119.69	-154.08	2491.83	-329.35	-760.18
2.75	16326.90	847.56	-100.72	-146.46	2680.07	-390.55	-727.50
3.0	19310.30	276.56	-59.30	-139.52	2851.81	-452.00	-614.85
4.0	33007.90	-4711.82	663.64	-117.16	3291.32	-622.31	-338.63
5.0	46306.70	-12448.80	2292.51	-113.20	3504.56	-662.41	-73.12
6.0	57461.80	-20693.50	4405.17	-121.14	3633.52	-614.82	1.66
7.0	66450.10	-28314.10	6635.00	-129.61	3706.00	-510.84	-113.74
8.0	73652.70	-34919.90	8759.36	-136.91	3752.43	-391.36	-311.56
9.0	79378.80	-40316.60	10606.30	-141.55	3784.66	-280.29	-485.97
10.0	84125.10	-44860.80	12239.70	-143.00	3777.62	-152.58	-635.70
11.0	88066.60	-48540.60	13594.30	-142.74	3758.54	-33.78	-726.86
12.0	91416.80	-51619.90	14789.00	-141.31	3742.31	64.80	-833.14
13.0	94657.90	-54579.30	15916.70	-137.14	3652.04	215.05	-967.41
14.0	97332.40	-56854.80	16823.50	-133.83	3610.21	315.79	-959.48
15.0	99866.10	-58816.70	17560.80	-128.68	3529.41	430.14	-991.32
16.0	102093.00	-60412.40	18171.30	-124.64	3469.67	535.07	-1078.73
17.0	104419.00	-62150.90	18846.80	-118.62	3363.97	674.13	-1092.27
18.0	106439.00	-63357.20	19259.50	-114.31	3300.43	769.38	-1137.26
19.0	108613.00	-64655.80	19660.70	-107.71	3182.61	904.63	-1084.05
20.0	110475.00	-65506.20	19883.50	-103.32	3125.81	988.08	-1062.86
22.0	114223.00	-66854.40	19969.00	-91.34	2899.19	1260.81	-1076.58
24.0	117822.00	-67556.70	19641.80	-79.56	2684.32	1499.23	-1011.23
26.0	121396.00	-67752.70	18783.80	-68.61	2465.91	1753.65	-940.82
28.0	125040.00	-67445.30	17353.90	-55.51	2184.99	2059.27	-883.36
30.0	128075.00	-65562.60	14994.70	-45.58	2003.10	2244.12	-819.25
35.0	136419.00	-58633.40	6027.48	-15.81	1354.94	2757.84	-687.83
40.0	144776.00	-48670.50	-4898.54	5.02	1019.97	2652.57	-507.64

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14B						
	A	B	C	D	E	F	G
2.0	7962.04	1332.84	-83.96	-165.28	1836.65	-176.15	-859.65
2.25	10055.50	1296.32	-100.51	-156.80	2012.11	-217.67	-907.11
2.5	12332.50	1153.20	-110.56	-149.42	2185.46	-264.52	-845.06
2.75	15072.80	715.71	-82.42	-140.68	2336.22	-310.64	-833.26
3.0	18034.30	64.77	-24.88	-130.87	2450.80	-348.00	-857.34
4.0	30007.50	-4046.37	538.96	-110.22	2792.92	-469.98	-371.81
5.0	41033.00	-9824.17	1644.13	-108.10	2979.87	-509.22	122.91
6.0	50398.10	-16082.00	3115.79	-113.75	3084.72	-485.25	117.44
7.0	57782.60	-21657.00	4602.39	-121.19	3161.16	-433.49	-112.57
8.0	63670.20	-26431.00	6006.16	-127.70	3227.81	-382.20	-74.84
9.0	68390.50	-30359.70	7246.09	-131.82	3277.23	-336.08	-200.60
10.0	72284.50	-33630.50	8335.68	-132.71	3293.15	-279.98	-291.73
11.0	75584.30	-36387.10	9298.07	-132.38	3295.07	-227.50	-340.65
12.0	78425.20	-38681.30	10125.90	-130.36	3283.13	-176.12	-462.22
13.0	80928.60	-40624.70	10848.10	-127.28	3259.89	-127.73	-563.09
14.0	83136.90	-42279.70	11500.20	-124.50	3249.69	-97.40	-565.79
15.0	85398.00	-44023.70	12192.30	-119.64	3186.24	-30.11	-665.54
16.0	87257.50	-45137.70	12617.40	-113.94	3127.01	22.40	-678.95
17.0	89196.20	-46520.30	13209.90	-110.27	3091.45	63.17	-713.69
18.0	90991.80	-47570.50	13623.80	-104.55	3008.16	136.69	-772.63
19.0	92591.90	-48339.00	13957.70	-99.63	2967.34	161.34	-697.42
20.0	94285.30	-49165.00	14265.20	-93.25	2875.59	235.94	-721.92
22.0	97593.80	-50692.00	14904.40	-82.77	2745.24	324.79	-695.61
24.0	100677.00	-51565.30	15201.30	-71.53	2596.73	409.91	-701.93
26.0	103715.00	-52185.40	15380.80	-60.88	2445.30	499.31	-581.96
28.0	106669.00	-52197.30	15136.20	-49.42	2276.34	582.57	-547.22
30.0	109832.00	-52431.30	15114.20	-38.14	2103.73	641.34	-544.99
35.0	116933.00	-49435.10	12742.20	-10.82	1691.80	667.30	-388.35
40.0	123932.00	-43775.70	9268.80	15.25	1356.03	327.73	-339.10

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14C						
	A	B	C	D	E	F	G
2.0	7910.49	1566.52	-112.06	-173.92	1975.67	-202.57	-1582.61
2.25	10090.40	1516.08	-132.53	-164.22	2144.71	-245.91	-1586.24
2.5	12672.30	1230.85	-127.98	-154.40	2293.68	-288.88	-1526.05
2.75	15404.70	785.48	-103.88	-146.02	2435.58	-333.58	-1526.92
3.0	18263.20	174.52	-57.73	-138.13	2539.97	-369.83	-1372.54
4.0	30052.40	-3931.93	484.14	-116.91	2815.30	-467.36	-710.84
5.0	40995.00	-9796.91	1583.72	-113.09	2900.21	-451.56	-204.87
6.0	49804.50	-15620.10	2905.31	-119.64	2970.21	-399.85	-228.44
7.0	56671.50	-20724.30	4228.04	-129.87	3058.54	-347.83	-244.26
8.0	62114.70	-24957.40	5410.68	-135.49	3080.42	-267.82	-216.83
9.0	66532.70	-28492.00	6458.64	-138.92	3102.21	-196.64	-343.21
10.0	70257.00	-31538.30	7424.54	-139.96	3109.64	-131.37	-466.58
11.0	73240.40	-33856.10	8182.60	-139.49	3113.36	-77.52	-528.62
12.0	75830.10	-35829.20	8857.54	-137.30	3097.43	-23.81	-597.83
13.0	78304.00	-37697.30	9499.38	-132.64	3034.49	60.52	-690.28
14.0	80401.00	-39162.40	10022.20	-129.04	3004.11	112.39	-819.41
15.0	82413.50	-40565.20	10547.80	-125.00	2972.01	159.60	-815.35
16.0	84138.60	-41575.10	10920.50	-121.03	2935.91	206.01	-844.59
17.0	85994.20	-42654.40	11295.20	-113.82	2848.12	279.72	-924.47
18.0	87721.10	-43657.50	11664.00	-108.56	2775.07	353.35	-960.97
19.0	89122.20	-44109.80	11806.40	-103.94	2740.54	384.66	-864.21
20.0	90678.60	-44723.70	11996.00	-97.44	2648.86	459.77	-907.84
22.0	93894.70	-46071.00	12444.30	-85.57	2487.47	593.03	-912.09
24.0	96742.60	-46597.20	12482.60	-75.19	2358.14	688.79	-833.76
26.0	99697.50	-47055.90	12472.30	-63.23	2185.39	810.10	-803.84
28.0	102343.00	-46639.70	11970.90	-52.13	2038.03	893.63	-704.66
30.0	105173.00	-46148.00	11326.10	-41.21	1856.73	1002.71	-620.51
35.0	111963.00	-42828.60	8640.91	-13.96	1473.64	1063.44	-455.86
40.0	118574.00	-36526.50	4330.66	12.00	1111.29	892.32	-351.40

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15A/B/C						
	A	B	C	D	E	F	G
2.0	6771.99	897.63	-45.95	-155.96	1478.91	-112.57	-571.21
2.25	8543.84	862.70	-53.16	-148.35	1638.47	-142.90	-603.00
2.5	10454.10	757.88	-56.51	-143.91	1802.08	-178.39	-613.38
2.75	12589.40	536.75	-50.58	-136.31	1939.28	-212.48	-598.75
3.0	15043.50	106.18	-18.51	-127.37	2049.65	-242.76	-584.58
4.0	25256.40	-2809.40	320.40	-108.47	2382.23	-339.78	-246.30
5.0	34995.70	-7157.77	1037.70	-104.27	2547.85	-373.57	64.26
6.0	43079.90	-11755.40	1968.81	-110.42	2669.55	-367.08	207.73
7.0	49495.50	-15880.10	2915.99	-117.70	2745.06	-335.00	79.17
8.0	54674.20	-19541.50	3863.26	-124.97	2823.26	-307.52	-139.52
9.0	58746.90	-22465.30	4666.71	-128.88	2870.36	-274.05	-284.74
10.0	62159.00	-24900.00	5358.04	-129.81	2882.28	-231.65	-307.41
11.0	64980.00	-26916.40	5974.92	-128.99	2890.02	-197.70	-320.91
12.0	67449.80	-28657.30	6533.20	-126.96	2889.14	-168.72	-358.64
13.0	69587.80	-30096.10	7005.49	-125.03	2881.70	-138.49	-417.57
14.0	71617.00	-31412.90	7443.05	-120.37	2839.04	-95.47	-497.72
15.0	73320.90	-32442.90	7811.27	-117.59	2836.73	-78.55	-582.44
16.0	75078.70	-33504.10	8184.69	-111.70	2773.08	-28.70	-569.58
17.0	76605.90	-34256.30	8446.38	-106.43	2722.31	10.58	-648.37
18.0	78201.90	-35135.30	8779.71	-102.00	2687.99	34.04	-637.10
19.0	79683.00	-35825.50	9024.65	-96.68	2626.60	78.21	-644.17
20.0	81040.00	-36264.40	9175.96	-90.42	2571.71	105.53	-621.79
22.0	83842.80	-37347.80	9582.93	-79.77	2452.81	179.87	-678.83
24.0	86457.20	-37934.30	9779.99	-69.09	2348.63	223.29	-555.43
26.0	89143.70	-38488.40	9965.70	-58.22	2222.80	276.21	-541.65
28.0	91552.10	-38289.80	9775.89	-47.03	2083.59	328.54	-483.47
30.0	93976.80	-37775.30	9380.97	-35.17	1933.91	367.06	-412.13
35.0	99743.70	-35109.80	7937.17	-10.10	1701.23	242.55	-292.95
40.0	105747.00	-30710.40	5734.70	16.14	1409.70	-19.63	-330.25

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15D/E/F/H/I						
	A	B	C	D	E	F	G
2.0	6290.79	883.39	-49.29	-150.42	1348.67	-93.23	-194.84
2.25	7850.16	906.09	-62.37	-145.85	1507.07	-121.33	-234.20
2.5	9917.64	729.63	-57.61	-138.51	1649.34	-150.19	-389.61
2.75	12039.70	498.88	-50.28	-132.19	1776.46	-179.02	-384.86
3.0	14308.20	140.88	-27.37	-126.11	1896.47	-208.80	-424.35
4.0	24246.40	-2585.64	274.38	-105.96	2197.31	-292.15	-98.88
5.0	33660.00	-6672.88	931.23	-104.57	2380.99	-330.06	323.27
6.0	41534.90	-11039.20	1790.84	-111.20	2485.37	-318.04	436.06
7.0	47737.40	-14940.00	2668.46	-119.75	2572.84	-293.94	394.87
8.0	52510.40	-18097.60	3446.19	-126.75	2647.38	-274.16	310.51
9.0	56484.50	-20845.30	4162.00	-129.08	2662.71	-225.75	158.84
10.0	59692.00	-23093.90	4799.05	-130.53	2692.07	-199.57	18.86
11.0	62307.70	-24865.90	5320.34	-130.34	2710.88	-176.52	-96.66
12.0	64497.20	-26247.00	5725.38	-127.89	2691.98	-137.42	-152.99
13.0	66473.70	-27479.90	6111.71	-124.64	2678.39	-110.34	-220.62
14.0	68322.50	-28605.10	6471.87	-120.12	2648.26	-78.83	-317.16
15.0	69880.10	-29416.90	6732.96	-115.83	2620.06	-52.26	-351.02
16.0	71504.30	-30337.40	7046.36	-110.89	2583.27	-22.60	-386.91
17.0	72938.30	-31008.00	7269.02	-105.81	2541.55	5.22	-421.21
18.0	74306.50	-31601.90	7471.26	-100.67	2498.95	31.67	-421.69
19.0	75649.10	-32149.50	7661.36	-95.47	2449.77	61.38	-439.23
20.0	76868.40	-32525.30	7793.09	-90.99	2421.09	73.14	-450.75
22.0	79592.40	-33604.00	8197.86	-78.90	2293.07	142.14	-486.11
24.0	81996.10	-34015.70	8295.91	-67.98	2173.93	196.55	-435.49
26.0	84232.50	-34067.60	8271.85	-57.61	2083.11	215.81	-374.64
28.0	86620.60	-34049.50	8171.94	-45.82	1954.61	249.73	-400.41
30.0	88983.60	-33826.80	8026.95	-34.27	1835.41	255.33	-353.18
35.0	94579.10	-31817.80	7120.43	-8.81	1596.94	131.34	-263.56
40.0	100058.00	-27653.80	5318.64	17.12	1355.45	-187.62	-273.88

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 16x16A						
	A	B	C	D	E	F	G
2.0	7213.94	1062.48	-60.18	-163.16	1632.73	-137.39	-660.90
2.25	9068.86	1052.65	-73.90	-157.12	1812.61	-174.53	-682.57
2.5	11282.40	881.74	-74.10	-149.28	1970.43	-212.23	-710.99
2.75	13602.30	625.18	-68.06	-143.44	2124.68	-253.65	-734.52
3.0	16226.30	143.97	-32.51	-136.73	2255.52	-291.73	-699.79
4.0	27528.60	-3346.42	393.54	-115.66	2587.71	-397.43	-273.55
5.0	38357.70	-8605.59	1312.06	-110.58	2719.25	-409.35	60.77
6.0	47353.00	-14184.20	2511.45	-117.96	2810.98	-373.58	26.38
7.0	54492.70	-19227.40	3751.22	-126.74	2889.14	-321.58	-84.61
8.0	60159.30	-23487.00	4884.62	-133.44	2918.29	-242.53	-126.66
9.0	64663.30	-26994.20	5900.01	-137.02	2946.64	-181.25	-285.69
10.0	68346.00	-29851.40	6755.60	-138.49	2958.18	-120.30	-384.11
11.0	71361.10	-32184.10	7502.54	-138.40	2964.72	-68.91	-497.04
12.0	74014.20	-34136.30	8127.59	-135.73	2938.32	-7.78	-627.98
13.0	76326.40	-35820.10	8697.58	-132.72	2908.57	49.64	-715.32
14.0	78450.30	-37288.70	9197.21	-128.85	2871.70	104.32	-771.96
15.0	80439.10	-38636.00	9667.15	-124.14	2815.86	168.64	-851.14
16.0	82142.00	-39610.20	10013.20	-120.20	2790.66	203.72	-859.48
17.0	83886.70	-40590.10	10336.30	-114.04	2714.78	270.50	-870.62
18.0	85580.90	-41545.60	10677.80	-108.53	2648.66	332.69	-921.15
19.0	87028.10	-42030.60	10787.80	-102.57	2576.39	390.15	-880.17
20.0	88490.60	-42584.60	10956.70	-97.67	2529.96	430.91	-912.08
22.0	91586.50	-43770.60	11272.60	-85.21	2343.82	579.90	-878.01
24.0	94293.80	-44158.40	11248.70	-74.44	2224.40	656.22	-824.58
26.0	97086.50	-44420.30	11078.90	-62.82	2045.62	784.53	-737.98
28.0	99965.10	-44515.00	10777.60	-51.29	1871.32	897.77	-719.30
30.0	102352.00	-43418.60	9831.79	-40.46	1725.50	957.49	-626.62
35.0	109039.00	-40353.50	7075.81	-12.07	1286.03	1106.60	-531.72
40.0	115345.00	-34020.20	2448.15	13.49	928.92	963.44	-395.64

Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17A/16x16B/C						
	A	B	C	D	E	F	G
2.0	7482.84	749.18	-32.06	-153.69	1490.54	-111.64	-301.94
2.25	9138.06	783.14	-45.73	-148.43	1678.27	-147.42	-271.38
2.5	11115.90	682.88	-49.46	-143.38	1855.10	-184.65	-248.90
2.75	13492.40	392.81	-34.32	-137.63	2018.42	-224.60	-364.95
3.0	15985.10	3.54	-9.05	-128.84	2149.50	-260.42	-263.00
4.0	27326.30	-3316.13	388.73	-110.89	2545.62	-376.10	-60.44
5.0	38630.20	-8729.17	1335.65	-109.86	2754.84	-407.49	244.70
6.0	48364.20	-14788.30	2652.90	-117.55	2878.88	-375.72	252.15
7.0	56144.10	-20415.70	4068.96	-128.12	2970.68	-312.43	-145.42
8.0	62319.20	-25122.10	5332.37	-133.94	2986.20	-212.65	-192.32
9.0	67097.40	-28916.30	6441.26	-139.07	3028.70	-142.12	-304.90
10.0	71141.80	-32210.80	7461.17	-140.60	3037.68	-63.75	-484.40
11.0	74293.50	-34623.40	8214.63	-140.16	3026.35	11.71	-567.89
12.0	77101.60	-36783.10	8922.19	-138.37	3008.48	83.17	-677.97
13.0	79705.10	-38760.90	9576.13	-134.21	2949.33	173.71	-820.83
14.0	81840.20	-40208.40	10063.30	-130.61	2915.99	236.79	-867.80
15.0	83845.30	-41560.10	10535.80	-126.12	2867.51	306.60	-940.08
16.0	85751.10	-42671.70	10876.60	-120.77	2799.15	386.28	-990.12
17.0	87613.20	-43744.30	11214.60	-114.75	2722.88	466.15	-1028.96
18.0	89198.60	-44487.50	11451.40	-110.00	2673.61	522.32	-974.28
19.0	90843.80	-45204.50	11637.70	-103.89	2591.93	602.99	-1048.14
20.0	92361.20	-45701.20	11710.50	-98.45	2507.40	689.65	-1034.50
22.0	95455.20	-46715.70	11886.10	-86.86	2353.10	835.28	-1006.44
24.0	98319.40	-46988.20	11622.80	-74.63	2169.86	995.06	-941.81
26.0	101240.00	-47039.80	11136.00	-62.32	1971.79	1168.97	-907.73
28.0	103863.00	-46243.10	10186.30	-51.51	1822.28	1270.39	-758.20
30.0	106638.00	-45299.90	9011.04	-39.38	1598.42	1447.93	-698.69
35.0	113059.00	-40056.10	4113.55	-12.17	1169.02	1660.44	-557.52
40.0	119131.00	-30799.70	-3521.78	14.35	791.94	1564.09	-401.82



Table 2.1.28 (cont'd)  
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17B/C						
	A	B	C	D	E	F	G
2.0	6766.33	744.89	-33.96	-154.11	1387.05	-99.30	-455.94
2.25	8406.78	735.84	-42.13	-148.76	1546.40	-127.76	-412.22
2.5	10326.00	618.40	-42.67	-140.84	1696.17	-158.83	-428.21
2.75	12425.70	400.95	-35.11	-134.79	1833.92	-190.65	-448.69
3.0	14787.40	16.36	-8.09	-128.41	1953.16	-221.24	-426.08
4.0	25076.00	-2855.35	319.19	-107.73	2268.19	-307.82	-118.54
5.0	34842.80	-7144.52	1015.11	-107.42	2457.65	-342.14	294.08
6.0	43259.40	-11920.40	1970.81	-113.08	2547.52	-316.78	82.08
7.0	49884.40	-16230.60	2962.56	-122.92	2650.94	-291.11	127.95
8.0	55105.20	-19804.80	3845.74	-128.64	2682.52	-232.47	-61.87
9.0	59268.90	-22820.00	4674.45	-133.56	2742.72	-203.91	-265.03
10.0	62653.20	-25227.80	5347.65	-134.19	2744.28	-150.34	-229.28
11.0	65528.50	-27328.80	5990.85	-134.07	2759.67	-117.12	-349.73
12.0	67925.00	-28930.10	6470.25	-131.66	2738.04	-69.75	-467.93
13.0	70014.00	-30295.30	6903.21	-128.41	2714.49	-27.74	-580.42
14.0	71939.40	-31542.90	7318.09	-124.70	2688.09	8.93	-630.83
15.0	73678.50	-32578.30	7669.57	-120.41	2659.19	41.04	-637.54
16.0	75313.80	-33488.20	7973.96	-115.46	2610.74	86.53	-708.01
17.0	76870.20	-34276.40	8238.11	-110.15	2563.22	123.29	-739.52
18.0	78338.30	-34971.50	8477.60	-104.26	2505.00	166.49	-731.14
19.0	79849.90	-35703.80	8726.57	-99.14	2447.13	211.29	-756.38
20.0	81109.20	-36047.10	8827.48	-93.99	2404.21	235.46	-751.74
22.0	83793.40	-36898.90	9088.73	-82.74	2281.57	313.80	-704.73
24.0	86424.70	-37453.70	9205.18	-70.11	2134.35	393.96	-654.44
26.0	88971.30	-37671.00	9134.01	-58.64	1983.82	478.46	-659.93
28.0	91497.60	-37723.60	9032.79	-47.61	1861.20	520.75	-564.47
30.0	93706.20	-36961.70	8512.11	-37.17	1743.83	543.52	-523.93
35.0	99798.50	-34670.70	6911.55	-9.53	1376.43	593.61	-406.67
40.0	105384.00	-29185.20	3708.34	16.92	1086.25	354.06	-343.59

Table 2.1.29  
BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 7x7B & 10x10F <sup>†</sup>						
	A	B	C	D	E	F	G
2.0	15761.10	10171.40	-1983.74	-180.41	4533.44	-1035.69	-1020.71
2.25	20683.90	10100.50	-2362.96	-171.37	4924.21	-1259.16	-1149.28
2.5	25710.50	9847.51	-2788.08	-162.18	5329.88	-1548.05	-1048.31
2.75	31858.60	7767.18	-2661.83	-154.93	5675.76	-1804.31	-992.87
3.0	38703.40	4333.22	-2101.88	-144.94	5898.42	-1990.59	-1030.87
4.0	65948.40	-16991.70	3924.57	-118.43	6390.16	-2406.62	-614.30
5.0	90881.20	-47264.90	16771.40	-112.75	6498.93	-2241.12	-192.49
6.0	111776.00	-79261.50	33399.20	-115.32	6416.04	-1620.07	-84.57
7.0	127348.00	-107023.00	50534.70	-139.25	6848.43	-1458.29	-14.89
8.0	140072.00	-130028.00	65223.10	-144.93	6836.24	-857.79	-99.75
9.0	150749.00	-150213.00	79005.50	-147.77	6773.51	-231.87	-331.15
10.0	158943.00	-167178.00	92612.70	-164.66	7287.36	-461.83	-382.12
11.0	165714.00	-179168.00	101557.00	-164.07	7241.92	-45.10	-521.50
12.0	171975.00	-190727.00	110548.00	-161.09	7166.19	380.43	-589.16
13.0	177624.00	-200947.00	118921.00	-158.82	7131.17	664.17	-667.75
14.0	182802.00	-210117.00	126526.00	-154.60	7016.50	1083.45	-747.88
15.0	186884.00	-214518.00	128584.00	-147.82	6809.36	1591.41	-783.35
16.0	191316.00	-221293.00	134071.00	-142.04	6646.92	2019.29	-841.16
17.0	195369.00	-231600.00	147624.00	-158.43	7404.40	946.55	-820.02
18.0	199404.00	-236224.00	150408.00	-148.69	7053.70	1655.35	-883.27
19.0	203726.00	-243272.00	157476.00	-143.31	6936.71	1903.09	-895.71
20.0	206861.00	-245479.00	159023.00	-137.13	6829.41	2091.47	-903.40
22.0	213325.00	-250875.00	163825.00	-127.55	6623.17	2500.20	-800.98
24.0	220063.00	-255065.00	166460.00	-114.40	6330.37	2896.83	-803.85
26.0	226903.00	-262541.00	177379.00	-115.77	6627.51	2189.72	-651.65
28.0	234964.00	-270961.00	187677.00	-102.37	6255.46	2595.08	-735.34
30.0	241796.00	-272482.00	188002.00	-88.80	5779.54	3315.93	-731.24
35.0	257457.00	-265751.00	183333.00	-71.68	5676.93	1648.24	-511.23
40.0	282525.00	-292276.00	240288.00	-43.47	4948.25	152.96	-833.96

†Array/Class 10x10F for MPC-68M only.

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8B						
	A	B	C	D	E	F	G
2.0	15913.80	11664.70	-2480.99	-179.56	4694.73	-1100.00	-1003.87
2.25	20652.00	12023.80	-3025.66	-174.12	5204.92	-1412.29	-979.17
2.5	26986.10	10399.30	-3032.60	-163.94	5594.88	-1694.85	-1213.71
2.75	33074.30	8670.65	-3129.69	-156.84	5959.94	-1975.74	-1054.90
3.0	39987.50	5388.94	-2722.03	-146.15	6189.85	-2184.18	-1039.58
4.0	68821.60	-18071.10	4016.97	-119.21	6655.64	-2578.72	-677.77
5.0	95032.70	-50959.00	18228.50	-113.67	6737.08	-2341.46	-253.74
6.0	117864.00	-88879.60	39468.80	-128.75	6937.68	-1918.61	-203.01
7.0	133919.00	-117151.00	56431.30	-139.69	6960.80	-1212.83	-123.38
8.0	147621.00	-142952.00	73246.80	-143.67	6879.18	-441.73	-342.11
9.0	158036.00	-165478.00	90946.70	-167.32	7480.35	-551.45	-378.22
10.0	166796.00	-181378.00	101771.00	-165.98	7346.03	114.50	-504.04
11.0	174312.00	-195869.00	112810.00	-165.26	7291.07	642.48	-648.03
12.0	180736.00	-207916.00	122412.00	-163.34	7243.01	1055.04	-742.81
13.0	187002.00	-219945.00	132127.00	-159.70	7084.08	1641.84	-903.88
14.0	192382.00	-229413.00	139613.00	-156.32	7001.62	2085.84	-972.60
15.0	196087.00	-233618.00	142299.00	-151.48	6860.06	2570.55	-883.73
16.0	202268.00	-249608.00	159974.00	-162.80	7359.57	1999.93	-1048.13
17.0	206376.00	-256109.00	166401.00	-159.20	7309.03	2257.68	-1062.93
18.0	209117.00	-255071.00	162389.00	-151.82	7125.28	2596.49	-891.61
19.0	213124.00	-261295.00	168674.00	-146.82	7004.96	2966.11	-951.40
20.0	217047.00	-267281.00	175609.00	-141.96	6943.62	3118.99	-1012.59
22.0	223569.00	-268761.00	171389.00	-127.42	6436.52	4175.11	-877.23
24.0	233533.00	-291046.00	200512.00	-131.73	6830.33	3613.57	-988.74
26.0	238557.00	-284966.00	188216.00	-118.63	6424.02	4316.86	-862.50
28.0	245385.00	-285588.00	185055.00	-105.51	6116.61	4651.69	-844.39
30.0	254559.00	-295608.00	196106.00	-100.36	6027.39	4465.31	-886.90
35.0	272231.00	-295589.00	203313.00	-71.05	5259.94	4464.18	-744.47
40.0	290782.00	-286198.00	204311.00	-50.38	4868.38	2364.75	-614.59

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8C/D/E						
	A	B	C	D	E	F	G
2.0	16663.00	10889.80	-2211.52	-182.17	4831.25	-1175.27	-1260.49
2.25	21598.90	10980.20	-2691.18	-176.65	5300.72	-1453.46	-1219.04
2.5	27348.40	10071.30	-2967.33	-165.41	5680.31	-1735.86	-1252.79
2.75	33467.10	8232.39	-2999.52	-158.56	6061.56	-2033.93	-1086.98
3.0	40382.30	4849.42	-2525.53	-148.53	6314.10	-2257.89	-1075.95
4.0	68954.10	-18263.30	4048.93	-123.13	6850.62	-2734.70	-652.59
5.0	96324.30	-53730.10	19778.60	-114.90	6841.59	-2381.30	-353.71
6.0	118229.00	-89906.60	39997.30	-134.45	7190.60	-2120.86	-143.41
7.0	134948.00	-119919.00	58227.10	-143.18	7200.03	-1397.69	-170.37
8.0	149092.00	-147517.00	76590.50	-149.16	7110.00	-528.97	-313.19
9.0	159771.00	-170139.00	93968.00	-170.19	7649.69	-595.38	-403.04
10.0	168715.00	-187828.00	107088.00	-172.19	7651.82	-46.57	-555.81
11.0	176169.00	-201821.00	117349.00	-170.83	7550.84	552.84	-651.76
12.0	182662.00	-214445.00	127628.00	-169.36	7519.56	997.32	-756.73
13.0	189114.00	-227085.00	137699.00	-166.11	7388.07	1583.27	-844.97
14.0	195273.00	-239345.00	148361.00	-160.79	7228.22	2124.28	-1017.11
15.0	199939.00	-249862.00	159949.00	-174.10	7782.47	1566.35	-1026.32
16.0	204899.00	-258274.00	166856.00	-167.77	7534.06	2227.05	-1070.51
17.0	209356.00	-265290.00	173458.00	-161.96	7463.49	2386.89	-1040.14
18.0	213546.00	-272476.00	180667.00	-158.41	7387.49	2763.66	-1098.37
19.0	217506.00	-277100.00	183949.00	-150.21	7155.18	3240.82	-1107.07
20.0	219837.00	-275266.00	179705.00	-145.05	7009.96	3638.55	-1007.16
22.0	228092.00	-285272.00	186688.00	-133.55	6672.08	4473.64	-1122.87
24.0	237213.00	-304032.00	211958.00	-136.95	7000.92	4086.48	-1049.61
26.0	242060.00	-297359.00	199620.00	-125.83	6734.22	4465.79	-972.10
28.0	249432.00	-299622.00	196900.00	-111.26	6222.03	5440.43	-914.71
30.0	263307.00	-334844.00	247655.00	-111.83	6452.32	4775.31	-1191.53
35.0	273393.00	-291765.00	178985.00	-83.84	5736.80	4650.87	-621.35
40.0	293153.00	-283353.00	175255.00	-57.06	4937.79	3684.27	-559.25

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9A						
	A	B	C	D	E	F	G
2.0	16564.30	12063.20	-2586.67	-184.87	4976.49	-1228.06	-894.91
2.25	22071.80	11834.70	-3015.91	-174.88	5443.22	-1518.94	-1014.33
2.5	27866.60	10993.50	-3286.54	-168.71	5965.88	-1909.06	-1027.88
2.75	34375.10	9004.62	-3367.62	-158.97	6305.05	-2182.06	-933.24
3.0	41566.50	5392.11	-2800.23	-149.79	6613.45	-2462.36	-904.38
4.0	72006.50	-20264.40	4921.01	-123.85	7211.86	-3004.62	-603.22
5.0	100197.00	-57315.80	21669.60	-118.72	7356.33	-2796.24	-243.52
6.0	124367.00	-99348.10	46264.80	-136.71	7648.05	-2394.38	-67.58
7.0	143009.00	-134740.00	68824.10	-143.35	7544.90	-1403.30	-173.80
8.0	157479.00	-165996.00	92255.30	-168.05	8114.30	-1315.88	-266.71
9.0	169636.00	-191379.00	110928.00	-172.50	8069.55	-500.37	-450.57
10.0	179282.00	-211202.00	125969.00	-172.12	7976.57	283.36	-617.13
11.0	187512.00	-228637.00	140325.00	-172.16	7928.03	894.69	-760.39
12.0	195321.00	-245580.00	154682.00	-170.38	7824.20	1596.02	-863.97
13.0	202110.00	-263050.00	173293.00	-187.18	8470.09	1003.55	-953.17
14.0	208171.00	-274758.00	183332.00	-179.75	8249.83	1717.21	-1103.07
15.0	213590.00	-284590.00	191650.00	-175.64	8098.33	2289.04	-1165.13
16.0	218091.00	-292503.00	199557.00	-171.84	8035.82	2659.38	-1119.03
17.0	223491.00	-302449.00	208733.00	-164.92	7833.36	3192.21	-1255.80
18.0	226523.00	-304524.00	209895.00	-162.71	7829.04	3410.57	-1091.33
19.0	231702.00	-312496.00	215730.00	-153.73	7552.13	4052.91	-1189.12
20.0	236531.00	-324776.00	232293.00	-164.72	8073.05	3368.73	-1233.57
22.0	244888.00	-335452.00	241932.00	-150.44	7566.26	4642.58	-1160.69
24.0	252171.00	-340795.00	244542.00	-141.18	7321.23	5355.16	-1142.40
26.0	259438.00	-343494.00	244340.00	-129.66	7094.56	5645.82	-1119.92
28.0	268823.00	-359239.00	266068.00	-130.16	7204.93	5605.85	-1064.30
30.0	277221.00	-363922.00	268930.00	-116.96	6799.84	6219.78	-1037.79
35.0	294285.00	-351643.00	245914.00	-99.35	6404.25	5923.44	-713.23
40.0	324174.00	-389397.00	319233.00	-77.68	5933.52	3992.56	-1188.62

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9B						
	A	B	C	D	E	F	G
2.0	17186.80	11657.20	-2441.58	-183.45	5049.98	-1246.51	-1156.40
2.25	21800.20	12295.50	-3074.77	-180.94	5660.86	-1631.58	-1064.82
2.5	28010.00	11198.70	-3349.88	-169.84	6074.18	-1943.73	-1220.46
2.75	34607.80	9092.75	-3327.98	-161.55	6476.70	-2279.47	-1090.70
3.0	41425.40	6300.12	-3202.59	-151.95	6782.84	-2566.85	-1000.46
4.0	71942.80	-18734.90	3920.65	-125.38	7367.52	-3119.27	-631.75
5.0	101151.00	-57291.00	21182.10	-118.05	7377.24	-2721.50	-361.88
6.0	125823.00	-99944.80	45636.60	-136.47	7588.00	-2124.69	-262.67
7.0	144638.00	-135378.00	67687.60	-143.88	7447.72	-995.76	-340.94
8.0	159872.00	-168383.00	91921.20	-168.66	7933.70	-673.04	-395.74
9.0	172305.00	-194121.00	110332.00	-172.16	7831.09	301.31	-634.37
10.0	181683.00	-213140.00	124418.00	-173.36	7740.03	1165.16	-753.12
11.0	190922.00	-232977.00	140095.00	-171.28	7581.53	2053.29	-1027.00
12.0	198213.00	-248066.00	152236.00	-170.70	7492.96	2781.03	-1087.99
13.0	205947.00	-268590.00	173240.00	-187.42	8096.44	2390.78	-1199.48
14.0	211867.00	-280583.00	184192.00	-183.14	8023.23	2903.27	-1325.04
15.0	217071.00	-289407.00	190649.00	-177.77	7760.30	3819.17	-1355.68
16.0	221340.00	-294404.00	193178.00	-173.59	7653.54	4235.81	-1282.26
17.0	227205.00	-306489.00	204027.00	-164.96	7309.81	5290.73	-1440.44
18.0	231085.00	-310612.00	206608.00	-160.03	7176.88	5715.32	-1383.11
19.0	236345.00	-320398.00	215697.00	-153.84	7020.00	6284.82	-1522.44
20.0	240125.00	-328538.00	227545.00	-170.25	7836.24	5008.11	-1382.77
22.0	245672.00	-325279.00	216287.00	-158.18	7517.98	5919.63	-1187.15
24.0	256479.00	-345503.00	236771.00	-144.07	6970.57	7508.12	-1317.75
26.0	260950.00	-331434.00	205388.00	-130.57	6497.58	8638.70	-1076.78
28.0	269984.00	-343628.00	218366.00	-134.58	6861.68	8165.52	-1062.58
30.0	278259.00	-348285.00	221391.00	-123.31	6538.19	8720.28	-1076.88
35.0	297697.00	-344053.00	202586.00	-105.06	6094.38	9194.58	-852.15
40.0	331243.00	-401432.00	313358.00	-81.82	5561.33	7636.50	-1470.42

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9C/D						
	A	B	C	D	E	F	G
2.0	16691.80	11823.60	-2447.14	-185.99	5008.36	-1243.90	-1059.30
2.25	21740.60	12301.10	-3136.66	-173.22	5422.51	-1511.79	-1061.56
2.5	27709.70	11300.00	-3398.46	-167.10	5898.90	-1850.17	-1171.40
2.75	33988.10	9774.59	-3696.16	-158.15	6268.38	-2155.04	-974.14
3.0	41117.20	6515.41	-3381.03	-148.32	6548.78	-2413.74	-948.98
4.0	71428.60	-18297.80	3576.44	-123.51	7125.21	-2923.50	-632.21
5.0	100397.00	-56458.80	20611.70	-115.75	7125.58	-2528.06	-313.97
6.0	124283.00	-97234.10	43750.10	-135.36	7393.89	-2038.45	-178.07
7.0	142677.00	-131502.00	64937.90	-142.42	7276.64	-994.67	-255.89
8.0	158111.00	-164750.00	89150.00	-165.13	7682.79	-614.18	-382.56
9.0	169539.00	-187815.00	105688.00	-170.16	7701.54	95.21	-536.66
10.0	179168.00	-207560.00	120407.00	-172.05	7615.14	907.40	-757.15
11.0	187428.00	-224318.00	133228.00	-170.11	7472.64	1710.47	-885.30
12.0	195546.00	-241540.00	147050.00	-166.19	7281.30	2560.85	-1135.94
13.0	202256.00	-258699.00	164971.00	-182.40	7906.42	2044.37	-1182.19
14.0	207838.00	-268927.00	173192.00	-178.93	7770.91	2703.98	-1224.09
15.0	213979.00	-281611.00	184781.00	-172.75	7552.21	3409.13	-1276.86
16.0	217809.00	-285839.00	187221.00	-168.56	7458.11	3805.42	-1317.69
17.0	223749.00	-297214.00	196642.00	-160.86	7141.47	4676.19	-1362.21
18.0	226075.00	-295937.00	193130.00	-157.66	7127.19	4895.03	-1291.13
19.0	230997.00	-304670.00	201281.00	-150.53	6907.85	5558.32	-1353.07
20.0	238022.00	-324930.00	227066.00	-158.32	7284.25	5103.45	-1464.16
22.0	243676.00	-322706.00	217208.00	-147.77	6978.74	5979.30	-1239.05
24.0	251683.00	-332524.00	227486.00	-137.48	6744.91	6651.45	-1261.39
26.0	256408.00	-321812.00	204514.00	-125.79	6394.39	7373.18	-1135.32
28.0	264537.00	-330729.00	215269.00	-131.03	6864.20	6415.84	-1014.55
30.0	273958.00	-341208.00	225146.00	-115.29	6196.43	7947.39	-1073.39
35.0	292385.00	-333153.00	204415.00	-98.00	5956.86	7222.98	-860.79
40.0	329247.00	-419504.00	371883.00	-71.42	4943.73	7633.01	-1618.27

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9E/F						
	A	B	C	D	E	F	G
2.0	16854.60	11084.70	-2322.04	-181.73	4769.99	-1147.12	-810.29
2.25	21630.80	11546.20	-2940.96	-172.49	5228.13	-1436.00	-839.61
2.5	27849.90	10029.20	-2985.66	-164.15	5650.51	-1736.59	-1040.92
2.75	34540.60	7548.11	-2786.62	-154.38	5990.92	-2013.50	-935.15
3.0	41307.10	4337.80	-2362.16	-146.82	6295.85	-2275.82	-884.96
4.0	70768.40	-20480.20	5197.61	-121.39	6876.47	-2797.83	-537.40
5.0	98180.80	-56583.30	21720.10	-115.24	7004.63	-2612.66	-168.15
6.0	120573.00	-94683.40	43765.30	-134.45	7390.91	-2400.88	20.85
7.0	138493.00	-128353.00	65326.00	-141.23	7368.45	-1657.87	2.12
8.0	151304.00	-154813.00	84923.70	-165.48	7997.42	-1799.73	-3.75
9.0	162835.00	-178601.00	102770.00	-169.20	8012.87	-1222.27	-178.21
10.0	173089.00	-200396.00	119704.00	-169.43	7906.04	-489.94	-481.35
11.0	180227.00	-213998.00	130552.00	-169.48	7924.61	-143.28	-537.04
12.0	188058.00	-230819.00	144797.00	-165.45	7782.15	482.35	-705.69
13.0	193490.00	-240795.00	153382.00	-163.80	7756.04	834.76	-753.66
14.0	199338.00	-255751.00	170303.00	-178.59	8424.78	16.81	-795.55
15.0	204471.00	-264530.00	177215.00	-172.61	8186.47	708.91	-873.25
16.0	209807.00	-275635.00	189071.00	-167.97	8087.71	1042.99	-936.73
17.0	214452.00	-282609.00	194830.00	-159.86	7819.12	1616.41	-906.17
18.0	217197.00	-283928.00	195786.00	-157.56	7869.81	1568.69	-890.15
19.0	221266.00	-288837.00	199363.00	-149.64	7592.40	2213.50	-965.82
20.0	225737.00	-295774.00	205279.00	-143.23	7337.40	2875.11	-876.23
22.0	234598.00	-314227.00	231133.00	-148.51	7825.76	2021.35	-879.15
24.0	242046.00	-320606.00	235951.00	-134.75	7367.58	2926.98	-913.50
26.0	247960.00	-318479.00	229552.00	-123.51	7133.33	3171.11	-783.22
28.0	261521.00	-352854.00	278305.00	-120.41	7120.21	3024.72	-1121.44
30.0	264913.00	-340198.00	263913.00	-111.92	6968.28	2888.33	-788.23
35.0	288082.00	-360268.00	293412.00	-86.40	6220.44	2894.70	-961.02
40.0	298948.00	-303570.00	215523.00	-55.72	5417.82	785.23	-415.39



Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9G						
	A	B	C	D	E	F	G
2.0	18157.70	12664.10	-2736.69	-182.35	5344.31	-1383.11	-916.79
2.25	23646.70	12752.10	-3248.16	-178.95	5971.94	-1793.73	-925.78
2.5	29660.10	12309.80	-3821.64	-169.21	6473.09	-2183.65	-879.92
2.75	36525.80	10358.80	-3962.11	-162.46	6968.29	-2613.38	-863.49
3.0	44006.40	7030.85	-3698.49	-153.38	7336.54	-2971.63	-809.92
4.0	77288.30	-21207.50	4543.15	-125.70	8058.78	-3705.78	-537.87
5.0	110686.00	-69960.20	29062.30	-130.54	8442.77	-3626.36	-336.85
6.0	137786.00	-118830.00	58088.00	-136.52	8339.36	-2532.48	-201.40
7.0	160795.00	-169293.00	94340.50	-161.16	8672.27	-1671.25	-379.07
8.0	177763.00	-207034.00	122389.00	-170.18	8619.96	-400.24	-562.99
9.0	193108.00	-243101.00	150849.00	-171.94	8368.05	1156.18	-881.11
10.0	205042.00	-275555.00	181997.00	-195.35	9071.69	1098.87	-1083.51
11.0	215280.00	-300568.00	204362.00	-194.55	8934.09	2200.13	-1266.10
12.0	223585.00	-319189.00	220301.00	-191.69	8775.21	3201.84	-1325.62
13.0	230947.00	-335777.00	234994.00	-189.96	8659.97	4110.52	-1472.39
14.0	239135.00	-355478.00	253619.00	-183.93	8406.36	5194.67	-1726.13
15.0	245572.00	-374776.00	278406.00	-203.34	9278.36	4194.86	-1666.34
16.0	251881.00	-387322.00	288544.00	-193.80	8836.24	5557.89	-1689.56
17.0	257861.00	-401610.00	304798.00	-189.68	8737.81	6220.47	-1840.71
18.0	262232.00	-408488.00	311370.00	-185.11	8602.16	6925.67	-1728.75
19.0	265329.00	-406025.00	301388.00	-178.52	8347.70	7730.36	-1689.95
20.0	271234.00	-419055.00	315509.00	-171.72	8067.36	8751.47	-1705.40
22.0	283895.00	-451199.00	356261.00	-175.40	8389.72	8926.87	-1890.66
24.0	288388.00	-437401.00	323902.00	-164.80	8075.31	9968.86	-1575.02
26.0	299757.00	-459004.00	349014.00	-154.15	7793.16	11086.10	-1690.60
28.0	312233.00	-487890.00	389532.00	-156.41	8001.62	11248.70	-1695.28
30.0	317451.00	-470929.00	352843.00	-144.12	7616.90	12129.50	-1519.49
35.0	340908.00	-472938.00	320383.00	-126.33	6958.19	14189.40	-1265.87
40.0	355826.00	-406707.00	181832.00	-109.88	6567.54	13350.90	-690.33

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 10x10A/B/G <sup>†</sup>						
	A	B	C	D	E	F	G
2.0	16284.00	11316.60	-2373.42	-183.95	4757.49	-1129.72	-908.53
2.25	21494.10	11161.90	-2738.06	-174.87	5233.98	-1435.08	-1029.88
2.5	27378.90	10122.70	-3001.13	-163.37	5590.72	-1687.18	-1133.76
2.75	33997.50	7667.21	-2796.85	-154.59	5934.47	-1960.21	-1063.93
3.0	40669.30	4604.85	-2427.68	-146.64	6233.46	-2224.40	-1023.08
4.0	69456.60	-19048.60	4510.80	-121.07	6769.53	-2693.26	-595.32
5.0	96363.50	-53810.50	20060.80	-115.15	6852.01	-2455.28	-235.29
6.0	118075.00	-89649.00	40101.30	-135.03	7207.34	-2199.03	-31.82
7.0	135465.00	-121448.00	59891.00	-141.81	7176.22	-1464.52	-84.35
8.0	149172.00	-147759.00	77477.10	-146.29	7123.94	-720.75	-270.69
9.0	160098.00	-171854.00	96698.30	-168.49	7716.07	-861.33	-341.94
10.0	168703.00	-188210.00	108590.00	-170.65	7707.01	-369.98	-413.26
11.0	176895.00	-205123.00	122221.00	-167.56	7590.63	267.07	-597.28
12.0	183500.00	-217775.00	132403.00	-165.29	7503.92	748.16	-696.44
13.0	189527.00	-229054.00	141757.00	-162.77	7481.92	1050.96	-848.98
14.0	195892.00	-241671.00	152138.00	-155.37	7192.81	1854.09	-983.23
15.0	199561.00	-249322.00	161820.00	-172.75	7962.69	824.80	-863.19
16.0	204447.00	-258563.00	171271.00	-167.33	7839.02	1163.01	-928.77
17.0	209187.00	-266807.00	178586.00	-160.49	7588.94	1870.46	-983.28
18.0	212908.00	-270532.00	180865.00	-155.48	7487.99	2077.63	-955.84
19.0	216478.00	-274912.00	185127.00	-150.92	7417.63	2302.50	-949.30
20.0	219761.00	-276790.00	185299.00	-144.53	7207.71	2794.21	-860.04
22.0	230330.00	-297894.00	208958.00	-142.95	7317.84	2710.62	-1141.54
24.0	235204.00	-296597.00	207242.00	-136.96	7299.78	2658.68	-881.02
26.0	243035.00	-302622.00	210474.00	-120.72	6753.85	3686.66	-891.14
28.0	250446.00	-307503.00	216130.00	-107.51	6366.92	4185.55	-863.84
30.0	265199.00	-348982.00	280458.00	-107.22	6539.80	3562.03	-1192.36
35.0	273468.00	-298369.00	203934.00	-79.97	5875.23	3082.40	-627.85
40.0	292898.00	-285148.00	187876.00	-50.41	4835.07	2436.15	-509.94

<sup>†</sup>Array/Class 10x10G for MPC-68M only.

Table 2.1.29 (cont'd)  
 BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
 (ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 10x10C						
	A	B	C	D	E	F	G
2.0	17325.30	11490.30	-2423.96	-183.30	5030.60	-1243.75	-1042.41
2.25	22130.00	11951.30	-2993.28	-179.73	5638.45	-1641.98	-1049.38
2.5	28141.40	10893.00	-3249.42	-171.97	6092.80	-1970.42	-1042.25
2.75	35001.90	8485.77	-3132.08	-161.49	6464.02	-2288.54	-1064.03
3.0	41817.40	5588.18	-2935.15	-152.37	6778.27	-2580.33	-960.42
4.0	72503.80	-20126.90	4676.40	-126.12	7389.26	-3161.51	-598.75
5.0	101686.00	-58844.80	22172.30	-118.88	7430.83	-2824.08	-314.90
6.0	125964.00	-100714.00	46115.40	-137.38	7670.65	-2280.40	-139.13
7.0	145279.00	-138063.00	69971.00	-145.81	7593.29	-1239.47	-240.17
8.0	160736.00	-171770.00	94922.90	-169.48	8074.18	-936.98	-413.14
9.0	173109.00	-198050.00	114195.00	-173.24	7952.04	107.22	-587.69
10.0	183348.00	-219689.00	130706.00	-174.38	7886.25	887.26	-747.19
11.0	192349.00	-239413.00	146643.00	-173.03	7738.68	1801.89	-960.79
12.0	198722.00	-251849.00	156661.00	-174.40	7779.41	2247.21	-1024.32
13.0	206317.00	-271870.00	177242.00	-191.21	8405.58	1825.60	-1138.70
14.0	212647.00	-284224.00	187282.00	-183.63	8103.28	2759.09	-1219.61
15.0	218920.00	-297923.00	200391.00	-179.50	7978.82	3335.37	-1313.57
16.0	223379.00	-304963.00	206476.00	-175.76	7922.23	3689.54	-1328.16
17.0	228676.00	-314595.00	214380.00	-168.29	7569.76	4728.35	-1384.57
18.0	233175.00	-321606.00	220636.00	-164.63	7582.84	4872.65	-1394.73
19.0	238334.00	-334048.00	236292.00	-170.69	7886.97	4618.40	-1403.78
20.0	242429.00	-340497.00	242818.00	-172.36	8094.92	4434.37	-1437.97
22.0	251428.00	-353397.00	253878.00	-156.59	7500.41	6060.21	-1412.04
24.0	257957.00	-354461.00	249954.00	-147.71	7305.10	6634.39	-1346.94
26.0	272010.00	-391459.00	299301.00	-145.25	7227.25	7258.81	-1619.05
28.0	273995.00	-368436.00	261102.00	-136.90	7071.78	7562.48	-1159.20
30.0	279666.00	-356857.00	232864.00	-125.34	6696.43	8273.08	-973.58
35.0	297242.00	-340805.00	191056.00	-108.66	6404.77	8127.91	-777.55
40.0	330405.00	-398218.00	299749.00	-84.01	5531.03	7980.06	-1232.79

Table 2.1.30

MPC Regionalized Loading Heat Load Limits ( $q_1$  and  $q_2$ )<sup>1</sup> for Discrete Values of X

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

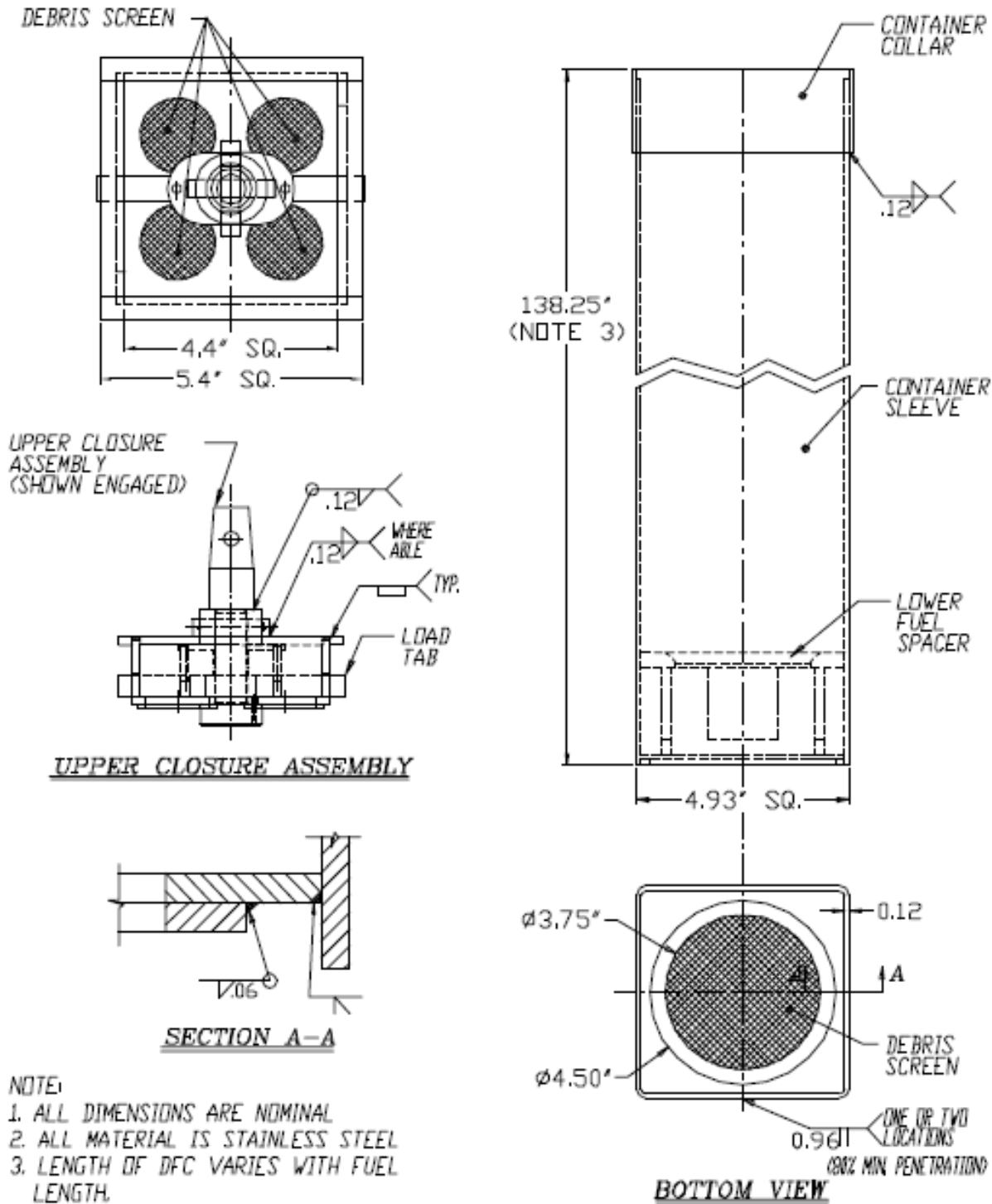
\*See Table 2.1.27 for the number of storage cells (n) in each region for the specific MPC type listed.

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

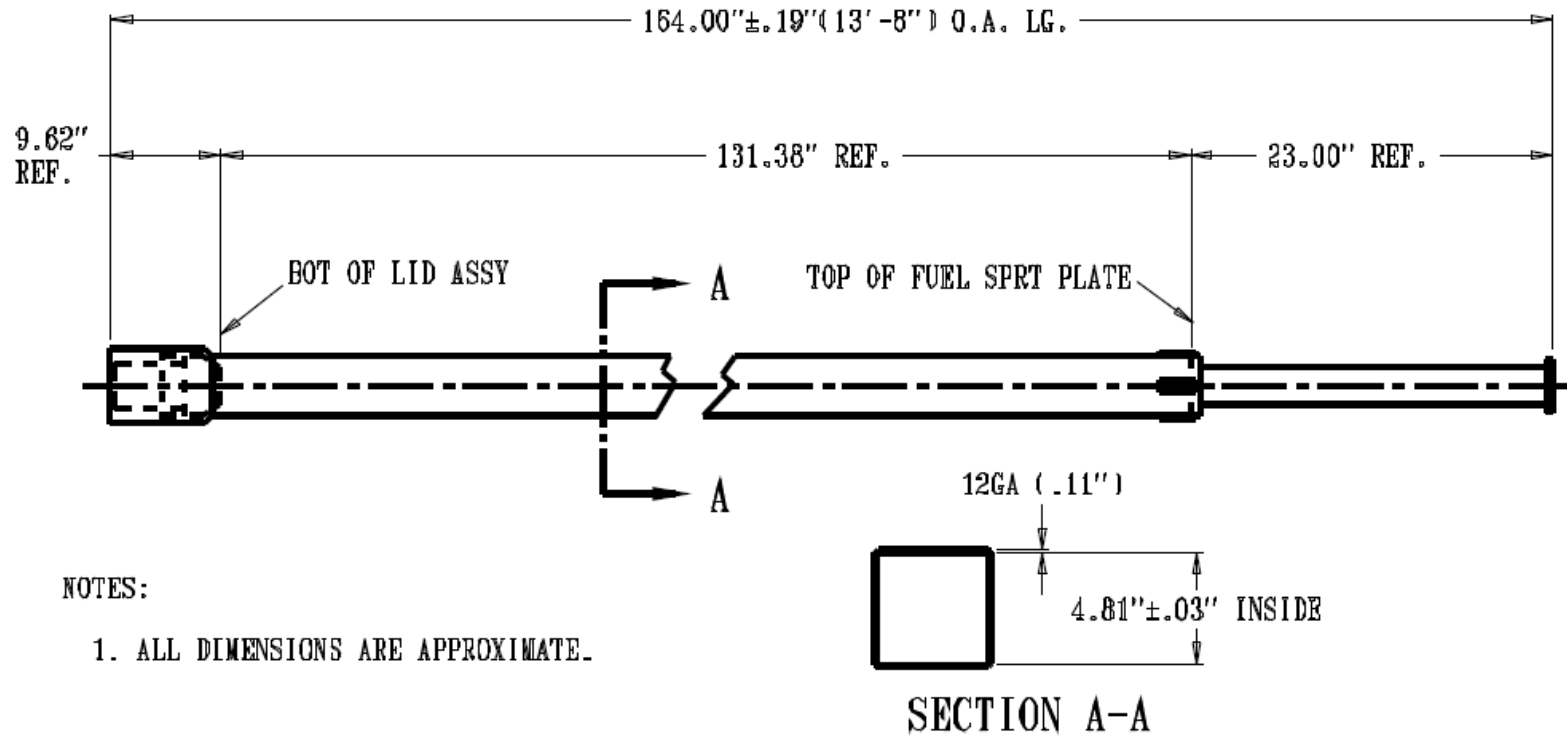
Table 2.1.31

MPC Heat Load Limits for Lower Helium Backfill Range<sup>Note 2</sup>

<u>Uniform Loading</u>				
<u>MPC Type</u>	<u>Number of Cells</u>		<u>Storage Cell Heat Load Limit (kW)</u>	
<u>MPC-24</u>	<u>24</u>		<u>1.157</u>	
<u>MPC-24E/EF</u>	<u>24</u>		<u>1.173</u>	
<u>MPC-32/32F</u>	<u>32</u>		<u>0.898</u>	
<u>MPC-68/68F/68FF</u>	<u>68</u>		<u>0.414</u>	
<u>Regionalized Loading</u>				
<u>MPC Type</u>	<u>Number of Cells in Inner Region</u> <sup>Note 1</sup>	<u>Storage Cell Heat Load (Inner Region) (kW)</u>	<u>Number of Cells in Outer Region</u> <sup>Note 1</sup>	<u>Storage Cell Heat Load</u>
<u>MPC-24</u>	<u>4</u>	<u>1.470</u>	<u>20</u>	<u>0.900</u>
<u>MPC-24E/EF</u>	<u>4</u>	<u>1.540</u>	<u>20</u>	<u>0.900</u>
<u>MPC-32/32F</u>	<u>12</u>	<u>1.131</u>	<u>20</u>	<u>0.600</u>
<u>MPC-68/68F/68FF</u>	<u>32</u>	<u>0.500</u>	<u>36</u>	<u>0.275</u>
<u>Note 1: For regionalized loading:</u>				
<ul style="list-style-type: none"> <li>• <u>MPC-24/24E/24EF – Inner Region: Cells 9, 10, 15, 16 Outer Region: Cells 1-8, 11-14, 17-24 in Figure 1.2.4</u></li> <li>• <u>MPC-32/32F – Inner Region: Cells 7, 8, 12-15, 18-21, 25, 26 Outer Region: Cells 1-4, 5, 6, 9-11, 16, 17, 22-24, 27-32 in Figure 1.2.3</u></li> <li>• <u>MPC-68/68F/68FF – Inner Region: Cells 11-14, 18-23, 27-32, 37-42, 46-51, 55-58 Outer Region: Cells 1-10, 15-17, 24-26, 33-36, 43-45, 52-54, 59-68 in Figure 1.2.2</u></li> </ul>				
<u>Note 2: Under SCS mandatory conditions evaluated in HI-TRAC operations Section 4.5, the storage cell heat loads tabulated herein are limited by the heat load reduction factor defined in Table 4.5.4</u>				



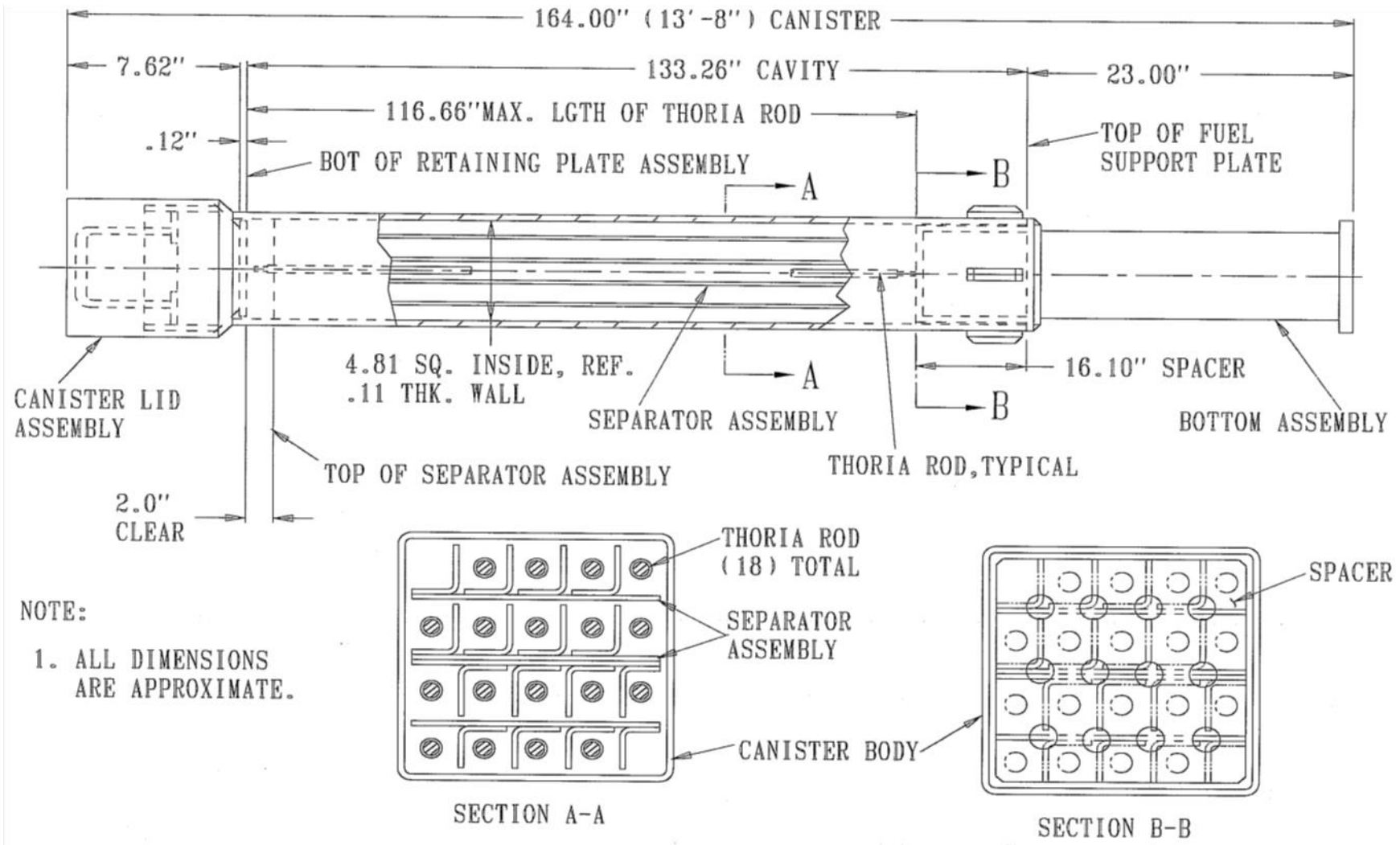
**FIGURE 2.1.1: DAMAGED FUEL CONTAINER FOR DRESDEN UNIT-1/  
HUMBOLDT BAY SNF**



NOTES:

1. ALL DIMENSIONS ARE APPROXIMATE.

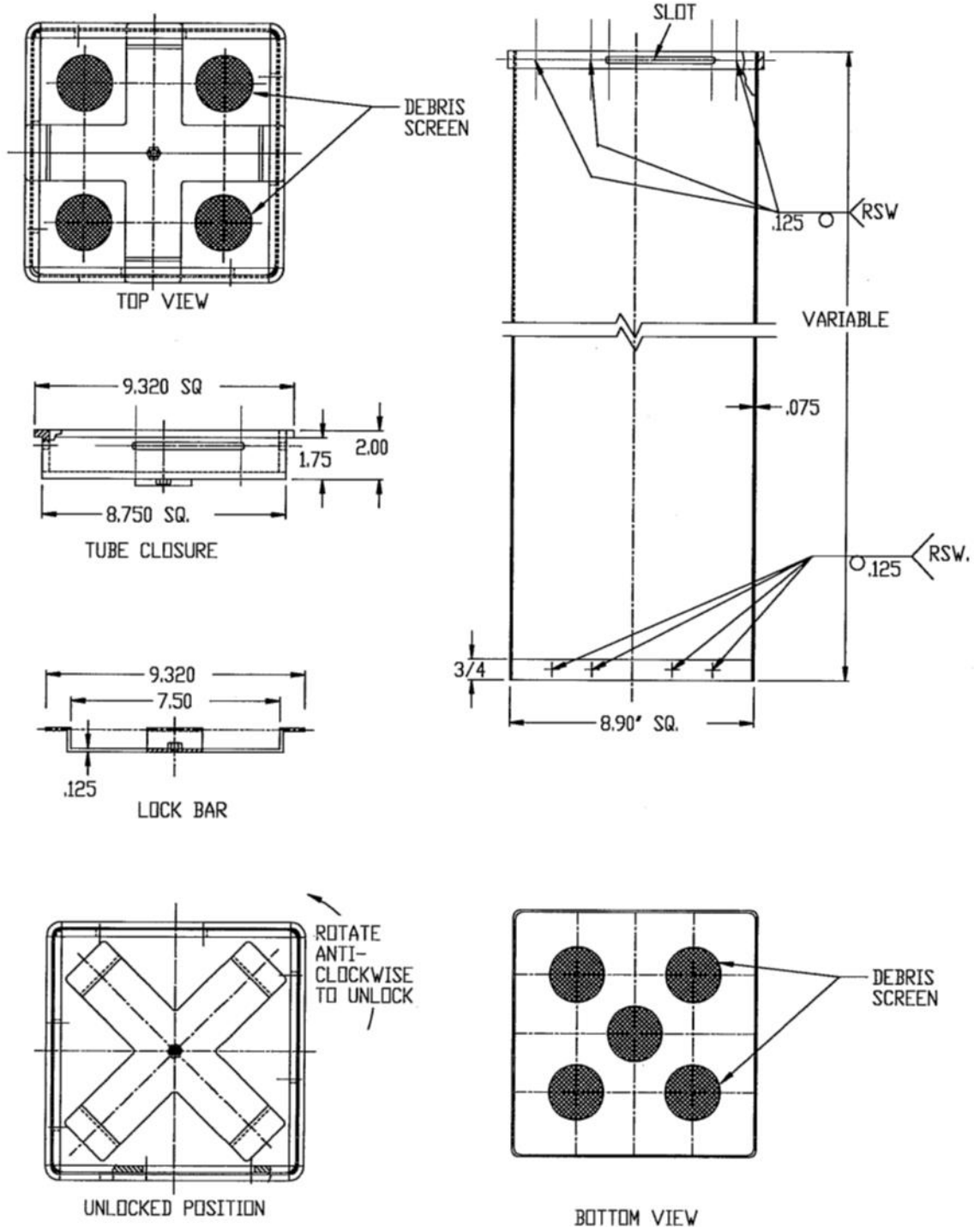
FIGURE 2.1.2: TN DAMAGED FUEL CANISTER FOR DRESDEN UNIT-1



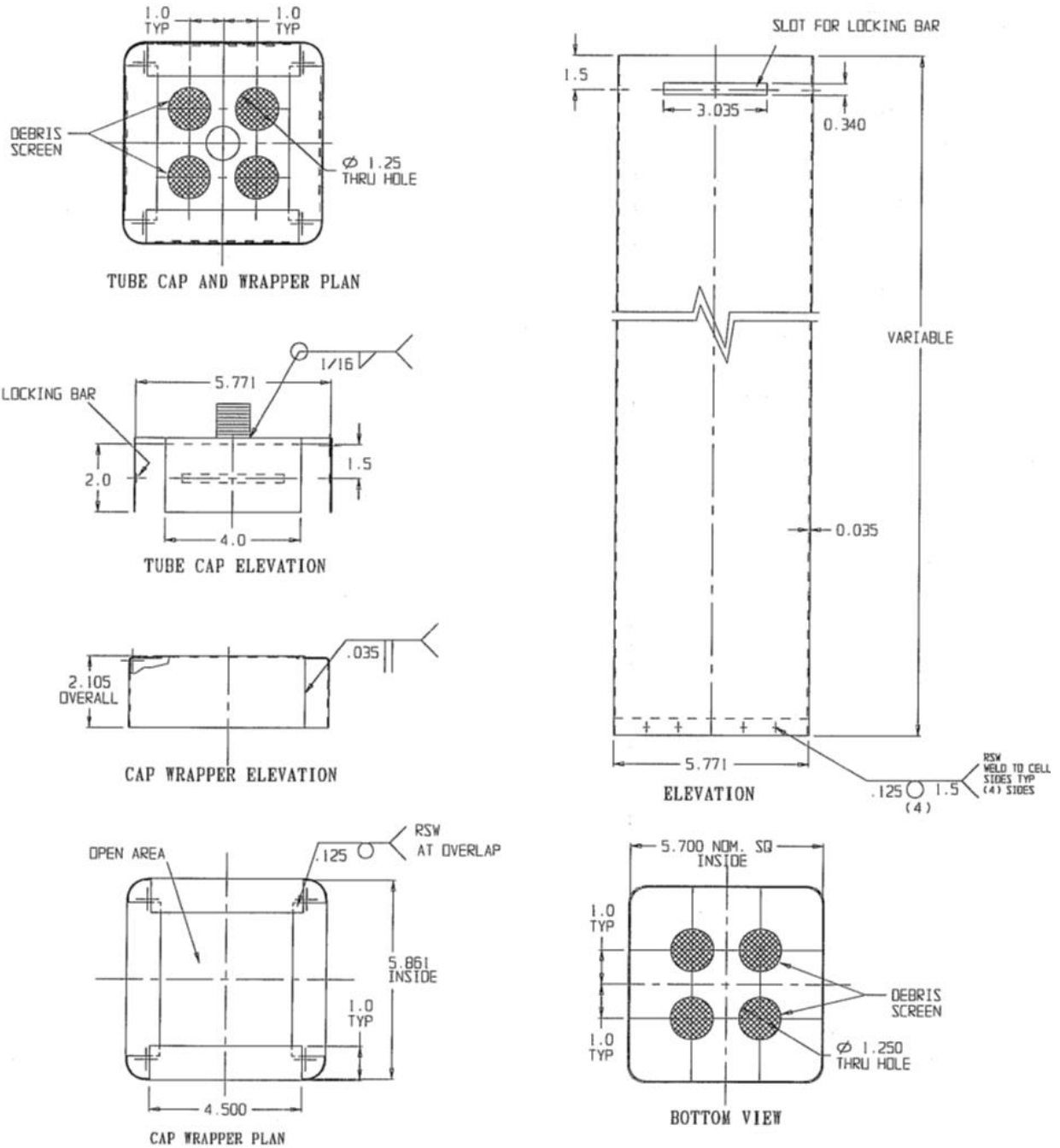
**FIGURE 2.1.2A: TN THORIA ROD CANISTER FOR DRESDEN UNIT-1**



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

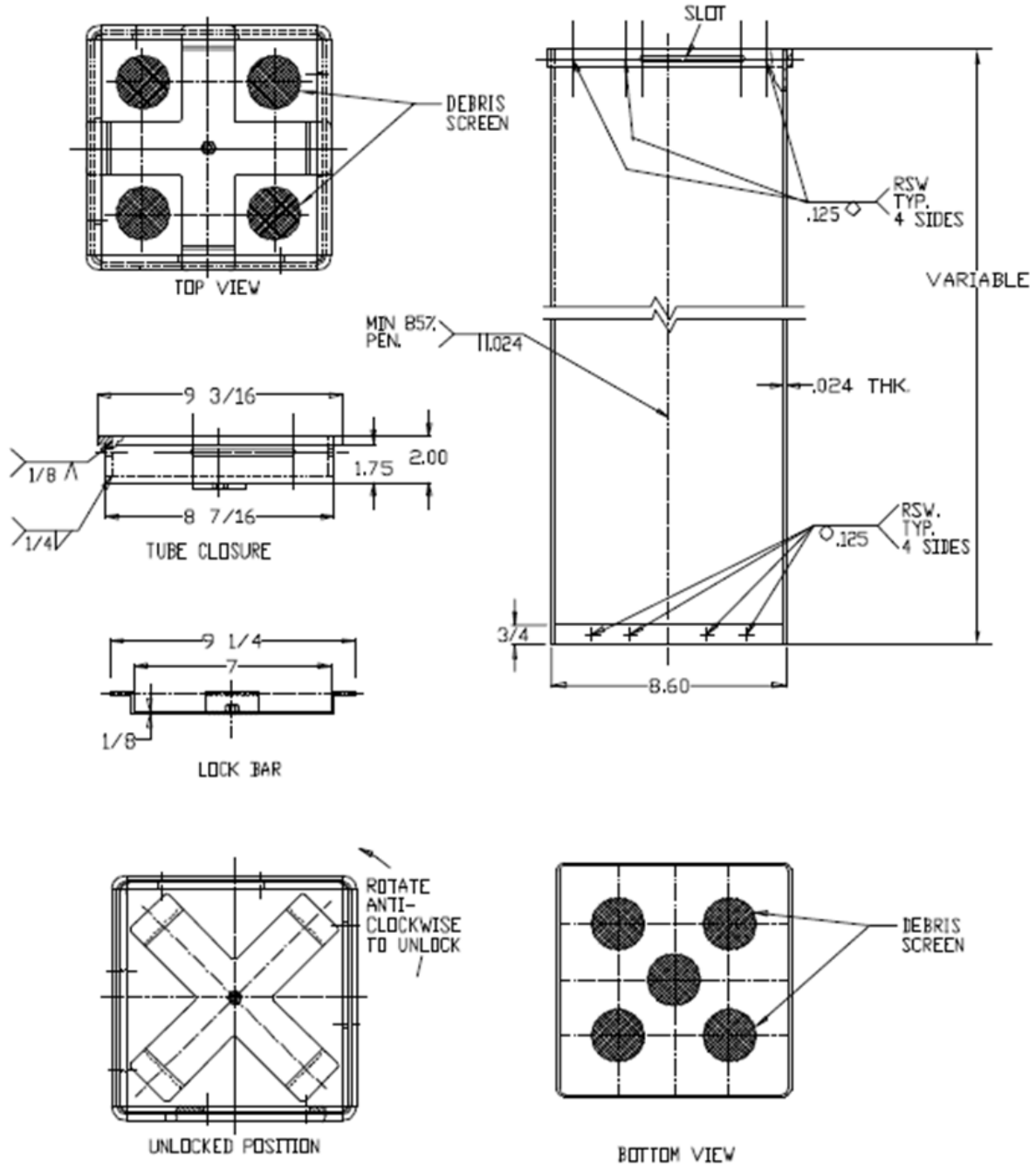


**FIGURE 2.1.2B: HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-24E/24EF**  
 (For Illustrative Purposes Only)

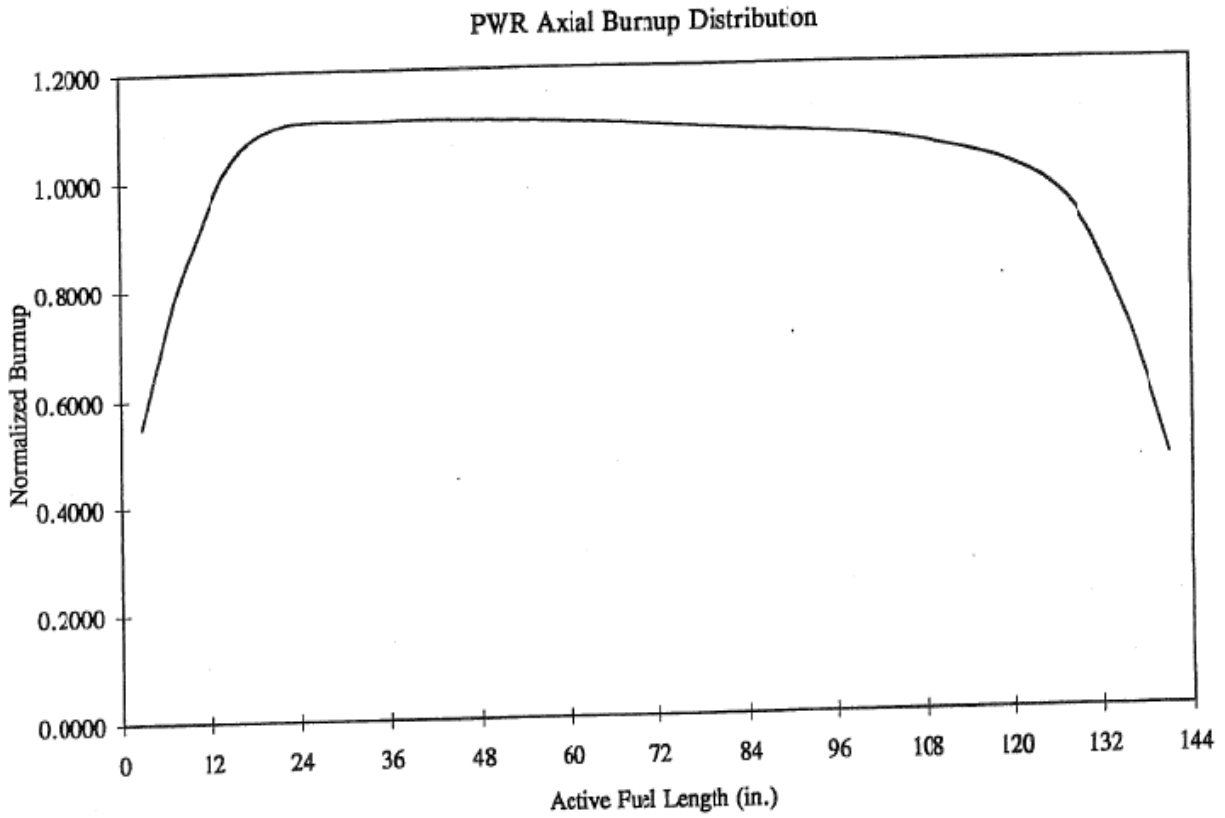


**FIGURE 2.1.2C: HOLTEC DAMAGED FUEL CONTAINER FOR BWR SNF IN MPC-68/68FF**  
 (For Illustrative Purposes Only)

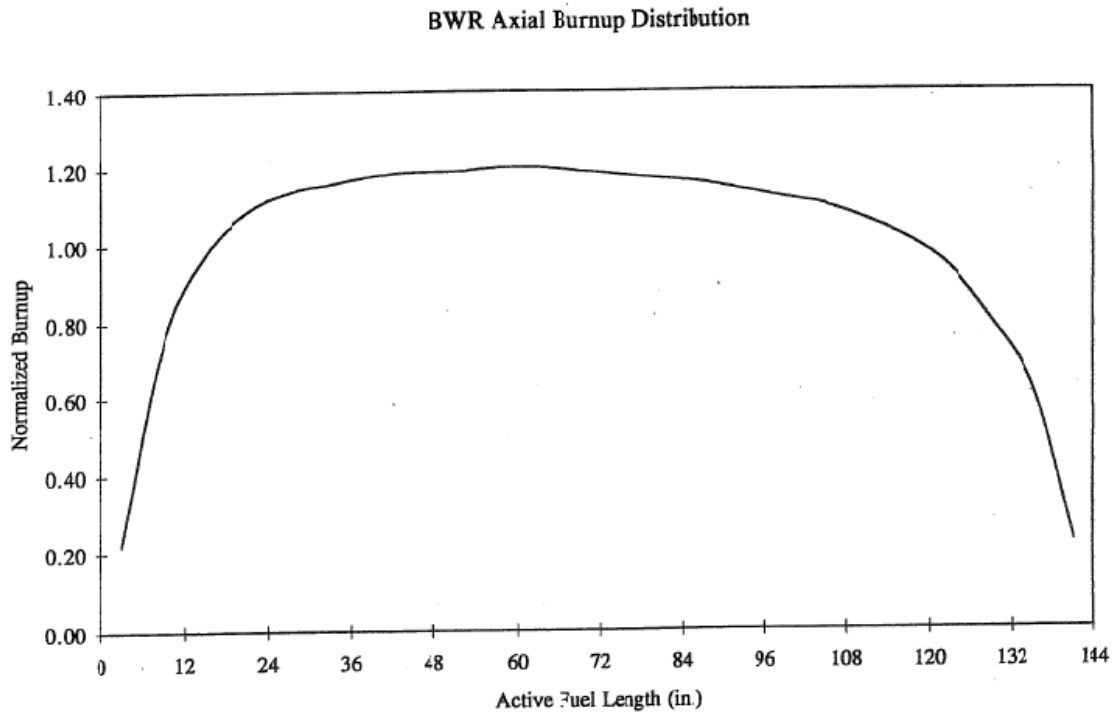
- NOTES:  
 1. ALL DIMENSIONS ARE IN INCHES AND ARE APPROXIMATE.  
 2. ALL MATERIAL IS STAINLESS STEEL.



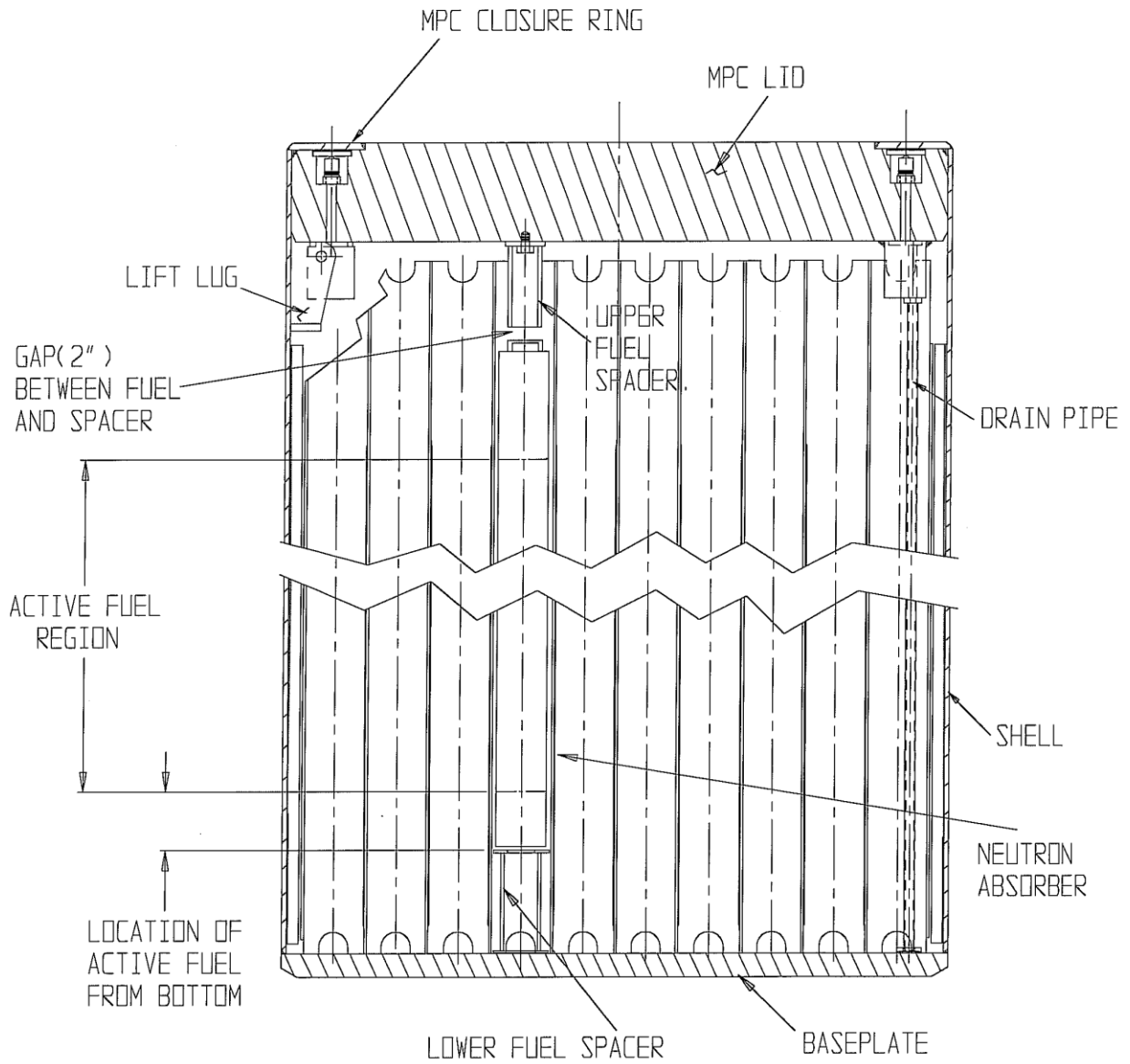
**FIGURE 2.1.2D: HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-32/32F**  
 (For Illustrative Purposes Only)



**FIGURE 2.1.3: PWR Axial Burnup Profile with Normalized Distribution**



**FIGURE 2.1.4: BWR Axial Burnup Profile with Normalized Distribution**



**FIGURE 2.1.5: MPC WITH UPPER AND LOWER FUEL SPACERS**

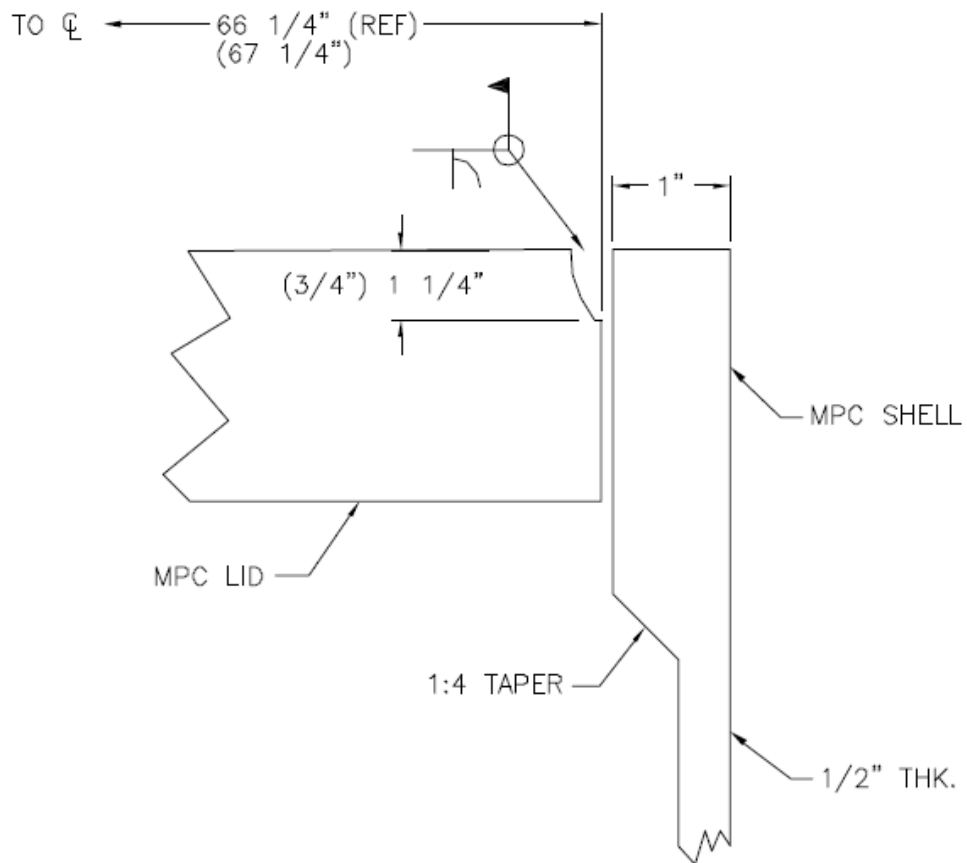
**FIGURE 2.1.6**

**INTENTIONALLY DELETED**

**FIGURE 2.1.7: DELETED**



**FIGURE 2.1.8: DELETED**



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

**FIGURE 2.1.9: FUEL DEBRIS MPC ("F" MODEL)**

## 2.2 HI-STORM 100 DESIGN CRITERIA

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

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

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

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

Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Partial Blockage of Air Inlets, Off-Normal Handling of HI-TRAC, Malfunction of Forced Helium Dehydrator System, Supplemental Cooling System Power Failure

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

Short-Term Operations: This loading condition is defined to accord with ISG-11, Revision 3 guidance [2.0.8]. This includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC transfer cask.

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

## 2.2.1 Normal Condition Design Criteria

### 2.2.1.1 Dead Weight

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

### 2.2.1.2 Handling

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

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

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

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

Special lifting devices shall meet the requirements of ANSI N14.6<sup>†</sup> [2.2.3].

### 2.2.1.3 Pressure

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

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

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

Table 2.2.1 provides the pressure limits for the HI-STORM 100 System.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container, it is conservatively assumed that 100% of the fuel rods are ruptured with 100% of the rod fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released for both normal and off-normal conditions. For PWR assemblies stored with non-fuel hardware, it is assumed that 100% of the gasses in the non-fuel hardware (e.g., BPRAs) is also released. This condition is bounded by the pressure calculation for design basis intact fuel with 100% of the fuel rods ruptured in all of the fuel assemblies. It is shown in Chapter 4 that the accident condition design pressure is not exceeded with 100% of the fuel rods ruptured in all of the design basis fuel assemblies. Therefore, rupture of 100% of the fuel rods in the damaged fuel assemblies or fuel debris will not cause the MPC internal pressure to exceed the accident design pressure.

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

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

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

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

#### 2.2.1.4 Environmental Temperatures

To evaluate the long-term effects of ambient temperatures on the HI-STORM 100 System, an upper bound value on the annual average ambient temperatures for the continental United States is used. The normal temperature specified in Table 2.2.2 is bounding for all reactor sites in the contiguous United States. The "normal" temperature set forth in Table 2.2.2 is intended to ensure that it is greater than the annual average of ambient temperatures at any location in the continental United States. In the northern region of the U.S., the design basis "normal" temperature used in this FSAR will be exceeded only for brief periods, whereas in the southern U.S, it may be straddled daily in summer months. Inasmuch as the sole effect of the "normal" temperature is on the computed fuel cladding temperature to establish long-term fuel integrity, it should not lie below the time averaged

yearly mean for the ISFSI site. Previously licensed cask systems have employed lower "normal" temperatures (viz. 75° F in Docket 72-1007) by utilizing national meteorological data.

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

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

#### 2.2.1.5 Design Temperatures

The ASME Boiler and Pressure Vessel Code (ASME Code) requires that the value of the vessel design temperature be established with appropriate consideration for the effect of heat generation internal or external to the vessel. The decay heat load from the spent nuclear fuel is the internal heat generation source for the HI-STORM 100 System. The ASME Code (Section III, Paragraph NCA-2142) requires the design temperature to be set at or above the maximum through thickness mean metal temperature of the pressure part under normal service (Level A) condition. Consistent with the terminology of NUREG-1536, we refer to this temperature as the "Design Temperature for Normal Conditions". Conservative calculations of the steady-state temperature field in the HI-STORM 100 System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STORM component temperatures at or below the normal condition design temperatures for the HI-STORM 100 System defined in Table 2.2.3. The temperature limits in Table 2.2.3 have been set down for the various parts in the MPC, HI-STORM and HI-TRAC to ensure that the subject part will not degrade if subject to the limiting temperature under long term storage. The normal condition temperature limit may be used as the reference temperature for defining the applicable part's allowable stress levels or for estimating the free expansion of the part if the actual computed service condition temperature is not available.

Except for the MPC Confinement Boundary, there are no Section III Subsection NB components in the HI-STORM system. Therefore, computation of secondary stresses (i.e., those that arise from thermal gradients at gross structural discontinuities) is not required for any component except the MPC pressure boundary. It should be noted that a HI-STORM MPC is uniquely protected against a sharp thermal gradient anywhere within its pressure boundary because of the continuously circulating helium gas which sweeps the canister's internal surfaces by the thermos-siphon action. As a result, the thermal stresses in the pressure boundary of the MPC are quite small. However, if required, the actual metal temperatures under the applicable service condition must be used to compute the Secondary stresses; the limiting temperatures listed in Table 2.2.3 can't be used.

Maintaining fuel rod cladding integrity is also a design consideration. The fuel rod peak cladding

temperature (PCT) limits for the long-term storage and short-term normal operating conditions meet the intent of the guidance in ISG-11, Revision 3 [2.0.8]. For moderate burnup fuel, the previously licensed PCT limit of 570°C (1058°F) may be used [2.0.9] (see also Section 4.5).

#### 2.2.1.6 Snow and Ice

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

#### 2.2.2 Off-Normal Conditions Design Criteria

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

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

##### 2.2.2.1 Pressure

The HI-STORM 100 System must withstand loads due to off-normal pressure. The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released in accordance with NUREG-1536.

##### 2.2.2.2 Environmental Temperatures

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

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

the definition of the off-normal temperature recognizes the considerable thermal inertia of the HI-STORM 100 storage system which reduces the effect of undulations in instantaneous temperature on the internals of the multi-purpose canister.

#### 2.2.2.3 Design Temperatures

In addition to the normal condition design temperatures, which apply to the long-term storage condition, we also define an "off-normal/accident condition temperature" pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61. This is, in effect, the temperature, which may exist during a transient event (an example of such an instance is the overpack blocked air duct off-normal event). The off-normal/accident design temperatures of Table 2.2.3 are set down to bound the maximax (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during the transient event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects during or immediately after, a transient event. Additionally, temperature limits are also defined for short-term normal operating conditions which include but are not limited to MPC drying operations and onsite transport operations.

The off-normal/accident design temperatures for stainless steel and carbon steel components are chosen such that the material's ultimate tensile strength does not fall below 30% of its room temperature value, based on data in published references [2.2.12 and 2.2.13]. This ensures that the material will not fail due to creep rupture during these short duration transient events. In addition, the off-normal and accident condition temperature limits in Table 2.2.3 have been set down for the various parts in the MPC, HI-STORM and HI-TRAC to ensure that the subject part will continue to render its intended function if subject to the limiting temperature under the applicable service condition. The limiting service condition temperature may be used as the reference temperature for defining the applicable part's allowable stress levels (for stress analysis purposes) or for estimating the free expansion of the part if the actual computed service condition temperature is not available.

#### 2.2.2.4 Leakage of One Seal

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

The confinement boundary is defined by the MPC shell, baseplate, MPC lid, port cover plates, closure ring, and associated welds. MPC shell welds and shell to baseplate weld are subject to helium leakage testing. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method as specified in Chapter 9 using proven inspection and testing techniques to verify the integrity of the confinement boundary.

#### 2.2.2.5 Partial Blockage of Air Inlets

The HI-STORM 100 System must withstand the partial blockage of the overpack air inlets. This



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

#### 2.2.2.6 Off-Normal HI-TRAC Handling

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

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

#### 2.2.2.7 Malfunction of Forced Helium Dehydrator (FHD)

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

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

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

#### 2.2.2.8 Supplemental Cooling System (SCS) Power Failure

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

### 2.2.3 Environmental Phenomena and Accident Condition Design Criteria

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

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

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

#### 2.2.3.1 Handling Accident

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

The loaded HI-STORM overpack will be lifted so that the bottom of the cask is at a height less than the vertical lift limit (see Table 2.2.8) above the ground. For conservatism, the postulated drop event assumes that the loaded HI-STORM 100 overpack falls freely from the vertical lift limit height before impacting a thick reinforced concrete pad. The deceleration of the cask must be maintained below 45 g's. Additionally, the overpack must continue to suitably shield the radiation emitted from the loaded MPC. The use of special lifting devices designed in accordance with ANSI N14.6 having redundant drop protection features to lift the loaded overpack will eliminate the lift height limit. The lift height limit is dependent on the characteristics of the impacting surface, which are specified in Table 2.2.9, and the number of fuel assemblies being stored (full or partially loaded MPC). For site-specific conditions including impact surfaces not encompassed by Table 2.2.9 or handling a partially loaded MPC, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the analyses in Appendix 3.A and shall be reviewed by the Certificate Holder.

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

Analysis is provided that demonstrates that the HI-TRAC continues to suitably shield the radiation

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

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

The loaded MPC is lowered into the HI-STORM or HI-STAR overpack or raised from the overpack using the HI-TRAC transfer cask and a MPC lifting system designed in accordance with ANSIN14.6 and having redundant drop protection features. Therefore, the possibility of a loaded MPC falling freely from its highest elevation during the MPC transfer operations into the HI-STORM or HI-STAR overpacks is not credible.

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

### 2.2.3.2 Tip-Over

The free-standing HI-STORM 100 System is demonstrated by analysis to remain kinematically stable under the design basis environmental phenomena (tornado, earthquake, etc.). However, the HI-

STORM 100 overpack and MPC shall also withstand impacts due to a hypothetical tip-over event. The structural integrity of a loaded HI-STORM 100 System after a tip-over onto a reinforced concrete pad is demonstrated by analysis. The cask tip-over is not postulated as an outcome of any environmental phenomenon or accident condition. The cask tip-over is a non-mechanistic event.

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

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

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

If the HI-STORM 100 System is deployed in an anchored configuration (HI-STORM 100A), then tip-over of the cask is structurally precluded along with the requirement of target compliance, which warrants setting specific limits on the concrete compressive strength and subgrade Young's Modulus. Rather, at the so-called high seismic sites (ZPAs greater than the limit set forth in the CoC for free standing casks), the ISFSI pad must be sufficiently rigid to hold the anchor studs and maintain the integrity of the fastening mechanism embedded in the pad during the postulated seismic event. The ISFSI pad must be designed to minimize a physical uplift during extreme environmental event (viz., tornado missile, DBE, etc.). The requirements on the ISFSI pad to render the cask anchoring function under long-term storage are provided in Section 2.0.4.

### 2.2.3.3 Fire

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

The HI-STORM 100 System must withstand temperatures due to a fire event. The HI-STORM overpack and HI-TRAC transfer cask fire accidents for storage are conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible transporter fuel. The HI-STORM overpack and HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Table 2.2.8 provides the fire durations for the HI-STORM overpack and HI-TRAC transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475° F in accordance with 10CFR71.73.

The accident condition design temperatures for the HI-STORM 100 System and the fuel rod cladding limits are specified in Table 2.2.3. The specified fuel cladding temperature limits are based on the temperature limits specified in ISG-11, Rev. 3 [2.0.9].

#### 2.2.3.4 Partial Blockage of MPC Basket Vent Holes

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

#### 2.2.3.5 Tornado

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

The kinematic stability of the HI-STORM overpack, and continued integrity of the MPC confinement boundary, while within the storage overpack or HI-TRAC transfer cask, must be demonstrated under impact from tornado-generated missiles in conjunction with the wind loadings. Standard Review Plan (SRP) 3.5.1.4 of NUREG-0800 [2.2.9] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile that deforms on impact (large missile); a rigid missile to test penetration resistance (penetrant missile); and a small rigid missile of

a size sufficient to pass through any openings in the protective barriers (micro-missile). SRP 3.5.1.4 suggests an automobile for a large missile, a rigid solid steel cylinder for the penetrant missile, and a solid sphere for the small rigid missile, all impacting at 35% of the maximum horizontal wind speed of the design basis tornado. Table 2.2.5 provides the missile data used in the analysis, which is based on the above SRP guidelines. The effects of a large tornado missile are considered to bound the effects of a light general aviation airplane crashing on an ISFSI facility.

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

#### 2.2.3.6 Flood

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

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

The flood water is assumed to be nonstagnant. The maximum allowable flood water velocity is determined by calculating the equivalent pressure loading required to slide or tip over the HI-STORM 100 System. The design basis flood water velocity is stated in Table 2.2.8. Site-specific safety reviews by the licensee must confirm that flood parameters do not exceed the flood depth, slide, or tip-over forces.

If the flood water depth exceeds the elevation of the top of the HI-STORM overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the overpack. Blockage of the air inlets is addressed in Subsection 2.2.3.13.

Most reactor sites are hydrologically characterized as required by Paragraph 100.10(c) of 10CFR100 and further articulated in Reg. Guide 1.59, "Design Basis Floods for Nuclear Power Plants" and Reg. Guide 1.102, "Flood Protection for Nuclear Power Plants." It is assumed that a complete characterization of the ISFSI's hydrosphere including the effects of hurricanes, floods, seiches and tsunamis is available to enable a site-specific evaluation of the HI-STORM 100 System for kinematic

stability. An evaluation for tsunamis<sup>†</sup> for certain coastal sites should also be performed to demonstrate that sliding or tip-over will not occur and that the maximum flood depth will not be exceeded.

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

#### 2.2.3.7 Seismic Design Loadings

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

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

#### 2.2.3.8 100% Fuel Rod Rupture

The HI-STORM 100 System must withstand loads due to 100% fuel rod rupture. For conservatism, 100 percent of the fuel rods are assumed to rupture with 100 percent of the fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released in accordance with NUREG-1536. All of the fill gas contained in non-fuel hardware, such as Burnable Poison Rod Assemblies (BPRAs) is also assumed to be released in analyzing this event.

#### 2.2.3.9 Confinement Boundary Leakage

No credible scenario has been identified that would cause failure of the confinement system. Section 7.1 provides a discussion as to why leakage of any magnitude from the MPC is not credible, based on the materials and methods of fabrication and inspection.

#### 2.2.3.10 Explosion

The HI-STORM 100 System must withstand loads due to an explosion. The accident condition MPC

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

external pressure and overpack pressure differential specified in Table 2.2.1 bounds all credible external explosion events. There are no credible internal explosive events since all materials are compatible with the various operating environments, as discussed in Section 3.4.1, or appropriate preventive measures are taken to preclude internal explosive events (see Section 1.2.1.3.1.1). The MPC is composed of stainless steel, neutron absorber material, and prior to CoC Amendment 2, possibly optional aluminum alloy 1100 heat conduction elements. For these materials, and considering the protective measures taken during loading and unloading operations there is no credible internal explosive event.

#### 2.2.3.11 Lightning

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

#### 2.2.3.12 Burial Under Debris

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

#### 2.2.3.13 100% Blockage of Air Inlets

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

#### 2.2.3.14 Extreme Environmental Temperature

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

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

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



environmental hazards.

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

#### 2.2.3.16 Supplemental Cooling System (SCS) Failure

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

Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated. Possible causes of SCS failure are: (a) Simultaneous loss of external and backup power, or (b) Complete loss of annulus water from an uncontrolled leak or line break.

#### 2.2.4 Applicability of Governing Documents

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition, with Addenda through 1997 [2.2.1], is the governing code for the structural design of the MPC, the metal structure of the HI-STORM 100 overpack, and the HI-TRAC transfer cask, except for Sections V and IX. The latest effective editions of ASME Section V and IX may be used, provided a written reconciliation of the later edition against the 1995 Edition, including addenda, is performed by the certificate holder. The MPC enclosure vessel and fuel basket are designed in accordance with Section III, Subsections NB Class 1 and NG Class 1, respectively. The metal structure of the overpack and the HI-TRAC transfer cask are designed in accordance with Section III, Subsection NF Class 3. The ASME Code is applied to each component consistent with the function of the component.

ACI 349 is the governing code for the plain concrete in the HI-STORM 100 overpack. ACI 318.1-85(92) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited during structural analysis. Appendix 1.D provides the sections of ACI 349 and ACI 318.1-85(92) applicable to the plain concrete.

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

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

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

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

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

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

### 2.2.5 Service Limits

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

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

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

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

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

#### 2.2.6 Loads

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

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

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

pressure vessel. The specific load combination for each component is specified in Subsection 2.2.7.

### 2.2.7 Load Combinations

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

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

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

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

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

### 2.2.8 Allowable Stresses

The stress intensity limits for the MPC confinement boundary for the design condition and the service conditions are provided in Table 2.2.10. The MPC confinement boundary stress intensity limits are obtained from ASME Code, Section III, Subsection NB. The stress intensity limits for the MPC fuel basket are presented in Table 2.2.11 (governed by Subsection NG of Section III). The steel

structure of the overpack and the HI-TRAC meet the stress limits of Subsection NF of ASME Code, Section III for plate and shell components. Limits for the Level D condition are obtained from Appendix F of ASME Code, Section III for the steel structure of the overpack. The ASME Code is not applicable to the HI-TRAC transfer cask for accident conditions, service level D conditions. The HI-TRAC transfer cask has been shown by analysis to not deform sufficiently to apply a load to the MPC, have any shell rupture, or have the top lid, pool lid, or transfer lid (as applicable) detach.

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

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

$S_y$ : Minimum yield strength at temperature

$S_u$ : Minimum ultimate strength at temperature

## 2.2.9 Requirements on Lifting and Special Lifting Devices

2.2.9.1 Definitions: The lifting and handling systems used in Holtec's used fuel management program are made up of individual components or devices. These components can be further classified as either "*lifting devices*" or "*special lifting devices*." (See Glossary)

The term *special lifting device* refers to components to which ANSI N14.6 applies. As stated in ANSI N14.6 (both 1978 and 1993 versions), "This standard shall apply to *special lifting devices* that transmit the load from lifting attachments, which are structural parts of a container to the hook(s) of an overhead hoisting system." Examples of special lifting devices used with Holtec's systems include MPC lift cleats, lift brackets, and lift yokes

The term *lifting device* refers to components of a lifting and handling system that are not classified as *special lifting devices*. ANSI N14.6 is not applicable to these *lifting devices*. These include non-active structural components (components that bear the primary load but are not a constituent of a moving part, e.g., gear train, hydraulic cylinder) of the system. Examples of lifting devices used with Holtec's systems include: a vertical cask transporter's overhead beam, the structural members (viz. the main girder) of a gantry crane or a cask crane used to handle the MPC inside a part 50 structure.

The design of all lifting devices is governed by a Purchasing Specification prepared under the system designer's QA program which shall contain appropriate interpretations of the applicable codes and standards, required material properties, extreme environmental loadings (viz., earthquakes) and the like. The qualification for seismic and other applicable environmental loads is not required for transient states such as when the load is being emplaced and fastened or the lifting device is in motion.

### 2.2.9.2 Stress compliance criteria applicable to *Lifting Devices and Special Lifting Devices*:

The stress compliance criteria for *lifting devices* are taken from the code applicable to the specific component defined in the system designer's Purchasing Specification. For example, slings are required to meet the guidelines of ANSI B30.9 and overhead beams are required to meet the guidelines of an applicable consensus national standard selected by the designer, such as AISC, CMAA, or ASME Code (Subsection NF). Where a suitable consensus standard does not exist, the system designer is required to specify the necessary stress and strength requirements appropriate to the hardware.

The stress compliance criteria for *special lifting devices* are taken directly from ANSIN14.6, which requires safety factors of three against the yield strength and five times against ultimate strength under the dead load to be lifted.

### 2.2.9.3 Single Failure Proof Criteria

In order for a *lifting device* or *special lifting device* to be considered *single failure proof*, the design must also follow the guidance in NUREG-0612, which requires that a single failure proof device have twice the normal safety margin. This designation can be achieved by either providing redundant devices or providing twice the design safety factor as required by the applicable code. Therefore, for a *lifting device* to be considered single failure proof, the applicable code requirements should be doubled, or a redundant *lifting device* should be provided. The load drop protection feature incorporated in the vertical towers of a cask transporter is an example of redundant lifting part.

The horizontal transporters, referred to as Hauling transporter, Low Profile transporter, etc., are characterized by the absence of a lifting feature. Such ground supported equipment is considered single failure proof if the stresses developed under the design basis dead load are <50% of the allowable limit set down in the system designer's specification.

Likewise, for cask handling purposes, a plant's main crane can be treated as single failure proof if the structural factors of safety against the applicable code limit are a minimum of 2.

Similarly for a *special lifting device* to be considered *single failure proof*, the design safety factors in ANSIN14.6 should be doubled, or a redundant *special lifting device* should be provided.

Alternatively, the designer may perform a load drop analysis (permitted by both NUREG-0612 and ANSIN14.6). If the analyses support the conclusion that, after the physically admissible drop accident, the permissible dose rate from the cask does not exceed the plant's accident condition dose limit and the MPC meets the sub-criticality criterion of §72.124 then the increased safety factors are not required. In addition, for a drop scenario involving a loaded MPC, the confinement integrity of the MPC must remain intact and the MPC must remain retrievable subsequent to the drop event.

Table 2.2.1

## PRESSURE LIMITS

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

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

Table 2.2.2

## ENVIRONMENTAL TEMPERATURES

Condition	Temperature (°F)	Comments
HI-STORM 100 Overpack		
Normal Ambient (Bounding Annual Average)	80	
Normal Soil Temperature (Bounding Annual Average)	77	
Off-Normal Ambient (3-Day Average)	-40 and 100	-40°F with no insolation  100°F with insolation
Extreme Accident Level Ambient (3-Day Average)	125	125°F with insolation starting at steady-state off-normal high environment temperature
HI-TRAC Transfer Cask		
Inside Building Short- Term Operations (3- Day Average)	110	110°F with no insolation
Outside Building Short- Term Operations (3- Day Average)	90	90°F with insolation

Note:

- 2) Handling operations with the loaded HI-STORM overpack and HI-TRAC transfer cask are limited to working area ambient temperatures greater than or equal to 0°F as specified in Subsection 2.2.1.2.



Sub-component or Part**	Long Term, Normal Condition Temperature Limits (Long-Term Events) (° F)	Short-Term Events <sup>††</sup> Temperature Limits (° F)	Off-Normal and Accident Condition Temperature Limits <sup>†</sup> (° F)	30-Day Accident Condition Temperature Limit (° F) <sup>†††</sup>
MPC shell	600‡	775‡	775	572
MPC basket	752	1058	1058	752
MPC Neutron Absorber	752	1058	1058	752
MPC lid	600‡	775‡	775	572
MPC closure ring	500‡	775‡	775	572
MPC baseplate	400‡	775‡	775	572
HI-TRAC inner shell	-	500	800	-
HI-TRAC pool lid/transfer lid	-	350	800	-
HI-TRAC top lid	-	400	800	-
HI-TRAC top flange	-	400	700	-
HI-TRAC pool lid seals	-	350	N/A	-
HI-TRAC bottom lid bolts	-	350	800	-

\*\* Wherever applicable, the limiting temperatures under all service conditions in this table for each sub-component or part have been aligned with the corresponding values in Table 2.2.3 of the HI-STORM FW FSAR [2.2.17].

††† 30-day accident event is defined as 100% blocked vent condition at threshold heat loads defined in Section 4.6.

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

† For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the ISFSI fire event, the local temperature limit of HI-STORM concrete is 1100°F (Appendix 1.D) and the overpack steel structure is required to remain physically stable (i.e., so there will be no risk of structural instability such as gross buckling the maximum temperature shall be less than 50% of the component's melting temperature and the specific temperature limits in Table 2.2.3 for the overpack steel structure do not apply.) Concrete that exceeds 1100°F shall be considered unavailable for shielding of the overpack.

‡ Temperature limits in Table 1.A.6 shall take precedence if duplex stainless steels are used for the fabrication of confinement boundary components as described in Appendix 1.A

<b>Sub-component or Part**</b>	<b>Long Term, Normal Condition Temperature Limits (Long-Term Events) (° F)</b>	<b>Short-Term Events<sup>††</sup> Temperature Limits (° F)</b>	<b>Off-Normal and Accident Condition Temperature Limits<sup>†</sup> (° F)</b>	<b>30-Day Accident Condition Temperature Limit (° F)<sup>†††</sup></b>
HI-TRAC bottom flange	-	350	800	-
HI-TRAC top lid neutron shielding	-	300	350	-
HI-TRAC radial neutron shield	-	307	N/A	-
HI-TRAC radial lead gamma shield	-	350	600	-
Remainder of HI-TRAC	-	350	800	-
Fuel Cladding	752	752 or 1058 <sup>††</sup>	1058	752
Overpack concrete	300 <sup>†</sup>	572 (on local temperature of shielding concrete)	572 (on local temperature of shielding concrete except under fire <sup>†</sup> )	450 (on local temperature of shielding concrete)
Overpack Lid Top and Bottom Plate	450	700	800	450
Remainder of overpack steel structure	400	700	800	450

General note: All short-term, off-normal and accident condition structural evaluations are based on bounding temperatures from thermal evaluations presented in Chapter 4. If the actual computed service condition temperature is not available for a particular sub-component or part, the normal condition temperature limit can be used conservatively in the design basis structural evaluations for MPC and HI-STORM. Similarly, the short-term condition temperature limits can be used for HI-TRAC. The thermal evaluations presented in Chapter 4 must comply with the above temperature limits as well as the temperatures used to inform the structural design basis calculations, which in some cases use actual computed temperatures as opposed to the above limit values. Changes to these computed temperatures should be evaluated under 10CFR72.48 as necessary.

Table 2.2.4

## TORNADO CHARACTERISTICS

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

Table 2.2.5

## TORNADO-GENERATED MISSILES

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

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Confinement	Shell	A	ASME Section III; Subsection NB	Alloy X <sup>(5)</sup>	See Appendix 1.A	NA	NA
Confinement	Baseplate	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Lid (One-piece design and top portion of optional two-piece design)	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Closure Ring	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Port Cover Plates	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Basket Cell Plates	A	ASME Section III; Subsection NG core support structures (NG-1121)	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Neutron Absorber	A	Non-code	NA	NA	NA	Aluminum/SS
Shielding	Drain and Vent Shield Block	C	Non-code	Alloy X	See Appendix 1.A	NA	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Shielding	Plugs for Drilled Holes	NITS	Non-code	SA 193B8 (or equivalent)	See Appendix 1.A	NA	NA
Shielding	Bottom portion of optional two-piece MPC lid design	B	Non-code	Alloy X or Carbon Steel	See Appendix 1.A for Alloy X, Table 3.3.6 for Carbon Steel	Stainless Steel coating when using Carbon Steel	Stainless Steel when using Carbon Steel
Structural Integrity	Upper Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Sheathing	A	Non-code	Alloy X	See Appendix 1.A	Aluminum/SS	NA
Structural Integrity	Shims	NITS	Non-code (shims, welded directly to angle or parallel plate basket supports, are ASME Section II)	Alloy X	See Appendix 1.A	NA	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Structural Integrity	Basket Supports (Angled Plate or Parallel Plates with connecting end shim) (If Basket Shims are not used)	A	ASME Section III; Subsection NG internal structures (NG-1122)	Alloy X	See Appendix 1.A	NA	NA
Structural Form	Basket Supports (Flat Plates)	NITS	Non-Code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (If Basket Shims are used) (for MPC-32 only)	NITS	ASME Section III; Subsection NG internal structures (NG-1122)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Shims (for MPC-32 only)	A	ASME Section III; Subsection NG internal structures (NG-1122)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug	C	NUREG-0612	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug Baseplate	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Bolt	NITS	Non-code	A193-B8 (or equiv.)	Per ASME Section II	NA	NA
Structural Integrity	Upper Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Structural Integrity	Lower Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Stainless Steel. See Note 6	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Vent Shield Block Spacer	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Vent and Drain Tube	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	NA
Operations	Vent & Drain Cap	C	Non-code	S/S	Per ASME Section II	NA	NA
Operations	Vent & Drain Cap Seal Washer	NITS	Non-code	Aluminum	NA	NA	Aluminum/SS
Operations	Vent & Drain Cap Seal Washer Bolt	NITS	Non-code	Aluminum	NA	NA	NA
Operations	Reducer	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Drain Line	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Damaged Fuel Container	C	ASME Section III;	S/S	See Appendix 1.A	NA	NA



Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
			Subsection NG	(Primarily 304 S/S)			
Operations	Drain Line Guide Tube	NITS	Non-code	S/S	NA	NA	NA
Operations	Vent and Drain Tube, Optional	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	N/A
Operations	Threaded Disc, Plug Adjustment	C	Non-code	S/S	Per ASME Section II	N/A	N/A
Operations	Vent and Drain Plug	C	Non-code	Aluminum	N/A	N/A	N/A
Operations	Thread Shield Cap	NITS	Non-code	Aluminum	N/A	N/A	N/A
Operations	Retaining Ring	NITS	Non-code	S/S	N/A	N/A	N/A

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Shielding	Radial Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Block Ring (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Shielding	Lid Shield Ring (100S and 100S Version B) and Shield Block Shell (100S)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70 (SA515-70 not permitted for 100S Version B)	See Table 3.3.2	See Note 5	NA
Shielding	Shield Block Shell (100)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Shielding	Pedestal Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Lid Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Shell (eliminated from design 6/01)	B	See Note 6	SA516-70	See Table 3.3.2	NA	NA
Shielding	Shield Block	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Gamma Shield Cross	C	Non-code	SA240-304	NA	NA	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
	Plates & Tabs						
Structural Integrity	Baseplate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.3	See Note 5	NA
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Concrete Form	Pedestal Shell	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Concrete Form	Pedestal Plate (100) Pedestal Baseplate (100S)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Table 3.3.2	NA
Structural Integrity	Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inlet Vent Vertical & Horizontal Plates	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Thermal	Exit Vent Horizontal Plate (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Thermal	Exit Vent Vertical/Side Plate	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Thermal	Heat Shield	B	N/A	C/S	N/A	See Note 5	N/A
Thermal	Heat Shield Ring	B	N/A	C/S	N/A	See Note 5	N/A
Structural Integrity	Top Plate, including shear ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Top (Cover) Plate, including shear ring (100 and 100S)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Stud & Nut	B	ASME Section II	SA564-630 or SA 193-B7 (stud) SA 194-2H (nut)	See Table 3.3.4 (stud) Per ASME Section II (nut)	Threads to have cadmium coating (or similar lubricant for corrosion protection)	NA
Structural Integrity	100S Lid Washer	B	Non-Code	SA240-304	Per ASME	NA	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
					Section II		
Structural Integrity	Bolt Anchor Block	B	ASME Section III; Subsection NF Regulatory Guide 3.61	SA350-LF3, SA350-LF2, or SA203E	See Table 3.3.3	See Note 5	NA
Structural Integrity	Channel	B	ASME Section III; Subsection NF	SA516-70 (galvanized) or SA240-304	See Table 3.3.2 or Table 3.3.1	See Note 5	NA
Structural Integrity	Channel Mounts	B	ASME Section III; Subsection NF	SA36 or equivalent	Per ASME Section II	See Note 5	NA
Structural Integrity	Cylindrical Guide Tubes	B	ASME Section III; Subsection NF	Type 304 or 316 S/S	Per ASME Section II	NA	NA
Shielding	Pedestal Platform	B	Non-Code	SA36 or equivalent	NA	See Note 5	NA
Operations	Storage Marking Nameplate	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Exit Vent Screen Sheet	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Drain Pipe	NITS	Non-code	C/S or S/S	NA	See Note 5	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Operations	Exit & Inlet Screen Frame	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Temperature Element & Associated Temperature Monitoring Equipment	C	Non-code	NA	NA	NA	NA
Operations	Screen	NITS	Non-code	Mesh Wire	NA	NA	NA
Operations	Paint	NITS	Non-code	Per Appendix 1.C	NA	NA	NA
Structural Integrity	100S Version B Base Bottom Plate	B	ASME III; Subsection NF	Carbon Steel	See Table 3.3.6	See Note 5	NA
Structural Integrity	100S Version B Base Spacer Block	B	Non-code	Carbon Steel	See Table 3.3.6	See Note 5	NA
Shielding	100S Version B Base Shield Block	B	Non-code	Carbon Steel	NA	See Note 5	NA
Structural Integrity	100S Version B Base Top Plate	B	ASME III; Subsection NF	SA 516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	100S Version B Base MPC Support	B	Non-code	Carbon Steel	See Table 3.3.6	See Note 5	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Shielding	100S Version B Lid Outer Ring	B	ASME III; Subsection NF	SA516-70 or SA36	See Table 3.3.2 or Table 3.3.6	See Note 5	NA
Operations	100S Version B Lid Vent Duct	NITS	Non-code	Carbon Steel	NA	See Note 5	NA
Structural Integrity	100S Version B Lid Inner Ring	B	ASME III; Subsection NF	Carbon Steel	See Table 3.3.6	See Note 5	NA
Operations	100S Version B Lid Stud Pipe	NITS	Non-code	Carbon Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Spacer	NITS	Non-code	Carbon Steel	NA	See Note 5	NA
Operations	100S Version B Lid Lift Block	B	ASME III; Subsection NF	SA36	See Table 3.3.6	See Note 5	NA
Shielding	100S Version B Lid Vent Shield	B	Non-code	Carbon Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Washer	C	Non-code	Stainless Steel	NA	See Note 5	NA
Operations	100S Version B Lid Stud Cap	NITS	Non-code	PVC	NA	See Note 5	NA

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	100S Version B Radial Gusset	B	ASME III; Subsection NF	SA 516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	100S Version B Lid Closure Bolt and Closure Bolt Handle	B (bolt)  NITS (bolt Handle)	ASME Section II	SA 193-B7 (bolt)  C/S (bolt handle)	See Table 3.3.4 (bolt)  NA (bolt handle)	Threads to have cadmium coating (or similar lubricant for corrosion protection)	NA
Structural Integrity	100S Version B Lid Top (Cover) Plate	B	ASME Section III; Subsection NF	SA516-70 or SA36	See Table 3.3.2 or Table 3.3.6	See Note 5	NA
Structural Integrity	100S Version B Shear Ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA



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

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

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Pool Lid Bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA
Structural Integrity	Lifting Trunnion Block	B	ASME Section III; Subsection NF	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lifting Trunnion	A	NUREG-0612 Regulatory Guide 3.61	SB637 (N07718) or SA564-630H1100 (For HI-TRAC125D only)	See Table 3.3.4	NA	NA
Structural Integrity	Pocket Trunnion (HI-TRAC 100 and HI-TRAC 125 only)	B	ASME Section III; Subsection NF NUREG-0612 Regulatory Guide 3.61	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Dowel Pins	B	ASME Section III; Subsection NF	SA564-630	See Table 3.3.4	NA	SA350-LF3
Structural Integrity	Water Jacket End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA

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

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

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

Table 2.2.7

## HI-STORM 100 ASME BOILER AND PRESSURE VESSEL CODE APPLICABILITY

<b>HI-STORM 100 Component</b>	<b>Material Procurement</b>	<b>Design</b>	<b>Fabrication</b>	<b>Inspection**</b>
Overpack steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3200	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5350, NF-5360 and Section V
Anchor Studs for HI-STORM 100A	Section II, Section III, Subsection NF, NF-2000*	Section III, Subsection NF, NF- 3300	NA	NA
MPC confinement boundary	Section II, Section III, Subsection NB, NB-2000	Section III, Subsection NB, NB-3200	Section III, Subsection NB, NB-4000	Section III, Subsection NB, NB-5000 and Section V
MPC fuel basket	Section II, Section III, Subsection NG, NG-2000; core support structures (NG-1121)	Section III, Subsection NG, NG-3300 and NG-3200; core support structures (NG-1121)	Section III, Subsection NG, NG-4000; core support structures (NG-1121)	Section III, Subsection NG, NG-5000 and Section V; core support structures (NG-1121)
HI-TRAC Trunnions	Section II, Section III, Subsection NF, NF-2000	NUREG-0612 and Regulatory Guide 3.61	Section III, Subsection NF, NF-4000	See Chapter 9
MPC basket supports (Angled Plates)	Section II, Section III, Subsection NG, NG-2000; internal structures (NG-1122)	Section III, Subsection NG, NG-3300 and NG-3200; internal structures (NG-1122)	Section III, Subsection NG, NG-4000; internal structures (NG-1122)	Section III, Subsection NG, NG-5000 and Section V; internal structures (NG-1122)
Basket Shims (for MPC-32 only)	Section II, Section III, Subsection NG, NG-2000; internal structures (NG-1122)	Section III, Subsection NG, NG-3300 and NG-3200; internal structures (NG-1122)	Section III, Subsection NG, NG-4000; internal structures (NG-1122)	Section III, Subsection NG, NG-5000 and Section V; internal structures (NG-1122)
HI-TRAC steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3300	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5360 and Section V

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

\*\* Section V applies to Code welds only unless otherwise noted.

<b>HI-STORM 100 Component</b>	<b>Material Procurement</b>	<b>Design</b>	<b>Fabrication</b>	<b>Inspection**</b>
Damaged fuel container	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V
Overpack concrete	ACI 349 as specified by Appendix 1.D	ACI 349 and ACI 318.1-89(92) as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D



Table 2.2.8

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

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

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

††† For ISFSI and subgrade design parameter Sets A and B and fully loaded MPC. Users may also develop a site-specific lift height limit.

Table 2.2.9

## EXAMPLES OF ACCEPTABLE ISFSI PAD DESIGN PARAMETERS

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

NOTE: A static coefficient of friction of 0.53 between the ISFSI pad and the bottom of the overpack shall be used. If for a specific ISFSI a higher value of the coefficient of friction is used, it shall be verified by test. The test procedure shall follow the guidelines included in the Sliding Analysis in Subsection 3.4.7.1.

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

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

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

STRESS CATEGORY	DESIGN	LEVELS A & B <sup>†††</sup>	LEVEL D <sup>††</sup>
Primary Membrane, $P_m$	$S_m$	$S_m$	AMIN ( $2.4S_m$ , $0.7S_u$ )
Local Membrane, $P_L$	$1.5S_m$	$1.5S_m$	150% of $P_m$ Limit
Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of $P_m$ Limit
Primary Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of $P_m$ Limit
Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
Average Shear Stress <sup>††††</sup>	$0.6S_m$	$0.6S_m$	$0.42S_u$

<sup>†</sup> Stress combinations including F (peak stress) apply to fatigue evaluations only.

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

<sup>†††</sup> For stress intensities generated by Level B Service Loadings, allowable stress intensity values are 110% of Level A limits per NB-3223.

<sup>††††</sup> Governed by NB-3227.2 or F-1331.1(d).

Table 2.2.11

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

<b>STRESS CATEGORY</b>	<b>DESIGN</b>	<b>LEVELS A &amp; B</b>	<b>LEVEL D<sup>†</sup></b>
Primary Membrane, $P_m$	$S_m$	$S_m$	AMIN ( $2.4S_m$ , $0.7S_u$ ) <sup>††</sup>
Primary Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of $P_m$ Limit
Primary Membrane plus Primary Bending plus Secondary	N/A <sup>†††</sup>	$3S_m$	N/A

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

<sup>††</sup> Governed by F-1331.1(a).

<sup>†††</sup> No specific stress intensity limit applicable.

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

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

Definitions:

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

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

$S_u$  = Ultimate Strength

<sup>†</sup> Governed by Appendix F, Paragraph F-1332 of the ASME Code, Section III.

Table 2.2.13

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

NORMAL CONDITION	
LOADING	NOTATION
Dead Weight	D
Handling Loads	H
Design Pressure (Internal)	$P_i$
Short-Term Normal Pressure (Internal)	$P_{st}$
Design Pressure (External) <sup>†</sup>	$P_o$
Snow	S
Operating Temperature	T
OFF-NORMAL CONDITION	
Loading	Notation
Off-Normal Pressure (Internal)	$P_i'$
Off-Normal Pressure (External) <sup>†</sup>	$P_o$
Off-Normal Temperature	$T'$
Off-Normal HI-TRAC Handling	$H'$

ACCIDENT CONDITIONS	
LOADING	NOTATION
Handling Accident	H'
Earthquake	E
Fire	T*
Tornado Missile	M
Tornado Wind	W'
Flood	F
Explosion	E*
Accident Pressure (Internal)	P <sub>i</sub> *
Accident Pressure (External)	P <sub>o</sub> *

Table 2.2.14  
 APPLICABLE LOAD CASES AND COMBINATIONS FOR EACH CONDITION AND COMPONENT<sup>†, ††</sup>

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

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

<sup>††</sup> N/A stands for "Not Applicable."

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

<sup>††††</sup>  $P_o^*$  bounds the external pressure due to explosion.

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



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

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

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

<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Alternative, Justification &amp; Compensatory Measures</b>
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC	NB-2121	Provides permitted material specification for pressure-retaining material, which must conform to Section II, Part D, Tables 2A and 2B	Certain duplex stainless steels are not included in Section II, Part D, Tables 2A and 2B. UNS S31803 duplex stainless steel alloy is evaluated in the HI-STORM 100 FSAR and meet the required design criteria for use in the HI-STORM 100 system per ASME Code Case N-635-1. Appendix 1.A provides the required property data for the necessary safety analysis
MPC, MPC basket assembly, HI-STORM overpack, and HI-TRAC transfer cask	NB-3100 NG-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.

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

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

<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Alternative, Justification &amp; Compensatory Measures</b>
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT examination alone is used, at a minimum, it will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be pressure tested as defined in Chapter 9. Accessibility for leakage inspections precludes a Code compliant pressure test. Since the shell welds of the MPC cannot be checked for leakage during this pressure test, the shop leakage test to $10^{-7}$ ref cc/sec (as described in Chapter 9) provides reasonable assurance as to its leak tightness. All MPC vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			<p>cause the primary stress limits of NB-3000 to be exceeded.</p> <p>The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.</p>
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NG-2000 requirements.

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	NG-4420	NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	<p>Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis.</p> <p>From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators (“peak stress”, F, in the ASME Code for which specific stress intensity limits do not apply).</p>



<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Alternative, Justification &amp; Compensatory Measures</b>
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package is to be in conformance with Holtec's QA program.
Overpack Steel Structure	NF-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NF-2000 requirements.
HI-TRAC Steel Structure	NF-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NF-2000 requirements.
Overpack Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member thickness of 1" or greater is loaded to transmit loads in the through thickness direction.	The margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have large safety factors during lifting. Likewise, the top lid plate to lid shell weld has a large structural margin under the inertia loads imposed during a non-mechanistic tipover event.

<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Alternative, Justification &amp; Compensatory Measures</b>
Overpack Steel Structure	NF-3256 NF-3266	Provides requirements for welded joints.	<p>Welds for which no structural credit is taken are identified as “Non-NF” welds in the design drawings by an “*”. These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX.</p> <p>Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.</p>

<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Alternative, Justification &amp; Compensatory Measures</b>
HI-STORM Overpack and HI-TRAC Transfer Cask	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting	<p>These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The overpack and transfer cask bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).</p> <p>Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.</p>

Table 2.2.16

COMPARISON BETWEEN HI-STORM MPC LOADINGS WITH HI-STAR MPC LOADINGS<sup>†</sup>

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

<sup>†</sup> HI-STAR MPC loadings are those specified in the HI-STAR SAR under docket number 71-9261, which does not impose any off-normal condition loadings.

## 2.3 SAFETY PROTECTION SYSTEMS

### 2.3.1 General

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

1. The MPC confinement barrier is an enclosure vessel designed in accordance with the ASME Code, Subsection NB with confinement welds inspected by radiography (RT) or ultrasonic testing (UT). Where RT or UT is not possible, a redundant closure system is provided with field welds, which are tested as specified in Section 9.1.
2. The MPC confinement barrier is surrounded by the HI-STORM overpack which provides for the physical protection of the MPC.
3. The HI-STORM 100 System is designed to meet the requirements of storage while maintaining the safety of the SNF.
4. The SNF once initially loaded in the MPC does not require opening of the canister for repackaging to transport the SNF.
5. The decay heat emitted by the SNF is rejected from the HI-STORM 100 System through passive means. No active cooling systems are employed.

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

The structures, systems, and components of the HI-STORM 100 System designated as important to safety are identified in Table 2.2.6. Similar categorization of structures, systems, and components, which are part of the ISFSI, but not part of the HI-STORM 100 System, will be the responsibility of the 10CFR72 licensee. For HI-STORM 100A, the ISFSI pad is designated ITS, Category C as discussed in Subsection 2.0.4.1.

## 2.3.2 Protection by Multiple Confinement Barriers and Systems

### 2.3.2.1 Confinement Barriers and Systems

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

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

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

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

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

Confinement is discussed further in Chapter 7. MPC shop and field weld examinations using non-destructive methods as specified in Chapters 8 and 9 are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 9, to verify the confinement boundary.

### 2.3.2.2 Cask Cooling

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

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

### 2.3.3 Protection by Equipment and Instrumentation Selection

#### 2.3.3.1 Equipment

Design criteria for the HI-STORM 100 System are described in Section 2.2. The HI-STORM 100 System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized outside of the reactor facility's 10CFR Part 50 structures may be broken down into two broad categories, namely Important to Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", provides guidance for the determination of a component's safety classification. Certain ancillary equipment (such as trailers, rail cars, skids, portable cranes, transporters, or air pads) are not required to be designated as ITS for most ISFSI implementations, if the HI-STORM 100 is designed to withstand the failure of these components.

The listing and ITS designation of ancillary equipment in Table 8.1.6 follows NUREG/CR-6407. Ancillary equipment is classified as the highest ITS classification of any of its constituent parts/components. The ITS classification of the ancillary equipment constituent parts/components

are determined and documented in the quality program documents such as purchase specifications. ITS ancillary equipment utilized in activities that occur outside the 10CFR Part 50 structure shall be engineered to meet all functional, strength, service life, and operational safety requirements to ensure that the design and operation of the ancillary equipment is consistent with the intent of this Safety Analysis Report. The design for these components shall consider the following information, as applicable:

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

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

Users may effectuate the inter-cask transfer of the MPC between the HI-TRAC transfer cask and either the HI-STORM 100 or the HI-STAR 100 overpack in a location of their choice, depending upon site-specific needs and capabilities. For those users choosing to perform the MPC inter-cask transfer using devices not integral to structures governed by the regulations of 10 CFR Part 50 (e.g., fuel handling or reactor building), a Cask Transfer Facility (CTF) is required. The CTF may be any of the following types to effectuate the cask manipulations and MPC transfers:

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

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



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

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

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

The specific requirements for the CTF employing stationary and mobile lifting devices are somewhat different. The requirements provided in the following specification for the CTF apply to both types of lifting devices, unless explicitly differentiated in the text. The numbers in brackets { } after each design criterion indicate which of the 3 types of CTF design they apply to.

#### A General Specifications:

- i. The cask handling functions which may be required of the Cask Transfer Facility include:
  - a. Upending and downending of a HI-STAR 100 overpack on a flatbed rail car or other transporter (see Figure 2.3.1 for an example). {1, 2}
  - b. Upending and downending of a HI-TRAC transfer cask on a heavy-haul transfer trailer or other transporter (see Figure 2.3.2 for an example). {1, 2}
  - c. Raising and placement of a HI-TRAC transfer cask on top of a HI-STORM 100 overpack for MPC transfer operations (see Figure 2.3.3 for an example of the cask arrangement with the standard design HI-TRAC transfer cask. The HI-TRAC 100D and 125D designs would include the mating device and no transfer lid). {1, 2, 3}
  - d. Raising and placement of a HI-TRAC transfer cask on top of a HI-STAR 100 overpack for MPC transfer operations (see Figure 2.3.4 for an example of the cask arrangement with the standard design HI-

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

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

ii. Other Functional Requirements:

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

- a. Devices and areas to support installation and removal of the HI-STORM overpack lid. {1, 2, 3}
- b. Devices and areas to support installation and removal of the HI-STORM 100 overpack vent shield block inserts. {1, 2, 3}
- c. Devices and areas to support installation and removal of the HI-STAR 100 closure plate. {1, 2, 3}
- d. Devices and areas to support installation and removal of the HI-STAR 100 transfer collar. {1, 2, 3}
- e. Features to support positioning and alignment of the HI-STORM overpack and the HI-TRAC transfer cask. {1, 2, 3}
- f. Features to support positioning and alignment of the HI-STAR 100 overpack and the HI-TRAC transfer cask. {1, 2, 3}
- g. Areas to support jacking of a loaded HI-STORM overpack for insertion of a translocation device underneath. {1, 2, 3}
- h. Devices and areas to support placement of an empty MPC in the HI-TRAC transfer cask or HI-STAR 100 overpack. {1, 2, 3}
- i. Devices and areas to support receipt inspection of the MPC, HI-TRAC transfer cask, HI-STORM overpack, and HI-STAR overpack. {1, 2, 3}

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

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

- Connector Brackets: The mechanical part used in the load path which connects to the cask trunnions. A fabricated weldment, slings, and turnbuckles are typical examples of connector brackets. {1, 2, 3}
- CTF structure: The CTF structure is the stationary, anchored portion of the CTF which provides the required structural function to support MPC transfer operations, including lateral stabilization of the HI-TRAC transfer cask and, if required, the overpack, to protect against seismic events. The MPC lifter, if used in the CTF design, is integrated into the CTF structure (see Lifter Mount). {1}
- HI-TRAC lifter(s): The HI-TRAC lifter is the mechanical lifting device, typically consisting of jacks or hoists, that is utilized to lift a loaded or unloaded HI-TRAC to the required elevation in the CTF so that it can be mounted on the overpack.<sup>†</sup> {1, 2, 3}
- Lifter Mount: A beam-like structure (part of the CTF structure) that supports the HI-TRAC and MPC lifter(s). {1}
- Lift Platform: The lift platform is the intermediate structure that transfers the vertical load of the HI-TRAC transfer cask to the HI-TRAC lifters. {1}
- Mobile lifting devices: A mobile lifting device is a device defined in ASME B30.5-1994, Mobile and Locomotive Cranes. A mobile lifting device may be used in lieu of the HI-TRAC lifter and/or an MPC

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

lifter provided all requirements set forth in this subsection are satisfied. {2}

- MPC lifter: The MPC lifter is a mechanical lifting device, typically consisting of jacks or hoists, that is utilized to vertically transfer the MPC between the HI-TRAC transfer cask and the overpack. {1}
- Pier: The portion of the reinforced concrete foundation which projects above the concrete floor of the CTF. {1}
- Single-Failure-Proof (SFP): A single-failure-proof handling device is one wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria given in of NUREG-0612 and/or is designed in accordance with ANSIN14.6 and employs redundant drop protection features. {1, 2, 3}
- Translocation Device: A low vertical profile device used to laterally position an overpack such that the bottom surface of the overpack is fully supported by the top surface of the device. Typical translocation devices are air pads and Hillman rollers. {1,2}
- Vertical Cask Transporter: A device which is capable of performing the CTF functions as well as transporting the transfer cask and overpack to and from the CTF. A vertical cask transporter may be used in lieu of the CTF structure, HI-TRAC lifter, and/or an MPC lifter provided all requirements set forth in this subsection are satisfied. {3}

iv. Important to Safety Designation:

All components and structures which comprise the CTF shall be given an ITS category designation in accordance with a written procedure which is consistent with NUREG/CR-6407 and the Holtec quality assurance program. {1,2,3}

B. Environmental and Design Conditions

- i. Lowest Service Temperature (LST): Unless otherwise specified, the LST for the CTF is 0°F (consistent with the specification for the HI-TRAC transfer cask in Subsection 3.1.2.3). Based on its local meteorological data, a host nuclear site may use a higher LST value for the Cask Transfer Facility. {1, 2, 3}

- ii. Snow and Ice Load, S: The CTF structure shall be designed to withstand the dead weight of snow and ice for unheated structures as set forth in ASCE 7-88 [2.2.2] for the specific ISFSI site. { 1 }
- iii. Tornado Missile, M, and Tornado Wind, W': The tornado wind and tornado-generated missile data applicable to the HI-STORM 100 System (Tables 2.2.4 and 2.2.5) will be used in the design of the CTF unless existing site design basis data or a probabilistic risk assessment (PRA) for the CTF site with due consideration of short operation durations indicates that a less severe tornado missile impact or wind loading on the CTF can be postulated. The PRA analysis can be performed in the manner of the EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology Computer Code Manual". USNRC Reg. Guide 1.117 and Section 2.2.3 of NUREG-800 may be used for guidance in establishing the appropriate tornado missile and wind loading for the CTF.

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

- The missile has a planform area of 20 sq. ft. and impact force characteristics consistent with the HI-TRAC missile impact analysis.
- The large missile can strike the CTF in any orientation up to an elevation of 15 feet.

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

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

Intermediate penetrant missile and small missiles postulated in this FSAR are

not considered to be a credible threat to the functional integrity of the CTF and, therefore, need not be considered. {1, 2, 3}

- iv. Flood: The CTF will be assumed to be flooded to the highest elevation for the CTF facility determined from the local meteorological data. The flood velocity shall be taken as the largest value defined for the ISFSI site. {1, 2, 3}
- v. Lightning: Meteorological data for the region surrounding the ISFSI site shall be used to specify the applicable lightning input to the CTF for personnel safety evaluation purposes. {1, 2, 3}
- vi. Water Waves (Tsunami, Y): Certain coastal CTF sites may be subject to sudden, short duration waves of water, denoted in the literature by various terms, such as tsunami. If the applicable meteorological data for the CTF site indicates the potential of such water-borne loadings on the CTF, then such a loading, with due consideration of the short duration of CTF operations, shall be defined for the CTF. {1, 2, 3}
- vii. Design Basis Earthquake (DBE), E: The DBE event applicable to the CTF facility pursuant to 10CFR100, Appendix A, shall be specified. The DBE should be specified as a set of response spectra or acceleration time-histories for use in the CTF structural and impact consequence analyses. {1, 2, 3}
- viii. Design Temperature: Unless a lower value can be justified for a specific site, all material properties used in the stress analysis of the CTF structure shall utilize a reference design temperature of 150°F. {1, 2, 3}

C. Heavy Load Handling:

- i. Apparent dead load, D\*: The dead load of all components being lifted shall be increased in the manner set forth in Subsection 3.4.3 to define the Apparent Dead Load, D\*. {1, 2, 3}
- ii. NUREG-0612 Conformance:

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

- a. Mobile lifting devices at the CTF shall conform to the guidelines of

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

- b. Section 5.1.6(2) of NUREG-0612 specifies that new cranes should be designed to meet the requirements of NUREG-0554. For mobile lifting devices, the guidance of Section 5.1.6(2) of NUREG-0612 does not apply. {2}
- c. Vertical cask transporters shall be designed in accordance with ANSI N14.6 and shall employ redundant drop protection features. {3}

iii. Defense-in-Depth Measures:

- a. The lift platform and the lifter mount shall be designed to ensure that the stresses produced under the apparent dead load,  $D^*$ , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures. {1}
- b. The CTF structure shall be designed to ensure that the stresses produced in it under the apparent dead load,  $D^*$ , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures. {1}
- c. Maximum deflection of the lift platform and the lifter mount under the apparent dead load shall comply with the limits set forth in CMAA-70. {1}
- d. When the HI-TRAC transfer cask is stacked on the overpack, HI-TRAC shall be either held by the lifting device or laterally restrained by the CTF structure. Furthermore, when the HI-TRAC transfer cask is placed atop the overpack, the overpack shall be laterally restrained from uncontrolled movement, if required by the analysis specified in Subsection 2.3.3.1.N. {1}
- e. The design of the lifting system shall ensure that the lift platform (or lift frame) is held essentially horizontal at all times and that the symmetrically situated axial members are symmetrically loaded.

{1,3}

- f. In order to minimize occupational radiation exposure to ISFSI personnel, design of the MPC lifting attachment (viz., sling) should not require any human activity inside the HI-TRAC cylindrical space. {1, 2, 3}
- g. The HI-TRAC lifter and MPC lifter shall possess design features to avoid excessive side-sway of the payload during lifting operations. {1, 2, 3}
- h. The lifter (HI-TRAC and MPC) design shall ensure that any electrical malfunction in the motor or the power supply will not lead to an uncontrolled lowering of the load. {1, 2, 3}
- i. The kinematic stability of HI-TRAC or HI-STORM standing upright in an unrestrained configuration (if such a condition exists during the use of the CTF) shall be analytically evaluated and ensured under all postulated extreme environmental phenomena loadings for the CTF facility. {1, 2, 3}

iv. Shielding Surety:

The design of the HI-TRAC and MPC lifters shall preclude the potential for the MPC to be removed, completely or partially, from the cylindrical space formed by the HI-TRAC and the underlying overpack. {1, 2, 3}

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

A mobile lifting device, if used in the CTF in the role of the HI-TRAC lifter or MPC lifter is governed in part by ANSI/ASME N45.2.15 with technical requirements specified in ANSI B30.5 (1994). {2}

When lifting the MPC from an overpack to the HI-TRAC transfer cask, limit switches or load limiters shall be set to ensure that the lifted load does not exceed 110% of the loaded MPC weight. {2,3}

An analysis of the consequences of a potential MPC vertical drop which conforms to the guidelines of Appendix A to NUREG-0612 shall be performed. The analysis shall demonstrate that a postulated drop would not result in the MPC developing a thru-wall breach resulting in loss of confinement or experiencing a deceleration in excess of its design basis



deceleration specified in this FSAR. {2}

- vi. Lift Height Limitation: The HI-TRAC lift heights shall be governed by the Technical Specifications. {1,2,3}
- vii. Control of Side Sway: Procedures shall provide provisions to ensure that the load is lifted essentially vertically with positive control of the load. Key cask lifting and transfer procedures, as determined by the user, should be reviewed by the Certificate Holder before their use. {1, 2, 3}

D. Loads and Load Combinations for the CTF Structure

The applicable loadings for the CTF have been summarized in paragraph B in the preceding. A stress analysis of the CTF structure shall be performed to demonstrate compliance with the Subsection NF stress limits for Class 3 linear structures for the service condition germane to each load combination. Table 2.3.2 provides the load combinations (the symbols in Table 2.3.2 are defined in the preceding text and in Table 2.2.13). {1}

E. Materials and Failure Modes

- i. Acceptable Materials and Material Properties: All materials used in the design of the CTF shall be ASME or ASTM approved or equal, consistent with the ITS category of the part (see discussion in subsection 2.I.0). Reinforced concrete, if used, shall comply with the provisions of ACI 318 (89). The material property and allowable stress values for all steel structures shall be taken from the ASME and B&PV Code, Section II, wherever such data is available; otherwise, the data provided in the ASTM standards shall be used. {1, 2, 3}
- ii. Brittle Fracture: All materials used for structural components of the CTF structure and the lift platform, designated as primary load bearing, shall meet the fracture toughness requirements for ASME Section III, Subsection NF, and Class 3 structures (NF-2300). If required, Charpy impact test temperature shall be equal to the designated LST or lower. {1, 2, 3}
- iii. Fatigue: Fatigue failure modes of primary structural members in the CTF structure whose failure may result in uncontrolled lowering of the HI-TRAC transfer cask or the MPC (critical members) shall be evaluated. A minimum factor of safety of 2 on the number of permissible loading cycles on the critical members shall apply. {1, 2, 3}

- iv. Buckling: For all critical members in the CTF structure (defined above), potential failure modes through buckling under axial compression shall be considered. The margin of safety against buckling shall comply with the provisions of ASME Section III, Subsection NF, for Class 3 linear structures. {1, 2, 3}

F. CTF Pad

A reinforced concrete pad in conformance with the specification for the ISFSI pad set forth in this FSAR (see Table 2.2.9) may be used in the region of the CTF where the overpack and HI-TRAC are stacked for MPC transfer. Alternatively, the pad may be designed using the guidelines of ACI-318(89). {1, 2, 3}

G. Miscellaneous Components

Hoist rings, turnbuckles, slings, and other appurtenances which are in the load path during heavy load handling at the CTF shall have enhanced margin of safety per ANSIN14.6 or be single-failure-proof. {1, 2, 3}

H. Structural Welds

All primary structural welds in the CTF structure shall comply with the specifications of ASME Section III for Class 3 NF linear structures. {1}

I. Foundation

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

J. Rail Access

The rail lines that enter the Cask Transfer Facility shall be set at grade level with no exposed rail ties or hardware other than the rail itself. {1,2}

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

- i. The cask handling bay in the CTF shall allow access of a vertical cask crawler or translocation device carrying a transfer cask or overpack. The building floor shall be equipped with a smooth transition to the cask travel route such that the vertical cask crawler tracks do not have to negotiate sharp lips or slope transitions and the translocation devices have a smooth

transition. Grading of exterior aprons shall be no more than necessary to allow water drainage. {1}

- ii. If roll-up doors are used, the roll up doors shall have no raised threshold that could damage the vertical cask crawler tracks (if a crawler is used). {1}
- iii. Exterior aprons shall be of a material that will not be damaged by the vertical cask crawler tracks, if a crawler is used. {1}

L. Facility Floor

- i. The facility floor shall be sufficiently flat to allow optimum handling of casks with a translocation device. {1}
- ii. Any floor penetrations, in areas where translocation device operations may occur, shall be equipped with flush inserts. {1}
- iii. The rails, in areas where translocation device operations may occur shall be below the finish level of the floor. Flush inserts, if necessary, shall be sized for installation by hand. {1}

M. Cask Connector Brackets

- i. Primary lifting attachments between the cask and the lifting platform are the cask connector brackets. The cask connector brackets may be lengthened or shortened to allow for differences in the vehicle deck height of the cask delivery vehicle and the various lifting operations. The connector brackets shall be designed to perform cask lifting, upending and downending functions. The brackets shall be designed in accordance with ANSI N14.6 [Reference 2.2.3] and load tested at 300% of the load applied to them during normal handling. {1, 2, 3}
- ii. The connector brackets shall be equipped with a positive engagement to ensure that the cask lifting attachments do not become inadvertently disconnected during a seismic event and during normal cask handling operations. {1, 2, 3}
- iii. The design of the connector brackets shall ensure that the HI-TRAC transfer cask is fully secured against slippage during MPC transfer operations. {1, 2, 3}

N. Cask Restraint System

A time-history analysis of the stacked overpack/HI-TRAC transfer cask assemblage under the postulated ISFSI Level D events in Table 2.3.2 shall be performed to demonstrate that a minimum margin of safety of 1.1 against overturning or kinematic instability exists and that the CTF structure complies with the applicable stress limits (Table 2.3.2) and that the maximum permissible deceleration loading specified in the FSAR is not exceeded. If required to meet the minimum margin of safety of 1.1, a cask restraining system shall be incorporated into the design of the Cask Transfer Facility to provide lateral restraint to the overpack (HI-STORM or HI-STAR 100). If the HI-STORM/HI-TRAC stack is laterally supported such that the fundamental natural frequency of the beam mode vibration of the stack is in the rigid range of the horizontal acceleration time histories (or the corresponding response spectra), then the dynamic time history solution converges to the static solution using the ZPA (see Glossary) of the corresponding time history. In such a case, a time history analysis or a static analysis may be performed since the maximum predicted responses from both solutions are identical. {1, 2, 3}

O. Design Life

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

P. Testing Requirements

In addition to testing recommended in NUREG-0612 (1980), a structural adequacy test of the CTF structure at 125% of its operating load prior to its first use in a cask loading campaign shall be performed. This test should be performed in accordance with the guidance provided in the CMAA Specification 70 [2.2.16]. {1}

Q. Quality Assurance Requirements

All components of the CTF shall be manufactured in full compliance with the quality assurance requirements applicable to the ITS category of the component as set forth in the Holtec QA program. {1, 2, 3}

R. Documentation Requirements

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

- Maintenance Drawings

- Operating Procedures

- ii. Design Report: if required by the safety classification, a QA-validated design report documenting full compliance with the provisions of this specification shall be prepared and archived for future reference in accordance with the provisions of the Holtec QA program. {1, 2, 3}

### 2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STORM 100 System, instrumentation which is important to safety is not necessary. No instrumentation is required or provided for HI-STORM 100 storage operations, other than normal security service instruments and TLDs.

However, in lieu of performing the periodic inspection of the HI-STORM overpack vent screens, temperature elements may be installed in two of the overpack exit vents to continuously monitor the air temperature. If the temperature elements and associated temperature monitoring instrumentation are used, they shall be designated important to safety as specified in Table 2.2.6.

The temperature elements and associated temperature monitoring instrumentation provided to monitor the air outlet temperature shall be suitable for a temperature range of -40°F to 500°F. At a minimum, the temperature elements and associated temperature monitoring instrumentation shall be calibrated for the temperatures of 32°F (ice point), 212°F (boiling point), and 449°F (melting point of tin) with an accuracy of +/- 4°F.

### 2.3.4 Nuclear Criticality Safety

The criticality safety criteria stipulates that the effective neutron multiplication factor,  $k_{\text{eff}}$ , including statistical uncertainties and biases, is less than 0.95 for all postulated arrangements of fuel within the cask under all credible conditions.

#### 2.3.4.1 Control Methods for Prevention of Criticality

The control methods and design features used to prevent criticality for all MPC configurations are the following:

- a. Incorporation of permanent neutron absorbing material in the MPC fuel basket walls.
- b. Favorable geometry provided by the MPC fuel basket

Additional control methods used to prevent criticality for the MPC-24, MPC-24E, and MPC-24EF (all with higher enriched fuel), and the MPC-32 and MPC-32F are the following:

- a. Loading of PWR fuel assemblies must be performed in water with a minimum boron content as specified in Table 2.1.14 or 2.1.16, as applicable.
- b. Prevention of fresh water entering the MPC internals.

Administrative controls and shall be used to ensure that fuel placed in the HI-STORM 100 System meets the requirements described in Chapters 2 and 6. All appropriate criticality analyses are presented in Chapter 6.

#### 2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criticality analyses performed in Chapter 6. Because biases and uncertainties are explicitly evaluated in the analysis, it is not necessary to introduce additional contingency for error.

#### 2.3.4.3 Verification Analyses

In Chapter 6, critical experiments are selected which reflect the design configurations. These critical experiments are evaluated using the same calculation methods, and a suitable bias is incorporated in the reactivity calculation.

#### 2.3.5 Radiological Protection

##### 2.3.5.1 Access Control

As required by 10CFR72, uncontrolled access to the ISFSI is prevented through physical protection means. A peripheral fence with an appropriate locking and monitoring system is a standard approach to limit access. The details of the access control systems and procedures, including division of the site into radiation protection areas, will be developed by the licensee (user) of the ISFSI utilizing the HI-STORM 100 System.

##### 2.3.5.2 Shielding

The shielding design is governed by 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive doses in excess of the limits given in Table 2.3.1 for normal, off-normal, and accident conditions.

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

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

Dose rates in the immediate vicinity of the loaded overpack are important in consideration of occupational exposure. Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on the contents of the BWR and PWR MPCs permitted for storage as described in Section 2.1.9. Actual dose rates in operation will be lower than those reported in Chapter 5 for the following reasons:

- The shielding evaluation model has a number of conservatisms, as discussed in Chapter 5.
- No single cask will likely contain design basis fuel in each fuel storage location and the full compliment of non-fuel hardware allowed by Section 2.1.9.
- No single cask will contain fuel and non-fuel hardware at the limiting burnups and cooling times allowed by Section 2.1.9.

Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM 100 System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask. A design objective for the maximum average radial surface dose rate has been established as 300 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 175 mrem/hr. The average dose rate at the top of the overpack is limited to below 60 mrem/hr. Chapter 5 of this FSAR presents the analyses and evaluations to establish HI-STORM 100 compliance with these design objectives.

Because of the passive nature of the HI-STORM 100 System, human activity related to the system is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 10. Chapter 10 also provides information concerning temporary shielding which may be utilized to reduce the personnel dose during loading, unloading, transfer, and handling operations. The estimated occupational doses for personnel comply with the requirements of 10CFR20.

For the loading and unloading of the HI-STORM overpack with the MPC, several transfer cask designs are provided (i.e., HI-TRAC 125, HI-TRAC 100, HI-TRAC 100D and HI-TRAC 125D). The two 125 ton HI-TRAC provide better shielding than the HI-TRAC 100D and 125D due to the increased shielding thickness and corresponding greater weight. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would normally dictate that the 125 ton HI-TRAC should be used. However, sites may not be capable of utilizing the 125 ton HI-TRAC due to crane capacity limitations, floor loading limitations, or other site-specific considerations. As with other dose reduction-based plant activities, individual users who cannot accommodate the 125 ton HI-TRAC should perform a cost-benefit analysis of the actions (e.g., plant modifications) that would

be necessary to use the 125 ton HI-TRAC. The cost of the action(s) would be weighed against the value of the projected reduction in radiation exposure and a decision made based on each plant's particular ALARA implementation philosophy.

Dose rates at the restricted area and site boundaries shall be in accordance with applicable regulations. Licensees shall demonstrate compliance with 10CFR72.104 and 10CFR72.106 for the actual fuel being stored, the ISFSI storage array, and the controlled area boundary distances.

The analyses presented in Chapters 5, 10, and 11 demonstrate that the HI-STORM 100 System is capable of meeting the above radiation dose limits.

#### 2.3.5.3 Radiological Alarm System

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

#### 2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM 100 System. No such materials would be stored within an ISFSI. However, for conservatism we have analyzed a hypothetical fire accident as a bounding condition for HI-STORM 100. An evaluation of the HI-STORM 100 System in a fire accident is discussed in Chapter 11.

Small overpressures may result from accidents involving explosive materials which are stored or transported near the site. Explosion is an accident loading condition considered in Chapter 11.



Table 2.3.1

## RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

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

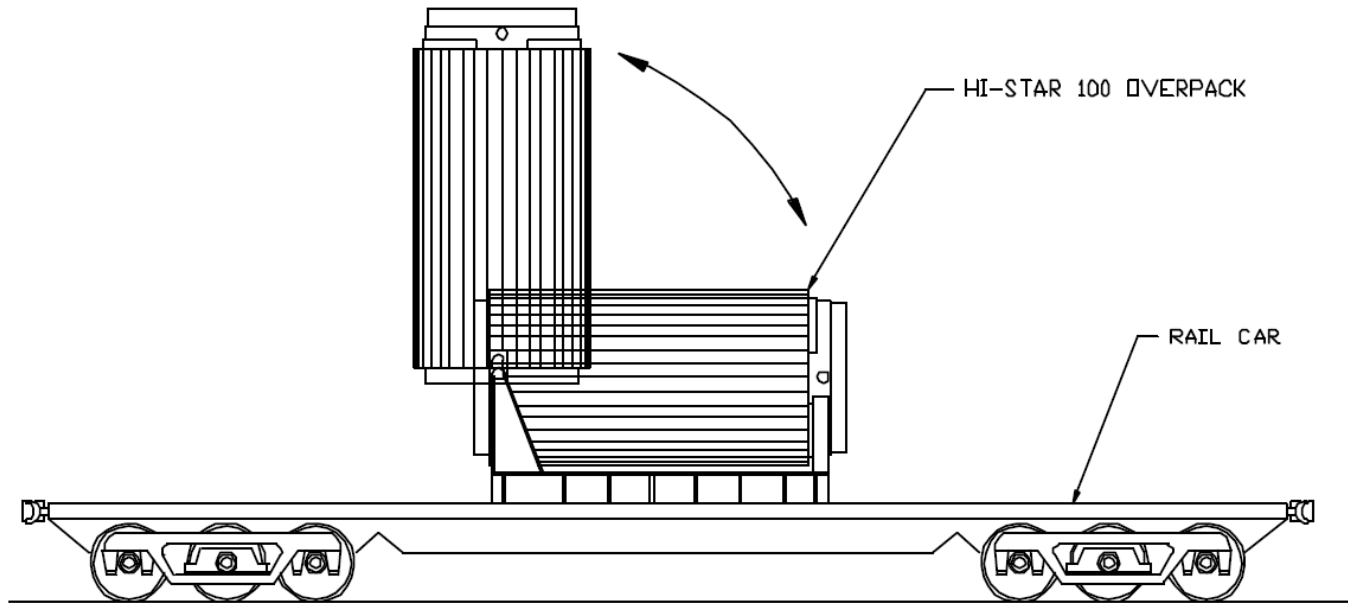
Table 2.3.2

Load Combinations<sup>†</sup> and Service Condition Definitions for the CTF Structure

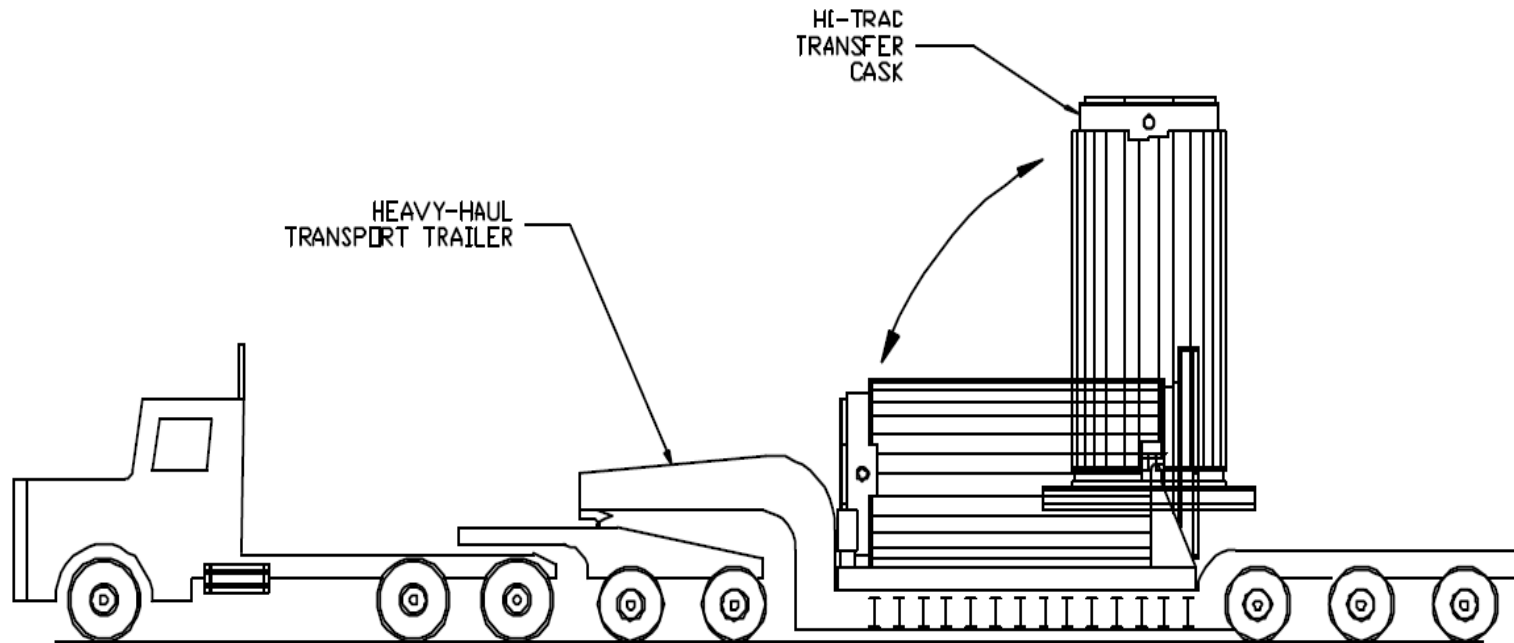
Load Combination	Service Condition for Section III of the ASME Code for Definition of Allowable Stress	Comment
D*	Level A	All primary load bearing members must satisfy Level A stress limits.
D+S	Level A	
D+M <sup>††</sup> +W <sup>'</sup>  D+F  D+E or D+Y	Level D	Factor of safety against overturning shall be $\geq 1.1$

<sup>†</sup> The reinforced concrete portion of the CTF structure shall also meet factored combinations of the above loads set forth in ACI-318(89).

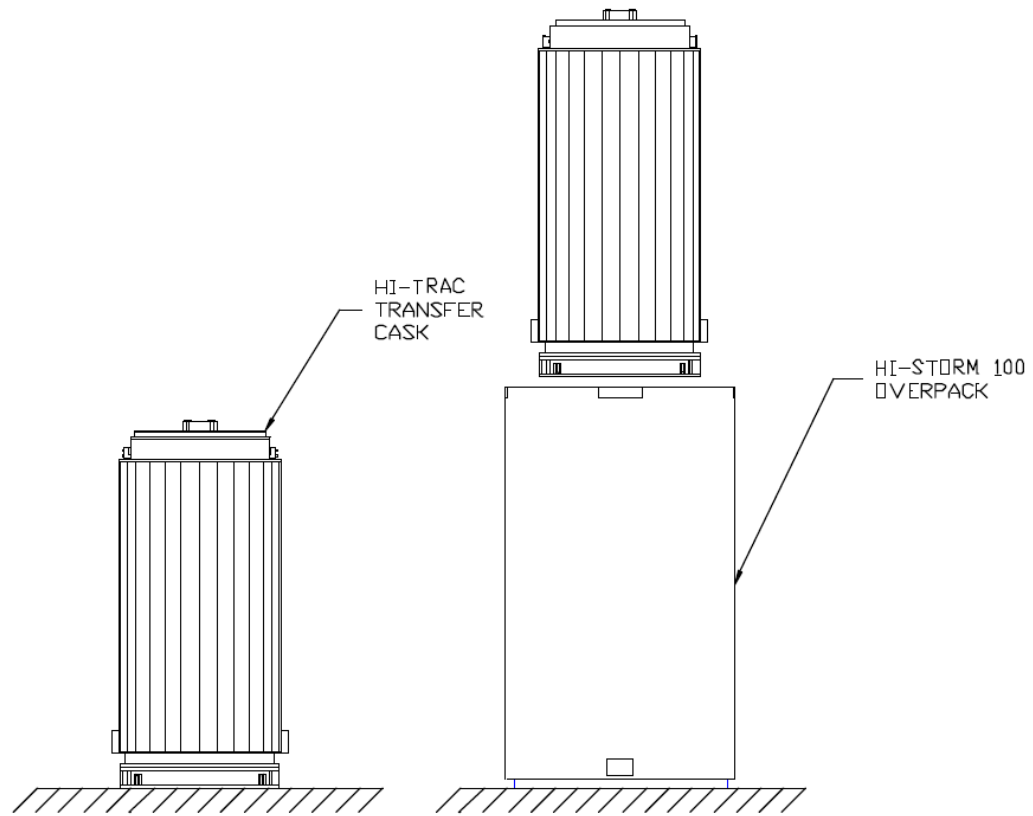
<sup>††</sup> This load may be reduced or eliminated based on a PRA for the CTF site.



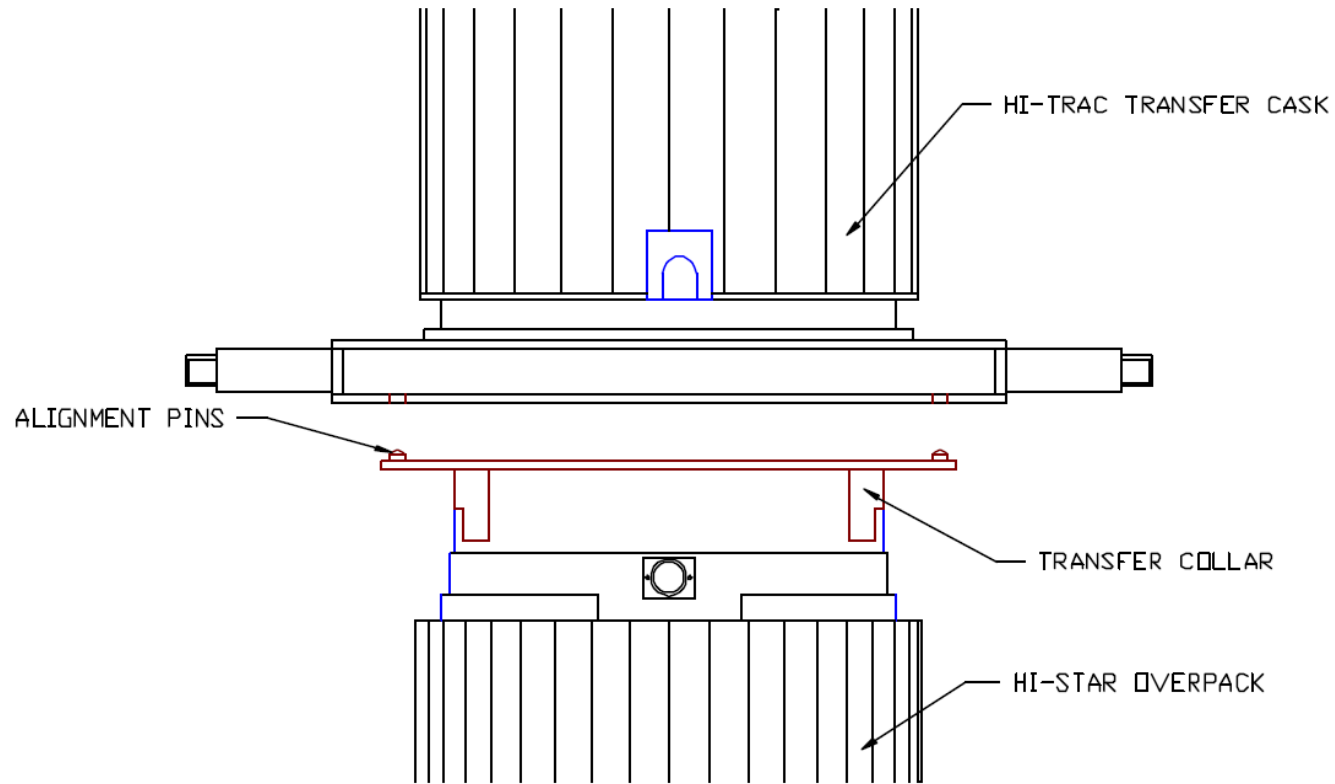
**FIGURE 2.3.1: HI-STAR 100 UPENDING AND DOWNENDING ON A RAIL CAR**



**FIGURE 2.3.2: HI-TRAC UPENDING AND OWNENDING ON A HEAVY-HAUL TRANSPORT TRAILER**



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



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

## 2.4 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STORM 100 System. The HI-STORM 100 System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site.

The MPC is being licensed for transport off-site in the HI-STAR 100 dual-purpose cask system (Reference Docket No. 71-9261). No further handling of the SNF stored in the MPC is required prior to transport to a licensed centralized storage facility or licensed repository.

The MPC which holds the SNF assemblies is engineered to be suitable as a waste package for permanent internment in a deep Mined Geological Disposal System (MGDS). The materials of construction permitted for the MPC are known to be highly resistant to severe environmental conditions. No carbon steel, paint, or coatings are used or permitted in the MPC in areas where they could be exposed to spent fuel pool water or the ambient environment. Therefore, the SNF assemblies stored in the MPC should not need to be removed. However, to ensure a practical, feasible method to defuel the MPC, the top of the MPC is equipped with sufficient gamma shielding and markings locating the drain and vent locations to enable semiautomatic (or remotely actuated) boring of the MPC lid to provide access to the MPC vent and drain. The circumferential welds of the MPC lid closure ring can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the overpack consists of steel and concrete rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the overpack, and the latter reused for storage of other MPCs.

In either case, the overpack would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

If the MPC needs to be opened and separated from the SNF before the fuel is placed into the MGDS, the MPC interior metal surfaces will be decontaminated using existing mechanical or chemical methods. This will be facilitated by the MPC fuel basket and interior structures' smooth metal surfaces designed to minimize crud traps. After the surface contamination is removed, the MPC radioactivity will be diminished significantly, allowing near-surface burial or secondary applications at the licensee's facility.

It is also likely that both the overpack and MPC, or extensive portions of both, can be further decontaminated to allow recycle or reuse options. After decontamination, the only radiological hazard the HI-STORM 100 System may pose is slight activation of the HI-STORM 100 materials caused by irradiation over a 40-year storage period.

Due to the design of the HI-STORM 100 System, no residual contamination is expected to be left behind on the concrete ISFSI pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last overpack is removed.

To evaluate the effects on the MPC and HI-STORM overpack caused by irradiation over a 40-year storage period, the following analysis is provided. Table 2.4.1 provides the conservatively determined quantities of the major nuclides after 40 years of irradiation. The calculation of the material activation is based on the following:

- Beyond design basis fuel assemblies (B&W 15x15, 4.8% enrichment, 70,000 MWD/MTU, and five-year cooling time) stored for 40 years. A constant source term for 40 years was used with no decrease in the neutron source term. This bounds the source term associated with the limiting PWR burnup of 68,200 MWD/MTU.
- Material quantities based on the drawings in Section 1.5.
- A constant flux equal to the initial loading condition is conservatively assumed for the full 40 years.
- Material activation is based on MCNP-4A calculations.

As can be seen from the material activation results presented in Table 2.4.1, the MPC and HI-STORM overpack activation is very low, even including the very conservative assumption of a constant flux for 40 years. The results for the concrete in the HI-STORM overpack can be conservatively applied to the ISFSI pad. This is extremely conservative because the overpack shields most of the flux from the fuel and, therefore, the ISFSI pad will experience a minimal flux.

In any case, the HI-STORM 100 System would not impose any additional decommissioning requirements on the licensee of the ISFSI facility per 10CFR72.30, since the HI-STORM 100 System could eventually be shipped from the site.



Table 2.4.1  
MPC ACTIVATION

<b>Nuclide</b>	<b>Activity After 40-Year Storage (Ci/m<sup>3</sup>)</b>
<sup>54</sup> Mn	2.20e-3
<sup>55</sup> Fe	3.53e-3
<sup>59</sup> Ni	2.91e-6
<sup>60</sup> Co	3.11e-4
<sup>63</sup> Ni	9.87e-5
<b>Total</b>	<b>6.15e-3</b>

HI-STORM OVERPACK ACTIVATION

<b>Nuclide</b>	<b>Activity After 40-Year Storage (Ci/m<sup>3</sup>)</b>
Overpack Steel	
<sup>54</sup> Mn	3.62e-4
<sup>55</sup> Fe	6.82e-3
<b>Total</b>	<b>7.18e-3</b>
Overpack Concrete	
<sup>39</sup> Ar	3.02e-6
<sup>41</sup> Ca	2.44e-7
<sup>54</sup> Mn	1.59e-7
<sup>55</sup> Fe	2.95e-5
<b>Total</b>	<b>3.43e-5</b>

## 2.5 REGULATORY COMPLIANCE

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

## 2.6 REFERENCES

- [2.0.1] American Concrete Institute, "Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)".
- [2.0.2] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan<sup>†</sup>
- [2.0.3] Deleted.
- [2.0.4] NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," USNRC, Washington, D.C. Rev. 1 (1986).
- [2.0.5] J.W. McConnell, A.L. Ayers, and M.J. Tyacke, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Component According to Important to Safety," Idaho Engineering Laboratory, NUREG/CR-6407, INEL-95-0551, 1996.
- [2.0.6] NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities, March 2000.
- [2.0.7] ASME Code, Section III, Subsection NF and Appendix F, and Code Section II, Part D, Materials, 1995, with Addenda through 1997.
- [2.0.8] "Cladding Considerations for the Transportation and Storage of Spent Fuel," USNRC Interim Staff Guidance-11, Revision 3, November 17, 2003.
- [2.0.9] USNRC Memorandum from Christopher L. Brown to M. Wayne Hodges, "Scoping Calculations for Cladding Hoop Stresses in Low Burnup Fuel," dated January 29, 2004.
- [2.1.1] ORNL/TM-10902, "Physical Characteristics of GE BWR Fuel Assemblies", by R.S. Moore and K.J. Notz, Martin Marietta (1989).
- [2.1.2] U.S. DOE SRC/CNEAF/96-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1994, Feb. 1996.
- [2.1.3] Deleted.

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

- [2.1.4] Deleted.
- [2.1.5] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.
- [2.1.6] DOE Multi-Purpose Canister Subsystem Design Procurement. Specification.
- [2.1.7] S.E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks", SAND-89-0018, Sandia National Laboratory, Oct., 1989.
- [2.1.8] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.
- [2.2.1] ASME Boiler & Pressure Vessel Code, American Society of Mechanical Engineers, 1995 with Addenda through 1997.
- [2.2.2] ASCE 7-88 (formerly ANSI A58.1), "Minimum Design Loads for Buildings and Other Structures", American Society of Civil Engineers, New York, NY, 1990.
- [2.2.3] ANSI N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More", June 1993.
- [2.2.4] Holtec Report HI-2012610, "Final Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 72-1008, latest revision.
- [2.2.5] Holtec Report HI-951251, "Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 71-9261, latest revision.
- [2.2.6] "Debris Collection System for Boiling Water Reactor Consolidation Equipment", EPRI Project 3100-02 and ESEERCO Project EP91-29, October 1995.
- [2.2.7] Design Basis Tornado for Nuclear Power Plants, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, April 1974.
- [2.2.8] ANSI/ANS 57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (dry type)", American Nuclear Society, LaGrange Park, Illinois.
- [2.2.9] NUREG-0800, SRP 3.5.1.4, USNRC, Washington, DC.

- [2.2.10] United States Nuclear Regulatory Commission Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", August 1973 and Rev. 1, April 1976.
- [2.2.11] "Estimate of Tsunami Effect at Diablo Canyon Nuclear Generating Station, California." B.W. Wilson, PG&E (September 1985, Revision 1).
- [2.2.12] D. Peckner and I.M. Bernstein, "Handbook of Stainless Steels," McGraw-Hill Book Company, 1977.
- [2.2.13] "Nuclear Systems Materials Handbook," Oak Ridge National Laboratory, TID 26666, Volume 1.
- [2.2.14] Deleted.
- [2.2.15] Deleted.
- [2.2.16] Crane Manufacturer's Association of America (CMAA), Specification #70, 1988, Section 3.3.
- [2.2.17] Holtec Report HI-2114830, "Final Safety Analysis Report on the HI-STORM FW MPC Storage System", NRC Docket No. 72-1032, Revision 4.

## APPENDIX 2.A

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

#### 2.A.1 General Comments

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

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

#### 2.A.2 General Requirements for ISFSI Pad

- Consistent with the provisions of NUREG-1567 [2.0.6], all concrete work shall comply with the requirements of ACI-349-85 [2.0.2].
- All reinforcing steel shall be manufactured from high strength billet steel conforming to ASTM designation A615 Grade 60.
- The ISFSI owner shall develop appropriate mixing, pouring, reinforcing steel placement, curing, testing, and documentation procedures to ensure that all provisions of ACI 349-85 [2.0.2] are met.
- The placement, depth, and design and construction of the slab shall take into account the depth of the frost line at the ISFSI location. The casks transmit a very small amount of heat into the cask pad through conduction. The American Concrete Institute guidelines on reinforced concrete design of ground level slabs to minimize thermal and shrinkage induced cracking shall be followed.

5. General Requirements for Steel Embedment: The steel embedment, excluding the pre-tensioned anchorage studs, is required to follow the provisions stipulated in ACI 349-85 [2.0.2], Appendix B "Steel Embedment" and the associated Commentary on Appendix B, as applicable. Later editions of this Code may be used provided a written reconciliation is performed. An example of one acceptable embedment configuration is provided in Figure 2.A.1. Site-specific embedment designs may vary from this example, depending on the geotechnical characteristics of the site-specific foundation. The embedment designer shall consider any current, relevant test data in designing the pad embedment for HI-STORM 100A and HI-STORM 100SA.
6. The ISFSI owner shall ensure that pad design analyses, using interface loads provided in this report, demonstrate that all structural requirements of NUREG-1567 and ACI-349-85 are satisfied.
7. Unless the load handling device is designed in accordance with ANSIN14.6 and incorporates redundant drop protection features, the ISFSI owner shall ensure that a permissible cask carry height is computed for the site-specific pad/foundation configuration such that the design basis deceleration set forth in this FSAR are not exceeded in the event of a handling accident involving a vertical drop.
8. The ISFSI owner shall ensure that the pad/foundation configuration provides sufficient safety margins for overall kinematic stability of the cask/pad/foundation assemblage.
9. The ISFSI owner shall ensure that the site-specific seismic inputs, established at the top surface of the ISFSI pad, are bounded by the seismic inputs used as the design basis for the attachment components. If required, the ISFSI owner shall perform additional analyses to ensure that the site-specific seismic event or durations greater than the design basis event duration analyzed in this report, do not produce a system response leading to structural safety factors (defined as allowable stress (load) divided by calculated stress (load)) less than 1.0. Table 2.0.5 and Table 2.2.8 provide the limiting values of ZPAs in the three orthogonal directions that must not be exceeded at an ISFSI site (on the pad top surface) to comply with the general CoC for the HI-STORM 100A (and 100SA) System.
10. An ISFSI pad used to support anchored HI-STORM overpacks, unlike the case of free standing overpacks, may experience tensile (vertically upward) anchorage forces in addition to compression loads. The reinforcing steel (pattern and quantity) must be selected to meet the demands of the anchorage forces under seismic and other environmental conditions that involve destabilizing loadings (such as the large tornado missile defined in this FSAR).

### 2.A.3 Steel Embedment for Anchored Casks

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

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

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

- i. The weakest structural link in the system shall be in the ductile member. In other words, the tension capacity of the anchor stud/anchor receptacle group (based on the material ultimate strengths) shall be less than the concrete pull-out strength (computed with due recognition of the rebars installed in the pad).
- ii. The maximum ratio of embedment plus cask anchor stud effective tensile stiffness to the effective compressive stiffness of the embedment plus concrete shall not exceed 0.25 in order to ensure the effectiveness of the pre-load.
- iii. The maximum axial stress in the cask anchor studs under normal and seismic conditions shall be governed by the provisions of ASME Section III Subsection NF (1995).



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

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

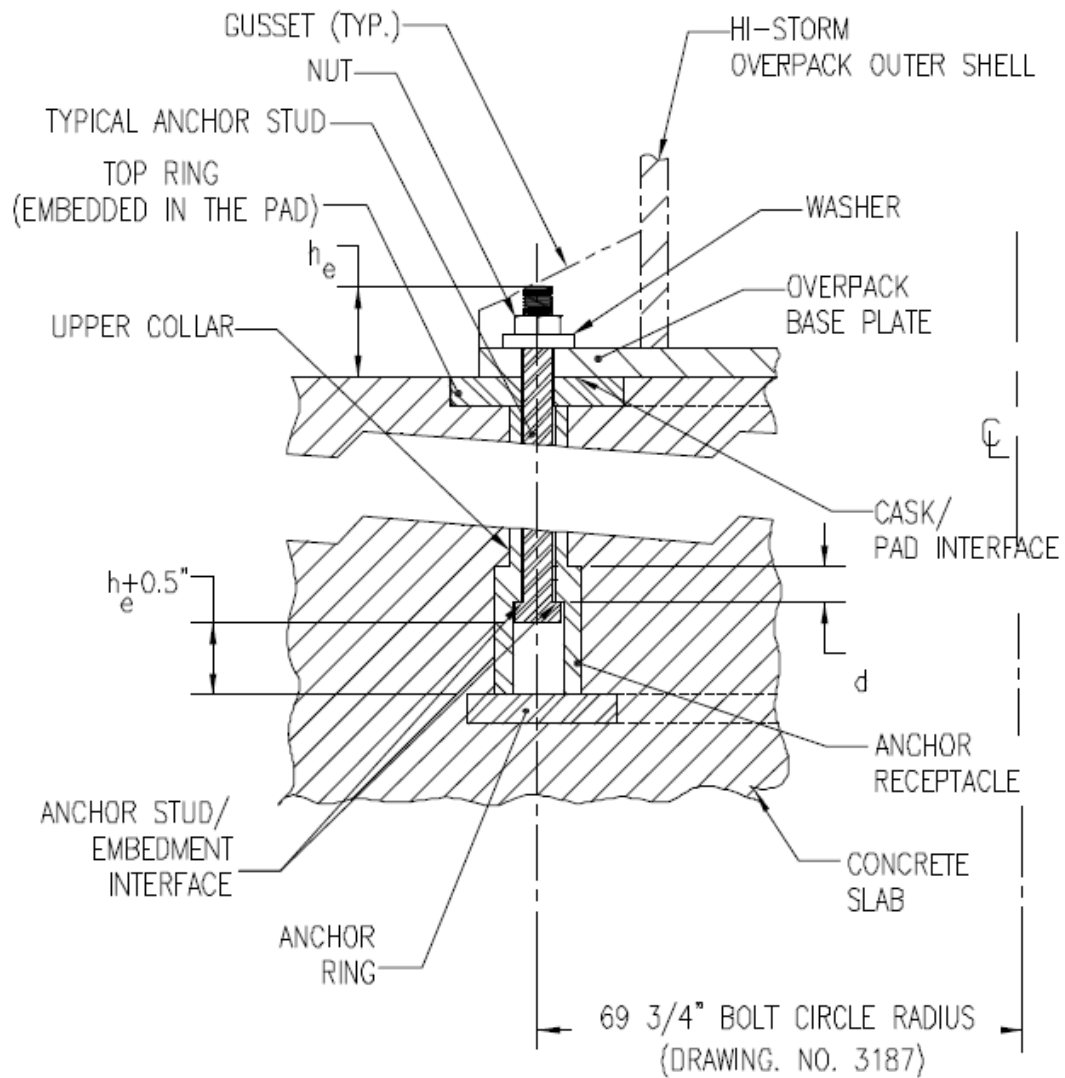
Table 2.A.1

## Typical Embedment and Anchoring Data\*

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

\* Refer to Figure 2.A.1

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



**FIGURE 2.A.1: Typical HI-STORM/ISFSI pad Fastening Detail**

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

## Appendix 2.B The Forced Helium Dehydration (FHD) System

### 2.B.1 System Overview

The Forced Helium Dehydration (FHD) system is used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump or an inert gas has been expelled in the water blowdown operation. The FHD system is required to be used for MPCs containing at least one high burnup fuel assembly. The FHD method of moisture removal is optional for all other MPCs.

Expelling the water from the MPC using a conventional pump or a water displacement method using inert gas would remove practically all of the contained water except for the small quantity remaining on the MPC baseplate below the bottom of the drain line and an even smaller adherent amount wetting the internal surfaces. A skid-mounted, closed loop dehydration system will be used to remove the residual water from the MPC such that the partial pressure of the trace quantity of water vapor in the MPC cavity gas is brought down to  $\leq 3$  torr. The FHD system, engineered for this purpose, shall utilize helium gas as the working substance.

The FHD system, schematically illustrated in Figure 2.B.1, can be viewed as an assemblage of four thermal modules, namely, (i) the condensing module, (ii) the demister module, (iii) the helium circulator module and (iv) the pre-heater module. The condensing module serves to cool the helium/vapor mixture exiting the MPC to a temperature well below its dew point such that water may be extracted from the helium stream. The condensing module is equipped with suitable instrumentation to provide a direct assessment of the extent of condensation that takes place in the module during the operation of the FHD system. The demister module, engineered to receive partially cooled helium exiting the condensing module, progressively chills the recirculating helium gas to a temperature that is well below the temperature corresponding to the partial pressure of water vapor at 3 torr.

The motive energy to circulate helium is provided by the helium circulator module, which is sized to provide the pressure rise necessary to circulate helium at the requisite rate. The last item, labeled the pre-heater module, serves to pre-heat the flowing helium to the desired temperature such that it is sufficiently warm to boil off any water present in the MPC cavity.

The pre-heater module, in essence, serves to add supplemental heat energy to the helium gas (in addition to the heat generated by the stored SNF in the MPC) so as to facilitate rapid conversion of water into vapor form. The heat input from the pre-heater module can be adjusted in the manner of a conventional electric heater so that the recirculating helium entering the MPC is sufficiently dry and hot to evaporate water, but not unduly hot to place unnecessary thermal burden on the condensing module.

The FHD system described in the foregoing performs its intended function by continuously removing water entrained in the MPC through successive cooling, moisture removal and reheating of the working substance in a closed loop. In a classical system of the FHD genre, the moisture removal

operation occurs in two discrete phases. In the beginning of the FHD system's operation (Phase 1), the helium exiting the MPC is laden with water vapor produced by boiling of the entrained bulk water. The condensing module serves as the principal device to condense out the water vapor from the helium stream in Phase 1. Phase 1 ends when all of the bulk water in the MPC cavity is vaporized. At this point, the operation of the FHD system moves on to steadily lowering the relative humidity and bulk temperature of the circulating helium gas (Phase 2). The demister module, equipped with the facility to chill flowing helium, plays the principal role in the dehydration process in Phase 2.

## 2.B.2 Design Criteria

The design criteria set forth below are intended to ensure that design and operation of the FHD system will drive the partial pressure of the residual vapor in the MPC cavity to  $\leq 3$  torr if the gas has met the specified temperature or dew point value and duration criteria. The FHD system shall be designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

- i. The temperature of helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.
- ii. The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).
- iii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulent flow regime in the MPC cavity.
- iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. The limit will be met if the gas temperature at the demister outlet is verified by measurement to remain  $\leq 21^\circ\text{F}$  for  $\geq 30$  minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain  $\leq 22.9^\circ\text{F}$  for  $\geq 30$  minutes.

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

- i. The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point of 120°F or less.
- ii. The demister module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in Section 2.B.1) has been completed.
- iii. The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

- iv. The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

### 2.B.3 Analysis Requirements

The design of the FHD system shall be subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this FSAR.

- i. System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.
- ii. System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demoinsturizer modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach  $\leq 3$  torr if the temperature of the helium gas exiting the demoinsturizer is predicted to be at a maximum of 21°F for 30 minutes.
- iii. Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

### 2.B.4 Acceptance Testing

The first FHD system designed and built for the MPC drying function required by HI-STORM's technical specifications shall be subject to confirmatory testing as follows:

- a. A representative quantity of water shall be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.
- b. The MPC cavity drying test shall be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).

- c. The drain and vent line RVOAs on the MPC lid shall be connected to the terminals located in the pre-heater and condensing modules of the FHD system.
  
- d. The FHD system shall be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demister module has been at or below 21°F for thirty minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

## **Appendix 2.C**

### **The Supplemental Cooling System**

#### 2.C.1 Purpose

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

#### 2.C.2 General Description and Requirements

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

- (i) If the system uses water as the coolant, the system is sized to limit the coolant temperature to below 180°F under steady-state conditions for the design basis heat load at an ambient air temperature of 110°F. Active components (i.e., pump or air-cooler fan) are powered by electric motors with a backup power supply for uninterrupted operation.
- (ii) The system will utilize a contamination-free fluid medium in contact with the external surfaces of the MPC and inside surfaces of the HI -TRAC transfer cask to minimize corrosion. Figure 2.C.1 shows a typical P&ID for a SCS.
- (iii) The number of active components in the SCS will be minimized.
- (iv) All passive components such as tubular heat exchangers, manually operated valves and fittings shall be designed to applicable standards (TEMA, ANSI).

#### 2.C.3 Thermal/Hydraulic Design Criteria

- (i) The heat dissipation capacity of the SCS shall be equal to or greater than the minimum necessary to ensure that the peak cladding temperature of High-Burnup fuel assemblies is below the ISG-11, Rev. 3 limit of 400°C (752°F). All heat transfer surfaces in any heat exchangers shall be assumed to be fouled to the maximum limits specified in a widely used heat exchange equipment standard such as the Standards of Tubular Exchanger Manufacturers Association.
- (ii) The coolant utilized to extract heat from the MPC shall be either high purity water or air. Anti-freeze may be used to prevent water from freezing if warranted by operating conditions.



## 2.C.4 Mechanical Requirements

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

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

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

- (iv) Not Used.
- (v) The SCS skid shall be equipped with appropriate lifting lugs to permit its handling by the plant's lifting devices in full compliance with NUREG-0612 provisions.

## 2.C.5 Regulatory Requirements

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

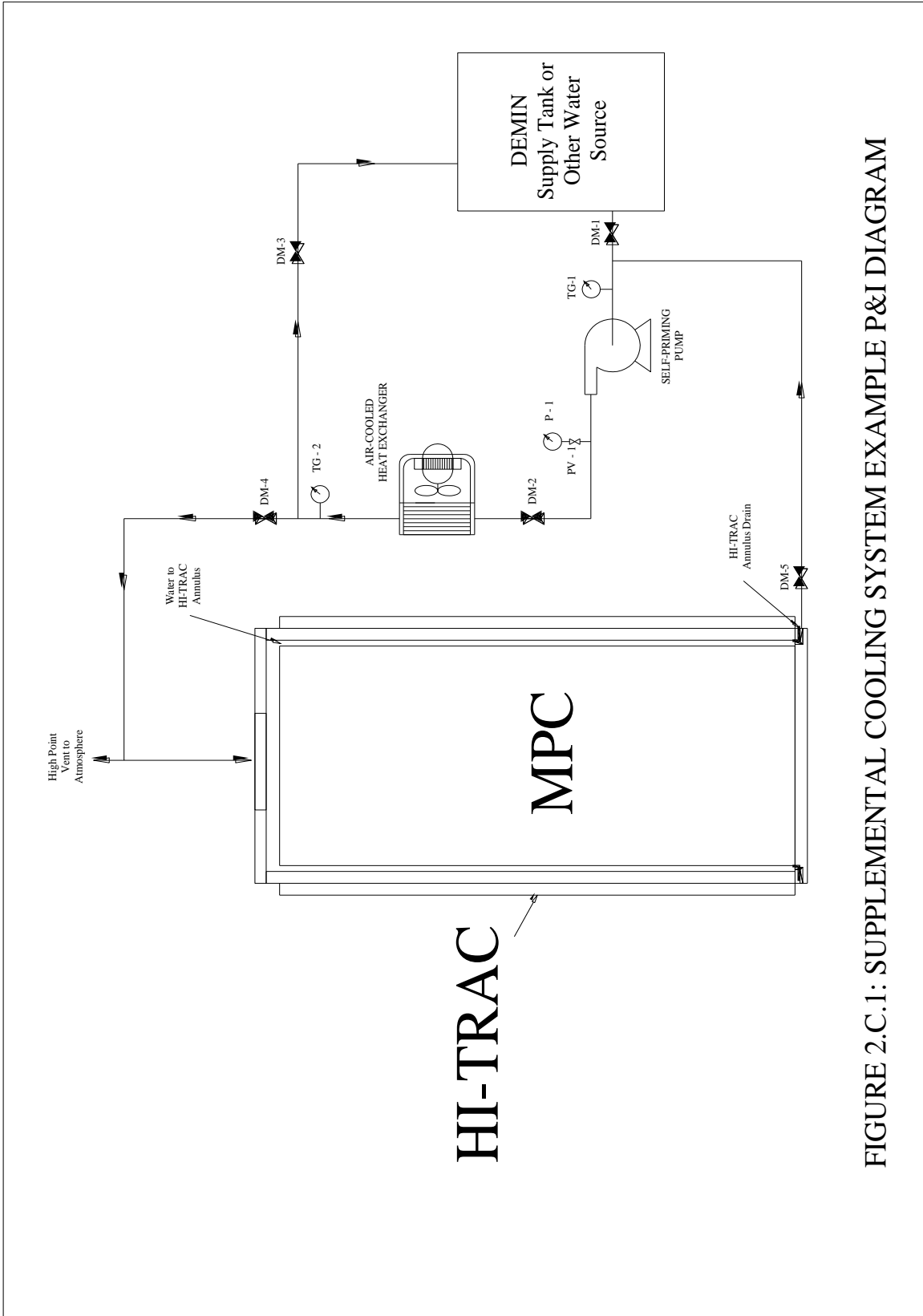


FIGURE 2.C.1: SUPPLEMENTAL COOLING SYSTEM EXAMPLE P&I DIAGRAM

## SUPPLEMENT 2.I

### PRINCIPAL DESIGN CRITERIA FOR THE HI-STORM 100U SYSTEM

#### 2.I.0 OVERVIEW OF THE PRINCIPAL DESIGN CRITERIA

##### General

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

- i. VVM components
- ii. ISFSI structures

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

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

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

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

## Structural

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

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

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

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

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

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

values are required. In this category lie VVM Component properties such as specific gravity and thermal expansion coefficient.

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

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

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

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

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

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

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

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

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

The appropriate critical characteristics for structural components of the VVM, therefore, are:

- i. Material yield strength,  $\sigma_y$
- ii. Material ultimate strength,  $\sigma_u$
- iii. Elongation,  $\epsilon$
- iv. Charpy impact strength at the lowest service temperature for the part,  $C_i$

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than the minimum values used in the qualifying analyses in this FSAR. The above *critical characteristics* apply to all materials used in the structural parts of the CEC. Table 2.I.9 provides guidance for the critical characteristics associated with the steels used in the VVM.

In the event that one or more of the *critical characteristics* of the replacement material is slightly lower than the original material, then the use of the §72.48 process is necessary to ensure that all regulatory predicates for the material substitution are fully satisfied.

In addition to the design configuration, the maximum magnitude of Design Basis Earthquake for the “100U” ISFSI is also specified in this FSAR. A three-dimensional non-linear time-history solution procedure implemented on LS-DYNA is used in Supplement 3.I to qualify the ISFSI including the storage system. This same three-dimensional non-linear time-history solution procedure may be used to perform safety evaluation under 10CFR72.212 at a host site, as indicated in Paragraph 2.I.6(v). Likewise, the loadings from the extreme environmental phenomena, defined in the main body of Chapter 2, are considered in Supplement 3.I. Site specific loadings that deviate from those analyzed in Supplement 3.I are subject to 72.212 safety evaluations in the manner of all HI-STORM models.

To serve their intended functions, the CEC and Closure Lid shall ensure physical protection, biological shielding, and allow the retrieval of the MPC under all conditions of storage (10 CFR 72.122(l)). Because the VVM is an in-ground structure, drops and tip-over of the VVM are not credible events and, therefore, do not warrant analysis. The load cases germane to establishing the structural adequacy of the VVM pursuant to 10 CFR 72.24(c) are compiled in Table 2.I.5. The physical characteristics of the MPCs intended for storage in the VVMs are presented in the main body of Chapter 1.

The design bases and criteria provided in this supplement are intended to quantify the safety margins in the VVM design with respect to all applicable loadings that follow from the provisions of 10CFR72.24(c)(3), §72.122(b) and §72.122(c).

### Thermal

The engineered thermal performance of the HI-STORM 100U system is essentially equivalent to its aboveground counterparts under quiescent conditions. Ambient air enters from a circumferential opening provided in the Closure Lid. The intake air flows downward through an annular passage or intake plenum formed between the CEC and the Divider Shell. At the bottom of the intake plenum

the air turns inwards through openings or cutouts provided in the Divider Shell bottom and rises up through an annular gap formed between the MPC and the Divider Shell. Heat is dissipated from the MPC to this upward rising column of air. The rising air column enters the curved flow passages engineered in the Closure Lid and exhausts from the top through a large central opening (see Figure 1.I.4). To minimize the heating of the downward flowing inlet air and the upward column of heated air, the divider shell is insulated on its outside surface. The *critical characteristic* of the insulation is specified in Table 2.I.1. This thermal insulation material is required to meet the service conditions (temperature and humidity) for the design life of the VVM. Because the thermal performance of the HI-STORM 100U relies on buoyancy-driven convection of air and because of the relative proximity of the inlet and outlet vents to each other, the effect of wind on its thermal performance is also considered.

The allowable long-term and short term section-average temperature limits for concrete (used in the Closure Lid) are established in Appendix 1.D. Section-average temperature limits for structural steel in the VVM are provided in Table 2.I.8.

The VVM is designed for extreme cold conditions, as discussed in Subsection 2.2.2.2. The safety of structural steel material used for the VVM from brittle fracture is discussed in Subsection 3.1.2.3.

The VVM is designed to reject the maximum allowable heat load as defined below in a reliable and testable manner consistent with its important-to-safety designation (10CFR72.128(a)(4)).

The maximum permissible HI-STORM 100U heat load  $Q(X)$  is a function of the parameter “X” defined as the ratio of the maximum permissible inner region assembly heat load  $q_1$ , and outer region assembly heat load  $q_2$ . The inner and outer fuel storage regions are defined in Table 2.1.27. The functional relationship  $Q(X)$  is presented below:

$$Q(X) = 2 \cdot \alpha \cdot Q_d / (1 + X^y) \text{ where } y = 0.23/X^{0.1}$$

$Q_d$  is the maximum heat load where  $X=1$  (34kW) and  $\alpha$  is a penalty factor for underground storage discussed in Supplement 4.I.

### Shielding

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

The VVM is designed to limit the dose rates for all MPCs to ALARA values. The VVM is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The underground location of the VVM significantly reduces the radiation from the ISFSI at the site boundary compared to an aboveground cask. The calculated VVM dose rates are discussed in Supplement 5.I, which also discusses dose rates during site construction next to an operating ISFSI.

The dose rate calculations presented in Supplement 5.I conservatively use a lower density for the subgrade than is specified in Table 2.I.2. For dose rate calculation at a particular ISFSI, the spatial average of the actual subgrade density shall be used.

### Criticality

The VVM does not perform any criticality control function. The MPCs provide criticality control for all design basis normal, off-normal and postulated accident conditions, as discussed in Chapter 6.

### Confinement

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

### Operations

MPC preparation for storage and onsite transport of the MPC in the HI-TRAC transfer cask is the same for the VVM as for the aboveground overpack designs. The cask transporter is used to move the loaded transfer cask to the ISFSI and to transfer the MPC into the VVM. Generic operating instructions for the use of the HI-STORM 100U System that parallel those for the aboveground overpack are provided in Supplement 8.I.

### Acceptance Tests and Maintenance

The fabrication acceptance bases and maintenance program to be applied to the VVM are described in Supplement 9.I. Application of these requirements will assure that the VVM is fabricated and maintained in a manner that satisfies the design criteria defined in this FSAR.

### Decommissioning

Decommissioning considerations for the HI-STORM 100U System, including the VVM, are addressed in Section 2.I.11.

#### 2.I.1 SPENT FUEL TO BE STORED



There is no difference in the authorized contents of the HI-STORM 100U VVM and the aboveground HI-STORM systems. The information in Section 2.1 is applicable.

## 2.I.2 HI-STORM 100U VVM COMPONENTS, ISFSI STRUCTURES, AND CORROSION MITIGATION MEASURES

The VVM is engineered for below-grade storage for the duration of its design life, and is designed to withstand normal, off-normal, and extreme environmental phenomena as well as accident conditions of storage with appropriate margins of safety.

As discussed in Supplement 1.I, the VVM Components are (see Figure 1.I.2):

1. The MPC Cavity Enclosure Container (CEC), and
2. The Closure Lid

The CEC is comprised of the following subcomponents:

1. Container Shell (a cylindrical enclosure shell)
2. Bottom Plate
3. Container Flange (a top ring flange)
4. Divider Shell with insulation and MPC Guides
5. MPC bearing pads

The Closure Lid consists of:

1. The integral steel weldment (filled with shielding concrete), and
2. The removable vent screen assemblies (inlet and outlet).

The structural limit criteria imposed on the VVM Components are selected to comply with the provisions of 10CFR72, with an embedded large margin of safety. Table 2.I.1 provides the principal design criteria applicable to the VVM Components. The specifications of the materials of construction for the load bearing and non-load bearing parts are provided in Table 2.I.8 along with their maximum permissible temperature for different conditions of storage.

The interfacing SSCs, the proximate structures, and corrosion mitigation measures germane to the design of a HI-STORM 100U ISFSI are:

- i) The SFP that supports the weight of the loaded VVMs.
- ii) The ISFSI pad consisting of the VIP which provides a water seepage barrier against rainwater and melting snow and also acts as a missile barrier, and the TSP which serves as a water seepage barrier as well as the riding surface for the transporter.
- iii) The lateral subgrade (natural or engineered fill) surrounding the CEC (Space “A” in

Figure 2.I.5).

- iv) The impressed current cathodic protection system (ICCPS) that may be used as a corrosion mitigation measure for the CEC in accordance with the Technical Specifications.
- v) The concrete encasement that may be used as a corrosion mitigation measure for the CEC in accordance with Technical Specifications. Reference is made to Figure 2.I.3 for typical concrete encasement of the CEC.
- vi) The retaining wall used to protect the soil column in the Radiation Protection Space from excavation activities adjacent to the ISFSI.

Each of these is discussed below:

i. The Support Foundation Pad (SFP) (Interfacing Structure)

The structural requirements on the SFP are focused on providing a robust support to the CEC structure (for shear and compression), and to limit the long-term settlement of the SFP. The minimum structural design requirements on the SFP are provided in Table 2.I.2 and the licensing drawing in 1.I.5.

ACI-318(2005) is the prescribed Code for SFP design. As specified in ACI-318(2005), the applicable loads on the SFP are:

1. Dead load (from the TSP, the VIP, the loaded VVM, and the mass of soil above the SFP).
2. Live load (from the loaded vertical cask transporter bearing on the TSP).
3. Seismic load (the additional inertia load, in excess of the dead weight, live load transmitted to the SFP from the loaded VVM and the transporter under the ISFSI's DBE event).
4. Long-term differential settlement.

The load combinations for the structural analysis of the SFP pursuant to ACI-318(2005) are provided in Table 2.I.11.

Of the above loads, the effect of long-term settlement on the SFP is treated together with the Dead load. The standard approach to compute the long-term settlement is provided in [2.I.6]. This methodology, which is based on classical soil mechanics and is utilized in the structural analysis in Supplement 3.I, is summarized below.

1. Compute the total long-term settlement, "d", of the subgrade under the SFP (Space C) over the Design Life assuming that the total load "P" (modeled as a uniform pressure at the top of the subgrade) is equivalent to that produced by the SFP fully populated with loaded VVMs for the entire life using the methodology in [2.I.6].

2. Determine an “effective” elastic spring constant “K” of Space C that emulates the cumulative settlement:

$$K = P/d.$$

3. Using the spring constant computed above, which accounts for the effect of long-term settlement under static loading, an appropriate elastic modulus is defined for the soil column under the SFP. The degraded soil modulus so defined is used in the finite element model of the SFP to evaluate the pad flexure under the factored dead load.

The maximum permitted settlement of the SFP is limited to the value specified in Table 2.I.2. Remedial measures such as pilings must be used if the Table 2.I.2 limit cannot be met.

In the structural qualification of the SFP, the loading from the seismic event is computed using the dynamic elastic modulus corresponding to the minimum strain wave velocity of the subgrades specified in Table 2.I.2.

ii. VVM Interface Pad (VIP) (Interfacing Structure) and Top Surface Pad (TSP) (Proximate Structure)

The VIP portion of the ISFSI Pad serves no structural function in supporting the VVM. However, it girdles the Container Shell and underlies the Container Flange to form a leak tight interface, and directs water away from the CEC. The principal functions of the TSP are to provide the riding surface for the loaded transporter and also to enable rainwater to be channeled away from the storage arrays and into the site’s storm drain system. The TSP is isolated from the VIP by appropriately located expansion joints to isolate the CEC from any unbalanced loads imparted by the transporter. Similarly, an expansion joint between the CEC and the VIP is incorporated to permit differential movement between the two. The licensing drawing in Section 1.I.5 provides details for the expansion joint and typical drainage and sealing details. Because the sealing is visible and accessible, re-sealing, when and if necessary, is easily accomplished. Thus, continued sealing is assured. A specific brand of sealant is noted on the expansion joint detail, but there are several equivalent\* proven sealant materials commercially available that are ideal for this application and the expected ambient conditions.

In summary, the design objectives for the VIP and the TSP are: to provide a leak tight interface and to provide a sufficiently inflexible travel surface for the loaded transporter, respectively. The top surface of the VIP, as shown in the 100U licensing drawing package, also serves to keep rain water away from the VVM. The minimum structural design requirements on the VIP and TSP are provided in Table 2.I.2 and the licensing drawing in 1.I.5. The applicable loads on the TSP and VIP are:

1. Dead load (Self weight including settlement effects) (TSP and VIP)
2. Live load (Weight of a loaded cask transporter) (TSP only)

\* The definition of the term “equivalent” is provided in the Glossary.

3. Seismic Load (Inertia load from the concrete pad and the transporter under the ISFSI's DBE event) (TSP only).

The applicable load combinations for the structural analysis of the VIP and TSP pursuant to ACI-318(2005) are provided in Table 2.I.11.

The effect of settlement is incorporated in the stress analysis of the TSP using the same procedure as the SFP discussed above. As in the case of the SFP stress analysis, the settlement of the TSP from Dead load (self-weight) relative to the SFP over their Design Life is computed and incorporated in the stress analysis. The maximum permissible settlement of the TSP with respect to the SFP is required to be limited to the value in Table 2.I.2.

The design of the TSP together with the lateral subgrade must also satisfy the allowable bearing capacity requirement of ACI 360R-06 [2.I.8] for slabs on grade. In particular, the total load imparted by the TSP on the lateral subgrade, including the live load and seismic load from the transporter, shall be less than 50 percent of the allowable bearing capacity thereof when the load is applied uniformly.

iii. Lateral Subgrade (Interfacing SSC)

The physical characteristics of the subgrade surrounding the VVM vary from site-to-site. Further, an ISFSI owner may elect to excavate the natural subgrade and replace it with an engineered fill of an appropriate density and composition to fulfill shielding demand. While the surrounding subgrade may not provide a structural support function to the CEC structure, as an interfacing SSC, it plays a role in the loading applied to the CEC under certain scenarios, namely:

- a. during an earthquake event;
- b. during movement of the cask transporter along the Top Surface Pad;
- c. normal storage condition from the natural overburden or under the state of maximum soil saturation (hydraulic buoyancy).

During a seismic event, the subgrade surrounding the VVM may exert a time-varying lateral pressure loading on the Container Shell, which, in principle, may ovalize the Container Shell and possibly bend it like a beam.

During the movement of the cask transporter, which is loaded with the transfer cask (see Supplement 8.I for operational details), the vertical load of the cask transporter results in a lateral pressure on the Container Shell. Although the lateral pressure is apt to be quite small due to the physical restriction on how close to the Container Shell the transporter can ride, mandatory limits on the lateral separation and subgrade properties are necessary to ensure a design with adequate safety margins. Accordingly, the minimum average density and the minimum shear wave velocity in the lateral subgrade surrounding the VVMs have been specified in Table 2.I.2.

The soil overburden pressure on the Container Shell is the third loading category which must be evaluated. Also, the condition of maximum soil saturation applies a hydrostatic pressure on the CEC. The maximum value depends on the depth of the MPC storage cavity and the effective density of the saturated soil.

iv. Impressed Current Cathodic Protection System (ICCPS) (Corrosion Mitigation Measure)

If an ICCPS is required by the technical specifications, it shall be implemented in accordance with the requirements in Supplement 3.I, Subsection 3.I.4.1 and appropriate references. The following general design procedure may be followed:

1. Select the current density to be applied.
2. Compute the total current required to achieve the selected current density.
3. Design the ground bed system or distributed anode system.
4. Select a rectifier of proper voltage and current output.
5. Design all electrical circuits, fittings, and switchgear in accordance with good electrical practice.
6. Locate the cathodic protection test stations.
7. Prepare the necessary drawings and specifications for the project.

An example design is provided in this subsection for illustrative purposes and should not be interpreted as implying to present the best design or the only possible design. Because there are a multitude of ISFSI variables that will bear upon the design of the ICCPS for a particular site including differing ISFSI layouts, certain simplifying assumptions are made throughout the example. The example provides the user with insight on the types of design decisions that will need to be made. For example, because of possible shielding effects between CECs, as well as other SSC obstructions, the design implements a layout with closely distributed anodes to provide more uniform current distribution. Also, the example design implements closed loop electrical connections such that if the wire/cable is severed at any one place, electrical continuity is maintained to all anodes. Another item to be considered during the design phase is whether or not a test station is needed for each and every CEC.

Figure 2.I.1 presents an example ICCPS design layout for a 2x6 Array of VVMs. The ICCPS consists of the following four main subsystems/components:

- 1) Rectifier
- 2) Anodes
- 3) Test Stations
- 4) Wires and Cables

Figure 2.I.2 presents an example ICCPS test station.

The following is an example computation for determining the required current (approximate dimensions and quantities are used) as applicable to Figure 2.I.1:

Assume a CEC length (determined from “top of grade” to bottom of CEC bottom plate): 219.5 in.  
CEC outside diameter: 86 in.  
CEC condition: exterior is coated  
Coating efficiency: 91.5% (i.e. 8.5% of the coated CEC surface is considered bare metal)  
Cathodic Protection: Rectifier and distributed Natural Graphite Anodes with carbonaceous backfill  
Soil resistivity: 4,000 ohm/cm<sup>2</sup>  
Current density: 1 mA/ft<sup>2</sup> exposed metal  
Outside area of each CEC: 59,300 in<sup>2</sup> (412 ft<sup>2</sup>)  
Total area for an array of twelve CECs: 4,944 ft<sup>2</sup>  
Bare CEC metal exposed: 4,944 ft<sup>2</sup> x 0.085 or 420 ft<sup>2</sup>  
Current required: 420 ft<sup>2</sup> x 1 mA/ft<sup>2</sup> or 420 mA

The following is additional data applicable to Figure 2.I.1.

Approximate Anode quantity: 11  
Approximate Anode size: 5 in dia. x 120 in. long  
Approximate Backfill quantity: 6,000 lbs of carbonaceous backfill

The total number of anodes required is determined primarily by the total current requirements of the CEC metal to be protected and the optimum current density of the anode material selected.

Graphite is a semi-consumable anode. Graphite typically has experienced corrosion rates of 1.5 to 2.16 lbs /amp year [2.I.3] or as determined by experiment, 0.08 grams per square meter of anode per amp-hour of current (at 30 C, 40 mA/cm<sup>2</sup> anode current density) [2.I.4]. A computed anode life of less than 40 years is acceptable as long as appropriate measures are taken to facilitate the replacement of anodes during the design phase and appropriate maintenance planning measures are implemented. Use of carbonaceous backfill should be considered since it can substantially lengthen the anode life. Inert (non-consumable) platinized anodes may also be considered.

v. Concrete Encasement (Corrosion Mitigation Measure)

If concrete encasement is used, it shall be implemented in accordance with the requirements in Supplement 3.I, Subsection 3.I.4.1 and appropriate references.

The following points shall also be taken into consideration:

- The effect of the concrete encasement on the ICCPS, if an ICCPS is also implemented.
- The concrete encasement should not interfere with the settlement of the TSP (which provides the transporter support surface) without appropriate evaluation.

vi. Retaining Wall

Because the subgrade within and around an operating 100U ISFSI serves a principal shielding

function, it is essential that any excavation activity adjacent to the ISFSI (e.g., to build an extension of the ISFSI), must not disturb the soil in the Radiation Protection Space (RPS) shown in the licensing drawings (Section 1.I.5).

The extent of the RPS is set down to ensure, with sufficient margin of safety, that the ISFSI will continue to meet all relevant safety criteria under all applicable conditions of storage including normal, off-normal, extreme environmental phenomena and accident conditions. For example, the RPS must provide sufficient buffer so that design basis projectiles (large, medium, and penetrant missiles) will not access an MPC stored in a VVM cavity. In this case, as explained in Supplement 3.I, the incident missile is assumed to act when a deep cavity has been excavated contiguous to the RPS and the direction of action of the missile is oriented to achieve maximum penetration of the sub-grade towards the CEC shell.

A retaining wall at the edge of or beyond the RPS is recommended if an excavation activity is planned adjacent to the RPS boundary while the ISFSI is in active service. The retaining wall, as shown in the licensing drawing, shall be keyed to the TSP and connected using dowels to the SFP so that it is laterally restrained from movement but does not transmit any bending moment to the SFP or the TSP. The minimum structural design requirements on the retaining wall are provided in Table 2.I.2 and the licensing drawing in 1.I.5. The applicable load combinations for the structural analysis of the retaining wall pursuant to ACI-318(2005) are provided in Table 2.I.11.

When a retaining wall is installed on one or more sides of the 100U ISFSI, excavation activities associated with the construction of a new underground ISFSI can be performed directly adjacent to the retaining wall(s) at depths above the bottom surface of the existing SFP. Soil excavations below the bottom surface of the existing SFP shall be treated as though no retaining walls are installed and, therefore, are subject to the limitations of the following paragraph.

For the case where a retaining wall is not installed, no excavation activities associated with the construction of a new underground ISFSI shall take place within a distance from the RPS equal to ten times the planned excavation depth. Alternatively, the Excavation Exclusion Zone (EEZ), defined as the minimum distance from the centerline of a VVM located on the periphery of the ISFSI to where the effect of DBE is sufficiently attenuated such that a full depth excavation will not cause collapse of the lateral sub-grade at the RPS boundary during an earthquake, can be determined by a site specific seismic analysis. If a retaining wall is installed at or beyond the RPS then the wall becomes the EEZ boundary, but only for excavation depths above the bottom surface of the existing SFP.

### 2.I.3 Service Conditions and Applicable Loads

The categories of loads on the HI-STORM 100U VVM are identified below. They parallel those for the aboveground systems.

- Normal Condition: dead weight, handling of the Closure Lid, soil overburden pressure from subgrade, live load due to cask transporter movement, snow loads, and buoyancy effect of

water saturation of surrounding subgrade and foundation. Most normal condition loadings occur at an ambient temperature denoted as the “normal storage condition temperature”; however, for calculations involving the Closure Lid, a higher temperature is assumed when the VVM carries a loaded MPC since the Closure Lid outlet ducts will be subject to heated air.

- Off-Normal Condition: elevated ambient temperature and partial blockage of air inlets.
- Extreme Environmental Phenomena and Accident Condition: handling accidents, fire, tornado, flood, earthquake, explosion, lightning, burial under debris, 100% blockage of air inlets, extreme environmental temperature, 100% fuel rod rupture, and an accident during construction in the vicinity of a loaded ISFSI.

The design basis magnitudes of the above loads, as applicable, are provided in Tables 2.I.1 and 2.I.5, and are discussed further in the following subsections. Applicable loads for an MPC contained in a VVM or for a HI-TRAC that services a VVM are identical to those already identified in the main body of Chapter 2 and, therefore, are not repeated or discussed within this supplement. However, recognizing that the support of an MPC in a VVM is different from the support provided in an above ground HI-STORM, the design basis dynamic analysis model includes the fuel assemblies, the fuel basket, and the enclosure vessel so that the loads described above are properly distributed within the VVM.

#### 2.I.4 Normal Condition Operating Parameters and Loads

##### i. Dead Load

The HI-STORM 100U System must withstand the static loads due to the weight of each of its components. As the support provided by the subgrade and the VVM Interface Pad from lateral friction is apt to be negligible, the weight of the Closure Lid is assumed to bear on the Container Flange and the Container Shell; the load to the VVM Support Foundation is transferred through direct bearing action.

##### ii. Handling Loads

The only instance of a handling load occurs during emplacement or removal of the Closure Lid while the CEC contains a loaded MPC. To provide defense-in-depth, Closure Lid lifting attachments are conservatively designed to meet the design requirements of ANSI N14.6 [2.2.3].

Lift locations for the CEC and the Divider Shell are used for lifting only during construction, and possibly during maintenance and decommissioning of the VVM with no loaded MPC present; therefore, these lifting locations are not subject to the defense-in-depth measures of NUREG-0612. They are therefore considered as a part of the site construction safety plan,



site-specific maintenance program, or site decommissioning plan, as applicable, and as such are treated as being outside the scope of this FSAR.

iii. Live Loads

a. Subgrade Pressure Due to Transporter Movement

The properties of the surrounding subgrade and the presence of a loaded cask transporter affect the state of stress in the subgrade continuum. This stress field may produce a lateral compressive load on the Container Shell, which acts together with the effect from soil overburden.

b. MPC Transfer Operation

The VVM must withstand the weight of the loaded HI-TRAC transfer cask and the mating device during MPC transfer operations. Bounding weights for these components are used in the qualifying analysis.

iv. Ambient Temperature

The HI-STORM 100U System is analyzed for the same maximum yearly average ambient air temperature as that used for the aboveground overpacks. This normal operating condition temperature bounds all locations in the continental United States.

v. Snow

An appropriately conservative snow load on the Closure Lid is considered as a potential bounding case (see Table 2.I.1).

vi. Long-Term Settlement

There is no mechanism for an appreciable long-term settlement of Support Foundation Pad from loaded VVMs because the equivalent density of the loaded VVM is nearly equal to the density of the removed subgrade. Therefore, at an ISFSI site, depending on the density of the subgrade, there may be a small mismatch between the mass of the subgrade displaced by the loaded VVM leading to a minor amount of long-term settlement.

The TSP and the VIP are founded on well conditioned subgrade that is typically installed after the CECs are emplaced on the SFP. In addition to its own weight, the sole long-term load is the dead weight of the pads (TSP and VIP) which is evidently insufficient to cause appreciable long-term settlement in a subgrade continuum installed to meet or exceed a specific shear wave velocity and density criterion (see Table 2.I.2). Therefore, the long-term settlement of the TSP and the VIP relative to the SFP is expected to be small. Limiting allowable values of settlement for the SFP and the TSP have been specified in Table 2.I.2

for a conservative stress analysis of the TSP and SFP under the load combinations including the generic Design Basis Earthquake .

The effect of long-term settlement on the SFP and TSP shall be considered as a concurrent load with Dead load in all load combinations in the manner described in Section 2.I.2.

#### 2.I.5 Off-Normal Condition Design Criteria

##### i. Elevated Ambient Air Temperature

The HI-STORM 100U System must be able to reject the design basis heat load under short-term conditions of elevated ambient air temperature.

##### ii. Partial Blockage of Inlet Air Ducts

The HI-STORM 100U System must withstand 50% blockage of the inlet air flow area without exceeding allowable temperature and pressure limits.

#### 2.I.6 Environmental Phenomena and Accident Condition Criteria

The extreme environmental phenomena and accident conditions applicable to the HI-STORM 100U System are listed below. The loadings apply to either or both the VVM components and ISFSI structures.

##### i. Handling Accidents (Drops and Tipover)

Because the VVM is situated underground and cannot be moved, drop and tipover events are not credible accidents for this design. The Closure Lid, as discussed in Supplement 1.I, cannot strike the MPC lid due to geometry constraints if it were to undergo a free fall. Further, because the load handling device and lifting equipment are required to meet the defense-in-depth criteria set down in this FSAR, the drop of the Closure Lid or transfer cask during handling operation is termed non-credible (as is the case for the aboveground HI-STORM system MPC transfer operations at the ISFSI).

##### ii. Fire

The VVM must withstand the effects of a fire that consumes the maximum volume of fuel permitted to be in the fuel tank of the cask transporter. The duration of the fire for the VVM is conservatively assumed to be the same as that used for the aboveground overpacks. As is the case for aboveground overpacks, the fuel is assumed to spill, surround one storage system and burn until it is depleted. Because the VVM is configured to have a surrounding built-in step or spill barrier (see Figure 1.I.3), the spilled fuel will collect and burn over the Top Surface Pad, also referred to as Top-of-Grade (see Figure 1.I.2). Therefore, the location of fuel combustion will be somewhat removed from the CEC. Also, the natural grade in the

TSP surface, engineered to direct the rainwater away from the VVMs, will do the same to the spilled fuel, further ameliorating the thermal consequence of the fire to the stored SNF.

The closed-end geometry of the MPC storage cavity ensures that a sustained combustion of the fuel, even if it were to be hypothesized to enter the VVM cavity, is not possible.

The loss of shielding effectiveness due to heat up of the concrete and the surrounding SSCs is primarily due to vaporization of the small amount of volatiles, including the contained moisture present in the concrete. This reduction in shielding is small and is permitted under the regulations. Therefore, the fire analysis of the VVM is focused on determining safety against a structural collapse due to elevation in the structure's metal temperature.

The sole effect of fire on the VVM structure is to raise the metal temperature of the structural members surrounding the shielding concrete in the Closure Lid. The analysis for the fire event accordingly seeks to establish that the load bearing structure will not be weakened by the rise in its metal temperature (and a consequent reduction in the yield and ultimate strength) and result in its structural collapse.

iii. Tornado

The HI-STORM 100U System is protected from the effects of a tornado and accompanying missiles by virtue of its underground configuration. The only VVM component that warrants evaluation for the effects of a tornado-induced missile strike is the Closure Lid, which is made of a steel weldment with encased concrete.

The HI-STORM 100U System is inherently stable under tornado missile impact. The impact of a large missile (1800kg Automobile) is evaluated to determine whether the Closure Lid continues to maintain its required shielding function. Penetration and perforation issues associated with the Closure Lid due to intermediate missiles that constitute the Extreme Environmental Phenomena loads for the HI-STORM 100U system are also addressed. The Closure Lid is analyzed for penetration of a solid steel cylinder traveling at a high speed consistent with the characteristics of the intermediate missile listed in Table 2.2.5. As there is no direct line of sight to the MPC, small missiles are not considered. Also, since a tornado is a short duration event, the effect of extremely high tornado winds on the thermal performance of the VVM would be negligible due to the system's thermal inertia. Therefore, the effect of tornado wind on the thermal performance of the HI-STORM 100U system is not analyzed.

iv. Flood

As discussed in Subsection 1.I.2, the HI-STORM 100U System is engineered to be flood resistant. However, even though the potential water ingress passages are elevated in the HI-STORM 100U (in contrast to the pad level inlet ducts in typical ventilated overpacks), submersion flooding that fills all or a portion of the ducts could occur at certain ISFSI sites

located in flood zones. The MPC is designed to withstand 125 feet of water submergence. The VVM will clearly withstand this static head of water above the surface of the ISFSI because all structural members either are not subject to any pressure differential from the flood or are backed by the subgrade, which resists the flood water directly. Full or partial submergence of the MPC is not a concern from a thermal perspective, as discussed in Supplement 1.I, because heat removal is enhanced by the floodwater.

The most severe flooding event from a thermal perspective would be the partial filling of the intake plenum such that airflow is blocked but the MPC is not submerged in water. To mitigate the consequences of this event, the height of the Divider Shell cutouts is purposely located well above the bottom elevation of the MPC. Therefore, if the flood level is just high enough to block air flow, the lower portion of the MPC will be submerged in water. The wetted MPC bottom region serves as an efficient means of heat rejection to the floodwater. This accident event is described in Supplement 11.I.

v. Earthquake

As explained herein and in Subsection 3.I.4.7, the generic seismic loading for the HI-STORM 100U system is established using the combination of an earthquake and soil subgrade properties that maximize the severity of the inertia forces on the ISFSI structures and components.

As required by 10CFR72.102(f), the Design Basis Earthquake for the ISFSI must be specified. For the HI-STORM 100U system, a generic Design Basis Earthquake is specified with horizontal and vertical ZPAs intended to envelope the site-specific DBEs at all U.S. plant sites (See Table 2.I.2). For purposes of the generic seismic analysis in this FSAR, the Design Basis Earthquake for the HI-STORM 100U system is defined by two sets of response spectra specified at the SFP bottom surface elevation and at the TSP top surface elevation, as shown in Figure 2.I.4. These two spectra sets together exhibit the severity of the earthquake experienced by the ISFSI structures and VVM Components and are henceforth referred to as the governing spectra. The two sets of response spectra are obtained from the two-step SHAKE/LS-DYNA seismic response analyses performed using a lower-bound soil shear wave velocity profile (see Figure 2.I.6). This lower bound profile was established in [2.I.10] based on the geotechnical data of typical U.S. nuclear power plant sites. To develop the governing spectra, the input seismic acceleration time history for the SHAKE analysis is derived from the Regulatory Guide 1.60 seismic response spectrum and designated as the rock outcrop motion. The synthetic time history complies with the response spectrum and power density enveloping criteria in SRP 3.7.1 in NUREG-0800, Rev 2. The input acceleration time history is scaled to yield ground surface ZPAs (at the top of grade elevation) specified in Table 2.I.2. The average strain-compatible shear wave velocities of the soil column obtained from the SHAKE analysis are used to specify the minimum shear wave velocity values in Table 2.I.2. The ZPAs of the rock outcropping acceleration that yielded the governing spectra are 0.538 g's for the horizontal direction and 0.483 g's for the vertical direction.

The soil model for the subsequent LS-DYNA seismic response analysis uses the average strain-compatible wave velocities obtained from the SHAKE analysis (i.e., minimum shear wave velocity values in Table 2.I.2) to define the structural characteristics of the soil layers above and below the SFP elevation (see Figure 2.I.5 for sub-grade and under-grade space nomenclature). The acceleration time history at the soil column bottom surface, also obtained from the above-mentioned SHAKE analysis, is used as the input seismic motion for the LS-DYNA seismic response analysis performed in Supplement 3.I. The response spectrum plots shown in Figure 2.I.4 are the results of the LS-DYNA seismic response analysis (in the absence of the ISFSI). The same soil model and input seismic motion used in the LS-DYNA seismic response analysis is used for the LS-DYNA Soil-Structure Interaction (SSI) analysis (with the ISFSI included in the model) in Supplement 3.I.

The combination of weak soil properties and strong earthquake, as specified in Table 2.I.2 and Figure 2.I.4 for the structural evaluation of the underground ISFSI, has been selected to ensure that the Design Basis Earthquake response spectra at the ISFSI location will uniformly envelope those at most U.S. nuclear plants and that the Design Basis structural evaluation for the “100U” system is performed conservatively based on the lower bound support from the sub-grade and the under-grade. Thus, the HI-STORM 100U system can be deployed in most U.S. nuclear power plant sites without the need for a site-specific analysis to satisfy the requirements of 72.212. Specifically, a candidate 100U ISFSI site will be exempt from a detailed SSI analysis if the seismic response analysis for the site (using SHAKE or similar program) can demonstrate that the following two criteria are met:

1. The site’s response spectra at both TSP and SFP elevations are enveloped by the Design Basis Earthquake response spectra shown in Figures 2.I.4-A and 2.I.4-B, respectively;
2. The soil properties of the candidate site are greater than the minimum values specified in Table 2.I.2.

In order to satisfy the first criterion, the site must consider multiple time history sets as input to the seismic response analysis based on the guidelines set forth in SRP 3.7.1 [2.I.12] and ASCE 4-98 [2.I.11]. The site’s response spectra at both the TSP and SFP elevations must be bounded by the Design Basis Earthquake response spectra in Figure 2.I.4 for all acceleration time histories sets used as input.

For the case where only one of the above two criteria is not satisfied, a site-specific evaluation under 10CFR72.212 is permitted. Typical scenarios that warrant a site specific evaluation are discussed below:

Scenario A: The site’s response spectra are not completely enveloped by the Design Basis Earthquake response spectra in Figure 2.I.4. However, the site’s overall earthquake strength, represented by the resultant ZPA (see Table 2.I.2 for definition) is bounded by that of the Design Basis Earthquake at both TSP and SFP elevations.

While the ZPA represents the strength of the earthquake (in terms of the maximum value of the seismic acceleration time history), the shape of the seismic response spectrum is affected by many factors such as the overall stiffness of the site and the stiffness profile of soil layers.

Therefore, for the same input seismic time history at the base of the soil column, a stiffer site could have a peak response that is not enveloped by the Design Basis Earthquake response spectrum (as demonstrated in the SHAKE parametric study results presented in Table 2.I.4, where the only difference between the two analyzed cases is the stiffness (i.e., shear wave velocity) of the soil column). Although it is expected that the 100U system would exhibit a greater safety margin against the earthquake loading at the stiffer subgrade/undergrade site, a site-specific evaluation under 10CFR72.212 is the appropriate vehicle to confirm the structural integrity in this situation.

Scenario B: The strain compatible wave velocity of the soil in Space B and/or Space D of the ISFSI site (see Figure 2.I.5) is less than the required minimum value specified in Table 2.I.2.

Typically, Spaces B and D (in Figure 2.I.5) contain native soils whose properties are not affected by the ISFSI construction. More importantly, the loaded VVMs are not directly supported by the soil in the two spaces. Therefore, it is reasonable to assume that a small reduction of soil stiffness in these two spaces would not significantly modify the structural response of the VVM system. Structural compliance through a site specific analysis is assured if the ZPA of the DBE is well below the Design Basis value set down in this FSAR (Figure 2.I.4).

The site-specific safety analysis, if performed, shall follow the methodology set down in Supplement 3.I. In addition, since the soil and rock configuration varies from site to site, the total depth of the soil model for site-specific analysis shall be determined following the guideline in Section 3.3.3.2 of ASCE 4-98 [2.I.11]. Uncertainties in SSI analysis for a candidate 100U ISFSI site shall be accounted for by varying the best estimate low strain shear modulus of the substrates between the best estimate values times  $(1+c)$  and the best estimate value divided by  $(1+c)$ . If sufficient, adequate soil investigation data is available, the mean and standard deviation of the low strain shear modulus shall be established for every soil layer. The value of  $c$  may be established so that it will cover the mean plus or minus one standard deviation for every layer; however, the minimum value for  $c$  shall be no less than 0.5. If sufficient data is not available to determine a statistically meaningful mean and standard deviation, then the value for  $c$  shall be no less than 1.0.

The qualification of the ISFSI under the system's DBE event involves the following safety determinations:

1. Compliance of the VVM components (Divider shell, CEC shell, etc.) to the applicable stress/deformation limits specified in Table 2.I.6.

2. Strength compliance of the ISFSI reinforced concrete structures under ACI -318(2005) load combinations listed in Table 2.I.11.

A candidate 100U ISFSI site that does not meet the requirement discussed above for seismic qualification shall not be allowed for the consideration of a 100U general license.

vi. Explosion

The HI-STORM 100U System must withstand the pressure pulse due to a design basis explosion event. The effect of overpressure due to an explosion near the VVM is evaluated. The overpressure design value applied to the Closure Lid outer shell surface is intended to bound all credible explosion events because no combustible material is permitted to be stored near the VVM, and all materials of construction are engineered to be compatible with the operating environment. However, site-specific explosion scenarios that are not evidently bounded by the design basis explosion load considered herein (see Table 2.I.1) shall be evaluated under the provisions of 10CFR72.212.

vii. Lightning

The HI-STORM 100U System must withstand a lightning strike without a significant loss in its shielding capability. The effect of a lightning strike on the VVM is the same as that described for the aboveground overpack design, even though the likelihood of a lightning strike on the VVM is lower due to its low height above grade. Lightning is treated as an Extreme Environmental Phenomena event in Supplement 11.I. Because of its non-significant structural effect on the VVM, it is not considered as a load that warrants analysis in Supplement 3.I.

viii. Burial Under Debris

The burial under debris event for the HI-STORM 100U System is bounded by the evaluation performed for the aboveground overpacks, as discussed in Supplement 4.I.

ix. 100% Blockage of Air Inlets

The blockage of the entire inlet air flow area is analyzed as an accident event and is described in Supplement 11.I and analyzed in Supplement 4.I.

x. Extreme Environmental Temperature

An extremely high ambient air temperature is analyzed as an extreme environmental event and is described in Supplement 11.I and analyzed in Supplement 4.I.

xi. 100% Fuel Rod Rupture

This loading condition is specific to the MPC thermal evaluation and treated in Supplement 11.I.

2.I.7 Codes, Standards, and Practices to Ensure Regulatory Compliance

There is no U.S. or international code that is sufficiently comprehensive to provide a completely prescriptive set of requirements for the design, manufacturing, and structural qualification of the VVM. The various sections of the ASME Codes, however, contain a broad range of specifications that can be assembled to provide a complete set of requirements for the design, analysis, shop manufacturing, and field erection of the VVM. The portions of the ASME Codes that are invoked for the various elements of the VVM design, analysis, and manufacturing activities are summarized in Table 2.I.3.

The ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NF Class 3, 1995 Edition, with Addenda through 1997 [2.2.1], is the applicable code to determine stress limits for the metallic structural components of the VVM when required by the acceptance criteria listed in Table 2.I.5. Table 2.I.3 summarizes considerations for design, fabrication, materials, and inspection. The permitted material types and their permissible temperature limits for long-term use are listed in Table 2.I.8. Manufacturing requirements are set down in licensing and design drawings.

ACI-318(2005) [2.I.5] is the applicable reference code to establish applicable limits on unreinforced concrete (in the Closure Lid), which is subject to secondary structural loadings. Appendix 1.D contains the design, construction, and testing criteria applicable to the plain concrete in the VVM's Closure Lid. The load combinations applicable to the TSP, SFP, and the retaining wall, pursuant to ACI-318(2005) are summarized in Table 2.I.11. Since the VIP carries no load except for the self-weight and is thicker than the TSP, the structural evaluation of the VIP is not necessary. Applicable sections of ACI-318(2005) should be used in the design of the interfacing SSCs and proximate structures.

The selection of the ISFSI site shall be made with due consideration of the potential of liquefaction. The host plant's criteria with respect to liquefaction for siting the Part 50 structures shall be used.

As mandated by 10CFR72.24(c)(3) and §72.44(d), Holtec International's quality assurance program requires all constituent parts of an SSC subject to NRC's certification under 10CFR72 to be assigned an ITS category appropriate to its function in the control and confinement of radiation. The ITS designations for the constituent parts of the HI-STORM 100U VVM, using the guidelines of NUREG-CR/6407 [2.0.5], are provided in Table 2.I.8.

The aggregate of the citations from the codes, standards, and generally recognized industry publications invoked in this FSAR, supplemented by the commitments in Holtec's quality assurance procedures, provide the necessary technical framework to ensure that the as-installed VVM would meet the intent of §72.24(c), §72.120(a) and §72.236(b). As required by Holtec's QA Program



(discussed in Chapter 13), all operations on ITS components must be performed under QA validated written procedures and specifications that are in compliance with the governing citations of codes, standards, and practices set down in this FSAR. For activities that may be performed by others, such as site construction work to install the VVM, Holtec International requires that all activities be formalized in procedures and subject to the CoC holder's as well as the ISFSI owner's review and approval.

An ITS designation is also applied to the interfacing SSCs (such as the Support Foundation), which requires that all quality assurance measures set down in Holtec's Quality Assurance Procedure Manual be complied with by the entity performing the site construction work. In this manner, the compliance of the as-built VVMs with its engineered safety margins under all design basis scenarios of loading is assured.

#### 2.I.8 Service Limits

No new service limits are defined for the HI-STORM 100U System beyond those described in Subsection 2.2.5.

#### 2.I.9 Loads and Acceptance Criteria

Subsections 2.I.4, 2.I.5, and 2.I.6 describe the loadings for normal, off-normal, and extreme environmental phenomena and accident conditions, respectively, for the HI-STORM 100U System. Tables 2.I.1 and 2.I.2, respectively, provide the design loads and seismic load parameters in terms of ZPA values for a bounding analysis using the methodology of Subsection 3.I.4.7.

Bounding load cases that are significant to the structural performance of the VVM and require evaluation are compiled in Table 2.I.5 using information provided in Sections 2.I.4, 2.I.5, and 2.I.6. Supplement 3.I contains a description of the evaluations, establishes the evaluation methodology, and provides evaluation results that demonstrate compliance of the VVM to the applicable load cases and acceptance criteria described below. The load cases and acceptance criteria are explained in subsequent paragraphs and summarized in Table 2.I.5. Table 2.I.6 summarizes the acceptance criteria for the CEC and internals under extreme environmental events.

Each loading case in Table 2.I.5 is distinct in respect of the sub-component of the VVM that it affects most significantly. The acceptance criteria consist of demonstrating that (i) radiation shielding does not degrade under normal and off-normal conditions of storage loadings, (ii) the system does not deform under credible loading conditions in a manner that would jeopardize the subcritical condition or retrievability of the MPC, and (iii) the MPC maintains confinement. For accident condition loadings, any permissible degradation in shielding must be shown to result in dose rates sufficiently low to permit recovery of the MPC from the damaged cask, including unloading if necessary, and loss of function must be readily visible, apparent or detectable.

The above set of criteria, extracted from NUREG-1536, is further particularized in a more conservative form for each applicable loading case in this subsection.

### Load Case 01: Buoyant Force

This loading case pertains to the scenario wherein a VVM has been built, but the Closure Lid and MPC are not yet installed. Strictly speaking, this condition is not important to storage safety because the MPC is not present. However, considerations of long-term service life warrant that a minimum weight CEC, subject to the maximum buoyant force of water under an assumed hypothetical condition of submergence in water with a head equal to the length of the CEC, does not float. This evaluation sets a minimum additional weight (usually on a temporary cover) that will be set in place during construction to protect the CEC from construction debris, to provide for construction worker safety, and to insure that the CEC does not suffer uplift from buoyant forces. In addition, the Bottom Plate of the CEC must have sufficient flexural strength such that under a buoyant uplift pressure, its primary bending stress intensity remains below the ASME Level D allowable stress intensity at the reference metal temperature (assumed to be same as the extreme environmental condition temperature specified in Table 2.I.1 of this FSAR).

### Load Case 02: Dead Load plus Design Basis Explosion Pressure

The dead weight loading, explained in Paragraph 2.I.4(i) is accentuated by the design basis explosion loading defined in Paragraph 2.I.6(vi). The explosion load is stated in terms of an equivalent static pressure. The affected sub-components are:

- a. The Container Shell, subjected to a compressive state of stress under the combined effect of dead weight of the Closure Lid and surface pressure on the Closure Lid under the explosion event.
- b. The Closure Lid, subject to self-weight and the Closure Lid surface pressure under the explosion event.

Other VVM components are not in the direct path of this loading. The explosion pressure envelops other mechanical loads such as snow and flood. Load Case 02, therefore, is a bounding load combination that conservatively subsumes a number of normal and extreme environmental phenomena loads. As this load case is intended to bound any normal condition, Level A stress limits are applicable to this case based on reference metal temperatures that bound all mechanical loading scenarios.

### Load Case 03: Tornado Missile Impact

The Closure Lid is the only exposed portion of the VVM. Therefore, the tornado-borne missile strikes must be postulated to occur on the lid. The only other affected VVM part is the Container Flange, which prevents lateral sliding of the lid.

When subject to a tornado missile strike, the Closure Lid must not be dislodged, resulting in a direct line of sight from the top of the MPC to the outside. For the intermediate missile, the Closure Lid

must resist full penetration. Finally, any CEC deformation from the compressive axial impulse due to the missile strike must not prevent MPC retrievability.

#### Load Case 04: Design Basis Seismic Event

The Design Basis Seismic Event is classified as an extreme environmental phenomenon. As such the Level D service condition limits are applicable to the Code components, such as the MPC Enclosure Vessel. The MPC Enclosure Vessel and fuel basket have been qualified to a 60g deceleration limit in the HI-STAR 100 (Docket Nos. 72-1008, 71-9261); this deceleration exceeds the expected deceleration from a seismic event. However, to ensure an accurate structural evaluation of the VVM, the evaluation of the response of the VVM to the design basis seismic event shall include a detailed model of the MPC, the fuel basket, and the contained fuel; this model, referred to as the Design Basis Seismic Model, should capture impacts between the fuel and the fuel basket, between the fuel basket and the MPC, and between the MPC and applicable components of the VVM.

The CEC shell is subject to performance-based limits, which require that the deformation of the CEC does not prevent MPC retrievability, does not cause loss of MPC confinement, and that the system remains subcritical. This is accomplished by demonstrating that after the seismic event, permanent ovalization of the Container Shell and/or Divider Shell does not result in a geometry that precludes retrievability of the MPC and that the impact loadings on the MPC due to its rattling inside the CEC do not cause a breach of the MPC confinement boundary.

The Divider Shell's sole function is to direct the airflow inside the CEC cavity and to hold MPC Guides that serve to restrain the MPC from excessive rattling motion during an earthquake event. The MPC guides welded to the Divider Shell are subject to compressive impacts from the "hard points" on the MPC (the approximately 2.5-inch thick baseplate at the bottom and the 9.5-inch thick lid at the top). The MPC tubular guides are engineered to serve as "impact limiters" to minimize the local plastic strains in the MPC Confinement Boundary.

Finally, because the MPC Enclosure Vessel is designed to meet ASME Section III, Subsection "NB" (Class 1) stress intensity limits, and the earthquake is categorized as a "Level D" event, the primary stress intensities in the MPC Enclosure Vessel must meet Level D limits. The primary stress intensity in the MPC shell is the maximum longitudinal flexural stress intensity, which is compared against the primary membrane stress intensity limit for the material (Alloy X) at the applicable service temperature. The fuel basket is a multi-flange 3-D beam structure, designed to meet the stress limits of Subsection "NG" of the Code. The maximum longitudinal primary stress intensity in the basket, calculated from the 3-D fuel basket/fuel assembly model, must be less than the corresponding Level D condition limit at the service temperature. In addition to the primary stress based limits it is also necessary to demonstrate that the transverse bending stress in any panel normalized over the length of the fuel basket is less than the Level D primary stress limit.

The limits on the primary stresses in the MPC components for the DBE condition are also applicable to other Level D (faulted) events. Dynamic analysis using a 3-D detailed model of the MPC (which includes the Confinement Boundary, the internal fuel basket, and the fuel assemblies inside the

basket) is the vehicle for performing the structural qualification. In addition to the primary stress limits, the local strain in the Confinement Boundary due to the impact between the MPC and the MPC guides under the Design Basis Earthquake requires evaluation.

Table 2.I.5 summarizes the above discussion in tabular form.

#### Load Case 05: Closure Lid Handling

The Closure Lid lifting attachments are conservatively designed to meet the strength limits of ANSI N14.6 for heavy load handling. The metal load bearing parts shall satisfy the requirements of Reg. Guide 3.61 for primary stresses near the lifting locations and shall satisfy ASME NF Level A limits away from the lifting locations.

Yield and ultimate strength values used in the stress compliance demonstration per ANSIN14.6 shall utilize confirmed material test data through either independent coupon testing or material suppliers' CMTR or COC, as appropriate.

#### Load Case 06: Design Basis Fire Event

The exposed portion of the VVM, namely the Closure Lid, will experience the heat input and temperature rise under the fire event. The balance of the VVM, because of its underground location, will be subject to only a secondary temperature increase.

It is required to demonstrate that the structural collapse of the Closure Lid cannot occur due to the reduction of its structural material's (low carbon steel) strength at the elevated temperatures from the fire.

#### Load Case 07: CEC Loading From Surrounding Subgrade

The CEC is subject to a lateral pressure from the soil in the non-seismic condition. This pressure is affected by the presence of a loaded cask transporter adjacent to the CEC. The CEC must be shown to provide adequate resistance to this loading.

This load case tends to ovalize the CEC; the maximum primary membrane plus bending stress is limited to the material yield strength under normal conditions of storage.

In evaluating the structural safety margins in Supplement 3.I for the load cases described above, design data for the interfacing SSCs presented in Table 2.I.2 is used as applicable.

### 2.I.10 Safety Protection Systems

The HI-STORM 100U System, featuring the VVM with the stored MPC, provides for confinement, criticality control, and heat removal for the stored spent nuclear fuel in the manner of the aboveground overpacks. The VVM provides better shielding and protection from environmental

events, such as tornado missiles, because of its underground configuration. The information in Section 2.3 also applies to the HI-STORM 100U System, with the recognition that the air ventilation system is modified. Instead of the ambient air entering through inlet ducts at the bottom, the cooling air enters the circumferentially symmetric passage at the top of the VVM and is directed to the bottom of the VVM cavity along a radially symmetric annulus (Figure 1.I.4). However, the mechanism of heat transfer from the MPC to the cooling air is identical to the aboveground overpack designs.

The HI-STORM 100U System is completely passive requiring no active components or instrumentation to perform its design functions. Temperature monitoring or scheduled visual verification of the integrity of the air passages is used to verify continued operability of the VVM heat removal system, as set down in the system's Technical Specification.

#### 2.I.11 Decommissioning Considerations

The HI-STORM 100U VVM is specifically engineered to facilitate convenient decommissioning. As discussed in Supplement 1.I, the component most proximate to the active fuel and, hence, likely to be the most activated, is the Divider Shell. The Divider Shell is not welded to the CEC structure; therefore, it can be conveniently removed for decommissioning. The CEC structure can be removed by excavating the surrounding subgrade. Alternatively, the cavity can be filled with suitable fill materials and the CEC left in place. While the above discussion is unique to the VVM design, the information in Section 2.4 pertaining to decommissioning of other HI-STORM models is also applicable to the VVM. Even if the decision is made to dispose of all activated material, the VVM, due to differences in its geometry and construction (particularly, use of the native soil as the biological shield to the extent possible) will result in less steel and concrete to be disposed of. In the aggregate, it is estimated that less material will need to be disposed of to decommission a VVM ISFSI in comparison to an ISFSI containing aboveground overpacks.

Finally, the activation estimate in Table 2.4.1 for the aboveground overpack inner shell is conservatively applicable to the VVM steel shell enclosure.

#### 2.I.12 Regulatory Compliance

Pursuant to the guidance provided in NUREG-1536, the foregoing material in this supplement provides:

- i. a complete set of principal design criteria for the VVM as mandated by 10CFR72.24I(1), §72.24(c)(2), §72.120(a) and §72.236(b);
- ii. a clear identification of VVM structural parts subject to a fully articulated design subject to certification under 10CFR72 and of interfacing SSCs;
- iii. the required set of limiting critical characteristics of the interfacing SSCs to ensure that the VVM will render its intended function under all design basis scenarios of operation;

- iv. a complete set of requirements premised on well-recognized codes and standards to govern the design and analysis (to establish safety margins) and manufacturing of the VVM; and
- v. a table containing cross-reference between the applicable 10CFR72 requirements and the location in this FSAR where the fulfillment of each specific requirement is demonstrated.

It is noted that the requirements of 10CFR72 do not preclude the use of an underground storage system such as the HI-STORM 100U. The VVM concept, while not specifically mentioned in the regulatory guidance literature associated with implementing the requirements in 10CFR72 (i.e., NUREG-1536), meets and exceeds the intent of the guidance in that it provides an enhanced protection of the stored spent nuclear fuel and a significantly reduced site boundary dose, enables a more convenient handling operation, and presents a much smaller target for missiles/projectiles compared to an aboveground storage system.

### 2.I.13 References

The references in Section 2.6 apply to the VVM to the extent that they are appropriate for use with an underground system.

- [2.I.1] NACE Standard RP0104-2004 “The Use of Coupons for Cathodic Protection Monitoring Applications”, NACE International.
- [2.I.2] NACE Standard TM0101-2001 “Measurement Techniques Related to Criteria for Cathodic Protection on Underground or Submerged Metallic Tank Systems”, NACE International.
- [2.I.3] Federal Construction Council Technical Report No. 32, Cathodic Protection As Applied to Underground Metal Structures”, National Academy of Sciences – National Research Council, Publication 741, 1959.
- [2.I.4] Rabah, M.A., et al., "Electrochemical Wear of Graphite Anodes during Electrolysis of Brine," *Carbon*, Vol. 29, No. 2, pp. 165-171, 1991.
- [2.I.5] ACI-318(2005), Building Code Requirements for Structural Concrete (ACI 318-05) and Commentary (ACI 318R-05), Chapter 22, American Concrete Institute, 2005.
- [2.I.6] Holtec Position Paper, DS-338, “A Methodology to Compute the Equivalent Elastic Properties of the Subgrade Continuum to Incorporate the Effect of Long- Term Settlement,” A.I. Soler and C. Bullard (2010) (Holtec Proprietary)
- [2.I.7] Basic Soils Engineerings, B.H.Hough, Second Edition.

- [2.I.8] ACI 360R-06, Design of Slabs on Grade, American Concrete Institute, 2006.
- [2.I.9] “2009 International Building Code,” International Code Council, Inc.
- [2.I.10] NUREG/CR-6865, “Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Storage Systems,” U.S. Nuclear Regulatory Commission, February 2005.
- [2.I.11] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 2000.
- [2.I.12] NUREG-0800, SRP 3.7.1, “Seismic Design Parameters”, USNRC, Revision 3, March 2007.
- [2.I.13] N.M. Newmark, “Seismic Design Criteria for Structures and Facilities: Trans-Alaska Pipeline System,” proceedings of U.S. national Conference on Earthquake Engineering, Ann Arbor, Michigan, June 18-20, 1975.
- [2.I.14] USNRC Regulatory Guide 1.92, “Combining Modal Responses and Spatial Components in Seismic Responses Analysis,” Revision 2, July 2006.

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
<b>Life:</b>		
Design Life	40 yrs, Section 3.I.4	-
License Life	20 yrs, Section 3.I.4	10CFR72.42(a) & 10CFR72.236(g)
<b>Structural:</b>		
Design & Fabrication Codes: Foundation Pad; VVM Interface Pad and Top Surface Pad	ACI-318(2005)	10CFR 72.24
Unreinforced Concrete Stress Limits (Closure Lid)	Applicable Sections of ACI-318(2005)	10CFR72.24(c)(4)
Structural Steel	Section 2.I.7, Tables 2.I.5, 2.I.6	10CFR72.24(c)(4)
VVM Closure Lid Dead Weight <sup>†</sup> :	Table 3.I.1	R.G. 3.61
Design Internal Pressure	Atmospheric, Supplement 1.I	Ventilated Module
Response and Degradation Limits	Section 3.I.4	10CFR72.122(b), (c)
Corrosion Allowance	1/8" on surfaces directly in contact with subgrade	Standard industry practice
<b>Thermal:</b>		
Maximum Design Temperatures:		
Closure Lid Concrete		
Through-Thickness Section Average (Normal)	Table 1.D.1	ACI 349-85, Appendix A, (Paragraph A.4.3)
Through-Thickness Section Average (Off-Normal and Accident)	Table 1.D.1	ACI 349-85, Appendix A, (Paragraph A.4.2)
Structural Steel	Table 2.I.8	ASME Code, Section II, Part D
VVM Divider Shell Thermal Insulation	Heat transfer resistance $\geq 4$ hr-ft <sup>2</sup> -°F/Btu. Must be stable at temperatures $\leq 800^\circ\text{F}$	N/A
<b>Confinement:</b>		
	N/A, Provided by MPC; Supplement 7.I	10CFR72.128(a)(3) and 10CFR72.236(d) & (e)
<b>Retrievability:</b>		
Normal/Off-Normal/Accident	No damage that precludes MPC retrieval or threatens subcriticality of fuel. MPC maintains confinement, Supplement 3.I	10CFR72.122(f), (h), (1), & (l)

<sup>†</sup> All weights listed in Table 3.I.1 are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.



Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
<b>Criticality:</b>	N/A; Provided by MPC; Supplement 6.I	10CFR72.124 and 10CFR72.128(a)(2)
<b>Radiation Protection/Shielding:</b>		
Normal/Off-Normal	Provide capability to meet controlled area boundary dose limits under 10CFR72 for all normal and off-normal conditions; Supplement 5.I	10CFR72.104 and 10CFR72.212
	Ensure dose rates on and around the VVM during MPC transfer and lid installation operations are ALARA; Supplement 10.I	10CFR20
Accident or Conditions of Extreme Environmental Phenomena	Meet controlled area boundary dose limits in regulations for all accidents; Supplement 5.I	10CFR72.106
<b>Design Bases:</b>		
Spent Fuel Specification	Table 2.0.1; Section 2.I.1	10CFR72.236(a)
<b>Normal Design Event Conditions:</b>		
Ambient Outside Temperature:	-	-
Max. Yearly Average	80°F; Subsection 2.2.1.4	ANSI/ANS 57.9
Live Load <sup>†</sup> :		
Loaded HI-TRAC 125D and Mating Device	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Dry Loaded MPC	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Cask Transporter	Table 3.I.1, Subsection 2.I.9	-
<b>Handling:</b>	Subsection 2.I.4	-
VVM Closure Lid Lift Points	Subsection 3.I.4	NUREG-0612 Regulatory Guide 3.61
Minimum Temperature During Closure Lid Handling Operations	0°F; Subsection 2.2.1.2	ANSI/ANS 57.9
Snow and Ice Load	100 lb/ft <sup>2</sup> ; Subsection 2.I.4	ASCE 7-88
Wet/Dry Loading	Dry; Supplement 1.I, 8.I	-
Storage Orientation	Vertical; Supplement 1.I	-

<sup>†</sup> Weights listed in Table 3.I.1 are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
<b>Off-Normal Design Event Conditions:</b>		
Ambient Temperature:	Subsection 2.I.5	-
Minimum	-40°F; Subsection 2.2.2.2	ANSI/ANS 57.9
Maximum	100°F; Subsection 2.2.2.2	ANSI/ANS 57.9
Partial Blockage of Air Inlets	50% blockage of air inlet flow area; Supplement 4.I	-
<b>Design Basis Accident Events and Conditions:</b>		
Drop Cases:		
End Drop	Not credible; Subsection 2.I.6	In-ground VVM is not lifted
Tipover	Not credible; Subsection 2.I.6	In-ground VVM is constrained by subgrade and foundation
Fire:	-	-
Duration	217 seconds; Supplement 11.I	10CFR72.122(c)
Temperature	1475°F; Supplement 11.I	10CFR72.122(c)
Fuel Rod Rupture	See Table 2.0.1; Subsection 2.2.3.8	-
Air Flow Blockage	100% blockage of air inlet flow area; Subsection 2.I.6	10CFR72.128(a)(4)
Explosive Overpressure External Differential Pressure	10 psi steady state; Subsection 2.I.6 and Table 2.2.1	10CFR72.128(a)(4)
<b>Extreme Environmental Phenomenon Events and Conditions:</b>		
Flood:	Subsection 2.I.6	-
Height	125 ft	R.G. 1.59
Velocity	N/A; Supplement 1.I	In-ground VVM is not subject to tipover or sliding. Loads on the Closure Lid are bounded by missile impact loads.
Max. Earthquake	Table 2.I.2, Figure 2.I.4	10CFR72.102(f)
Tornado:	Subsection 2.I.6	-

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Tornado-Borne Missiles:		
i. Automobile	Ensure shielding, subcriticality and retrievability MPC maintains confinement Subsection 2.I.6 and Supplement 3.I	NUREG-1536
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
ii. Rigid Solid Steel Cylinder (intermediate tornado missile)	Ensure shielding, subcriticality and retrievability, MPC maintains confinement	NUREG-1536
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
iii. Steel Sphere	Subsection 2.I.6	NUREG-1536 In-ground VVM has no penetrations that provide line-of-sight to MPC
▪ Weight	Table 2.2.5	NUREG-0800
▪ Velocity	Table 2.2.5	NUREG-0800
Burial Under Debris	Maximum decay heat load and adiabatic heat-up; Subsection 2.I.6	-
Lightning	Bounded by aboveground evaluation (resistance heat-up); Subsection 2.I.6	In-ground VVM contains less metal
Extreme Environmental Temp.	125°F; Subsection 2.I.6 and Table 2.2.2	-
<b>Load Cases for Structural Qualification:</b>	Subsection 2.I.9 and Table 2.I.5	ANSI/ANS 57.9 and NUREG-1536

**TABLE 2.I.2  
DESIGN DATA FOR HI-STORM 100U ISFSI**

	<b>Item</b>	<b>Value(Minimum or nominal, as applicable)</b>	<b>Comment</b>
1.	Support Foundation Pad, VVM Interface Pad and Top Support Pad, and Retaining Wall	<ul style="list-style-type: none"> <li>▪ Minimum Concrete density = 145 lb/ft<sup>3</sup></li> <li>▪ Minimum concrete compressive strength @ ≤ 28 days = 4,500 psi</li> <li>▪ Grade 60 Rebar - Minimum yield strength of rebar = 60,000 psi; rebar is #11@9" (each face, each direction)</li> <li>▪ Minimum concrete cover on rebar per section 7.7.1 of ACI-318(05)</li> </ul>	See Licensing Drawings in Section 1.I.5 for detailed concrete pad/wall thickness.
2.	Depth averaged density of subgrade in Space A (see Figure 2.I.5), lb/ft <sup>3</sup>	120	A lower average density value may be used in shielding analysis in Supplement 5.I for conservatism.
3.	Depth averaged density of subgrade in Space B (see Figure 2.I.5), lb/ft <sup>3</sup>	110	A lower average density value may be used in shielding analysis in Supplement 5.I for conservatism.
4.	Depth depth averaged density of subgrade in Space C (see Figure 2.I.5), lb/ft <sup>3</sup>	120	Not required for shielding.
5.	Depth depth averaged density of subgrade in Space D (see Figure 2.I.5), lb/ft <sup>3</sup>	120	This space will typically contain native soil. Not required for shielding.
6.	Lower bound, strain compatible effective shear wave velocity in Space A, V ft/sec (see Notes 1 and 2)	500	This space will typically contain engineered fill.
7.	Lower bound, strain compatible effective shear wave velocity in Space B, V ft/sec (see Notes 1 and 2)	450	This space will typically contain native soil.
8.	Lower bound, strain compatible effective shear wave velocity in Space C, V ft/sec (see Notes 1 and 2)	485	This space may be remediated with vertical reinforcement such as pilings to enhance V.

**TABLE 2.I.2 (continued)**  
**DESIGN DATA FOR HI-STORM 100U ISFSI**

	<b>Item</b>	<b>Value</b>	<b>Comment</b>
9.	Lower bound, strain compatible effective shear wave velocity in Space D, $V$ ft/sec (see Notes 1 and 2)	485	This space will typically contain native soil.
10.	Design Basis Earthquake	<p>Ground surface spectra per Figure 2.I.4-A with horizontal ZPA, <math>a_H</math> and vertical ZPA, <math>a_V</math> as:</p> <p style="text-align: center;"><math>a_H = 1.0g</math>  <math>a_V = 0.75g</math></p> <p>and foundation surface spectra per Figure 2.I.4-B.</p>	<p>Horizontal and vertical spectra shown in Figures 2.I.4-A and 2.I.4-B are based on 5% damping.</p> <p>Following the Newmark 100-40-40 response combination technique [2.I.13] endorsed by the Regulatory Guide 1.92 [2.I.14], the <i>resultant ZPA</i> for a 3-D earthquake site is defined as: <math>a_R = a_1 + 0.4a_2 + 0.4a_3</math>, where <math>a_1</math>, <math>a_2</math> and <math>a_3</math> are the site's ZPAs in three orthogonal directions and <math>a_1 \geq a_2 \geq a_3</math>.</p> <p>Hence, the DBE <i>resultant ZPAs</i> at ground surface and foundation surface elevations are <math>1.3 \text{ g's}</math> (<math>=1.0 \times 1.0 \text{ g's} + 0.4 \times 0.75 \text{ g's}</math>) and <math>1.228 \text{ g's}</math> (<math>=1.0 \times 0.94 \text{ g's} + 0.4 \times 0.72 \text{ g's}</math>), respectively.</p>
11.	Maximum permissible long-term settlement of the SFP	0.2 inches	
12.	Maximum permissible long-term settlement of the TSP with respect to the SFP	0.4 inches	

**TABLE 2.I.2 (continued)**  
**DESIGN DATA FOR HI-STORM 100U ISFSI**

**Note 1:**

Strain compatible shear wave velocities in each space at an ISFSI site (see Figure 2.I.5) shall be computed using the guidance provided in Section 16 of the International Building Code, 2009 Edition [2.I.9]. The equivalent wave velocity is defined so that the wave transit time for an equivalent homogeneous material of the same total depth is the same as the actual layered substrate.

$$V = \frac{d}{\sum \frac{d_i}{v_i}}$$

$d_i$  = thickness of  $i^{\text{th}}$  layer within the region (ft.);

$v_i$  = strain compatible shear wave velocity of  $i^{\text{th}}$  layer within the region (ft./sec.);

$d$  = total thickness of substrate region (e.g. 20', 80')

$V$  = Equivalent Strain Compatible Shear Wave Velocity for substrate thickness "d".

**Note 2:**

The lower bound, strain compatible effective shear wave velocities at a particular site must account for the potential variability (i.e., uncertainty) in the site soil properties in accordance with Section 3.7.2 of NUREG-0800. This means that a site must demonstrate that, when the lower bound values for shear wave velocity (based on the site soil investigation data) are used as input, the free-field site response analysis yields a strain compatible effective shear wave velocity greater than the minimum value provided in this table for each space. The lower bound shear wave velocities used as input to the free-field site response analysis shall be determined using the following formula:

$$V_{LB} = \frac{V_{BE}}{\sqrt{1 + COV}}$$

where  $V_{BE}$  is the best estimate shear wave velocity for a given soil layer based on the soil investigation data, and  $COV$  is the coefficient of variation for the site soil properties. For well-investigated sites, the  $COV$  should be no less than 0.5. For sites that are not well investigated, the  $COV$  shall be set equal to 1.0.

**TABLE 2.I.3**  
**REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS**

	<b>Item</b>	<b>Code Paragraph<sup>†</sup></b>	<b>Explanation and Applicability</b>
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The “intervening elements” are termed interfacing SSCs in this FSAR.
3.	Certification of material	NF-2130(b) and (c)	Materials shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2400	-
6.	Welding procedure	Section IX	-
7.	Welding material	Section II	-
8.	Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3424	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 plate and shell supports and for linear supports are applicable for Closure Lid and Container Shell, respectively.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	Applies to the Divider Shell and Container Shell
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Storage of Welding Materials	NF-4411	-
19.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural welds
20.	Backing Strips, Peening	NF-4421 NF-4422	Applies to structural and non-structural welds
21.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
22.	Non-Destructive Examination	NF-5360	Invokes Section V
23.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	-

<sup>†</sup> All references to the ASME Code refer to applicable sections of the 1995 edition with addenda through 1997.

**TABLE 2.I.4**  
**SHAKE PARAMETRIC STUDY OF THE EFFECT OF SUBGRADE PROPERTIES**  
**ON SOIL RESPONSES AT 100U ISFSI TOP & BOTTOM ELEVATIONS**

Elevation & Direction	Acceleration Response	Value (g's)	
		Lower Bound Shear Wave Velocity Profile (see Figure 2.I.6)	Upper Bound Shear Wave Velocity Profile (see Figure 2.I.6)
TSP Top Surface Horizontal Direction	ZPA	1.008	0.897
	Peak	3.851	4.040
SFP Bottom Surface Horizontal Direction	ZPA	0.945	0.790
	Peak	3.590	3.848
TSP Top Surface Vertical Direction	ZPA	0.751	0.539
	Peak	3.912	2.377
SFP Bottom Surface Vertical Direction	ZPA	0.724	0.523
	Peak	3.674	2.314



**TABLE 2.I.5**  
**LOAD CASES AND ACCEPTANCE CRITERION APPLICABLE TO VVM**  
**COMPONENTS**

Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data		Acceptance Criterion
			Magnitude of Loading (Ref. Table I.D.)	Value of Coincident Metal Temperature used (Deg. F)	
01	Condition with no MPC or Closure Lid installed; buoyancy from a water head equal to the distance between TOG and TOF.	• Temporary Cover	Buoyant Force From CEC Displaced Volume	125	The minimum weight of the anti-buoyancy cover is 16,000lb.
		• CEC Bottom Plate	< 8 psi	125	Maximum primary bending stress intensity in the CEC Bottom Plate must be below Level D limit.
02	Normal operation condition; dead load plus design basis explosion pressure	• Container Shell structure	2.I.1; 3.I.1	125	Primary stresses do not exceed applicable Level A stress limits of ASME Subsection NF (or Level D limits with explosion)
		• Closure Lid	2.I.1	350	
03	Design basis missile	Closure Lid	2.I.1 and 2.2.5	350	Closure Lid does not collapse, is not dislodged from the cavity, and is not perforated by the missile.
04	Design basis earthquake	Container Shell	Figure 2.I.6	125	After the DBE event, MPC retrievability, subcriticality and confinement must not be compromised. Additional criteria for the CEC and its contents are defined in Table 2.I.6.
05	Closure lid handling	Lid Lift Lugs; all metal structure in Lid	1.15 x Closure Lid Weight (From Table 3.I.1)	125	NUREG-0612 and Regulatory Guide 3.61 limits based on yield or ultimate strength including magnified inertia loads. Meet Reg. Guide 3.61 and Level A limits as applicable. (see Section 2.I.9)

**TABLE 2.I.5 (continued)**  
**LOAD CASES AND ACCEPTANCE CRITERION APPLICABLE TO VVM**  
**COMPONENTS**

Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data		Acceptance Criterion
			Magnitude of Loading (Ref. Table I.D.)	Limiting Value of Coincident Metal Temperature (Deg. F)	
06	Design basis fire	Closure Lid	2.I.1	800	The Closure Lid structure does not collapse under its dead weight due to elevated metal temperatures.
07	CEC loading from subgrade	Container Shell	Calculated in 3.I	125	Service A stress limit for NF Class 3 plate and shell structure for the maximum "body extensive" membrane plus bending stress (body extensive defined as the region whose characteristic dimension exceeds $2.5 \sqrt{R \cdot T}$ ), where R and T are, respectively, the radius and thickness of the CEC shell.
<p>Note 1. Structural loads and acceptance criteria for each load case are further explained in Section 2.I.9.</p> <p>Note 2: Materials of construction are identified in Table 2.I.8.</p> <p>Note 3: Design attributes of the VVM are explained in Section 1.I.2 and details are presented in the drawings in Section 1.I.5.</p> <p>Note 4: The limiting value of coincident metal temperature is used to establish material properties and allowable stress (or stress intensity) when applicable.</p>					

**TABLE 2.I.6**  
**ACCEPTANCE CRITERIA FOR THE HI-STORM 100U VVM AND INTERNALS**  
**UNDER EXTREME ENVIRONMENTAL CONDITIONS**

Component	Calculated Value	Allowable Limit
CEC Container Shell and Divider Shell	Radial gap between CEC Shell and Divider Shell Insulation after the seismic event	Nominal Gap (based on OD of Divider Shell Insulation and ID of CEC Shell) must remain open at end of event.
MPC Guides	Maximum compressive load	Minimum of limiting buckling load or ultimate load
MPC Shell	Longitudinal flexural stress intensity in shell wall from bending of the MPC shell as a beam. The local true strain in the MPC shell in the region of MPC guide/MPC impact.	ASME Level D primary membrane stress intensity limit The local strain from impact must be less than 10%, which has been established as a conservative limit in [3.I.31]
MPC Fuel Basket	Primary flexural stress intensity in basket panel from bending of the fuel basket as a beam	ASME Level D primary membrane stress intensity limit
MPC Fuel Basket	Maximum transverse bending stress in most heavily loaded basket panel, averaged over the panel length	ASME Level D primary membrane + bending stress intensity limit

Table 2.I.7  
Intentionally Deleted

**TABLE 2.I.8**  
**PERMISSIBLE MATERIALS FOR HI-STORM 100U VVM COMPONENTS AND ISFSI STRUCTURES**

	Primary Function	Part	ITS Category	Material (note6)	Normal Storage (Long-Term Limit)	Max. Permissible Temperature (°F)		Interfacing Matl. (if dissimilar)
						Off-normal, extreme environmental phenomena, and accident conditions	Special Surface Finish/ Coating (note 1)	
1	Shielding	Closure Lid Concrete	C	Shielding Concrete per Appendix 1.D (note 2)	300 (note 3)	572 on local temperature of concrete except for fire 1100 for fire accidents (note 3)	NA	Steel
2	Shielding	Closure Lid Steel	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Concrete/Elastomer
3	Structural	CEC (Container Shell, Bottom Plate and Container Flange)	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Subgrade/Concrete
4	Thermal	Insulation	C	Commercial	800	800	NA	Steel
5	Thermal	Inlet/Outlet Vent Screens and associated hardware	NITS	Carbon steel, stainless steel, aluminum, a polymeric fabric capable of 400°F (min.) service temperature or commercial	800 (note 4) if all metallic 400 otherwise	800 (note 4) if all metallic 400 otherwise	(note 5)	variable
6	Thermal	Outlet Vent Cover and associated hardware	NITS	Carbon steel, stainless steel, aluminum or commercial	800 (note 4)	800 (note 4)	(note 5)	variable
7	Thermal	Divider Shell and Divider Shell Restraints	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	Insulation

**TABLE 2.I.8 (continued)**  
**PERMISSIBLE MATERIALS FOR HI-STORM 100U VVM COMPONENTS AND ISFSI STRUCTURES**

	Primary Function	Part	ITS Category	Material (note 6)	Max. Permissible Temperature (°F)		Special Surface Finish/Coating (note 1)	Interfacing Matl. (if dissimilar)
					Normal Storage (Long-Term Limit)	Off-normal, extreme environmental phenomena, and accident conditions		
8	Structural	Upper and Lower MPC Guides	C	ASTM A516, Gr. 70, A515 Gr. 70 or equivalent	800 (note 4)	800 (note 4)	(note 5)	-
9	Structural	MPC Bearing Pads	C	Carbon Steel (with stainless steel liners)	800 (note 4)	800 (note 4)	(note 5)	Stainless steel
10	Shielding and Physical Protection to the CEC	VVM Interface Pad (VIP)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	Steel
11	Shielding and Physical Protection	Top Surface Pad (TSP)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	—
12	Shielding and Physical Protection	Subgrade Surrounding the VVMs	B	Engineered fill, natural soil, or treated soil	150	350	N/A	Steel or Concrete
13	Structural Support	Support Foundation Pad (SFP)	C	Reinforced Concrete per ACI-318 (2005)	150	350	N/A	Soil, rock, mud mat, piling, etc., as appropriate
14	Shielding and Physical Protection	Retaining Wall (if used)	B	Reinforced Concrete Per ACI-318 (2005)	150	350	N/A	—
<p>Note 1 Materials identified by a supplier's trademark may be replaced with an equivalent product after an appropriate evaluation of acceptability.</p> <p>Note 2 All requirements are identical to the shielding concrete in aboveground HI-STORMs.</p> <p>Note 3 Limit per Appendix 1.D.</p> <p>Note 4 Permissible temperature limit from ASME Code, Section II, is used as guidance to define all long and short-term loading limits. The metal temperature limits do not apply to the fire event (see Subsection 2.I.6).</p> <p>Note 5 Surface preservative per Subsection 3.I.4.</p> <p>Note 6 Materials listed as "or equivalent" may be replaced with "equivalent materials" as defined in the glossary. The critical characteristics for these materials are given in Table 2.I.9.</p>								

**TABLE 2.I.9**  
**CRITICAL CHARACTERISTICS OF EQUIVALENT MATERIALS USED IN THE VVM COMPONENTS**

Designated Material	Item	Critical Characteristic
ASTM A515 or A516, Gr. 70	Yield Strength	Yield strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures. Applicable Code year is the same as used for the above ground HI-STORM.
	Ultimate Strength	Ultimate strength vs. Temperature data must exceed values from appropriate tables for 515/516 Gr.70 materials in ASME Code, Section II, Part D at all applicable temperatures. Applicable Code year is the same as used for the above ground HI-STORM.
	Elongation	Elongation must equal or exceed value(s) for 515/516 Gr. 70
	Charpy Impact	Values that measure resistance to impact must equal or exceed corresponding values for 515/516 Gr. 70.

**TABLE 2.I.10  
CRITICAL CHARACTERISTICS OF MATERIALS REQUIRED FOR SAFETY EVALUATION OF  
STORAGE AND TRANSPORT SYSTEMS**

	<b>Property</b>	<b>Type</b>	<b>Purpose</b>	<b>Bounding Acceptable Value</b>
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of $\delta_{min}$ , %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at ambient conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum allowable creep rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Thermal conductivity (minimum averaged value in the range of ambient to maximum service temperature, $t_{max}$ )	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
8.	Minimum Emissivity	T	To ensure that the thermal calculations are performed conservatively	Min.
9.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
10.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. and Max.
11.	Boron-10 Content	R	To control reactivity	Min.



**TABLE 2.I.11  
LOAD COMBINATIONS FOR THE TOP SURFACE PAD, VVM INTERFACE  
PADS, SUPPORT FOUNDATION PAD, AND THE RETAINING WALL PER  
ACI-318 (2005)**

Load Combination	
LC-1	1.4D
LC-2	1.2D + 1.6L
LC-3	1.2D + E + L

where:

D: Dead Load including long-term differential settlement effects.  
L: Live Load  
E: DBE for the Site

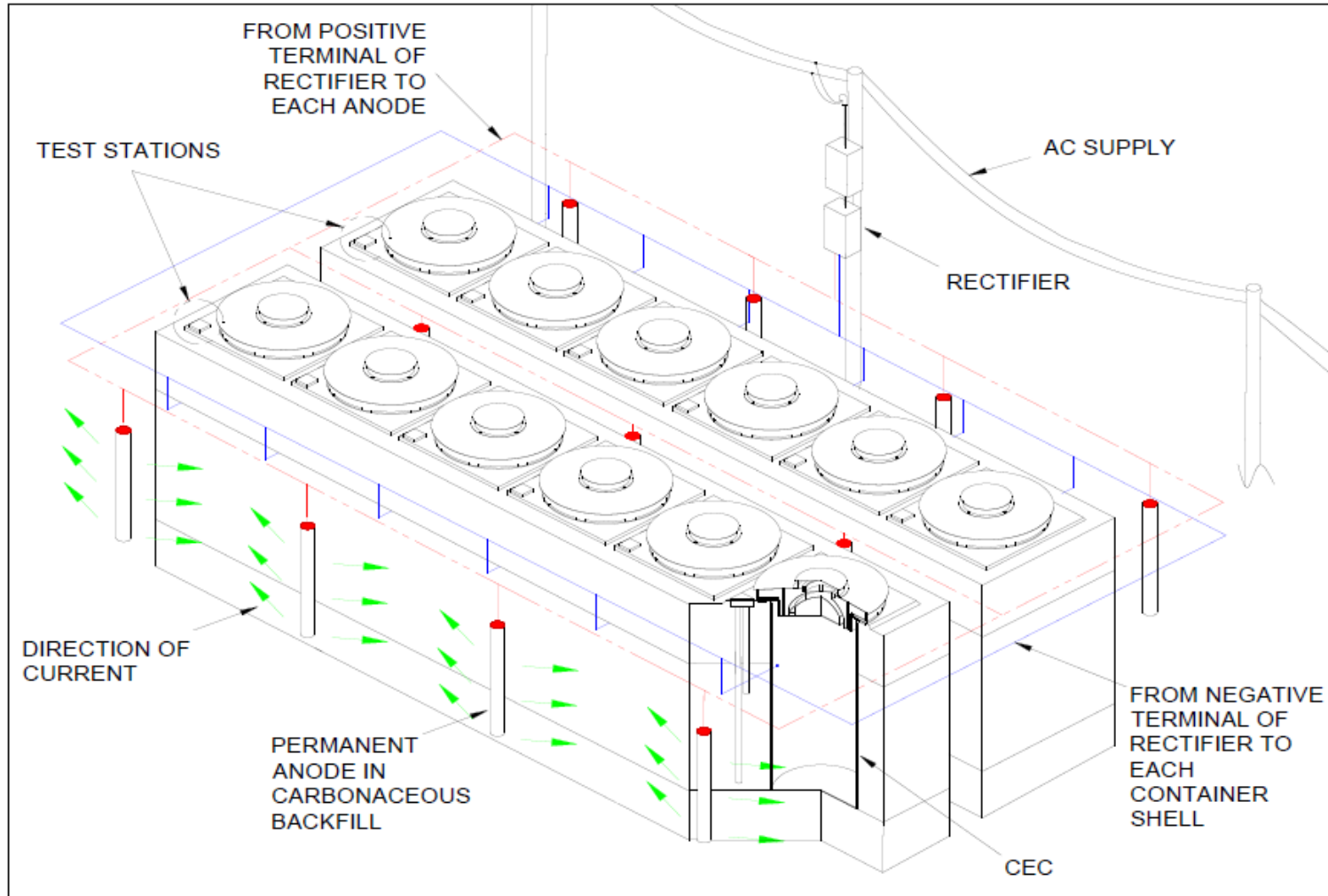


FIGURE 2.I.1: HI-STORM 100U SYSTEM EXAMPLE ICCPS DESIGN – 2 X 6 ARRAY DESIGN LAYOUT\*

\* The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Expansion joints between the VVM Interface Pad and the Top Surface Pad are not shown in this figure.

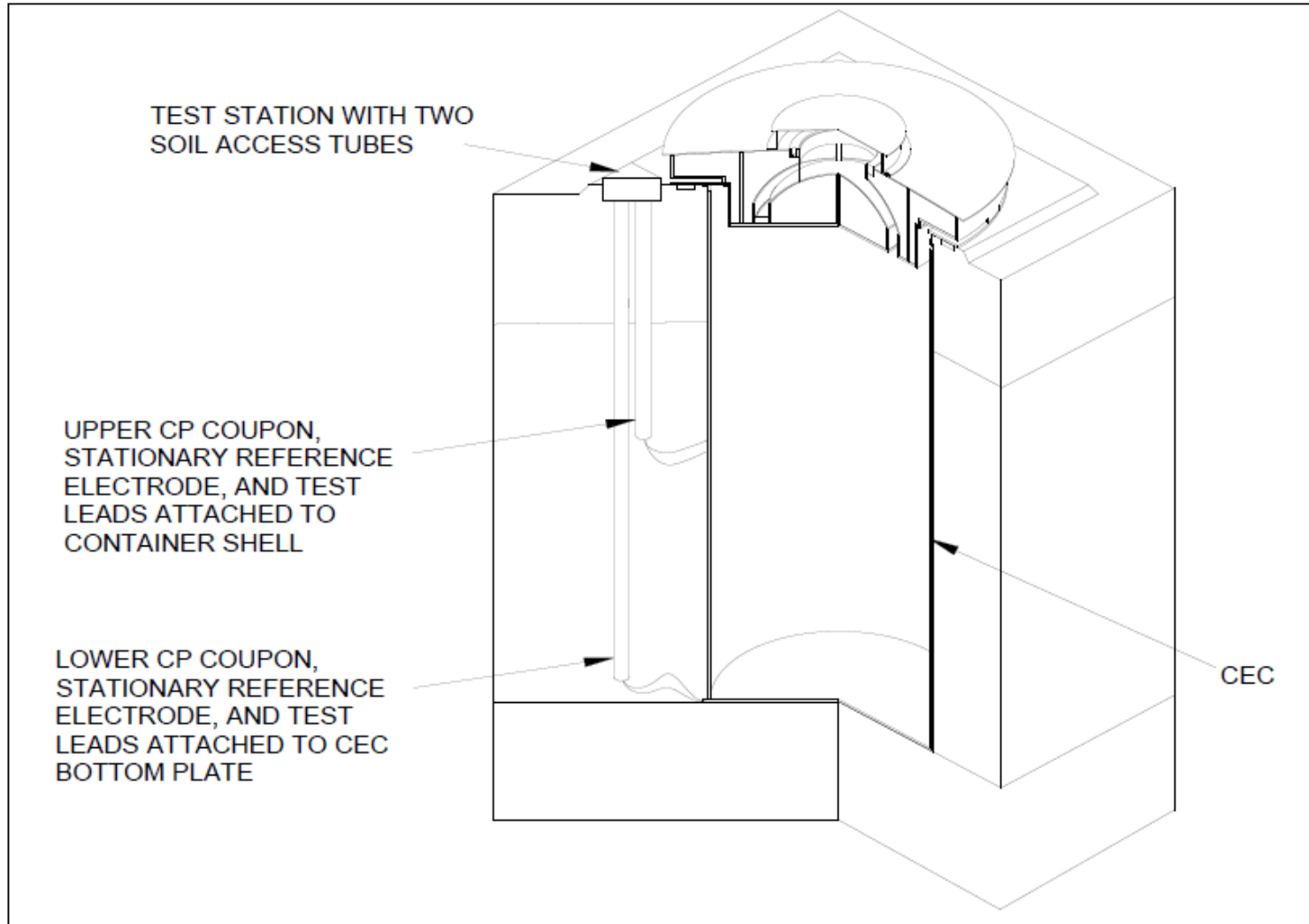


FIGURE 2.I.2: HI-STORM 100U SYSTEM EXAMPLE ICCPS DESIGN – TEST STATION\*

\*The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patent right laws. Expansion joints between VVM Interface Pad and Top Surface Pad are omitted from this figure.

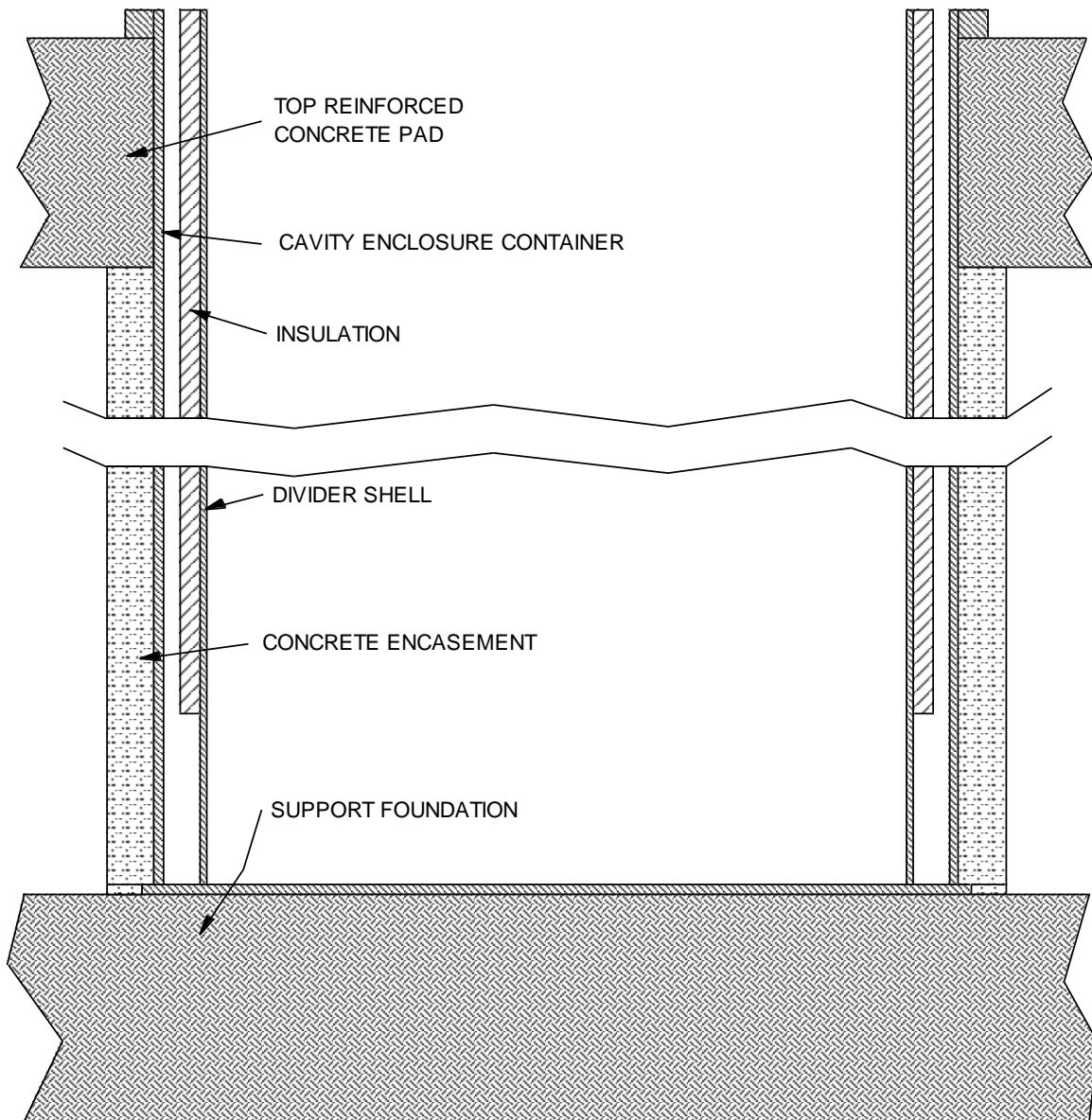
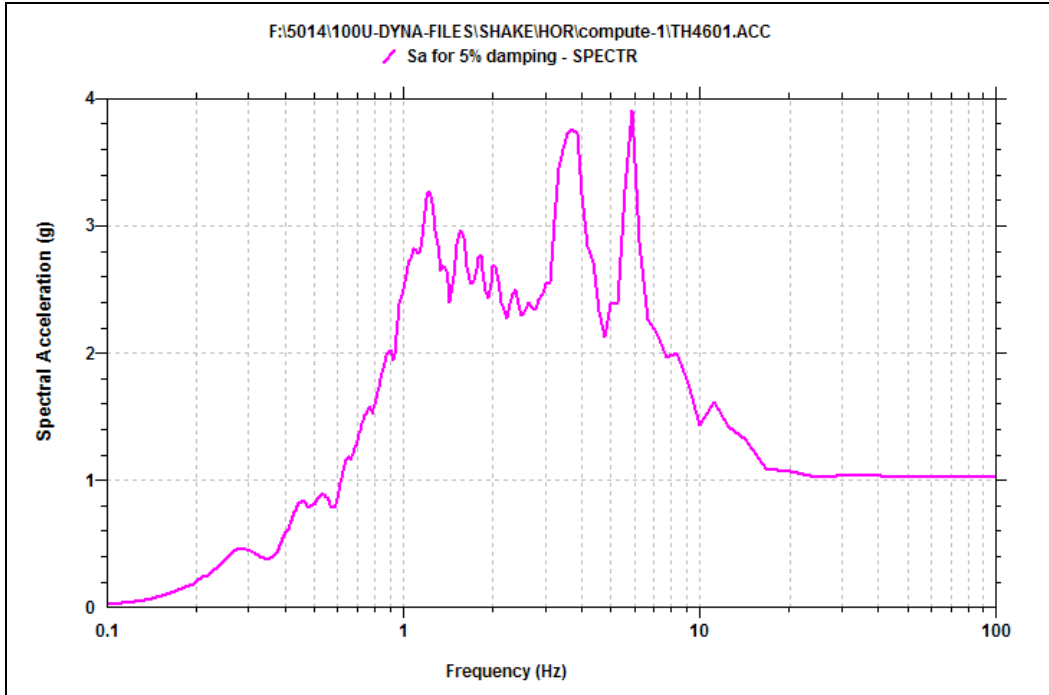
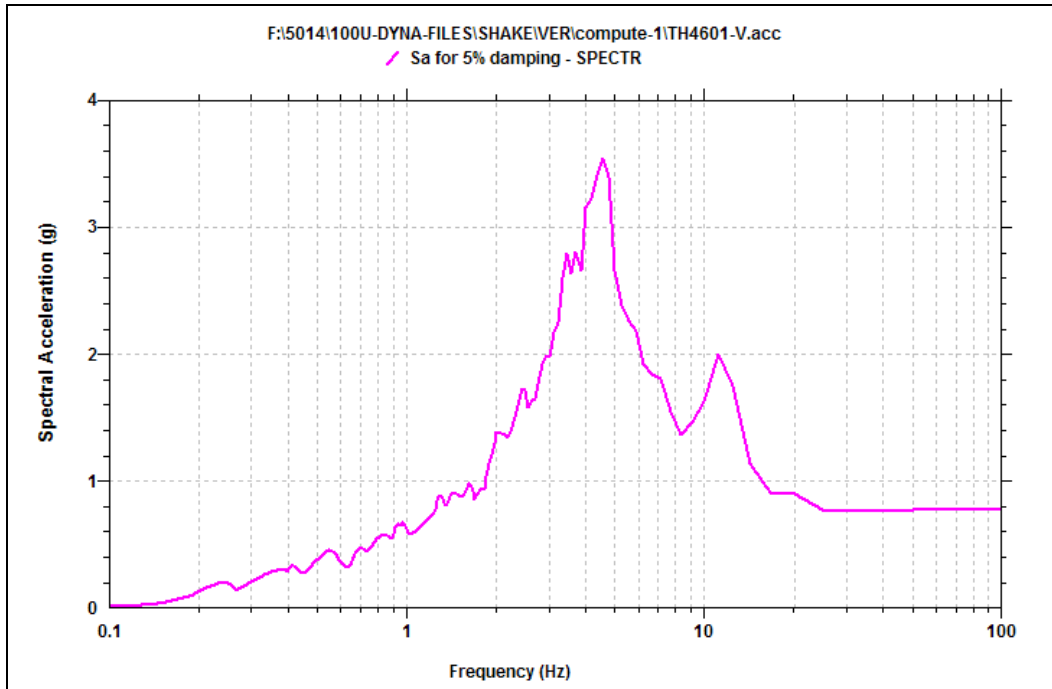


FIGURE 2.I.3: TYPICAL CONCRETE ENCASEMENT OF THE CEC



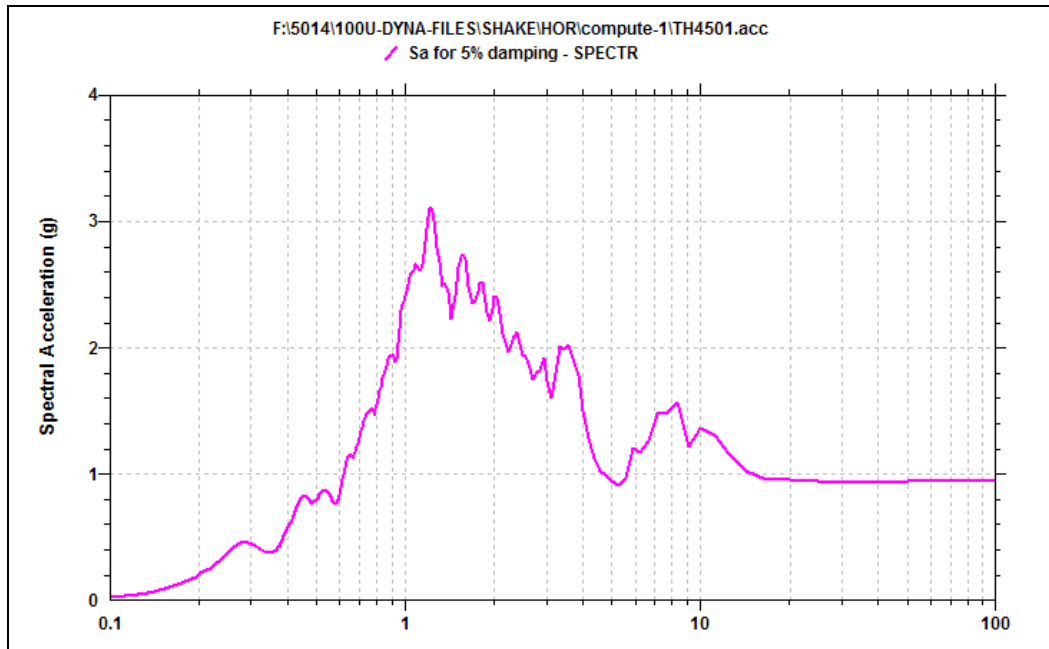
(a)



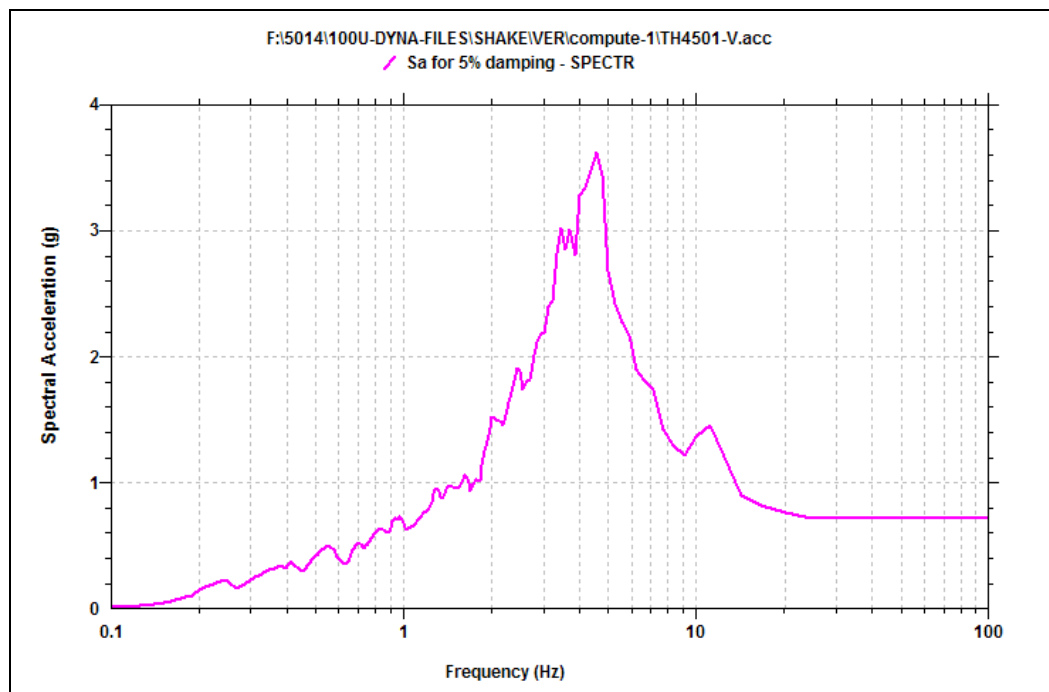
(b)

FIGURE 2.1.4-A: DESIGN BASIS SPECTRUM AT THE GROUND SURFACE (TOP OF TSP) ELEVATION

(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION



(a)



(b)

FIGURE 2.I.4-B: DESIGN BASIS SPECTRUM AT THE 100U FOUNDATION SURFACE  
(BOTTOM OF SFP) ELEVATION  
(a) HORIZONTAL DIRECTION; (b) VERTICAL DIRECTION

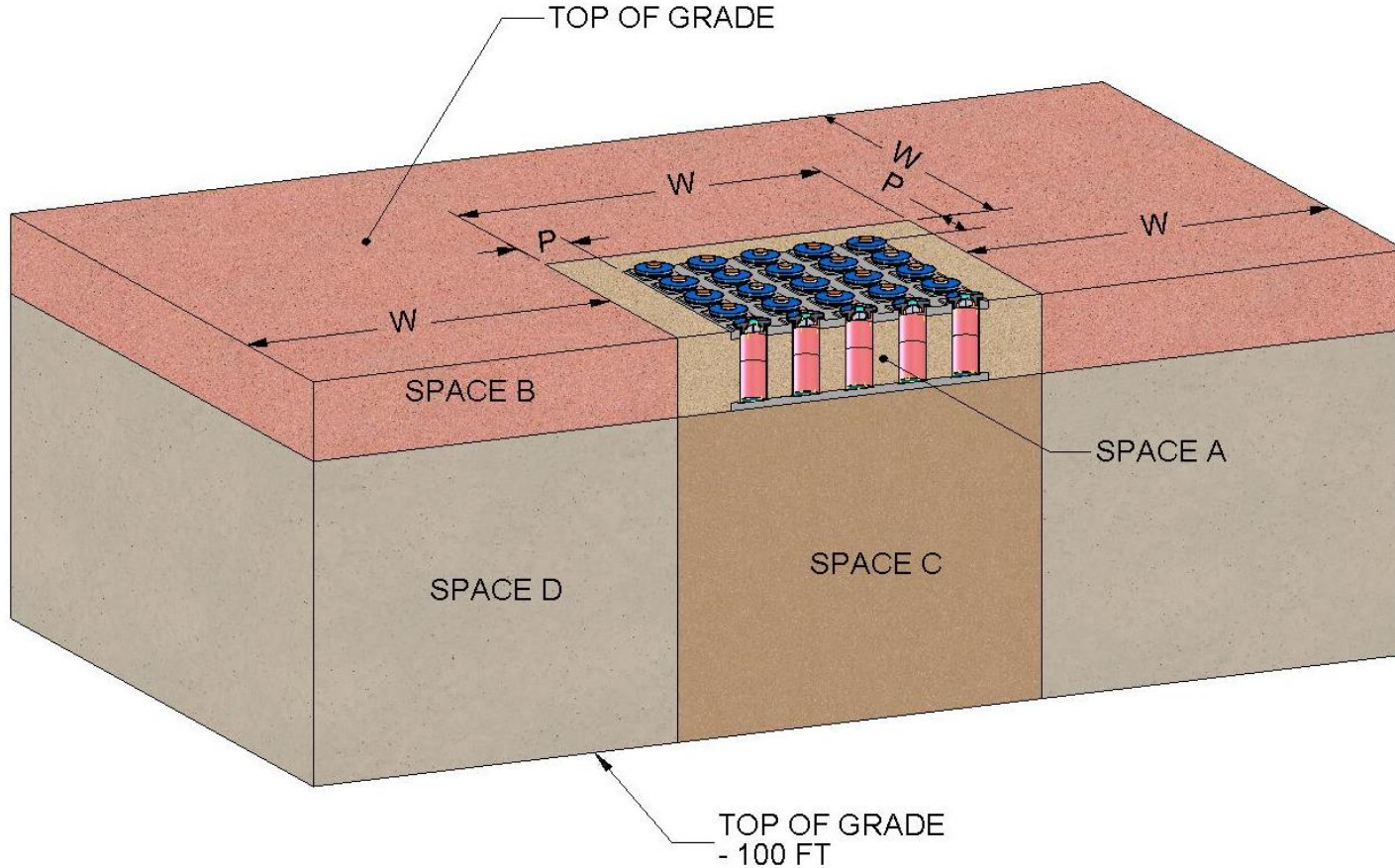


FIGURE 2.I.5 SUB-GRADE AND UNDER-GRADE SPACE NOMENCLATURE

Note: The figure shows a 5 by 5 array with a slice through the centerline of the first row of VVMs facing the reader. Space A is the lateral subgrade space in and around the VVMs which may be excavated and refilled with engineered fill. Space B is the lateral subgrade that extends by the amount  $W$  around the ISFSI where  $W$  is the characteristic dimension of the ISFSI. Space C is the undergrade below the SFP. Space D is the undergrade surrounding Space C.  $P$  is the distance to the Retaining wall.

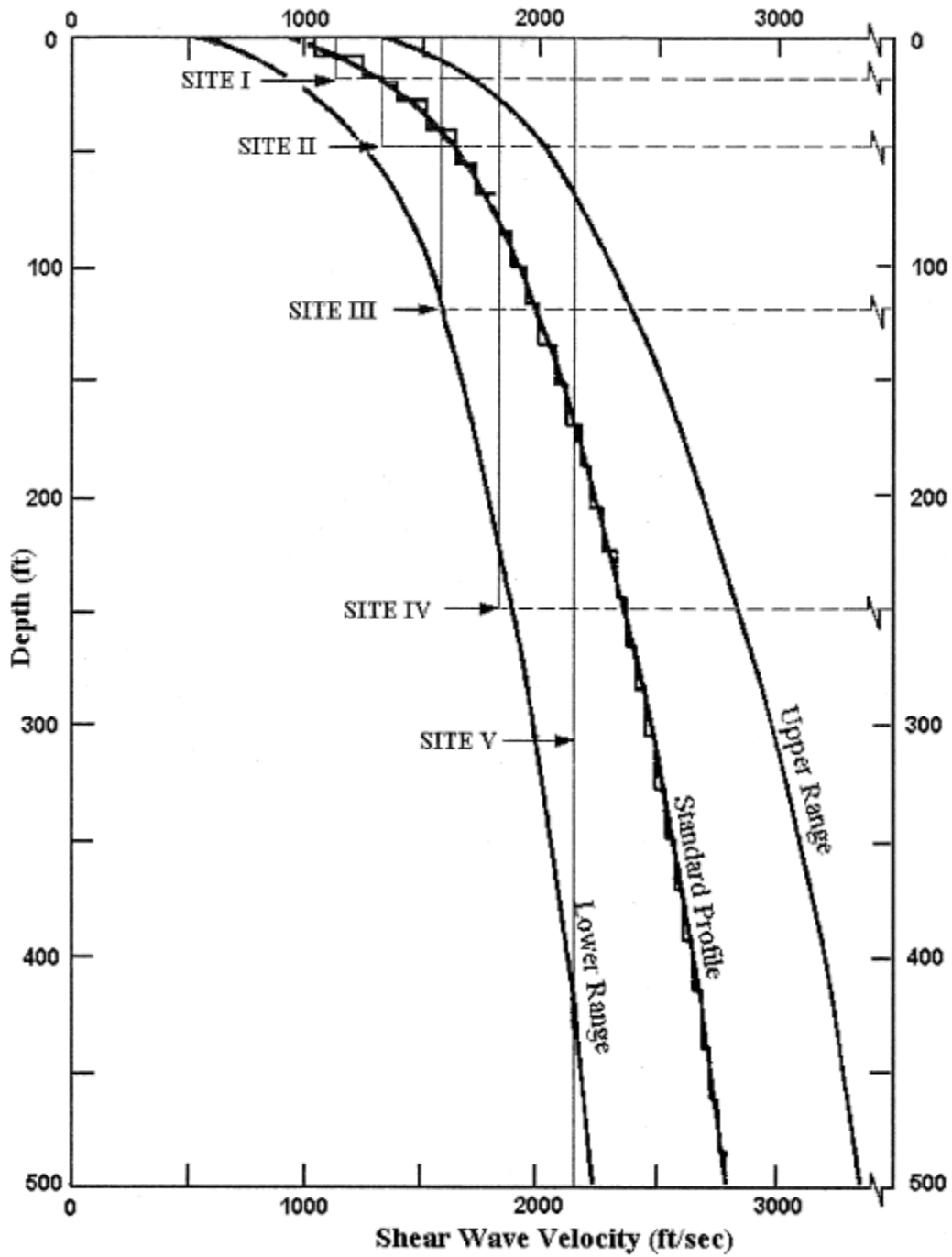


FIGURE 2.I.6 TYPICAL SHEAR WAVE VELOCITY PROFILES FOR NUCLEAR POWER PLANT SITES (REPRODUCED FROM FIGURE I-1 OF [2.I.10])



## SUPPLEMENT 2.III

### PRINCIPAL DESIGN CRITERIA FOR THE MPC-68M

#### 2.III.0 OVERVIEW OF THE PRINCIPAL DESIGN CRITERIA

A general description of the MPC-68M is provided in Supplement 1.III. This supplement specifies the loading conditions and associated design criteria applicable to the MPC-68M fuel basket. The loads, loading conditions, and design criteria presented in Chapter 2 are applicable to the HI-STORM 100 System using the MPC-68M unless otherwise specified in this supplement. The drawing package for the MPC-68M fuel basket is provided in Section 1.5. The safety classifications of Metamic-HT basket and basket shims are ITS-A and NITS, respectively.

The design criteria pertaining to the HI-STORM overpack and the HI-TRAC transfer cask are completely unaffected by the incorporation of the MPC-68M. Likewise, the structural demands on the MPC Enclosure Vessel (whose design remains unchanged) are also unaffected. The design criteria in this supplement pertain to the loading conditions that bear upon the fuel basket's function and performance.

#### 2.III.0.1 MPC-68M Design Criteria

##### i. Structural

The fuel basket is designed to meet a more stringent displacement limit under mechanical loadings than those implicit in the stress limits of the ASME code (see Section 2.III.2). The basket shims are designed to remain below the yield limit of the selected aluminum alloy. Fuel basket welds are designed and fabricated in accordance with Supplement 9.III and the drawing package in Section 1.5. Fuel basket structural welds are designed to the minimum weld strength specified in the drawing package in Section 1.5. Metamic-HT is a Holtec proprietary (non-ASME code) material. The *critical characteristics* and the attainment of the required critical characteristics through a comprehensive qualification process and production testing are discussed in Supplement 1.III with acceptance criteria established in Supplement 9.III.

All normal and off-normal conditions (including pressures) for the MPC-68M are the same as those described in Section 2.2. All loads on the HI-STORM 100 overpack and HI-TRAC transfer cask described in Section 2.2 remain applicable when using the MPC-68M.

The main acceptance criterion for the evaluation of accident conditions on the MPC-68M fuel basket is for the basket structure to maintain the fuel contents in a subcritical configuration. The structural design criteria for the MPC-68M basket are provided in Table 2.III.4.

The fuel basket material is subject to the requirements in Supplement 1.III and is designed to a specific (lateral) deformation limit of its walls under accident conditions of loading (credible and

non-mechanistic). The basis for the lateral deflection limit in the active fuel region,  $\theta$ , is provided in [2.III.6.1] as

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

where  $\delta$  is defined as the maximum total deflection sustained by the basket panels under the loading event and  $w$  is the nominal inside (width) dimension of the storage cell. The limiting value of  $\theta$  is provided in Table 2.III.4. The above deflection-based criterion has been used previously in the HI-STAR 180 Transportation Package [2.III.6.2] to qualify similar Metamic-HT fuel baskets.

ii. Thermal

The design and operation of the HI-STORM 100 System with the MPC-68M must meet the intent of the review guidance contained in ISG-11, Revision 3 [2.0.8] as described in Subsection 2.0.1.

All applicable material design temperature limits in Section 2.2 and 4.3 continue to apply to the MPC-68M. Temperature limits of MPC-68M fuel basket and basket shim materials are specified in Table 4.III.2.

The MPC-68M is designed for both uniform and regionalized fuel loading strategies as described in Subsection 2.0.1. The regions for the MPC-68M are given in Table 2.III.1. Additionally, the quarter-symmetric heat load pattern has been defined for MPC-68M as shown in Figure 2.III.1. The same temperature limits apply to this configuration.

iii. Shielding

Same as Subsection 2.0.1.

iv. Criticality

Same as Subsection 2.0.1 with the clarifications herein.

Criticality control is maintained by the geometric spacing of the fuel assemblies and spatially distributed B-10 isotope in the Metamic-HT. No soluble boron is required in the MPC-68M water. The minimum specified boron concentration in the Metamic-HT purchasing specification must be met in every lot of the material manufactured. No credit is taken for burnup. Enrichment limits are delineated in Table 2.III.2.

v. Confinement

Same as Subsection 2.0.1

vi. Operations

Same as Subsection 2.0.1. Generic operating procedures for the HI-STORM 100 System with MPC-68M are provided in Chapter 8 with certain limitations and clarifications provide in Supplement 8.III.

vii. Acceptance Tests and Maintenance

Same as Subsection 2.0.1. The acceptance criteria for the HI-STORM 100 System with MPC-68M are provided in Chapter 9 and Supplement 9.III.

vi. Decommissioning

The MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM 100 System is addressed in Section 2.III.4.

2.III.1 SPENT FUEL TO BE STORED

Table 2.1.22 and the limitations/clarification in this supplement provide the limits for material to be stored in the MPC-68M. All BWR fuel assembly array/classes which are authorized for the MPC-68 are authorized in the MPC-68M except fuel assembly array/classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D, and 10x10E. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for the fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt.% U-235 are authorized in the MPC-68M. The maximum planar-average initial enrichments acceptable for loading in the MPC-68M, for each fuel assembly array/class given in Table 2.1.4, are provided in Table 2.III.2. Table 2.III.3 provides the description of two new fuel assembly array/classes which are added as acceptable contents to the MPC-68M only, 10x10F and 10x10G. No credit is taken for fuel burnup or integral poisons such as gadolinia for any fuel assembly array/class. The maximum allowable initial enrichment for fuel assemblies are consistent with the criticality analysis described in Supplement 6.III.

Fuel classified as damaged fuel assemblies or fuel debris will be loaded into damaged fuel containers (DFCs) for storage in the MPC-68M. The appropriate thermal and criticality analyses have been performed to account for damaged fuel and fuel debris and are described in Supplements 4.III and 6.III, respectively. The restrictions on the number and location of damaged fuel containers authorized for loading in the MPC-68M is the same as MPC-68 (see Section 1.III.2.3 and Figure 1.III.2). Non-fuel hardware is not applicable to all the BWR fuel classes/arrays.

The heat generation rate, axial burnup distribution, and all other bounding radiological, thermal, and criticality parameters specified for MPC-68 are used to ensure the performance of the HI-STORM SYSTEM with the MPC-68M.

2.III.2 MPC-68M DESIGN LOADINGS

Design loadings in Section 2.2 apply to the HI-STORM 100 System using the MPC-68M.

2.III.3        SAFETY PROTECTION SYSTEMS

Same as Section 2.3.

2.III.4        DECOMMISSIONING CONSIDERATION

Same as Section 2.4.

2.III.5        REGULATORY COMPLIANCE

Same as Section 2.5.

2.III.6        REFERENCES

[2.III.6.1]     Holtec Proprietary Position Paper DS-331, “Structural Acceptance Criteria for the Metamic-HT Fuel Basket”, (USNRC Docket No. 71-9325).

[2.III.6.2]     HI-STAR 180 Transportation Package, USNRC Docket No. 71-9325.

Table 2.III.1: MPC-68M FUEL STORAGE REGIONS

Number of Storage Cells		Storage Cell IDs*	
Inner Region (n <sub>1</sub> )	Outer Region (n <sub>2</sub> )	Inner Region	Outer Region
32	36	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All other locations
* See Figure 1.III.2 for storage cell numbering			

Table 2.III.2: BWR FUEL ASSEMBLY INITIAL ENRICHMENTS  
FOR LOADING IN MPC-68M (Note 1)

<b>Fuel Assembly Array and Class</b>	<b>Maximum Planar-Average Initial Enrichment (wt.%<sup>235</sup>U) (Note 3, 4)</b>
<b>7x7 B</b>	4.8
<b>8x8 B</b>	4.8
<b>8x8 C</b>	4.8
<b>8x8 D</b>	4.8
<b>8x8 E</b>	4.8
<b>8x8 F</b>	4.5 (Note 2)
<b>9x9 A</b>	4.8
<b>9x9 B</b>	4.8
<b>9x9 C</b>	4.8
<b>9x9 D</b>	4.8
<b>9x9 E</b>	4.5 (Note 2)
<b>9x9 F</b>	4.5 (Note 2)
<b>9x9 G</b>	4.8
<b>10x10 A</b>	4.8
<b>10x10 B</b>	4.8
<b>10x10 C</b>	4.8

Notes:

1. All other fuel assembly array/class specifications from Table 2.1.4 apply.
2. Fuel assemblies classified as damaged fuel assemblies are limited to 4.0 wt.% U-235.
3. For MPC-68M loaded with both undamaged fuel assemblies and damaged fuel assemblies or fuel debris, the maximum planar average initial enrichment for the undamaged fuel assemblies is limited to the enrichment of the damaged assembly.
4. In accordance with the definition of undamaged fuel assembly, certain fuel assemblies may be limited to 3.3 wt.% U-235. When loading these fuel assemblies, all fuel assemblies in the MPC are limited to 3.3 wt.% U-235.

Table 2.III.3: BWR FUEL ASSEMBLY CHARACTERISTICS FOR LOADING IN MPC-68M (Note 1)

<b>Fuel Assembly Array and Class</b>	<b>10x10F</b>	<b>10x10G</b>
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 188
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 8, 9)	4.7 (Note 7)	4.75 (Note 7)
Initial Rod Maximum Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 4)	96/84
Fuel Clad O.D. (in.)	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150
No. of Water Rods (Note 6)	2	5 (Note 5)
Water Rod Thickness (in.)	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.120	≤ 0.060

Table 2.III.3 (continued)  
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. See Table glossary for the definition of “ZR.”
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users’ fuel records to account for manufacturer tolerances.
4. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
5. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
6. These rods may also be sealed at both ends and contain ZR material in lieu of water.
7. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% U-235 for the 10x10F and 10x10G arrays/classes.
8. For MPC-68M loaded with both undamaged fuel assemblies and damaged fuel assemblies or fuel debris, the maximum planar average initial enrichment for the undamaged fuel assemblies is limited to the enrichment of the damaged assembly.
9. In accordance with the definition of UNDAMAGED FUEL ASSEMBLY, certain assemblies may be limited to up to 3.3 wt.% U-235. When loading these fuel assemblies, all other undamaged fuel assemblies in the MPC are limited to enrichments specified in this table and Table 2.III.2.



Table 2.III.4	
STRUCTURAL DESIGN CRITERIA FOR THE FUEL BASKET	
PARAMETER	ALLOWABLE VALUE
Minimum service temperature	-40°F
Maximum total (lateral) deflection in the active fuel region, $\theta$ (dimensionless)	0.005

				1 0.5*	2 0.5*				
		3 0.5*	4 0.5	5 1.2	6 1.2	7 0.5	8 0.5*		
	9 0.5*	10 0.5	11 1.2	12 0.4	13 0.4	14 1.2	15 0.5	16 0.5*	
	17 0.5	18 1.2	19 0.4	20 0.4	21 0.4	22 0.4	23 1.2	24 0.5	
25 0.5*	26 1.2	27 0.4	28 0.4	29 0.4	30 0.4	31 0.4	32 0.4	33 1.2	34 0.5*
35 0.5*	36 1.2	37 0.4	38 0.4	39 0.4	40 0.4	41 0.4	42 0.4	43 1.2	44 0.5*
	45 0.5	46 1.2	47 0.4	48 0.4	49 0.4	50 0.4	51 1.2	52 0.5	
	53 0.5*	54 0.5	55 1.2	56 0.4	57 0.4	58 1.2	59 0.5	60 0.5*	
		61 0.5*	62 0.5	63 1.2	64 1.2	65 0.5	66 0.5*		
				67 0.5*	68 0.5*				

Cell ID
Heat Load (kW)

\* Note: This figure provides per cell allowable heat loads for MPC-68M with all UNDAMAGED FUEL assemblies. For MPC-68M with DAMAGED FUEL and/or FUEL DEBRIS stored in this location (in a DFC), the per cell allowable heat load of the cell is limited to 0.35 kW.

**Figure 2.III.1**  
**Per Cell Allowable Heat Loads (kW) for Quarter-Symmetric Pattern - MPC-68M**

## CHAPTER 3: STRUCTURAL EVALUATION<sup>†</sup>

### 3.0 OVERVIEW

In this chapter, the structural components of the HI-STORM 100 System that are important to safety (ITS) are identified and described. The objective of the structural analyses is to ensure that the integrity of the HI-STORM 100 System is maintained under all credible loads for normal, off-normal, and design basis accident/natural phenomena. The chapter results support the conclusion that the confinement, criticality control, radiation shielding, and retrievability criteria set forth by 10CFR72.236(l), 10CFR72.124(a), 10CFR72.104, 10CFR72.106, and 10CFR72.122(l) are met. In particular, the design basis information contained in the previous two chapters and in this chapter provides sufficient data to permit structural evaluations to demonstrate compliance with the requirements of 10CFR72.24. To facilitate regulatory review, the assumptions and conservatisms inherent in the analyses are identified along with a complete description of the analytical methods, models, and acceptance criteria. A summary of other material considerations, such as corrosion and material fracture toughness is also provided. Design calculations for the HI-TRAC transfer cask are included where appropriate to comply with the guidelines of NUREG-1536.

The organization of technical information in this chapter follows the format and content guidelines of USNRC Regulatory Guide 3.61 (February 1989). The FSAR ensures that the responses to the review requirements listed in NUREG-1536 (January 1997) are complete and comprehensive. The areas of NRC staff technical inquiries, with respect to structural evaluation in NUREG-1536, span a wide array of technical topics within and beyond the material in this chapter. To facilitate the staff's review to ascertain compliance with the stipulations of NUREG-1536, Table 3.0.1 "Matrix of NUREG-1536 Compliance - Structural Evaluation", is included in this chapter. A comprehensive cross-reference of the topical areas set forth in NUREG-1536, and the location of the required compliance information is contained in Table 3.0.1.

Section 3.7 describes in detail HI-STORM 100 System's compliance to NUREG-1536 Structural Evaluation Requirements.

The HI-STORM 100 System matrix of compliance table given in this section is developed with the supposition that the storage overpack is designated as a steel structure that falls within the purview of subsection 3.V.3 "Other Systems Components Important to Safety" (page 3-28 of NUREG-1536), and therefore, does not compel the use of reinforced concrete. (Refer to Table 1.0.2 for an explicit statement of exception on this matter). The concrete mass installed in the HI-STORM 100 overpack is accordingly equipped with "plain concrete" for which the sole applicable industry code is ACI 318.1 (92). Plain concrete, in contrast to reinforced concrete, is the preferred shielding material HI-STORM 100 because of three key considerations:

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

- (i) Plain concrete is more amenable to a void free pour than reinforced concrete in narrow annular spaces typical of ventilated vertical storage casks.
- (ii) The tensile strength bearing capacity of reinforced concrete is not required to buttress the steel weldment of the HI-STORM 100 overpack.
- (iii) The compression and bearing strength capacity of plain concrete is unaffected by the absence of rebars. A penalty factor, on the compression strength, pursuant to the provisions of ACI-318.1 is, nevertheless, applied to insure conservatism. However, while plain concrete is the chosen shielding embodiment for the HI-STORM 100 storage overpack, all necessary technical, procedural Q.C., and Q.A. provisions to insure nuclear grade quality will be implemented by utilizing the relevant sections from ACI-349 (85) as specified in Appendix 1.D.

In other words, guidelines of NUREG 1536 pertaining to reinforced concrete are considered to insure that the material specification, construction quality control and quality assurance of the shielding concrete comply with the provisions of ACI 349 (85). These specific compliance items are listed in the compliance matrix.

**TABLE 3.0.1**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
IV.1.a	ASME B&PV Compliance		
	NB	3.1.1	Tables 2.2.6,2.2.7
	NG	3.1.1	Tables 2.2.6,2.2.7
IV.2	Concrete Material Specification		Appendix 1.D
IV.4	Lifting Devices	3.1; 3.4	
V.	Identification of SSC that are ITS		Table 2.2.6
“	Applicable Codes/Standards	3.6.1	Table 2.2.6
“	Loads		Table 2.2.13
“	Load Combinations	3.1.2.1.2; Tables 3.1.1-3.1.5	Table 2.2.14
“	Summary of Safety Factors	3.4.3; 3.4.4.2; 3.4.4.3.1-3.4.6-3.4.9; Tables 3.4.3-3.4.9	
“	Design/Analysis Procedures	Chapter 3	
“	Structural Acceptance Criteria		Tables 2.2.10-2.2.12

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Material/QC/Fabrication	Table 3.4.2	Chap. 9; Chap. 13
“	Testing/In-Service Surveillance		Chap. 9; Chap. 12
“	Conditions for Use		Table 1.2.6; Chaps. 8,9,12
V.1.a	Description of SSC	3.1.1	1.2
V.1.b.i.(2)	Identification of Codes & Standards		Tables 2.2.6, 2.2.7
V.1.b.ii	Drawings/Figures		1.5
“	Identification of Confinement Boundary		1.5; 2.3.2; 7.1; Table 7.1.1
“	Boundary Weld Specifications	3.3.1.4	1.5; Table 7.1.2
“	Boundary Bolt Torque	NA	
“	Weights and C.G. Location	Tables 3.2.1-3.2.4	
“	Chemical/Galvanic Reactions	3.4.1; Table 3.4.2	
V.1.c	Material Properties	3.3; Tables 3.3.1-3.3.5	1.A; 1.C; 1.D
“	Allowable Strengths	Tables 3.1.6-3.1.17	Tables 2.2.10-2.2.12; 1.D
“	Suitability of Materials	3.3; Table 3.4.2	1.A; 1.B; 1.D
“	Corrosion	3.3	
“	Material Examination before Fabrication		9.1.1

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Material Testing and Analysis		9.1; Table 9.1.1; 1.D
“	Material Traceability		9.1.1
“	Material Long Term Performance	3.3; 3.4.11; 3.4.12	9.2
“	Materials Appropriate to Load Conditions		Chap. 1
“	Restrictions on Use		Chap. 12
“	Temperature Limits	Table 3.1.17	Table 2.2.3
“	Creep/Slump	3.4.4.3.3.2	
“	Brittle Fracture Considerations	3.1.2.3; Table 3.1.18	
“	Low Temperature Handling		2.2.1.2
V.1.d.i.(1)	Normal Load Conditions		2.2.1; Tables 2.2.13,2.2.14
“	Fatigue	3.1.2.4	
“	Internal Pressures/Temperatures for Hot and Cold Conditions	3.4.4.1	2.2.2; Tables 2.2.1,2.2.3
“	Required Evaluations		
“	Weight+Pressure	3.4.4.3.1.2	
“	Weight/Pressure/Temp.	3.4.4.3.1.2	
“	Free Thermal Expansion	3.4.4.2	4.4.6; Table 4.4.10

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
V.1.d.i.(2)	Off-Normal Conditions		2.2.2; Tables 2.2.13, 2.2.14; 11.1
V.1.d.i.(3)	Accident Level Events and Conditions	Tables 3.1.1, 3.1.2	2.2.3; Tables 2.2.13, 2.2.14; 11.2
V.1.d.i.(3).(a)	Storage Cask Vertical Drop	3.1.2.1.1.2; 3.4.10; 3.A	2.2.3.1
“	Storage Cask Tipover	3.1.2.1.1.1; 3.4.10; 3.A	2.2.3.2
“	Transfer Cask Horizontal Drop	3.4.9	2.2.3.1
V.1.d.i.(3).(b)	Explosive Overpressure	3.1.2.1.1.4	2.2.3.10
V.1.d.i.(3).(c)	Fire		
“	Structural Evaluations	3.4.4.2	2.2.3.3
“	Material Properties		11.2
“	Material Suitability	3.1.2.2; 3.3.1.1	Table 2.2.3; 11.2
V.1.d.i.(3).(d)	Flood		
“	Identification	3.1.2.1.1.3; 3.4.6	2.2.3.6
“	Cask Tipover	3.4.6	
“	Cask Sliding	3.4.6	
“	Hydrostatic Loading	3.1.2.1.1.3; 3.4.6	72-1008(3.H)
“	Consequences		11.2
V.1.d.i.(3).(e)	Tornado Winds		
“	Specification	3.1.2.1.1.5	2.2.3.5; Table 2.2.4
“	Drag Coefficients	3.4.8	
“	Load Combination	3.4.8	



**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Overturning –Transfer	NA	
V.1.d.i.(3).(f)	Tornado Missiles		
“	Missile Parameters	3.1.2.1.1.5	Table 2.2.5
“	Tipover	3.4.8	
“	Damage	3.4.8.1; 3.4.8.2	
“	Consequences	3.4.8.1; 3.4.8.2	11.2
V.1.d.i.(3).(g)	Earthquakes		
“	Definition of DBE	3.1.2.1.1.6; 3.4.7	2.2.3.7; Table 2.2.8
“	Sliding	3.4.7	
“	Overturning	3.4.7	
“	Structural Evaluations	3.4.7	11.2
V.1.d.i.(4).(a)	Lifting Analyses		
“	Trunnions		
“	Requirements	3.1.2.1.2; 3.4.3.1; 3.4.3.2	72-1008(3.4.3); 2.2.1.2
“	Analyses	3.4.3.1; 3.4.3.2	72-1008(3.4.3)
“	Other Lift Analyses	3.4.3.7-3.4.3.9	
V.1.d.i.(4).(b)	Fuel Basket		
“	Requirements	3.1.2.1.2; Table 3.1.3	
“	Specific Analyses	3.4.4.2; 3.4.4.3; 3.6.3	72-1008(3.4.4.3.1.2; 3.4.4.3.1.6; 3.M; 3.H; 3.I)

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Dynamic Amplifiers	3.4.4.4.1	
“	Stability	3.4.4.3; 3.4.4.4	72-1008(Figures 3.4.27-32)
V.1.d.i.(4).(c)	Confinement Closure Lid Bolts		
“	Pre-Torque	NA	
“	Analyses	NA	
“	Engagement Length	NA	
“	Miscellaneous Bolting		
“	Pre-Torque	3.4.3.7; 3.4.3.8	
“	Analyses	3.4.4.3.2.2	
“	Engagement Length	3.4.3.5; 3.4.3.7; 3.4.3.8	
V.1.d.i.(4)	Confinement		
“	Requirements	3.1.2.1.2; Table 3.1.4	Chap. 7
“	Specific Analyses	3.6.3; Tables 3.4.3, 3.4.4	72-1008(3.E; 3.K; 3.I)
“	Dynamic Amplifiers	3.4.4.1	
“	Stability	3.4.4.3.1	72-1008(3.H)
“	Overpack		
“	Requirements	3.1.2.1.2; Tables 3.1.1, 3.1.5	
“	Specific Analyses	3.6.3; 3.4.4.3	

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Dynamic Amplifiers	3.4.4.3.2	
“	Stability	3.4.4.3; Table 3.1.1; 3.4.4.5	
“	Transfer Cask		
“	Requirements	3.1.2.1.2; Table 3.1.5	
“	Specific Analyses	3.4.4.3; 3.6.3	
“	Dynamic Amplifiers	3.4.4.4.1	
“	Stability	NA	2.2.3.1

† Legend for Table 3.0.1

Per the nomenclature defined in Chapter 1, the first digit refers to the chapter number, the second digit is the section number within the chapter; an alphabetic character in the second place means it is an appendix to the chapter.

72-1008

HI-STAR 100 Docket Number where the referenced item is located

NA

Not Applicable for this item

### 3.1 STRUCTURAL DESIGN

#### 3.1.1 Discussion

The HI-STORM 100 System consists of three principal components: the Multi-Purpose Canister (MPC), the storage overpack, and the transfer cask. The MPC is a hermetically sealed, welded structure of cylindrical profile with flat ends and a honeycomb fuel basket. A complete description is provided in Subsection 1.2.1.1 wherein the anatomy of the MPC and its fabrication details are presented with the aid of figures. The MPCs utilized in the HI-STORM 100 System are identical to those for the HI-STAR 100 System submitted under Dockets 72-1008 and 71-9261. The evaluation of the MPCs presented herein draws upon the work described in those earlier submittals. In this section, the discussion is confined to characterizing and establishing the structural features of the MPC, the storage overpack, and the HI-TRAC transfer cask. Since a detailed discussion of the HI-STORM 100 Overpack and HI-TRAC transfer cask geometries is presented in Section 1.2, attention is focused here on structural capabilities and their inherent margins of safety for housing the MPC. Detailed design drawings for the HI-STORM 100 System are provided in Section 1.5.

The design of the MPC seeks to attain three objectives that are central to its functional adequacy, namely:

- **Ability to Dissipate Heat:** The thermal energy produced by the stored spent fuel must be transported to the outside surface of the MPC such that the prescribed temperature limits for the fuel cladding and for the fuel basket metal walls are not exceeded.
- **Ability to Withstand Large Impact Loads:** The MPC, with its payload of nuclear fuel, must be sufficiently robust to withstand large impact loads associated with the postulated handling accident events. Furthermore, the strength of the MPC must be sufficiently isotropic to meet structural requirements under a variety of handling and tip-over accidents.
- **Restraint of Free End Expansion:** The membrane and bending stresses produced by restraint of free-end expansion of the fuel basket are categorized as primary stresses. In view of the concentration of heat generation in the fuel basket, it is necessary to ensure that structural constraints to its external expansion do not exist.

Where the first two criteria call for extensive inter-cell connections, the last criterion requires the opposite. The design of the MPC seeks to realize all of the above three criteria in an optimal manner.

From the description presented in Chapter 1, the MPC enclosure vessel is the confinement vessel designed to meet ASME Code, Section III, Subsection NB stress limits. The enveloping canister shell, the baseplate, and the lid system form a complete confinement boundary for the stored fuel that is referred to as the "enclosure vessel". Within this cylindrical shell confinement vessel is an integrally welded assemblage of cells of square cross sectional openings for fuel storage, referred to herein as the fuel basket. The fuel basket is analyzed under the provisions of Subsection NG of Section III of the ASME Code. All multi-purpose canisters designed for deployment in the HI-

STORM 100 and HI-STAR 100 systems are exactly alike in their external dimensions. The essential difference between the MPCs lies in the fuel baskets. Each fuel storage MPC is designed to house fuel assemblies with different characteristics. Although all fuel baskets are configured to maximize structural ruggedness through extensive inter-cell connectivity, they are sufficiently dissimilar in structural details to warrant separate evaluations. Therefore, analyses for each of the MPC types were carried out to ensure structural compliance. Inasmuch as no new MPC designs are introduced in this application, and all MPC designs were previously reviewed by the USNRC under Docket 72-1008, the MPC analyses submitted under Docket Numbers 72-1008 and 71-9261 for the HI-STAR 100 System are not reproduced herein unless they need to be modified by HI-STORM 100 conditions or geometry differences. Analyses provided in the HI-STAR 100 System safety analysis reports that are applicable to the HI-STORM 100 System are referenced in this FSAR by docket number and subsection or appendix.

Components of the HI-STORM 100 System that are important to safety and their applicable design codes are defined in Chapter 2.

Some of the key structural functions of the MPC in the storage mode are:

1. To position the fuel in a subcritical configuration, and
2. To provide a confinement boundary.

Some of the key structural functions of the overpack in the storage mode are:

1. To serve as a missile barrier for the MPC,
2. To provide flow paths for natural convection,
3. To ensure stability of the HI-STORM 100 System, and
4. To maintain the position of the radiation shielding.
5. To allow movement of the overpack with a loaded MPC inside.

Some structural features of the MPCs that allow the system to perform these functions are summarized below:

- There are no gasketed ports or openings in the MPC. The MPC does not rely on any sealing arrangement except welding. The absence of any gasketed or flanged joints makes the MPC structure immune from joint leaks. The confinement boundary contains no valves or other pressure relief devices.

- The closure system for the MPCs consists of two components, namely, the MPC lid and the closure ring. The MPC lid can be either a single thick circular plate continuously welded to the MPC shell along its circumference or two dual lids welded around their common periphery. The MPC closure system is shown in the design drawings in Section 1.5. The MPC lid is equipped with vent and drain ports which are utilized for evacuating moisture and air from the MPC following fuel loading, and subsequent backfilling with an inert gas (helium) at a specified mass. The vent and drain ports are covered by a cover plate and welded before the closure ring is installed. The closure ring is a circular annular plate edge-welded to the MPC lid and shell. The two closure members are interconnected by welding around the inner diameter of the ring. Lift points for the MPC are provided in the MPC lid.
- The MPC fuel baskets consist of an array of interconnecting plates. The number of storage cells formed by this interconnection process varies depending on the type of fuel being stored. Basket designs containing cell configurations for PWR and BWR fuel have been designed and are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell. Welding of the basket plates along their edges essentially renders the fuel basket into a multiflange beam. Figure 3.1.1 provides an isometric illustration of a fuel basket for the MPC-68 design.
- The MPC basket is separated from its basket supports by a gap. The gap size decreases as a result of thermal expansion (depending on the magnitude of internal heat generation from the stored spent fuel). The provision of a small gap between the basket and the basket support structure is consistent with the natural thermal characteristics of the MPC. For MPCs that contain basket shims welded to the MPC basket, the MPC shell wall is separated from the basket shims by a gap. The planar temperature distribution across the basket, as shown in Section 4.4, approximates a shallow parabolic profile. This profile will create high thermal stresses unless structural constraints at the interface between the basket and the basket support structure (or between the MPC shell wall and the basket shims) are removed.
- The MPCs will be loaded with fuel with widely varying heat generation rates. The basket/basket support structure (or MPC shell/basket shim) gap tends to be reduced for higher heat generation rates due to increased thermal expansion rates. Gaps between the fuel basket and the basket support structure (or between the MPC shell wall and the basket shim) are specified to be sufficiently large such that a gap exists around the periphery after any thermal expansion.
- In some early vintage MPCs, a small number of flexible thermal conduction elements (thin aluminum tubes) are interposed between the basket and the MPC shell. The elements are designed to be resilient. They do not provide structural support for the basket, and thus their resistance to thermal growth is negligible.

It is quite evident from the geometry of the MPC that a critical loading event pertains to the drop condition when the MPC is postulated to undergo a handling side drop (the longitudinal axis of the MPC is horizontal) or tip-over. Under the side drop or tip-over condition the flat panels of the fuel basket are subject to an equivalent pressure loading that simulates the deceleration-magnified inertia load from the stored fuel and the MPC's own metal mass.

The MPC fuel basket maintains the spent nuclear fuel in a subcritical arrangement. Its safe operation is assured by maintaining the physical configuration of the storage cell cavities intact in the aftermath of a drop event. This requirement is considered to be satisfied if the MPC fuel basket meets the stress intensity criteria set forth in the ASME Code, Section III, Subsection NG. Therefore, the demonstration that the fuel basket meets Subsection NG limits ensures that there is no impairment of ready retrievability (as required by NUREG-1536), and that there is no unacceptable effect on the subcritical arrangement.

The MPC confinement boundary contains no valves or other pressure relief devices. The MPC enclosure vessel is shown to meet the stress intensity criteria of the ASME Code, Section III, Subsection NB for design and all service conditions. Therefore, the demonstration that the enclosure vessel meets Subsection NB limits ensures that there is no unacceptable release of radioactive materials.

The HI-STORM 100 storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of unreinforced (plain) concrete. Additional regions of fully confined (by enveloping steel structure) unreinforced concrete are attached to the lid and to the baseplate depending on the specific configuration (see applicable figures in previous chapters). The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a cushion for the MPC in the event of a tip-over accident. The storage overpack is not a pressure vessel since it contains cooling vents that do not allow for a differential pressure to develop across the overpack wall. The structural steel components of the HI-STORM 100 Overpack are designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3. Short versions of the HI-STORM 100 overpack, designated as the HI-STORM 100S, and the HI-STORM 100S Version B, are included in this revision. To accommodate nuclear plants with limited height access, the HI-STORM 100S has a re-configured lid and a lower overall height. There are minor weight redistributions but the overall bounding weight of the system is unchanged. The HI-STORM 100S Version B incorporates other improvements and modifications designed to improve fabricability and enhance some margins. Structural analyses are revisited if and only if the modified configuration cannot be demonstrated to be bounded by the original calculation. New or modified calculations focused on the HI-STORM 100S and the HI-STORM 100S Version B are clearly identified within the text of this chapter. Unless otherwise designated, general statements using the terminology "HI-STORM 100" also apply to the HI-STORM 100S and to the HI-STORM 100S Version B. The HI-STORM 100S overpacks can carry all MPCs and transfer casks that can be carried in the HI-STORM 100.

As discussed in Chapters 1 and 2, and Section 3.0, the principal shielding material utilized in the HI-STORM 100 Overpack is plain concrete. Plain concrete was selected for the HI-STORM 100 Overpack in lieu of reinforced concrete, because there is no structural imperative for incorporating tensile load bearing strength into the contained concrete. From a purely practical standpoint, the absence of rebars facilitate pouring and curing of concrete with minimal voids, which is an important consideration in light of its shielding function in the HI-STORM 100 Overpack. Plain concrete, however, acts essentially identical to reinforced concrete under compressive and bearing loads, even though ACI standards apply a penalty factor on the compressive and bearing strength of concrete in the absence of rebars (vide ACI 318.1).

Accordingly, the plain concrete in the HI-STORM 100 is considered as a structural material only to the extent that it may participate in supporting direct compressive loads. The allowable compression/bearing resistance is defined and quantified in the ACI 318.1-89(92) Building Code for Structural Plain Concrete.

In general, strength analysis of the HI-STORM 100 Overpack and its confined concrete is carried out only to demonstrate that the concrete is able to perform its radiation protection function and that retrievability of the MPC subsequent to any postulated accident condition of storage or handling is maintained.

A discrete ITS component in the HI-STORM 100 System is the HI-TRAC transfer cask. The HI-TRAC serves to provide a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM 100 Overpack. The HI-TRAC body is a double-walled steel cylinder that constitutes its structural system. Contained between the two steel shells is an intermediate lead cylinder. Attached to the exterior of the HI-TRAC body outer shell is a water jacket that acts as a radiation barrier. The HI-TRAC is not a pressure vessel since it contains a penetration in the HI-TRAC top lid that does not allow for a differential pressure to develop across the HI-TRAC wall. Nevertheless, in the interest of conservatism, structural steel components of the HI-TRAC are subject to the stress limits of the ASME Code, Section III, Subsection NF, Class 3.

Since both the HI-STORM 100 and HI-TRAC may serve as an MPC carrier, their lifting attachments are designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and Regulatory Guide 3.61 [1.0.2].

Table 2.2.6 provides a listing of the applicable design codes for all structures, systems, and components which are designated as ITS. Since no structural credit is required for the weld between the adjustable basket support pieces (i.e., shims and basket support flat plates), the adjustable basket supports are classified as NITS. In addition, for MPCs using basket shims welded to the MPC fuel basket, the basket supports are classified as NITS.

### 3.1.2 Design Criteria

Principal design criteria for normal, off-normal, and accident/environmental events are discussed in Section 2.2. In this section, the loads, load combinations, and allowable stresses used in the structural evaluation of the HI-STORM 100 System are presented in more detail.



Consistent with the provisions of NUREG-1536, the central objective of the structural analysis presented in this chapter is to ensure that the HI-STORM 100 System possesses sufficient structural capability to withstand normal and off-normal loads and the worst case loads under natural phenomenon or accident events. Withstanding such loadings enables the HI-STORM 100 System to successfully preclude the following negative consequences:

- unacceptable risk of criticality
- unacceptable release of radioactive materials
- unacceptable radiation levels
- impairment of ready retrievability of the SNF

The above design objectives for the HI-STORM 100 System can be particularized for individual components as follows:

- The objectives of the structural analysis of the MPC are to demonstrate that:
  1. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.
  2. The MPC basket does not deform under credible loading conditions such that the subcriticality or retrievability of the SNF is jeopardized.
- The objectives of the structural analysis of the storage overpack are to demonstrate that:
  1. Tornado-generated missiles do not compromise the integrity of the MPC confinement boundary.
  2. The overpack can safely provide for on-site transfer of the loaded MPC and ensure adequate support to the HI-TRAC transfer cask during loading and unloading of the MPC.
  3. The radiation shielding remains properly positioned in the case of any normal, off-normal, or natural phenomenon or accident event.
  4. The flow path for the cooling airflow shall remain available under normal and off-normal conditions of storage and after a natural phenomenon or accident event.
  5. The loads arising from normal, off-normal, and accident level conditions exerted on the contained MPC do not exceed the structural design criteria of the MPC.
  6. No geometry changes occur under any normal, off-normal, and accident level conditions of storage that may preclude ready retrievability of the contained MPC.
  7. A freestanding storage overpack can safely withstand a non-mechanistic

tip-over event with a loaded MPC within the overpack. The HI-STORM 100A is specifically engineered to be permanently attached to the ISFSI pad. The ISFSI pad engineered for the anchored cask is designated as “Important to Safety”. Therefore, the non-mechanistic tipover is not applicable to the HI-STORM 100A.

8. The inter-cask transfer of a loaded MPC can be carried out without exceeding the structural capacity of the HI-STORM 100 Overpack, provided all required auxiliary equipment and components specific to an ISFSI site comply with their design criteria set forth in this FSAR and the handling operations are in full compliance with operational limits and controls prescribed in this FSAR.
- The objective of the structural analysis of the HI-TRAC transfer cask is to demonstrate that:
    1. Tornado generated missiles do not compromise the integrity of the MPC confinement boundary while the MPC is contained within HI-TRAC.
    2. No geometry changes occur under any postulated handling or storage conditions that may preclude ready retrievability of the contained MPC.
    3. The structural components perform their intended function during lifting and handling with the loaded MPC.
    4. The radiation shielding remains properly positioned under all applicable handling service conditions for HI-TRAC.
    5. The lead shielding, top lid, and transfer lid doors remain properly positioned during postulated handling accidents.

The aforementioned objectives are deemed to be satisfied for the MPC, the overpack, and the HI-TRAC, if stresses (or stress intensities, as applicable) calculated by the appropriate structural analyses are less than the allowables defined in Subsection 3.1.2.2, and if the diametral change in the storage overpack (or HI-TRAC), if any, after any event of structural consequence to the overpack (or transfer cask), does not preclude ready retrievability of the contained MPC.

Stresses arise in the components of the HI-STORM 100 System due to various loads that originate under normal, off-normal, or accident conditions. These individual loads are combined to form load combinations. Stresses and stress intensities resulting from the load combinations are compared to their respective allowable stresses and stress intensities. The following subsections present loads, load combinations, and the allowable limits germane to them for use in the structural analyses of the MPC, the overpack, and the HI-TRAC transfer cask.

### 3.1.2.1 Loads and Load Combinations

The individual loads applicable to the HI-STORM 100 System and the HI-TRAC cask are defined in Section 2.2 of this report (Table 2.2.13). Load combinations are developed by assembling the individual loads that may act concurrently, and possibly, synergistically (Table 2.2.14). In this subsection, the individual loads are further clarified as appropriate and the required load combinations are identified. Table 3.1.1 contains the load combinations for the storage overpack where kinematic stability is of primary importance. The load combinations where stress or load level is of primary importance are set forth in Table 3.1.3 for the MPC fuel basket, in Table 3.1.4 for the MPC confinement boundary, and in Table 3.1.5 for the storage overpack and the HI-TRAC transfer cask. Load combinations are applied to the mathematical models of the MPCs, the overpack, and the HI-TRAC. Results of the analyses carried out under bounding load combinations are compared with their respective allowable stresses (or stress intensities, as applicable). The analysis results from the bounding load combinations are also assessed, where warranted, to ensure satisfaction of the functional performance criteria discussed in the preceding subsection.

#### 3.1.2.1.1 Individual Load Cases

The individual loads that address each design criterion applicable to the structural design of the HI-STORM 100 System are catalogued in Table 2.2.13. Each load is given a symbol for subsequent use in the load combination listed in Table 2.2.14.

Accident condition and natural phenomena-induced events, collectively referred to as the "Level D" condition in Section III of the ASME Boiler & Pressure Vessel Codes, in general, do not have a universally prescribed limit. For example, the impact load from a tornado-borne missile, or the overturning load under flood or tsunami, cannot be prescribed as design basis values with absolute certainty that all ISFSI sites will be covered. Therefore, as applicable, allowable magnitudes of such loadings are postulated for the HI-STORM 100 System. The allowable values are drawn from regulatory and industry documents (such as for tornado missiles and wind) or from an intrinsic limitation in the system (such as the permissible "drop height" under a postulated handling accident). In the following, the essential characteristic of each "Level D" type loading is explained.

##### 3.1.2.1.1.1 Tip-Over

It is required to demonstrate that the free-standing HI-STORM 100 storage overpack, containing a loaded MPC, will not tip over as a result of a postulated natural phenomenon event, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tip-over scenario per NUREG-1536 is analyzed. Since the HI-STORM 100S and the HI-STORM 100S Version B have an overall length that is less than the regular HI-STORM 100, the maximum impact velocity of the overpack will be reduced. Therefore, the results of the tipover analysis for the HI-STORM 100 (reported in Appendix 3.A) are bounding for the HI-STORM 100S and HI-STORM 100S Version B. The potential of the HI-STORM 100 Overpack tipping over during the lowering (or raising) of the loaded MPC into (or out of) it with the HI-TRAC cask mounted on it is ruled out because of the safeguards and devices mandated by this FSAR for such operations (Subsection 2.3.3.1 and

Technical Specification 4.9). The physical and procedural barriers under the MPC handling operations have been set down in the FSAR to preclude overturning of the HI-STORM/HI-TRAC assemblage with an extremely high level of certainty. Much of the ancillary equipment needed for the MPC transfer operations must be custom engineered to best accord with the structural and architectural exigencies of the ISFSI site. Therefore, with the exception of the HI-TRAC cask, their design cannot be prescribed, a priori, in this FSAR. However, carefully drafted Design Criteria and conditions of use set forth in this FSAR eliminate the potential of weakening of the safety measures contemplated herein to preclude an overturning event during MPC transfer operations. Subsection 2.3.3.1 contains a comprehensive set of design criteria for the ancillary equipment and components required for MPC transfer operations to ensure that the design objective of precluding a kinematic instability event during MPC transfer operations is met. Further information on the steps taken to preclude system overturning during MPC transfer operations may be found in Chapter 8, Section 8.0.

In the HI-STORM 100A configuration, wherein the overpack is physically anchored to the ISFSI pad, the potential for a tip-over is a priori precluded. Therefore, the ISFSI pad need not be engineered to be sufficiently compliant to limit the peak MPC deceleration to Table 2.2.8 values. The stiffness of the pad, however, may be controlled by the ISFSI structural design and, therefore, may result in a reduced “carry height” from that specified for a freestanding cask. If a non-single failure proof lifting device is employed to carry the cask over the pad, determination of maximum carry height must be performed by the ISFSI owner once the ISFSI pad design is formalized.

#### 3.1.2.1.1.2 Handling Accident

A handling accident during transport of a loaded HI-STORM 100 storage overpack is assumed to result in a vertical drop. The HI-STORM 100 storage overpack will not be handled in a horizontal position while containing a loaded MPC. Therefore, a side drop is not considered a credible event.

HI-TRAC can be carried in a horizontal orientation while housing a loaded MPC. Therefore, a handling accident during transport of a loaded HI-TRAC in a horizontal orientation is considered to be a credible accident event.

As discussed in the foregoing, the vertical drop of the HI-TRAC and the tip-over of the assemblage of a loaded HI-TRAC on the top of the HI-STORM 100 storage overpack during MPC transfer operations do not need to be considered.

#### 3.1.2.1.1.3 Flood

The postulated flood event results into two discrete scenarios which must be considered; namely,

1. stability of the HI-STORM 100 System due to flood water velocity, and
2. structural effects of hydrostatic pressure and water velocity induced lateral pressure.

The maximum hydrostatic pressure on the cask in a flood where the water level is conservatively

set at 125 feet is calculated as follows:

Using

$p$  = the maximum hydrostatic pressure on the system (psi),

$\gamma$  = weight density of water = 62.4 lb/ft<sup>3</sup>

$h$  = the height of the water level = 125 ft;

The maximum hydrostatic pressure is

$$p = \gamma h = (62.4 \text{ lb/ft}^3)(125 \text{ ft})(1 \text{ ft}^2/144 \text{ in}^2) = 54.2 \text{ psi}$$

The accident condition design external pressure for the MPC (Table 2.2.1) bounds the maximum hydrostatic pressure exerted by the flood.

#### 3.1.2.1.1.4 Explosion

The potential for explosive materials shall be evaluated based on site specific conditions as required in the HI-STORM 100 Technical Specification.

Pressure waves from an explosive blast in a property near the ISFSI site result in an impulsive aerodynamic loading on the stored HI-STORM 100 Overpacks. Depending on the rapidity of the pressure build-up, the inside and outside pressures on the HI-STORM METCON™ shell may not equalize, leading to a net lateral loading on the upright overpack as the pressure wave traverses the overpack. The magnitude of the dynamic pressure wave is conservatively set to a value below the magnitude of the pressure differential that would cause a tip-over of the cask if the pulse duration were set at one second. With the maximum design basis pressure pulse established (by setting the design basis pressure differential sufficiently low that cask tip-over is not credible due to the travelling pressure wave), the stress state under this condition requires analysis. The lateral pressure difference, applied over the overpack full height, causes axial and circumferential stresses and strains to develop. Level D stress limits must not be exceeded under this state of stress. It must also be demonstrated that no permanent ovalization of the cross section occurs that leads to loss of clearance to remove the MPC after the explosion.

Once the pressure wave traverses the cask body, then an elastic stability evaluation is warranted. An all-enveloping pressure from the explosion may threaten safety by buckling the overpack outer shell.

In contrast to the overpack, the MPC is a closed pressure vessel. Because of the enveloping overpack around it, the explosive pressure wave would manifest as an external pressure on the external surface of the MPC.

The maximum overpressure on the MPC resulting from an explosion is limited to the accident condition design external pressure specified in Table 2.2.1. The maximum external pressure differential on the overpack is limited to the accident condition design pressure differential

specified in Table 2.2.1.

### 3.1.2.1.1.5 Tornado

The three components of a tornado load are:

1. pressure changes,
2. wind loads, and
3. tornado-generated missiles.

Wind speeds and tornado-induced pressure drop are specified in Table 2.2.4. Tornado missiles are listed in Table 2.2.5. A central functional objective of a storage overpack is to maintain the integrity of the “confinement boundary”, namely, the multi-purpose canister stored inside it. This operational imperative requires that the mechanical loadings associated with a tornado at the ISFSI do not jeopardize the physical integrity of the loaded MPC. Potential consequences of a tornado on the cask system are:

- Instability (tip-over) due to tornado missile impact plus either steady wind or impulse from the pressure drop (only applicable for free-standing cask).
- Stress in the overpack induced by the lateral force caused by the steady wind or missile impact.
- Loadings applied on the MPC transmitted to the inside of the overpack through its openings or as a secondary effect of loading on the enveloping overpack structure.
- Excessive storage overpack permanent deformation that may prevent ready retrievability of the MPC.
- Excessive storage overpack permanent deformation that may significantly reduce the shielding effectiveness of the storage overpack.

Analyses must be performed to ensure that, due to the tornado-induced loadings:

- The loaded overpack does not become kinematically unstable (only applicable for free-standing cask).
- The overpack does not deform plastically such that the retrievability of the stored MPC is threatened.
- The MPC does not sustain an impact from an incident missile.
- The MPC is not subjected to inertia loads (acceleration or deceleration) in excess of its design basis limit set forth in Chapter 2 herein.

- The overpack does not deform sufficiently due to tornado-borne missiles such that the shielding effectiveness of the overpack is significantly affected.

The results obtained for the HI-STORM 100 bound the corresponding results for the HI-STORM 100S versions because of the reduced height. In the anchored configuration (HI-STORM 100A), the kinematic stability requirement stated above is replaced with the requirement that the stresses in the anchor studs do not exceed level D stress limits for ASME Section III, Class 3, Subsection NF components.

#### 3.1.2.1.1.6 Earthquake

Subsections 2.2.3.7 and 3.4.7 contain the detailed specification of the seismic inputs applied to the HI-STORM 100 System. The design basis earthquake is assumed to be at the top of the ISFSI pad. Potential consequences of a seismic event are sliding/overturning of a free-standing cask, overstress of the sector lugs and anchor studs for the anchored HI-STORM 100A, and lateral force on the overpack causing excessive stress and deformation of the storage overpack.

In the anchored configuration (HI-STORM 100A), a seismic event results in a fluctuation in the state of stress in the anchor bolts and a local bending action on the sector lugs.

Analyses must be performed to ensure that:

- The maximum axial stress in the anchor bolts remains below the Level D stress limits for Section III Class 3 Subsection NF components.
- The maximum primary membrane plus bending stress intensity in the sector lugs during the DBE event satisfies Level D stress limits of the ASME Code, Subsection NF.
- The anchor bolts will not sustain fatigue failure due to pulsation in their axial stress during the DBE event.
- The stress in the weld line joining the sector lugs to the HI-STORM 100 weldment is within Subsection NF limits for Level D condition.

#### 3.1.2.1.1.7 Lightning

The HI-STORM 100 Overpack contains over 25,000 lb of highly conductive carbon steel with over 700 square feet of external surface area. Such a large surface area and metal mass is adequate to dissipate any lightning that may strike the HI-STORM 100 System. There are no combustible materials on the HI-STORM 100 surface. Therefore, lightning will not impair the structural performance of components of the HI-STORM 100 System that are important to safety.

#### 3.1.2.1.1.8 Fire

The potential structural consequences of a fire are: the possibility of an interference developing

between the storage overpack and the loaded MPC due to free thermal expansion; and, the degradation of material properties to the extent that their structural performance is affected during a subsequent recovery action. The fire condition is addressed to the extent necessary to demonstrate that these adverse structural consequences do not materialize.

#### 3.1.2.1.1.9 100% Fuel Rod Rupture

The effect on structural performance by 100% fuel rod rupture is felt as an increase in internal pressure. The accident internal pressure limit set in Chapter 2 bounds the pressure from 100% fuel rod rupture. Therefore, no new load condition has been identified.

#### 3.1.2.1.2 Load Combinations

Load combinations are created by summing the effects of several individual loads. The load combinations are selected for the normal, off-normal, and accident conditions. The loadings appropriate for HI-STORM 100 under the various conditions are presented in Table 2.2.14. These loadings are combined into meaningful combinations for the various HI-STORM 100 System components in Tables 3.1.1, and 3.1.3-3.1.5. Table 3.1.1 lists the load combinations that address overpack stability. Tables 3.1.3 through 3.1.5 list the applicable load combinations for the fuel basket, the enclosure vessel, and the overpack and HI-TRAC, respectively.

As discussed in Subsection 2.2.7, the number of discrete load combinations for each situational condition (i.e., normal, off-normal, etc.) is consolidated by defining bounding loads for certain groups of loadings. Thus, the accident condition pressure  $P_o^*$  bounds the surface loadings arising from accident and extreme natural phenomenon events, namely, tornado wind  $W'$ , flood  $F$ , and explosion  $E^*$ .

As noted previously, certain loads, namely earthquake  $E$ , flowing water under flood condition  $F$ , force from an explosion pressure pulse  $F^*$ , and tornado missile  $M$ , act to destabilize a cask. Additionally, these loads act on the overpack and produce essentially localized stresses at the HI-STORM 100 System to ISFSI interface. Table 3.1.1 provides the load combinations that are relevant to the stability analyses of freestanding casks. The site ISFSI DBE zero period acceleration (ZPA) must be bounded by the design basis seismic ZPA defined by the Load Combination C of Table 3.1.1 to demonstrate that the margin against tip-over during a seismic event is maintained.

The major constituents in the HI-STORM 100 System are: (i) the fuel basket, (ii) the enclosure vessel, (iii) the HI-STORM 100 (or HI-STORM 100S versions) Overpack, and (iv) the HI-TRAC transfer cask. The fuel basket and the enclosure vessel (EV) together constitute the multi-purpose canister. The multi-purpose canister (MPC) is common to HI-STORM 100 and HI-STAR 100, and as such, has been extensively analyzed in the storage FSAR and transport SAR (Dockets 72-1008 and 71-9261) for HI-STAR 100. Many of the loadings on the MPC (fuel basket and enclosure vessel) are equal to or bounded by loadings already considered in the HI-STAR 100 SAR documents. Where such analyses have been performed, their location in the HI-STAR 100 SAR documents is indicated in this HI-STORM 100 SAR for continuity in narration. A complete



account of analyses and results for all load combinations for all four constituents parts is provided in Section 3.4 as required by Regulatory Guide 3.61.

In the following, the loadings listed as applicable for each situational condition in Table 2.2.14 are addressed in meaningful load combinations for the fuel basket, enclosure vessel, and the overpack. Each component is considered separately.

### Fuel Basket

Table 3.1.3 summarizes all loading cases (derived from Table 2.2.14) that are germane to demonstrating compliance of the fuel baskets to Subsection NG when these baskets are housed within HI-STORM 100 or HI-TRAC.

The fuel basket is not a pressure vessel; therefore, the pressure loadings are not meaningful loads for the basket. Further, the basket is structurally decoupled from the enclosure vessel. The gap between the basket and the enclosure vessel is sized to ensure that no constraint of free-end thermal expansion of the basket occurs. The demonstration of the adequacy of the basket-to -enclosure vessel (EV) gap to ensure absence of interference is a physical problem that must be analyzed.

The normal handling loads on the fuel basket in an MPC within the HI-STORM 100 System or the HI-TRAC transfer cask are identical to or bounded by the normal handling loads analyzed in the HI-STAR 100 FSAR Docket Number 72-1008.

Three accident condition scenarios must be considered: (i) drop with the storage overpack axis vertical; (ii) drop with the HI-TRAC axis horizontal; and (iii) storage overpack tipover. The vertical drop scenario is considered in the HI-STAR 100 FSAR.

The horizontal drop and tip-over must consider multiple orientation of the fuel basket, as the fuel basket is not radially symmetric. Therefore, two horizontal drop orientations are considered which are referred to as the 0 degree drop and 45 degree drop, respectively. In the 0 degree drop, the basket drops with its panels oriented parallel and normal to the vertical (see Figure 3.1.2). The 45-degree drop implies that the basket's honeycomb section is rotated meridionally by 45 degrees (Figure 3.1.3).

### Enclosure Vessel

Table 3.1.4 summarizes all load cases that are applicable to structural analysis of the enclosure vessel to ensure integrity of the confinement boundary.

The enclosure vessel is a pressure vessel consisting of a cylindrical shell, a thick circular baseplate at the bottom, and a thick circular lid at the top. This pressure vessel must be shown to meet the primary stress intensity limits for ASME Section III Class 1 at the design temperature and primary plus secondary stress intensity limits under the combined action of pressure plus thermal loads.

Normal handling of the enclosure vessel is considered in Docket 72-1008; the handling loads are

independent of whether the enclosure vessel is within HI-STAR 100, HI-STORM 100, or HI-TRAC.

The off-normal condition handling loads are identical to the normal condition and, therefore, a separate analysis is not required.

Analyses presented in this chapter are intended to demonstrate that the maximum decelerations in drop and tip-over accident events are limited by the bounding values in Table 3.1.2. The vertical drop event is considered in the HI-STAR 100 FSAR Docket 72-1008.

The deceleration loadings developed in the enclosure vessel during a horizontal drop event are combined with those due to  $P_i$  (internal pressure) acting alone. The accident condition pressure is bounded by  $P_i^*$ . The design basis deceleration for the MPC in the HI-STAR 100 System is 60g's, whereas the design basis deceleration for the MPC in the HI-STORM 100 System is 45g's. The design pressures are identical. The fire event ( $T^*$  loading) is considered for ensuring absence of interference between the enclosure vessel and the fuel basket and between the enclosure vessel and the overpack.

It is noted that the MPC basket-enclosure vessel thermal expansion and stress analyses are reconsidered in this submittal to reflect the different MPC-to-overpack gaps that exist in the HI-STORM 100 Overpack versus the HI-STAR 100 overpack, coupled with the different design basis decelerations.

### Storage Overpack

Table 3.1.5 identifies the load cases to be considered for the overpack. These are in addition to the kinematic criteria listed in Table 3.1.1. Within these load cases and kinematic criteria, the following items must be addressed:

#### Normal Conditions

- The dead load of the HI-TRAC with the heaviest loaded MPC (dry) on top of the HI-STORM 100 Overpack must be shown to be able to be supported by the metal-concrete (METCON™) structure consisting of the two concentric steel shells and the radial ribs.
- The dead load of the HI-STORM 100 Overpack itself must be supportable by the steel structure with no credit for concrete strength other than self-support in compression.
- Normal handling loads must be accommodated without taking any strength credit from the contained concrete other than self-support in compression.

#### Accident Conditions

- Maximum flood water velocity for the overpack with an empty MPC must be specified to ensure that no sliding or tip-over occurs.

- Tornado missile plus wind on an overpack with an empty MPC must be specified to demonstrate that no cask tip-over occurs.
- Tornado missile penetration analysis must demonstrate that the postulated large and penetrant missiles cannot contact the MPC. The small missile must be shown not to penetrate the MPC pressure vessel boundary, since it can enter the overpack cavity through the vent ducts.
- Under seismic conditions, a fully loaded, free-standing HI-STORM 100 overpack must be demonstrated to not tip over under the maximum ZPA event. The maximum sliding of the overpack must demonstrate that casks will not impact each other.
- Under a non-mechanistic postulated tip-over of a fully loaded, freestanding HI-STORM 100 overpack, the overpack lid must not dislodge.
- Accident condition stress levels must not be exceeded in the steel and compressive stress levels in the concrete must remain within allowable limits.
- Accident condition induced gross general deformations of the storage overpack must be limited to values that do not preclude ready retrievability of the MPC.

As noted earlier, analyses performed using the HI-STORM 100 generally provide results that are identical to or bound results for the shorter HI-STORM 100S versions; therefore, analyses are not repeated specifically for the HI-STORM 100S unless the specific geometry changes significantly influence the safety factors.

### HI-TRAC Transfer Cask

Table 3.1.5 identifies load cases applicable to the HI-TRAC transfer cask.

The HI-TRAC transfer cask must provide radiation protection, must act as a handling cask when carrying a loaded MPC, and in the event of a postulated accident must not suffer permanent deformation to the extent that ready retrievability of the MPC is compromised. This submittal includes four types of transfer casks: a 125-ton HI-TRAC (referred to as the HI-TRAC 125), a modified version of the HI-TRAC 125 called the HI-TRAC 125D, a 100-ton HI-TRAC (referred to as the HI-TRAC 100), and a modified version of the HI-TRAC 100 called the HI-TRAC 100D. The details of these four transfer casks are provided in the design drawings in Section 1.5. The same steel structures (i.e., shell thicknesses, lid thicknesses, etc.) are maintained with the only major differences being in the amount of lead shielding, the water jacket configuration, the bottom flange, and the lower dead weight loading. Therefore, all structural analyses performed for the HI-TRAC 125 are repeated for the HI-TRAC 125D, the HI-TRAC 100, and the HI-TRAC 100D only if it cannot be clearly demonstrated that the HI-TRAC 125 calculation is bounding.

### 3.1.2.2 Allowables

The important to safety components of the HI-STORM 100 System are listed in Table 2.2.6. Allowable stresses, as appropriate, are tabulated for these components for all service conditions.

In Subsection 2.2.5, the applicable service level from the ASME Code for determination of allowables is listed. Table 2.2.14 provides a tabulation of normal, off-normal, and accident conditions and the service levels defined in the ASME Code, along with the applicable loadings for each service condition.

Allowable stresses and stress intensities are calculated using the data provided in the ASME Code and Tables 2.2.10 through 2.2.12. Tables 3.1.6 through 3.1.16 contain numerical values of the stresses/stress intensities for all MPC, overpack, and HI-TRAC load bearing materials as a function of temperature.

In all tables the terms  $S$ ,  $S_m$ ,  $S_y$ , and  $S_u$ , respectively, denote the design stress, design stress intensity, minimum yield strength, and the ultimate strength. Property values at intermediate temperatures that are not reported in the ASME Code are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the Code in any structural calculation.

Additional terms relevant to the analyses are extracted from the ASME Code (Figure NB-3222-1, for example) as follows:

Symbol	Description	Notes
$P_m$	Average primary stress across a solid section	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.
$P_L$	Average stress across any solid section	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
$P_b$	Primary bending stress	Component of primary stress proportional to the distance from the centroid of a solid section. Excludes the effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
$P_e$	Secondary expansion stress	Stresses that result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration. (Not applicable to vessels.)
$Q$	Secondary membrane plus bending stress	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion.
$F$	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may

Symbol	Description	Notes
		cause fatigue but not distortion. This value is not used in the tables.

It is shown that there is no interference between component parts due to free thermal expansion. Therefore,  $P_e$  does not develop within any HI-STORM 100 component.

It is recognized that the planar temperature distribution in the fuel basket and the overpack under the maximum heat load condition is the highest at the cask center and drops monotonically, reaching its lowest value at the outside surface. Strictly speaking, the allowable stresses/stress intensities at any location in the basket, the enclosure vessel, or the overpack should be based on the coincident metal temperature under the specific operating condition. However, in the interest of conservatism, reference temperatures are established for each component, which are upper bounds on the metal temperature for each situational condition. Table 3.1.17 provides the reference temperatures for the fuel basket and the MPC canister utilizing Tables 3.1.6 through 3.1.16, and provides conservative numerical limits for the stresses and stress intensities for all loading cases. Reference temperatures for the MPC baseplate and the MPC lid are 400 degrees F and 550 degrees F, respectively, as specified in Table 2.2.3.

Finally, the lifting attachments or the interfacing lift points in the HI-STORM 100 and HI-TRAC casks and the multi-purpose canisters must meet the requirements of NUREG- 0612 and/or Regulatory Guide 3.61 as described in Subsection 3.4.3 and Tables 2.0.1, 2.0.2 and 2.0.3 under a normal handling condition (Load Case 01 in Table 3.1.5). The load combination D+H in Table 3.1.5 is equivalent to 1.15D. This is further explained in Subsection 3.4.3.

The region around the trunnions is part of the NF structure in HI-STORM 100 and HI-TRAC and NB pressure boundary in the MPC, and as such, must satisfy the applicable stress (or stress intensity) limits for the load combination. In addition to meeting the applicable Code limits, it is further required that the primary stress required to maintain equilibrium at the defined trunnion/mother structure interface must not exceed the material yield stress at three times the handling condition load (1.15D). This criterion, mandated by Regulatory Guide 3.61, Section 3.4.3, insures that a large safety factor exists on non-local section yielding at the trunnion/mother structure interface that would lead to unacceptable section displacement and rotation.

### 3.1.2.3 Brittle Fracture

The MPC canister and basket are constructed from a series of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition (DBT) in the operating temperature range of the HI-STORM 100 System. Therefore, brittle fracture is not a concern for the MPC components fabricated using austenitic stainless steel. Such an assertion cannot be made a priori for the MPC confinement boundary components fabricated using duplex stainless steel grade of Alloy X, or for the HI-STORM storage overpack and the HI-TRAC transfer cask that contain ferritic steel parts.

The use of duplex stainless steel grade of Alloy X material is limited to the MPC confinement boundary components and shall be restricted to the maximum temperatures specified in Table 1.A.6 as the material may suffer from precipitation of brittle micro-constituents above 600°F.

The duplex stainless steel material undergoes DBT below the temperature of -40°F/-40°C [3.1.4] (which is equal to the Lowest Service Temperature (LST) of MPC). In addition, Holtec Position Paper DS-213 [9.1.6] demonstrates that crack propagation in MPC lid-to-shell weld is not credible for austenitic and duplex stainless steel grades of Alloy X. Therefore, brittle fracture is not a concern for the MPC confinement boundary components fabricated using duplex stainless steel grade of Alloy X as well.

In general, the impact testing requirements for the HI-STORM overpack and the HI-TRAC transfer cask are a function of two parameters: the LST and the normal stress level. The significance of these two parameters, as they relate to impact testing of the overpack and the transfer cask, is discussed below.

In normal storage mode, the LST of the HI-STORM storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). During the HI-STORM handling operations, the applicable lowest service temperature is 0°F (which is the threshold ambient temperature below which lifting and handling of the HI-STORM 100 Overpack or the HI-TRAC cask is not permitted by the Technical Specification). Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM 100 Overpack and the HI-TRAC cask; namely,

LST = 0°F for the HI-STORM overpack during handling operations and for the HI-TRAC transfer cask during all normal operating conditions.

LST = -40°F for the HI-STORM overpack during all non-handling operations (i.e., normal storage mode).

Parts used to lift the overpack or the transfer cask, which include the anchor block in the HI-STORM 100 overpack, and the pocket trunnions, the lifting trunnions and the lifting trunnion block in HI-TRAC, will henceforth be referred to as “significant-to-handling” (STH) parts. The applicable design codes for these elements of the structure are NUREG-0612 and Regulatory Guide 3.61. All other parts of the overpack and the transfer cask will be referred to as “NF” components. It is important to ensure that all materials designated as “NF” or “STH” parts possess sufficient fracture toughness to preclude brittle fracture. For the STH parts, the necessary level of protection against brittle fracture is deemed to exist if the NDT (nil ductility transition) temperature of the part is at least 40° below the LST. Therefore, the required NDT temperature for all STH parts is -40°F.

It is well known that the NDT temperature of steel is a strong function of its composition, manufacturing process (viz., fine grain vs. coarse grain practice), thickness, and heat treatment. For example, according to Burgreen [3.1.3], increasing the carbon content in carbon steels from

0.1% to 0.8% leads to the change in NDT from -50°F to approximately 120°F. Likewise, lowering of the normalizing temperature in the ferritic steels from 1200°C to 900°C lowers the NDT from 10°C to -50°C [3.1.3]. It, therefore, follows that the fracture toughness of steels can be varied significantly within the confines of the ASME Code material specification set forth in Section II of the Code. For example, SA516 Gr. 70 (which is a principal NF material in the HI-STORM 100 Overpack) can have a maximum carbon content of up to 0.3% in plates up to four inches thick. Section II further permits normalizing or quenching followed by tempering to enhance fracture toughness. Manufacturing processes which have a profound effect on fracture toughness, but little effect on tensile or yield strength of the material, are also not specified with the degree of specificity in the ASME Code to guarantee a well defined fracture toughness. In fact, the Code relies on actual coupon testing of the part to ensure the desired level of protection against brittle fracture. For Section III, Subsection NF Class 3 parts, the desired level of protection is considered to exist if the lowest service temperature is equal to or greater than the NDT temperature (per NF 2311(b)(10)). Accordingly, the required NDT temperature for all load bearing metal parts in the HI-STORM 100 Overpack (NF and STH) is -40°F. Likewise, the NDT temperature for all NF parts in HI-TRAC (except for STH parts) is set equal to 0°F.

The STH components (HI-STORM bolt anchor block, HI-TRAC lifting trunnion, HI-TRAC lifting trunnion block, and HI-TRAC pocket trunnion) have thicknesses greater than 2". SA350-LF3 has been selected as the material for these items (except for the lifting trunnions) due to its capability to maintain acceptable fracture toughness at low temperatures (see Table 5 in SA350 of ASME Section IIA). Additionally, material for the HI-TRAC top flange, pool lid (100 ton) and pool lid outer ring (125 ton) has been defined as SA350-LF3, SA350-LF2, or SA203E (see Table A1.15 of ASME Section IIA) in order to achieve low temperature fracture toughness. The HI-TRAC lifting trunnion is fabricated from SB-637 Grade N07718, a high strength nickel alloy material. This material has a high resistance to fracture at low temperatures. All other steel structural materials in the HI-STORM 100 overpack and HI-TRAC cask are made of SA516 Gr. 70, SA515 Gr. 70, or SA36 (with some components having an option for SA203E or SA350-LF3 depending on material availability).

Either as-rolled or normalized SA516 Gr. 70 plate can be used to fabricate the overpack and the transfer cask. If the SA516 Gr. 70 plate is normalized, then it is exempt from impact testing per NF-2311(b). The specific reasons are:

1. The LST for handling operations is above the Minimum Design Temperature of SA516 Gr. 70 normalized plate (for thickness less than 2-1/2") per Figure NF-2311(b)-1, and;
2. During non-handling operations (i.e., normal storage mode), the maximum tensile stress in the HI-STORM overpack is less than the threshold limit of 6,000 psi specified in NF-2311(b)(7).

If the SA516 Gr. 70 plate is as-rolled (i.e., not normalized), then impact testing is required, except when the material is used for the following HI-STORM components: inner and outer shells, top plate, lid shear ring, lid shield ring, lid outer ring (for 100S Version B), and lid cover plate (for

100S Version B) and Base-Bottom Plate (for 100S Version B). The material for these components is exempt from impact testing per NF-2311(b)(7) since the maximum stress under normal conditions, including handling operations, does not exceed 6,000 psi tension or is compressive.

Even though SA516 Gr. 70 normalized plate is exempt from impact testing per the above, certain components of the HI-STORM 100A overpack, namely the lug support ring, the gussets, and the baseplate, are impact tested, as a defense-in-depth measure, because they are potentially subject to high tensile stress levels (i.e., greater than 6,000 psi) during an earthquake.

Table 3.1.18 provides a summary of impact testing requirements to satisfy the requirements for prevention of brittle fracture.

#### 3.1.2.4 Fatigue

In storage, the HI-STORM 100 System is not subject to significant cyclic loads. Failure due to fatigue is not a concern for the HI-STORM 100 System.

In an anchored installation, however, the anchor studs sustain a pulsation in the axial load during the seismic event. The amplitude of axial stress variation under the DBE event is computed in this chapter and a significant margin of safety against fatigue failure during the DBE event is demonstrated.

The system is subject to cyclic temperature fluctuations. These fluctuations result in small changes of thermal expansions and pressures in the MPC. The loads resulting from these changes are small and do not significantly contribute to the "usage factor" of the cask.

Inspection of the HI-TRAC trunnions specified in Chapter 9 will preclude use of a trunnion that exhibits visual damage.

#### 3.1.2.5 Buckling

Certain load combinations subject structural sections with relatively large slenderness ratios (such as the enclosure vessel shell) to compressive stresses that may actuate buckling instability before the allowable stress is reached. Tables 3.1.4 and 3.1.5 list load combinations for the enclosure vessel and the HI-STORM 100/HI-TRAC structures; the cases which warrant stability (buckling) check are listed therein (note that a potential buckling load has already been identified as a consequence of a postulated explosion).



**TABLE 3.1.1****LOAD COMBINATIONS SIGNIFICANT TO HI-STORM 100 OVERPACK  
KINEMATIC STABILITY ANALYSIS**

<b>Loading Case</b>	<b>Combinations<sup>†</sup></b>	<b>Comment</b>	<b>Analysis of this Load Case Presented in:</b>
A	D + F	This case establishes flood water flow velocity with a minimum safety factor of 1.1 against overturning and sliding.	Subsection 3.4.6
B	D + M + W'	Demonstrate that the HI-STORM 100 Overpack with minimum SNF stored (minimum D) will not tip over.	Subsection 3.4.8
C	D + E	Establish the value of ZPA <sup>††</sup> that will not cause the overpack to tip over.	Subsection 3.4.7

† Loading symbols are defined in Table 2.2.13

†† ZPA is zero period acceleration

**TABLE 3.1.2****DESIGN BASIS DECELERATIONS FOR THE DROP EVENTS**

<b>Case</b>	<b>Value<sup>†</sup> (in multiples of acceleration due to gravity)</b>
Vertical axis drop (HI-STORM 100 Overpack only)	45
Horizontal axis (side) drop (HI-TRAC only)	45

<sup>†</sup> The design basis value is set from the requirements of the HI-STORM 100 System, as its components are operated as a storage system. The MPC is designed to higher loadings (60g's vertical and horizontal) when in a HI-STAR 100 overpack. Analysis of the MPC in a HI-STAR 100 overpack under a 60g loading is provided in HI-STAR 100 Docket Numbers 71-9261 and 72-1008.

**TABLE 3.1.3**  
**LOADING CASES FOR THE FUEL BASKET**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location Where this Case is Evaluated</b>
F1	T, T'	Demonstrate that the most adverse of the temperature distributions in the basket will not cause fuel basket to expand and contact the enclosure vessel wall. Compute the secondary stress intensity and show that it is small.	Subsection 3.4.4.2
F2 (Note 1)	D + H	Conservatively add the stresses in the basket due to vertical and horizontal orientation handling to form a bounding stress intensity.	Table 3.4.9 of HI-STAR FSAR (Docket 72-1008)
F3 F3.a (Note 2)	D + H'	Vertical axis drop event	HI-STAR FSAR, Subsection 3.4.4.3.1.3
F3.b (Note 3)	D + H'	Side Drop, 0 degree orientation (Figure 3.1.2)	Table 3.4.6
F3.c (Note 3)	D + H'	Side Drop, 45 degree orientation (Figure 3.1.3)	Table 3.4.6

## Notes:

1. Load Case F2 for the HI-STORM 100 System is identical to Load Case F2 for the HI-STAR 100 System in Docket Number 72-1008, Table 3.1.3.
2. Load Case F3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008, Subsection 3.4.4.3.1.3. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases F3.b and F3.c are analyzed here for a 45g deceleration, while the MPC is housed within a HI-STORM 100 Overpack or a HI-TRAC transfer cask. The initial clearance at the interface between the MPC shell and the HI-STORM 100 Overpack or HI-TRAC transfer cask is greater than or equal to the initial clearance between the MPC shell and the HI-STAR 100 overpack. This difference in clearance directly affects the stress field. The side drop analysis for the MPC in the HI-STAR 100 overpack under 60g's bounds the corresponding analysis of the MPC in HI-TRAC for 45 g's.

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.1.4****LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†</sup></b>	<b>Notes</b>	<b>Comments and Location Where this Case is Analyzed</b>
E1 (Note 1)			
E1.a	Design internal pressure, $P_i$	Primary stress intensity limits in the shell, baseplate, and closure ring	E1.a Lid Docket 72-1008 3.E.8.1.1 Baseplate Docket 72-1008 3.I.8.1 Shell 3.4.4.3.1.2 Supports N/A
E1.b	Design external pressure, $P_o$	Primary stress intensity limits, buckling stability	E1.b Lid $P_i$ bounds Baseplate $P_i$ bounds Shell Docket 72-1008 Buckling methodology in 3.H Supports N/A
E1.c	Short-term normal internal pressure, $P_{st}$ , plus Temperature, T	Primary plus secondary stress intensity under Level A condition	E1.c Lid, Baseplate, and Shell Section 3.4.4.3.1.2
E2	$D + H + (P_i, P_{st}, P_o)^{\dagger\dagger}$	Vertical lift, internal operating pressure conservatively assumed to be equal to the maximum normal pressure. Principal areas of concern are the lid and the baseplate.	Lid Docket 72-1008 3.E.8.1.2 Baseplate 3.4.3.6 Shell 3.4.4.3.1.2 Supports Docket 72-1008 Buckling methodology in 3.H Docket 72-1008 Table 3.4.9

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

<sup>††</sup> The notation  $(P_i, P_{st}, P_o)$  means that all cases are checked with either  $P_o$  or  $P_{st}$  or  $P_i$  applied.

**TABLE 3.1.4 (CONTINUED)****LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†</sup></b>	<b>Notes</b>	<b>Comments and Location Where this Case is Analyzed</b>
E3 E3.a (Note 2)	$D + H' + (P_o, P_i)$	Vertical axis drop event	E3.a Lid Docket 72-10083.E.8.2.1-2 Baseplate Docket 72-10083.I.8.3 Shell Docket 72-1008 Buckling methodology in 3.H Supports N/A
E3.b (Note 3)	$D + H' + (P_i, P_o)$	Side drop, 0 degree orientation (Figure 3.1.2)	E3.b Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6
E3.c (Note 3)	$D + H' + (P_i, P_o)$	Side drop, 45 degree orientation (Figure 3.1.3)	E3.c Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6
E4	T	Demonstrate that interference with the overpack will not develop for T.	Section 3.4.4.2

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.1.4 (CONTINUED)****LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†</sup></b>	<b>Notes</b>	<b>Comments and Location Where this Case is Analyzed</b>	
E5	$P_i^*$ or $P_o^* + D + T^*$	Demonstrate compliance with Level D stress limits – buckling stability.	Lid	3.4.4.3.1.10
			Baseplate	3.4.4.3.1.10
			Shell	Docket 72-1008 Buckling methodology in 3.H 3.4.4.3.1.2 (stress)
			Supports	N/A

## Notes:

1. Load Cases E1.a and E1.b are identical to the load cases presented in Docket Number 72-1008, Table 3.1.4. Design pressures and MPC weights are identical.
2. Load Case E3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases E3.b and E3.c are analyzed in this HI-STORM 100 SAR for a 45g deceleration, while the MPC is housed within the HI-STORM 100 storage overpack. The interface between the MPC shell and storage overpack is not identical to the MPC shell and HI-STAR 100 overpack. The analysis for an MPC housed in HI-TRAC is not performed since results are bounded by those reported in the HI-STAR 100 TSAR for a 60g deceleration.

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.1.5****LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location in FSAR</b>
01	D + H + T + (P <sub>o</sub> ,P <sub>i</sub> )	Vertical load handling of HI-STORM 100 Overpack/HI-TRAC.	Overpack 3.4.3.5 HI-TRAC Shell 3.4.3.3, 3.4.3.4 Pool lid 3.4.3.8 Transfer lid 3.4.3.9
02			
02.a	D + H' + (P <sub>o</sub> ,P <sub>i</sub> )	Storage Overpack: End drop; primary stress intensities must meet level D stress limits.	Overpack 3.4.4.3.2.3
02.b	D + H' + (P <sub>o</sub> ,P <sub>i</sub> )	HI-TRAC: Horizontal (side) drop; meet level D stress limits for NF Class 3 components away from the impacted zone; show lids stay in-place. Show primary and secondary impact decelerations are within design basis. (This case is only applicable to HI-TRAC.)	HI-TRAC Shell 3.4.9.1 Transfer Lid 3.4.4.3.3.3 Slapdown 3.4.9.2
02.c	D + H'	Storage Overpack: Tip-over; any permanent deformations must not preclude ready retrieval of the MPC.	Overpack 3.4.10, 3.A

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13

**TABLE 3.1.5 (CONTINUED)****LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location in FSAR</b>	
03	D (water jacket)	Satisfy primary membrane plus bending stress limits for water jacket (This case is only applicable to HI-TRAC).	3.4.4.3.3.4	
04	M (penetrant missiles)	Demonstrate that no thru-wall breach of the HI-STORM overpack or HI-TRAC transfer cask occurs, and the primary stress levels are not exceeded. Small and intermediate missiles are examined for HI-STORM and HI-TRAC. Large missile penetration is also examined for HI-TRAC.	Overpack	3.4.8.1
			HI-TRAC	3.4.8.2.1, 3.4.8.2.2
05	P <sub>o</sub>	Explosion: must not produce buckling or exceed primary stress levels in the overpack structure.	3.4.4.5.2, 3.4.7.2	

## Notes:

1. Under each of these load cases, different regions of the structure are analyzed to demonstrate compliance.

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13



**TABLE 3.1.6****DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA203-E  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	23.3	23.3	35.0	35.0	69.9	69.9
300	23.3	23.3	35.0	35.0	69.9	69.9
400	22.9	22.9	34.4	34.4	68.7	68.7
500	21.6	21.6	32.4	32.4	64.8	64.8

## Definitions:

- $S_m$  = Stress intensity values per ASME Code  
 $P_m$  = Primary membrane stress intensity  
 $P_L$  = Local membrane stress intensity  
 $P_b$  = Primary bending stress intensity  
 $P_e$  = Expansion stress  
 $Q$  = Secondary stress  
 $P_L + P_b$  = Either primary or local membrane plus primary bending

Definitions for Table 3.1.6 apply to all following tables unless modified.

## Notes:

- Limits on values are presented in Table 2.2.10.
- For stress intensities generated by Level B Service Loadings, allowable stress intensity values are 110% of Level A limits per NB-3223.

† Evaluation required for Design condition only.

†† Evaluation required for Levels A and B only.  $P_e$  not applicable to vessels.

**TABLE 3.1.7****LEVEL D: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA203-E  
**Service Condition:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>P<sub>m</sub></b>	<b>P<sub>L</sub></b>	<b>P<sub>L</sub> + P<sub>b</sub></b>
-20 to 100	49.0	70.0	70.0
200	49.0	70.0	70.0
300	49.0	70.0	70.0
400	48.2	68.8	68.8
500	45.4	64.9	64.9

## Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed 0.42 S<sub>u</sub>.
3. Limits on values are presented in Table 2.2.10.
4. P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.6.

**TABLE 3.1.8**  
**DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code: ASME NB**  
**Material: SA350-LF3**  
**Service Conditions: Design, Levels A and B**  
**Item: Stress Intensity**

Temp. (Deg. F)	Classification and Value (ksi)					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	22.8	22.8	34.2	34.2	68.4	68.4
300	22.2	22.2	33.3	33.3	66.6	66.6
400	21.5	21.5	32.3	32.3	64.5	64.5
500	20.2	20.2	30.3	30.3	60.6	60.6
600	18.5	18.5	27.75	27.75	55.5	55.5
700	16.8	16.8	25.2	25.2	50.4	50.4

## Notes:

1. Source for  $S_m$  is ASME Code
2. Limits on values are presented in Table 2.2.10.
3.  $S_m$ ,  $P_m$ ,  $P_L$ ,  $P_b$ ,  $Q$ , and  $P_e$  are defined in Table 3.1.6.
4. For stress intensities generated by Level B Service Loadings, allowable stress intensity values are 110% of Level A limits per NB-3223.

† Evaluation required for Design condition only.

†† Evaluation required for Levels A and B conditions only.  $P_e$  not applicable to vessels.

**TABLE 3.1.9****LEVEL D, STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA350-LF3  
**Service Conditions:** Level D  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	$P_m$	$P_L$	$P_L + P_b$
-20 to 100	49.0	70.0	70.0
200	48.0	68.5	68.5
300	46.7	66.7	66.7
400	45.2	64.6	64.6
500	42.5	60.7	60.7
600	38.9	58.4	58.4
700	35.3	53.1	53.1

## Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed  $0.42 S_u$ .
3. Limits on values are presented in Table 2.2.10.
4.  $P_m$ ,  $P_L$ , and  $P_b$  are defined in Table 3.1.6.

**TABLE 3.1.10****DESIGN AND LEVEL A: STRESS**

**Code:** ASME NF  
**Material:** SA516, Grade 70, SA350-LF3, SA203-E  
**Service Conditions:** Design and Level A  
**Item:** Stress

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S</b>	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	17.5	17.5	26.3
700	16.6	16.6	24.9

## Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Limits on values are presented in Table 2.2.12.

**TABLE 3.1.11****LEVEL B: STRESS**

**Code:** ASME NF  
**Material:** SA516, Grade 70, SA350-LF3, and SA203-E  
**Service Conditions:** Level B  
**Item:** Stress

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>	
	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	23.3	34.9
700	22.1	33.1

## Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.10.

**TABLE 3.1.12****LEVEL D: STRESS INTENSITY**

**Code:** ASME NF  
**Material:** SA516, Grade 70  
**Service Conditions:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S<sub>m</sub></b>	<b>P<sub>m</sub></b>	<b>P<sub>m</sub> + P<sub>b</sub></b>
-20 to 100	23.3	45.6	68.4
200	23.1	41.5	62.3
300	22.5	40.4	60.6
400	21.7	39.1	58.7
500	20.5	36.8	55.3
600	18.7	33.7	50.6
650	18.4	33.1	49.7
700	18.3	32.9	49.3

## Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S<sub>m</sub> = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. Limits on values are presented in Table 2.2.12.
4. P<sub>m</sub> and P<sub>b</sub> are defined in Table 3.1.6.

**TABLE 3.1.13**  
**DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** Alloy X  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Numerical Value					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	20.0	20.0	30.0	30.0	60.0	60.0
200	20.0	20.0	30.0	30.0	60.0	60.0
300	20.0	20.0	30.0	30.0	60.0	60.0
400	18.7	18.7	28.1	28.1	56.1	56.1
500	17.5	17.5	26.3	26.3	52.5	52.5
600	16.4	16.4	24.6	24.6	49.2	49.2
650	16.0	16.0	24.0	24.0	48.0	48.0
700	15.6	15.6	23.4	23.4	46.8	46.8
750	15.2	15.2	22.8	22.8	45.6	45.6
800	14.9	14.9	22.4	22.4	44.7	44.7

## Notes:

1.  $S_m$  = Stress intensity values per Table 2A of ASME II, Part D for austenitic stainless steels of Alloy X and Appendix 1.A for duplex stainless steel of Alloy X.
2. Alloy X  $S_m$  values are the lowest values for each of the candidate materials at temperature.
3. Stress classification per NB-3220.
4. Limits on values are presented in Table 2.2.10.
5.  $P_m$ ,  $P_L$ ,  $P_b$ ,  $Q$ , and  $P_e$  are defined in Table 3.1.6.
6. For stress intensities generated by Level B Service Loadings, allowable stress intensity values are 110% of Level A limits per NB-3223.

† Evaluation required for Design condition only.

†† Evaluation required for Levels A, B conditions only.  $P_e$  not applicable to vessels.



**TABLE 3.1.14****LEVEL D: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** Alloy X  
**Service Conditions:** Level D  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	$P_m$	$P_L$	$P_L + P_b$
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

## Notes:

1. Level D stress intensities per ASME NB-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed  $0.42 S_u$ .
3. Limits on values are presented in Table 2.2.10.
4.  $P_m$ ,  $P_L$ , and  $P_b$  are defined in Table 3.1.6.

**TABLE 3.1.15**  
**DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NG  
**Material:** Alloy X  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)				
	$S_m$	$P_m$	$P_m+P_b$	$P_m+P_b+Q$	$P_e$
-20 to 100	20.0	20.0	30.0	60.0	60.0
200	20.0	20.0	30.0	60.0	60.0
300	20.0	20.0	30.0	60.0	60.0
400	18.7	18.7	28.1	56.1	56.1
500	17.5	17.5	26.3	52.5	52.5
600	16.4	16.4	24.6	49.2	49.2
650	16.0	16.0	24.0	48.0	48.0
700	15.6	15.6	23.4	46.8	46.8
750	15.2	15.2	22.8	45.6	45.6
800	14.9	14.9	22.4	44.7	44.7

## Notes:

1.  $S_m$  = Stress intensity values per Table 2A of ASME, Section II, Part D for austenitic stainless steels of Alloy X and Appendix 1.A for duplex stainless steel of Alloy X.
2. Alloy X  $S_m$  values are the lowest values for each of the candidate materials at temperature.
3. Classifications per NG-3220.
4. Limits on values are presented in Table 2.2.11.
5.  $P_m$ ,  $P_b$ ,  $Q$ , and  $P_e$  are defined in Table 3.1.6.
6. For stress intensities generated by Level B Service Loadings, allowable stress intensity values are 110% of Level A limits per NG-3223.

**TABLE 3.1.16****LEVEL D: STRESS INTENSITY**

**Code:** ASME NG  
**Material:** Alloy X  
**Service Conditions:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>P<sub>m</sub></b>	<b>P<sub>L</sub></b>	<b>P<sub>L</sub> + P<sub>b</sub></b>
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

## Notes:

1. Level D stress intensities per ASME NG-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed  $0.42 S_u$ .
3. Limits on values are presented in Table 2.2.11.
4.  $P_m$ ,  $P_L$ , and  $P_b$  are defined in Table 3.1.6.

**TABLE 3.1.17**  
**REFERENCE TEMPERATURES AND STRESS LIMITS**  
**FOR THE VARIOUS LOAD CASES**

Load Case I.D.	Material	Reference Temperature <sup>†</sup> , ° F	Stress Intensity Allowables, ksi		
			P <sub>m</sub>	P <sub>L</sub> + P <sub>b</sub>	P <sub>L</sub> + P <sub>b</sub> + Q
F1	Alloy X	725	15.4	23.1	46.2
F2	Alloy X	725	15.4	23.1	46.2
F3	Alloy X	725	36.9	55.4	NL
E1	Alloy X	500	17.5	26.3	52.5
E2	Alloy X	500	17.5	26.3	52.5
E3	Alloy X	500	42.0	63.0	NL <sup>††</sup>
E4	Alloy X	500	17.5	26.3	52.5
E5	Alloy X	775	36.15	54.25	NL

## Notes:

1. Q, P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.6.
2. Reference temperatures for Load Cases E1-E4 are for MPC shell; for MPC lid and MPC baseplate, reference temperatures are 550 deg. F and 400 deg. F, respectively (per Table 2.2.3) and stress intensity allowables should be adjusted accordingly.

<sup>†</sup> Values for reference temperatures are chosen to bound the thermal results in Chapter 4. Lower temperature values may be used provided that they are at least equal to the calculated temperature for the specific component and location or otherwise justified.

<sup>††</sup> NL: No specified limit in the Code

**TABLE 3.1.17 (CONTINUED)****REFERENCE TEMPERATURES AND STRESS LIMITS FOR THE VARIOUS LOAD CASES**

Load Case I.D.	Material	Reference Temperature, <sup>†,††</sup> ° F	Stress Intensity Allowables, ksi		
			P <sub>m</sub>	P <sub>L</sub> + P <sub>b</sub>	P <sub>L</sub> + P <sub>b</sub> + Q
O1	SA203-E	400	17.5	26.3	NL <sup>†††</sup>
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O2	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL
O3	SA203-E	400	17.5	26.3	NL
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O4	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL

## Note:

1. P<sub>m</sub>, P<sub>L</sub>, P<sub>b</sub>, and Q are defined in Table 3.1.6.
2. Load Cases O1 and O3 are for Normal Conditions; therefore the values listed refer to allowable stress, not allowable stress intensity

† Values for reference temperatures are chosen to bound the thermal results in Chapter 4. Lower temperature values may be used provided that they are at least equal to the calculated temperature for the specific component and location or otherwise justified.

†† For storage fire analysis, temperatures are defined by thermal solution

††† NL: No specified limit in the Code

**TABLE 3.1.18  
FRACTURE TOUGHNESS TEST REQUIREMENTS**

<b>Material</b>	<b>Test Requirement</b>	<b>Test Temperature</b>	<b>Acceptance Criterion</b>
Bolting (SA193 B7)	Not required (per NF-2311(b)(13) and Note (e) to Figure NF-2311(b)-1)	-	-
Ferritic steel with nominal section thickness of 5/8" or less	Not required per NF-2311(b)(1)	-	-
SA36 (thickness greater than 5/8")	Not required per NF-2311(b)(7)	-	-
Normalized SA516 Gr. 70 (thickness greater than 5/8", but less than or equal to 2-1/2"), except when used for HI-STORM 100A baseplate, lug support ring, and gussets	Not required per NF-2311(b)(7), NF-2311(b)(13), and curve D in Figure NF-2311(b)-1	-	-
HI-STORM 100A baseplate, lug support ring, and gussets (See Note 2)	Per NF-2331	See Note 1. (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2 (Also must meet ASME Section IIA requirements)
As-rolled SA516 Gr. 70, except when used for HI-STORM inner and outer shells, top plate, lid shear ring, lid shield ring, lid outer ring (for 100S Version B), and lid cover plate (for 100S Version B)	Per NF-2331	See Note 1. (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2 (Also must meet ASME Section IIA requirements)
As-rolled SA516 Gr. 70 used for HI-STORM inner and outer shells, top plate, lid shear ring, lid shield ring, lid outer ring (for 100S Version B), and lid cover plate (for 100S Version B) and Base- Bottom Plate (for 100S Version B)	Not required per NF-2311(b)(7)	-	-
SA203, SA515 Gr. 70, SA350-LF2, SA350-LF3 (thickness greater than	Per NF-2331	See Note 1. (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2

5/8")			(Also must meet ASME Section IIA requirements)
Weld material	Test per NF-2430 if: 1) either of the base materials of the production weld requires impact testing, or; 2) either of the base materials is SA516 Gr. 70 with nominal section thickness greater than 5/8".	See Note 1	Per NF-2330

## Notes:

1. Required NDT temperature = -40 deg. F for all materials in the HI-STORM 100 Overpack, -40 deg. F for HI-TRAC "STH" materials, and 0 deg. F for HI-TRAC "NF" materials.
2. In accordance with ASME Code Subsection NF, impact testing is not required for these components; specified testing is performed strictly for defense-in-depth.

**TABLE 3.1.19****DESIGN AND LEVEL A: STRESS**

**Code:** ASME NF  
**Material:** SA36  
**Service Conditions:** Design and Level A  
**Item:** Stress

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S</b>	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	14.5	14.5	21.8
700	13.9	13.9	20.9

## Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Limits on values are presented in Table 2.2.12.



**TABLE 3.1.20****LEVEL B: STRESS**

**Code:** ASME NF  
**Material:** SA36  
**Service Conditions:** Level B  
**Item:** Stress

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>	
	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	19.3	28.9
700	18.5	27.7

## Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.19.

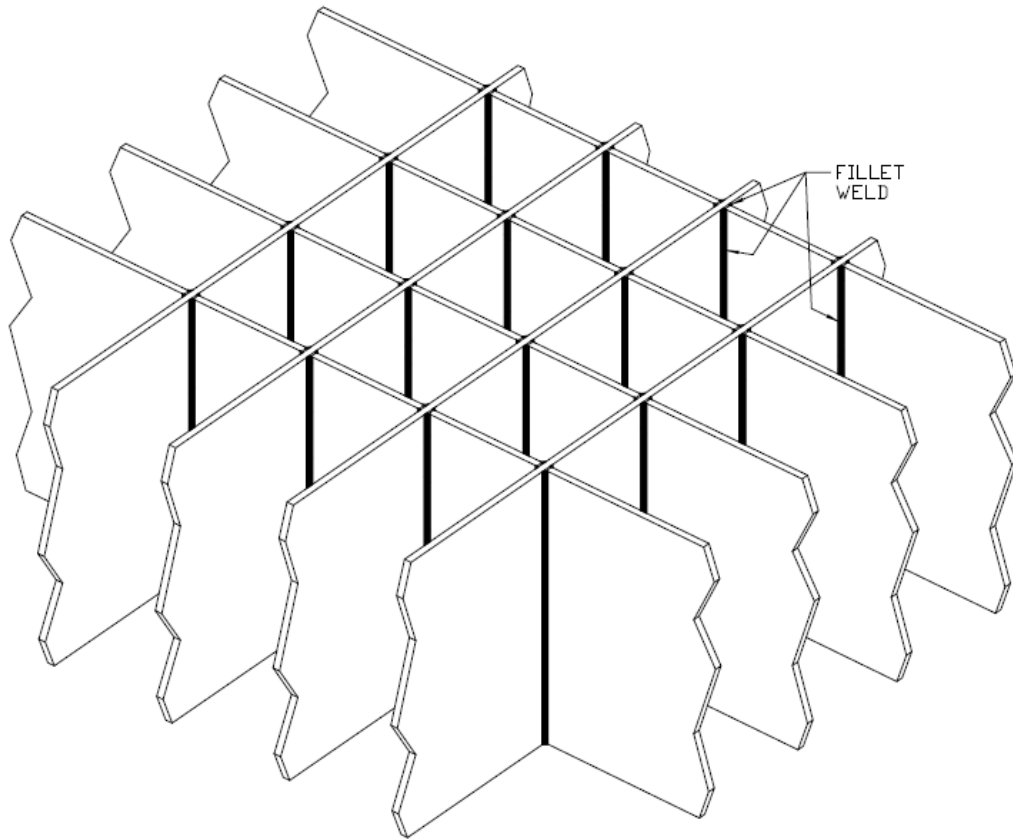
**TABLE 3.1.21****LEVEL D: STRESS INTENSITY**

**Code:** ASME NF  
**Material:** SA36  
**Service Conditions:** Level D  
**Item:** Stress Intensity

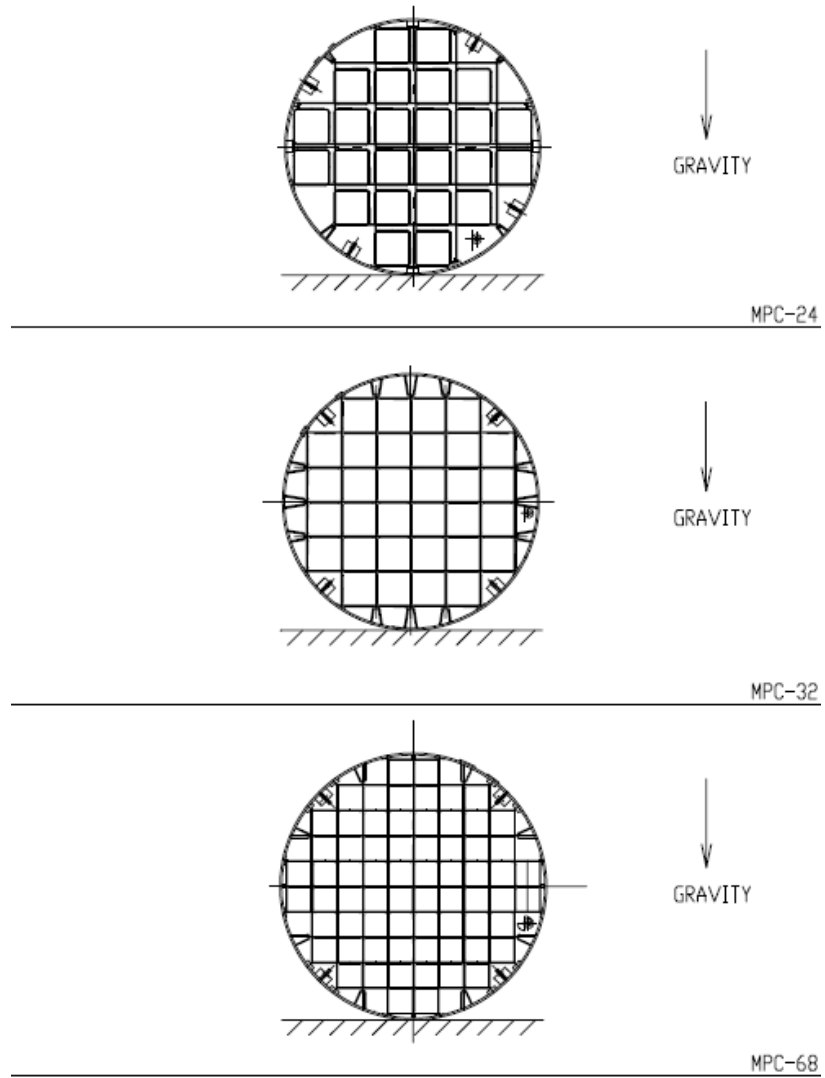
<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S<sub>m</sub></b>	<b>P<sub>m</sub></b>	<b>P<sub>m</sub> + P<sub>b</sub></b>
-20 to 100	19.3	43.2	64.8
200	19.3	37.0	55.5
300	19.3	36.0	54.0
400	19.3	34.7	52.1
500	19.3	32.8	49.2
600	17.7	30.0	45.0
650	17.4	29.5	44.3
700	17.3	29.2	43.8

## Notes:

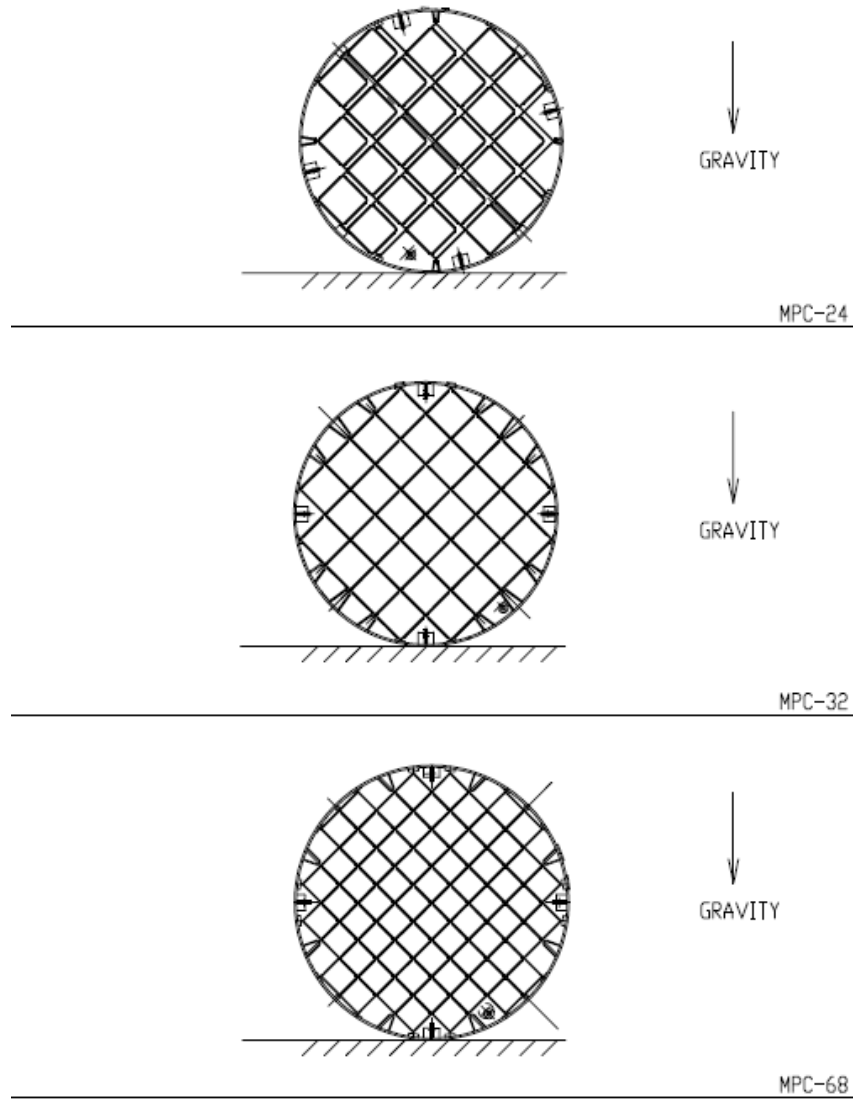
1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S<sub>m</sub> = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. Limits on values are presented in Table 2.2.12.
4. P<sub>m</sub> and P<sub>b</sub> are defined in Table 3.1.6.



**FIGURE 3.1.1: MPC-68 AND MPC-32 FUEL BASKET GEOMETRY**



**FIGURE 3.1.2: 0° DROP ORIENTATIONS FOR THE MPCs**



**FIGURE 3.1.3: 45° DROP ORIENTATIONS FOR THE MPCs**

### 3.2 WEIGHTS AND CENTERS OF GRAVITY

Tables 3.2.1 and 3.2.2 provide the calculated weights of the individual HI-STORM 100 components as well as the total system weights. The actual weights will vary within a narrow range of the calculated values due to the tolerances in metal manufacturing and fabrication permitted by the ASME Codes. Contained water mass during fuel loading is not included in this table.

The locations of the calculated centers of gravity (CGs) are presented in Table 3.2.3. All centers of gravity are located on the cask centerline since the non-axisymmetric effects of the cask system plus contents are negligible.

Table 3.2.4 provides the lift weight when the HI-TRAC transfer cask with the heaviest fully loaded MPC is being lifted from the fuel pool. The effect of buoyancy is neglected, and the weight of rigging is set at a conservative value.

In all weight tables, bounding values are also listed where necessary for use in structural calculations where their use will provide a conservative result.

**TABLE 3.2.1  
HI-STORM OVERPACK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>MPC-24</b>	
• Without SNF	42,000
• Fully loaded with SNF and Fuel Spacers	90,000 <sup>†††</sup>
<b>MPC-32</b>	
• Without SNF	36,000
• Fully loaded with SNF and Fuel Spacers	90,000 <sup>†††</sup>
<b>MPC-68/68F/68FF</b>	
• Without SNF	39,000
• Fully loaded with SNF and Fuel Spacers	90,000 <sup>†††</sup>
<b>MPC-24E/EF</b>	
• Without SNF	45,000
• Fully loaded with SNF and Fuel Spacers	90,000 <sup>†††</sup>
<b>HI-STORM 100 Overpack<sup>†</sup></b>	
• Overpack top lid	23,000
• Overpack w/ lid (empty)	270,000
• Overpack w/ fully loaded MPC-24	360,000
• Overpack w/ fully loaded MPC-32	360,000
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000
• Overpack w/ fully loaded MPC-24E/EF	360,000
<b>HI-STORM 100S(232) Overpack<sup>†</sup></b>	
• Overpack top lid	25,500 <sup>††</sup>
• Overpack w/ lid (empty)	270,000
• Overpack w/ fully loaded MPC-24	360,000
• Overpack w/ fully loaded MPC-32	360,000
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000
• Overpack w/ fully loaded MPC-24E/EF	360,000

**TABLE 3.2.1 (CONTINUED)  
HI-STORM OVERPACK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>HI-STORM 100S(243) Overpack<sup>†</sup></b>	
• Overpack top lid	25,500 <sup>††</sup>
• Overpack w/ lid (empty)	270,000
• Overpack w/ fully loaded MPC-24	360,000
• Overpack w/ fully loaded MPC-32	360,000
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000
• Overpack w/ fully loaded MPC-24E/EF	360,000
<b>HI-STORM 100A Overpack<sup>†</sup></b>	Same as above
<b>HI-STORM 100S Version B(218) Overpack (values in parentheses use high density concrete in overpack body)</b>	
• Overpack top lid	29,000
• Overpack w/ lid (empty)	270,000 (305,000)
• Overpack w/ fully loaded MPC-24	360,000 (395,000)
• Overpack w/ fully loaded MPC-32	360,000 (395,000)
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000 (395,000)
• Overpack w/ fully loaded MPC-24E/EF	360,000 (395,000)
<b>HI-STORM 100S Version B(229) Overpack (values in parentheses use high density concrete in overpack body)</b>	
• Overpack top lid	29,000
• Overpack w/ lid (empty)	270,000 (320,000)
• Overpack w/ fully loaded MPC-24	360,000 (410,000)
• Overpack w/ fully loaded MPC-32	360,000 (410,000)
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000 (410,000)
• Overpack w/ fully loaded MPC-24E/EF	360,000 (410,000)

<sup>†</sup> The bounding weights for the HI-STORM 100S(232) and 100S(243) overpacks listed in the above table are based on a maximum concrete (dry) density of 160.8 pcf. For improved shielding effectiveness, higher density concrete (up to 200 pcf dry) can be poured in the radial cavity of each of the HI-STORM 100S overpacks. At 200 pcf, the bounding weights of an empty overpack and a fully loaded overpack increase to 320,000 lb and 410,000 lb, respectively. Higher density concrete cannot be used in the HI-STORM 100 or 100A overpacks.

<sup>††</sup> Based on a maximum concrete (dry) density of 155 pcf. For improved shielding effectiveness, higher density concrete (up to 200 pcf dry) can be poured in the HI-STORM 100S lids. At 200 pcf, the bounding weight of the lid increases to 28,000 lb.

<sup>†††</sup> Based on the following maximum fuel assembly weights (as applicable):  
 1,680 lb per assembly (including non-fuel hardware) for PWR fuel that requires fuel spacers  
 1,720 lb per assembly (including non-fuel hardware) for PWR fuel that does not require fuel spacers  
 830 lb per assembly (including channels and DFCs) for BWR fuel



**TABLE 3.2.2**  
**HI-TRAC 125 TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	2,750
Pool Lid	12,500
Transfer Lid	24,500
HI-TRAC 125 w/ Top Lid and Pool Lid (water jacket filled)	143,500
HI-TRAC 125 w/ Top Lid and Transfer Lid (water jacket filled)	155,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	226,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	233,500
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF(water jacket filled)	231,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	229,000
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	237,500
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	245,000
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	242,500
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	240,500

**TABLE 3.2.2 (CONTINUED)**  
**HI-TRAC 100 TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	1,500
Pool Lid	8,000
Transfer Lid	17,000
HI-TRAC 100 w/ Top Lid and Pool Lid (water jacket filled)	102,000
HI-TRAC 100 w/ Top Lid and Transfer Lid (water jacket filled)	111,000
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	183,500
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	191,000
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	188,500
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	186,500
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	192,000
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	199,000
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	196,500
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	194,500

**TABLE 3.2.2 (CONTINUED)**  
**HI-TRAC 125D TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	2,750
Pool Lid	12,500
HI-TRAC 125D w/ Top Lid and Pool Lid (water jacket filled)	146,000
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	228,500
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	236,000
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	233,500
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	231,500

**TABLE 3.2.2 (CONTINUED)**  
**HI-TRAC 100D TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	1,500
Pool Lid	8,000
HI-TRAC 100D w/ Top Lid and Pool Lid (water jacket filled)	102,000
HI-TRAC 100D w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	183,500
HI-TRAC 100D w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	191,000
HI-TRAC 100D w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	188,500
HI-TRAC 100D w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	186,500

**TABLE 3.2.3**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

<b>Component</b>	<b>Height of CG Above Datum<sup>†</sup> (in)</b>
MPC-24 (empty)	109.0
MPC-32 (empty)	113.2
MPC-68/68F/68FF (empty)	111.5
MPC-24E/EF (empty)	108.9
HI-STORM 100 Overpack (empty)	116.8
HI-STORM 100S(232) Overpack (empty)	111.7
HI-STORM 100S(243) Overpack (empty)	117.4
HI-STORM 100S Version B(218) Overpack (empty)(height using standard weight concrete bounds height calculated using high density concrete)	108.77(108.45)
HI-STORM 100S Version B(229) Overpack (empty)(height using standard weight concrete bounds height calculated using high density concrete)	114.27(113.94)
HI-STORM 100 Overpack w/ fully loaded MPC-24	118.8
HI-STORM 100 Overpack w/ fully loaded MPC-32	118.7
HI-STORM 100 Overpack w/ fully loaded MPC-68/68F/68FF	119.0
HI-STORM 100 Overpack w/ fully loaded MPC-24E/EF	119.2
HI-STORM 100S(232) Overpack w/ fully loaded MPC-24	113.8
HI-STORM 100S(232) Overpack w/ fully loaded MPC-32	113.7
HI-STORM 100S(232) Overpack w/ fully loaded MPC-68/68F/68FF	114.0
HI-STORM 100S(232) Overpack w/ fully loaded MPC-24E/EF	114.2
HI-STORM 100S(243) Overpack w/ fully loaded MPC-24	118.1
HI-STORM 100S(243) Overpack w/ fully loaded MPC-32	117.9
HI-STORM 100S(243) Overpack w/ fully loaded MPC-68/68F/68FF	118.2
HI-STORM 100S(243) Overpack w/ fully loaded MPC-24E/EF	118.4
HI-STORM 100S Version B(218) Overpack w/ fully loaded MPC-24	110.83
HI-STORM 100S Version B(218) Overpack w/ fully loaded MPC-32	111.88
HI-STORM 100S Version B(218) Overpack w/ fully loaded MPC-68/68F/68FF	111.45
HI-STORM 100S Version B(218) Overpack w/ fully loaded MPC-24E/EF	110.80

<sup>†</sup> See notes at end of table.

**TABLE 3.2.3 (CONTINUED)**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

<b>Component</b>	<b>Height of CG Above Datum<sup>†</sup> (in)</b>
HI-STORM 100S Version B(229) Overpack w/ fully loaded MPC-24	114.95
HI-STORM 100S Version B(229) Overpack w/ fully loaded MPC-32	116.00
HI-STORM 100S Version B(229) Overpack w/ fully loaded MPC-68/68F/68FF	115.58
HI-STORM 100S Version B(229) Overpack w/ fully loaded MPC-24E/EF	114.93
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	99.5
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	99.5
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	99.8
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	100.1
HI-TRAC 100 Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	91.0
HI-TRAC 100 Transfer Cask w/ Top Lid and Transfer Lid (water jacket filled)	91.1
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	97.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	97.2
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	97.6
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	98.0
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	100.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	100.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	100.7
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	101.0

<sup>†</sup> See notes at end of table.

**TABLE 3.2.3 (CONTINUED)**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

<b>Component</b>	<b>Height of CG Above Datum (in)</b>
HI-TRAC 125D Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	92.4
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	97.6
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	97.5
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	97.8
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	98.2
HI-TRAC 100D Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	87.0
HI-TRAC 100D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	99.2
HI-TRAC 100D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	101.2
HI-TRAC 100D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	100.4
HI-TRAC 100D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	99.1

## Notes:

1. The datum used for calculations involving the HI-STORM is the bottom of the overpack baseplate. The datum used for calculations involving the HI-TRAC is the bottom of the pool lid or transfer lid, as appropriate.
2. The datum used for calculations involving only the MPC is the bottom of the MPC baseplate.
3. The CG heights of the HI-STORM overpacks are calculated based on standard density concrete (i.e., 150 pcf dry). At higher densities, the CG heights are slightly lower, which makes the HI-STORM overpacks less prone to tipping.

**TABLE 3.2.4  
LIFT WEIGHT ABOVE POOL WITH HI-TRAC 125**

Item	Estimated Weight (lb)	Bounding Weight (lb)
HI-TRAC 125 w/ Top Lid and Pool Lid (water jacket filled)	142,976	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
HI-TRAC 125 Top Lid	-2,730 <sup>††</sup>	
Water in MPC and HI-TRAC 125 Annulus	16,570	
Water in Water Jacket	-9,757 <sup>†††</sup>	
Lift yoke	4,200	
Inflatable annulus seal	50	
<b>TOTAL</b>	241,074	250,000

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 125 includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.



**TABLE 3.2.4 (CONTINUED)**  
**LIFT WEIGHT ABOVE POOL WITH HI-TRAC 100**

Item	Estimated Weight (lb)	Maximum Weight (lb)
HI-TRAC 100 w/ Top Lid and Pool Lid (water jacket filled)	100,194	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
MPC closure ring	-140	
HI-TRAC 100 Top Lid	-1,203 <sup>††</sup>	
Water in MPC and HI-TRAC 100 Annulus	16,570	
Water in Water Jacket	-7,562 <sup>†††</sup>	
Lift yoke	3,200	
Inflatable annulus seal	50	
<b>TOTAL</b>	200,874 <sup>††††</sup>	200,000

Note: HI-TRAC transfer cask weight is without removable portion of pocket trunnion.

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 100 includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

<sup>††††</sup> Under worst case conditions, removal of a portion of water from the MPC may be required to reduce the in-pool lift weight.

**TABLE 3.2.4 (CONTINUED)**  
**LIFT WEIGHT ABOVE POOL WITH HI-TRAC 125D**

Item	Estimated Weight (lb)	Bounding Weight (lb)
HI-TRAC 125D w/ Top Lid and Pool Lid (water jacket filled)	145,635	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
HI-TRAC 125D Top Lid	-2,575 <sup>††</sup>	
Water in MPC and HI-TRAC 125D Annulus	16,570	
Water in Water Jacket	-8,955 <sup>†††</sup>	
Lift yoke	4,200	
Inflatable annulus seal	50	
<b>TOTAL</b>	244,690	250,000

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 125D includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

**TABLE 3.2.4 (CONTINUED)**  
**LIFT WEIGHT ABOVE POOL WITH HI-TRAC 100D**

Item	Estimated Weight (lb)	Maximum Weight (lb)
HI-TRAC 100D w/ Top Lid and Pool Lid (water jacket filled)	99,106	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
MPC closure ring	-140	
HI-TRAC 100D Top Lid	-1,213 <sup>††</sup>	
Water in MPC and HI-TRAC 100D Annulus	16,570	
Water in Water Jacket	-7,665 <sup>†††</sup>	
Lift yoke	4,200	
Inflatable annulus seal	50	
<b>TOTAL</b>	200,673 <sup>††††</sup>	200,000

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 100D includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

<sup>††††</sup> Under worst case conditions, removal of a portion of water from the MPC may be required to reduce the in-pool lift weight.

### 3.3 MECHANICAL PROPERTIES OF MATERIALS

Table 2.2.6 provides a comprehensive listing of materials of construction, applicable code, and ITS designation for all functional parts in the HI-STORM 100 System. This section provides the mechanical properties used in the structural evaluation. The properties include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. Values are presented for a range of temperatures which envelopes the maximum and minimum temperatures under all service conditions discussed in the preceding section where structural analysis is performed.

The materials selected for use in the MPC, HI-STORM 100 Overpack, and HI-TRAC transfer cask are presented in the Bills-of-Material in Section 1.5. In this chapter, the materials are divided into two categories, structural and nonstructural. Structural materials are materials that act as load bearing members and are, therefore, significant in the stress evaluations. Materials that do not support mechanical loads are considered nonstructural. For example, the HI-TRAC inner shell is a structural material, while the lead between the inner and outer shell is a nonstructural material. For nonstructural materials, the only property that is used in the structural analysis is weight density. In local deformation analysis, however, such as the study of penetration from a tornado-borne missile, the properties of lead in HI-TRAC and plain concrete in HI-STORM 100 are included.

#### 3.3.1 Structural Materials

##### 3.3.1.1 Alloy X

A hypothetical material termed Alloy X is defined for all MPC structural components. The material properties of Alloy X are the least favorable values from the set of candidate alloys. The purpose of a least favorable material definition is to ensure that all structural analyses are conservative, regardless of the actual MPC material. For example, when evaluating the stresses in the MPC, it is conservative to work with the minimum values for yield strength and ultimate strength. This guarantees that the material used for fabrication of the MPC will be of equal or greater strength than the hypothetical material used in the analysis. In the structural evaluation, the only property for which it is not always conservative to use the set of minimum values is the coefficient of thermal expansion. Two sets of values for the coefficient of thermal expansion are specified, a minimum set and a maximum set. For each analysis, the set of coefficients, minimum or maximum that causes the more severe load on the cask system is used.

Table 3.3.1 lists the numerical values for the material properties of Alloy X versus temperature. These values, taken from the ASME Code, Section II, Part D [3.3.1] and Appendix 1.A, are used in all structural analyses. A minimum requirement on yield strength is set for MPC Lids in Table 3.3.1, to ensure that the lid internal threads have enough capacity under the lifted load as mandated by NUREG-0612. The maximum temperatures in some MPC components may exceed the long term normal allowable temperature limits during short term loading operations, off-normal events, or storage accident events. However, it is ensured that the maximum temperature of austenitic stainless steel grades of Alloy X used for or within the confinement boundary does not

exceed 1000°F under any condition and the maximum temperature of duplex stainless steel (UNS S31803) grade of Alloy X used for the confinement boundary does not exceed 600°F under any condition. As shown in ASME Code Case N-47-33 (Class 1 Components in Elevated Temperature Service, 1995 Code Cases, Nuclear Components), the strength properties of austenitic stainless steels do not change due to exposure to 1000°F temperature for up to 10,000 hours. In addition, per ASME Code Case N-635-1 (Use of 22Cr-5Ni-3Mo-N (Alloy UNS S31803) Forgings, Plate, Bar, Welded and Seamless Pipe, and/or Tube, Fittings, and Fusion Welded Pipe with Additional of Filler Metal, Classes 1, 2, and 3, Section III, Division 1), the maximum permissible temperature for duplex stainless steel grade of Alloy X is 600°F. Therefore, there is no significant effect on mechanical properties of the confinement or basket material during the short time duration loading. It is noted that duplex stainless steel material shall not be used for the fabrication of MPC baskets and MPC internal components. A further description of Alloy X, including the materials from which it is derived, is provided in Appendix 1.A.

Two properties of Alloy X that are not included in Table 3.3.1 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses, regardless of temperature. The values used are shown in the table below.

<b>PROPERTY</b>	<b>VALUE</b>
Weight Density (lb/in <sup>3</sup> )	0.290
Poisson's Ratio	0.30

### 3.3.1.2 Carbon Steel, Low-Alloy and Nickel Alloy Steel

The carbon steels used in the structural qualification of the HI-STORM 100 System are SA516 Grade 70, SA515 Grade 70, and SA36. The nickel alloy and low alloy steels are SA203-E and SA350-LF3, respectively. These steels are not constituents of Alloy X. The material properties of SA516 Grade 70 and SA515 Grade 70 are shown in Tables 3.3.2. The material properties of SA203-E and SA350-LF3 are given in Table 3.3.3. The material properties of SA36 are shown in Table 3.3.6.

Two properties of these steels that are not included in Tables 3.3.2, 3.3.3 and 3.3.6 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses. The values used are shown in the table below.

<b>PROPERTY</b>	<b>VALUE</b>
Weight Density (lb/in <sup>3</sup> )	0.283
Poisson's Ratio	0.30

### 3.3.1.3 Bolting Materials

Material properties of the bolting materials used in the HI-STORM 100 System and HI-TRAC lifting trunnions are given in Table 3.3.4. The properties of representative anchor studs used to fasten HI-STORM 100A are listed in Table 1.2.7.

### 3.3.1.4 Weld Material

All weld materials utilized in the welding of the Code components comply with the provisions of the appropriate ASME subsection (e.g., Subsection NB for the MPC enclosure vessel) and Section IX. All non-code welds will be made using weld procedures that meet the requirements specified in the licensing drawings. The minimum tensile strength of the weld wire and filler material (where applicable) will be equal to or greater than the tensile strength of the base metal listed in the ASME Code.

## 3.3.2 Nonstructural Materials

### 3.3.2.1 Solid Neutron Shield

The solid neutron shielding material in the HI-TRAC top lid and transfer lid doors is not considered as a structural member of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analyses except where such omission would be non-conservative. The only material property of the solid neutron shield that is important to the structural evaluation is weight density ( $1.63\text{g/cm}^3$ ).

### 3.3.2.2 Solid Neutron Absorber

The fuel basket solid neutron absorber is not a structural member of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analyses. The only material property of the solid neutron absorber that is important to the structural evaluation is weight density. As the MPC fuel baskets can be constructed with neutron absorber panels of variable areal density, the weight that produces the most severe cask load is assumed in each analysis (density  $2.644\text{ g/cm}^3$ ).

### 3.3.2.3 Concrete

The primary function of the plain concrete in the HI-STORM storage overpack is shielding. Concrete in the HI-STORM 100 Overpack is not considered as a structural member, except to withstand compressive, bearing, and penetrant loads. While concrete is not considered a structural member, its mechanical behavior must be quantified to determine the stresses in the structural members (steel shells surrounding it) under accident conditions. Table 3.3.5 provides the concrete mechanical properties. Allowable, bearing strength in concrete for normal loading conditions is calculated in accordance with ACI 318.1 -89 (92) [3.3.2]. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. In addition, although the concrete is not reinforced (since the absence of reinforcement does not degrade the compressive strength), the requirements of ACI-349-85 [3.3.3] are imposed to

insure the suitability of the concrete mix. Appendix 1.D provides additional information on the requirements on plain concrete for use in HI-STORM 100 storage overpack.

To enhance the shielding performance of the HI-STORM storage overpack, high density concrete can be used during fabrication. The permissible range of concrete densities is specified in Table 1.D.1. The structural calculations consider the most conservative density value (i.e., maximum or minimum weight), as appropriate.

#### 3.3.2.4 Lead

Lead is not considered as a structural member of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analysis, except in the analysis of a tornado missile strike where it acts as a missile barrier. Applicable mechanical properties of lead are provided in Table 3.3.5.

#### 3.3.2.5 Aluminum Heat Conduction Elements

In early vintage MPCs, aluminum heat conduction elements may be located between the fuel basket and MPC vessel. They are optional thin flexible elements whose sole function is to transmit heat as described in Chapter 4. They are not credited with any structural load capacity and are shaped to provide negligible resistance to basket thermal expansion. The total weight of the aluminum inserts is less than 1,000 lb. per MPC.

**TABLE 3.3.1  
ALLOY X MATERIAL PROPERTIES**

Temp. (Deg. F)	Alloy X				
	$S_y^{\ddagger}$	$S_u^{\dagger}$	$\alpha_{\min}$	$\alpha_{\max}$	E
-40	30.0 (33.0)	75.0 (70.0)	8.54	8.55	28.82
100	30.0 (33.0)	75.0 (70.0)	8.54	8.55	28.14
150	27.5 (30.25)	73.0 (68.1)	8.64	8.67	27.87
200	25.0 (27.5)	71.0 (66.2)	8.76	8.79	27.6
250	23.75 (26.12)	68.5 (63.85)	8.88	8.9	27.3
300	22.5 (24.75)	66.0 (61.5)	8.97	9.0	27.0
350	21.6 (23.76)	65.2 (60.75)	9.10	9.11	26.75
400	20.7 (22.77)	64.4 (60.0)	9.19	9.21	26.5
450	20.05 (22.06)	64.0 (59.65)	9.28	9.32	26.15
500	19.4 (21.34)	63.5 (59.3)	9.37	9.42	25.8
550	18.8 (20.68)	63.3 (59.1)	9.45	9.50	25.55
600	18.2 (20.02)	63.1 (58.9)	9.53	9.6	25.3
650	17.8 (19.58)	62.8 (58.6)	9.61	9.69	25.05
700	17.3 (19.03)	62.5 (58.4)	9.69	9.76	24.8
750	16.9 (18.59)	62.2 (58.1)	9.76	9.81	24.45
800	16.6 (18.26)	61.7 (57.6)	9.82	9.90	24.1

## Definitions:

 $S_y$  = Yield Stress (ksi) $\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>) $S_u$  = Ultimate Stress (ksi)E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for  $S_y$  values is Table Y-1 of [3.3.1] for austenitic stainless steels of Alloy X and Appendix 1.A for duplex stainless steel of Alloy X.
2. Source for  $S_u$  values is Table U of [3.3.1] for austenitic stainless steels of Alloy X and Appendix 1.A for duplex stainless steel of Alloy X.

$\ddagger$  Values in the parentheses correspond to yield stress of MPC Lids which are 10% greater than the minimum yield stress values tabulated in Table 1.A.3. These higher values are only credited for the stress analysis of the MPC Lid lifting holes.

$\dagger$  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.



3. Source for  $\alpha_{\min}$  and  $\alpha_{\max}$  values is Table TE-1 of [3.3.1] for austenitic stainless steels of Alloy X. Values of  $\alpha$  for duplex stainless steel grade of Alloy X can be obtained from Appendix 1.A.
4. Source for E values is material group G in Table TM-1 of [3.3.1] for austenitic stainless steels of Alloy X and Appendix 1.A for duplex stainless steel of Alloy X.
5. Minimum values of  $S_y$ ,  $S_u$  and E from all candidate Alloy X materials are listed.
6. Duplex stainless steel grade of Alloy X is only used in fabrication of MPC confinement boundary components and its use is limited to 600 °F per Appendix 1.A.

**TABLE 3.3.2**  
**SA516 AND SA515, GRADE 70 MATERIAL PROPERTIES**

Temp. (Deg. F)	SA516 and SA515, Grade 70			
	S <sub>y</sub>	S <sub>u</sub>	α	E
-40	38.0	70.0	---	29.95
100	38.0	70.0	5.53 (5.73)	29.34
150	36.3	70.0	5.71 (5.91)	29.1
200	34.6	70.0	5.89 (6.09)	28.8
250	34.15	70.0	6.09 (6.27)	28.6
300	33.7	70.0	6.26 (6.43)	28.3
350	33.15	70.0	6.43 (6.59)	28.0
400	32.6	70.0	6.61 (6.74)	27.7
450	31.65	70.0	6.77 (6.89)	27.5
500	30.7	70.0	6.91 (7.06)	27.3
550	29.4	70.0	7.06 (7.18)	27.0
600	28.1	70.0	7.17 (7.28)	26.7
650	27.6	70.0	7.30 (7.40)	26.1
700	27.4	70.0	7.41 (7.51)	25.5
750	26.5	69.3	7.50 (7.61)	24.85

## Definitions:

S<sub>y</sub> = Yield Stress (ksi)

α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress (ksi)

E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α values is material group C in Table TE-1 of [3.3.1].
4. Source for E values is "Carbon steels with C less than or equal to 0.30%" in Table TM-1 of [3.3.1].
5. Values for SA515 are given in parentheses where different from SA516.

**TABLE 3.3.3**  
**SA350-LF3 AND SA203-E MATERIAL PROPERTIES**

Temp. (Deg. F)	SA350-LF3 and LF2			SA350-LF3/SA203-E		SA203-E		
	S <sub>m</sub>	S <sub>y</sub>	S <sub>u</sub>	E	α	S <sub>m</sub>	S <sub>y</sub>	S <sub>u</sub>
-20	23.3	37.5 (36.0)	70.0	28.2	---	23.3	40.0	70.0
100	23.3	37.5 (36.0)	70.0	27.6	6.27	23.3	40.0	70.0
200	22.8 (21.9)	34.2 (32.9)	68.5 (70.0)	27.1	6.54	23.3	36.5	70.0
300	22.2 (21.3)	33.2 (31.9)	66.7 (70.0)	26.7	6.78	23.3	35.4	70.0
400	21.5 (20.6)	32.2 (30.9)	64.6 (70.0)	26.1	6.98	22.9	34.3	68.8
500	20.2 (19.4)	30.3 (29.2)	60.7 (70.0)	25.7	7.16	21.6	32.4	64.9
600	18.5 (17.8)	- (26.6)	- (70.0)	-	-	-	-	-
700	16.8 (17.3)	- (26.0)	- (70.0)	-	-	-	-	-

## Definitions:

- S<sub>m</sub> = Design Stress Intensity (ksi)  
S<sub>y</sub> = Yield Stress (ksi)  
S<sub>u</sub> = Ultimate Stress (ksi)  
α = Coefficient of Thermal Expansion (in./in. per degree F x 10<sup>-6</sup>)  
E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>m</sub> values is ASME Code.
2. Source for S<sub>y</sub> values is ASME Code.
3. Source for S<sub>u</sub> values is ratioing S<sub>m</sub> values.
4. Source for α values is material group E in Table TE-1 of [3.3.1].
5. Source for E values is material group B in Table TM-1 of [3.3.1].
6. Values for LF2 are given in parentheses where different from LF3.

**TABLE 3.3.4  
BOLTING MATERIAL PROPERTIES**

SB637-N07718					
Temp. (Deg. F)	$S_y$	$S_u$	E	$\alpha$	$S_m$
-100	150.0	185.0	29.9	---	50.0
-20	150.0	185.0	---	---	50.0
70	150.0	185.0	29.0	7.05	50.0
100	150.0	185.0	---	7.08	50.0
200	144.0	177.6	28.3	7.22	48.0
300	140.7	173.5	27.8	7.33	46.9
400	138.3	170.6	27.6	7.45	46.1
500	136.8	168.7	27.1	7.57	45.6
600	135.3	166.9	26.8	7.67	45.1
SA193 Grade B7 (2.5 to 4 inches diameter)					
Temp. (Deg. F)	$S_y$	$S_u$	E	$\alpha$	-
100	95.0	115.00	-	5.73	-
200	88.5	107.13	-	6.09	-
300	85.1	103.02	-	6.43	-
400	82.3	99.63	-	5.9	-

## Definitions:

$S_m$  = Design stress intensity (ksi)

$S_y$  = Yield Stress (ksi)

$\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x  $10^{-6}$ )

$S_u$  = Ultimate Stress (ksi)

E = Young's Modulus (psi x  $10^6$ )

## Notes:

1. Source for  $S_m$  values is Table 4 of [3.3.1].
2. Source for  $S_y$  values is ratioing design stress intensity values.
3. Source for  $S_u$  values is ratioing design stress intensity values.
4. Source for  $\alpha$  values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
5. Source for E values is Table TM-1 of [3.3.1].
6. Source for  $S_y$  values for SA193 bolts is Table Y-1 of [3.3.1]; source for  $S_u$  is by ratioing  $S_y$ .

**TABLE 3.3.4 (CONTINUED)**  
**BOLTING MATERIAL PROPERTIES**

Temp. (Deg. F)	$S_y$	$S_u$	E	$\alpha$	$S_m$
SA193 Grade B7 (less than 2.5 inch diameter)					
100	105.0	125.00	-	5.73	-
200	98.0	116.67	-	6.09	-
300	94.1	112.02	-	6.43	
400	91.5	108.93	-	6.74	-
SA705-630/SA564-630 (Age Hardened at 1075 degrees F)					
200	115.6	145.0	28.5	5.9	---
300	110.7	145.0	27.9	5.9	---
400	106.9	145.0	27.3	5.91	---
SA705-630/SA564-630 (Age Hardened at 1100 degrees F)					
200	106.3	140.0	28.5	5.9	---
300	101.9	140.0	27.9	5.9	---
400	98.3	136.3	27.3	5.91	---
SA705-630/SA564-630 (Age Hardened at 1150 degrees F)					
200	97.1	135.0	28.5	5.9	---
300	93.0	135.0	27.9	5.9	---

## Definitions:

$S_m$  = Design stress intensity (ksi)

$S_y$  = Yield Stress (ksi)

$\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x  $10^{-6}$ )

$S_u$  = Ultimate Stress (ksi)

E = Young's Modulus (psi x  $10^6$ )

## Notes:

1. Source for  $S_y$  values is Table Y-1 of [3.3.1].
2. Source for  $S_u$  values is Table U of [3.3.1].
3. Source for  $\alpha$  values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
4. Source for E values is Table TM-1 of [3.3.1].

**TABLE 3.3.5  
CONCRETE AND LEAD MECHANICAL PROPERTIES**

PROPERTY	VALUE					
<b>CONCRETE:</b>						
Compressive Strength (psi)	See Table 1.D.1					
Nominal Density (lb/ft <sup>3</sup> )	See Table 1.D.1					
Allowable Bearing Stress (psi)	1,823 <sup>†</sup>					
Allowable Axial Compression (psi)	1,266 <sup>†</sup>					
Allowable Flexure, extreme fiber tension (psi)	187 <sup>†,††</sup>					
Allowable Flexure, extreme fiber compression (psi)	2,145 <sup>†</sup>					
Mean Coefficient of Thermal Expansion (in/in/deg. F)	5.5E-06					
Modulus of Elasticity (psi)	57,000 (compressive strength (psi)) <sup>1/2</sup>					
<b>LEAD:</b>	-40°F	-20°F	70°F	200°F	300°F	600°F
Yield Strength (psi)	700	680	640	490	380	20
Modulus of Elasticity (ksi)	2.4E+3	2.4E+3	2.3E+3	2.0E+3	1.9E+3	1.5E+3
Coefficient of Thermal Expansion (in/in/deg. F)	15.6E-6	15.7E-6	16.1E-6	16.6E-6	17.2E-6	20.2E-6
Poisson's Ratio	0.40					
Density (lb/cubic ft.)	708					

## Notes:

1. Concrete allowable stress values based on ACI 318.1-89 (92).
2. Lead properties are from [3.3.5].

<sup>†</sup> Values listed correspond to concrete compressive stress = 3,300 psi

<sup>††</sup> No credit for tensile strength of concrete is taken in the calculations

**TABLE 3.3.6**  
**SA36 AND CARBON STEEL MATERIAL PROPERTIES**

Temp. (Deg. F)	SA36 AND CARBON STEEL			
	S <sub>y</sub>	S <sub>u</sub>	α	E
-40	36.0	58.0	---	29.95
100	36.0	58.0	5.53	29.34
150	34.4	55.4	5.71	29.1
200	32.8	52.8	5.89	28.8
250	32.35	52.1	6.09	28.6
300	31.9	51.4	6.26	28.3
350	31.35	50.5	6.43	28.0
400	30.8	49.6	6.61	27.7
450	29.95	48.3	6.77	27.5
500	29.1	46.9	6.91	27.3
550	27.85	44.9	7.06	27.0
600	26.6	42.9	7.17	26.7
650	26.1	42.1	7.30	26.1
700	25.9	41.7	7.41	25.5

Definitions:

S<sub>y</sub> = Yield Stress (ksi)

α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress (ksi)

E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is ratioing S<sub>y</sub> values.
3. Source for α values is material group C in Table TE-1 of [3.3.1].
4. Source for E values is "Carbon steels with C less than or equal to 0.30%" in Table TM-1 of [3.3.1].

### 3.4 GENERAL STANDARDS FOR CASKS

#### 3.4.1 Chemical and Galvanic Reactions

In this section, it is shown that there is no credible mechanism for significant chemical or galvanic reactions in the HI-STORM 100 System during long-term storage operations (including HI-STORM 100S and HI-STORM 100A).

The MPC, which is filled with helium, provides a nonaqueous and inert environment. Insofar as corrosion is a long-term time-dependent phenomenon, the inert gas environment in the MPC precludes the incidence of corrosion during storage on the ISFSI. Furthermore, the only dissimilar material groups in the MPC are: (1) the neutron absorber material and stainless steel and (2) aluminum (found in some early vintage MPCs) and stainless steel. Neutron absorber materials and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from neutron absorber materials and stainless steel materials, with geometries similar to the MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This experience provides a sound basis to conclude that corrosion will not occur in these materials. Additionally, the aluminum conduction inserts and stainless steel basket are very close on the galvanic series chart. Aluminum, like other metals of its genre (e.g., titanium and magnesium) rapidly passivates in an aqueous environment, leading to a thin ceramic ( $\text{Al}_2\text{O}_3$ ) barrier, which renders the material essentially inert and corrosion-free over long periods of application. The physical properties of the material, e.g., thermal expansion coefficient, diffusivity, and thermal conductivity, are essentially unaltered by the exposure of the aluminum metal stock to an aqueous environment.

The aluminum in the optional heat conduction elements (found in some early vintage MPCs) will quickly passivate in air and in water to form a protective oxide layer that prevents any significant hydrogen production during MPC cask loading and unloading operations. The aluminum in the neutron absorber material may also react with the water to generate hydrogen gas. The exact rate of generation and total amount of hydrogen generated is a function of a number of variables (see Section 1.2.1.3.1) and cannot be predicted with any certainty. Therefore, to preclude the potential for hydrogen ignition during lid welding or cutting, the operating procedures in Chapter 8 require monitoring for combustible gas and purging the space beneath the MPC lid with an inert gas during these activities. Once the MPC cavity is drained, dried, and backfilled with helium, the source of the hydrogen gas (the aluminum-water reaction) is eliminated.

The HI-STORM 100 storage overpack and the HI-TRAC transfer cask each combine low alloy and nickel alloy steels, carbon steels, neutron and gamma shielding materials, and bolting materials. All of these materials have a long history of non-galvanic behavior within close proximity of each other. The internal and external steel surfaces of each of the storage overpacks are sandblasted and coated to preclude surface oxidation. The HI-TRAC coating does not chemically react with borated water. Therefore, chemical or galvanic reactions involving the storage overpack materials are highly unlikely and are not expected.

In accordance with NRC Bulletin 96-04 [3.4.7], a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-STORM 100 System, its contents and the



operating environments, which may produce adverse reactions, has been performed. Table 3.4.2 provides a listing of the materials of fabrication for the HI-STORM 100 System and evaluates the performance of the material in the expected operating environments during short-term loading/unloading operations and long-term storage operations. As a result of this review, no operations were identified which could produce adverse reactions beyond those conditions already analyzed in this FSAR.

### 3.4.2 Positive Closure

There are no quick-connect/disconnect ports in the confinement boundary of the HI-STORM 100 System. The only access to the MPC is through the storage overpack lid, which weighs over 23,000 pounds (see Table 3.2.1). The lid is fastened to the storage overpack with large bolts. Inadvertent opening of the storage overpack is not feasible; opening a storage overpack requires mobilization of special tools and heavy-load lifting equipment.

### 3.4.3 Lifting Devices

As required by Reg. Guide 3.61, in this subsection, analyses for all lifting operations applicable to the deployment of a member of the HI-STORM 100 family are presented to demonstrate compliance with applicable codes and standards.

The HI-STORM 100 System has the following components and devices participating in lifting operations: lifting trunnions located at the top of the HI-TRAC transfer cask, lid lifting connections for the HI-STORM 100 lid and for other lids in the HI-TRAC transfer cask, connections for lifting and carrying a loaded HI-STORM 100 vertically, and lifting connections for the loaded MPC.

Analyses of HI-STORM 100 storage overpack and HI-TRAC transfer cask lifting devices are reported in this submittal. Analyses of MPC lifting operations are presented in the HI-STAR 100 FSAR (Docket Number 72-1008, Subsection 3.4.3) and are also applicable here.

The evaluation of the adequacy of the lifting devices entails careful consideration of the applied loading and associated stress limits. The load combination  $D+H$ , where  $H$  is the "handling load", is the generic case for all lifting adequacy assessments. The term  $D$  denotes the dead load. Quite obviously,  $D$  must be taken as the bounding value of the dead load of the component being lifted. In all lifting analyses considered in this document, the handling load  $H$  is assumed to be  $0.15D$ . In other words, the inertia amplifier during the lifting operation is assumed to be equal to  $0.15g$ . This value is consistent with the guidelines of the Crane Manufacturer's Association of America (CMAA), Specification No. 70, 1988, Section 3.3, which stipulates a dynamic factor equal to  $0.15$  for slowly executed lifts. Thus, the "apparent dead load" of the component for stress analysis purposes is  $D^* = 1.15D$ . Unless otherwise stated, all lifting analyses in this report use the "apparent dead load",  $D^*$ , as the lifted load.

Analysis methodology to evaluate the adequacy of the lifting device may be analytical or numerical. For the analysis of the trunnion, an accepted conservative technique for computing the bending stress is to assume that the lifting force is applied at the tip of the trunnion "cantilever" and that the stress state is fully developed at the base of the cantilever. This conservative technique,

recommended in NUREG-1536, is applied to all trunnion analyses presented in this FSAR and has also been applied to the trunnions analyzed in the HI-STAR 100 FSAR.

In general, the stress analysis to establish safety pursuant to NUREG-0612, Regulatory Guide 3.61, and the ASME Code, requires evaluation of three discrete zones which may be referred to as (i) the trunnion, (ii) the trunnion/component interface, hereinafter referred to as Region A, and (iii) the rest of the component, specifically the stressed metal zone adjacent to Region A, herein referred to as Region B. During this discussion, the term “trunnion” applies to any device used for lifting (i.e., trunnions, lift bolts, etc.)

Stress limits germane to each of the above three areas are discussed below:

- i. Trunnion: NUREG-0612 and Reg. Guide 3.61 require that under the “apparent dead load”,  $D^*$ , the maximum primary stress in the trunnion be less than  $1/10^{\text{th}}$  of the trunnion material ultimate strength and less than  $1/3^{\text{rd}}$  of the trunnion material yield strength, respectively. Because of the materials of construction selected for trunnions in all HI-STORM 100 System components, the ultimate strength-based limit is more restrictive in every case. Therefore, all trunnion safety factors reported in this document pertain to the ultimate strength-based limit.
- ii. Region A: Trunnion/Component Interface: Stresses in Region A must meet ASME Code Level A limits under applied load  $D^*$ . Additionally, Regulatory Guide 3.61 requires that the primary stress under  $3D^*$ , associated with the cross-section, be less than the yield strength of the applicable material. In cases involving section bending, the developed section moment may be compared against the plastic moment at yield. The circumferential extent of the characteristic cross-section at the trunnion/component interface is calculated based on definitions from ASME Section III, Subsection NB and is defined in terms of the shell thickness and radius of curvature at the connection to the trunnion block. By virtue of the construction geometry, only the mean shell stress is categorized as “primary” for this evaluation.
- iii. Region B: Typically, the stresses in the component in the vicinity of the trunnion/component interface are higher than elsewhere. However, exceptional situations exist. For example, when lifting a loaded MPC, the MPC baseplate, which supports the entire weight of the fuel and the fuel basket, is a candidate location for high stress even though it is far removed from the lifting location (which is located in the top lid).

Even though the baseplate in the MPC would normally belong to the Region B category, for conservatism it was considered as Region A in the HI-STAR 100 SAR. The pool lid and the transfer lid of the HI-TRAC transfer cask also fall into this dual category. In general, however, all locations of high stress in the component under  $D^*$  must also be checked for compliance with ASME Code Level A stress limits.

The acceptance criteria for lifting operations summarized above are consistent with those used in

the HI-STORM FW FSAR, which has been approved by the NRC.

Unless explicitly stated otherwise, all analyses of lifting operations presented in this report follow the load definition and allowable stress provisions of the foregoing. Consistent with the practice adopted throughout this chapter, results are presented in dimensionless form, as safety factors, defined as

$$\text{Safety Factor, } \beta = \frac{\text{Allowable Stress in the Region Considered}}{\text{Computed Maximum Stress in the Region}}$$

The safety factor, defined in the manner of the above, is the added margin over what is mandated by the applicable code (NUREG-0612 or Regulatory Guide 3.61).

In the following subsections, we briefly describe each of the lifting analyses performed to demonstrate compliance with regulations. Summary results are presented for each of the analyses.

It is recognized that stresses in Region A are subject to two distinct criteria, namely Level A stress limits under D\* and yield strength at 3D\*. We will identify the applicable criteria in the summary tables, under the column heading “Item”, using the “3D\*” identifier.

All of the lifting analyses reported on in this Subsection are designated as Load Case 01 in Table 3.1.5.

#### 3.4.3.1 125 Ton HI-TRAC Lifting Analysis - Trunnions

The lifting device in the HI-TRAC 125 cask is presented in Holtec Drawing 1880 (Section 1.5 herein). The two lifting trunnions for HI-TRAC are spaced at 180 degrees. The trunnions are designed for a two-point lift in accordance with the aforementioned NUREG-0612 criteria. Figure 3.4.21 shows the overall lifting configuration. The lifting analysis demonstrates that the stresses in the trunnions, computed using the conservative methodology described previously, comply with NUREG-0612 provisions.

Specifically, the following results are obtained:

<b>HI-TRAC 125 Lifting Trunnions<sup>†</sup></b>		
	<b>Value (ksi)</b>	<b>Safety Factor</b>
Bending stress	16.09	1.13
Shear stress	7.26	1.50
<sup>†</sup> The lifted load is 245,800 lb.(a value that bounds the actual lifted weight from the pool after the lift yoke weight is eliminated per Table 3.2.4).		

Note that the safety factor presented in the previous table represents the additional margin beyond

the mandated limit of 3 on yield strength and 10 on tensile strength.

Similar calculations have been performed for the HI-TRAC 125D cask, which differs from the HI-TRAC 125 with respect to the material options for the lifting trunnions. The lifting trunnions for the HI-TRAC 125 are fabricated from SB637-N07718; the lifting trunnions for the HI-TRAC 125D can be fabricated from either SB637-N07718 or SA564-630. The bounding results for the HI-TRAC 125D are:

<b>HI-TRAC 125D Lifting Trunnions<sup>†</sup></b>		
	<b>Value (ksi)</b>	<b>Safety Factor</b>
Bending stress	13.57	1.03
Shear stress	7.26	1.16
<sup>†</sup> The lifted load is 245,800 lb.(a value that bounds the actual lifted weight from the pool after the lift yoke weight is eliminated per Table 3.2.4).		

### 3.4.3.2 125 Ton HI-TRAC Lifting - Trunnion Lifting Block Welds, Bearing, and Thread Shear Stress (Region A)

As part of the Region A evaluation, the weld group connecting the lifting trunnion block to the inner and outer shells, and to the HI-TRAC top flange, is analyzed. Conservative analyses are also performed to determine safety factors for bearing stress and for thread shear stress at the interface between the trunnion and the trunnion block. The following results are obtained for the HI-TRAC 125 and 125D transfer casks:

<b>125 Ton HI-TRAC Lifting Trunnion Block (Region A Evaluation)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Trunnion Block Bearing Stress	5.95	11.4	1.91
Trunnion Block Thread Shear Stress	5.05	6.84	1.35
Weld Shear Stress (3D*)	4.35 <sup>†</sup>	11.4	2.62

<sup>†</sup> No quality factor has been applied to the weld group. (Subsection NF or NUREG-0612 do not apply penalty factors to the structural welds).

### 3.4.3.3 125 Ton HI-TRAC Lifting - Structure near Trunnion (Region B/Region A)

A three-dimensional elastic model of the HI-TRAC 125 metal components is analyzed using the

ANSYS finite element code. The structural model includes, in addition to the trunnion and the trunnion block, a portion of the inner and outer HI-TRAC shells and the HI-TRAC top flange. Stress results over the characteristic interface section are summarized and compared with allowable strength limits per ASME Section III, Subsection NF, and per Regulatory Guide 3.61. The results show that the primary stresses in the HI-TRAC 125 structure comply with the level A stress limits for Subsection NF structures.

The results from the analysis are summarized below:

<b>HI-TRAC 125 Trunnion Region (Regions A and B)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Membrane Stress	6.50	17.5	2.69
Membrane plus Bending Stress	8.71	26.25	3.01
Membrane Stress (3D*)	19.5	34.6	1.77

The results above are also valid for the HI-TRAC 125D since the dimensions and the configuration of the inner shell, outer shell, top flange, and the trunnion block are the same in both the HI-TRAC 125 and 125D transfer casks, and the dimension used in the finite element model for the trunnion length conservatively bounds both transfer casks.

#### 3.4.3.4 100 Ton HI-TRAC Lifting Analysis

The lifting trunnions and the trunnion blocks for the 100 Ton HI-TRAC are identical to the trunnions analyzed for the 125 Ton HI-TRAC. However, the outer shell geometry (outer diameter) is different. A calculation performed in the spirit of strength-of-materials provides justification that, despite the difference in local structure at the attachment points, the stresses in the body of the HI-TRAC 100 Ton unit meet the allowables set forth in Subsection 3.1.2.2.

Figure 3.4.10 illustrates the differences in geometry, loads, and trunnion moment arms between the body of the 125-Ton HI-TRAC and the body of the 100-Ton HI-TRAC. It is reasonable to assume that the level of stress in the 100 Ton HI-TRAC body, in the immediate vicinity of the interface (Section X-X in Figure 3.4.10), is proportional to the applied force and the bending moment applied. In the figure, the subscripts 1 and 0 refer to 100 Ton and 125 Ton casks, respectively. Figure 3.4.10 shows the location of the area centroid (with respect to the outer surface) and the loads and moment arms associated with each construction. Conservatively, neglecting all other interfaces between the top of the trunnion block and the top flange and between the sides of the trunnion block and the shells, equilibrium is maintained by developing a force and a moment in the section comprised of the two shell segments interfacing with the base of the trunnion block.

The most limiting stress state is in the outer shell at the trunnion block base interface. The stress

level in the outer shell at Section X-X is proportional to  $P/A + Mc/I$ . Evaluating the stress for a unit width of section permits an estimate of the stress state in the HI-TRAC 100 outer shell if the corresponding stress state in the HI-TRAC 125 is known (the only changes are the applied load, the moment arm and the geometry). Using the geometry shown in Figure 3.4.10 gives the result as:

$$\text{Stress (HI-TRAC 100 outer shell)} = 1.242 \times \text{Stress (HI-TRAC 125 outer shell)}$$

The tabular results in the previous subsection can be adjusted accordingly and are reported below:

<b>100 Ton HI-TRAC Near Trunnion (Region A and Region B)</b>	
<b>Item</b>	<b>Safety Factor</b>
Membrane Stress	2.17
Membrane plus Bending Stress	2.42
Membrane Stress (3D*)	1.43

#### 3.4.3.5 HI-STORM 100 Lifting Analyses

There are two vertical lifting scenarios for the HI-STORM 100 storage overpack carrying a fully loaded MPC. Figure 3.4.17 shows a schematic of these lifting scenarios. Both lifting scenarios are examined using finite element models that focus on the local regions near the lift points. The analysis is based on the geometry of the HI-STORM 100; the alterations to the lid and to the length of the overpack barrel to configure the HI-STORM 100S have no effect on the conclusions reached in the area of the baseplate. Therefore, there is no separate analysis for the baseplate, inboard of the inner shell, for the HI-STORM 100S as the results are identical to or bounded by the results presented here. Since the upper portion of the HI-STORM 100S, the HI-STORM 100S lid, and the radial ribs and anchor block have a different configuration than the HI-STORM 100, separate calculations have been performed for these areas of the HI-STORM 100S. Similarly, where differences in construction between the HI-STORM 100 and the HI-STORM 100S Version B exist, separate calculations have been performed and the results summarized here.

Scenario #1 considers a "bottom lift" where the fully loaded HI-STORM 100 storage overpack is lifted vertically by four synchronized hydraulic jacks each positioned at one of the four inlet air vents. This lift allows for installation and removal of "air pads" which may be used for horizontal positioning of HI-STORM 100 at the ISFSI pad.

Scenario #2, labeled the "top lift scenario" considers the lifting of a fully loaded HI-STORM 100 vertically through the four lifting lugs located at the top end.

No structural credit is assumed for the HI-STORM concrete in either of the two lifting scenarios except as a vehicle to transfer compressive loads.

For the bottom lift, a three-dimensional one-quarter symmetry finite element model of the bottom

region of the HI-STORM 100 storage overpack is constructed. The model includes the inner shell, the outer shell, the baseplate, the inlet vent side and top plates, and the radial plates connecting the inner and outer shells.

In the finite element analysis, the concrete is modeled as an equivalent pressure load applied over the baseplate as well as the four horizontal inlet vent plates. In reality, the concrete is supported only at the four inlet vents, directly above the hydraulic jacks. In other words, the concrete has sufficient strength to carry its own weight between these four support locations. The average shear stress in the concrete on a vertical cross section at the edge of an inlet vent is calculated as:

$$\tau_{concrete} = \frac{\rho V}{2A}$$

where:

- $\rho$  equals the weight density of concrete;
- $V$  equals the volume of unsupported concrete between two adjacent inlet vents;
- $A$  cross-sectional area of concrete at location of maximum shear stress;

For  $\rho = 160.8 \text{ lb/ft}^3$ ,  $V = 231 \text{ ft}^3$ , and  $A = 5,665 \text{ in}^2$  ( $= 27.5 \text{ in} \times 206 \text{ in}$ ), the average shear stress is only 3.28 psi, which is negligible compared to the allowable shear stress of 126.5 psi for 4,000 psi compressive strength concrete. If the density of concrete is increased to  $200 \text{ lb/ft}^3$ , the shear stress increases by roughly 0.8 psi. Clearly, the concrete can support this load. Moreover, the positive effect that the concrete strength has on the results outweighs any adverse impact due to high density concrete. Therefore, the safety factors reported in this subsection (where the concrete is treated like water) for the bottom lift remain conservative for concrete densities up to  $200 \text{ lb/ft}^3$ .

For the analysis of the "top lift" scenario, a three-dimensional 1/8-symmetry finite element model of the top segment of HI-STORM 100 storage overpack is constructed. The metal HI-STORM 100 material is modeled (shells, radial plates, lifting block, ribs, vent plates, etc.) using shell or solid elements. Lumped weights are used to ensure that portions of the structure not modeled are, in fact, properly represented as part of a lifted load. The model is supported vertically at the lifting lug. The results are reported in tabular form at the end of this subsection.

The finite element results for the HI-STORM 100, as well as the results of similar analyses for the HI-STORM 100S, are based on inner and outer shell thicknesses of 1-1/4" and 3/4", respectively. Per Bill of Material 1575 and Drawing 3669, the thickness of both shells may be changed to 1" as an option for the HI-STORM 100 and 100S overpacks. With respect to the lifting analyses, the 1" thick inner and outer shells would have a negligible effect on the maximum calculated stress in the inlet vent horizontal plate, the HI-STORM baseplate, and the radial ribs. Therefore, the safety factors reported below for the HI-STORM 100 and 100S are valid for either thickness option.

To provide an alternate calculation to demonstrate that the bolt anchor blocks are adequate, we compute the average normal stress in the net metal area of the block under three times the lifted load. Further conservatism is introduced by including an additional 15% for dynamic amplification, i.e., the total load is equal to  $3D^*$ .

The average normal load in one bolt anchor block is

$$\text{Load} = 3 \times 1.15 \times 360,000 \text{ lb.}/4 = 310,500 \text{ lb.} \quad (\text{Weight comes from Table 3.2.1})$$

The net area of the bolt anchor block is

$$\text{Area} = (3.14159)/4 \times (5'' \times 5'' - 3.25'' \times 3.25'') = 11.34 \text{ sq. inch} \quad (\text{Dimensions from BM-1575})$$

Therefore, the safety factor (yield strength at 350 degrees F/calculated stress from Table 3.3.3) is

$$\text{SF} = 31,400 \text{ psi}/ (\text{Load}/\text{Area}) = 1.14$$

The shear stress in the threads of the lifting block is also examined. This analysis considers a cylindrical area of material under an axial load resisting the load by shearing action. The diameter of the area is the basic pitch diameter of the threads, and the length of the cylinder is the thread engagement length.

The analysis also examines the capacity of major welds in the load path and the compression capacity of the pedestal shield and pedestal shield shell.

The table below summarizes key results obtained from the analyses described above for the HI-STORM 100.

<b>HI-STORM 100 Top and Bottom Lifting Analyses<sup>†‡</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Primary Membrane plus Bending - Bottom Lift - Inlet Vent Plates - Region B	8.0	26.3	3.28
Primary Membrane - Top Lift - Radial Rib Under Lifting Block - Region B	6.67	17.5	2.63
Primary Membrane plus Bending – Top Lift - Baseplate – Region B	7.0	26.3	3.75
Primary Membrane Region A (3D*)	19.97	33.15	1.66
Primary Membrane plus Bending Region A (3D*)	24.02	33.15	1.38
Lifting Block Threads - Top Lift – Region A (3D*)	10.67	18.84	1.76
Lifting Stud - Top Lift –Region A (3D*)	43.733	108.8	2.49
Welds – Anchor Block-to-Radial Rib Region B	5.74	19.695	3.43
Welds – Anchor Block-to-Radial Rib Region A (3D*)	17.21	19.62	1.14



<b>HI-STORM 100 Top and Bottom Lifting Analyses<sup>†‡</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Welds – Radial Rib-to-Inner and Outer Shells Region B	5.83	21.00	3.60
Welds – Radial Rib-to-Inner and Outer Shells Region A (3D*)	17.49	19.89	1.13
Weld – Baseplate-to Inner Shell Region A (3D*)	1.59	19.89	12.48
Weld – Baseplate-to-Inlet Vent Region A (3D*)	14.89	19.89	1.33
Pedestal Shield Concrete (3D*)	0.096	1.266	13.19
Pedestal Shell (3D*)	3.269	33.15	10.14

<sup>†</sup> Regions A and B are defined at beginning of Subsection 3.4.3

<sup>‡</sup> The lifted load is 360000 lb. and an inertia amplification of 15% is included.

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100 storage overpack under either the top lift or the bottom lift scenario. All factors of safety are greater than 1.0 using criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports and from USNRC Regulatory Guide 3.61.

Similar calculations have been performed for the HI-STORM 100S where differences in configuration warrant. The results are summarized in the table below:

<b>HI-STORM 100S Top and Bottom Lifting Analyses<sup>†‡</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Primary Membrane plus Bending - Bottom Lift - Inlet Vent Plates - Region A (3D*)	9.824	33.15	3.374
Lifting Block Threads - Top Lift – Region A (3D*)	7.950	18.840	2.37
Lifting Stud - Top Lift –Region A (3D*)	49.806	83.7	1.68
Welds – Anchor Block-to-Radial Rib Region B	5.556	21.0	3.78
Welds – Anchor Block-to-Radial Rib Region A (3D*)	16.670	18.84	1.13
Welds – Radial Rib-to-Inner and Outer Shells Region B	5.631*	21.00	3.73*
Welds – Radial Rib-to-Inner and Outer Shells Region A (3D*)	16.895*	19.89	1.18*

Weld – Baseplate-to Inner Shell Region A (3D*)	3.188	19.89	6.24
Weld – Baseplate-to-Inlet Vent Region A (3D*)	13.47	19.89	1.476
Radial Rib Membrane Stress – Bottom Lift Region A (3D*)	10.58	33.15	3.132
Pedestal Shield Concrete (3D*)	0.095	1.535	16.17
Pedestal Shell (3D*)	3.235	33.15	10.24

† Regions A and B are defined at beginning of Subsection 3.4.3

‡ The lifted load is 410,000 lb and an inertia amplification of 15% is included. The increased weight (over the longer HI-STORM 100) comes from conservatively assuming an increase in concrete weight density in the HI-STORM 100S overpack and lid to provide additional safety margin.

\* Result is specific to HI-STORM 100S overpacks fabricated with full height radial plates. For HI-STORM 100S overpacks fabricated with shorter top and bottom radial plates (i.e., two-piece configuration), the results tabulated below for the HI-STORM 100S Version B overpack, for the radial rib to inner and outer shell welds, are bounding.

Similar calculations have been performed for the HI-STORM 100S, Version B where differences in configuration warrant. The results are summarized in the table below for the heaviest HI-STORM 100S Version B (using high density concrete and with SA 564-630 stud material):

<b>HI-STORM 100S Version B Top and Bottom Lifting Analyses</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Primary Membrane - Bottom Lift - Inlet Vent Plates - Region A (3D*)	27.06	33.15	1.22
Primary Membrane + Bending - Bottom Lift - Inlet Vent Plates - Region A (3D*)	20.455	33.15	1.62
Lifting Block Threads - Top Lift –Region A (3D*)	9.315	19.620	2.11
Lifting Stud - Top Lift –Region A (3D*)	49.369	108.8	2.20
Welds – Anchor Block-to-Radial Rib Region B	5.507	19.695	3.58
Welds – Anchor Block-to-Radial Rib Region A (3D*)	16.523	19.620	1.19
Welds – Radial Rib-to-Inner and Outer Shells Region B	6.120	21.00	3.43
Welds – Radial Rib-to-Inner and Outer Shells Region A (3D*)	18.36	19.89	1.08
Weld – Baseplate-to Inner Shell Region A (3D*)	2.724	19.89	7.302
Radial Rib to Inner and Outer Shell – Bottom Lift Region A (3D*)	18.360	19.89	1.08

For the longest HI-STORM 100S, Version B, with high-density concrete, the lifted load is 406,400 lb.

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100, HI-STORM 100S, and HI-STORM 100S, Version B storage overpacks under either the top lift or the bottom lift scenario. All factors of safety are greater than 1.0 using criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports and from USNRC Regulatory Guide 3.61.

### 3.4.3.6 MPC Lifting Analysis

The MPC can be inserted or removed from an overpack by lifting cleats that are designed for installation into threaded holes in the top lid. The strength requirements of the attachment bolts and base metal are examined based on the requirements of NUREG 0612. Sufficiency of thread engagement length and bolt pre-load are also considered. The MPC top closure is examined considering the top lid as “Region B”, where satisfaction of ASME Code Level A requirements is demonstrated. The analysis also considers highly stressed regions of the top closure as “region A” where applied load is 3D\*. The MPC baseplate is analyzed under normal handling and subject to the allowable strengths appropriate to a component considered in “Region B”. Finally, the baseplate region is further analyzed where loading is “3D\*” consistent with the MPC baseplate being considered as “Region A”. The definitions of “Region A”, “Region B”, and “3D\*” as they apply to lifting analyses have been introduced at the beginning of this subsection.

The following table summarizes the minimum safety factors from these analyses. As stated earlier, safety factors tabulated below represent margins that are over and beyond those implied by the loading magnification mandated in NUREG 0612 or Regulatory Guide 3.61, as appropriate.

<b>Summary of MPC Lifting Analyses</b>			
<b>Item</b>	<b>MPC Lid Internal Threads Safety Factor (NUREG-0612/Reg. Guide 3.61)</b>	<b>Region A Safety Factor (Note 1)</b>	<b>Region B Safety Factor (Note 1)</b>
MPC	1.67	1.54	1.08

Notes:

1. Safety factor is for MPC baseplate.

When dual lids are used on the MPC, the outer lid transfers the entire lifted load to the peripheral weld. The maximum bending stress in the outer lid from the lifted load can be conservatively computed by strength of materials theory using the solution for a simply supported circular plate under a central concentrated load equal to 115% of the bounding MPC load. The calculation and result are presented below using tabular results from Timoshenko, Strength of Materials, Vol. II, 3<sup>rd</sup> Edition.

$$P = 90,000 \text{ lb.} \times 1.15$$

$$\text{Outer Diameter } a = 67.375''$$

$$\text{Effective Central Diameter where load is applied } b = 13.675'' \text{ (conservative assumption)}$$

$$a/b = 5$$

$$\text{Lid thickness} = 4.75'' \text{ (Dual lids)}$$

From the reference,  $k=1.745$  and the maximum bending stress under the amplified lifted load is

$$\sigma = kP/h^2 = 8005 \text{ psi}$$

Supplement 38 of [3.4.6] provides results for the stress in the lid under maximum normal condition internal pressure. For the case with dual lids, the stress must be doubled. Therefore, the maximum

primary stress is

$$S = 2 \times 1,795 \text{ psi}$$

Therefore, the combined bending stress at the center of the dual lid is 11,595. Using the allowable strength from Table 3.4.7, the safety factor is

$$SF = 25,450 \text{ psi}/11,595 \text{ psi} = 2.195$$

#### 3.4.3.7 Miscellaneous Lid Lifting Analyses

The HI-STORM 100 lid lifting analysis is performed to ensure that the threaded connections provided in the lid are adequately sized. The lifting analysis of the top lid is based on a vertical orientation of loading from an attached lifting device. The top lid of the HI-STORM 100 storage overpack is lifted using four lugs that are threaded into holes in the top plate of the lid (Holtec Drawing 1495, Section 1.5). It is noted that failure of the lid attachment would not result in any event of safety consequence because a free-falling HI-STORM 100 lid cannot strike a stored MPC (due to its size and orientation). Operational limits on the carry height of the HI-STORM 100 lid above the top of the storage overpack containing a loaded MPC preclude any significant lid rotation out of the horizontal plane in the event of a handling accident. Therefore, contact between the top of the MPC and the edge of a dropped lid due to uncontrolled lowering of the lid during the lid placement operation is judged to be a non-credible scenario. Except for location of the lift points, the lifting device for the HI-STORM 100S and for the HI-STORM 100S, Version B lid is the same as for the regular HI-STORM 100 lid. Since the lid weight for the HI-STORM 100S, Version B bounds the HI-STORM 100 and the HI-STORM 100S, the calculated safety factors for the lifting of the HI-STORM 100S lid are reduced and are also reported in the summary table below.

In addition to the HI-STORM 100 top lid lifting analysis, the strength qualification of the lid lifting holes, and associated lid lifting devices, for the HI-TRAC pool lid and top lid has been performed. The qualification is based on the Regulatory Guide 3.61 requirement that a load factor of 3 results in stresses less than the yield stress. The results for the HI-TRAC 125 bound the results for the HI-TRAC 125D, the HI-TRAC 100, and the HI-TRAC 100D, since the lid weights used in the calculation are greater than or equal to all other HI-TRAC lid weights. Example commercially available lifting structures are considered and it is shown that thread engagement lengths are acceptable. Loads to lifting devices are permitted to be at a maximum angle of 45 degrees from vertical. A summary of results, pertaining to the various lid lifting operations, is given in the table below:

<b>Summary of HI-STORM 100 Lid Lifting Analyses</b>		
<b>Item</b>	<b>Dead Load (lb)</b>	<b>Minimum Safety Factor</b>
HI-STORM 100 (100S) Top Lid Lifting	23,000 (29,000 <sup>†</sup> )	4.925 (3.906)
HI-TRAC Pool Lid Lifting		

	12,500	3.594 (Note 1)
HI-TRAC Top Lid Lifting	2,750	11.3
† Bounding weight of HI-STORM 100S, Version B top lid with 200 pcf concrete.		

Note 1: Safety Factor is calculated based on a conservative thread engagement of 1”.

The analysis demonstrates that thread engagement is sufficient for the threaded holes used solely for lid lifting and that commercially available lifting devices engaging the threaded holes, are available. We note that all reported safety factors are based on an allowable strength equal to 33.3% of the yield strength of the lid material when evaluating shear capacity of the internal threads and based on the working loads of the commercially available lifting devices associated with the respective threaded holes.

#### 3.4.3.8 HI-TRAC Pool Lid Analysis - Lifting MPC From the Spent Fuel Pool (Load Case 01 in Table 3.1.5)

During lifting of the MPC from the spent fuel pool, the HI-TRAC pool lid supports the weight of a loaded MPC plus water (see Figure 3.4.21). Calculations are performed to show structural integrity under this condition for both the HI-TRAC 100 and the HI-TRAC 125 transfer casks. In accordance with the general guidelines set down at the beginning of Subsection 3.4.3, the pool lid is considered as both Region A and Region B for evaluating safety factors. The analysis shows that the stress in the pool lid top plate is less than the Level A allowable stress under pressure equivalent to the heaviest MPC, contained water, and lid self-weight (Region B evaluation). Stresses in the lids and bolts are also shown to be below yield under three times the applied lifted load (Region A evaluation using Regulatory Guide 3.61 criteria). The threaded holes in the HI-TRAC pool lid are also examined for acceptable engagement length under the condition of lifting the MPC from the pool. It is demonstrated that the pool lid peripheral bolts have adequate engagement length into the pool lid to permit the transfer of the required load. The safety factor is defined based on the strength limits imposed by Regulatory Guide 3.61.

The following table summarizes the results of the analyses for the HI-TRAC pool lid for each of the four transfer cask types. Results given in the following table compare calculated stress (or load) and allowable stress (or load). In all cases, the safety factor is defined as the allowable value divided by the calculated value.

<b>HI-TRAC Pool Lid Lifting a Loaded MPC Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Bending Stress - HI-TRAC 125/125D - Region B Analysis - Pool Lid Top Plate	10.1	26.3	2.604
Lid Bending Stress - HI-TRAC 125/125D - Region B Analysis - Pool Lid Bottom Plate	5.05	26.3	5.208
Lid Bending Stress - HI-TRAC 100/100D - Region B Analysis- Pool Lid Top Plate	10.06	26.3	2.614
Lid Bending Stress - HI-TRAC 100/100D - Region B Analysis- Pool Lid Bottom Plate	6.425	26.3	4.093
Lid Bolt Stress - HI-TRAC 125 – (3D*)	18.92	95.0	5.02
Lid Bolt Stress - HI-TRAC 100 – (3D*)	18.21	95.0	5.216
Lid Bolt Force - HI-TRAC 125D – (3D*)	25.77‡	84.05‡	3.262
Lid Bolt Force - HI-TRAC 100D – (3D*)	24.80‡	84.05‡	3.389
Lid Bending Stress - HI-TRAC 125/125D - Region A Analysis - Pool Lid Top Plate (3D*)	30.3	33.15	1.094
Lid Bending Stress - HI-TRAC 125/125D - Region A Analysis - Pool Lid Bottom Plate (3D*)	15.15	33.15	2.188
Lid Bending Stress –HI-TRAC 100/100D – Region A Analysis- Pool Lid Top Plate (3D*)	30.19	33.15	1.098
Lid Bending Stress –HI-TRAC 100/100D – Region A Analysis- Pool Lid Bottom Plate (3D*)	19.28	33.15	1.72
Lid Thread Engagement Length (HI-TRAC 125)	137.4‡	267.9‡	1.949

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3.

<sup>‡</sup> Calculated and allowable value for this item in (kips).

### 3.4.3.9 HI-TRAC Transfer Lid Analysis - Lifting MPC Away from Spent Fuel Pool (Load Case 01 in Table 3.1.5)

During transfer to or from a storage overpack using a HI-TRAC 125 or a HI-TRAC 100, the HI-TRAC transfer lid supports the weight of a loaded MPC. Figure 3.4.21 illustrates the lift operation. In accordance with the general lifting analysis guidelines, the transfer lid should be considered as both a Region A (Regulatory Guide 3.61 criteria) and a Region B location (ASME Section III, Subsection NF for Class 3 plate and shell structures) for evaluation of safety factors. The HI-TRAC 125 transfer lid and the HI-TRAC 100 transfer lid are analyzed separately because of differences in geometry. The HI-TRAC 125D and HI-TRAC 100D employ specially designed mating devices in combination with the pool lid to transfer a loaded MPC to or from a storage overpack. Thus, a transfer lid analysis is not performed for the HI-TRAC 125D or the HI-TRAC 100D. Results for the HI-TRAC 125D and HI-TRAC 100D pool lids are presented in the previous subsection.

It is shown that the transfer lid doors can support a loaded MPC together with the door weight without exceeding ASME NF stress limits and the more conservative limits of Regulatory Guide 3.61. It is also shown that the connecting structure transfers the load to the cask body without overstress. The following tables summarize the results for both HI-TRAC casks:

<b>HI-TRAC 125 Transfer Lid – Lifting Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 125 - Door Plate – (3D*)	9.381	32.7	3.486
HI-TRAC 125 - Door Plate – Region B	3.127	26.25	8.394
HI-TRAC 125 – Wheel Track (3D*)	26.91	36.0	1.338
HI-TRAC 125 - Door Housing Bottom Plate-Region B	7.701	26.25	3.409
HI-TRAC 125 - Door Housing Bottom Plate-(3D*)	23.103	32.7	1.415
HI-TRAC 125 - Door Housing Stiffeners- (3D*)	4.131	32.7	7.913
HI-TRAC 125 - Housing Bolts-Region B	29.96	57.5	1.919
HI-TRAC 125 – Housing Bolts (3D*)	89.88	95.0	1.057
HI-TRAC 125 – Lid Top Plate (3D*)	30.907	32.7	1.058

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3

<b>HI-TRAC 100 Transfer Lid – Lifting Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 100 - Door Plate – (3D*)	22.188	32.7	1.474
HI-TRAC 100 - Door Plate – Region B	7.396	26.25	3.549
HI-TRAC 100 – Wheel Track (3D*)	13.011	36.0	2.767
HI-TRAC 100 – Door Housing Bottom Plate- Region B	7.447	26.25	3.525
HI-TRAC 100 – Door Housing Bottom Plate- (3D*)	22.336	32.7	1.464
HI-TRAC 100 – Door Housing Stiffeners- (3D*)	4.917	32.7	6.65
HI-TRAC 100 – Welds Connecting Door Housing Stiffeners (3D*)	11.802	32.7	2.771
HI-TRAC 100 - Housing Bolts-Region B	22.478	57.5	2.558
HI-TRAC 100 – Housing Bolts (3D*)	67.423	95.0	1.409
HI-TRAC 100 – Lid Top Plate (3D*)	19.395	32.7	1.686

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3

### 3.4.3.10 HI-TRAC Bottom Flange Evaluation during Lift (Load Case 01 in Table 3.1.5)

During a lifting operation, the HI-TRAC transfer cask body supports the load of a loaded MPC, and the transfer lid (away from the spent fuel pool) or the pool lid plus contained water (lifting from the spent fuel pool). In either case, the load is transferred to the bottom flange of HI-TRAC through the bolts and a state of stress in the flange and the supporting inner and outer shells is developed. Figure 3.4.21 illustrates the lifting operation. This area of the HI-TRAC 125 is analyzed to demonstrate that the required limits on stress are maintained for both ASME and Regulatory Guide 3.61. The bottom flange is considered as an annular plate subject to a total bolt load acting at the bolt circle and supported by reaction loads developed in the inner and outer shells of HI-TRAC. The solution for maximum flange bending stress is found in the classical literature and stresses and corresponding safety factors developed for the bottom flange and for the outer and inner shell weld shear stress. Since the welds are partial penetration, weld stress evaluation bounds an evaluation of direct stress. The table below summarizes the results of the evaluation.

<b>Safety Factors in HI-TRAC Bottom Flange During a Lift Operation</b>			
<b>Item</b>	<b>Value(ksi)</b>	<b>Allowable(ksi)</b>	<b>Safety Factor</b>
Bottom Flange – Region B	7.798	26.25	3.37
Bottom Flange (3D*)	23.39	33.15	1.42
Outer Shell (3D*)	4.773	33.15	6.94

The bottom flanges of the HI-TRAC 125D and HI-TRAC 100D are different from the HI-TRAC 125 in several respects. Namely, the thickness of the bottom flange is less, and the groove weld connecting the bottom flange to the inner shell is smaller. In addition, the bottom flange of the HI-TRAC 125D and HI-TRAC 100D is reinforced by eight gusset plates, whereas the HI-TRAC 125 bottom flange is not reinforced. Therefore, to account for these differences, the evaluation described above has been repeated for the HI-TRAC 125D and the HI-TRAC 100D. The results are summarized in the tables below. Note that the following results are conservative since the HI-TRAC 125D and HI-TRAC 100D bottom flange evaluations neglect the reinforcing strength of the gusset plates.

<b>Safety Factors in HI-TRAC 125D Bottom Flange During a Lift Operation</b>			
<b>Item</b>	<b>Value(ksi)</b>	<b>Allowable(ksi)</b>	<b>Safety Factor</b>
Bottom Flange – Region B	9.594	26.25	2.74
Bottom Flange (3D*)	28.78	33.15	1.15
Outer Shell (3D*)	4.710	33.15	7.04



<b>Safety Factors in HI-TRAC 100D Bottom Flange During a Lift Operation</b>			
<b>Item</b>	<b>Value(ksi)</b>	<b>Allowable(ksi)</b>	<b>Safety Factor</b>
Bottom Flange – Region B	8.646	26.25	3.04
Bottom Flange (3D*)	25.94	33.15	1.28
Outer Shell (3D*)	5.499	33.15	6.03

#### 3.4.3.11 Conclusion

Synopses of lifting device, device/component interface, and component stresses, under all contemplated lifting operations for the HI-STORM 100 System have been presented in the foregoing. The HI-STORM storage overpack and the HI-TRAC transfer cask have been evaluated for limiting stress states. The results show that all factors of safety are greater than 1.

#### 3.4.4 Heat

The thermal evaluation of the HI-STORM 100 System is reported in Chapter 4.

##### 3.4.4.1 Summary of Pressures and Temperatures

Design pressures and design temperatures for all conditions of storage are listed in Tables 2.2.1 and 2.2.3, respectively.

##### 3.4.4.2 Differential Thermal Expansion

Consistent with the requirements of Reg. Guide 3.61, Load Cases F1 (Table 3.1.3) and E4 (Table 3.1.4) are defined to study the effect of differential thermal expansion among the constituent components in the HI-STORM 100 System. The temperatures necessary to perform the differential thermal expansion analyses for the MPC in the HI-STORM 100 and HI-TRAC casks are provided in Chapter 4. The material presented in Subsection 4.4.5 demonstrates that a physical interference between discrete components of the HI-STORM 100 System (e.g. storage overpack and enclosure vessel) will not develop due to differential thermal expansion during any operating condition.

###### 3.4.4.2.1 Normal Hot Environment

Closed form calculations are performed in Subsection 4.4.6 to demonstrate that initial gaps between the HI-STORM 100 storage overpack or the HI-TRAC transfer cask and the MPC canister, and between the MPC canister and the fuel basket, will not close due to thermal expansion of the system components under loading conditions, defined as F1 and E4 in Tables 3.1.3 and 3.1.4, respectively. To assess this in the most conservative manner, the thermal solutions computed in Chapter 4, including the thermosiphon effect, are surveyed for the following information.

- The radial temperature distribution in each of the fuel baskets at the location of peak center metal temperature.

- The highest and lowest mean temperatures of the canister shell for the hot environment condition.

Using the above temperature information, simplified thermoelastic solutions of equivalent axisymmetric problems are used to obtain conservative estimates of gap closures. The following procedure, which conservatively neglects axial variations in temperature distribution, is utilized.

1. Use the surface temperature information for the fuel basket to define a parabolic distribution in the fuel basket that bounds (from above) the actual temperature distribution. Using this result, generate a conservatively high estimate of the radial and axial growth of the different fuel baskets using classical closed form solutions for thermoelastic deformation in cylindrical bodies.
2. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-basket gaps.
3. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-storage overpack and canister-to-HI-TRAC gaps.
4. For given initial clearances, compute the operating clearances.

The results are summarized in Table 4.4.10 for normal storage conditions. The clearances between the MPC basket and canister structure, as well as between the MPC shell and storage overpack or HI-TRAC inside surface, are sufficient to preclude a temperature induced interference from differential thermal expansions under normal operating conditions.

#### 3.4.4.2.2 Fire Accident

It is shown in Chapter 4 that the fire accident has a small effect on the MPC temperatures because of the short duration of the fire accidents and the large thermal inertia of the storage overpack. Therefore, a structural evaluation of the MPC under the postulated fire event is not required. The conclusions reached in Subsection 3.4.4.2.1 are also appropriate for the fire accident with the MPC housed in the storage overpack. Analysis of fire accident temperatures of the MPC housed within the HI-TRAC for thermal expansion is unnecessary, as the HI-TRAC, directly exposed to the fire, expands to increase the gap between the HI-TRAC and MPC.

As described in the footnote under Table 2.2.3 in Chapter 2, during a fire accident, no specific temperature limits apply to the overpack structure and the structure is only required to remain physically stable, i.e., the maximum temperature shall be less than 50% of the component's melting temperature.

As expected, the external surfaces of the HI-STORM 100 storage overpack that are directly exposed to the fire event experience maximum rise in temperature. The outer shell and top plate in the top lid are the external surfaces that are in direct contact with heated air from fire. The table

below, extracted from data provided in Chapter 4, provides the maximum temperatures attained at the key locations in HI-STORM 100 storage overpack under the postulated fire event.

<b>Component</b>	<b>Maximum Fire Condition Temperature (Deg. F)</b>
Storage Overpack Inner Shell	300
Storage Overpack Radial Concrete Mid-Depth	184
Storage Overpack Outer Shell	570
Storage Overpack Lid	<570

The following conclusions are readily reached from the above table.

- The maximum metal temperature of the components described above is less than 50% of their melting temperature.
- The local concrete temperature is below its short term temperature limit specified in Table 2.2.3.
- The presence of a stitch weld between the overpack inner shell and the overpack top plate ensures that there will be no pressure buildup in the concrete annulus due to the concrete losing water that then turns to steam.

The above summary confirms that the postulated fire event will not jeopardize the structural integrity of the HI-STORM 100 Overpack or significantly diminish its shielding effectiveness.

The above conclusions, as relevant, also apply to the HI-TRAC fire considered in Chapter 4. Water jacket over-pressurization is precluded by the safety valve set point. The non-structural effects of loss of water have been evaluated in Chapter 5 and shown to meet regulatory limits. Therefore, it is concluded that the postulated fire event will not cause significant loss in storage overpack or HI-TRAC shielding function.

#### 3.4.4.3 Stress Calculations

This subsection presents calculations of the stresses in the different components of the HI-STORM 100 System from the effects of mechanical load case assembled in Section 3.1. Loading cases for the MPC fuel basket, the MPC enclosure vessel, the HI-STORM 100 storage overpack and the HI-TRAC transfer cask are listed in Tables 3.1.3 through 3.1.5, respectively. The load case identifiers defined in Tables 3.1.3 through 3.1.5 denote the cases considered.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability of fuel from the MPC and the MPC from the HI-STORM 100 storage overpack or from the HI-TRAC transfer cask.

For all stress evaluations, the allowable stresses and stress intensities for the various HI-STORM

100 System components are based on bounding high metal temperatures to provide additional conservatism (Table 3.1.17 for the MPC basket, for example).

In addition to the loading cases germane to stress evaluations mentioned above, three cases pertaining to the stability of HI-STORM 100 are also considered (Table 3.1.1).

The results of various stress calculations on components are reported. The calculations are either performed directly as part of the text, or carried out in a separate calculation report that provides details of strength of materials evaluations or finite element numerical analysis. The specific calculations reported in this subsection are:

1. MPC stress calculations
2. HI-STORM 100 storage overpack stress calculations
3. HI-TRAC stress calculations

The MPC calculations reported in this document are complemented by analyses in the HI-STAR 100 Dockets. As noted earlier in this chapter, calculations for MPC components that are reported in HI-STAR 100 FSAR and SAR (Docket Numbers 72-1008 or 71-9261) are not repeated here unless geometry or load changes warrant reanalysis. For example, analysis of the MPC lid under normal conditions is not included in this submittal since neither the MPC lid loading nor geometry is affected by the MPC being placed in HI-TRAC or HI-STORM 100. MPC stress analyses reported herein focus on the basket and canister stress distributions due to the design basis (45g) lateral deceleration imposed by a non-mechanistic tip-over of the HI-STORM 100 storage overpack or a horizontal drop of HI-TRAC. In the submittals for the HI-STAR 100 FSAR and SAR (Docket Numbers 72-1008 and 71-9261, for storage and transport, respectively), the design basis deceleration was 60g. In this submittal the design basis deceleration is 45g. However, since the geometry of the MPC external boundary condition, viz. canister-to-storage overpack gap, has changed, a reanalysis of the MPC stresses under the lateral deceleration loads is required. This analysis is performed and the results are summarized in this subsection.

The HI-STORM 100 storage overpack and the HI-TRAC transfer cask have been evaluated for certain limiting load conditions that are germane to the storage and operational modes specified for the system in Tables 3.1.1 and 3.1.5. The determination of component safety factors at the locations considered in the HI-STORM 100 storage overpack and in the HI-TRAC transfer cask is based on the allowable stresses permitted by the ASME Code Section III, Subsection NF for Class 3 plate and shell support structures.

#### 3.4.4.3.1 MPC Stress Calculations

The structural function of the MPC in the storage mode is stated in Section 3.1. The calculations presented here demonstrate the ability of the MPC to perform its structural function. The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, or impairment of ready retrievability.

#### 3.4.4.3.1.1 Analysis of Load Cases E.3.b, E.3.c (Table 3.1.4) and F.3.b, F.3.c (Table 3.1.3)

Analyses are performed for each of the MPC designs. The following subsections describe the model, individual loads, load combinations, and analysis procedures applicable to the MPC. Unfortunately, unlike vertical loading cases, where the analyses performed in the HI-STAR 100 dockets remain fully applicable for application in HI-STORM 100, the response of the MPC to a horizontal loading event is storage overpack-geometry dependent. Under a horizontal drop event, for example, the MPC and the fuel basket structure will tend to flatten. The restraint to this flattening offered by the storage overpack will clearly depend on the difference in the diameters of the storage overpack internal cavity and that of the outer surface of the MPC. In the HI-STORM 100 storage overpack, the diameter difference is larger than that in HI-STAR 100; therefore, the external restraint to MPC ovalization under a horizontal drop event is less effective. For this reason, the MPC stress analysis for lateral loading scenarios must be performed anew for the HI-STORM 100 storage overpack; the results from the HI-STAR 100 analyses will not be conservative. The HI-TRAC transfer casks and HI-STAR 100 overpack inner diameters are identical. Therefore, the analysis of the MPC in the HI-STAR 100 overpack under 60g's for the side impact (Docket 72-1008) bounds the analysis of the MPC in the HI-TRAC under 45g's.

#### Description of Finite Element Models of the MPCs Under Lateral Loading

A finite element model of each MPC is used to assess the effects of the accident loads. The models are constructed using ANSYS [3.4.1], and they are identical to the models used in Holtec's HI-STAR 100 submittals in Docket Numbers 72-1008 and 71-9261. The following model description is common to all MPCs except for the MPCs using basket shims welded to the MPC fuel basket, which are evaluated in a separate analysis outside of ANSYS. A basket shim is defined as any structural member that is welded to the outside surface of the fuel basket panel.

The MPC structural model is two-dimensional. It represents a one-inch long cross section of the MPC fuel basket and MPC canister.

The MPC model includes the fuel basket, the basket support structures, and the MPC shell. A basket support is defined as any structural member (excluding the lift lugs) that is welded to the inside surface of the MPC shell. A portion of the storage overpack inner surface is modeled to provide the correct restraint conditions for the MPC. Figures 3.4.1 through 3.4.9 show typical MPC models. The fuel basket support structure shown in the figures is a multi-plate structure consisting of solid shims or support members having two separate compressive load supporting members. For conservatism in the finite element model some dual path compression members (i.e., "V" angles) are simulated as single columns. Therefore, the calculated stress intensities in the fuel basket angle supports from the finite element solution are conservatively overestimated in some locations.

The ANSYS model is not intended to resolve the detailed stress distributions in weld areas. Individual welds are not included in the finite element model. A separate analysis for basket welds and for the basket supports is performed outside of ANSYS.

No credit is taken for any load support offered by the neutron absorber panels, sheathing, and the

aluminum heat conduction elements. Therefore, these so-called non-structural members are not represented in the model. The bounding MPC weight used, however, does include the mass contributions of these non-structural components.

The model is built using five ANSYS element types: BEAM3, PLANE82, CONTAC12, CONTAC26, and COMBIN14. The fuel basket and MPC shell are modeled entirely with two-dimensional beam elements (BEAM3). Plate-type basket supports are also modeled with BEAM3 elements. Eight-node plane elements (PLANE82) are used for the solid-type basket supports. The gaps between the fuel basket and the basket supports are represented by two-dimensional point-to-point contact elements (CONTAC12). Contact between the MPC shell and the storage overpack is modeled using two-dimensional point-to-ground contact elements (CONTAC26) with an appropriate clearance gap.

Two orientations of the deceleration vector are considered. The 0-degree drop model includes the storage overpack-MPC interface in the basket orientation illustrated in Figure 3.1.2. The 45-degree drop model represents the storage overpack-MPC interface with the basket oriented in the manner of Figure 3.1.3. The 0-degree and the 45-degree drop models are shown in Figures 3.4.1 through 3.4.6. Table 3.4.1 lists the element types and number of elements for current MPCs.

A contact surface is provided in the model used for drop analyses to represent the interface between the storage overpack channels and the MPC. As the MPC makes contact with the storage overpack, the MPC shell deforms to mate with the channels that are welded at equal intervals around the storage overpack inner surface. The nodes that define the elements representing the fuel basket and the MPC shell are located along the centerline of the plate material. As a result, the line of nodes that forms the perimeter of the MPC shell is inset from the real boundary by a distance that is equal to half of the shell thickness. In order to maintain the specified MPC shell/storage overpack gap dimension, the radius of the storage overpack channels is decreased by an equal amount in the model.

The three discrete components of the HI-STORM 100 System, namely the fuel basket, the MPC shell, and the storage overpack or HI-TRAC transfer cask, are engineered with small diametral clearances which are large enough to permit unconstrained thermal expansion of the three components under the rated (maximum) heat duty condition. A small diametral gap under ambient conditions is also necessary to assemble the system without physical interference between the contiguous surfaces of the three components. The required gap to ensure unrestricted thermal expansion between the basket and the MPC shell is small and will further decrease under maximum heat load conditions, but will introduce a physical nonlinearity in the structural events involving lateral loading (such as side drop of the system) under ambient conditions. It is evident from the system design drawings that the fuel basket that is non-radially symmetric is in proximate contact with the MPC shell at a discrete number of locations along the circumferences. At these locations, the MPC shell, backed by the channels attached to the storage overpack, provides a support line to the fuel basket during lateral drop events. Because the fuel basket, the MPC shell, and the storage overpack or HI-TRAC are all three-dimensional structural weldments, their inter-body clearances may be somewhat uneven at different azimuthal locations. As the lateral loading is increased, clearances close at the support locations, resulting in the activation of the support from the storage overpack or HI-TRAC.

The bending stresses in the basket and the MPC shell at low lateral loading levels which are too small to close the support location clearances are secondary stresses since further increase in the loading will activate the storage overpack's or HI-TRAC's transfer cask support action, mitigating further increase in the stress. Therefore, to compute primary stresses in the basket and the MPC shell under lateral drop events, the gaps should be assumed to be closed. However, in the analyses, we have conservatively assumed that an initial gap of 0.1875" exists, in the direction of the applied deceleration, at all support locations between the fuel basket and the MPC shell and that the clearance gap between the shell and the storage overpack at the support locations is 3/16". In the evaluation of safety factors for the MPC-24, MPC-32, and MPC-68, the total stress state produced by the applied loading on these configurations is conservatively compared with primary stress levels, even though the self-limiting stresses should be considered secondary in the strict definition of the Code. To illustrate the conservatism, we have eliminated the secondary stress (that develops to close the clearances) in the comparison with primary stress allowable values and report safety factors for the MPC-24E that are based only on primary stresses necessary to maintain equilibrium with the inertia forces.

ANSYS requires that for a static solution all bodies be constrained to prevent rigid body motion. Therefore, in the 0 degree and 45 degree drop models, two-dimensional linear spring elements (COMBIN14) join the various model components, i.e., fuel basket and enclosure vessel, at the point of initial contact. This provides the necessary constraints for the model components in the direction of the impact. By locating the springs at the points of initial contact, where the gaps remain closed, the behavior of the springs is identical to the behavior of a contact element. Linear springs and contact elements that connect the same two components have equal stiffness values.

#### Description of Individual Loads and Boundary Conditions Applied to the MPCs

The method of applying each individual load to the MPC model is described in this subsection. The individual loads are listed in Table 2.2.14. A free-body diagram of the MPC corresponding to each individual load is given in Figures 3.4.7-3.4.9. In the following discussion, reference to vertical and horizontal orientations is made. Vertical refers to the direction along the cask axis, and horizontal refers to a radial direction.

Quasi-static structural analysis methods are used. The effects of any dynamic load factors (DLFs) are included in the final evaluation of safety factors. All analyses are carried out using the design basis decelerations in Table 3.1.2.

The MPC models used for side drop evaluations are shown in Figures 3.4.1 through 3.4.6. In each model, the fuel basket and the enclosure vessel are constrained to move only in the direction that is parallel to the acceleration vector. The storage overpack inner shell, which is defined by three nodes needed to represent the contact surface, is fixed in all degrees of freedom. The fuel basket, enclosure vessel, and storage overpack inner shell are all connected at one location by linear springs, as described in Subsection 3.4.4.3.1.1 (see Figure 3.4.1, for example). Detailed side drop evaluations here focus on an MPC within a HI-STORM 100 storage overpack. Since the analyses performed in Docket Number 72-1008 for the side drop condition in the HI-STAR 100 storage overpack demonstrates a safe condition under a 60g deceleration, no new analysis is required for the MPC and contained fuel basket and fuel during a side drop in the HI-TRAC, which is limited

to a 45g deceleration (HI-TRAC and HI-STAR 100 overpacks have the same inside dimensions).

### Accelerations

During a side impact event, the stored fuel is directly supported by the cell walls in the fuel basket. Depending on the orientation of the drop, 0 or 45 degrees (see Figures 3.4.8 and 3.4.9), the fuel is supported by either one or two walls. In the finite element model this load is effected by applying a uniformly distributed pressure over the full span of the supporting walls. The magnitude of the pressure is determined by the weight of the fuel assembly, the axial length of the fuel basket support structure, the width of the cell wall, and the impact acceleration. It is assumed that the load is evenly distributed along an axial length of basket equal to the fuel basket support structure. For example, the pressure applied to an impacted cell wall during a 0-degree side drop event is calculated as follows:

$$p = \frac{a_n W}{L c}$$

where:

p = pressure

a<sub>n</sub> = ratio of the impact acceleration to the gravitational acceleration

W = weight of a stored fuel assembly

L = axial length of the fuel basket support structure

c = width of a cell wall

For the case of a 45-degree side drop the pressure on any cell wall equals p (defined above) divided by the square root of 2.

It is evident from the above that the effect of deceleration on the fuel basket and canister metal structure is accounted for by amplifying the gravity field in the appropriate direction.

### Internal Pressure

Design internal pressure is applied to the MPC model. The inside surface of the enclosure vessel shell is loaded with pressure. The magnitude of the internal pressure applied to the model is taken from Table 2.2.1.

For this load condition, the center node of the fuel basket is fixed in all degrees of freedom to numerically satisfy equilibrium.



## Temperature

Temperature distributions are developed in Chapter 4 and applied as nodal temperatures to the finite element model of the MPC enclosure vessel (confinement boundary). Maximum design heat load has been used to develop the temperature distribution used to demonstrate compliance with ASME Code stress intensity levels.

## Analysis Procedure

The analysis procedure for this set of load cases is as follows:

1. The stress intensity and deformation field due to the combined loads is determined by the finite element solution. Results are postprocessed and tabulated in the calculation package associated with this FSAR.
2. The results for each load combination are compared to allowables. The comparison with allowable values is made in Subsection 3.4.4.4.

### 3.4.4.3.1.2 Analysis of Load Cases E1.a, E1.c, E2, and Accident MPC Internal Pressure (Table 3.1.4)

Since the MPC shell is a pressure vessel, the classical Lamé's calculations should be performed to demonstrate the shell's performance as a pressure vessel. We note that dead load has an insignificant effect on this stress state. We first perform calculations for the shell under internal pressure. Subsequently, we examine the entire confinement boundary as a pressure vessel subject to both internal pressure and temperature gradients. Finally, we perform confirmatory hand calculations to gain confidence in the finite element predictions.

The stress from internal pressure is found for normal and accident pressures conditions using classical formulas:

Define the following quantities:

$P$  = pressure,  $r$  = MPC radius, and  $t$  = shell thickness.

Using classical thin shell theory, the circumferential stress,  $\sigma_1 = Pr/t$ , the axial stress  $\sigma_2 = Pr/2t$ , and the radial stress  $\sigma_3 = -P$  are computed for both normal and accident internal pressures. The results are given in the following table (conservatively using the outer radius for  $r$ ):

<b>Classical Shell Theory Results for Normal and Accident Internal Pressures</b>				
<b>Item</b>	<b><math>\sigma_1</math> (psi)</b>	<b><math>\sigma_2</math> (psi)</b>	<b><math>\sigma_3</math> (psi)</b>	<b><math>\sigma_1 - \sigma_3</math> (psi)</b>
P= 110 psi	7522	3761	-110	7632
P= 200 psi	13675	6838	-200	13875

Maximum MPC weight,  $W_{MPC} = 90,000$  lbs per Table 3.2.1

Outer diameter of MPC shell,  $OD_{MPC} = 68.375''$

Thickness of MPC shell,  $t_{MPC} = 0.5''$

Dynamic load factor,  $DLF = 1.15$  (Table 2.0.1)

For Load Case E2, the maximum axial stress ( $\sigma_a$ ) in the MPC shell is calculated below.

$$\sigma_a = \sigma_2 + \frac{W_{MPC} \cdot DLF}{\frac{\pi}{4} \cdot [OD_{MPC}^2 - (OD_{MPC} - 2 \cdot t_{MPC})^2]}$$

$$\sigma_a = 4,732 \text{ psi}$$

Therefore, the maximum primary membrane stress ( $\sigma_{PM}$ ) in the MPC shell for Load Cases E1.a, E1.c and E2 is,

$$\sigma_{PM} = \max(\sigma_a, \sigma_1 - \sigma_3) = 7,632 \text{ psi}$$

The primary membrane stress allowable for MPC shell at 500 °F (per Table 3.1.17) is,

$$\sigma_{all} = 17,500 \text{ psi}$$

Therefore, the safety factor against maximum primary membrane stress in the MPC shell for Load Cases E1.a, E1.c and E2 is,

$$SF_{shell} = (\sigma_{all} / \sigma_{PM}) = 2.29$$

- Finite Element Analysis (Load Case E1.a and E1.c of Table 3.1.4)

The MPC shell, the top lid, and the baseplate together form the confinement boundary (enclosure vessel) for storage of spent nuclear fuel. In this section, the bounding normal operating conditions consisting of dead weight, design and short-term normal internal pressure limits, and thermal effects for the hot condition of storage, are evaluated. The top and bottom plates of the MPC enclosure vessel (EV) are modeled using plane axisymmetric elements, while the shell is modeled using the axisymmetric thin shell element. The thickness of the top lid varies in the different MPC types and can be either a single thick lid, or two dual lids welded around their common periphery; the minimum thickness top lid is modeled in the finite element analysis. As applicable, the results for the MPC top lid are modified to account for the fact that in the dual lid configuration, the two lids act independently under mechanical loading. The temperature distributions for all MPC constructions are nearly identical in magnitude and gradient and reflect the thermosiphon effect inside the MPC. Temperature differences across the thickness of both the baseplate and the top lid exist during HI-STORM 100's operations. There is also a thermal gradient from the center of the top lid and baseplate out to the shell wall. The metal temperature profile is essentially parabolic from the centerline of the MPC out to the MPC shell. There is also a parabolic temperature profile along the length of the MPC canister. Figure 3.4.11 shows a sketch of the confinement boundary structure with identifiers A-I locating points where temperature input data is used to represent a

continuous temperature distribution for analysis purposes. The overall dimensions of the confinement boundary are also shown in the figure.

The temperatures for confinement thermal stress analysis are determined from the thermal numerical analyses supporting the results in Chapter 4. Based on the maximum through thickness thermal gradients, a reference temperature profile is defined for all MPC types and used as input for thermal stress analysis. In reality, the temperature profile for a particular MPC will vary based upon the total heat load and the fuel loading pattern and will differ from the temperature profile analyzed herein. In order to insure that the slight variations in the temperature profile do not adversely affect the design of the MPC enclosure vessel, the allowable stress intensity values used to compute the safety factors reported in Tables 3.4.7 and 3.4.8 correspond to base metal temperatures that exceed the applied temperature profile for each MPC component at all locations.

Because of the intimate contact between the two lid plates when the MPC lid is a two-piece unit, there is no significant thermal discontinuity through the thickness; thermal stresses arising in the MPC top lid will be bounding when there is only a single lid. Therefore, for thermal stresses, results from the analysis that considers the lid as a one-piece unit are used and are amplified to reflect the increase in stress in the dual lid configuration.

Figure 3.4.12 shows details of the finite element model of the top lid (considered as a single piece), canister shell, and baseplate. The top lid is modeled with 40 axisymmetric quadrilateral elements; the weld connecting the lid to the shell is modeled by a single element solely to capture the effect of the top lid attachment to the canister offset from the middle surface of the top lid. The MPC canister is modeled by 50 axisymmetric shell elements, with 20 elements concentrated in a short length of shell appropriate to capture the so-called "bending boundary layer" at both the top and bottom ends of the canister. The remaining 10 shell elements model the MPC canister structure away from the shell ends in the region where stress gradients are expected to be of less importance. The baseplate is modeled by 20 axisymmetric quadrilateral elements. Deformation compatibility at the connections is enforced at the top by the single weld element, and deformation and rotation compatibility at the bottom by additional shell elements between nodes 106-107 and 107-108.

The geometry of the model is listed below (terms are defined in Figure 3.4.12):

$H_t =$	9.5" (the minimum total thickness lid is assumed)
$R_L =$	0.5 x 67.25" (Bill of Materials for Top Lid)
$L_{MPC} =$	190.5" (Design Drawings in Section 1.5)
$t_s =$	0.5"
$t_{BP} =$	2.5"
$\beta =$	$2\sqrt{R_s t_s} \approx 12"$ (the "bending boundary layer")

Stress analysis results are obtained for two cases as follows:

- a. internal pressure = 100 psi
- b. internal pressure = 110 psi plus short-term normal condition temperatures

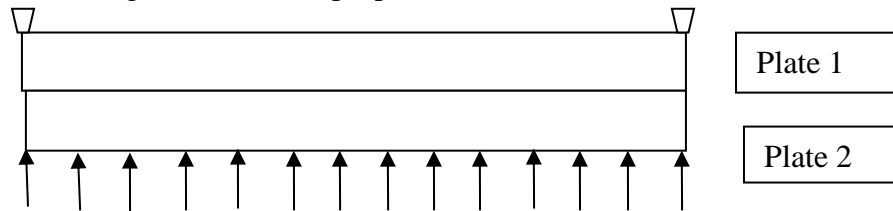
For this configuration, dead weight of the top lid acts to reduce the stresses due to pressure. For example, the equivalent pressure simulating the effect of the weight of the top lid is an external pressure of 3 psi, which reduces the pressure difference across the top lid to 97 psi under long-term normal internal pressure limit and 107 psi under short-term normal internal pressure limit. The dead weight of the top lid is neglected to provide additional conservatism in the results. The dead weight of the baseplate, however, adds approximately 0.73 psi to the effective internal pressure acting on the base. The effect of dead weight is still insignificant compared to the 100 psi design pressure limit and 110 psi short-term normal condition pressure limit, and is therefore neglected. The thermal loading in the confinement vessel is obtained by developing a parabolic temperature profile to the entire length of the MPC canister and to the top lid and baseplate. The temperature data provided at locations A-I in Figures 3.4.11 and 3.4.12 are sufficient to establish the profiles. Through-thickness temperatures are assumed linearly interpolated between top and bottom surfaces of the top lid and baseplate. Finally, in the analysis, all material properties and expansion coefficients are considered to be temperature-dependent in the model.

Results for stress intensities are reported for the case of design internal pressure limit alone (Load Case E.1.a in Table 3.1.4) and for the combined loading of short-term normal pressure limit plus temperature (Load Case E1.c in Table 3.1.4). Tables 3.4.7 and 3.4.8 report results at the inside and outside surfaces of the top lid and baseplate at the centerline and at the extreme radius. Canister results are reported in the "bending boundary layer" and at a location near mid-length of the MPC canister. In the tables, the calculated value is the value from the finite element analysis, the categories are  $P_m$  = primary membrane;  $P_L + P_b$  = local membrane plus primary bending; and  $P_L + P_b + Q$  = primary plus secondary stress intensity. The allowable strength value is obtained from the appropriate table in Section 3.1 for Level A conditions, and the safety factor SF is defined as the allowable strength divided by the calculated value. Allowable strengths for Alloy X are taken at 550 degrees F, 400 degrees F, and 500 degrees F, respectively, for the MPC lid, baseplate, and canister shell. These temperatures bound the actual component temperatures during both of the above loading cases. The results given in Tables 3.4.7 and 3.4.8 demonstrate the ruggedness of the MPC as a confinement boundary. It is noted that secondary stresses are conservatively presented in Table 3.4.7 (for Design condition) even though it is not required by Table 2.2.10. The Level B off-normal condition is bounded by the short-term normal condition which has an identical pressure limit, comparable temperatures, and is evaluated against Level A allowables. Since mechanically induced stresses in the top lid are increased when a dual lid configuration is considered, the stress results obtained from an analysis of a single top lid must be corrected to reflect the maximum stress state when a dual lid configuration is considered. The modifications required are based on the following logic:

Consider the case of a simply supported circular plate of thickness  $h$  under uniform lateral pressure "q". Classical strength of materials provides the solution for the maximum stress, which occurs at the center of the plate, in the form:

$$\sigma_s = 1.2375 \cdot q \cdot \left(\frac{a}{h}\right)^2 \quad \text{where } a \text{ is the radius of the plate and } h \text{ is the plate thickness.}$$

Now consider the MPC simply supported top lid as fabricated from two plates “1” and “2”, of thickness  $h_1$  and  $h_2$ , respectively, where the lower surface of plate 2 is subjected to the internal pressure “ $q$ ”, the upper surface of plate 1 is the outer surface of the helium retention boundary, and the lower surface of plate 1 and the upper surface of plate 2 are in contact. The following sketch shows the dual lid configuration for the purposes of this discussion:



From classical plate theory, if it is assumed that the interface pressure between the two plates is uniform and that both plates deform to the same central deflection, then if

$$h_1 + h_2 = h, \text{ and if } h_2/h_1 = r$$

the following relations exist between the maximum stress in the two individual plates,  $\sigma_1$ ,  $\sigma_2$  and the maximum stress  $\sigma_s$  in the single plate of thickness “ $h$ ” (assuming that all three plates are the same material):

$$\frac{\sigma_1}{\sigma_s} = \frac{(1+r)^2}{(1+r^3)} \qquad \frac{\sigma_2}{\sigma_s} = \frac{(1+r)^2}{(1+r^3)} r$$

Since the two lid thicknesses are the same in the dual lid configuration,  $r = 1.0$  so that the stresses in plates 1 and 2 are both two times larger than the maximum stress computed for the single plate lid having the same total thickness. In Tables 3.4.7 and 3.4.8, bounding results for the single lid configuration are reported; a doubling of the calculated stress values (and a halving of the top lid safety factors) results when the dual lid configuration is considered.

Per the design drawings in Section 1.5, the top plate in the dual lid configuration is stainless steel (Alloy X), and the bottom plate is fabricated from stainless steel or carbon steel with a stainless steel covering. Although the top and bottom plates may be different materials, the stress amplification factor of 2, which is used to convert the results from the one-piece MPC lid to the dual lid configuration as explained above, is conservative since carbon steel has a higher Young’s modulus than stainless steel. Therefore, the carbon steel bottom plate would be stiffer, and the stress in the top plate would be less versus the dual lid configuration with stainless steel top and bottom plates.

- Evaluation of MPC Baseplate Alternate Support Configuration

The stress state in the MPC baseplate and adjacent canister is evaluated to assess the effect of the discrete support of the MPC under the action of vertical loading plus pressure and temperature.

The alternate MPC supports consist of bearing pads (shims) at six locations around the periphery plus a central support to transfer vertical loads to the HI-STORM. The baseplate of the MPC has been previously analyzed under loading from the fuel basket and the fuel assemblies assuming the baseplate plate continuously supported around the periphery by the MPC canister shell (e.g., this condition arises during lifting and lowering of the MPC into the storage overpack). To evaluate the effect of a discrete support configuration, a finite element model of ½ of the baseplate is constructed using shell elements and includes a sufficient portion of the MPC canister to simulate the canister-to-baseplate joint and the bending boundary layer in the canister shell. Vertical loads from fuel assemblies and fuel basket are applied to the baseplate as a uniform pressure and a ring loading, respectively (these loads have been applied in the same manner in the evaluation of the baseplate under the MPC lowering condition for HI-STORM 100 system). The total vertical load is resisted at the peripheral and central discrete support locations. Under normal conditions of storage, the baseplate/canister is subject to normal service pressure and temperature plus the one-g dead weight loading. The state of stress in the MPC under design pressure, and short-term pressure limit and operating temperatures, have been previously considered using an axisymmetric finite element model. The results are discussed above (see “Finite Element Analysis”) and summarized in Tables 3.4.7 (Design condition) and 3.4.8 (Level A condition). In Table 3.4.8, although the actual metal temperatures were used to develop the solution for the thermal stresses, the allowable stresses were conservatively chosen at the design temperature limits rather than at the actual operating temperatures.

The stress intensities arising in the MPC baseplate and in the lower portion of the canister from the added vertical load are added to the previously determined stress intensities arising from internal pressure and temperature (reported in Table 3.4.8 and adjusted downward for actual service pressure) to obtain the total stress intensity for the Level A normal operating condition. The computed stress intensities are then amplified to simulate the vertical seismic event and again summed with the results from Table 3.4.8 to obtain the total stress intensity for the Level D condition.

The primary and secondary stress intensities in the MPC baseplate and canister shell are computed for the Level A short-term normal operating condition. The Level B off-normal condition is bounded by the short-term normal condition which has an identical pressure limit, comparable temperatures, and is evaluated against Level A allowables. The maximum primary stress intensity in the MPC baseplate is also determined for the Level D vertical seismic event. All computed safety factors are above 1.0.

#### 3.4.4.3.1.3 Elastic Stability and Yielding of the MPC Basket under Compression Loads (Load Case F3 in Table 3.1.3)

This load case corresponds to the scenario wherein the loaded MPC is postulated to drop causing a compression state in the fuel basket panels.

##### a. Elastic Stability

Following the provisions of Appendix F of the ASME Code [3.4.3] for stability analysis of Subsection NG structures, (F-1331.5(a)(1)), a comprehensive buckling analysis is performed using

ANSYS. For this analysis, ANSYS's large deformation capabilities are used. This feature allows ANSYS to account for large nodal rotations in the fuel basket, which are characteristic of column buckling. The interaction between compressive and lateral loading, caused by the deformation, is exactly included. Subsequent to the large deformation analysis, the basket panel that is most susceptible to buckling failure is identified by a review of the results. The lateral displacement of a node located at the mid-span of the panel is measured for the range of impact decelerations. The buckling or collapse load is defined as the impact deceleration for which a slight increase in its magnitude results in a disproportionate increase in the lateral displacement.

The stability requirement for the MPC fuel basket under lateral loading is satisfied if two-thirds of the collapse deceleration load is greater than the design basis horizontal acceleration (Table 3.1.2). This analysis was performed for the HI-STAR 100 submittal (Docket Number 72-1008) under a 60g deceleration loading. Within the HI-STAR 100 FSAR (Docket Number 72-1008), Figures 3.4.27 through 3.4.32 are plots of lateral displacement versus impact deceleration for the MPC-24, MPC-32, and MPC-68. It should be noted that the displacements (in the HI-STAR 100 FSAR) in Figures 3.4.27 through 3.4.31 are expressed in  $1 \times 10^{-1}$  inch and Figure 3.4.32 is expressed in  $1 \times 10^{-2}$  inch. The plots in the HI-STAR 100 FSAR clearly show that the large deflection collapse load of the MPC fuel basket is greater than 1.5 times the design basis deceleration for all baskets in all orientations. The results for the MPC-24E are similar. Thus, the requirements of Appendix F are met for lateral deceleration loading under Subsection NG stress limits for faulted conditions.

An alternative solution for the stability of the fuel basket panel is obtained using the methodology espoused in NUREG/CR-6322 [3.4.13]. In particular, we consider the fuel basket panels as wide plates in accordance with Section 5 of NUREG/CR-6322. We use eq.(19) in that section with the "K" factor set to the value appropriate to a clamped panel. Material properties are selected corresponding to a metal temperature of 600 degrees F which bounds computed metal temperatures at the periphery of the basket. In general, the basket periphery sees the largest loading in an impact scenario. The critical buckling stress is:

$$\sigma_{cr} = \left( \frac{\pi}{K} \right)^2 \frac{E}{12(1-\nu^2)} \left( \frac{h}{a} \right)^2$$

where h is the panel thickness, a is the unsupported panel length, E is the Young's Modulus of Alloy X at 600 degrees F,  $\nu$  is Poisson's Ratio, and  $K=0.65$  (per Figure 6 of NUREG/CR-6322).

The MPC-24 has a small h/a ratio; the results of the finite element stress analyses under design basis deceleration load show that this basket is subject to the highest compressive load in the panel. Therefore, the critical buckling load is computed using the geometry of the MPC-24. The following table shows the results from the finite element stress analysis and from the stability calculation.

<b>Panel Buckling Results From NUREG/CR-6322</b>			
<b>Item</b>	<b>Finite Element Stress (ksi)</b>	<b>Critical Buckling Stress (ksi)</b>	<b>Factor of Safety</b>
Stress	12.585	44.44	3.531

For a stainless steel member under an accident condition load, the recommended safety factor is 2.12. We see that the calculated safety factor exceeds this value; therefore, we have independently confirmed the stability predictions of the large deflection analysis based on classical plate stability analysis by employing a simplified method.

Stability of the basket panels, under longitudinal deceleration loading, is demonstrated in the following manner. Under 60g deceleration in Docket Number 71-9261, the axial compressive stress was computed for all fuel basket types, and the bounding result was determined as:

$$4,074 \text{ psi} \quad (\text{for MPC 24E})$$

For the 45g design basis decelerations for HI-STORM 100, the basket axial stresses are reduced by 25%.

The above values represent the amplified weight, including the nonstructural sheathing and the neutron absorber material, divided by the bearing area resisting axial movement of the basket. To demonstrate that elastic instability is not a concern, the buckling stress for an MPC-24 flat panel is computed.

For elastic stability, Reference [3.4.8] provides the formula for critical axial stress as

$$\sigma_{cr} = \frac{4 \pi^2 E}{12 (1 - \nu^2)} \left( \frac{T}{W} \right)^2$$

where T is the panel thickness and W is the width of the panel, E is the Young's Modulus at the metal temperature and  $\nu$  is the metal Poisson's Ratio. The following table summarizes the calculation for the critical buckling stress using the formula given above:

<b>Elastic Stability Result for a Flat Panel</b>	
Reference Temperature	752 degrees F
T (MPC-24)	5/16 inch
W	10.777 inch
E	24,436,000 psi
Critical Axial Stress	74,280 psi

It is noted the critical axial stress is an order of magnitude greater than the computed basket axial stress reported in the foregoing and demonstrates that elastic stability under longitudinal deceleration load is not a concern for any of the fuel basket configurations.

#### b. Yielding

The safety factor against yielding of the basket under longitudinal compressive stress from a design basis inertial loading is given, using the bounding result for the MPC-24E, by

$$SF = 17,100/4,074 = 4.197$$



Therefore, plastic deformation of the fuel basket under design basis deceleration is not credible.

#### 3.4.4.3.1.4 MPC Baseplate Analysis (Load Case E2)

Minimum safety factors have been reported for Load Case E2 in Subsection 3.4.3.6 where an evaluation has been performed for stresses under three times the apparent load  $D^*$ . A bounding analysis is performed in the HI-STAR 100 FSAR (Docket Number 72-1008, Appendix 3.I) to evaluate the stresses in the MPC baseplate during a vertical end drop (Load Case E3.a). During a fire (Load Case E5), the MPC baseplate is subjected to the accident internal pressure, dead load, and the fire temperature (which serves only to lower the allowable strengths). The results for Load Cases E3.a and E5 are summarized below:

MPC Baseplate Minimum Safety Factors – Load Cases E3.a and E5			
Item	Value (ksi)	Allowable (ksi)	Safety Factor
Center of Baseplate – Primary Bending (Load Case E3.a) (Note 1)	35.93	67.32	1.87
Center of Baseplate – Primary Bending (Load Case E5)	46.38	54.23	1.17

Notes:

1. Detailed analysis presented in Appendix 3.I of HI-STAR 100 FSAR (Docket 72-1008).

#### 3.4.4.3.1.5 Analysis of the MPC Top Closure (Load Case E2)

The FSAR for the HI-STAR 100 System (Docket Number 72-1008, Appendix 3.E) contains stress analysis of the MPC top closure during lifting. Loadings in that analysis are also valid for the HI-STORM 100 System. In addition, the results from top closure lid structural evaluations under the short-term normal condition pressure limit (per Table 2.2.1) are presented in Subsection 3.4.3.6 and Table 3.4.4. From Table 2.2.1, the off-normal internal pressure limit is 110 psi, or ten percent greater than the long-term normal design pressure limit. Whereas Level A service limits are used to establish allowables for the normal design pressure, Level B service limits are used for off-normal loads. Since Subsection NB of the ASME Code permits an identical 10% increase in allowable stress intensity values for stress intensities generated by Level B Service Loadings, it stands to reason that the safety factors reported for normal pressure are also valid for the case of off-normal design internal pressure.

#### 3.4.4.3.1.6 Structural Analysis of the Fuel Support Spacers (Load Case E3.a)

Upper and lower fuel support spacers are utilized to position the active fuel region of the spent nuclear fuel within the poisoned region of the fuel basket. It is necessary to ensure that the spacers will continue to maintain their structural integrity after an accident event. Ensuring structural integrity implies that the spacer will not buckle under the maximum compressive load, and that the maximum compressive stress will not exceed the compressive strength of the spacer material (Alloy X). Detailed calculations in Docket Number 72-1008, Appendix 3.J, demonstrate that large structural margins in the fuel spacers are available for the entire range of spacer lengths which may

be used in HI-STORM 100 applications (for the various acceptable fuel types). The calculations for the HI-STORM 100 45g load are bounded by those for the HI-STAR 100 60g load.

#### 3.4.4.3.1.7 External Pressure (Load Case E1.b, Table 3.1.4)

The design external pressure for the MPC is zero psig. The outer surface of the MPC shell is conservatively subject to a net external pressure of 2 psig. The methodology for analysis of the MPC under this external pressure is provided in the HI-STAR 100 FSAR Docket Number 72-1008. Using the identical methodology with input loads and decelerations appropriate to the HI-STORM, safety factors  $> 1.0$  are obtained for all relevant load cases.

#### 3.4.4.3.1.8 Miscellaneous MPC Structural Evaluations

Calculations are performed to determine the minimum fuel basket weld size, the capacity of the sheathing welds, the stresses in the MPC cover plates, and the stresses in the fuel basket supports (or basket shims). The following paragraphs briefly describe each of these evaluations.

The fillet welds in the fuel basket honeycomb are made by an autogenous operation that has been shown to produce highly consistent and porosity free weld lines. However, Subsection NG of the ASME Code permits only 40% quality credit on double fillet welds which can be only visually examined (Table NG-3352-1). Subsection NG, however, fails to provide a specific stress limit on such fillet welds. In the absence of a Code mandated limit, Holtec International's standard design procedure requires that the weld section possess as much load resistance capability as the parent metal section. Since the loading on the honeycomb panels is essentially that of section bending, it is possible to develop a closed form expression for the required weld throat thickness "t" corresponding to panel thickness "h".

The sheathing is welded to the fuel basket cell walls to protect and position the neutron absorber material. Force equilibrium relationships are used to demonstrate that the sheathing weld is adequate to support a 45g deceleration load applied vertically and horizontally to the sheathing and the confined neutron absorber material. The analysis assumes that the weld is continuous and then modifies the results to reflect the amplification due to intermittent welding.

The MPC cover plates are welded to the MPC lid during loading operations. The cover plates are part of the confinement boundary for the MPC. No credit is taken for the pressure retaining abilities of the quick disconnect couplings for the MPC vent and drain. Therefore, the MPC cover plates must meet ASME Code, Section III, Subsection NB limits for normal, off-normal, and accident conditions. Conservatively, the accident condition pressure loading is applied, and it is demonstrated that the Level A limits for Subsection NB are met where possible.

The fuel basket internal to the MPC canister is supported by a combination of angle fuel basket supports and flat plate or solid bar fuel basket supports. Alternatively, the fuel basket internal to the MPC canister can be supported by basket shims. These fuel basket supports (or basket shims) are subject to significant load only when a lateral acceleration is applied to the fuel basket and the contained fuel. The quasi-static finite element analyses of the MPC's, under lateral inertia loading, focused on the structural details of the fuel basket and the MPC shell. Basket supports were

modeled in less detail, which served only to properly model the load transfer path between fuel basket and canister. Safety factors reported for the fuel basket supports from the finite element analyses, are overly conservative, and do not reflect available capacity of the fuel basket angle support or the basket shim. A strength of materials analysis of the fuel basket angle supports (or basket shims) is performed to complement the finite element results. Weld stresses in the load path are computed, direct stress in the support members is evaluated, and stability of the support legs is examined for all fuel basket support configurations.

The results of these evaluations are summarized in the tables below.

<b>Minimum Weld Sizes for Fuel Baskets</b>			
<b>Basket Type</b>	<b>Panel Thickness (h), in</b>	<b>t/h Ratio</b>	<b>Minimum Weld Size (t), in</b>
MPC-24	5/16	0.568	0.177
MPC-68	1/4	0.533	0.133
MPC-32	9/32	0.580	0.163
MPC-24E	5/16	0.519	0.162

<b>Miscellaneous Stress Results for MPC</b>			
<b>Item</b>	<b>Stress (ksi)</b>	<b>Allowable Stress (ksi)</b>	<b>Safety Factor</b>
Shear Stress in Sheathing Weld	7.724	27.93	3.62
Bending Stress in MPC Cover Plate (Level A)	15.49	24.6	1.59
Bending Stress in MPC Cover Plate (Level D)	28.16	59.04	2.10
Shear Stress in MPC Cover Plate Weld	7.55	18.93	2.51
Shear Stress in Fuel Basket Angle Support Weld	19.78	26.67	1.35
Combined Stress in Fuel Basket Support Plates	32.393	59.1	1.82
Load on Basket Support Legs (Stability)	2.185 kips/in	8.886 kips/in	4.07
Shear Stress in Basket Shim Weld	31.78	32.385	1.019
Combined Stress in Basket Shim	31.246	63	2.02
Load on Basket Shim (Stability)	8.015 kips/in	225.393 kips/in	28.12

#### 3.4.4.3.1.9 Structural Integrity of Damaged Fuel Containers

The damaged fuel containers or canisters (DFCs) to be deployed in the HI-STAR 100 System transport package have been evaluated to demonstrate that the containers are structurally adequate to support the mechanical loads postulated during normal operations, while in long-term storage, and during a hypothetical end drop. The evaluations address the following damaged/failed fuel

containers for transportation in the HI-STAR 100 System:

- Holtec-designed MPC-24E (PWR) DFC
- Holtec-designed MPC-68 (BWR) DFC
- Transnuclear-designed DFC for Dresden Unit 1 fuel
- Transnuclear-designed Thoria Rod Canister for Dresden Unit 1

The structural load path in each of the analyzed containers is evaluated using basic strength of materials formulations. The various structural components are modeled as axial or bending members and their stresses are computed. Depending on the particular DFC, the load path includes components such as the container sleeve and collar, various weld configurations, load tabs, closure components and lifting bolts. Axial plus bending stresses are computed, together with applicable bearing stresses and weld stresses. Comparisons are then made with the appropriate allowable strengths at temperature. Input data for all DFCs comes from the applicable drawings. The design temperature for lifting evaluations is set at 150°F (since the DFC is in the spent fuel pool). The design temperature for accident conditions is set at 725°F.

*For those DFCs designed to be handled when loaded, the DFC lift point(s) must be designed in accordance with Item 3 in Section 5.1.6 of NUREG-0612 [3.1.1]. The remaining components of the damaged fuel container are governed by the stress limits of the ASME Code Section III, Subsection NG [3.4.10] and Section III, Appendix F [3.4.3], as applicable.*

The following table presents the minimum safety factors, from all of the stress computations, for each of the above listed DFCs.

DFC Type	Loading Condition – DFC Component	Calculated Stress (ksi)	Allowable Stress (ksi)	Safety Factor = (Allowable Stress) / (Calculated Stress)	Remarks
Holtec-designed MPC-24E (PWR) DFC					
	60g End Drop – Baseplate-to-Container Sleeve Welds	3.95	26.59	6.73	ASME Level D stress limit
Holtec-designed MPC-68 (BWR) DFC					
	60g End Drop – Baseplate-to-Container Sleeve Welds	1.59	26.59	16.7	ASME Level D stress limit
Transnuclear-designed DFC for Dresden Unit 1	Normal Lift – Lid Frame Assembly	0.527	4.583	8.70	Bearing stress
	60g End Drop – Bottom Assembly	12.32	37.92	3.08	ASME Level D stress limit
Transnuclear-designed Thoria Rod Canister for Dresden Unit 1	Normal Lift – Lid Frame Assembly	0.373	4.583	12.3	Bearing stress
	60g End Drop – Bottom Assembly	8.73	37.92	4.34	ASME Level D stress limit

The table above demonstrates that the DFCs are structurally adequate to support the mechanical

loads postulated during normal operations and during a hypothetical end drop. Moreover, since the HI-STAR design basis handling accident bounds the corresponding load for HI-STORM (60g vs. 45g), the DFCs can be carried safely in both the HI-STAR and HI-STORM Systems.

#### 3.4.4.3.1.10 Analysis of MPC Baseplate and Closure Lid (Load Case E5)

During a fire (Load Case E5), the MPC baseplate is subjected to the accident pressure plus dead load, and the fire temperature (which serves only to lower the allowable strengths). The results of this analysis are summarized below:

MPC Baseplate Minimum Safety Factors – Load Cases E5			
Item	Value (ksi)	Allowable (ksi)	Safety Factor
Center of Baseplate – Primary Bending (Load Case E5)	46.58	54.25	1.16

The closure lid and the closure lid peripheral weld are also examined for maximum stresses developed during the storage fire. The closure lid is modeled as a single simply supported plate and is subject to dead load plus accident internal pressure. Results are presented for both the single and dual lid configuration (in parentheses). The results for minimum safety factor are reported in the table below:

MPC Top Closure Lid – Minimum Safety Factors – Load Case E5			
Item	Stress (ksi) or Load (lb.)	Allowable Stress (ksi) or Load Capacity (lb.)	Safety Factor
Lid Bending Stress – Load Case E5	3.158/(6.316)	54.225	17.17/(8.58)
Lid-to-Shell Peripheral Weld Load – Load Case E5	713,047	1,477,000 <sup>†</sup>	2.07

<sup>†</sup> Based on 0.625" single groove weld and conservatively includes a quality factor of 0.45.

We note from the above that all safety factors are greater than 1.0.

#### 3.4.4.3.2 HI-STORM 100 Storage Overpack Stress Calculations

The structural functions of the storage overpack are stated in Section 3.1. The analyses presented here demonstrate the ability of components of the HI-STORM 100 storage overpack to perform their structural functions in the storage mode. Load Cases considered are given in Table 3.1.5. The nomenclature used to identify the load cases (Load Case Identifier) considered is also given in Table 3.1.5.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability of the MPC from the storage overpack. Results obtained using the HI-STORM 100 configuration

are identical to or bound results for the HI-STORM 100S configuration.

#### 3.4.4.3.2.1 HI-STORM 100 Compression Under the Static Load of a Fully Loaded HI-TRAC Positioned on the Top of HI-STORM 100 (Load Case 01 in Table 3.1.5)

During the loading of HI-STORM 100, a HI-TRAC transfer cask with a fully loaded MPC may be placed on the top of a HI-STORM 100 storage overpack. During this operation, the HI-TRAC may be held by a single-failure-proof lifting device so a handling accident is not credible. The HI-STORM 100 storage overpack must, however, possess the compression capacity to support the additional dead load. The following analysis provides the necessary structural integrity demonstration; safety factors are large and results for the HI-STORM 100 overpack are representative of the margins for the 100S and 100S Version B overpacks.

Define the following quantities for analysis purposes:

$W_{HT}$  = Bounding weight of HI-TRAC 125D (loaded w/ MPC-32) = 236,000 lb (Table 3.2.2)

$W_{MD}$  = Weight of mating device = 15,000 lb

$W_{TOTAL} = W_{HT} + W_{MD} = 251,000$  lb

The total weight of the HI-TRAC 125D plus the mating device is greater than the weight of a loaded HI-TRAC 125 with the transfer lid. Therefore, the following calculations use the weight for the HI-TRAC 125D as input.

The dimensions of the compression components of HI-STORM 100 are as follows:

outer diameter of outer shell =	$D_o = 132.5''$
thickness of outer shell =	$t_o = 0.75''$ (1'' for HI-STORM 100S Version B)
outer diameter of inner shell =	$D_i = 76''$
thickness of inner shell =	$t_i = 1.25''$ (1'' for HI-STORM 100S Version B)
thickness of radial ribs =	$t_r = 0.75''$ (ribs are not full-length for HI-STORM 100S Version B; HI-STORM 100S can be fabricated with full or partial length ribs)

In what follows, detailed results are provided using the classic HI-STORM 100 dimensions. While Bill of Material 1575 provides the option to fabricate the inner and outer shells from 1'' thick material, the above dimensions (i.e.,  $t_o$  and  $t_i$ ) are used because they minimize the total cross sectional metal area.

The metal area of the outer shell is

$$A_o = \frac{\pi}{4} (D_o^2 - (D_o - 2t_o)^2) = \frac{\pi}{4} (132.5^2 - 131^2) \\ = 310.43 \text{ in}^2$$

The metal area of the radial ribs is

$$A_r = 4t_r (D_o - 2t_o - D_i) / 2 = \frac{3}{2} (131 - 76) = 82.5 \text{ in}^2$$

The metal area of the inner shell is

$$A_i = \frac{\pi}{4} (76^2 - 73.5^2) = 293.54 \text{ in}^2$$

In the HI-STORM 100, there are four radial ribs that extend full length and can carry load. For the HI-STORM 100S and the HI-STORM 100S Version B, the radial ribs are not counted as part of the compression load carrying area since they are not required to be full-length. The concrete radial shield can also support compression load. The area of concrete available to support compressive loading is

$$\begin{aligned} A_{\text{concrete}} &= \frac{\pi}{4} ((D_o - 2t_o)^2 - (D_i)^2) - A_r \\ &= \frac{\pi}{4} (131^2 - 76^2) - 82.5 \text{ in}^2 \\ &= (8,994 - 82.5) \text{ in}^2 = 8,859.5 \text{ in}^2 \end{aligned}$$

The areas computed above are calculated at a section below the air outlet vents. To correct the above areas for the presence of the air outlet vents (HI-STORM 100 only since HI-STORM 100S and HI-STORM 100S, Version B have the air outlet vents located in the lid), we note that Bill-of-Materials 1575 in Chapter 1 gives the size of the horizontal plate of the air outlet vents as:

Peripheral width =  $w = 16.5''$

Radial depth =  $d = 27.5''$  (over concrete in radial shield)

Using these values, the following final areas are obtained:

$$A_o = A_o(\text{no vent}) - 4t_o w = 260.93 \text{ sq. inch}$$

$$A_i = A_i(\text{no vent}) - 4t_i w = 211.04 \text{ sq. inch}$$

$$A_{\text{concrete}} = A_{\text{concrete}}(\text{no vent}) - 4dw = 7044.2 \text{ sq. inch}$$

The loading case is a Level A load condition. The load is apportioned to the steel and to the concrete in accordance with the values of EA for the two materials ( $E(\text{steel}) = 28,000,000 \text{ psi}$  and  $E(\text{concrete}) = 3,605,000 \text{ psi}$ ).

$$\begin{aligned} EA(\text{steel}) &= 28 \times 10^6 \text{ psi} \times (260.93 + 211.04 + 82.5) \text{ in}^2 \\ &= 15,525.2 \text{ lb} \times 10^6 \text{ lbs.} \end{aligned}$$

$$\begin{aligned} EA(\text{concrete}) &= 3.605 \times 10^6 \times (7044.2) \text{ in}^2 \\ &= 25,394.3 \times 10^6 \text{ lb.} \end{aligned}$$

Therefore, the total HI-TRAC load will be apportioned as follows:

$$F_{\text{STEEL}} = (15,525.2/40,919.5) \times 251,000 = 95,231.5 \text{ lb.}$$

$$F_{\text{CONCRETE}} = (25,394.3/40,919.5) \times 251,000 = 155,768.5 \text{ lb.}$$

Therefore, if the load is apportioned as above, with all load-carrying components in the path acting, the compressive stress in the steel is

If we conservatively neglect the compression load bearing capacity of concrete, then

$$\sigma_{\text{STEEL}} = \frac{251,000}{554.5} = 452.7 \text{ psi}$$

If we include the concrete, then the maximum compressive stress in the concrete is:

$$\sigma_{\text{CONCRETE}} = \frac{F_{\text{CONCRETE}}}{A_{\text{CONCRETE}}} = 22.1 \text{ psi}$$

It is clear that HI-STORM 100 storage overpack can support the dead load of a fully loaded HI-TRAC 125D and the mating device placed on top for MPC transfer into or out of the HI-STORM 100 storage overpack cavity. The calculated stresses at a cross-section through the air outlet ducts are small and give rise to large factors of safety. The metal cross-section at the base of the HI-STORM storage overpack will have a slightly larger metal area (because the width of the air-inlet ducts is smaller) but will be subject to additional dead load from the weight of the supported metal components of the HI-STORM storage overpack plus the loaded HI-TRAC weight. At the base of the storage overpack, the additional stress in the outer shell and the radial plates is due solely to the weight of the component. Based on the maximum concrete density, the additional stress in these components is computed as:

$$\Delta\sigma = (200\text{lb./cu.ft.}) \times 18.71 \text{ ft./}144 \text{ sq.in./sq.ft.} = 26.0 \text{ psi}$$

This stress will be further increased by a small amount because of the material cut away by the air-inlet ducts; however, the additional stress still remains small. The inner shell, however, is subject to additional loading from the top lid of the storage overpack and from the radial shield. From the Structural Calculation Package (HI-981928)(see Subsection 3.6.4 for the reference), and from Table 3.2.1, the following weights are obtained (for conservatism, use the 100S, Version B lid weight with 200 pcf concrete even though the shell geometry is for the classic HI-STORM 100):

HI-STORM 100S, Version B Top Lid weight < 29,000 lb.

HI-STORM 100 Inner Shell weight < 19,000 lb.

HI-STORM 100 Shield Shell weight < 11,000 lb.



Note that the shield shell was removed from the HI-STORM 100 design as of June 2001. However, it is conservative to include the shield shell weight in the following calculations.

Using the calculated inner shell area at the top of the storage overpack for conservatism, gives the metal area of the inner shell as:

$$A_i = A_i(\text{no vent}) - 4t_i w = 211.04 \text{ sq. inch}$$

Therefore, the additional stress from the HI-STORM 100S, Version B storage overpack components, at the base of the overpack, is:

$$\Delta\sigma = 280 \text{ psi}$$

and a maximum compressive stress in the inner shell predicted as:

$$\text{Maximum stress} = 453 \text{ psi} + 280 \text{ psi} = 733 \text{ psi}$$

The safety factor at the base of the storage overpack inner shell (minimum section) is

$$\text{SF} = 17,500\text{psi}/733 \text{ psi} = 23.9$$

The preceding analysis is bounding for the 100 Ton HI-TRAC transfer cask because of the lower HI-TRAC weight.

The preceding analysis is representative of all overpacks since the bounding lid weight from the Version B has been used. Based on the computed safety factor, it is concluded that all versions of the HI-STORM 100 and HI-STORM 100S can safely support the heaviest HI-TRAC while performing a vertical fuel transfer operation.

#### 3.4.4.3.2.2 HI-STORM 100 Lid Integrity Evaluation (Load Case 02.c, Table 3.1.5)

A non-mechanistic tip over of the HI-STORM 100 results in high decelerations at the top of the storage overpack. The storage overpack lid diameter is less than the storage overpack outer diameter. This ensures that the storage overpack lid does not directly strike the ground but requires analysis to demonstrate that the lid remains intact and does not separate from the body of the storage overpack. Figure 3.4.19 shows the scenario.

The HI-STORM 100 overpack has two lid designs, which rely on different mechanisms to resist separation from the overpack body. The original design relies solely on the lid studs to resist the shear and axial loads on the lid. In the new design, the bolt holes are enlarged and a shear ring is welded to the underside of the lid top plate. These changes insure that the lid studs only encounter axial (tensile) loads. The in-plane load is resisted by the shear ring as it bears against the top plate. The HI-STORM 100S and the HI-STORM 100S, Version B has only one lid design, which utilizes a shear ring. Calculations have been performed for both HI-STORM 100 lid configurations, as well as the HI-STORM 100S and the HI-STORM 100S, Version B lid geometry, to demonstrate that the lid can withstand a non-mechanistic tip-over.

The deceleration level for the non-mechanistic tip-over bounds all other decelerations, directed in the plane of the lid, experienced under other accident conditions such as flood or earthquake as can be demonstrated by evaluating the loads resulting from these natural phenomena events.

It is shown that the weight of the HI-STORM 100 lid, amplified by the design basis deceleration, can be supported entirely by the shear capacity available in the four studs<sup>†</sup>. If only a single stud is loaded initially during a tipover (because of tolerances), the stud hole will enlarge rather than the stud fail in shear. Therefore, it is assured that all four bolts will resist the tipover load regardless of the initial position of the HI-STORM 100 lid.

The following tables summarize the limiting results obtained from the detailed analyses, and from the similar detailed analysis for the HI-STORM 100 lid with shear ring, for the HI-STORM 100S(243), and for the HI-STORM 100S, Version B(229). The results for the longer HI-STORM 100S and HI-STORM 100S, Version B bound the corresponding results for the shorter versions of these units.

<b>HI-STORM 100 Top Lid Integrity (No Shear Ring)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Shell-Lid Top Plate Weld Shear Stress	6.733	29.4	4.367
Lid Shell-Lid Top Plate Combined Stress	9.11	29.4	3.226
Attachment Bolt Shear Stress	44.82	60.9	1.359
Attachment Bolt Combined Shear and Tension Interaction at Interface with Anchor Block	-----	-----	1.21

<b>HI-STORM 100 Top Lid Integrity (With Shear Ring)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Top Plate-to-Lid Shell Weld Combined Stress	7.336	29.4	4.007
Shield Block Shells-to-Lid Top Plate Weld Combined Stress	1.768	29.4	16.63
Attachment Bolt Tensile Stress	28.02	107.13	3.823
Shear Ring-to-Lid Top Plate Weld Stress	32.11	40.39	1.258
Shear Ring Bearing Stress	25.43	63.0	2.477
Top Plate-to-Outer Shell Weld Stress	35.61	40.39	1.134

<sup>†</sup> The tip-over event is non-mechanistic by definition since the HI-STORM 100 System is designed to preclude tip-over under all normal, off-normal, and accident conditions of storage, including extreme natural phenomena events. Thus, the tip-over event cannot be categorized as an operating or test condition as contemplated by ASME Section III, Article NCA-2141. The bolted connection between the overpack top lid and the overpack body provided by the top lid studs and nuts serves no structural function during normal or off-normal storage conditions, or for credible accident events. Therefore, the ASME Code does not apply to the construction of the HI-STORM top plate-to-overpack connection (the lid studs, nuts, and the through holes in the top plate). However, for conservatism, the stress limits from ASME III, Subsection NF are used for the analysis of the lid bolts.

<b>HI-STORM 100S(243) Top Lid Integrity<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Inner and Outer Shell Weld to Base	17.61	29.4	1.669
Shield Block Shell-to-Lid Weld Shear Stress	7.692	29.4	3.822
Attachment Bolt Tensile Stress	37.38	107.13	2.866
Shear Ring-to-Overpack Shell Weld Stress	33.24	42.0	1.264
Shear Ring Bearing Stress	19.36	63.0	3.254
Lid Shield Ring-to-Shear Ring Weld Stress	20.95	42.0	2.004

<sup>†</sup> Results are based on a bounding weight of 28,000 lb for the HI-STORM 100S top lid.

<b>HI-STORM 100S, Version B(229) Top Lid Integrity<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Outer Ring to Lid Shield Ring Weld	26.15	30.3	1.159
Shield Block Shell-to-Lid Weld Shear Stress	26.94	30.3	1.125
Attachment Bolt Tensile Stress	41.563	107.13	2.578
Shear Ring-to-Overpack Shell Weld Stress	30.57	42.0	1.374
Shear Ring Bearing Stress	20.59	63.0	3.06
Lid Shield Ring-to-Shear Ring Weld Stress	32.36	42.0	1.298

<sup>†</sup> Results are based on a bounding weight of 29,000 lb for the HI-STORM 100S, Version B top lid.

#### 3.4.4.3.2.3 Vertical Drop of HI-STORM 100 Storage Overpack (Load Case 02.a of Table 3.1.5)

A loaded HI-STORM 100, with the top lid in place, drops vertically and impacts the ISFSI. Figure 3.4.20 illustrates the drop scenario. The regions of the structure that require detailed examination are the storage overpack top lid, the inlet vent horizontal plate, the pedestal shield, the inlet vent vertical plate, and all welds in the load path. These components are examined for the Level D event of a HI-STORM 100 drop developing the design basis deceleration.

The table provided below summarizes the results of the analyses for the weight and configuration of the HI-STORM 100. The results for the HI-STORM 100S are bounded by the results given below, except as noted. Any calculation pertaining to the pedestal is bounding since the pedestal dimensions and corresponding weights are less in the HI-STORM 100S. The HI-STORM 100S, Version B, however, has sufficient differences in configuration to merit a separate evaluation using similar analyses; therefore, a separate summary table of results is provided for the HI-STORM 100S, Version B.

<b>HI-STORM 100 Load Case 02.a Evaluation</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Bottom Plate Bending Stress Intensity	6.00	58.7	9.777 <sup>†</sup>
Lid Bottom Plate Collapse Load	10450x1.06 (in.*lb./in.)	12730 (in.*lb./in.)	1.15 <sup>†</sup>
Weld- lid bottom plate-to-lid shell	10.91	29.4	2.695
Lid Shell – Membrane Stress Intensity	1.90	39.1	20.58
Lid Top (2" thick) Plate Bending Stress Intensity	11.27	58.7	5.208*
Inner Shell –Membrane Stress Intensity	11.46	39.1	3.41
Outer Shell –Membrane Stress Intensity	3.401	39.1	11.495
Inlet Vent Horizontal Plate Bending Stress Intensity	46.20	58.7	1.271
Inlet Vent Vertical Plate Membrane Stress Intensity	12.86	39.1	3.04
Pedestal Shield – Compression	1.252	1.266	1.011
Weld – outer shell-to-baseplate	2.569	29.4	11.443
Weld – outer shell-to-baseplate (100S)	3.854	29.4	7.629
Weld – inner shell-to-baseplate	6.644	29.4	4.425
Weld – inner shell-to-baseplate (100S)	8.627	29.4	3.408
Weld-Pedestal shell-to-baseplate	2.281	29.4	12.887

<sup>†</sup> Note that the dynamic load factor for the lid top plate is negligible and for the lid bottom plate is 1.06. This dynamic load factor has been incorporated in the above table.

\* For the HI-STORM 100S, this safety factor is conservatively evaluated to be 1.357 because of increased load on the upper of the two lid plates.

Applicable analyses are performed for the HI-STORM 100S, Version B for the amplified loads resulting from the vertical drop of a fully loaded cask with the top lid in place.

<b>HI-STORM 100S, Version B Load Case 02.a Evaluation</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Vent Shield Bending Stress Intensity	13.09	36.0	2.75
Lid Inner Ring Compression	16.80	24.0	1.43
Inner Shell Compression	8.180	35.94	4.39
Outer Shell Compression	2.604	35.94	13.8

Weld – outer shell-to-baseplate	5.404	29.4	5.44
Weld – inner shell-to-baseplate	7.183	29.4	4.093

An assessment of the potential for instability of the compressed inner and outer shells under the compressive loading during the drop event has also been performed. The methodology is from ASME Code Case N-284 (Metal Containment Shell Buckling Design Methods, Division I, Class MC (8/80)). This Code Case has been previously accepted by the NRC as an acceptable method for evaluation of stability in vessels. The results obtained are conservative in that the loading in the shells is assumed to be uniformly distributed over the entire length of the shells. In reality, the component due to the amplified weight of the shell varies from zero at the top of the shell to the maximum value at the base of the shell. It is concluded that large factors of safety exist so that elastic or plastic instability of the inner and outer shells does not provide a limiting condition. The results for the HI-STORM 100 bound similar results for the HI-STORM 100S since the total weight of the “S” configuration is decreased (see Subsection 3.2). The same methodology has been used for an assessment of the HI-STORM 100S Version B with the same conclusion; namely, that elastic or plastic instability of the inner and outer shells is not a concern under the postulated design basis load cases.

The results do not show any gross regions of stress above the material yield point that would imply the potential for gross deformation of the storage overpack subsequent to the handling accident. MPC stability has been evaluated in the HI-STAR 100 FSAR for a drop event with 60g deceleration and shown to satisfy the Code Case N-284 criteria. Therefore, ready retrievability of the MPC is maintained as well as the continued performance of the HI-STORM 100 storage overpack as the primary shielding device.

#### 3.4.4.3.3 HI-TRAC Transfer Cask Stress Calculations

The structural functions of the transfer cask are stated in Section 3.1. The analyses presented here demonstrate the ability of components of the HI-TRAC transfer cask to perform their structural functions in the transfer mode. Load Cases considered are given in Table 3.1.5.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability.

##### 3.4.4.3.3.1 Analysis of Pocket Trunnions (Load Case 01 of Table 3.1.5)

The HI-TRAC 125 and HI-TRAC 100 transfer casks have pocket trunnions attached to the outer shell and to the water jacket. During the rotation of HI-TRAC from horizontal to vertical or vice versa (see Figure 3.4.18), these trunnions serve to define the axis of rotation. The HI-TRAC is also supported by the lifting trunnions during this operation. Two load conditions are considered: Level A when all four trunnions support load during the rotation; and, Level B when the hoist cable is assumed slack so that the entire load is supported by the rotation trunnions. A dynamic amplification of 15% is assumed in both cases appropriate to a low-speed operation. Figure 3.4.23 shows a free body of the trunnion and shows how the applied force and moment are assumed to be resisted by the weld group that connects the trunnion to the outer shell. Drawings 1880 (sheet 10) and 2145 (sheet 10) show the configuration. An optional construction for the HI-TRAC 100

permits the pocket trunnion base to be split to reduce the “envelope” of the HI-TRAC. For that construction, bolts and dowel pins are used to insure that the force and moment applied to the pocket trunnions are transferred properly to the body of the transfer cask. The analysis also evaluates the bolts and dowel pins and demonstrates that safety factors greater than 1.0 exist for bolt loads, dowel bearing and tear-out, and dowel shear. Allowable strengths and loads are computed using applicable sections of ASME Section III, Subsection NF.

Unlike the HI-TRAC 125 and the HI-TRAC 100, the HI-TRAC 125D and HI-TRAC 100D are designed and fabricated without pocket trunnions. An L-shaped rotation frame is used to upend and downend the HI-TRAC 125D and HI-TRAC 100D, instead of pocket trunnions. Thus, a pocket trunnion analysis is not applicable to the HI-TRAC 125D or the HI-TRAC 100D.

The table below summarizes the results for the HI-TRAC 125 and the HI-TRAC 100:

<b>Pocket Trunnion Weld Evaluation Summary</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)<sup>†</sup></b>	<b>Safety Factor</b>
HI-TRAC 125 Pocket Trunnion-Outer Shell Weld Group Stress	7.979	23.275	2.917
HI-TRAC 125 Pocket Trunnion-Water Jacket Weld Group Stress	5.927	23.275	3.9
HI-TRAC 100 Pocket Trunnion-Outer Shell Weld Group Stress	6.603	23.275	3.525
HI-TRAC 100 Pocket Trunnion-Water Jacket Weld Group Stress	5.244	23.275	4.438
HI-TRAC 100 Pocket Trunnion-Bolt Tension at Optional Split	45.23	50.07	1.107
HI-TRAC 100 Pocket Trunnion-Bearing Stress on Base Surfaces at Dowel	6.497	32.7	5.033
HI-TRAC 100 Pocket Trunnion-Tear-out Stress on Base Surfaces at Dowel	2.978	26.09	8.763
HI-TRAC 100 Pocket Trunnion-Shear Stress on Dowel Cross Section at Optional Split	29.04	37.93	1.306

<sup>†</sup> Allowable stress is reported for the Level B loading, which results in the minimum safety factor.

To provide additional information on the local stress state adjacent to the rotation trunnion, a new finite element analysis is undertaken to provide details on the state of stress in the metal structure surrounding the rotation trunnions for the HI-TRAC 125. The finite element analysis has been based on a model that includes major structural contributors from the water jacket enclosure shell

panels, radial channels, end plates, outer and inner shell, and bottom flange. In the finite element analysis, the vertical trunnion load has been oriented in the direction of the HI-TRAC 125 longitudinal axis. The structural model has been confined to the region of the HI-TRAC adjacent to the rotation trunnion block; the extent of the model in the longitudinal direction has been determined by calculating the length of the “bending boundary layer” associated with a classical shell analysis. This was considered to be a sufficient length to capture maximum shell stresses arising from the Level B (off-normal) rotation trunnion loading. The local nature of the stress around the trunnion block is clearly demonstrated by the finite element results.

Consistent with the requirements of ASME Section III, Subsection NF, for Class 3 components, safety factors for primary membrane stress have been computed. Primary stresses are located away from the immediate vicinity of the trunnion; although the NF Code sets no limits on primary plus secondary stresses that arise from the gross structural discontinuity immediately adjacent to the trunnion, these stresses are listed for information. The results are summarized in the table below for the Level B load distribution for the HI-TRAC 125.

ITEM –HI-TRAC 125	CALCULATED VALUE	ALLOWABLE VALUE
Longitudinal Stress - (ksi) (Primary Stress – Inner Shell)	-0.956	23.275
Tangential Stress (ksi) (Primary Stress - Inner Shell)	-1.501	23.275
Longitudinal Stress (ksi) (Primary Stress – Outer Shell)	-0.830	23.275
Tangential Stress (ksi) (Primary Stress - Outer Shell)	-0.436	23.275
Longitudinal Stress - (ksi) (Primary Stress – Radial Channels)	2.305	23.275
Tangential Stress (ksi) (Primary Stress - Radial Channels)	-0.631	23.275
Longitudinal Stress - (ksi) (Primary plus Secondary Stress -Inner Shell)	1.734	No Limit (34.9)*
Tangential Stress (ksi) (Primary plus Secondary Stress - Inner Shell)	-1.501	NL
Longitudinal Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	2.484	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	-2.973	NL
Longitudinal Stress - (ksi) (Primary plus Secondary Stress - Radial Channels)	-13.87	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Radial Channels)	-2.303	NL

\* The NF Code sets no limits (NL) for primary plus secondary stress (see Table 3.1.17). Nevertheless, to demonstrate the robust design with its large margins of safety, we list here, for information only, the allowable value for Primary Membrane plus Primary Bending Stress appropriate to temperatures up to 650 degrees F.

The only stress of any significance is the longitudinal stress in the radial channels. This stress occurs immediately adjacent to the trunnion block/radial channel interface and by its localized

nature is identifiable as a stress arising at the gross structural discontinuity (secondary stress).

The finite element analysis has also been performed for the HI-TRAC 100 transfer cask. The following table summarizes the results:

<b>ITEM –HI-TRAC 100</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Longitudinal Stress - (ksi) (Primary Stress –Inner Shell)	-0.756	23.275
Tangential Stress (ksi) (Primary Stress - Inner Shell)	-2.157	23.275
Longitudinal Stress (ksi) (Primary Stress – Outer Shell)	-0.726	23.275
Tangential Stress (ksi) (Primary Stress - Outer Shell)	-0.428	23.275
Longitudinal Stress - (ksi) (Primary Stress – Radial Channels)	2.411	23.275
Tangential Stress (ksi) (Primary Stress - Radial Channels)	-0.5305	23.275
Longitudinal Stress - (ksi) (Primary plus Secondary Stress -Inner Shell)	2.379	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Inner Shell)	-2.157	NL
Longitudinal Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	3.150	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	-3.641	NL
Longitudinal Stress - (ksi) (Primary plus Secondary Stress - Radial Channels)	-15.51	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Radial Channels)	-2.294	NL

The finite element analyses of the metal structure adjacent to the trunnion block did not include the state of stress arising from the water jacket internal pressure. These stresses are conservatively computed based on a two-dimensional strip model that neglects the lower annular plate. The water jacket bending stresses are summarized below:

<b>Tangential Bending Stress in Water Jacket Outer Panel from Water Pressure (including hydrostatic and inertia effects)</b>	<b>Calculated Value (ksi)</b>
HI-TRAC 125	14.18
HI-TRAC 100	13.63

To establish a minimum safety factor for the outer panels of the water jacket for the Level A



condition, we must add primary membrane circumferential stress from the trunnion load analysis to primary circumferential bending stress from the water jacket bending stress. Then, the safety factors may be computed by comparison to the allowable limit for primary membrane plus primary bending stress. The following results are obtained:

<b>Results for Load Case 01 in Water Jacket (Load Case 01) – Level A Load</b>			
<b>Circumferential Stress in Water Jacket Outer Enclosure</b>	<b>CALCULATED VALUE (ksi)</b>	<b>ALLOWABLE VALUE (ksi)</b>	<b>SAFETY FACTOR (allowable value/calculated value)</b>
HI-TRAC 125	14.57	26.25	1.80
HI-TRAC 100	13.94	26.25	1.88

To arrive at minimum safety factors for primary membrane plus bending stress in the outer panel of the water jacket for the Level B condition, we amplify the finite element results the trunnion load analysis, add the appropriate stress from the two-dimensional water jacket calculation, and compare the results to the increased Level B allowable. The following results are obtained:

<b>Results for Load Case 01 in Water Jacket (Load Case 01) – Level B Load</b>			
<b>Circumferential Stress in Water Jacket Outer Enclosure</b>	<b>CALCULATED VALUE (ksi)</b>	<b>ALLOWABLE VALUE (ksi)</b>	<b>SAFETY FACTOR (allowable value/calculated value)</b>
HI-TRAC 125	14.81	35.0	2.36
HI-TRAC 100	14.16	35.0	2.47

All safety factors are greater than 1.0; the Level A load condition governs.

#### 3.4.4.3.3.2 Lead Slump in HI-TRAC 125 - Horizontal Drop Event (Case 02.b in Table 3.1.5)

During a side drop of the HI-TRAC 125 transfer cask, the lead shielding must be shown not to slump and cause significant amounts of shielding to be lost in the top area of the lead annulus. Slumping of the lead is not considered credible in the HI-TRAC transfer cask because of:

- a. the shape of the interacting surfaces
- b. the ovalization of the shell walls under impact
- c. the high coefficient of friction between lead and steel
- d. The inertia force from the MPC inside the HI-TRAC will compress the inner shell at the impact location and locally “pinch” the annulus that contains the lead; this opposes the tendency for the lead to slump and open up the annulus at the impact location.

Direct contact of the outer shell of the HI-TRAC with the ISFSI pad is not credible since there is a water jacket that surrounds the outer shell. The water jacket metal shell will experience most of the direct impact. Nevertheless, to conservatively analyze the lead slump scenario, it is assumed

that there is no water jacket, the impact occurs far from either end of the HI-TRAC so as to ignore any strengthening of the structure due to end effects, the impact occurs directly on the outer shell of the HI-TRAC, and the contact force between HI-TRAC and the MPC is ignored. All of these assumptions are conservative in that their imposition magnifies any tendency for the lead to slump.

To confirm that lead slump is not credible, a finite element analysis of the lead slump problem, incorporating the conservatisms listed above, during a postulated HI-TRAC 125 horizontal drop (see Figure 3.4.22) is carried out. The HI-TRAC 125 cask body modeled consists only of an inner steel shell, an outer steel shell, and a thick lead annulus shield contained between the inner and outer shell. A unit length of HI-TRAC is modeled and the contact at the lead/steel interface is modeled as a compression-only interface. Interface frictional forces are conservatively neglected. As the HI-TRAC 125 has a greater lead thickness, analysis of the HI-TRAC 125 is considered to bound the HI-TRAC 100 and the HI-TRAC 100D. Furthermore, since there are no differences between the HI-TRAC 125 and the HI-TRAC 125D with respect to the finite element model, the results are valid for both 125-Ton transfer casks.

The analysis is performed in two parts:

First, to maximize the potential for lead/steel separation, the shells are ignored and the gap elements grounded. This has the same effect as assuming the shells to be rigid and maximizes the potential and magnitude of any separation at the lead/steel interface (and subsequent slump). This also maximizes the contact forces at the portion of the interface that continues to have compression forces developed. The lead annulus is subjected to a 45g deceleration and the deformation, stress field, and interface force solution developed. This solution establishes a conservative result for the movement of the lead relative to the metal shells.

In the second part of the analysis, the lead is removed and replaced by the conservative (high) interface forces from the first part of the analysis. These interface forces, together with the 45g deceleration-induced inertia forces from the shell self weight are used to obtain a solution for the stress and deformation field in the inner and outer metal shells.

The results of the analysis are as follows:

- a. The maximum predicted lead slump at a location 180 degrees from the impact point is 0.1". This gap decreases gradually to 0.0" after approximately 25 degrees from the vertical axis. The decrease in the diameter of the inner shell of the transfer cask (in the direction of the deceleration) is approximately 0.00054". This demonstrates that ovalization of the HI-TRAC shells does not occur. Therefore, the lead shielding deformation is confined to a local region with negligible deformation of the confining shells.
- b. The stress intensity distribution in the shells demonstrates that high stresses are concentrated, as anticipated, only near the assumed point of impact with the ISFSI pad. The value of the maximum stress intensity (51,000 psi) remains below the allowable stress intensity for primary membrane plus primary bending for a Level D event (58,700 psi). Thus, the steel shells continue to perform their function and

contain the lead. The stress distribution, obtained using the conservatively large interface forces, demonstrates that permanent deformation could occur only in a localized region near the impact point. Since the “real” problem precludes direct impact with the outer shell, the predicted local yielding is simply a result of the conservatism imposed in the model.

It is concluded that a finite element analysis of the lead slump under a 45g deceleration in a side drop clearly indicates that there is no appreciable change in configuration of the lead shielding and no overstress of the metal shell structure. Therefore, retrievability of the MPC is not compromised and the HI-TRAC transfer cask continues to provide shielding.

#### 3.4.4.3.3.3 HI-TRAC Lid Stress Analysis During HI-TRAC Drop Accident (Load Case 02.b in Table 3.1.5)

The stress in the HI-TRAC 125 transfer lid is analyzed when the lid is subject to the deceleration loads of a side drop Figure 3.4.22 is a sketch of the scenario. The analysis shows that the cask body, under a deceleration of 45g's, will not separate from the transfer lid during the postulated side drop. This event is considered a Level D event in the ASME parlance.

The bolts that act as doorstops to prevent opening of the doors are also checked for their load capacity. It is required that sufficient shear capacity exists to prevent both doors from opening and exposing the MPC.

The only difference between the HI-TRAC 100 and the HI-TRAC 125 transfer lid doors is that the HI-TRAC 100 has less lead and has no middle steel plate. A similar analysis of the HI-TRAC 100 shows that all safety factors are greater than 1.0. The table given below summarizes the results for both units:

<b>Transfer Lid Attachment Integrity Under Side Drop</b>			
<b>Item – Shear Capacity</b>	<b>Value (kip) or (ksi)</b>	<b>Capacity (kip) or (ksi)</b>	<b>Safety Factor= Capacity/Value</b>
HI-TRAC 125 Attachment (kip)	1,272.0	1,475.0	1.159
HI-TRAC 125 Door Lock Bolts (ksi)	20.24	48.3	2.387
HI-TRAC 100 Attachment (kip)	1,129.0	1,503.0	1.331
HI-TRAC 100 Door Lock Bolts (ksi)	13.81	48.3	3.497

All safety factors are greater than 1.0 and are based on actual interface loads. For the HI-TRAC 125 and the HI-TRAC 100, the interface load (primary impact at transfer lid) computed from the handling accident analysis is bounded by the values given below:

<b>BOUNDING INTERFACE LOADS COMPUTED FROM HANDLING ACCIDENT ANALYSES</b>	
<b>Item</b>	<b>Bounding Value (kip)</b>
HI-TRAC 125	1,300
HI-TRAC 100	1,150

The HI-TRAC 125D and HI-TRAC 100D transfer casks do not utilize a transfer lid. Instead, the MPC is transferred to or from a storage overpack using the HI-TRAC pool lid and a special mating device. Therefore, an analysis is performed to demonstrate that the pool lid will not separate from the cask body during the postulated side drop. The results of the analyses are summarized in the following tables for the HI-TRAC 125D and the HI-TRAC 100D.

<b>HI-TRAC 125D Pool Lid Attachment Integrity Under Side Drop</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Limit</b>	<b>Safety Factor</b>
Lateral Shear Force (kips)	562.5	1085	1.929
Maximum Bolt Tensile Stress (ksi)	2.548	116.7	45.8
Combined Tension and Shear Interaction	0.269	1.00	3.72

<b>HI-TRAC 100D Pool Lid Attachment Integrity Under Side Drop</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Limit</b>	<b>Safety Factor</b>
Lateral Shear Force (kips)	360.0	1085	3.015
Maximum Bolt Tensile Stress (ksi)	1.477	116.7	79.0
Combined Tension and Shear Interaction	0.11	1.00	9.08

#### 3.4.4.3.3.4 Stress Analysis of the HI-TRAC Water Jacket (Load Case 03 in Table 3.1.5)

The water jacket is assumed subject to internal pressure from pressurized water and gravity water head. Calculations are performed for the HI-TRAC 125, the HI-TRAC 125D, the HI-TRAC 100, and the HI-TRAC 100D to determine the water jacket stress under internal pressure plus hydrostatic load. Results are obtained for the water jacket configuration and the connecting welds for all HI-TRAC transfer casks. The table below summarizes the results of the analyses.

<b>Water Jacket Stress Evaluation</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 125 Water Jacket Enclosure Shell Panel Bending Stress	14.18	26.25	1.851
HI-TRAC 100 Water Jacket Enclosure Shell Panel Bending Stress	13.63	26.25	1.926
HI-TRAC 125 Water Jacket Bottom Flange Bending Stress	18.3	26.25	1.434
HI-TRAC 100 Water Jacket Bottom Flange Bending Stress	16.92	26.25	1.551
HI-TRAC 125 Weld Stress – Bottom Flange to Outer Shell Double Fillet Weld	14.79	21.0	1.42
HI-TRAC 125 - Radial Rib Direct Stress	2.198	17.5	7.961
HI-TRAC 100 - Radial Rib Direct Stress	1.975	17.5	8.861
HI-TRAC 125D Water Jacket Bottom Flange Bending Stress	18.88	26.25	1.39
HI-TRAC 125D Water Jacket Enclosure Shell Panel Bending Stress	10.80	26.25	2.43
HI-TRAC 125D Weld Stress – Enclosure Panel to Radial Rib Plug Welds	1.093	17.5	16.01
HI-TRAC 125D Weld Stress – Bottom Flange to Outer Shell Single Fillet Weld	3.133	21.0	6.70
HI-TRAC 100D Water Jacket Bottom Flange Bending Stress	16.69	26.25	1.57
HI-TRAC 100D Water Jacket Enclosure Shell Panel Bending Stress	12.75	26.25	2.06
HI-TRAC 100D Weld Stress – Enclosure Panel to Radial Rib Plug Welds	0.680	17.5	25.7
HI-TRAC 100D Weld Stress – Bottom Flange to Outer Shell Single Fillet Weld	2.836	21.0	7.40

#### 3.4.4.3.3.5 HI-TRAC Top Lid Separation (Load Case 02.b in Table 3.1.5)

The potential of top lid separation under a 45g deceleration side drop event requires evaluation. It is concluded by analysis that the connection provides acceptable protection against top lid separation. It is also shown that the bolts and the lid contain the MPC within the HI-TRAC cavity during and after a drop event. The results from the HI-TRAC 125 bound the corresponding results from the HI-TRAC 100 because the top lid bolts are identical in the two units and the HI-TRAC 125 top lid weighs more. The analysis also bounds the HI-TRAC 125D and the HI-TRAC 100D because the postulated side drop of the HI-TRAC 125, during which the transfer lid impacts the target surface, produces a larger interface load between the MPC and the top lid of the HI-TRAC than the nearly horizontal drop of the HI-TRAC 125D and the HI-TRAC 100D. The table below provides the results of the bounding analysis.

<b>HI-TRAC Top Lid Separation Analysis</b>			
<b>Item</b>	<b>Value</b>	<b>Capacity</b>	<b>Safety Factor= Capacity/Value</b>
Attachment Shear Force (lb.)	123,750	957,619	7.738
Tensile Force in Stud (lb.)	132,000	1,117,222	8.464
Bending Stress in Lid (ksi)	35.56	58.7	1.65
Shear Load per unit Circumferential Length in Lid (lb./in)	533.5	29,400	55.10

#### 3.4.4.4 Comparison with Allowable Stresses

Consistent with the formatting guidelines of Reg. Guide 3.61, calculated stresses and stress intensities from the finite element and other analyses are compared with the allowable stresses and stress intensities defined in Subsection 3.1.2.2 per the applicable sections of [3.4.2] and [3.4.4] for defined normal and off-normal events and [3.4.3] for accident events (Appendix F).

##### 3.4.4.4.1 MPC

In Amendment #5 to the HI-STORM CoC, the weight limits for fuel assemblies to be stored in the MPCs were increased from 1,680 lbs to 1,720 lbs per assembly for PWR fuel. In order to account for this increase in fuel weight, the results of the MPC stress analysis under lateral loading, which is described in Subsection 3.4.4.3.1.1, are uniformly scaled based on the percentage weight increase. Specifically, the results for the PWR MPCs are scaled by a factor of 1.024 (=1720/1680). This approach is acceptable because (i) the finite element analysis results are based on linear elastic material properties and (ii) the percentage increases in total weight, considering the stored fuel, fuel basket, and MPC shell, are less than the factors above. Finally, since the stresses associated with closing the support clearance gaps between the fuel basket and the MPC shell and between the MPC shell and the overpack are secondary stress components, as explained in Subsection 3.4.4.3.1.1, the use of a linear scale factor is an appropriate means of computing the primary stresses in the fuel basket and MPC shell.

Table 3.4.6 provides summary data extracted from the numerical analysis results for the fuel basket, enclosure vessel, and fuel basket supports after scaling to adjust for the increased fuel assembly weights. The results presented in Table 3.4.6 are based on the design basis deceleration and do not include any dynamic amplification due to internal elasticity of the structure (i.e., local inertia effects). Calculations suggest that a uniform conservative dynamic amplifier for the fuel basket would be 1.08 independent of the duration of impact. If we recognize that the tip-over event for HI-STORM 100 is a long duration event, then a dynamic amplifier of 1.04 is appropriate. The summary data provided in Table 3.4.3 and 3.4.4 gives the lowest safety factor computed for the fuel basket and for the MPC, respectively. Safety factors reported for the MPC shell in Table 3.4.4 are based on allowable strengths at 500 deg. F. Modification of the fuel basket safety factor for dynamic amplification leaves considerable margin. Factors of safety greater than 1 indicate that

calculated results are less than the allowable strengths.

A perusal of the results in Tables 3.4.3 and 3.4.4 under different load combinations for the fuel basket and the enclosure vessel reveals that all factors of safety are above 1.0 even if we use the most conservative value for dynamic amplification factor. The relatively modest factor of safety in the fuel basket under side drop events (Load Case F3.b and F3.c) in Table 3.4.3 warrants further explanation since a very conservative finite element model of the structure has been utilized in the analysis.

The legacy 2-D finite element model dating back to the 1990s employed to simulate the MPC and the fuel basket, and other simplifications such as modeling the overpack as a rigid body and fuel as a pressure loading in some loading simulations used in this FSAR, lead to an understatement and associated uncertainty in the computed safety margins which can be alleviated by a 3-D analysis (under the §72.48 process). The use of a 3-D analysis has been utilized on the ANSYS and LS-DYNA platform (as appropriate) and approved by the NRC in the HI-STORM FW docket.

It should be noted that the change of the BWR fuel weight in Table 2.1.22 applies only to the specified DFC bearing locations but does not require the design basis gross weight of the MPC used in the structural analyses to be changed. Therefore, other safety analyses such as non-mechanistic tip-over and lifting and handling appurtenances remain unaffected and do not need to be revisited.

The wall thickness of the storage cells, which is by far the most significant variable in a fuel basket's structural strength, is significantly greater in the MPCs than in comparable fuel baskets licensed in the past. In spite of their relatively high flexural rigidities, computed margins in the fuel baskets are rather modest. This is because of some assumptions in the analysis that lead to an overstatement of the state of stress in the fuel basket. For example:

- i. The section properties of longitudinal fillet welds that attach contiguous cell walls to each other are completely neglected in the finite element model (Figure 3.4.7). The fillet welds strengthen the cell wall section modulus at the very locations where maximum stresses develop.
- ii. The radial gaps at the fuel basket-MPC shell and at the MPC shell-storage overpack interface are explicitly modeled. As the applied loading is incrementally increased, the MPC shell and fuel basket deform until a "rigid" backing surface of the storage overpack is contacted, making further unlimited deformation under lateral loading impossible. Therefore, some portion of the fuel basket and enclosure vessel (EV) stress has the characteristics of secondary stresses (which by definition, are self-limited by deformation in the structure to achieve compatibility). For conservativeness in the incremental analysis, we make no distinction between deformation controlled (secondary) stress and load controlled (primary) stress in the stress categorization of the MPC-24, 32, and 68 fuel baskets. We treat all stresses, regardless of their origin, as primary stresses. Such a conservative interpretation of the Code has a direct (adverse) effect on the computed safety factors. As noted earlier, the results for the MPC-24E are properly based only on

primary stresses to illustrate the conservatism in the reporting of results for the MPC-24, 32, and 68 baskets.

- iii. A uniform pressure simulates the SNF inertia loading on the cell panels, which is a most conservative approach for incorporating the SNF/cell wall structure interaction.

The above assumptions act to depress the computed values of factors of safety in the fuel basket finite element analysis and render conservative results.

The reported factors of safety do not include the effect of dynamic load amplifiers. The duration of impact and the predominant natural frequency of the basket panels under drop events result in the dynamic load factors that do not exceed 1.08. Therefore, since all reported factors of safety for all fuel basket types are greater than the DLF, the MPC is structurally adequate for its intended functions.

Tables 3.4.7 and 3.4.8 report stress intensities and safety factors for the confinement boundary subject to internal pressure alone and internal pressure plus the normal operating condition temperature with the most severe thermal gradient. The final values for safety factors in the various locations of the confinement boundary provide assurance that the MPC enclosure vessel is a robust pressure vessel.

#### 3.4.4.4.2 Storage Overpack and HI-TRAC

The result from analyses of the storage overpack and the HI-TRAC transfer cask is shown in Table 3.4.5. The location of each result is indicated in the table. Safety factors for lifting operations where three times the lifted load is applied are reported in Section 3.4.3.

The table shows that all allowable stresses are much greater than their associated calculated stresses and that safety factors are above the limit of 1.0.

#### 3.4.4.5 Elastic Stability Considerations

##### 3.4.4.5.1 MPC Elastic Stability

Stability calculations for the MPC have been carried out in the HI-STAR 100 FSAR, Docket Number 72-1008. Using identical methodology with input loads and decelerations appropriate to the HI-STORM, safety factors > 1.0 are obtained for all relevant load cases. Note that for HI-STORM, the design external pressure differential is reduced to 0.0 psi, and the peak deceleration under accident events is reduced from 60g's (HI-STAR) to 45g's.

##### 3.4.4.5.2 HI-STORM 100 Storage Overpack Elastic Stability

HI-STORM 100 (and 100S and the 100S Version B) storage overpack shell buckling is not a credible scenario since the two steel shells plus the entire radial shielding act to resist vertical compressive loading. Subsection 3.4.4.3.2.3 develops values for compressive stress in the steel



shells of the storage overpack. Because of the low value for compressive stress coupled with the fact that the concrete shielding backs the steel shells, we can conclude that instability is unlikely. Note that the entire weight of the storage overpack can also be supported by the concrete shielding acting in compression. Therefore, in the unlikely event that a stability limit in the steel was approached, the load would simply shift to the massive concrete shielding. Notwithstanding the above comments, stability analyses of the storage overpack have been performed for bounding cases of longitudinal compressive stress with nominal circumferential compressive stress and for bounding circumferential compressive stress with nominal axial compressive stress. This latter case is for a bounding all-around external pressure on the HI-STORM 100 outer shell. The latter case is listed as Load Case 05 in Table 3.1.5 and is performed to demonstrate that explosions or other environmental events that could lead to an all-around external pressure on the outer shell do not cause a buckling instability. ASME Code Case N-284, a methodology accepted by the NRC, has been used for this analysis. The storage overpack shells for the HI-STORM 100 are examined individually assuming that the four radial plates provide circumferential support against a buckling deformation mode. The analysis of the storage overpack outer shell for a bounding external pressure of

$$p_{\text{ext}} = 30 \text{ psi}$$

together with a nominal compressive axial load that bounds the dead weight load at the base of the outer shell, gives a safety factor against an instability of:

$$\text{Safety Factor} = (1/0.466) \times 1.34 = 2.88$$

The factor 1.34 is included in the above result since the analysis methodology of Code Case N-284 builds in this factor for a stability analysis for an accident condition. The suite of stability analyses have also been performed for the HI-STORM 100S Version B. No credit is taken for any support provided by the concrete shielding and the effect of support by radial ribs is conservatively neglected (since the ribs in the HI-STORM 100S Version B do not extend the full height of the overpack). It is shown that the safety factor computed for the classic HI-STORM 100 is a lower bound for all of the HI-STORM 100S versions.

The external pressure for the overpack stability considered here significantly bounds the short-time 10-psi differential pressure (between outer shell and internal annulus) specified in Table 2.2.1.

The same postulated external pressure condition can also act on the HI-TRAC during movement from the plant to the ISFSI pad. In this case, the lead shielding acts as a backing for the outer shell of the HI-TRAC transfer cask just as the concrete does for the storage overpack. The water jacket metal structure provides considerable additional structural support to the extent that it is reasonable to state that instability under external pressure is not credible. If it is assumed that the all-around water jacket support is equivalent to the four locations of radial support provided in the storage overpack, then it is clear that the instability result for the storage overpack bounds the results for the HI-TRAC transfer cask. This occurs because the R/t ratio (mean radius-to-wall thickness) of the HI-TRAC outer shell is less than the corresponding ratio for the HI-STORM storage overpack. Therefore, no HI-TRAC analysis is performed.

### 3.4.5 Cold

A discussion of the resistance to failure due to brittle fracture is provided in Subsection 3.1.2.3.

The value of the ambient temperature has two principal effects on the HI-STORM 100 System, namely:

- i. The steady-state temperature of all material points in the cask system will go up or down by the amount of change in the ambient temperature.
- ii. As the ambient temperature drops, the absolute temperature of the contained helium will drop accordingly, producing a proportional reduction in the internal pressure in accordance with the Ideal Gas Law.

In other words, the temperature gradients in the system under steady-state conditions will remain the same regardless of the value of the ambient temperature. The internal pressure, on the other hand, will decline with the lowering of the ambient temperature. Since the stresses under normal storage condition arise principally from pressure and thermal gradients, it follows that the stress field in the MPC under -40 degree F ambient would be smaller than the "heat" condition of storage, treated in the preceding subsection. Additionally, the allowable stress limits tend to increase as the component temperatures decrease.

Therefore, the stress margins computed in Section 3.4.4 can be conservatively assumed to apply to the "cold" condition as well.

Finally, it can be readily shown that the HI-STORM 100 System is engineered to withstand "cold" temperatures (-40 degrees F), as set forth in the Technical Specification, without impairment of its storage function.

Unlike the MPC, the HI-STORM 100 storage overpack is an open structure; it contains no pressure. Its stress field is unaffected by the ambient temperature, unless low temperatures produce brittle fracture due to the small stresses which develop from self-weight of the structure and from the minute difference in the thermal expansion coefficients in the constituent parts of the equipment (steel and concrete). To prevent brittle fracture, all steel material in HI-STORM 100 is qualified by impact testing as set forth in the ASME Code (Table 3.1.18).

The structural material used in the MPC (Alloy X) is recognized to be completely immune from brittle fracture in the ASME Codes.

As no liquids are included in the HI-STORM 100 storage overpack design, loads due to expansion of freezing liquids are not considered. The HI-TRAC transfer cask utilizes demineralized water in the water jacket. However, the specified lowest service temperature for the HI-TRAC is 0 degrees F and a 25% ethylene glycol solution is required for the temperatures from 0 degrees F to 32 degrees F. Therefore, loads due to expansion of freezing liquids are not considered.

There is one condition, however, that does require examination to insure ready retrievability of the fuel. Under a postulated loading of an MPC from a HI-TRAC transfer cask into a cold HI-STORM 100 storage overpack, it must be demonstrated that sufficient clearances are available to preclude interference when the “hot” MPC is inserted into a “cold” storage overpack. To this end, a bounding analysis for free thermal expansions has been performed in Subsection 4.4.6, wherein the MPC shell is postulated at its maximum design basis temperature and the thermal expansion of the overpack is ignored. The results from the evaluation of free thermal expansion are summarized in Table 4.4.10. The final radial clearance (greater than 0.25” radial) is sufficient to preclude jamming of the MPC upon insertion into a cold HI-STORM 100 storage overpack.

### 3.4.6 HI-STORM 100 Kinematic Stability under Flood Condition (Load Case A in Table 3.1.1)

The flood condition subjects the HI-STORM 100 System to external pressure, together with a horizontal load due to water velocity. Because the HI-STORM 100 storage overpack is equipped with ventilation openings, the hydrostatic pressure from flood submergence acts only on the MPC. As stated in subsection 3.1.2.1.1.3, the design external pressure for the MPC bounds the hydrostatic pressure from flood submergence. Subsection 3.4.4.5.2 has reported a positive safety factor against instability from external pressure in excess of that expected from a complete submergence in a flood. The analysis performed below is also valid for the HI-STORM 100S and the HI-STORM 100S, Version B.

The water velocity associated with flood produces a horizontal drag force, which may act to cause sliding or tip-over. In accordance with the provisions of ANSI/ANS 57.9, the acceptable upper bound flood velocity,  $V$ , must provide a minimum factor of safety of 1.1 against overturning and sliding. For HI-STORM 100, we set the upper bound flood velocity design basis at 15 feet/sec. Subsequent calculations conservatively assume that the flow velocity is uniform over the height of the storage overpack.

The overturning horizontal force,  $F$ , due to hydraulic drag, is given by the classical formula:

$$F = C_d A V^*$$

where:

$V^*$  is the velocity head =  $\frac{\rho V^2}{2g}$ ; ( $\rho$  is water weight density, and  $g$  is acceleration due to gravity).

A: projected area of the HI-STORM 100 cylinder perpendicular to the fluid velocity vector.

$C_d$ : drag coefficient

The value of  $C_d$  for flow past a cylinder at Reynolds number above  $5E+05$  is given as 0.5 in the literature (viz. Hoerner, Fluid Dynamics, 1965).

The drag force tending to cause HI-STORM 100's sliding is opposed by the friction force, which is given by

$$F_f = \mu K W$$

where:

$\mu$  = limiting value of the friction coefficient at the HI-STORM 100/ISFSI pad interface (conservatively taken as 0.25, although literature citations give higher values).

$K$  = buoyancy coefficient (documented in HI-981928, Structural Calculation Package for HI-STORM 100 (see citation in Subsection 3.6.4).

$W$ : Minimum weight of HI-STORM 100 with an empty MPC.

### Sliding Factor of Safety

The factor of safety against sliding,  $\beta_1$ , is given by

$$\beta_1 = \frac{F_f}{F} = \frac{\mu K W}{C_d A V^*}$$

It is apparent from the above equation,  $\beta_1$ , will be minimized if the empty weight of HI-STORM 100 is used in the above equation.

As stated previously,  $\mu = 0.25$ ,  $C_d = 0.5$ .

$V^*$  corresponding to 15 ft./sec. water velocity is 218.01 lb per sq. ft.

$A$  = length x diameter of HI-STORM 100 = 132.5" x 231.25"/144 sq. in./sq.ft. = 212.78 sq. ft.

$K$  = buoyancy factor = 0.64 (per calculations in HI-981928)

$W$  = empty weight of overpack w/ lid = 270,000 lbs. (Table 3.2.1)

Substituting in the above formula for  $\beta_1$ , we have

$$\beta_1 = 1.86 > 1.1 \text{ (required)}$$

Since the weight of the HI-STORM 100S or HI-STORM 100S, Version B, plus the weight of an empty MPC-32 (i.e., the lightest MPC) is greater than 270,000 lb, the above calculation is also valid for these two units for the entire range of concrete densities.

### Overtipping Factor of Safety

For determining the margin of safety against overturning  $\beta_2$ , the cask is assumed to pivot about a

fixed point located at the outer edge of the contact circle at the interface between HI-STORM 100 and the ISFSI. The overturning moment due to a force  $F_T$  applied at height  $H^*$  is balanced by a restoring moment from the reaction to the cask buoyant force  $KW$  acting at radius  $D/2$ .

$$F_T H^* = KW \frac{D}{2}$$

$$F_T = \frac{K W D}{2 H^*}$$

$W$  is the empty weight of the storage overpack.

We have,

$$W = 270,000 \text{ lb. (Table 3.2.1)}$$

$$H^* = 119.2" \text{ (maximum height of mass center per Table 3.2.3)}$$

$$D = 132.5" \text{ (Holtec Drawing 1495)}$$

$$K = 0.64 \text{ (calculated in HI-981928)}$$

$$F_T = 96,040 \text{ lb.}$$

$F_T$  is the horizontal drag force at incipient tip-over.

$$F = C_d A V^* = 23,194 \text{ lbs. (drag force at 15 feet/sec)}$$

The safety factor against overturning,  $\beta_2$ , is given as:

$$\beta_2 = \frac{F_T}{F} = 4.14 > 1.1 \text{ (required)}$$

This result bounds the result for the HI-STORM 100S, for the HI-STORM 100S Version B, as well as for the densified concrete shielding option, since the calculation uses a conservative lower bound weight and a bounding height for the center of gravity.

In the next subsection, results are presented to show that the load  $F$  (equivalent to an inertial deceleration of  $F/360,000 \text{ lb} = 0.0644 \text{ g}'s$  applied to the loaded storage overpack) does not lead to large global circumferential stress or ovalization of the storage overpack that could prevent ready retrievability of the MPC. It is shown in Subsection 3.4.7 that a horizontal load equivalent to  $0.47g's$  does not lead to circumferential stress levels and ovalization of the HI-STORM storage overpack to prevent ready retrievability of the MPC. The load used for that calculation clearly bounds the side load induced by flood.

### 3.4.7 Seismic Event and Explosion - HI-STORM 100

#### 3.4.7.1 Seismic Event (Load Case C in Table 3.1.1)

##### Overtuning Analysis

The HI-STORM 100 System plus its contents may be assumed to be subject to a seismic event consisting of three orthogonal statistically independent acceleration time-histories. For the purpose of performing a conservative analysis to determine the maximum ZPA that will not cause incipient tipping, the HI-STORM 100 System is considered as a rigid body subject to a net horizontal quasi-static inertia force and a vertical quasi-static inertia force. This is consistent with the approach used in previously licensed dockets. The vertical seismic load is conservatively assumed to act in the most unfavorable direction (upwards) at the same instant. The vertical seismic load is assumed to be equal to or less than the net horizontal load with  $\epsilon$  being the ratio of vertical component to one of the horizontal components. For use in calculations, define  $D_{BASE}$  as the contact patch diameter, and  $H_{CG}$  as the height of the centroid of an empty HI-STORM 100 System (no fuel). Conservatively, assume

$$D_{BASE} = 132.5" \text{ (Drawing 1495, Sheet 1 specifies } 133.875'' \text{ including overhang for welding)}$$

Tables 3.2.1 and 3.2.3 give HI-STORM 100 weight data and center-of-gravity heights.

The weights and center-of-gravity heights are reproduced here for calculation of the composite center-of-gravity height of the storage overpack together with an empty MPC.

<u>Weight (pounds)</u>	<u>C.G. Height (Inches); H</u>
Overpack - $W_o = 270,000$	116.8
MPC-24 - $W_{24} = 42,000$	$109.0 + 24 = 133.0^\dagger$
MPC-68 - $W_{68} = 39,000$	$111.5 + 24 = 135.5$
MPC-32 - $W_{32} = 36,000$	$113.2 + 24 = 137.2$
MPC-24E - $W_{24E} = 45,000$	$108.9 + 24 = 132.9$

The height of the composite centroid,  $H_{CG}$ , is determined from the equation

$$H_{cg} = \frac{W_o \times 116.8 + W_{MPC} \times H}{W_o + W_{MPC}}$$

Performing the calculations for all of the MPCs gives the following results:

$H_{cg}$  (inches)

<sup>†</sup> From Table 3.2.3, it is noted that MPC C.G. heights are measured from the base of the MPC. Therefore, the thickness of the overpack baseplate and the concrete MPC pedestal must be added to determine the height above ground.

MPC-24 with storage overpack	118.98
MPC-68 with storage overpack	119.16
MPC-32 with storage overpack	119.20
MPC-24E with storage overpack	119.10

A conservative overturning stability limit is achieved by using the largest value of  $H_{CG}$  (call it  $H$ ) from the above. Because the HI-STORM 100 System is a radially symmetric structure, the two horizontal seismic accelerations can be combined vectorially and applied as an overturning force at the C.G. of the cask. The net overturning static moment is

$$WG_H H$$

where  $W$  is the total system weight and  $G_H$  is the resultant zero period acceleration seismic loading (vectorial sum of two orthogonal seismic loads) so that  $WG_H H$  is the inertia load due to the resultant horizontal acceleration. The overturning moment is balanced by a vertical reaction force, acting at the outermost contact patch radial location  $r = D_{BASE}/2$ . The resistive moment is minimized when the vertical zero period acceleration  $G_V$  tends to reduce the apparent weight of the cask. At that instant, the moment that resists "incipient tipping" is:

$$W (1 - G_V) r$$

Performing a static moment balance and eliminating  $W$  results in the following inequality to ensure a "no-overturning condition":

$$G_H + \frac{r}{H} G_V \leq \frac{r}{H}$$

Using the values of  $r$  and  $H$  for the HI-STORM 100 ( $r = 66.25"$ ,  $H = 119.20"$ ), representative combinations of  $G_H$  and  $G_V$  that satisfy the limiting equality relation are computed and tabulated below:

Acceptable Net Horizontal G-Level (HI-STORM100), $G_H$	Acceptable Vertical G-Level, $G_V$
0.467	0.16
0.445	0.20
0.417	0.25
0.357	0.357

We repeat the above computations using the weight and c.g. location of the HI-STORM 100S(232). Because of the lowered center of gravity positions, the maximum net horizontal "G" levels are slightly increased.

Performing the calculations for all of the MPCs gives the following results:

<u>MPC</u>	<u>H<sub>cg</sub> (inches)</u>
MPC-24 with storage overpack	113.89
MPC-68 with storage overpack	114.07
MPC-32 with storage overpack	114.11
MPC-24E with storage overpack	114.01

Using the values of  $r$  and  $H$  for the HI-STORM 100S(232) ( $r = 66.25"$ ,  $H = 114.11"$ ), representative combinations of  $G_H$  and  $G_V$  that satisfy the limiting equality relation are computed and tabulated below:

<b>Acceptable Net Horizontal G-Level (HI-STORM 100S(232)), <math>G_H</math></b>	<b>Acceptable Vertical G-Level, <math>G_V</math></b>
0.488	0.16
0.464	0.20
0.435	0.25
0.367	0.367

The limiting values of  $G_H$  and  $G_V$  for the HI-STORM 100S(243), which is taller than the HI-STORM 100S(232), are the same as the HI-STORM 100.

If the HI-STORM 100 or the HI-STORM 100S is fabricated using high density concrete (i.e., above 160.8 pcf dry), the C.G. height of the overpack decreases and thereby enables the cask system to withstand higher g-loads. This conclusion becomes immediately clear when the maximum acceptable vertical g-level is expressed in the following form:

$$G_V = 1 - \frac{H}{r} G_H$$

For fixed values of  $G_H$  and  $r$ , the value of  $G_V$  increases as  $H$  decreases. Therefore, the representative combinations of  $G_H$  and  $G_V$  given above for the HI-STORM 100 and the HI-STORM 100S are conservative for the densified concrete shielding option.

Since the HI-STORM 100S, Version B has further reduced the centroid of the loaded units, it is expected that acceptable G-Levels are further increased. The following calculations provide the limiting G-level combinations for the HI-STORM 100S Version B with standard weight concrete. As noted previously, the result for standard weight concrete will bound the corresponding result for the high density concrete (densified) shielding option.

We repeat the above computations using the weight and c.g. location of the HI-STORM 100S(218). Because of the lowered center of gravity positions, the maximum net horizontal "G" levels are slightly increased.



Performing the calculations for all of the MPCs gives the following results:

$H_{cg}$  (inches)

MPC-24 with storage overpack	109.88
MPC-68 with storage overpack	110.12
MPC-32 with storage overpack	110.23
MPC-24E with storage overpack	109.93

Using the values of  $r$  and  $H$  for the HI-STORM 100S, Version B(218) ( $r = 66.25"$ ,  $H = 110.23"$ ), representative combinations of  $G_H$  and  $G_V$  that satisfy the limiting equality relation are computed and tabulated below:

Acceptable Net Horizontal G-Level (HI-STORM 100S, Version B(218)), $G_H$	Acceptable Vertical G-Level, $G_V$
0.505	0.16
0.481	0.20
0.451	0.25
0.376	0.375

The limiting values of  $G_H$  and  $G_V$  for the HI-STORM 100S, Version B(229), which is taller than the HI-STORM 100S, Version B(218), are bounded by the values listed for the HI-STORM 100.

Primary Stresses in the HI-STORM 100 Structure Under Net Lateral Load Over 180 degrees of the Periphery

Under a lateral loading, the storage overpack will experience axial primary membrane stress in the inner and outer shells as it resists bending as a “beam-like” structure. Under the same kind of lateral loading over one-half of the periphery of the cylinder, the shells will tend to ovalize under the loading and develop circumferential stress. Calculations for stresses in both the axial and circumferential direction are required to demonstrate satisfaction of the Level D structural integrity requirements and to provide confidence that the MPC will be readily removable after a seismic event, if necessary. An assessment of the stress state in the structure under the seismic induced load will be shown to bound the results for any other condition that induces a peripheral load around part of the HI-STORM 100 storage overpack perimeter. The specific analyses are performed using the geometry and loading for the HI-STORM 100; the results obtained for stress levels and the safety assessment are also applicable to an assessment of the HI-STORM 100S.

A simplified calculation to assess the flexural bending stress in the HI-STORM 100 structure under the limiting seismic event (at which tipping is incipient) is presented in the following:

A representative net horizontal acceleration of 0.47g is used to determine the primary stresses in the HI-STORM 100 storage overpack. The corresponding lateral seismic load,  $F$ , is given by

$$F = 0.47 W$$

This load will be maximized if the upper bound HI-STORM 100 weight ( $W = 410,000$  lbs. (Table 3.2.1)) is used. Accordingly,

$$F = (0.47) (410,000) = 192,700 \text{ lbs.}$$

No dynamic amplification is assumed as the overpack, considered as a beam, has a natural frequency well into the rigid range.

The moment,  $M$ , at the base of the HI-STORM 100 due to this lateral force is given by

$$M = \frac{F H}{2}$$

where  $H$  = height of HI-STORM 100 (taken conservatively as 235 inches). Note that the loading has now been approximated as a uniform load acting over the full height of the cask.

The flexural stress,  $\sigma$ , is given by the ratio of the moment  $M$  to the section modulus of the steel shell structure,  $z$ , which is computed to be  $12,640 \text{ in}^3$  for the HI-STORM 100 overpack with inner and outer shell thicknesses of  $1\text{-}1/4\text{'}$  and  $3/4\text{'}$ , respectively. The use of this value is conservative since the steel section modulus associated with the optional  $1\text{'}$  thick inner and outer shell design is slightly higher.

Therefore,

$$\sigma = \frac{(192,700) (235)}{(12,640) (2)} = 1,791 \text{ psi}$$

We note that the strength of concrete has been neglected in the above calculation.

The maximum axial stress in the storage overpack shell will occur on the "compressive" side where the flexural bending stress algebraically sums with the direct compression stress  $\sigma_d$  from vertical compression.

From the representative acceleration tables, the vertical seismic accelerations corresponding to the net  $0.47g$  horizontal acceleration is below  $0.25g$ .

Therefore, using the maximum storage overpack weight (bounded by  $410,000$  lbs. from data in Table 3.2.1)

$$\sigma_d = \frac{(410,000) (1.25)}{554.47} = 924 \text{ psi}$$

where  $554.47$  sq. inch is the metal area (cross section) of the steel structure in the HI-STORM 100

storage overpack as computed in Subsection 3.4.4.3.2.1. The total axial stress, therefore, is

$$\sigma_T = 1,791 + 924 = 2,715 \text{ psi}$$

Per Table 3.1.12, the allowable membrane stress intensity for a Level D event is 39,750 psi at 350 degrees F.

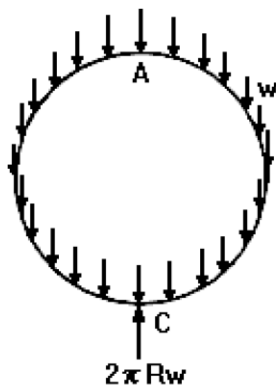
The Factor of Safety,  $\beta$ , is, therefore

$$\beta = \frac{39,750}{2,715} = 14.6$$

Examination of the stability calculations for the overpack outer shell under a 45-g vertical end drop demonstrates that no instability will result from this compressive load induced by a seismic or other environmental load leading to bending of the storage overpack as a beam.

The previous calculation has focused on the axial stress in the members developed assuming that the storage overpack does not overturn but resists the lateral load by remaining in contact with the ground and bending like a beam. Since the lateral loading is only over a portion of the periphery, there is also the potential for this load to develop circumferential stress in the inner and outer shells to resist ovalization of the shells. To demonstrate continued retrievability of the MPC after a seismic event, it must be shown that either the stresses remain in the elastic range or that any permanent deformation that develops due to plasticity does not intrude into the MPC envelope after the event is ended. In the following subsection, classical formulas for the deformation of rings under specified surface loadings are used to provide a conservative solution for the circumferential stresses in the HI-STORM 100. Specifically, the solution for a point-supported ring subject to a gravitational induced load, as depicted in the sketch below, is implemented. This solution provides a conservative estimate of the circumferential stress and the deformation of the ring that will develop under the actual applied seismic load.

Ring supported at base and loaded by its own weight,  $w$ , given per unit circumferential length.



The solution considers the geometry and load appropriate to a unit length of the inner and outer shells of the HI-STORM 100 storage overpack with a total weight equal to the overpack bounding weight (no MPC) subject to a 45g deceleration inertial loading. The numerical results for the 45g tipover event can be directly applied here by multiplying by the factor “X”, where “X” reflects the differences in the deceleration and the weights used for the tipover event and for the seismic load case here in this subsection.

$$X = (0.47g/45g) \times (410,000\text{lb.}/270,000\text{lb.}) = 0.0159$$

Using this factor on the tipover solution gives the following bounding results for maximum stresses (without regard for sign and location of the stress) and deformations:

$$\text{Maximum circumferential stress due to bending moment} = (29,310 \text{ psi} \times X) = 466 \text{ psi}$$

$$\text{Maximum circumferential stress due to mean tangential force} = (18,900 \text{ lb./2 sq.inch}) \times X = 150.3 \text{ psi}$$

$$\text{Change in diameter in the direction of the load} = -0.11'' \times X = -0.0017''$$

$$\text{Change in diameter perpendicular to the direction of the load} = +0.06'' \times X = 0.0010''$$

From the above results, it is clear that no permanent ovalization of the storage overpack occurs during the seismic event and that circumferential stresses will remain elastic and are bounded by the stresses computed based on considering the storage overpack as a simple beam. Therefore, the safety factors based on maximum values of axial stress are appropriate. The magnitudes of the diameter changes that are suggested by the ring solution clearly demonstrate that ready retrievability of the MPC is maintained after the seismic event.

Because of the low values for the calculated axial stress, the conclusions of the previous section are also valid for the HI-STORM 100S, and for the HI-STORM 100S, Version B.

### Potential for Concrete Cracking

It can be readily shown that the concrete shielding material contained within the HI-STORM 100 structure will not crack due to the flexuring action of HI-STORM 100 during a bounding seismic event that leads to a maximum axial stress in the storage overpack. For this purpose, the maximum axial strain in the steel shell is computed by dividing the tensile stress developed by the seismic G forces (for the HI-STORM 100, for example) by the Young's Modulus of steel.

$$\zeta = \frac{1,791 - 858}{28 \text{ E} + 06} = 33.3 \text{ E} - 06$$

where the Young's Modulus of steel is taken from Table 3.3.2 at 350 degrees F.

The acceptable concrete strain in tension is estimated from information in ACI-318.1 for plain concrete. The ratio of allowable tensile stress to concrete Young' Modulus is computed as

$$\text{Allowable Concrete Strain} = (5 \times (0.75) \times (f)^{1/2}) / (57,000(f)^{1/2}) = 65.8E-06$$

In the above expression,  $f$  is the concrete compressive strength.

Therefore, we conclude that considerable margins against tensile cracking of concrete under the bounding seismic event exist.

### Sliding Analysis

An assessment of sliding of the HI-STORM 100 System on the ISFSI pad during a postulated seismic event is performed using a one-dimensional "slider block on friction supported surface" dynamic model. The results for the shorter HI-STORM 100S are comparable. The HI-STORM 100 is simulated as a rigid block of mass ' $m$ ' placed on a surface, which is subject to a sinusoidal acceleration of amplitude ' $a$ '. The coefficient of friction of the block is assumed to be reduced by a factor  $\alpha$  to recognize the contribution of vertical acceleration in the most adverse manner (vertical acceleration acts to reduce the downward force on the friction interface). The equation of motion for such a "slider block" is given by:

$$m\ddot{x} = R + m a \sin \omega t$$

where:

- $\ddot{x}$ : relative acceleration of the slider block (double dot denotes second derivative of displacement ' $x$ ' in time)
- $a$ : amplitude of the sinusoidal acceleration input
- $\omega$ : frequency of the seismic input motion (radians/sec)
- $t$ : time coordinate

$R$  is the resistive Coulomb friction force that can reach a maximum value of  $\mu(mg)$  ( $\mu$  = coefficient of friction) and which always acts in the direction of opposite to  $\dot{x}(t)$ .

Solution of the above equation can be obtained by standard numerical integration for specified values of  $m$ ,  $a$ ,  $\alpha$  and  $\mu$ . The calculation is performed for representative horizontal and vertical accelerations of 0.47g and 0.16g, respectively. The input values are summarized below.

$$a = 0.47g$$

$$\alpha = 0.84 = 1 - \text{vertical acceleration} (= 0.16g)$$

$$m = 360,000 \text{ lbs/g}$$

$$\mu = 0.25$$

For establishing the appropriate value of  $\omega$ , reference is made to the USAEC publication TID-7024, "Nuclear Reactor and Earthquakes", page 35, 1963, which states that the significant energy of all seismic events in the U.S. essentially lies in the range of 0.4 to 10 Hz. Taking the mid-point value

$$\omega = (6.28) (0.5) (0.4+10) = 32.7 \text{ rad/sec.}$$

The numerical solution of the above equation yields the maximum excursion of the slider block  $x_{\max}$  as 0.12 inches, which is negligible compared to the spacing between casks.

Calculations performed at lower values of  $\omega$  show an increase in  $x_{\max}$  with reducing  $\omega$ . At 1 Hz, for example,  $x_{\max} = 3.2$  inches. It is apparent from the above that there is a large margin of safety against inter-module collision within the HI-STORM 100 arrays at an ISFSI, where the minimum installed spacing is over 2 feet (Table 1.4.1).

The above dynamic analysis indicates that the HI-STORM 100 System undergoes minimal lateral vibration under a seismic input with net horizontal ZPA g-values as high as 0.47 even under a bounding (from below) low interface surface friction coefficient of 0.25. Data reported in the literature (ACI-349R (97), Commentary on Appendix B) indicates that values of the coefficient of friction,  $\mu$ , as high as 0.7 are obtained at steel/concrete interfaces.

To ensure against unreasonably low coefficients of friction, the ISFSI pad design may require a "broom finish" at the user's discretion. The bottom surface of the HI-STORM 100 is manufactured from plate stock (i.e. non-machine finish). A coefficient of friction value of 0.53 is considered to be a conservative numerical value for the purpose of ascertaining the potential for incipient sliding of the HI-STORM 100 System. If a higher value is used, the coefficient of friction is required to be verified by test (see Table 2.2.9).

The relationship between the vertical ZPA,  $G_v$ , (conservatively assumed to act opposite to the normal gravitational acceleration), and the resultant horizontal ZPA  $G_H$  to insure against incipient sliding is given from static equilibrium considerations as:

$$G_H + \mu G_v \leq \mu$$

Using a conservative value of  $\mu$  equal to 0.53, the above relationship provides governing ZPA limits for a HI-STORM 100 (or 100S) System arrayed in a freestanding configuration. The table below gives representative combinations that meet the above limit.

$G_H$ (in g's)	$G_v$ (in g's)
0.445	0.16
0.424	0.20
0.397	0.25
0.350	0.34

Since the sliding inequality is independent of the weight and centroid of the cask system, the results above remain valid for HI-STORM overpacks with high density concrete and with different

heights.

If the values for the DBE event at an ISFSI site satisfy the above inequality relationship for incipient sliding with coefficient of friction equal to 0.53, then the non-sliding criterion set forth in NUREG-1536 is assumed to be satisfied a priori. However, if the ZPA values violate the inequality by a small amount, then it is permissible to satisfy the non-sliding criterion by implementing measures to roughen the HI-STORM 100/ISFSI pad interface to elevate the value of  $\mu$  to be used in the inequality relation. To demonstrate that the value of  $\mu$  for the ISFSI pad meets the required value implied by the above inequality, a series of Coulomb friction tests (under the QA program described in Chapter 13) shall be performed as follows:

Pour a concrete block with horizontal dimensions no less than 2' x 2' and a block thickness no less than 0.5'. Finish the top surface of the block in the same manner as the ISFSI pad surface will be prepared.

Prepare a 6" x 6" x 2" SA516 Grade 70 plate specimen (approximate weight = 20.25 lb.) to simulate the bottom plate of the HI-STORM 100 overpack. Using a calibrated friction gage attached to the steel plate, perform a minimum of twenty (20) pull tests to measure the static coefficient of friction at the interface between the concrete block and the steel plate. The pull tests shall be performed on at least ten (10) different locations on the block using varying orientations for the pull direction.

The coefficient of friction to be used in the above sliding inequality relationship will be set as the average of the results from the twenty tests.

The satisfaction of the "no-sliding" criterion set down in the foregoing shall be carried out along with the "no-overturning" qualification (using the static moment balance method in the manner described at the beginning of this subsection) and documented as part of the ISFSI facility's 10CFR72.212 evaluation.

#### Alternative Evaluation of Overturning and Sliding

In this subsection, an evaluation of the propensity for the free standing cask to be in a state of either incipient overturning or incipient sliding has been performed using a simple static analysis that is independent of time phasing of the input acceleration time histories and considers only the Zero Period Acceleration (ZPA) obtained from the response spectra. For both incipient overturning and incipient sliding, the following inequality must be satisfied to ensure satisfaction of the static criteria.

$$G_H + \mu G_V \leq \mu$$

For the incipient overturning evaluation,  $\mu$ =(radius of cask base/height to loaded cask center-of-gravity). For the incipient sliding evaluation,  $\mu$ = Coulomb coefficient of friction =0.53 at the cask/ISFSI pad interface (unless testing justifies use of a higher value). The inequality has been derived assuming that the cask is resting on a flat and level surface that is subject to a seismic event characterized by a response spectra set with the net horizontal and vertical Zero Period

Acceleration (ZPA) denoted by  $G_H$  and  $G_V$ , respectively.

This “screening” evaluation provides a conservative criterion to insure that top-of-pad acceleration time histories from the aggregate effect of soil structure interaction and free field acceleration would not predict initiation of overturning or sliding. If on-the-pad acceleration time histories are available, the applicable inequality (for overturning and sliding) may be satisfied at each time instant during the Design Basis Earthquake with  $G_H$  and  $G_V$  representing coincident values of the magnitude of the net horizontal and vertical acceleration vectors. One way of satisfying the use of the top-of-pad acceleration time history approach is to follow the guidelines described in the NRC supplemental technical summary resulting from the public meeting on ISFSI Pad Structural Analysis, dated February 27, 2013 [3.4.15].

### 3.4.7.2 Explosion (Load Case 05 in Table 3.1.5)

In the preceding subsection, it has been demonstrated that incipient tipping of the storage overpack will not occur under a side load equal to 0.47 times the weight of the cask. For a fully loaded cask with high density concrete, this side load is equal to

$$F = 192,700 \text{ lb.}$$

If it is assumed that this side load is uniformly distributed over the height of the cask and that the cask centroid is approximately at the half-height of the overpack, then an equivalent pressure,  $P$ , acting over 180 degrees of storage overpack periphery, can be defined as follows:

$$P \times (DH) = F$$

Where  $D$  = overpack outside diameter, and  $H$  = minimum height of a storage overpack (HI-STORM 100S Version B(218)).

For  $D = 132.5''$  and  $H = 218''$ , the equivalent pressure is

$$P = 192,700 \text{ lb}/(132.5'' \times 218'') = 6.67 \text{ psi}$$

Therefore, establishing 5 psi as the design basis steady state pressure differential (Table 2.2.1) across the overpack diameter is reasonable.

Since the actual explosion produces a transient wave, the use of a static incipient tip calculation is very conservative. To evaluate the margin against tip-over from a short-time pressure pulse, a Working Model analysis of the two-dimensional dynamic motion of the HI-STORM subject to a given initial angular velocity is carried out. Figures 3.4.25 and 3.4.26 provide details of the model and the solution for a HI-STORM 100 System (simulated as a rigid body) having a weight and inertia property appropriate to a minimum weight cask of height  $H=235''$ . The results show that an initial angular velocity of 0.626 radians/second does not lead to a tipover of the storage overpack. The results bound those obtained for the HI-STORM 100S(232) and for the HI-STORM 100S Version B (229) since the overall cask height is reduced. The results for the HI-STORM 100S(243) are roughly equal to the results for the HI-STORM 100 since the differences in height and weight are negligible. The results for the HI-STORM 100S Version B will be bounded by the



results presented because of lower centroid location.

Continuing, the initial angular velocity can be related to a square wave pressure pulse of magnitude  $P$  and time duration  $T$  by the following formula:

$$I\omega = (P \times D \times H) \times (0.5 \times H) \times T$$

The above formula relates the change in angular motion resulting from an impulsive moment about the base of the overpack.  $D$  is the diameter of the outer shell,  $H$  is the height of the storage overpack, and  $I$  is the mass moment of inertia of the storage overpack about the mass center (assumed to be at half-height). For  $D=132.5''$ ,  $H=235''$ ,  $P=10$  psi,  $T=1$  second, and  $I=64,277,000$  lb.inch<sup>2</sup>, the resulting initial angular velocity is:

$$\omega = 0.569 \text{ radians/second}$$

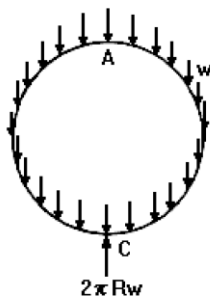
Therefore, an appropriate short time pressure limit is 10 psi with pulse duration less than or equal to 1 second. Table 2.2.1 sets this as the short-time external pressure differential.

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

The analysis in Subsection 3.4.7.1 evaluates ovalization of the shell by considering the seismically applied load as a line loading along the height of the overpack that is balanced by inertial body forces in the metal ring. The same solutions can be used to examine the circumferential stress state that would be induced to resist an external pressure that developed around one-half of the periphery. Such a pressure distribution may be induced by a pressure wave crossing the cask from a nearby explosion. It is shown here that a uniform pressure load over one-half of the overpack outer shell gives rise to an elastic stress state and deformation state that is bounded by a large margin by the results just presented for the seismic event in Subsection 3.4.7.1.

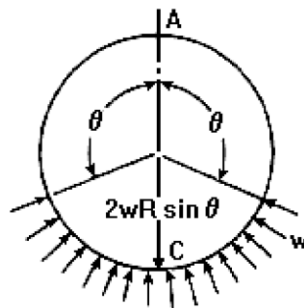
The case of an external pressure load from an explosion pressure wave (Load Case 05 in Table 3.1.5) is examined by combining the solutions for two different load cases. The combined case that results is a balance of pressure load over one-half the perimeter and inertial body forces. The sketch below describes this:

Case 1



+

Case 3



Both cases are considered under identical total loads (with the angle in case 3 set to 90 degrees). Therefore, adding the results from the two cases results in the desired combined case; namely, the balance of a peripheral external pressure with internal all around loading simulating an inertia load (since the reactions are identical in magnitude and opposite in direction, there is a complete cancellation of the concentrated loads).

Examination of the results shows that the algebraic sum of the two solutions gives results that are smaller in magnitude than the case 1 solution for a line loading balanced by inertially induced body forces. The applied loading used to develop the solution for case 1 is 56,180 lb. per inch of storage overpack axial length. This load is equivalent to an external pressure  $P = 424$  psi applied over one-half of the outer perimeter of the shell as is shown below:

$$P \times D = 56,180 \text{ lb./inch} \quad D = 132.5'' \quad \text{Therefore, } P = 424 \text{ psi}$$

Since this is higher by a large margin than any postulated external pressure load, circumferential stresses induced by the differential pressure specified in Table 2.2.1 are insignificant. Specifically, by adding the results from the two solutions (ring load case 1 for a point support reaction to a body force + ring load case 3 for a point support reaction to a lateral pressure over one-half of the perimeter), it is determined that the circumferential bending stress from case 1 is reduced by the factor "R" to obtain the corresponding stress from the combined case. R is computed as the ratio of moment magnitudes from the combined case to the results of case 1 alone.

$$R = (\text{maximum bending moment from case 1 + case 3}) / (\text{maximum bending moment from case 1}) \\ = 0.75 / 6.197 = 0.12$$

Examination of the graphs from the moment distribution from the two solutions shows that the individual terms always subtract and nearly cancel each other at every location.

Therefore, it is concluded that the maximum circumferential stress that develops under a pressure of 424 psi applied over one-half of the perimeter, and conservatively assumed balanced by inertia loading, is

$$\text{Stress} = 29,310 \text{ psi} \times 0.12 = 3517 \text{ psi}$$

The stress due to a differential pressure of 10 psi (Table 2.2.1) is only 2.36% of the above value and needs no further evaluation for stress limits or deformation to demonstrate retrievability of the MPC. Because of the large margin obtained for a specific set of values appropriate to the HI-STORM 100, the same conclusion is reached for the HI-STORM 100S and the HI-STORM 100S, Version B; that is, differential pressures of the postulated magnitude will not affect retrievability of the stored MPC.

### 3.4.7.3 Anchored HI-STORM Systems Under High-Seismic DBE (Load Case C in Table 3.1.1)

The anchored HI-STORM System (Figures 1.1.4 and 1.1.5) is assumed to be subjected to quasi-static inertial seismic loads corresponding to the ZPA design basis limits given in Table 2.2.8. The

results from this quasi-static analysis are used to evaluate structural margins for the preloaded anchor studs and the sector lugs. In the quasi-static evaluation, the effect of the “rattling” of the MPC inside of the overpack is accounted for by the imposition of a dynamic load factor of 2.0 on the incremental stresses that arise during the seismic event. In addition to the quasi-static analysis, confirmatory 3-D dynamic analyses are performed using base acceleration excitation histories developed from two sets of response spectra. Figure 3.4.30 shows the two sets of response spectra that are assumed to be imposed at the top of the ISFSI pad. One set of response spectra is the Regulatory Guide 1.60 spectra for 5% damping with zero period acceleration conservatively amplified to 1.5 in each direction. This spectra set has been used as the input spectra at many nuclear plants in the U.S. (although generally, the ZPA was much below 1.0). Three statistically independent acceleration time histories (two horizontal labeled as “H1”, “H2”) and one vertical (labeled as “VT”) have been developed. A twenty-second duration event was considered. Figures 3.4.31 to 3.4.33 show the time histories. The second set of response spectra used for time history analysis has similar levels of zero period acceleration but has higher peak spectral acceleration values in the low frequency range (2-3 Hz). This spectra set is the design basis set for a Pacific coast U.S. plant. Figures 3.4.34 to 3.4.36 (labeled as “FN”, “FP” for the two horizontal acceleration histories and “FV” for the vertical acceleration time history), show the corresponding time histories simulating a long duration seismic event (170 seconds).

The objectives of the quasi-static and dynamic seismic analyses are the following:

- i. Quantify the structural safety factor in the anchor studs and in the sector lugs that constitute the fastening system for the loaded HI-STORM 100A overpack. The structural safety factor is defined as the ratio of the permitted stress (stress intensity) per Subsection “NF” of the ASME Code to the maximum stress (stress intensity) developed in the loaded component.
- ii. Compute the safety factor against fatigue failure of the anchor studs from a single seismic event.
- iii. Quantify the interface loads applicable to the ISFSI pad to enable the ISFSI owner to design the ISFSI pad under the provisions of ACI-349 (85). The bounding interface loads computed for the maximum intensity seismic event (ZPA) and for extreme environmental loads may be used in pad design instead of the site-specific loads calculated for the loadings applicable to the particular ISFSI.

The above design objectives are satisfied by performing analyses of a loaded HI-STORM 100A System using a conservative set of input data and a conservative dynamic model. Calculations using the quasi-static model assume that the net horizontal inertia loads and the vertical inertia load correspond to the weight of the loaded cask times the appropriate ZPA. The results from the analyses are set down as the interface loads, and may be used in the ISFSI pad design work effort by the ISFSI owner. The information on the seismic analysis is presented in five paragraphs as follows:

- Input data for analysis
- Quasi-static model and results
- Dynamic model and modeling assumptions.

Results of dynamic analysis  
Summary of interface loads

a. Input Data for Analysis:

Key input data for the seismic analysis of a loaded HI-STORM 100A System is summarized in Table 3.4.10. As can be seen from Table 3.4.10, the input data used in the analysis is selected to bound the actual data, wherever possible, so as to maximize the seismic response. For example, a bounding weight of the loaded MPC and HI-STORM 100A overpack is used because an increase in the weight of the system directly translates into an increased inertial loading on the structure.

For quasi-static analysis, bounding ZPA values of 1.5 in all three directions are used with the vertical event directed upward to maximize the stud tension. The resulting ZPAs are then further amplified by the dynamic load factor (DLF=2.0) to reflect "rattling" of the MPC within the overpack. Input data for anchor stud lengths are representative. We consider long and short studs in order to evaluate the effect of stud spring rate.

For the confirmatory dynamic analyses, the time history base excitations are shown in Figures 3.4.31 through 3.4.36 and the propensity for "rattling" is included in the model.

b. Quasi-Static Model and Results:

We consider the HI-STORM100A baseplate as a rigid plate resting on the ISFSI pad with the twenty-eight studs initially preloaded so as to impart a compressive load at the baseplate pad interface that is balanced by a tensile load in the studs prior to the seismic event occurring. The discrete studs are replaced by a thin ring located at the stud circle radius for analysis purposes. The thickness of the thin ring is set so that the ring area is equal to the total stress area of the twenty-eight studs. Figure 3.4.37 shows a view of a segment of the baseplate with the outline of the ring. The ISFSI pad is represented by a linear spring and a rotational spring with spring constants determined from the exact solution for a rigid circular punch pressed into a elastic half-space. We assume that subsequent to pre-tensioning the studs, the seismic event occurs, represented by a net horizontal load DH and a net vertical load DV. In the analysis, the input loads DH and DV are:

$$G_H = (1.5^2 \times 2)^{1/2} \times DLF = 4.242 ; \quad G_V = 1.5 \times DLF = 3.0$$

$$DH = G_H \times 360,000 \text{ lb.} ; \quad DV = -G_V \times 360,000 \text{ lb}$$

DH is the magnitude of the vector sum of the two horizontal ZPA accelerations multiplied by the bounding HI-STORM 100A weight. Similarly, DV is an upward directed load due to the vertical ZPA acceleration. The upward direction is chosen in order to maximize the stud tension as the assemblage of studs and foundation resists overturning from the moment induced by DH applied at the centroid of the cask. Figure 3.4.38 shows the free-body diagram associated with the seismic event. Essentially, we consider an analysis of a pre-compressed interface and determine the interface joint behavior under the imposition of an external loading (note that this kind of analysis is well established in the pressure vessel and piping area where it is usually associated with establishing the effectiveness of a gasketed joint). An analysis is performed to determine the

maximum stud tension that results if the requirement of no separation between baseplate and pad is imposed under the imposed loading. The following result is obtained from static equilibrium, for a preload stress of 60 ksi, when the “no separation condition” is imposed:

$$\frac{2a/3h_{cg} (F_{\text{preload}}/W + 1)(1 + \alpha_1)}{G_H - 2a/3h_{cg} (G_V (1 + \alpha_1)/(1 + \alpha))} = 1.016$$

In the above equation,

$F_{\text{preload}} = (\text{Total stress area of twenty-eight, 2” diameter studs}) \times 60 \text{ ksi} = 4,200,000 \text{ lb.}$

$W = \text{Bounding weight of loaded HI-STORM 100A} = 360,000 \text{ lb.}$

$a = 73.25 \text{ inches,}$

$h_{cg} = 118.5 \text{ inches}$

The coefficients  $\alpha$  and  $\alpha_1$  relate the stiffness of the totality of studs to the stiffness of the foundation under direct loading and under rotation. The result given above is for the representative case of stud free length “L”, equal to

$L = 42 \text{ inches,}$  which gives  $\alpha$  and  $\alpha_1$  equal to 0.089 and 0.060, respectively.

A simplified confirmatory analysis of the above problem can be performed by considering the limiting case of a rigid baseplate and a rigid ISFSI pad. In the limit of a rigid ISFSI pad (foundation), the coefficients  $\alpha$  and  $\alpha_1$  go to zero. A related solution for the case of a rigid baseplate and a rigid foundation can be obtained when the criteria is not incipient separation, but rather, a more “liberal” incipient rotation about a point on the edge of the baseplate. That solution is given in “Mechanical Design of Heat Exchangers and Pressure Vessel Components”, by Singh and Soler (Arcturus Publishers, 1984). The result is (for 60 ksi pre-stress in each stud):

$$\frac{a/h_{cg} (F_{\text{preload}}/W + 1)}{G_H - a/h_{cg} (G_V)} = 1.284$$

Although not a requirement of any design code imposed herein, the right hand side of the previous relationships can be viewed as the safety factor against incipient separation (or rotation about an edge) at the radius “a”. Note that since we have assumed a bounding event, there is an additional margin of 1.5 in results since the Reg. Guide 1.60 event has not been applied with a ZPA in excess of 1.0.

For the real seismic event associated with a western U.S. plant having a slightly lower horizontal ZPA and a reduced vertical ZPA (see Figure 3.4.30). Using the same DLF =2.0 to account for “rattling” of the confined MPC:

$$G_H = 4.1 \quad ; \quad G_V = 2.6,$$

the aforementioned safety factors are:

$$\begin{aligned} \text{SF (incipient separation)} &= 1.076 \\ \text{SF (incipient edging)} &= 1.372 \end{aligned}$$

The increment of baseplate displacement and rotation, up to incipient separation, is computed from the equilibrium and compatibility equations associated with the free body in Figure 3.4.38 and the change in stud tension computed. The following formula gives the stud tensile stress in terms of the initial preload and the incremental change from the application of the horizontal and vertical seismic load.

$$\sigma_{\text{stud}} = \sigma_{\text{preload}} + \alpha \frac{W}{NA_{\text{stress}}} \left( \frac{-G_V}{1 + \alpha} + \left( \frac{3h_{cg}}{2a} \right) \left( \frac{c}{a} \right) \left( \frac{G_H}{1 + \alpha_1} \right) \right)$$

In the above formula,

$N$  = number of studs = 28 (maximum number based on HI-STORM dimensions). For lower seismic inputs, this might be reduced (in groups of 4 to retain symmetry).

$A_{\text{stress}}$  = tensile stress area of a 2" diameter stud

$2c$  = stud circle diameter

The results demonstrate that there is a relatively small change in stud stress from the initial pre-tension condition with the ISFSI pad foundation resisting the major portion of the overturning moment. For the geometry considered (maximum stud free length and nominal pre-stress), the maximum tensile stress in the stud increases by 9.1%. The following table summarizes the results from the quasi-static analysis using minimum ultimate strength for the stud to compute the safety factors. Note that under the seismic load, the direct stress in the stud is limited to 70% of the stud ultimate strength (per Appendix F of the ASME Code Section III). The allowable pad compressive stress is determined from the ACI Code assuming confined concrete and the minimum concrete compressive strength from Table 2.0.4. Because of the large compressive load at the interface from the pre-tensioning operation, the large frictional resistance inhibits sliding of the cask. Consequently, there will be no significant shear stress in the studs. Safety factors for sliding are obtained by comparing the ratio of horizontal load to vertical load with the coefficient of friction between steel and concrete (0.53). Values in parenthesis represent results obtained using ZPA values associated with the real seismic event for the western U.S. plant instead of the bounding Reg. Guide 1.60 event.

<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI-STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 60 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	65.48 (65.18)	87.5	1.336 (1.343)

Maximum Pad Pressure (ksi)(42" stud free length)	3.126 (3.039)	4.76	1.52 (1.57)
Stud Stress (ksi)(16" stud free length)	73.04 (72.34)	87.5	1.20 (1.21)
Maximum Pad Pressure(ksi) (16" stud free length)	2.977 (2.898)	4.76	1.60 (1.64)
Overpack Sliding	0.439 (0.407)	0.53	1.21 (1.31)

The effect of using a minimum stud free length in the embedment design is to increase the values of the coefficients  $\alpha$  and  $\alpha_1$  because the stud stiffness increases. The increase in stud stiffness, relative to the foundation stiffness results in an increase in incremental load on the studs. This is a natural and expected characteristic of preloaded configurations. It is noted that the stud safety factors are based on minimum ultimate strength and can be increased, without altering the calculated results, by changing the stud material.

The quasi-static analysis methodology has also been employed to evaluate the effects of variation in the initial pre-stress on the studs. The following tables reproduce the results above for the cases of lower bound stud pre-stress (55 ksi) and upper bound stud pre-stress (65 ksi) on the studs. Only the results using the values associated with the Reg. Guide 1.60 bounding event are reported.

<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI- STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 55 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	60.48	87.5	1.45
Maximum Pad Pressure (ksi)(42" stud free length)	3.012	4.76	1.58
Stud Stress (ksi)(16" stud free length)	68.07	87.5	1.29
Maximum Pad Pressure(ksi) (16" stud free length)	2.862	4.76	1.663
Overpack Sliding	0.488	0.53	1.09

<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI- STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 65 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	70.48	87.5	1.24
Maximum Pad Pressure (ksi)(42" stud free length)	3.24	4.76	1.47
Stud Stress (ksi)(16" stud free length)	78.07	87.5	1.12
Maximum Pad Pressure(ksi) (16" stud free length)	3.091	4.76	1.54

Overpack Sliding	0.399	0.53	1.33
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The results above confirm the expectations that an increase in preload increases the safety factor against sliding. The calculated coefficient of friction in the above tables is computed as the ratio of applied horizontal load divided by available vertical load. For all combinations examined, ample margin against incipient separation at the interface exists.

Based on the results from the quasi-static analysis, an assessment of the safety factors in the sector lugs is obtained by performing a finite element analysis of a repeated element of one of the sector lugs. Figure 3.4.39 shows the modeled section and the finite element mesh. The stud load is conservatively applied as a uniform downward pressure applied over a 5"x5" section of the extended baseplate simulating the washer between two gussets. This is conservative as the rigidity of the washer is neglected. The opposing pressure loading from the interface pressure is applied as a pressure over the entire extended baseplate flat plate surface. Only one half the thickness of each gusset plate is included in the model. The outer shell is modeled as 3/4" thick, which corresponds to the minimum thickness option per Bill of Material 1575.

Two cases are considered: (1) the pre-loaded state (a Normal Condition of Storage-Level A stress limits apply); and, (2), the seismic load condition at the location of the maximum tensile load in a stud (an Accident Condition of Storage – Level D stress intensity limits apply). Figures 3.4.40 and 3.4.41 present the stress results for the following representative input conditions:

Level A analysis - Preload stress/bolt = 60 ksi

Level D analysis - Maximum Bolt stress (includes seismic increment) = 65.5 ksi

In the Level A analysis, the resisting local foundation pressure exactly balances the preload. For the Level D analysis, the opposing local foundation pressure = 190 psi (average over the area between gussets). This represents the reduced pressure under the highest loaded stud under the induced rotation of the storage system.

The most limiting weld stress is obtained by evaluating the available load capacity of the fillet weld attaching the extended baseplate annulus region to the gussets (approximately 25 inches of weld per segment) using a limit strength equal to 42% of the ultimate strength of the base material.

The following table summarizes the limiting safety factors for the sector lugs. Allowable values for primary bending stress and stress intensity are from Tables 3.1.10 and 3.1.12 for SA-516 Grade 70 at 300 degrees F.

<b>SUMMARY OF RESULTS FOR SECTOR LUGS FROM QUASI-STATIC SEISMIC EVALUATION</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Maximum Primary Membrane + Bending Stress Away From Loaded Region and Discontinuity (ksi) – Case 1 - Preload	15.62	26.3	1.68



Maximum Primary Membrane + Bending Stress Intensity Away From Loaded Region and Discontinuity (ksi) – Case 2 - Preload + Seismic	36.67	60.6	1.65
Maximum Weld Shear Load (kips)	150.8	194.9	1.29

c. Dynamic Model and Modeling Assumptions:

The dynamic model of the HI-STORM 100A System consists of the following major components.

- i. The HI-STORM 100 overpack is modeled as a six degree-of-freedom (rigid body) component.
- ii. The loaded MPC is also modeled as a six degree-of-freedom (rigid body) component that is free to rattle inside the overpack shell. Gaps between the two bodies reflect the nominal dimensions from the drawings.
- iii. The contact between the MPC and the overpack is characterized by a coefficient of restitution and a coefficient of friction. For the dynamic analysis, the coefficient of restitution is set to 0.0, reflecting the large areas of nearly flat surface that come into contact and have minimal relative rebound. The coefficient of friction is set to 0.5 between all potentially contacting surfaces of the MPC/overpack interface.
- iv. The anchor studs, preloaded to axial stress  $\sigma_i$  (Table 3.4.10), induce a contact stress between the overpack base and the ISFSI pad. The loaded cask-pad interface can support a certain amount of overturning moment before an uplift (loss of circularity of the contact patch) occurs. The anchor studs are modeled as individual linear springs connecting the periphery of the extended baseplate to the ISFSI pad section. The resistance of the foundation is modeled by a vertical linear spring and three rotational springs connected between the cask baseplate center point and the surface of the flat plate modeling the driven ISFSI pad. The ISFSI pad is driven with the three components of acceleration time history applied simultaneously.

The HI-STORM 100A dynamic model described above is implemented on the public domain computer code WORKING MODEL (also known as VisualNastran) (See Subsection 3.6.2 for a description of the algorithm).

Figures 3.4.42 and 3.4.43 show the rigid body components of the dynamic model before and after assembly. The linear springs are not shown. Mass and inertia properties of the rigid bodies are consistent with the bounding property values in Table 3.4.10.

d. Results of Dynamic Analysis:

Figures 3.4.44 –3.4.47 show results of the dynamic analysis using the Reg. Guide 1.60 seismic time histories as input accelerations to the ISFSI pad. Figure 3.4.44 shows variation in the vertical

foundation compressive force. Figure 3.4.45 shows the corresponding load variation over time for the stud having the largest instantaneous tensile load. An initial preload of approximately 150,000 lb is applied to each stud (corresponding to 60,160 psi stud tensile stress). This induces an initial compression load at the interface approximately equal to 571,000 lb. (including the dead weight of the loaded HI-STORM). Figures 3.4.44 and 3.4.45 clearly demonstrate that the foundation resists the majority of the oscillatory and impactive loading as would be expected of a preloaded configuration. Figure 3.4.46 shows the impulse (between the MPC and HI-STORM 100A) as a function of time. It is clear that the “spikes” in both the foundation reaction and the stud load over the total time of the event are related to the impacts of the rattling MPC. The results provide a graphic demonstration that the rattling of the MPC inside the overpack must be accounted for in any quasi-static representation of the event. The quasi-static results presented herein for the anchored system, using a  $DLF = 2.0$ , are in excellent agreement with the dynamic simulation results.

We note that the dynamic simulation, which uses an impulse-momentum relationship to simulate the rattling contact, leads to results having a number of sharp peaks. Given that the stress intensity limits in the Code assume static analyses, filtering of the dynamic results is certainly appropriate prior to comparing with any static allowable strength. We conservatively do not perform any filtering of the results prior to comparison with the quasi-static analysis; we note only that any filtering of the dynamic results to eliminate high-frequency effects resulting from the impulse-momentum contact model would increase the safety factors. Finally, Figure 3.4.47 shows the ratio of the net interface horizontal force (needed to maintain equilibrium) to the instantaneous compression force at the ISFSI pad interface with the base of the HI-STORM 100A. This ratio, calculated at each instant of time from the dynamic analysis results using the Reg. Guide 1.60 event, represents an instantaneous coefficient of friction that is required to ensure no interface relative movement. Figure 3.4.47 demonstrates that the required coefficient of friction is below the available value 0.53. Thus, the dynamic analysis confirms that the foundation interface compression, induced by the preloading action, is sufficient to maintain a positive margin against sliding without recourse to any resistance from the studs.

The results of the dynamic analysis using acceleration time histories from the Reg. Guide 1.60 response spectra (grounded at 1.5 g's) confirm the ability of the quasi-static solution, coupled with a dynamic load factor, to correctly establish structural safety factors for the anchored cask. The dynamic analysis confirms that stud stress excursions from the preload value are minimal despite the large overturning moments that need to be balanced.

A second dynamic simulation has been performed using the seismic time histories appropriate to a pacific coast U.S. nuclear plant (Figures 3.4.34-3.4.36). The ZPA of these time histories are slightly less than the Reg. Guide 1.60 time histories but the period of relatively strong motion extends over a longer time duration. The results from this second simulation exhibit similar behavior as those results presented above and provide a second confirmation of the validity of the safety factors predicted by the quasi-static analysis. Reference [3.4.14] (see Subsection 3.8) provides archival information and backup calculations for the results summarized here.

Stress cycle counting using Figure 3.4.45 suggests 5 significant stress cycles per second provides a bounding number for fatigue analysis. A fatigue reduction factor of 4 is appropriate for the studs

(per ASME Code rules). Therefore, a conservative analysis of fatigue for the stud is based on an alternating stress range of:

$S(\text{alt}) = .5 \times (22,300 \text{ psi}) \times 4 = 44,600 \text{ psi}$  for 5 cycles per second. The value for the stress range is obtained as the difference between the largest tensile stress excursions from the mean value as indicated in the figure.

To estimate fatigue life, we use a fatigue curve from the ASME Code for high strength steel bolting materials (Figure I.9.4 in Appendix I, ASME Code Section III Appendices) For an amplified alternating stress intensity range of 44,600 psi, Figure I.9.4 predicts cyclic life of 3,000 cycles. Therefore, the safety factor for failure of a stud by fatigue during one Reg. Guide 1.60 seismic event is conservatively evaluated as:

$$SF(\text{stud fatigue}) = 3,000/100 = 30.$$

For the long duration event, even if we make the conservative assumption of a nine-fold increase in full range stress cycles, the safety factor against fatigue failure of an anchor stud from a single seismic event is 3.33. Recognizing that the fatigue curve itself is developed from test data with a safety factor of 20 on life and 4 on stress, the results herein demonstrate that fatigue failure of the anchor stud, from a single seismic event, is not credible.

e. **Summary of Interface Loads for ISFSI Pad Design:**

Bounding interface loads are set down for use by the ISFSI pad designer and are based on the validated quasi-static analysis and a dynamic load factor of 2.0:

<b>BOUNDING INTERFACE LOADS FOR ISFSI PAD STRUCTURAL/SEISMIC DESIGN</b>	
D (Cask Weight)	360 kips
D (Anchor Preload @ 65 ksi)	4,550 kips
E (Vertical Load)	1,080 kips
E (Net Horizontal Surface ShearLoad)	1,527.35 kips
E (Overturning Moment)	15,083 kip-ft.

3.4.8 **Tornado Wind and Missile Impact (Load Case B in Table 3.1.1 and Load Case 04 in Table 3.1.5)**

During a tornado event, the HI-STORM 100 System is assumed to be subjected to a constant wind force. It is also subject to impacts by postulated missiles. The maximum wind speed is specified in Table 2.2.4 and the three missiles, designated as large, intermediate, and small, are described in Table 2.2.5.

In contrast to a freestanding HI-STORM 100 System, the anchored overpack is capable of withstanding much greater lateral pressures and impulsive loads from large missiles. The quasi-static analysis result, presented in the previous subsection, can be used to determine a maximum permitted base overturning moment that will provide at least the same stud safety factors. This is accomplished by setting  $G_V = 0.0$ ,  $DLF = 1$  and finding an appropriate  $G_H$  that gives equal or better stud safety factors. The resulting value of  $G_H$  establishes the limit overturning moment for

combined tornado missile plus wind,  $M_L$ . ( $G_H \times \text{Weight} \times h_{cg}$ ) is conservatively set as the maximum permissible moment at the base of the cask due to combined action of lateral wind and tornado missile loading. Thus, if the lateral force from a tornado missile impact is  $F$  at height  $h$  and that from steady tornado wind action is a resultant force  $W$  acting at cask mid-height ( $0.5H$ ), and the two loads are acting synergistically to overturn the cask, then their magnitudes must satisfy the inequality

$$0.5WH + Fh \leq M_L$$

where the limit moment is established to ensure that the safety factors for seismic load remain bounding.

$$M_L = 18,667 \text{ kip-ft.}$$

Tornado missile impact factors should be factored into “F” prior to determining the validity of the above inequality for any specific site.

In the case of a freestanding system, the post impact response of the HI-STORM 100 System is required to assess stability. Both the HI-STORM 100 storage overpack, and the HI-TRAC transfer cask are assessed for missile penetration.

The results for the post-impact response of the HI-STORM 100 storage overpack demonstrate that the combination of tornado missile plus either steady tornado wind or instantaneous tornado pressure drop causes a rotation of the HI-STORM 100 to a maximum angle of inclination less than 3 degrees from vertical. This is much less than the angle required to overturn the cask. The results for the HI-STORM 100 are bounding since the HI-STORM 100S and the HI-STORM 100S Version B have a lower center of gravity when loaded. Since a lower bound cask weight of 302,000 lbs is used, the results are also bounding for HI-STORM overpacks that utilize high density concrete.

The maximum force (not including the initial pulse due to missile impact) acting on the projected area of the storage overpack is computed to be:

$$F = 91,920 \text{ lbs.}$$

The instantaneous impulsive force due to the missile strike is not computed here; its effect is felt as an initial angular velocity imparted to the storage overpack at time equal to zero. The net resultant force due to the simultaneous pressure drop is not an all-around distributed loading that has a net resultant, but rather is more likely to be distributed only over 180 degrees (or less) of the storage overpack periphery. The circumferential stress and deformation field will be of the same order of magnitude as that induced by a seismic loading. Since the magnitude of the force due to  $F$  is less than the magnitude of the net seismically induced force considered in Subsection 3.4.7, the storage overpack global stress analysis performed in Subsection 3.4.7 remains governing. In the next subsection, results are provided for the circumferential stress and ovalization of the portion of the storage overpack due to the bounding estimate for the impact force of the intermediate missile.

### 3.4.8.1 HI-STORM 100 Storage Overpack

This subsection considers the post impact behavior of the HI-STORM 100 System after impact from tornado missiles. During an impact, the system consisting of missile plus storage overpack and MPC satisfies conservation of linear and angular momentum. The large missile impact is assumed to be inelastic. This assumption conservatively transfers all of the momentum from the missile to the system. The intermediate missile and the small missile are assumed to be unyielding and hence the entire initial kinetic energy is assumed to be absorbed by motion of the cask and local yielding and denting of the storage overpack surface. It is shown that cask stability is maintained under the postulated wind and large missile loads. The conclusion is also valid for the HI-STORM 100S and for the HI-STORM 100S Version B with or without the densified concrete shielding option since their lower centers of gravity inherently provide additional stability margin.

The penetration potential of the missile strikes (Load Case 04 in Table 3.1.5) is examined first. The detailed calculations show that there will be no penetration through the concrete surrounding the inner shell of the storage overpack or penetration of the top closure plate. Therefore, there will be no impairment to the confinement boundary due to missile strikes during a tornado. Since the inner shell is not compromised by the missile strike, there will be no permanent deformation of the inner shell. Therefore, ready retrievability is assured after the missile strike. The following paragraphs summarize the analysis work for the HI-STORM 100.

- a. The small missile will dent any surface it impacts, but no significant puncture force is generated. The 1" missile can enter the air ducts, but geometry prevents a direct impact with the MPC.
- b. The following table summarizes the denting and penetration analysis performed for the intermediate missile. Denting is used to connote a local deformation mode encompassing material beyond the impacting missile envelope, while penetration is used to connote a plug type failure mechanism involving only the target material immediately under the impacting missile. The results are applicable to the HI-STORM 100 and to the HI-STORM 100S. The HI-STORM 100S version B has a thicker outer shell than the classic HI-STORM 100, and a lid configuration that consists of a 1" lid cover plate backed by concrete and a 3" thick lid vent shield plate that acts as a barrier to a top lid missile strike. Therefore, the tabular results presented below are bounding for the HI-STORM 100S Version B.

<b>Location</b>	<b>Denting (in.)</b>	<b>Thru-Thickness Penetration</b>
Storage overpack outer Shell	13.75 <sup>†</sup>	Yes (>0.75 in.)
Radial Concrete	18.54 <sup>††</sup>	No (<27.25 in.)
Storage overpack Top Lid	<2.0	No (<4 in.)

<sup>†</sup> Based on minimum outer shell thickness of 3/4". Penetration is less for HI-STORM 100 and 100S overpacks with 1" thick outer shell.

†† Based on concrete compressive strength equal to 50% of minimum value specified in Table 3.3.5 to account for exposure to high temperatures.

The primary stresses that arise due to an intermediate missile strike on the side of the storage overpack and in the center of the storage overpack top lid are determined next. The analysis of the storage lid for the HI-STORM 100 bounds that for the HI-STORM 100S; because of the additional energy absorbing material (concrete) in the direct path of a potential missile strike on the top lid of the HI-STORM 100S lid, the energy absorbing requirements of the circular plate structure are much reduced. The analysis demonstrates that Level D stress limits are not exceeded in either the overpack outer shell or the top lid. The safety factor in the storage overpack, considered as a cantilever beam under tip load, is computed, as is the safety factor in the top lids, considered as two centrally loaded plates. The applied load, in each case, is the missile impact load. Similar calculations are performed for the HI-STORM 100S Version B using the same model and methodology. A summary of the results for axial stress in the storage overpack is given in the table below with numbers in parentheses representing the results of calculations for the geometry of the HI-STORM 100S Version B:

<b>HI-STORM 100 MISSILE IMPACT - Global Axial Stress Results</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Outer Shell – Side Strike	14.35 <sup>†</sup> (15.17)	37.95	2.64 <sup>†</sup> (2.50)
Top Lid - End Strike	44.14(47.57)	57.0 (50.65)	1.29(1.065)

† Based on HI-STORM 100 overpack with inner and outer shell thicknesses of 1-1/4” and 3/4”, respectively. Result is bounding for HI-STORM 100 overpacks made with 1” thick inner and outer shells because the section modulus of the steel structure is greater.

To demonstrate ready retrievability of the MPC, we must show that the storage overpack suffers no permanent deformation of the inner shell that would prevent removal of the MPC after the missile strike. To demonstrate ready retrievability (for both HI-STORM 100 and for HI-STORM 100S) a conservative evaluation of the circumferential stress and deformation state due to the missile strike on the outer shell is performed. A conservative estimate for the 8” diameter missile impact force, “Pi”, on the side of the storage overpack is calculated as:

$$P_i = 1,063,000 \text{ lb.}$$

This force is conservative in that the target overpack is assumed rigid; any elasticity serves to reduce the peak magnitude of the force and increase the duration of the impact. The use of the upper bound value is the primary reason for the high axial stresses resulting from this force. To demonstrate continued ability to retrieve the MPC subsequent to the strike, circumferential stress and deformation that occurs locally in the ring section near the location of the missile strike are investigated.

Subsection 3.4.7 presents stress and displacement results for a composite ring of unit width consisting of the inner and outer shells of the storage overpack. The solution assumes that the net

loading is 56,184 lb. applied on the 1" wide ring (equivalent to a 45g deceleration applied uniformly along the height on a storage overpack weight of 270,000 lb.). This solution can be applied directly to evaluate the circumferential stress and deformation caused by a tornado missile strike on the outer shell. Using the results for the 45g tipover event, an attenuation factor to adjust the results is developed that reflects the difference in load magnitude and the width of the ring that is effective in resisting the missile strike force. The strike force  $P_i$  is resisted by a combination of inertia force and shear resistance from the portion of the storage overpack above and below the location of the strike. The ring theory solution to determine the circumferential stress and deformation conservatively assumes that inertia alone, acting on an effective length of ring, balances the applied point load  $P_i$ . The effective width of ring that balances the impact load is conservatively set as the diameter of the impacting missile (8") plus the effect of the "bending boundary layer" length. This boundary layer length is conservatively set as a multiple of twice the square root of the product of mean radius times the average thickness of two shells making up the cylindrical body of the storage overpack. The mean radius of the composite cylinder and the average thickness of the inner and outer shells are

$$R_{\text{mean}} = 48''$$

$$T = .5 \times (.75'' + 1.25'') = 1''$$

The bending boundary layer " $\beta$ " in a shell is generally accepted to be given as  $(2(R_{\text{mean}}T)^{1/2}) = 13.85''$  for this configuration. That is, the effect of a concentrated load is resisted mainly in a length along the shell equal to the bending boundary layer. For a strike away from the ends of the shell, a boundary layer length above and below the strike location would be effective (i.e., double the boundary layer length). However, to conservatively account for resistance above and below the location of the strike, this calculated result is only increased by 1.5 in the following analysis (rather than 2). Therefore, the effective width of ring is assumed as:

$$13.85'' \times 1.5 + 8'' = 28.78''$$

The solution for the 45g tipover event (performed for a unit ring width and a load of 56,184 lb.) is directly applicable if we multiply all stress and displacement results by the factor "Y" where

$$Y = (1''/28.78'') \times (1,063,000 \text{ lb.}/56,184 \text{ lb.}) = 0.658$$

Using this factor gives the following bounding results for maximum circumferential stresses (without regard for sign and location of the stress) and deformations due to the postulated tornado missile strike on the side of the storage overpack outer shell:

$$\text{Maximum circumferential stress due to bending moment} = (29,310 \text{ psi} \times Y) = 19,272 \text{ psi}$$

$$\text{Maximum circumferential stress due to mean tangential force} = (18,900 \text{ lb.}/2 \text{ sq.inch}) \times Y = 6,214 \text{ psi}$$

$$\text{Change in diameter in the direction of the load} = -0.11'' \times Y = -0.072''$$

Change in diameter perpendicular to the direction of the load =  $+0.06'' \times Y = 0.039''$

Based on the above calculation, the safety factor on maximum stress for this condition is

$$SF = 37,950\text{psi}/19,272 \text{ psi} = 1.97$$

The allowable stress for the above calculation is the Level D membrane stress intensity limit from Table 3.1.12 at 450°F . This is a conservative result since the stress intensity is localized and need not be compared to primary membrane stress intensity. Even with the overestimate of impact strike force used in the calculations here, the stresses remain elastic and the calculated diameter changes are small and do not prevent ready retrievability of the MPC. Note that because the stresses remain in the elastic range, there will be no post-strike permanent deformation of the inner shell.

The above calculations remain valid for the HI-STORM 100S, Version B using normal weight concrete and are bounding for the case where densified concrete is used.

### 3.4.8.2 HI-TRAC Transfer Cask

#### 3.4.8.2.1 Intermediate Missile Strike

HI-TRAC is always held by the handling system while in a vertical orientation completely outside of the fuel building (see Chapter 2 and Chapter 8). Therefore, considerations of instability due to a tornado missile strike are not applicable. However, the structural implications of a missile strike require consideration.

The penetration potential of the 8" missile strike on HI-TRAC (Load Case 04 in Table 3.1.5) is examined at two locations:

1. the lead backed outer shell of HI-TRAC.
2. the flat transfer lid consisting of multiple steel plates with a layer of lead backing.

In each case, it is shown that there is no penetration consequence that would lead to a radiological release. The following paragraphs summarize the analysis results.

- a. The small missile will dent any surface it impacts, but no significant puncture force is generated.
- b. The following table summarizes the denting and penetration analysis performed for the intermediate missile. Denting connotes a local deformation mode encompassing material beyond the impacting missile envelope, while penetration connotes a plug type failure mechanism involving only the target material immediately under the impacting missile. Where there is through-thickness penetration, the lead and the inner plate absorb any residual energy remaining after penetration of the outer plate in the 100 Ton HI-TRAC transfer lid. The table summarizes the bounding results for both transfer casks.



Location	Denting (in.)	Thru-Thickness Penetration
Outer Shell - lead backed	0.498	No (<1.0 in.)
Outer Transfer Lid Door	0.516	No (<0.75 in.) (HI-TRAC 125) Yes (>0.5 in.) (HI-TRAC 100)

Based on the above results, the intermediate missile will penetrate the ½” thick bottom plate of the HI-TRAC 100D pool lid. However, the lead and the pool lid top plate will absorb any residual energy remaining after penetration of the bottom plate. The 8” missile will not penetrate the pool lid for the HI-TRAC 125D because it has a thicker bottom plate than the HI-TRAC 125 transfer lid door. In addition, the results for the 8” missile strike on the HI-TRAC outer shell are valid for the HI-TRAC 125D and the HI-TRAC 100D since all four transfer casks have the same outer shell thickness.

While the transfer cask is being transported in a horizontal orientation, the MPC lid is exposed. We conservatively assume no protective plate in place during this transport operation and evaluate the capacity of the lid peripheral groove weld to resist the impact load. The calculated result is as follows:

<b>HI-TRAC MISSILE IMPACT - Capacity Results</b>			
Item	Value (lb)	Capacity (lb)	Safety Factor = Capacity/Value
Top Lid Weld	2,262,000	2,789,000	1.23

The final calculation in this subsection is an evaluation of the circumferential stress and deformation consequences of the horizontal missile strike on the periphery of the HI-TRAC shell. It is assumed that the HI-TRAC is simply supported at its ends (while in transit) and is subject to a direct impact from the 8” diameter missile. To compute stresses, an estimate of the peak impact force is required. The effect of the water jacket to aid in the dissipation of the impact force is conservatively neglected. The only portion of the HI-TRAC cylindrical body that is assumed to resist the impact load is the two metal shells. The lead is assumed only to act as a separator to maintain the spacing between the shells. The previous results from the lead slump analysis demonstrate that this conservative assumption on the behavior of the lead is valid. The peak value of the impact force is a function of the stiffness of the target. The target stiffness in this postulated event has the following contributions to the stiffness of the structure.

- a. a global stiffness based on a beam deformation mode, and
- b. a local stiffness based on a shell deformation mode

The global spring constant (i.e., the inverse of the global deflection of the cask body as a beam under a unit concentrated load) is a function of location of the strike along the length of the cask. The spring constant value varies from a minimum for a strike at the half-height to a maximum

value for a strike near the supports (the trunnions). Since the peak impact force is larger for larger stiffness, it is conservative to maximize the spring constant value. Therefore, in the calculation, we neglect this spring constant for the computation of peak impact force and focus only on the spring constant arising from the local deformation as a shell, in the immediate vicinity of the strike. To this end, the spring constant is estimated by considering the three-dimensional effects of the shell solution to be replaced by the two-dimensional action of a wide ring. The width of the ring is equal to the “bending boundary layer” length on either side of the strike location plus the diameter of the striking missile. Following the analysis methodology already utilized subsection 3.4.8.1, the following information is obtained:

The mean radius of the composite cylinder and the average thickness of the inner and outer shells, are (use the 100 Ton HI-TRAC data since it provides an upper bound on stress and deformation):

$$R_{\text{mean}} = 36.893$$

$$T = .5 \times (.75'' + 1.00'') = 0.875''$$

The bending boundary layer “ $\beta$ ” in a shell is generally accepted to be given as  $(2(R_{\text{mean}}T)^{1/2})$ . To account for resistance above and below the location of the strike, this calculated result is conservatively increased by multiplying by 1.5. Therefore, the effective width of ring is:

$$11.22'' \times 1.5 + 8'' = 24.84''$$

The missile impact is modeled as a point load, acting on the ring, of magnitude equal to  $P_i = 20,570$  lb. The use of a point load in the analysis is conservative in that it overemphasizes the local stress. The actual strike area is an 8” diameter circle (or larger, if the effect of the water jacket were included).

The force is assumed resisted by inertia forces in the ring section. From the results, a spring constant can be defined as the applied load divided by the change in diameter of the ring section in the direction of the applied load. Based on this approach, the following local spring constant is obtained:

$$K = P_i/D1_H = P_i/0.019'' = 1,083,000 \text{ lb./inch}$$

To determine the peak impact force, a dynamic analysis of a two-body system has been performed using the “Working Model” dynamic simulation code. A two mass-spring damper system is considered with the defined spring constant representing the ring deformation effect. Figure 3.4.24 shows the results from the dynamic analysis of the impact using the computer code “Working Model”. The small square mass represents the missile, while the larger mass represents the portion of the HI-TRAC “ring” assumed to participate in the local impact. The missile weight is 275.5 lb. and the participating HI-TRAC weight is set to the weight of the equivalent ring used to determine the spring constant.

The peak impact force that results in each of the two springs used to simulate the local elasticity of the HI-TRAC (ring) is:

$$F(\text{spring}) = 124,400 \text{ lb.}$$

Since there are two springs in the model, the total impact force is:

$$P(\text{impact}) = 248,800 \text{ lb.}$$

To estimate circumferential behavior of the ring under the impact, the previous solution (using a load of 20,570 lb.) is used and amplified by the factor “Z”, where:

$$Z = 248,800 \text{ lb.}/20,570 \text{ lb.} = 12.095$$

Consequently, the maximum circumferential stress due to the ring moment, away from the impact location, is:

$$3,037 \text{ psi} \times (69,260 \text{ in-lb}/180,900 \text{ in-lb}) \times Z = 14,230 \text{ psi}$$

At the same location, the mean stress adds an additional component (the ring area is computed based on the effective width of the ring).

$$(5,143 \text{ lb.}/43.47 \text{ sq.in}) \times Z = 1431 \text{ psi}$$

Therefore, the safety factor on circumferential stress causing ovalization of an effective ring section that is assumed to resist the impact is:

$$SF(\text{ring stress}) = 39,750 \text{ psi}/(1431 \text{ psi} + 14,230 \text{ psi}) = 2.54$$

The allowable stress for this safety factor calculation is obtained from Table 3.1.12 for primary membrane stress intensity for a Level D event at 350 degrees F material temperature. Noting that the actual circumferential stress in the ring remains in the elastic range, it is concluded that the MPC remains readily retrievable after the impact since there is no permanent ovalization of the cavity after the event. As noted previously, the presence of the water jacket adds an additional structural barrier that has been conservatively neglected in this analysis.

#### 3.4.8.2.2 Large Missile Strike

The effects of a large tornado missile strike on the side (water jacket outer enclosure) of a loaded HI-TRAC has been simulated using a transient finite element model of the transfer cask and loaded MPC. The transient finite element code LSDYNA3D has been used (approved by the NRC for use in impact analysis (see Appendix 3.A, reference [3.A.4] for the benchmarking of this computer code)). An evaluation of MPC retrievability and global stress state (away from the impact area) are of primary interest. The finite element model includes the loaded MPC, the HI-TRAC inner and outer shells, the HI-TRAC water jacket, the lead shielding, and the appropriate HI-TRAC lids. The water in the water jacket has been neglected for conservatism in the results. The large tornado missile has been simulated by an impact force-time pulse applied on an area representing the frontal area of an 1800-kg. vehicle. The force-time data used has been previously approved by the USNRC (Bechtel Topical Report BC-TOP-9A, “Design of Structures for Missile Impact”,

Revision 2, 9/1974). The frontal impact area used in the finite element analysis is that area recommended in NUREG-0800, SRP 3.5.1.4, Revision 2, 1981).

A summary of the results is presented below for the HI-TRAC 100 and HI-TRAC 125 transfer casks. Since the dimensions of the inner shell, the outer shell, the lead shielding, and the water jacket enclosure panels are the same in both the HI-TRAC 125 and the HI-TRAC 125D, the results from the HI-TRAC 125 are considered accurate for the HI-TRAC 125D. Likewise, the results from the HI-TRAC 100 are valid for the HI-TRAC 100D. The allowable value listed for the stress intensity for this Level D event comes from Table 3.1.17.

The results from the dynamic analysis have been summarized below.

<b>SUMMARY OF RESULTS FROM LARGE TORNADO MISSILE IMPACT ANALYSIS</b>		
<b>ITEM – HI-TRAC 100</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Maximum Stress Intensity in Water Jacket (ksi)	33.383	58.7
Maximum Stress Intensity in Inner Shell (ksi)	15.6	58.7
Maximum Plastic Strain in Water Jacket	0.0	-
Maximum Plastic Strain in Inner Shell	0.0	-

<b>ITEM – HI-TRAC 125</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Maximum Stress Intensity in Water Jacket (ksi)	33.697	58.7
Maximum Stress Intensity in Inner Shell (ksi)	18.669	58.7
Maximum Plastic Strain in Water Jacket	0.0	-
Maximum Plastic Strain in Inner Shell	0.0	-

The above results demonstrate that:

1. The retrievability of the MPC in the wake of a large tornado missile strike is not adversely affected since the inner shell does not experience any plastic deformation.
2. The maximum primary stress intensity, away from the impact interface on the HI-TRAC water jacket, is below the applicable ASME Code Level D allowable limit for NF, Class 3 structures.

#### 3.4.9 HI-TRAC Drop Events (Load Case 02.b in Table 3.1.5)

During transit, the HI-TRAC 125 or HI-TRAC 100 transfer cask may be carried horizontally with the transfer lid in place. Analyses have been performed to demonstrate that under a postulated carry height; the design basis 45g deceleration is not exceeded. The analyses have been performed

using two different simulation models. A simplified model of the drop event is performed using the computer simulation code “Working Model 2D”. The analysis using “Working Model 2D” assumed the HI-TRAC and the contained MPC acted as a single rigid body. A second model of the drop event uses DYNA3D, considers the multi-body analysis of HI-TRAC and the contained MPC as individual bodies, and is finite element based. In what follows, we outline the problem and the results obtained using each solution methodology.

#### 3.4.9.1 Working Model 2D Analysis of Drop Event

The analysis model conservatively neglects all energy absorption by any component of HI-TRAC; all kinetic energy is transferred to the ground through the spring-dampers that simulate the foundation (ground). If the HI-TRAC suffers a handling accident causing a side drop to the ground, impact will only occur at the top and bottom ends of the vessel. The so-called “hard points” are the top end lifting trunnions, the bottom end rotation trunnions, and the projecting ends of the transfer lid. Noting that the projecting hard points are of different dimensions and will impact the target at different times because of the HI-TRAC geometry, any simulation model must allow for this possibility.

A dynamic analysis of a horizontal drop, with the lowest point on the HI-TRAC assumed 50” above the surface of the target (larger than the design basis limit of 42”), is considered for the HI-TRAC 125 and for the HI-TRAC 100. Figure 3.4.22 shows the transfer cask orientation. The HI-TRAC is considered as a rigid body (calculations demonstrate that the lowest beam mode frequency is well above 33 Hz so that no dynamic amplification need be included). The effects of the ISFSI pad and the underlying soil are included using a simple spring-damper model based on a static classical Theory of Elasticity solution. The “worst” orientation of a horizontally carried HI-TRAC with the transfer cask impacting an elastic surface is considered. The HI-TRAC is assumed to initially impact the target with the impact force occurring over the rectangular surface of the transfer lid (11.875” x 81”). “Worst” is defined here as meaning an impact at a location having the maximum value of an elastic spring constant simulating the resistance of the target interface. The geometry and material properties reflect the USNRC accepted reference pad and soil (Table 2.2.9 - the pad thickness used is 36” and the Young’s Modulus of the elastic soil is the upper limit value  $E=28,000$  psi). The use of an elastic representation of the target surface is conservative as it minimizes the energy absorption capacity of the target and maximizes the deceleration loads developed during the impact. The spring constant is also calculated based on an assumption that impact at the lower end of HI-TRAC first occurs at the pocket trunnion. The results demonstrate that this spring constant is lower and therefore would lead to a lower impact force. Therefore, the dynamic analysis of the handling accident is performed assuming initial impact with the flat rectangular short end of the transfer lid. Subsequent to the initial impact, the HI-TRAC rotates in accordance with the dynamic equations of equilibrium and a secondary impact at the top of the transfer cask occurs. The impact is at the edge of the water jacket.

The following table summarizes the results from the dynamic analyses (using the Working Model 2D computer code):

<b>HI-TRAC Handling Analysis – Working Model Analysis of Horizontal Drop</b>			
<b>Item</b>	<b>Value</b>	<b>Allowable</b>	<b>Safety Factor</b>
HI-TRAC 125 – Primary Impact Deceleration (g's)	32.66	45	1.38
HI-TRAC 125 – Secondary Impact Deceleration (g's)	26.73	45	1.68
HI-TRAC 100 – Primary Impact Deceleration (g's)	33.18	45	1.36
HI-TRAC 100 – Secondary Impact Deceleration (g's)	27.04	45	1.66
Axial Membrane Stress Due to HI-TRAC 125 Bending as a Beam - Level D Drop (psi)	19.06	39.75	2.085
Axial Membrane Stress Due to HI-TRAC 100 Bending as a Beam - Level D Drop (psi)	15.77	39.75	2.52

In the table above, the decelerations are measured at points corresponding to the base and top of the fuel assemblies contained inside the MPC. The dynamic drop analysis reported above, using the Working Model 2D rigid body-spring model proved that decelerations are below the design basis value and that global stresses were within allowable limits.

#### 3.4.9.2 DYNA3D Analysis of Drop Event

An independent evaluation of the drop event to delineate the effect of target non-linearity and the flexibility of the transfer cask has been performed using DYNA3D. Both the HI-TRAC 125 and HI-TRAC 100 transfer casks are modeled as part of the cask-pad-soil interaction finite element model set forth in NUREG/CR-6608 and validated by an NRC reviewed and approved Holtec topical report (see reference [3.A.4] in Appendix 3.A). The model uses the identical MPC and target pad/soil models employed in the accident analyses of the HI-STORM 100 overpack. The HI-TRAC inner and outer shells, the contained lead, the transfer lid, the water jacket metal structure, and the top lids are included in the model. The water jacket is assumed empty for conservatism.

Two side drop orientations are considered (see Figures 3.4.27 and 3.4.28). The first drop assumes that the plane of the lifting and rotation trunnions is horizontal with primary impact on the short side of the transfer lid. This maximizes the angle of slapdown, and represents a credible drop configuration where the HI-TRAC cask is dropped while being carried horizontally. The second drop orientation assumes primary impact on the rotation trunnion and maximizes the potential for the lifting trunnion to participate in the secondary impact. This is a non-credible event that assumes complete separation from the transfer vehicle and a ninety-degree rotation prior to impact. Nevertheless, it is the only configuration where the trunnions could be involved in both primary and secondary impacts.

For each simulation performed, the lowest point on the HI-TRAC cask (either the transfer lid edge or the rotation trunnion) is set at 42" above the target interface. Decelerations are measured at the top lid, the cask centroidal position, and the transfer lid. Normal forces were measured at the primary impact interface, at the secondary impact interface, and at the top lid/MPC interface. Decelerations are filtered at 350 Hz.

The following key results summarize the analyses:

ITEM	HI-TRAC 125		HI-TRAC 100		ALLOWABLE
	Horizontal	Vertical	Horizontal	Vertical	
Max. Top Lid Vertical Deceleration – Secondary Impact (g's)	25.5	32	36.5	45 <sup>†</sup>	45
Centroid Vertical Deceleration – at Time of Secondary Impact (g's)	9.0	13.0	10.0	17.5	45
Max. Transfer Lid Vertical Deceleration – Primary Impact (g's)	30.8	23.5	35.0	31.75	45
Maximum Normal Force at Primary Impact Site (kips)	1,950.	1,700	1,700	1,700	-
Maximum Normal Force at Secondary Impact Site (kips)	1,300.	1,850.	1,500.	1,450.	-
Maximum MPC/Top Lid Interface Force (kips)	132.	-	39.	-	-
Maximum Diametral Change of Inner Shell (inch)	0.228	0.113	Not Computed	0.067	0.3725
Maximum Von Mises Stress (ksi)	37.577	38.367	40.690	40.444	58.7*

<sup>†</sup> The deceleration at the top of the basket is estimated at 41 g's

\* Allowable Level D Stress Intensity for Primary Plus Secondary Stress Intensity

The results summarized above demonstrate that both the HI-TRAC 125 and HI-TRAC 100 transfer casks are sufficiently robust to perform their function during and after the postulated handling accidents. We also note that the results, using the Working Model single rigid body dynamic model (see Subsection 3.4.9.1), are in reasonable agreement with the results predicted by the DYNA3D multi-body finite element dynamic model although performed for a different drop height with deceleration measurements at different locations on the HI-TRAC.

The results reported above for maximum interface force at the top lid/MPC interface are used as input to a separate analysis, which demonstrates that the top lid contains the MPC during and after a handling accident. The results reported above for the maximum normal force at the primary impact site (the transfer lid) have been used to calculate the maximum interface force at the bottom flange/transfer lid interface. This result is needed to insure that the interface forces used to evaluate transfer lid separation are indeed bounding. To obtain the interface force between the HI-TRAC transfer lid and the HI-TRAC bottom flange, it is sufficient to take a free-body of the transfer lid and write the dynamic force equilibrium equation for the lid. Figure 3.4.29 shows the free body with appropriate notation. The equation of equilibrium is:

$$M_{TL} a_{TL} = F_I - G_I$$

where

$M_{TL}$  = the mass of the transfer lid

$a_{TL}$  = the time varying acceleration of the centroid of the transfer lid

$F_I$  = the time varying contact force at the interface with the target

$G_I$  = the time varying interface force at the bottom flange/transfer lid interface

Solving for the interface force give the result

$$G_I = F_I - M_{TL} a_{TL}$$

Using the appropriate transfer lid mass and acceleration, together with the target interface force at the limiting time instant, provides values for the interface force. The table below provides the results of this calculation for the HI-TRAC 125 and HI-TRAC 100 transfer casks.

Item	Calculated from Equilibrium (kips)
HI-TRAC 125 – Trunnions Horizontal	1,183.
HI-TRAC 125 – Trunnions Vertical	1,272.
HI-TRAC 100 – Trunnions Horizontal	1,129.
HI-TRAC 100 – Trunnions Vertical	1,070.

#### 3.4.9.3 Horizontal Drop of HI-TRAC 125D

The previous subsection addressed the 42” horizontal drop of the HI-TRAC 125 and HI-TRAC 100, including an evaluation of the bolted connection between the transfer lid, which sustains the primary impact, and the cylindrical body of the loaded HI-TRAC. The HI-TRAC 125D does not have a bolted connection between the bottom flange and the cylindrical body of the cask. However, the transverse protrusions (bottom flange, lifting trunnions, and optional attachment lugs/support tabs at the top of the cask) spawn different impact scenarios. The uncontrolled lowering of the cask is assumed to occur from a height of 42” measured to the lowest location on the HI-TRAC 125D in the horizontal orientation.

The maximum decelerations for the HI-TRAC 125D are comparable to the drop results for the HI-TRAC 125 when the plane of the lifting and rotation trunnions is vertical. Although the HI-TRAC 125D has no rotation trunnions, its bottom flange extends radially beyond the water jacket shell by approximately the same amount as the HI-TRAC 125 rotation trunnions and thereby establishes a similar “hard point” for primary impact in terms of distance from the cask centerline. More important, because the bottom flange is positioned closer to the base of the HI-TRAC 125D than



the rotation trunnions are in the HI-TRAC 125, the slap-down angle for the HI-TRAC 125D is less. The shallower angle decreases the participation of the lifting trunnion during the secondary impact, and increases the participation of the water jacket shell. Since the water jacket shell is a more flexible structure than the lifting trunnion, the deceleration of the HI-TRAC 125D cask during secondary impact is slightly less than the calculated deceleration of the HI-TRAC 125. In the HI-TRAC 125D, there is no bolted connection at the bottom flange/cask body interface that is active in load transfer from the flange to the cask body. It is therefore concluded that this drop scenario for the HI-TRAC 125D is bounded by the similar evaluation for the HI-TRAC 125. The same rationale applies to the HI-TRAC 100D versus the HI-TRAC 100. In fact, the protruding segments of the HI-TRAC 100D bottom flange are not in the same impact plane as the lifting trunnions; therefore, a secondary impact involving a lifting trunnion is not possible. Therefore, the drop scenarios analyzed for the HI-TRAC 100 bound any 42" horizontal drop of a HI-TRAC 100D.

A second HI-TRAC 125D drop scenario, where the two attachment lugs/support tabs are oriented in a vertical plane, is the most limiting scenario. This drop event is unique to HI-TRAC 125D serial numbers 3 and 4, since these are the only two transfer casks fabricated with attachment lugs/support tabs. The tab dimensions are such that primary impact occurs at the top end of the cask when the support tabs impact the target surface, followed by a slap-down and a secondary impact at the bottom flange. Note that this drop scenario does not exist for the HI-TRAC 100D since it has no attachment lugs/support tabs.

The evaluation of the limiting HI-TRAC 125D drop scenario is performed using the computer code Working Model 3D (WM) (now known as Visual Nastran Desktop). First, the WM code is used to simulate the "Scenario A" drop of the HI-TRAC 125 in order to establish appropriate parameters to "benchmark" WM against the DYNA3D solution. The table below summarizes the results of the Working Model/DYNA3D benchmark comparison. Figure 3.4.48 shows the benchmark configuration after the drop event.

Comparison of HI-TRAC 125 Drop Results (Scenario A)		
	DYNA3D	Working Model
Vertical Deceleration of Top Lid (secondary impact) g's	32	33.49
Vertical Deceleration at Bottom Lid (primary impact on rotation trunnion) g's	23.5	23.59

The benchmarked Working Model simulation was then modified to simulate the second drop scenario of the HI-TRAC 125D with support tabs in a vertical plane; primary impact now occurred at the top end with secondary impact at the bottom flange. Figure 3.4.49 shows the configuration of the HI-TRAC 125D after this scenario. The impact parameters were unchanged from the benchmark model except for location. The acceleration results from the 42" horizontal drop of the HI-TRAC 125D in this second drop scenario are summarized below.

Results From HI-TRAC 125D 42" Drop	
Vertical Deceleration of Top Lid (primary impact on support tab) g's	36.75

Vertical Deceleration of Pool Lid (secondary impact on bottom flange) g's	29.27
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The resulting g loads at the top of the active fuel region for the HI-TRAC 125D, with primary impact on the support tabs, are increased over the loads computed for the HI-TRAC 125 but remain well below the design basis limit.

Finally, stress calculations similar to those presented in Subsection 3.4.9.1 for the HI-TRAC 125 have also been performed for the HI-TRAC 125D using the above maximum decelerations. The table below summarizes the results:

Item	Value	Allowable	Safety Factor
Axial Membrane Stress Due to HI-TRAC 125D Bending as a Beam - Level D Drop (psi)	26.13	39.75	1.521
Shear Stress in Outer Shell Circumferential Weld Due to HI-TRAC 125D Bending as a Beam - Level D Drop (psi)	27.43	29.40	1.072

#### 3.4.10 HI-STORM 100 Non-Mechanistic Tip-over and Vertical Drop Event (Load Cases 02.a and 02.c in Table 3.1.5)

Pursuant to the provision in NUREG-1536, a non-mechanistic tip-over of a loaded HI-STORM 100 System on to the ISFSI pad is considered in this report. Analyses are also performed to determine the maximum deceleration sustained by a vertical free fall of a loaded HI-STORM 100 System from an 11" height onto the ISFSI pad. The objective of the analyses is to demonstrate that the plastic deformation in the fuel basket is sufficiently limited to permit the stored SNF to be retrieved by normal means, does not have an adverse effect on criticality safety, and that there is no significant loss of radiation shielding in the system.

Ready retrievability of the fuel is presumed to be ensured: if global stress levels in the MPC structure meet Level D stress limits during the postulated drop events; if any plastic deformations are localized; and if no significant permanent ovalization of the overpack into the MPC envelope space, remains after the event.

Subsequent to the accident events, the storage overpack must be shown to contain the shielding so that unacceptable radiation levels do not result from the accident.

Appendix 3.A provides a description of the dynamic finite element analyses undertaken to establish the decelerations resulting from the postulated event. A non-mechanistic tip-over is considered together with an end drop of a loaded HI-STORM 100 System. A dynamic finite element analysis of each event is performed using a commercial finite element code well suited for such dynamic analyses with interface impact and non-linear material behavior. This code and methodology have been fully benchmarked against Lawrence Livermore Laboratories test data and correlation [3.4.12].

The table below provides the values of computed peak decelerations at the top of the fuel basket for the vertical drop and the non-mechanistic tipover scenarios. It is seen that the peak deceleration is below 45 g's.

### Filtered Results for Drop and Tip-Over Scenarios for HI-STORM

Drop Event	Max. Deceleration at the Top of the Basket (g's)	
	Set A(36" Thick Pad)	Set B(28" Thick Pad)
End Drop for 11 Inches	43.98	41.53
Non-Mechanistic Tip-over	42.85	39.91

The tipover analysis performed in Appendix 3.A is based on the HI-STORM 100 geometry and a bounding weight. The fact that the HI-STORM 100S(232) is shorter and has a lower center of gravity suggests that the impact kinetic energy is reduced so that the target would absorb the energy with a lower maximum deceleration. However, since the actual weight of a HI-STORM 100S(232) is less than that of a HI-STORM 100 by a significant amount, the predicted maximum rigid body deceleration would tend to increase slightly. Since there are two competing mechanisms at work, it is not a foregone conclusion that the maximum rigid body deceleration level is, in fact, reduced if a HI-STORM 100S(232) suffers a non-mechanistic tipover onto the identical target as the HI-STORM 100. The situation is clearer for the HI-STORM 100S(243), which is virtually equal in weight to the HI-STORM 100, yet its center of gravity when loaded is almost one inch lower. In what follows, we present a summary of the analysis undertaken to demonstrate conclusively that the result for maximum deceleration level in the HI-STORM 100 tipover event does bound the corresponding value for the HI-STORM 100S(232), and, therefore, we need only perform a detailed dynamic finite element analysis for the HI-STORM 100.

Appendix 3.A presents a result for the angular velocity of the cylindrical body representing a HI-STORM 100 just prior to impact with the defined target. The result is expressed in Subsection 3.A.6 in terms of the cask geometry, and the ratio of the mass divided by the mass moment of inertia about the corner point that serves as the rotation origin. Since the mass moment of inertia is also linearly related to the mass, the angular velocity at the instant just prior to target contact is independent of the cask mass. Subsequent to target impact, we investigate post-impact response by considering the cask as a cylinder rotating into a target that provides a resistance force that varies linearly with distance from the rotation point. We measure "time" as starting at the instant of impact, and develop a one-degree-of freedom equation for the post-impact response (for the rotation angle into the target) as:

$$\ddot{\theta} + \omega^2 \theta = 0$$

where

$$\omega^2 = \frac{kL^3}{3I_A}$$

The initial conditions at time=0 are: the initial angle is zero and the initial angular velocity is equal to the rigid body angular velocity acquired by the tipover from the center-of-gravity over corner position. In the above relation, L is the length of the overpack, I is the mass moment of inertia defined in Appendix 3.A, and k is a “spring constant” associated with the target resistance. If we solve for the maximum angular acceleration subsequent to time = 0, we obtain the result in terms of the initial angular velocity as:

$$\ddot{\theta}_{\max} = \omega \dot{\theta}_0$$

If we form the maximum linear acceleration at the top of the four-inch thick lid of the overpack, we can finally relate the decelerations of the HI-STORM 100 and the HI-STORM 100S(232) solely in terms of their geometry properties and their mass ratio. The value of “k”, the target spring rate is the same for both overpacks so it does not appear in the relationship between the two decelerations. After substituting the appropriate geometry and calculated masses, we determine that the ratio of maximum rigid body decelerations at the top surface of the four-inch thick top lid plates is:

$$A_{\text{HI-STORM 100S(232)}}/A_{\text{HI-STORM 100}} = 0.946$$

Therefore, as postulated, there is no need to perform a separate DYNA3D analysis for the HI-STORM 100S hypothetical tipover.

Moreover, according to Appendix 3.A, analysis of a single mass impacting a spring with a given initial velocity shows that the maximum deceleration “ $a_M$ ” of the mass is related to the dropped weight “w” and the drop height “h” as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}$$

In other words, as the dropped weight increases, the maximum deceleration of the mass decreases. Therefore, the rigid body decelerations calculated in Appendix 3.A serve as a conservative upper bound for the densified concrete shielding option.

The same considerations apply to the HI-STORM 100S Version B. The overall lengths are reduced from the classic HI-STORM 100, but the actual weights may be reduced. Therefore, calculations similar to those given above for the HI-STORM 100S are needed to conclusively demonstrate that the non-mechanistic tipover analysis of the classic HI-STORM 100 remains bounding. The results of the calculations, which demonstrate that the design basis limits are met, are presented below together with maximum G levels computed for the 11” vertical drop:

ITEM	A HI-STORM 100S VERSION B(218)/A HI-STORM 100	A HI-STORM 100S VERSION B(229)/A HI-STORM 100	Max. Calculated G Level 11" Drop
HI-STORM 100S Version B(218)	0.91	-	44.378
HI-STORM 100S Version B(229)	-	0.98	43.837

As discussed in Chapter 1, the channels attached to the inner shell of the HI-STORM 100S and HI-STORM 100 S Version B overpack may be replaced with cylindrical guide tubes as shown in the drawings in Section 1.5. A separate LS-DYNA analysis documented in Attachment E of [3.A.7] demonstrates that the top of the MPC fuel basket in the HI-STORM 100 S Version B with cylindrical guide tubes will not experience a g-load higher than the design limit for the non-mechanistic tipover. The HI-STORM 100S Version B analysis bounds the conditions of the HI-STORM 100S non-mechanistic tip-over case.

A simple elastic strength of materials calculation is performed to demonstrate that the cylindrical storage overpack will not permanently deform to the extent that the MPC cannot be removed by normal means after a tip-over event. The results demonstrate that the maximum diametrical closure of the cylindrical cavity is less than the initial clearance between the overpack MPC support channels and the MPC canister. Primary circumferential membrane stresses in the MPC shell remain in the elastic range during a tip-over (see Table 3.4.6 summary safety factors); therefore, no permanent global ovalization of the MPC shell occurs as a result of the drop.

To demonstrate that the shielding material will continue to perform its function after a tip-over accident, the stress and strain levels in the metal components of the storage overpack are examined at the end of the tip-over event. The results obtained in Appendix 3.A for impact decelerations conservatively assumed a rigid storage overpack model to concentrate nearly all energy loss in the target. However, to assess the state of stress and strain in the storage overpack after an accident causing a tip-over, the tip-over analysis was also performed using a non-rigid storage overpack model using overpack material properties listed in Appendix 3.A. Figure 3.4.13 shows the calculated von Mises stress in the top lid and outer shell at 0.08 seconds after the initiation of impact. Figure 3.4.14 shows the residual plastic strains in the same components. Figures 3.4.15 and 3.4.16 provide similar results for the inner shell, the radial plates, and the support channels<sup>†</sup>. The results show that while some plastic straining occurs, accompanied by stress levels above the yield stress of the material, there is no tearing in the metal structure which confines the radiation shielding (concrete). Therefore, there is no gross failure of the metal shells enclosing the concrete. The shielding concrete will remain inside the confines of the storage overpack and maintain its performance after the tipover event. Although the preceding results are based on an overpack model having inner and outer shell thicknesses of 1-1/4" and 3/4", respectively, the conclusion holds for the optional HI-STORM design with 1" thick inner and outer shells since having a thicker steel shell at the primary point of impact provides more strength and greater protection against a cavity breach. The results from these analyses are also applicable to the HI-STORM 100S and the

<sup>†</sup> During fabrication the channels are attached to the inner shell by one of two methods, either the channels are welded directly to the inner shell or they are welded to a pair of L-shaped angles (i.e., channel mounts) that are pre-fastened to the inner shell. The results presented in Figures 3.4.16a and 3.4.16b bound the results from both methods of attachment.

HI-STORM 100S, Version B since the structural material at the top of the cask that would be locally deformed after a tipover event is essentially the same.

### 3.4.11 Storage Overpack and HI-TRAC Transfer Cask Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years; therefore, the License Life (please see glossary) of all components is 20 years. Nonetheless, the HI-STORM 100 and 100S Storage overpacks and the HI-TRAC transfer cask are engineered for 40 years of design life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. In addition, the storage overpack and HI-TRAC are designed, fabricated, and inspected under the comprehensive Quality Assurance Program discussed in Chapter 13 and in accordance with the applicable requirements of the ACI and ASME Codes. This assures high design margins, high quality fabrication, and verification of compliance through rigorous inspection and testing, as describe in Chapter 9 and the design drawings in Section 1.5. Technical Specifications defined in Chapter 12 assure that the integrity of the cask and the contained MPC are maintained throughout the components' design life. The design life of a component, as defined in the Glossary, is the minimum duration for which the equipment or system is engineered to perform its intended function if operated and maintained in accordance with the FSAR. The design life is essentially the lower bound value of the service life, which is the expected functioning life of the component or system. Therefore, component longevity should be: licensed life < design life < service life. (The licensed life, enunciated by the USNRC, is the most pessimistic estimate of a component's life span.) For purposes of further discussion, we principally focus on the service life of the HI-STORM 100 System components that, as stated earlier, is the reasonable expectation of equipment's functioning life span.

The service life of the storage overpack and HI-TRAC transfer cask is further discussed in the following sections.

#### 3.4.11.1 Storage Overpack

The principal design considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

##### Exposure to Environmental Effects

In the following text, all references to HI-STORM 100 also apply to HI-STORM 100S and to the HI-STORM 100S Version B. All exposed surfaces of HI-STORM 100 are made from ferritic steels that are readily painted. Concrete, which serves strictly as a shielding material, is completely encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete, are ruled out for HI-STORM 100. Under normal storage conditions, the bulk temperature of the HI-STORM 100 storage overpack will, because of its large thermal inertia, change very gradually with time. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM 100 storage overpack. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM 100 because HI-STORM 100 does not rely on rebars (indeed, it contains no rebars). As discussed in Appendix 1.D, the aggregates, cement and water used in the storage cask concrete are

carefully controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation. In addition, the storage overpack is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the 40-year design life of the storage overpack as defined in Subsection 2.2.3 and evaluated in Chapter 11.

### Material Degradation

The relatively low neutron flux to which the storage overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The controlled environment of the ISFSI storage pad mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage overpack throughout the 40-year design life are defined in Chapter 9. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STORM 100 System is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [3.4.11], which concluded that dry storage systems designed, fabricated, inspected, and operate in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

#### 3.4.11.2 Transfer Cask

The principal design considerations that bear on the adequacy of the HI-TRAC Transfer Cask for the service life are addressed as follows:

### Exposure to Environmental Effects

All transfer cask materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The HI-TRAC is designed for repeated normal condition handling operations with high factor of safety, particularly for the lifting trunnions, to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the trunnion material, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading, as discussed in Chapter 12.

### Material Degradation

All transfer cask materials that are susceptible to corrosion are coated. The controlled environment in which the HI-TRAC is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the HI-TRAC materials are subjected do not result in radiation embrittlement or degradation of the HI-TRAC's shielding materials that could impair the HI-TRAC's intended safety function. The HI-TRAC transfer cask materials are selected for durability and wear resistance for their deployment.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the HI-TRAC transfer cask throughout the 40-year design life are defined in Chapter 9. These requirements include provisions for routine inspection of the HI-TRAC transfer cask for damage prior to each use, including an annual inspection of the lifting trunnions. Precautions are taken during lid handling operations to protect the sealing surfaces of the pool lid. The leak tightness of the liquid neutron shield is verified periodically. The water jacket pressure relief valves and other fittings used can be easily removed.

#### 3.4.12 MPC Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC (i.e., licensed life) is 20 years. Nonetheless, the HI-STORM 100 MPC is designed for 40 years of design life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. Additional assurance of the integrity of the MPC and the contained SNF assemblies throughout the 40-year life of the MPC is provided through the following:

- Design, fabrication, and inspection in accordance with the applicable requirements of the ASME Code as described in Chapter 2 assures high design margins.
- Fabrication and inspection performed in accordance with the comprehensive Quality Assurance program discussed in Chapter 13 assures competent compliance with the fabrication requirements.
- Use of materials with known characteristics, verified through rigorous inspection and testing, as described in Chapter 9, assures component compliance with design requirements.
- Use of welding procedures in full compliance with Section III of the ASME Code ensures high-quality weld joints.

Technical Specifications, as defined in Chapter 12, have been developed and imposed on the MPC that assure that the integrity of the MPC and the contained SNF assemblies are maintained throughout the 40-year design life of the MPC.

The principal design considerations bearing on the adequacy of the MPC for the service life are



summarized below.

### Corrosion

All MPC materials are fabricated from corrosion-resistant austenitic stainless steel and passivated aluminum. The corrosion-resistant characteristics of such materials for dry SNF storage canister applications, as well as the protection offered by these materials against other material degradation effects, are well established in the nuclear industry. The moisture in the MPC is removed to eliminate all oxidizing liquids and gases and the MPC cavity is backfilled with dry inert helium at the time of closure to maintain an atmosphere in the MPC that provides corrosion protection for the SNF cladding throughout the dry storage period. The preservation of this non-corrosive atmosphere is assured by the inherent sealworthiness of the MPC confinement boundary integrity (there are no gasketed joints in the MPC).

### Structural Fatigue

The passive non-cyclic nature of dry storage conditions does not subject the MPC to conditions that might lead to structural fatigue failure. Ambient temperature and insolation cycling during normal dry storage conditions and the resulting fluctuations in MPC thermal gradients and internal pressure is the only mechanism for fatigue. These low-stress, high-cycle conditions cannot lead to a fatigue failure of the MPC that is made from stainless alloy stock (endurance limit well in excess of 20,000 psi). All other off-normal or postulated accident conditions are infrequent or one-time occurrences, which cannot produce fatigue failures. Finally, the MPC uses materials that are not susceptible to brittle fracture.

### Maintenance of Helium Atmosphere

The inert helium atmosphere in the MPC provides a non-oxidizing environment for the SNF cladding to assure its integrity during long-term storage. The preservation of the helium atmosphere in the MPC is assured by the robust design of the MPC confinement boundary described in Section 7.1. Maintaining an inert environment in the MPC mitigates conditions that might otherwise lead to SNF cladding failures. The required mass quantity of helium backfilled into the canister at the time of closure and the associated fabrication and closure requirements for the canister are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout the 40-year design life.

### Allowable Fuel Cladding Temperatures

The helium atmosphere in the MPC promotes heat removal and thus reduces SNF cladding temperatures during dry storage. In addition, the SNF decay heat will substantially attenuate over a 40-year dry storage period. Maintaining the fuel cladding temperatures below allowable levels during long-term dry storage mitigates the damage mechanism that might otherwise lead to SNF cladding failures. The allowable long-term SNF cladding temperatures used for thermal acceptance of the MPC design are conservatively determined, as discussed in Section 4.3.

### Neutron Absorber Boron Depletion

The effectiveness of the fixed borated neutron absorbing material used in the MPC fuel basket design requires that sufficient concentrations of boron be present to assure criticality safety during worst case design basis conditions over the 40-year design life of the MPC. Information on the characteristics of the borated neutron absorbing material used in the MPC fuel basket is provided in Subsection 1.2.1.3.1. The relatively low neutron flux, which will continue to decay over time, to which this borated material is subjected, does not result in significant depletion of the material's available boron to perform its intended safety function. In addition, the boron content of the material used in the criticality safety analysis is conservatively based on the minimum specified boron areal density (rather than the nominal), which is further reduced by 25% for analysis purposes, as described in Section 6.1. Analysis discussed in Section 6.3.2 demonstrates that the boron depletion in the neutron absorber material is negligible over a 50-year duration. Thus, sufficient levels of boron are present in the fuel basket neutron absorbing material to maintain criticality safety functions over the 40-year design life of the MPC.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review, which concluded that dry storage systems designed, fabricated, inspected, and operated in the manner of the requirements set down in this document are adequate for a 100-year service life, while satisfying the requirements of 10CFR72.

#### 3.4.13 Design and Service Life

The discussion in the preceding sections seeks to provide the logical underpinnings for setting the design life of the storage overpacks, the HI-TRAC transfer cask, and the MPCs as forty years. Design life, as stated earlier, is a lower bound value for the expected performance life of a component (service life). If operated and maintained in accordance with this Final Safety Analysis Report, Holtec International expects the service life of its HI-STORM 100 and HI-STORM 100S Version's components to substantially exceed their design life values.

Table 3.4.1

## FINITE ELEMENTS IN THE MPC STRUCTURAL MODELS

MPC Type	Model Type			
	Element Type	Basic	0 Degree Drop	45 Degree Drop
<b>MPC-24</b>		1068	1114	1113
BEAM3		1028	1028	1028
PLANE82		0	0	0
CONTAC12		40	38	38
CONTAC26		0	45	45
COMBIN14		0	3	2
<b>MPC-32</b>		1374	1604	1603
BEAM3		1346	1346	1346
CONTAC12		28	27	24
CONTAC26		0	229	228
COMBIN14		0	2	5
<b>MPC-68</b>		1842	2066	2063
BEAM3		1782	1782	1782
PLANE82		16	16	16
CONTAC12		44	43	40
CONTAC26		0	223	222
COMBIN14		0	2	3
<b>MPC-24E</b>		1070	1124	1122
BEAM3		1030	1030	1030
PLANE82		0	0	0
CONTAC12		40	38	38
CONTAC26		0	53	52
COMBIN14		0	3	2

**TABLE 3.4.2**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

Material/Component	Fuel Pool (Borated and Unborated Water) <sup>†</sup>	ISFSI Pad (Open to Environment)
<u>Alloy X:</u>  - MPC Fuel Basket - MPC Baseplate - MPC Shell - MPC Lid - MPC Fuel Spacers	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions or interactions with spent fuel.	The MPC internal environment will be an inert (helium) atmosphere and the external surface will be exposed to ambient air. No adverse interactions identified.
<u>Aluminum:</u>  - Heat Conduction Elements	Aluminum and stainless steel form a galvanic couple. However, aluminum will be used in a passivated state. Upon passivation, aluminum forms a thin ceramic (Al <sub>2</sub> O <sub>3</sub> ) barrier. Therefore, during the short time they are exposed to pool water, significant corrosion of aluminum or production of hydrogen is not expected (see operational requirements under “Neutron Absorber Material” below).	In a non-aqueous atmosphere, galvanic corrosion is not expected.
<u>Neutron Absorber Material:</u>	Extensive in-pool experience on spent fuel racks with no adverse reactions. See Chapter 8 for additional requirements for combustible gas monitoring and required actions for control of combustible gas accumulation under the MPC lid.	No adverse potential reactions identified.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

<b>Material/Component</b>	<b>Fuel Pool (Borated and Unborated Water)<sup>†</sup></b>	<b>ISFSI Pad (Open to Environment)</b>
<u>Steels:</u> - SA350-LF2 - SA350-LF3 - SA203-E - SA515 Grade 70 - SA516 Grade 70 - SA193 Grade B7 - SA106 (HI-TRAC)	All exposed steel surfaces (except seal areas, and pocket trunnions) will be coated with paint specifically selected for performance in the operating environments. Even without coating, no adverse reactions (other than nominal corrosion) have been identified. Lid bolts are plated and the threaded portion of the bolt anchor blocks is coated to seal the threaded area.	Internal surfaces of the HI-TRAC will be painted and maintained. Exposed external surfaces (except those listed in fuel pool column) will be painted and will be maintained with a fully painted surface. No adverse reactions identified.
<u>Steels:</u> - SA516 Grade 70 - SA203-E - SA350-LF3 - A36 Storage Overpack	HI-STORM 100 storage overpack is not exposed to fuel pool environment.	Internal and external surfaces will be painted (except for bolt locations that will have protective coating). External surfaces will be maintained with a fully painted surface. No adverse reaction identified.
<u>Stainless Steels:</u> - SA240 304 - SA193 Grade B8 - 18-8 S/S  Miscellaneous Components	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions.	Stainless steel has a long proven history of corrosion resistance when exposed to the atmosphere. These materials are used for bolts and threaded inserts. No adverse reactions with steel have been identified. No impact on performance.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

<b>Material/Component</b>	<b>Fuel Pool (Borated and Unborated Water)<sup>†</sup></b>	<b>ISFSI Pad (Open to Environment)</b>
<u>Nickel Alloy:</u> - SB637-NO7718 - SA564-630 H1100 (for HI-TRAC 125D only)  Lifting Trunnion	No adverse reactions with borated or unborated water.	Exposed to weathering effects. No adverse reactions with storage overpack closure plate. No impact on performance.
<u>Brass/Bronze:</u> - Pressure Relief Valve HI-TRAC	Small surface of pressure relief valve will be exposed. No significant adverse impact identified.	Exposed to external weathering. No loss of function expected.
<u>Holtite-A:</u> - Solid Neutron Shield	The neutron shield is fully enclosed. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.	The neutron shield is fully enclosed in the outer enclosure. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

<b>Material/Component</b>	<b>Fuel Pool (Borated and Unborated Water)<sup>†</sup></b>	<b>ISFSI Pad (Open to Environment)</b>
<u>Paint:</u> - as per Appendix 1.C	Paint used for the HI-TRAC exterior surface has acceptable performance for short-term exposure in mild borated pool water.  Paint selected for HI-TRAC internal surfaces has excellent high temperature resistance properties. Will only be exposed to demineralized water during in-pool operations as annulus is filled prior to placement in the spent fuel pool and the inflatable seal prevents fuel pool water in-leakage. No adverse interaction identified which could affect MPC/fuel assembly performance.	Good performance on surfaces. Discoloration is not a concern.
<u>Elastomer Seals:</u>	No adverse reactions identified.	Only used during fuel pool operations.
<u>Lead:</u>	Enclosed by carbon steel. Lead is not exposed to fuel pool water. Lead has no interaction with carbon steel.	Enclosed by carbon steel. Lead is not exposed to ambient environment. Lead has no interaction with carbon steel.
<u>Concrete:</u>	Storage overpack is not exposed to fuel pool water.	Concrete is enclosed by carbon steel and not exposed to ambient environment. Concrete has no interaction with carbon steel.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.3  
FUEL BASKET RESULTS - MINIMUM SAFETY FACTORS**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Safety Factor</b>	<b>Location in FSAR</b>
F1	T, T'	No interference	Subsection 3.4.4.2
F2	D + H	2.87	Table 3.4.9 of Docket 72-1008
F3			
F3.a	D + H' (end drop)	3.6	3.4.4.3.1.3
F3.b	D + H' (side drop 0 deg.)	1.27	Table 3.4.6
F3.c	D + H' (side drop 45 deg.)	1.23	Table 3.4.6

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.



**TABLE 3.4.4**  
**MPC RESULTS - MINIMUM SAFETY FACTORS**

Load Case I.D.	Load Combination <sup>†,††</sup>	Safety Factor	Location in FSAR Where the Analysis is Performed
E1			
E1.a	Design internal pressure, $P_i$	4.30 <sup>2†††</sup> 1.326 1.20 N/A	E1.a Lid Table 3.4.7 Baseplate 3.I.8.1 of Docket 72-1008 Shell Table 3.4.7 Supports
E1.b	Design external pressure, $P_o$	4.30 <sup>†††</sup> 1.326 23.3  N/A	E1.b Lid $P_i$ bounds Baseplate $P_i$ bounds Shell 3.4.4.3.1.7 (buckling methodology in 3.H of Docket 72-1008)  Supports
E1.c	Short-term internal pressure, $P_{st}$ , plus Temperature T	1.04	E1.c Shell Table 3.4.8
E2	D + H + ( $P_i$ , $P_{st}$ , $P_o$ )	1.67 <sup>†††1</sup> 1.02 2.29 45.5  5.84	Lid Methodology per 3.E.8.1.2 of Docket 72-1008 Baseplate 3.4.3.6 Shell Primary membrane stress per 3.4.4.3.1.2 Buckling methodology in 3.H of Docket 72-1008  Supports Table 3.4.9 of Docket 72-1008

<sup>1</sup> The calculation from 3.E.8.1.2 is updated using the bounding short-term normal pressure,  $P_{st}$ , in Supplement 67 of [3.4.6].

† The symbols used for the loadings are defined in Table 2.2.13

†† Note that in analyses, bounding pressures are applied, i.e., in buckling calculations  $P_o$  is used, and in stress evaluations either  $P_o$  or  $P_i$  or  $P_{st}$  is appropriate

††† Minimum safety factor is based on the dual lid configuration.

**TABLE 3.4.4 (CONTINUED)**  
**MPC RESULTS - MINIMUM SAFETY FACTORS**

Load Case I.D.	Load Combination <sup>†,††</sup>	Safety Factor	Location in FSAR
E3 E3.a	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', end drop	1.4 <sup>†††</sup>	E3.a Lid 3.E.8.2.1.2 of Docket 72-1008 Baseplate 3.I.8.3 of Docket 72-1008 Shell Buckling methodology in 3.H of Docket 72-1008 Supports
		1.87	
		1.72	
		N/A	
E3.b	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', side drop 0 deg.	1.4 <sup>†††</sup>	E3.b Lid end drop bounds Baseplate end drop bounds Shell Table 3.4.6 Supports Table 3.4.6
		1.87	
		1.02	
		1.13	
E3.c	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', side drop 45 deg.	1.4 <sup>†††</sup>	E3.c Lid end drop bounds Baseplate end drop bounds Shell Table 3.4.6 Supports Table 3.4.6
		1.87	
		1.38	
		1.50	

† The symbols used for the loadings are defined in Table 2.2.13

†† Note that in analyses, bounding pressures are applied, i.e., in buckling calculations P<sub>o</sub> is used, and in stress evaluations either P<sub>o</sub> or P<sub>i</sub> is appropriate

††† Minimum safety factor is based on the dual lid configuration.

**TABLE 3.4.4 (CONTINUED)**  
**MPC RESULTS - MINIMUM SAFETY FACTORS**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†, ††</sup></b>	<b>Safety Factor</b>	<b>Location in FSAR</b>
E4	T	Subsection 3.4.4.2 shows there are no primary stresses from thermal expansion.	Subsection 3.4.4.2
E5	D + T* + (P <sub>i</sub> *, P <sub>o</sub> *)	8.6 <sup>†††</sup> 1.17 1.15 (buckling)  2.60 (stress)  N/A	Lid            3.4.4.3.1.10 Baseplate    3.4.4.3.1.10 Shell         Buckling methodology in 3.H of Docket 72-1008 3.4.4.3.1.2 (stress)  Supports     N/A

† The symbols used for the loadings are defined in Table 2.2.13.

†† Note that in analyses, bounding pressures are applied, i.e., in buckling calculations P<sub>o</sub> is used, and in stress evaluations either P<sub>o</sub> or P<sub>i</sub> is appropriate.

††† Minimum safety factor is based on the dual lid configuration.

**TABLE 3.4.5**  
**HI-STORM 100 STORAGE OVERPACK AND HI-TRAC RESULTS - MINIMUM SAFETY FACTORS**

Load Case I.D.	Loading <sup>†</sup>	Safety Factor	Location in FSAR
01	D + H + T + (P <sub>o</sub> , P <sub>i</sub> )	1.33	Overpack
		N/A	Shell (inlet vent)/Base 3.4.3.5 Top Lid N/A
02	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (end drop/tip-over)	2.69(125); 2.17(100)	HI-TRAC
		2.604 (ASME Code limit) 2.61 (ASME Code limit) 2.91; 1.11(optional bolts)	Shell 3.4.3.3; 3.4.3.4 Pool Lid 3.4.3.8 Top Lid N/A Pocket Trunnion 3.4.4.3.3.1
02.a	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (end drop/tip-over)	1.271	Overpack
		1.125	Shell 3.4.4.3.2.3 Top Lid 3.4.4.3.2.2
02.b	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (side drop)	1.52	HI-TRAC
		1.159 1.651	Shell 3.4.9.3 Transfer Lid 3.4.4.3.3.3 Top Lid 3.4.4.3.3.5
03	D (water jacket)	1.39	3.4.4.3.3.4
04	M (small and medium penetrant missiles)	2.60 (Side Strike); 1.065(End strike)	Overpack 3.4.8.1
		1.23 (End Strike)	HI-TRAC 3.4.8.2.1

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.4.6**  
**MINIMUM SAFETY FACTORS FOR MPC COMPONENTS DURING TIP-OVER**  
**45g DECELERATIONS**

Component - Stress Result	MPC-24		MPC-68	
	0 Degrees	45 Degrees	0 Degrees	45 Degrees
Fuel Basket - Primary Membrane ( $P_m$ )	3.38*0.987 (1134)	4.72*0.987 (396)	2.89*0.987 (1603)	4.18*0.987 (1603)
Fuel Basket - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	1.29*0.987 (1065)	1.30*0.987 (577)	1.97*0.987 (1590)	1.35*0.987 (1459)
Enclosure Vessel - Primary Membrane ( $P_m$ )	6.39*.967 (1354)	6.46*.967 (1370)	6.34 (2393)	6.64 (2377)
Enclosure Vessel - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	2.46*.967 (1278)	2.92*.967 (1247)	1.02 (1925)	1.45 (1925)
Basket Supports - Primary Membrane ( $P_m$ )	N/A	N/A	6.61 (1710)	8.61 (1699)
Basket Supports - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	N/A	N/A	1.09 (1715)	1.43 (1704)

## Notes:

1. Corresponding ANSYS element number shown in parentheses.
2. Multiplier of 0.967 reflects increase in Enclosure Vessel Design Temperature from 450 deg. F to 500 deg. F (Table 2.2.3); tabulated results for MPC-68 are based on higher temperature (500 deg. F) for Enclosure Vessel
3. Safety factors for the MPC-24 have been reduced (divided by factor of 1.024) to adjust for the fuel assembly weight increase (see Subsection 3.4.4.4.1)
4. Multiplier of 0.987 reflects increase in Fuel Basket Design Temperature from 725 deg. F to 752 deg. F (Table 2.2.3).

**TABLE 3.4.6 (CONTINUED)**  
**MINIMUM SAFETY FACTORS FOR MPC COMPONENTS DURING TIP-OVER**  
**45g DECELERATIONS**

Component - Stress Result	MPC-32	
	0 Degrees	45 Degrees
Fuel Basket - Primary Membrane (P <sub>m</sub> )	3.43*0.987 (715)	4.84*0.987 (366)
Fuel Basket - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	1.47*0.987 (390)	1.25*0.987 (19)
Enclosure Vessel - Primary Membrane (P <sub>m</sub> )	4.01*.967 (1091)	5.46*.967 (1222)
Enclosure Vessel - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	1.08*.967 (1031)	1.43*.967 (1288)
Basket Supports - Primary Membrane (P <sub>m</sub> )	3.36 (905)	4.74 (905)
Basket Supports - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	1.27 (901)	1.67 (908)

## Notes:

1. Corresponding ANSYS element number shown in parentheses.
2. Multiplier of 0.967 reflects increase in Enclosure Vessel Design Temperature from 450 deg. F to 500 deg. F (Table 2.2.3).
3. Safety factors for the MPC-32 has been reduced (divided by factor of 1.024) to adjust for the fuel assembly weight increase (see Subsection 3.4.4.4.1)
4. Multiplier of 0.987 reflects increase in Fuel Basket Design Temperature from 725 deg. F to 752 deg. F (Table 2.2.3).

**TABLE 3.4.6 (CONTINUED)**  
**MINIMUM SAFETY FACTORS FOR MPC-24E COMPONENTS DURING TIP-OVER**  
**45g DECELERATIONS**

<b>Components – Stress Result</b>	<b>0 Degrees</b>	<b>45 Degrees</b>
Fuel Basket – Primary Membrane ( $P_m$ )	-10,554 (3.50*0.987)	-7,608 (4.86*0.987)
Fuel Basket – Primary Membrane plus Primary Bending ( $P_L + P_b$ )	38,029 (1.46*0.987)	32,745 (1.70*0.987)
Enclosure Vessel – Primary Membrane ( $P_m$ )	6,611 (6.57*.967)	6,612 (6.57*.967)
Enclosure Vessel – Primary Membrane plus Primary Bending ( $P_L + P_b$ )	23,680 (2.75*.967)	16,868 (3.87*.967)

## Notes:

1. All stresses are reported in psi units and are based on closed gaps (primary stresses only).
2. The numbers shown in parentheses are the corresponding safety factors.
3. Multiplier of 0.967 reflects the increase in Enclosure Vessel Design Temperature from 450 deg. F to 500 deg. F (Table 2.2.3).
4. Stress results/safety factors for the MPC-24E have been multiplied/divided by a factor of 1.024 to adjust for the fuel assembly weight (See subsection 3.4.4.4.1).
5. Multiplier of 0.987 reflects increase in Fuel Basket Design Temperature from 725 deg. F to 752 deg. F (Table 2.2.3).

**TABLE 3.4.7**  
**STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY -**  
**INTERNAL PRESSURE ONLY**

Locations (Per Fig. 3.4.11)	Calculated Value of Stress Intensity (psi)	Category	Table 3.1.13 Allowable Value (psi) <sup>†</sup>	Safety Factor (Allowable/Calculated)
<u>Top Lid</u> <sup>††</sup>				
A	1,633	$P_L + P_b$	25,450	15.6
Neutral Axis	21.9	$P_m$	16,950	774
B	1,604	$P_L + P_b$	25,450	15.9
C	695	$P_L + P_b$	25,450	36.6
Neutral Axis	732	$P_m$	16,950	23.2
D	2,962	$P_L + P_b$	25,450	8.59
<u>Baseplate</u>				
E	19,773	$P_L + P_b$	28,100	1.42
Neutral Axis	415	$P_m$	18,700	45.1
F	20,601	$P_L + P_b$	28,100	1.36
G	9,610	$P_L + P_b$	28,100	2.92
Neutral Axis	2,268	$P_m$	18,700	8.25
H	8,279	$P_L + P_b$	28,100	3.39

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid), 400 degrees F (baseplate), and 500 degrees F (canister).

<sup>††</sup> Stresses for the top lid are reported for the single lid configuration; a doubling of the calculated stress values (and a halving of the top lid safety factors) results when the dual lid configuration is considered.



**TABLE 3.4.7 (CONTINUED)**  
**STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY -**  
**INTERNAL PRESSURE ONLY**

<b>Locations (Per Fig. 3.4.11)</b>	<b>Calculated Value of Stress Intensity (psi)</b>	<b>Category</b>	<b>Table 3.1.13 Allowable Value (psi)<sup>†</sup></b>	<b>Safety Factor (Allowable/Calculated)</b>
<u>Canister</u>				
I	6,788	$P_m$	17,500	2.58
Upper Bending Boundary Layer Region	7,202 7,014	$P_L + P_b + Q$ $P_L$	52,500 26,300	7.29 3.75
Lower Bending Boundary Layer Region	43,645 11,349	$P_L + P_b + Q$ $P_L$	52,500 26,300	1.20 2.32

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid), 400 degrees F (baseplate), and 500 degrees F (canister).

**TABLE 3.4.8**  
**STRESS INTENSITY RESULTS FOR**  
**CONFINEMENT BOUNDARY – SHORT-TERM NORMAL PRESSURE LIMIT PLUS**  
**THERMAL LOADING**

Locations	Calculated Value of Stress Intensity (psi)	Category	Allowable Stress Intensity (psi) <sup>†</sup>	Safety Factor (Allowable/Calculated)
<u>Top Lid</u> <sup>††</sup>				
A	6,418	$P_L + P_b + Q$	50,850	7.92
Neutral Axis	4,696	$P_m + P_L$	25,450	5.42
B	1,572	$P_L + P_b + Q$	50,850	32.34
C	10,093	$P_L + P_b + Q$	50,850	5.04
Neutral Axis	9,056	$P_m + P_L$	25,450	2.81
D	8,561	$P_L + P_b + Q$	50,850	5.94
<u>Baseplate</u>				
E	22,547	$P_L + P_b + Q$	56,100	2.49
Neutral Axis	604.8	$P_m + P_L$	28,100	46.46
F	21,372	$P_L + P_b + Q$	56,100	2.62
G	8,844	$P_m + P_L + Q$	56,100	6.34
Neutral Axis	3,784	$P_m + P_L$	28,100	7.43
H	11,331	$P_L + P_b + Q$	56,100	4.95
<u>Canister</u>				
I	7,569	$P_L$	26,300	3.47
Upper Bending	8,990	$P_L + P_b + Q$	52,500	5.84
Boundary	8,693	$P_L$	26,300	3.03
Layer Region				
Lower Bending	50,276	$P_L + P_b + Q$	52,500	1.04
Boundary Layer	11,777	$P_L$	26,300	2.23
Region				

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid), 400 degrees F (baseplate), and 500 degrees F (canister).

<sup>††</sup> Stresses for the top lid are reported for the single lid configuration; a doubling of the calculated stress values (and a halving of the top lid safety factors) results when the dual lid configuration is considered.

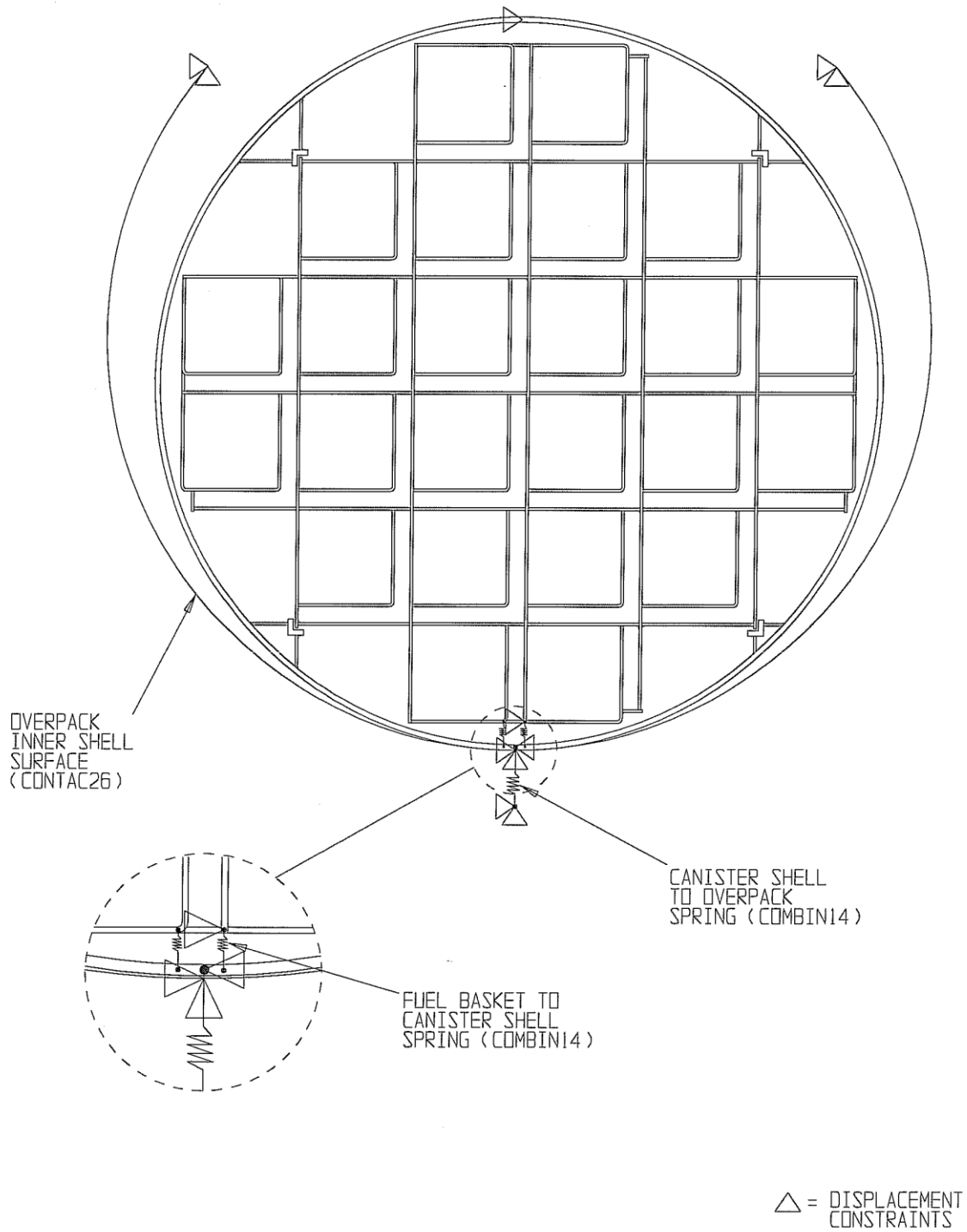
**TABLE 3.4.9**  
**SAFETY FACTORS FROM SUPPLEMENTARY CALCULATIONS**

<b>Item</b>	<b>Loading</b>	<b>Safety Factor</b>	<b>FSAR Location Where Details Are Provided</b>
HI-STORM Top Lid Weld Shear	Tipover	3.22	3.4.4.3.2.2
HI-STORM Lid Bottom Plate	End Drop	9.777	3.4.4.3.2.3
HI-STORM Lid Bottom Plate Welds	End Drop	2.695	3.4.4.3.2.3
Pedestal Shield Compression	End Drop	1.011	3.4.4.3.2.3
HI-STORM Inlet Vent Plate Bending Stress	End Drop	1.271	3.4.4.3.2.3
HI-STORM Lid Top Plate Bending	End Drop –100 100S	5.208 1.357	3.4.4.3.2.3
HI-TRAC Pocket Trunnion Weld	HI-TRAC Rotation	2.92	3.4.4.3.3.1
HI-TRAC 100 Optional Bolts - Tension	HI-TRAC Rotation	1.11	3.4.4.3.3.1
HI-STORM 100 Shell	Seismic Event	14.6	3.4.7
HI-TRAC Transfer Lid Door Lock Bolts	Side Drop	2.387	3.4.4.3.3.3
HI-TRAC Transfer Lid Separation	Side Drop	1.159	3.4.4.3.3.3
HI-STORM 100 Top Lid	Missile Impact	1.065	3.4.8.1
HI-STORM 100 Shell	Missile Impact	2.50	3.4.8.1
HI-TRAC Water Jacket –Enclosure Shell Bending	Pressure	1.85	3.4.4.3.3.4
HI-TRAC Water Jacket – Enclosure Shell Bending	Pressure plus Handling	1.80	3.4.4.3.3.1
HI-TRAC Water Jacket – Bottom Flange Bending	Pressure	1.39	3.4.4.3.3.4
HI-TRAC Water Jacket – Weld	Pressure	1.42	3.4.4.3.3.4
Fuel Basket Support Plate Bending	Side Drop	1.82	3.4.4.3.1.8
Fuel Basket Support Leg Stability	Side Drop	4.07	3.4.4.3.1.8
Fuel Basket Support Welds	Side Drop	1.35	3.4.4.3.1.8
Basket Shim Combined Bending and Shear	Side Drop	2.02	3.4.4.3.1.8
Basket Shim Stability	Side Drop	28.12	3.4.4.3.1.8
Basket Shim Welds	Side Drop	1.019	3.4.4.3.1.8
MPC Cover Plates in MPC Lid	Normal Condition Internal Pressure	1.59	3.4.4.3.1.8
MPC Cover Plate Weld	Accident Condition Internal Pressure	2.51	3.4.4.3.1.8
HI-STORM Storage Overpack	External Pressure	2.88	3.4.4.5.2
HI-STORM Storage Overpack Circumferential Stress	Missile Strike	1.97	3.4.8.1
HI-TRAC Transfer Cask Circumferential Stress	Missile Strike	2.61	3.4.8.2
HI-TRAC Transfer Cask Axial Membrane Stress	Side Drop	1.52	3.4.9.3

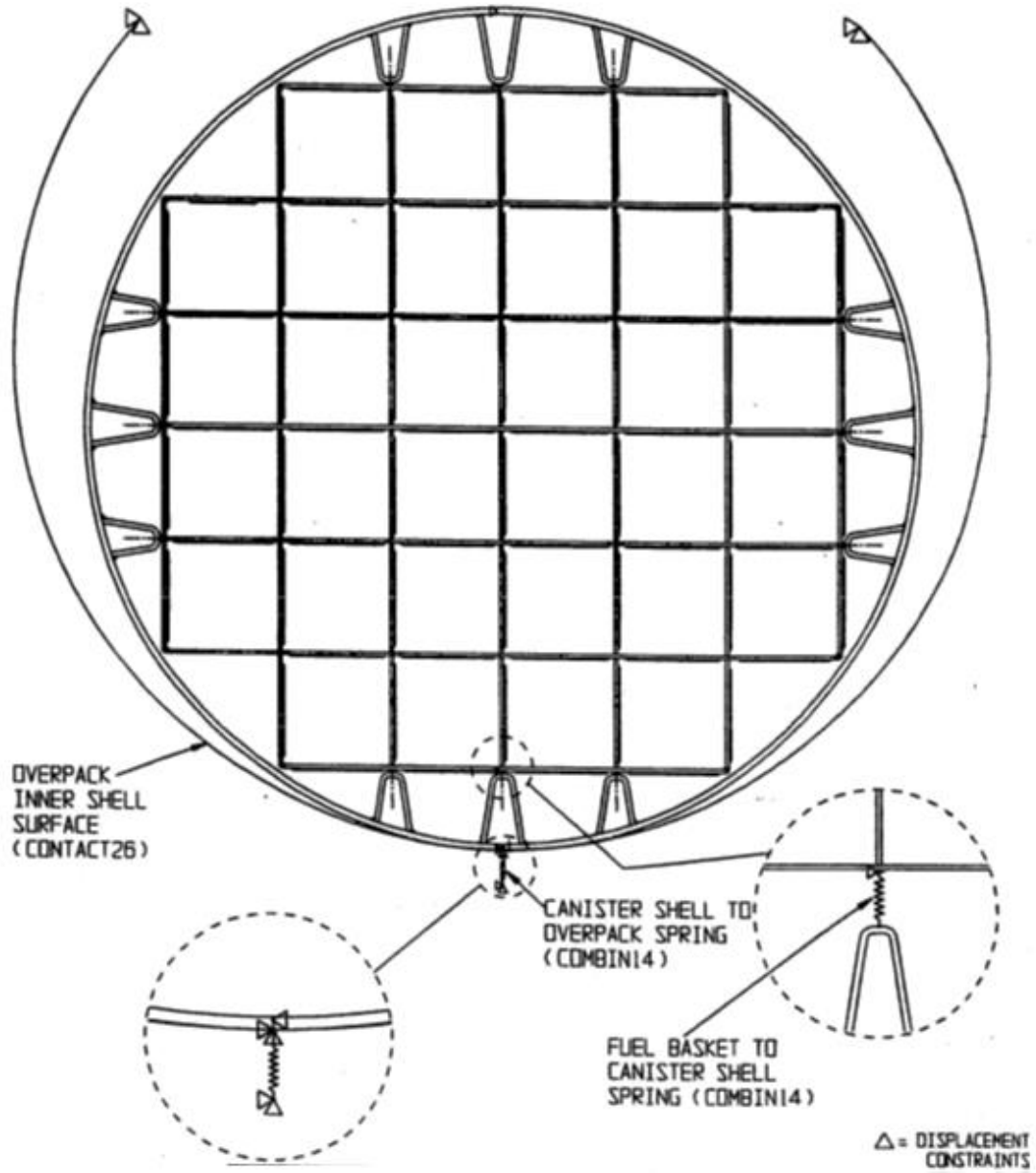
**TABLE 3.4.10**  
**INPUT DATA FOR SEISMIC ANALYSIS OF ANCHORED HI-STORM 100 SYSTEM**

Item	Data Used	Actual Value and Reference
Cask height, inch	231.25	231.25" (Dwg. 1495)
Contact diameter at ISFSI pad, inch	146.5	146.5 (Dwg. 3187)
Overpack empty, wt. Kips	270	267.87 (Table 3.2.1)
Bounding wt. of loaded MPC, kips	90	88.135 (Table 3.2.1)
Overpack-to-MPC radial gap (inch)	2.0	2.0' (Dwg. 1495, Sheets 2 and 5)
Overpack C.G. height above ISFSI pad, inch	117.0	116.8 (Table 3.2.3)
Overpack with Loaded MPC - C.G. height above ISFSI pad	118.5	118.5 (Table 3.2.3)
Applicable Response Spectra	Fig. 3.4-31 to 3.4-36	Figure 3.4-30
ZPA:	RG 1.60                      Western Plant	
Horizontal 1	1.5                              1.45	Site-Specific
Horizontal 2	1.5                              1.45	
Vertical	1.5                              1.3	
No. of Anchor Studs	28	Up to 28
Anchor Stud Diameter		
Inch	2.0	2.0 (BOM 3189)
Yield stress, ksi	80 (minimum)	Table 1.2.7
Ultimate stress, ksi	125 (minimum)	Table 1.2.7
Free length, inch*	16-42	Site-specific
Pre-load tensile stress, ksi*	55-65	55-65

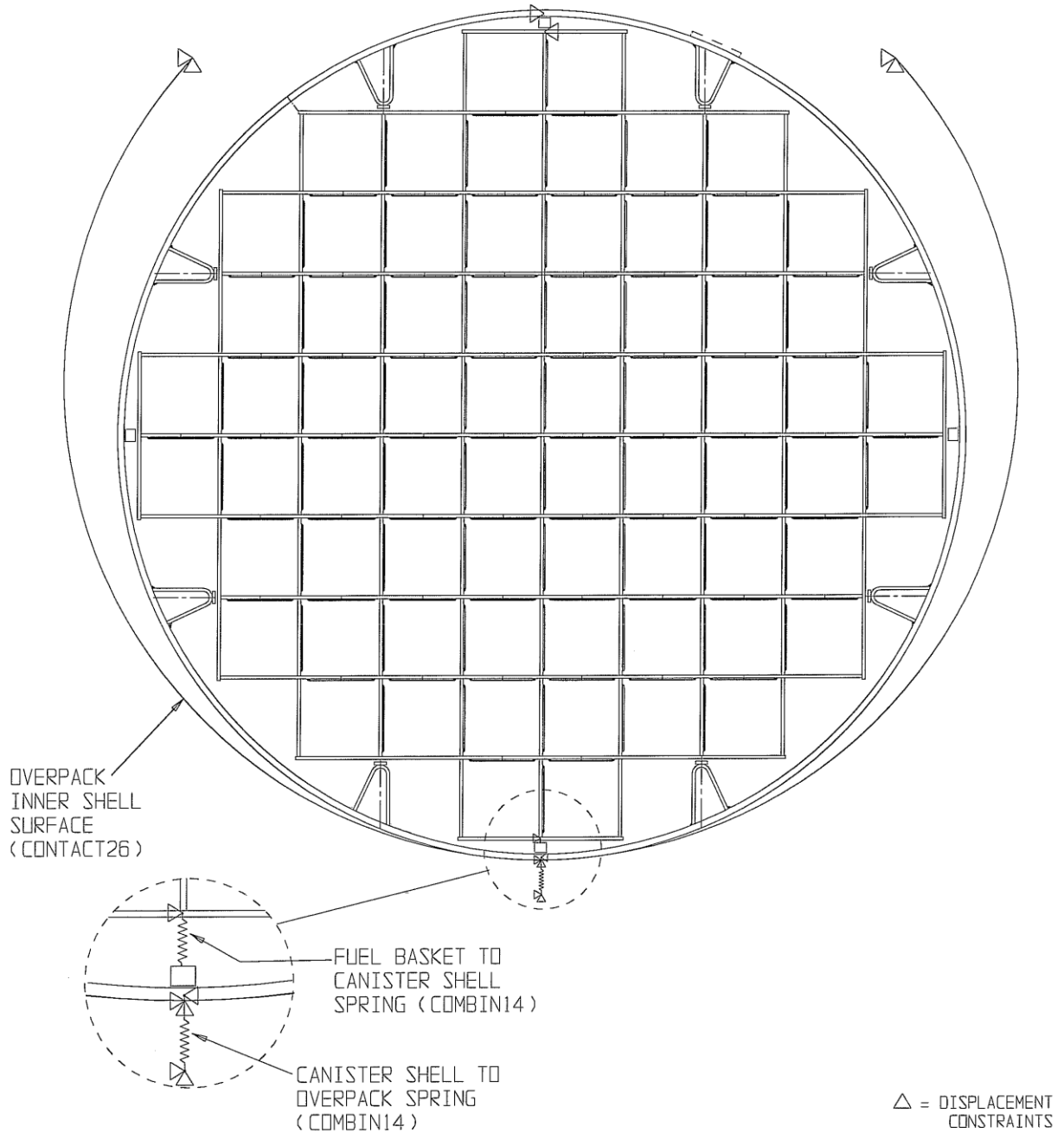
\*For the confirmatory dynamic analyses, bolt spring rates were computed using the maximum length, and the preload stress was slightly above 60.1 ksi. For the static analysis, all combinations were evaluated.



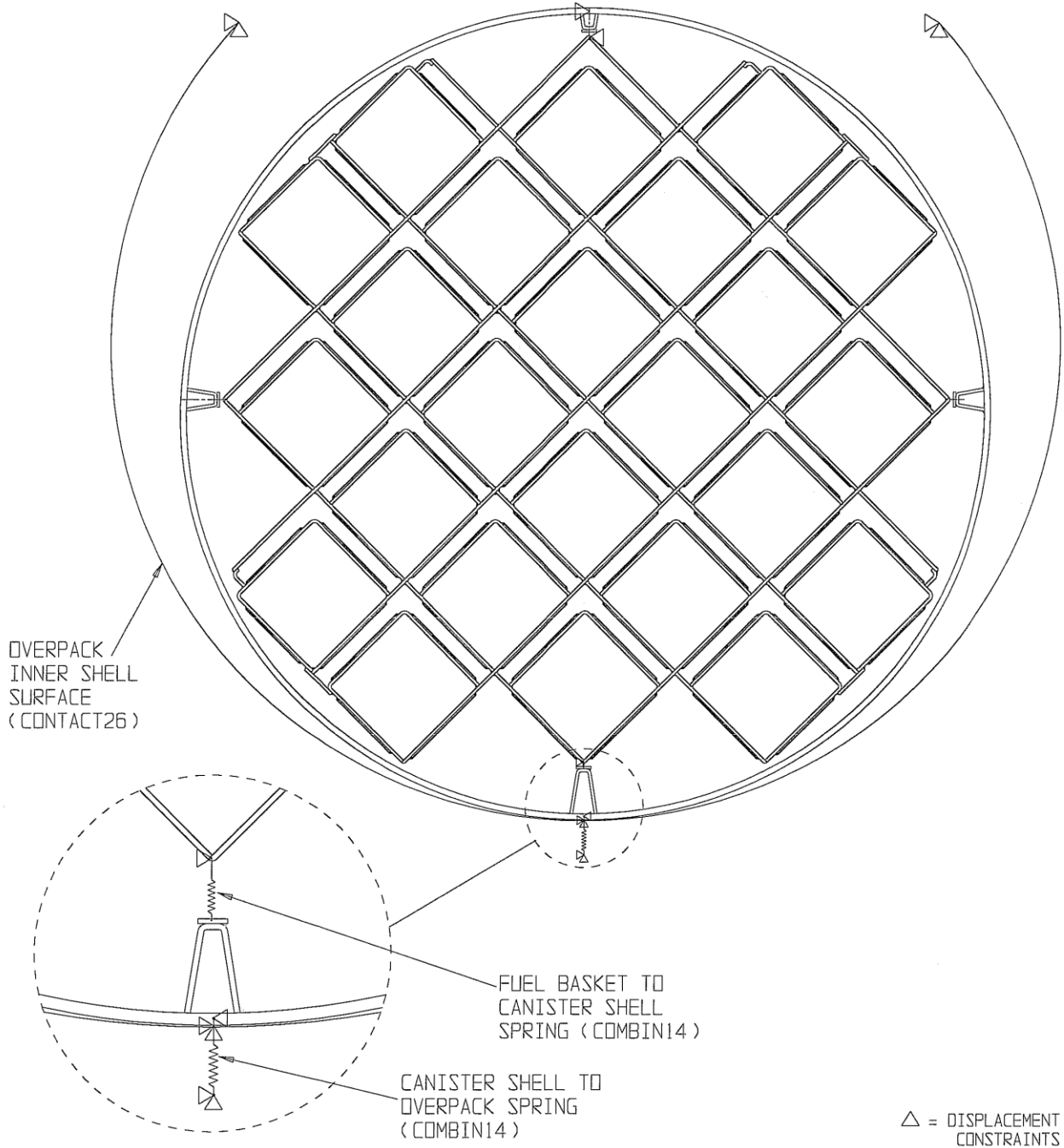
**FIGURE 3.4.1: FINITE ELEMENT MODEL OF MPC-24 (0 DEGREE DROP MODEL)**



**FIGURE 3.4.2: FINITE ELEMENT MODEL OF MPC-32 (0 DEGREE DROP MODEL)**

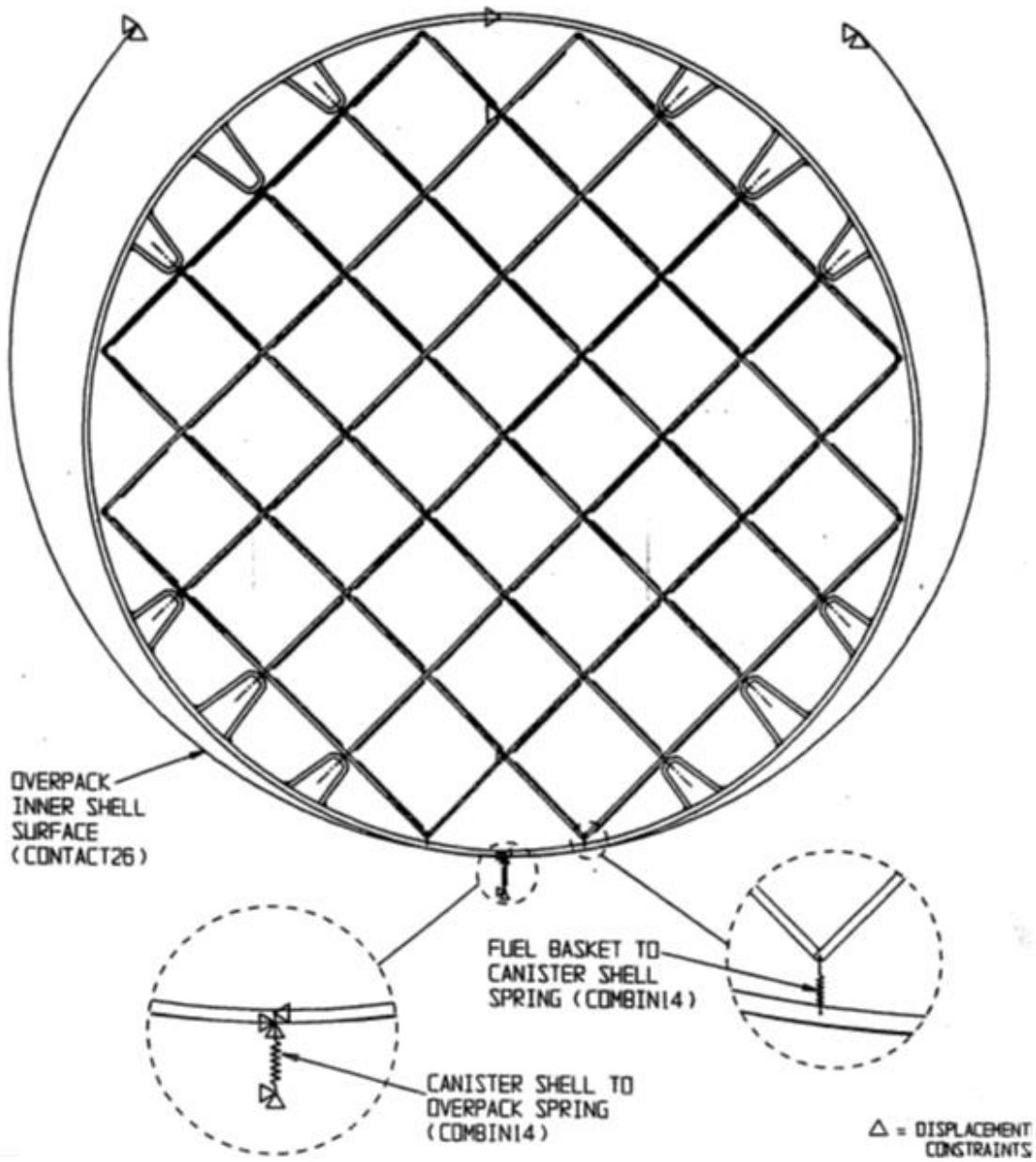


**FIGURE 3.4.3: FINITE ELEMENT MODEL OF MPC-68  
(0 DEGREE DROP MODEL)**

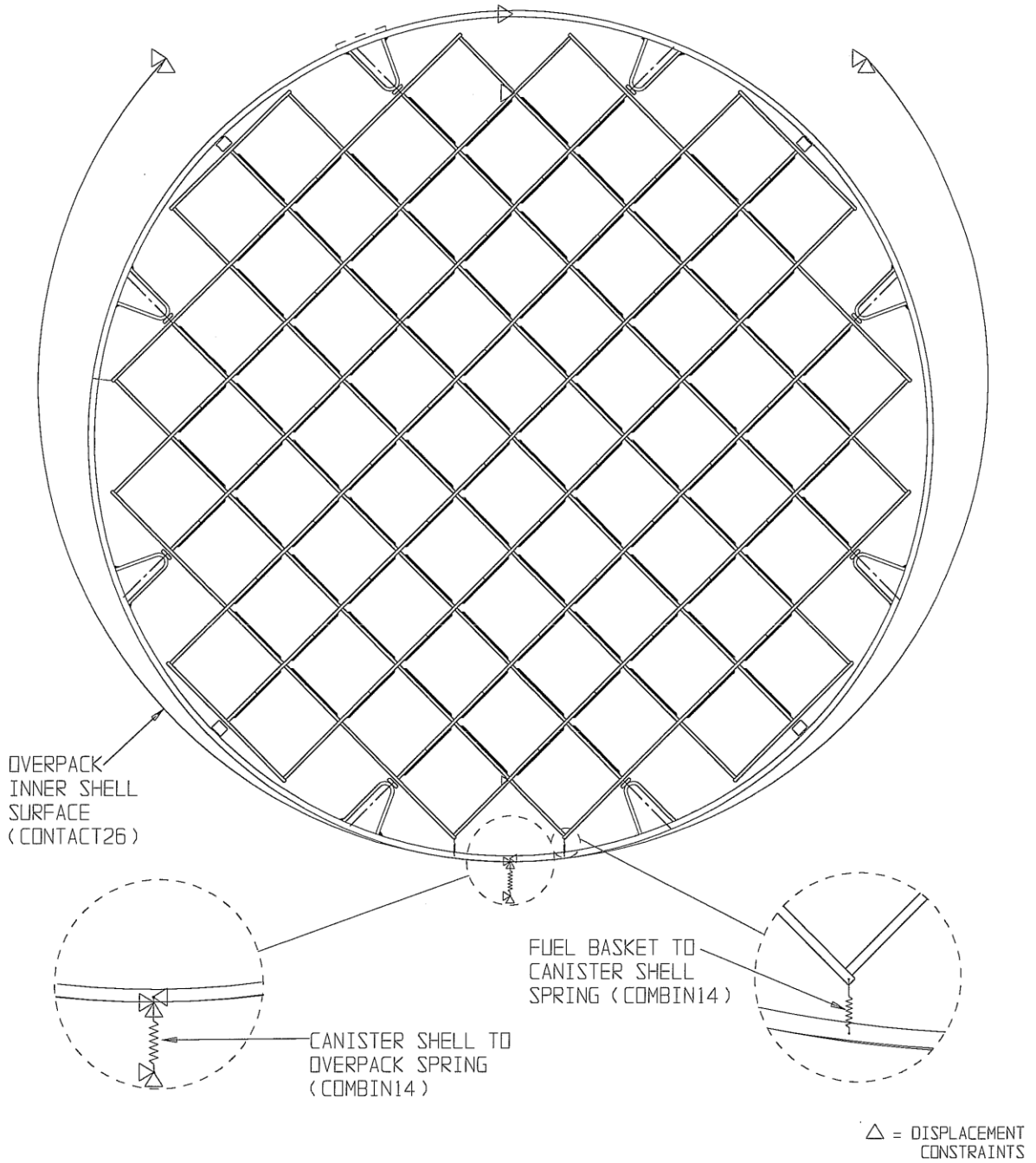


**FIGURE 3.4.4: FITE ELEMENT MODEL OF MPC-24  
(45 DEGREE DROP MODEL)**

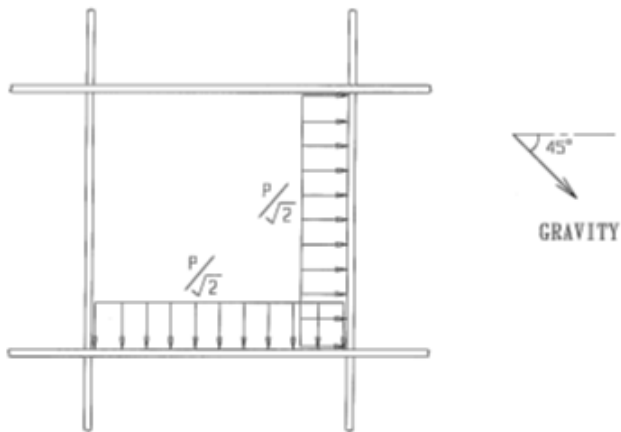
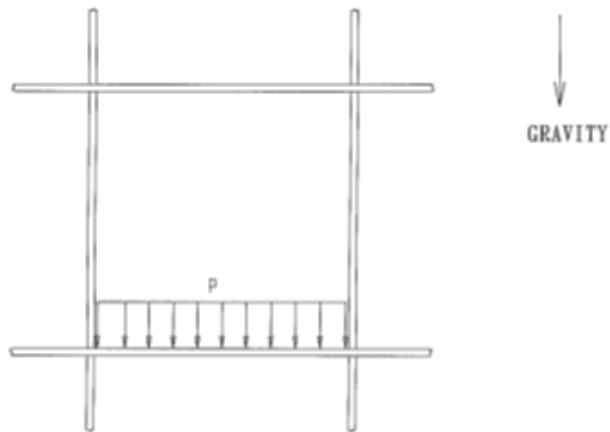




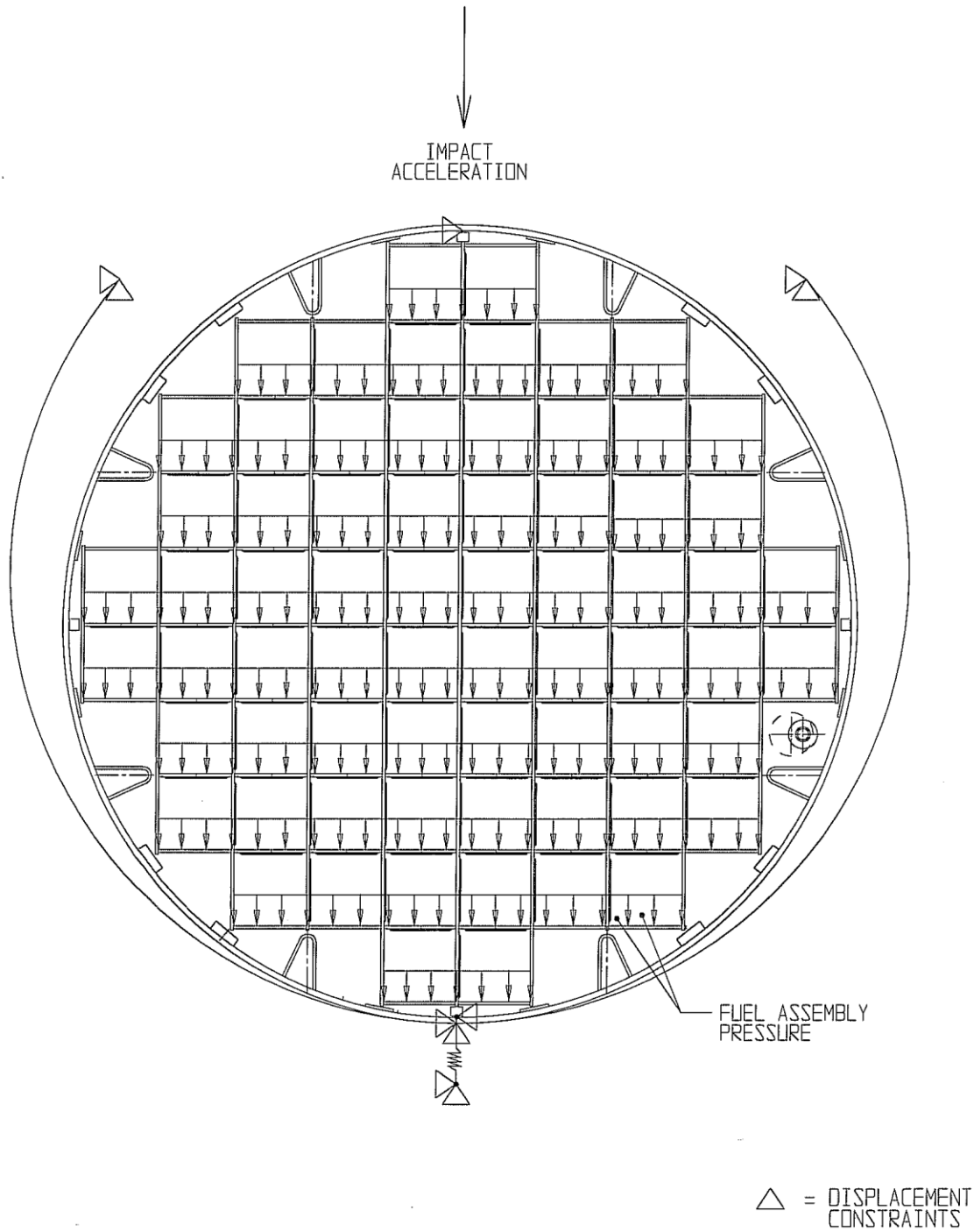
**FIGURE 3.4.5: FINITE ELEMENT MODEL OF MPC-32  
(45 DEGREE DROP MODEL)**



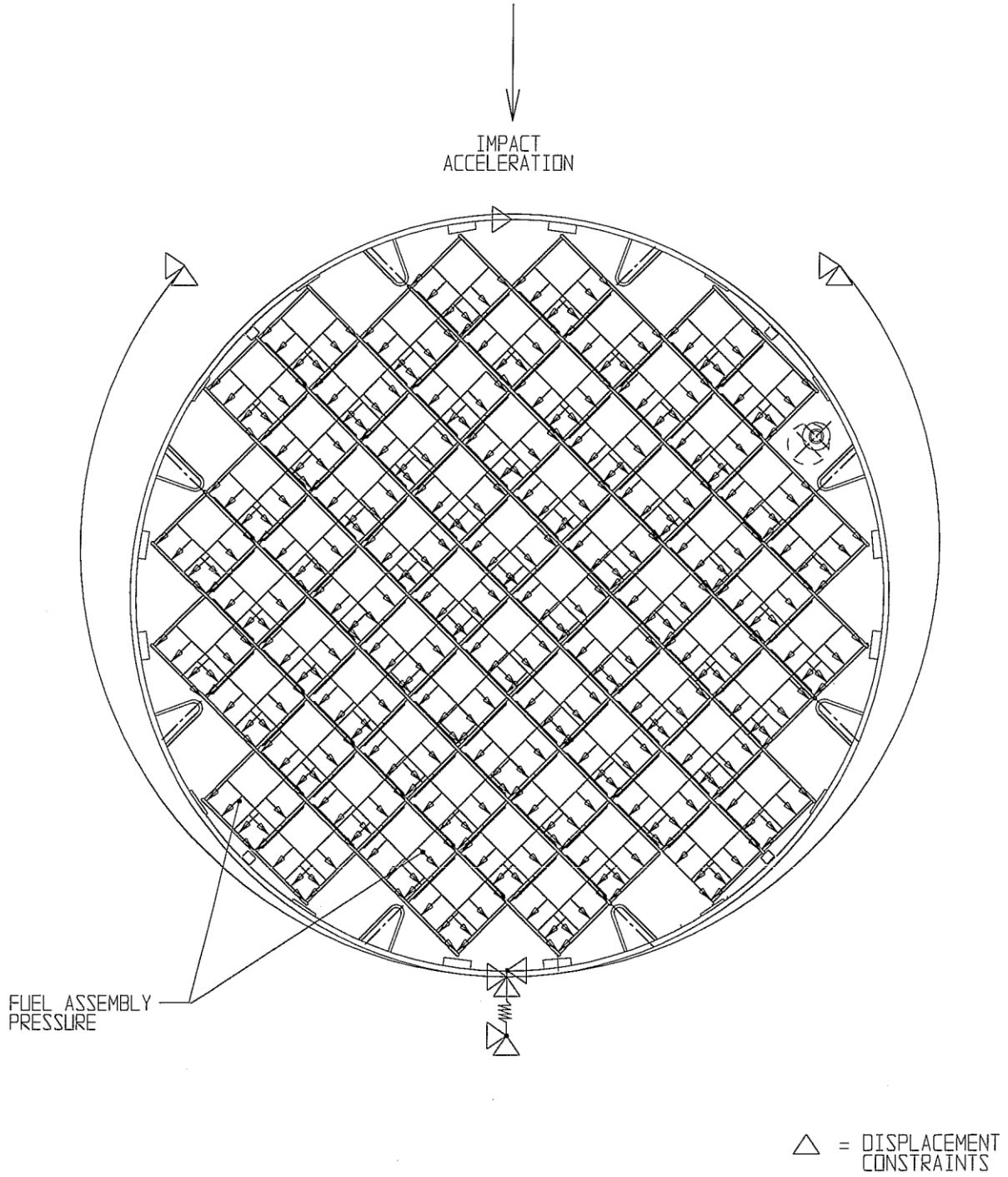
**FIGURE 3.4.6: FINITE ELEMENT MODEL OF MPC-68  
(45 DEGREE DROP MODEL)**



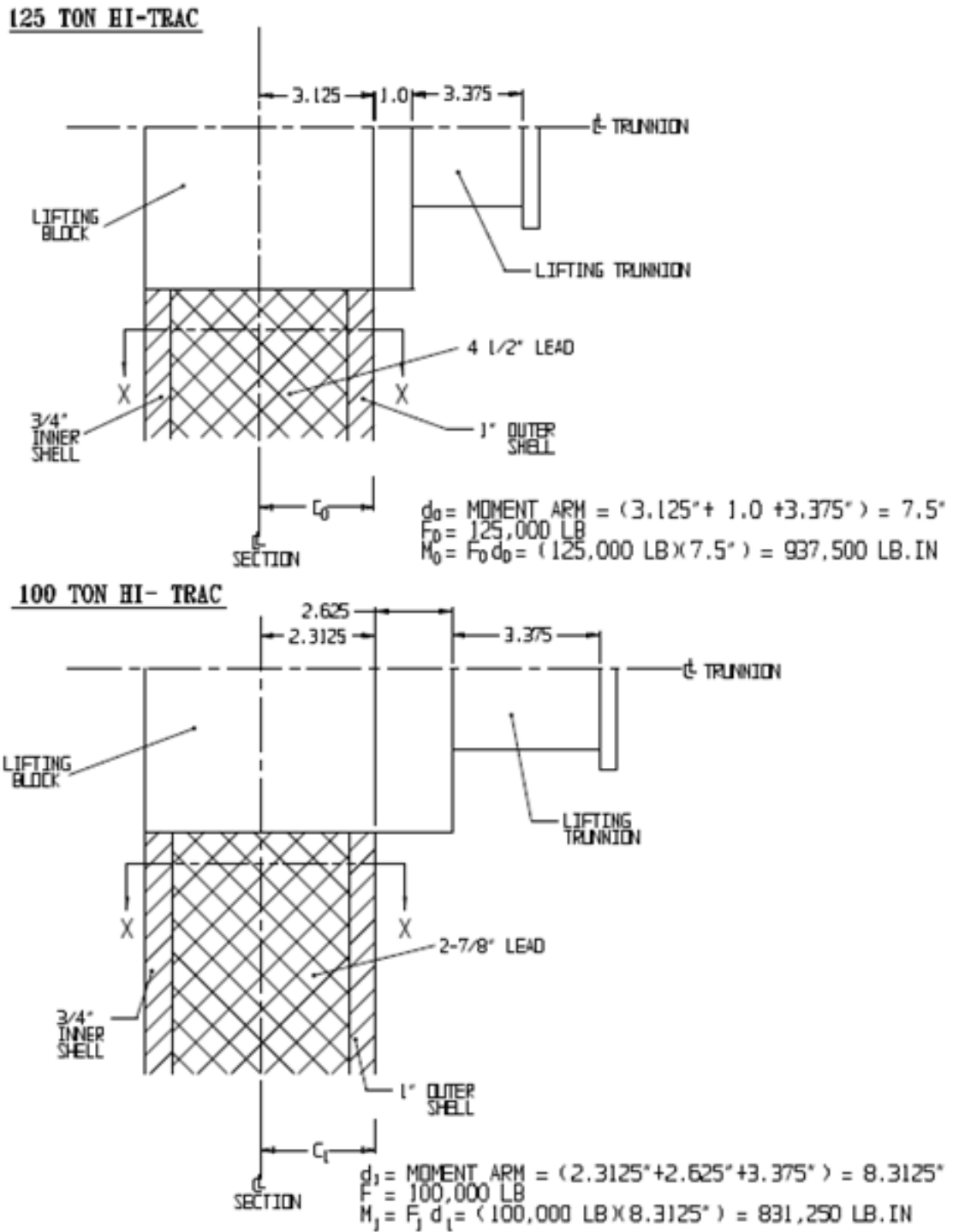
**FIGURE 3.4.7: DETAIL OF FUEL ASSEMBLY PRESSURE LOAD ON MPC BASKET**



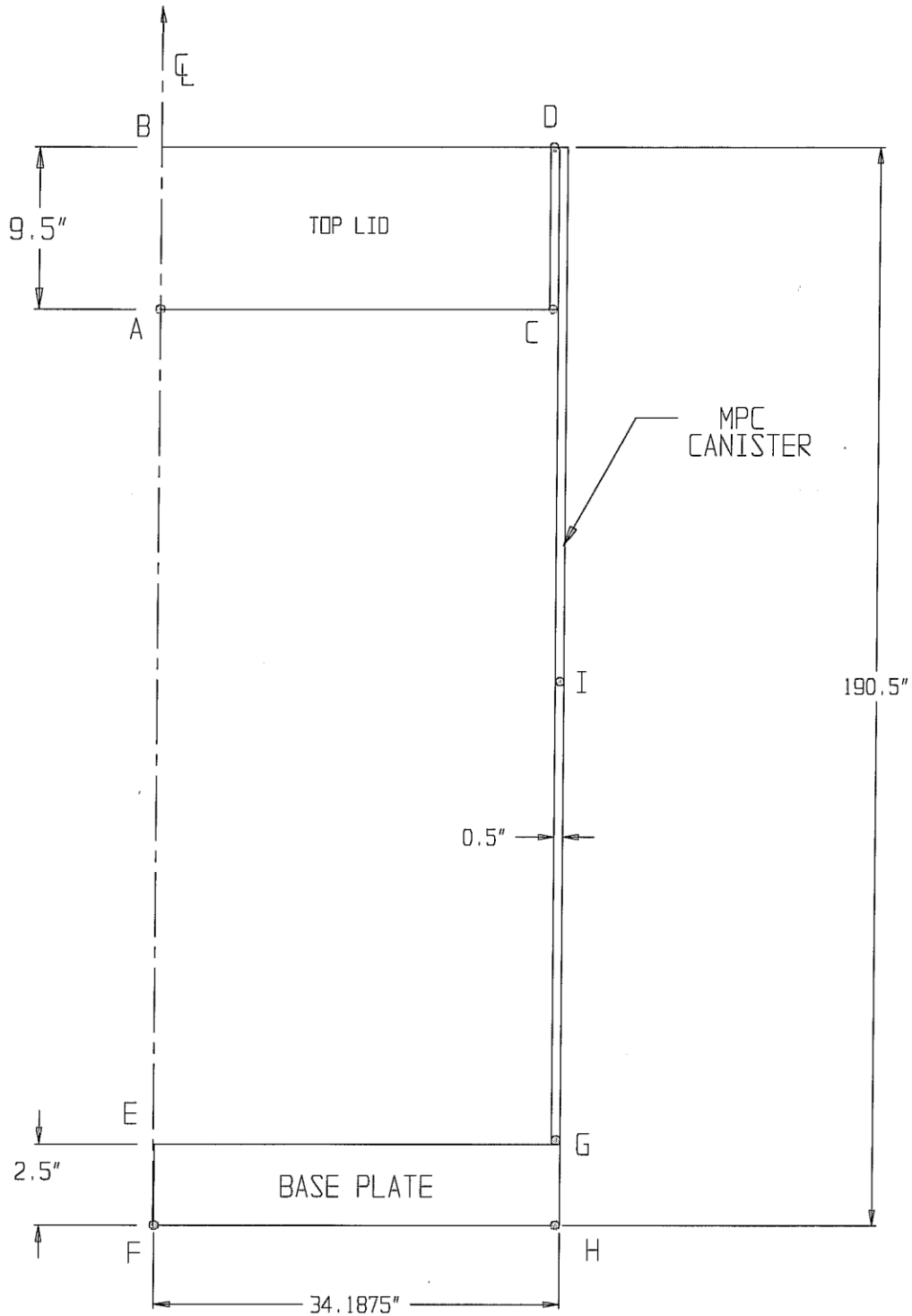
**FIGURE 3.4.8: 0 DEGREE SIDE DROP OF MPC**



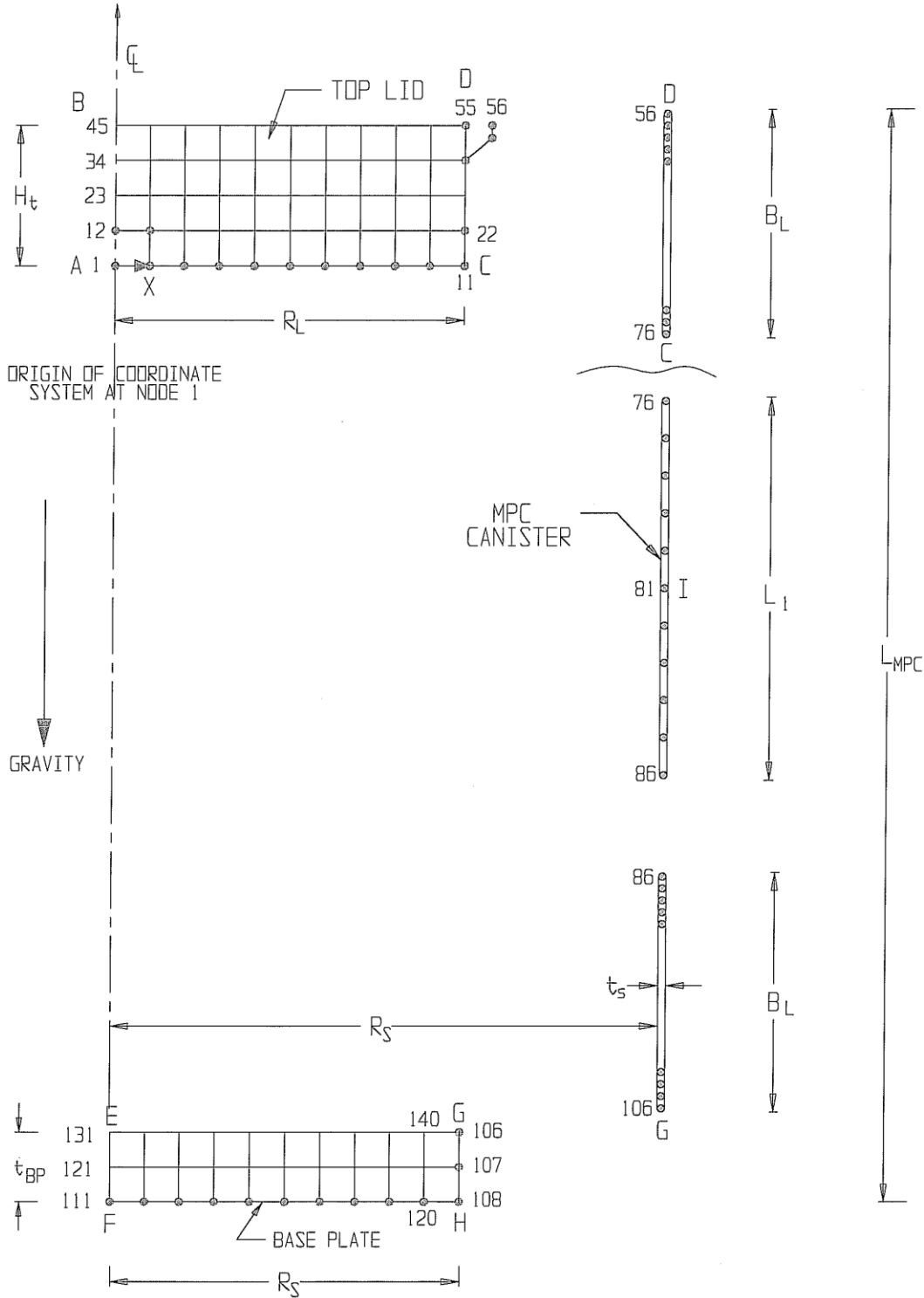
**FIGURE 3.4.9: 45 DEGREE SIDE DROP OF MPC**



**FIGURE 3.4.10: COMPARISON OF 125 TON AND 100 TON HI-TRAC LIFTING TRUNNION CONNECTION**

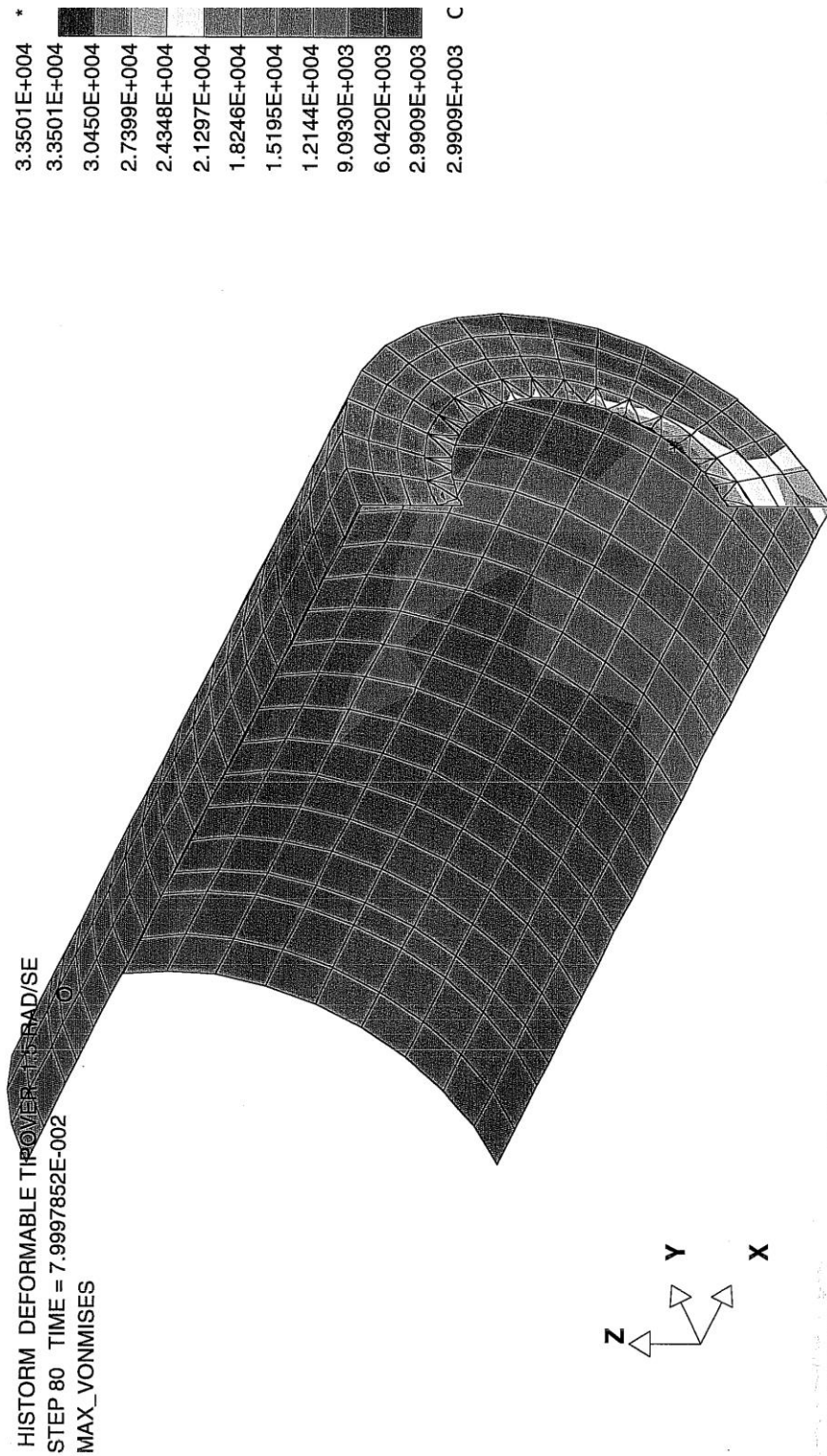


**FIGURE 3.4.11: CONFINEMENT BOUNDARY MODEL SHOWING TEMPERATURE DATA POINTS**

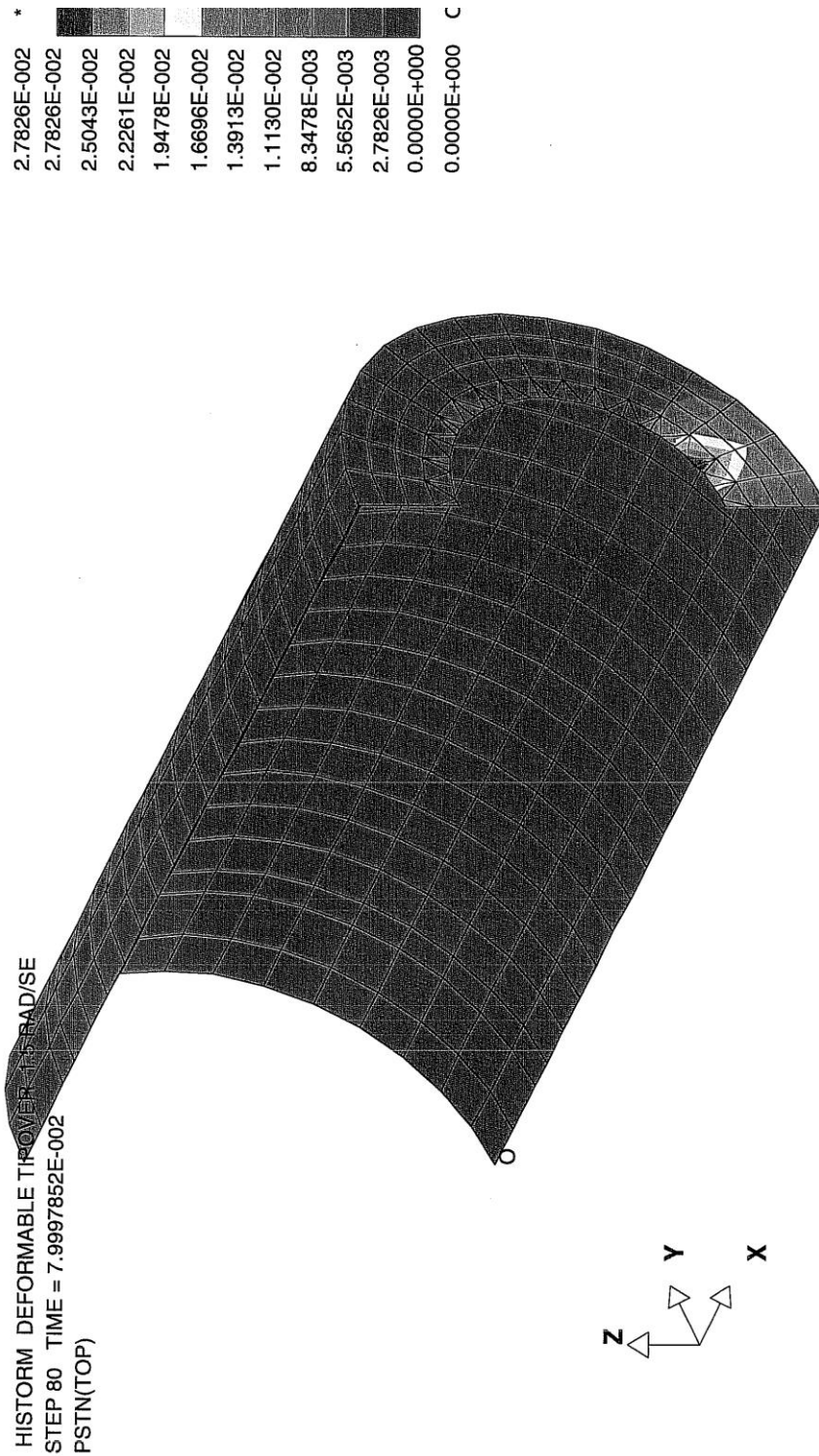


**FIGURE 3.4.12: MPC CONFINEMENT BOUNDARY FINITE ELEMENT GRID (EXPLODED VIEW)**

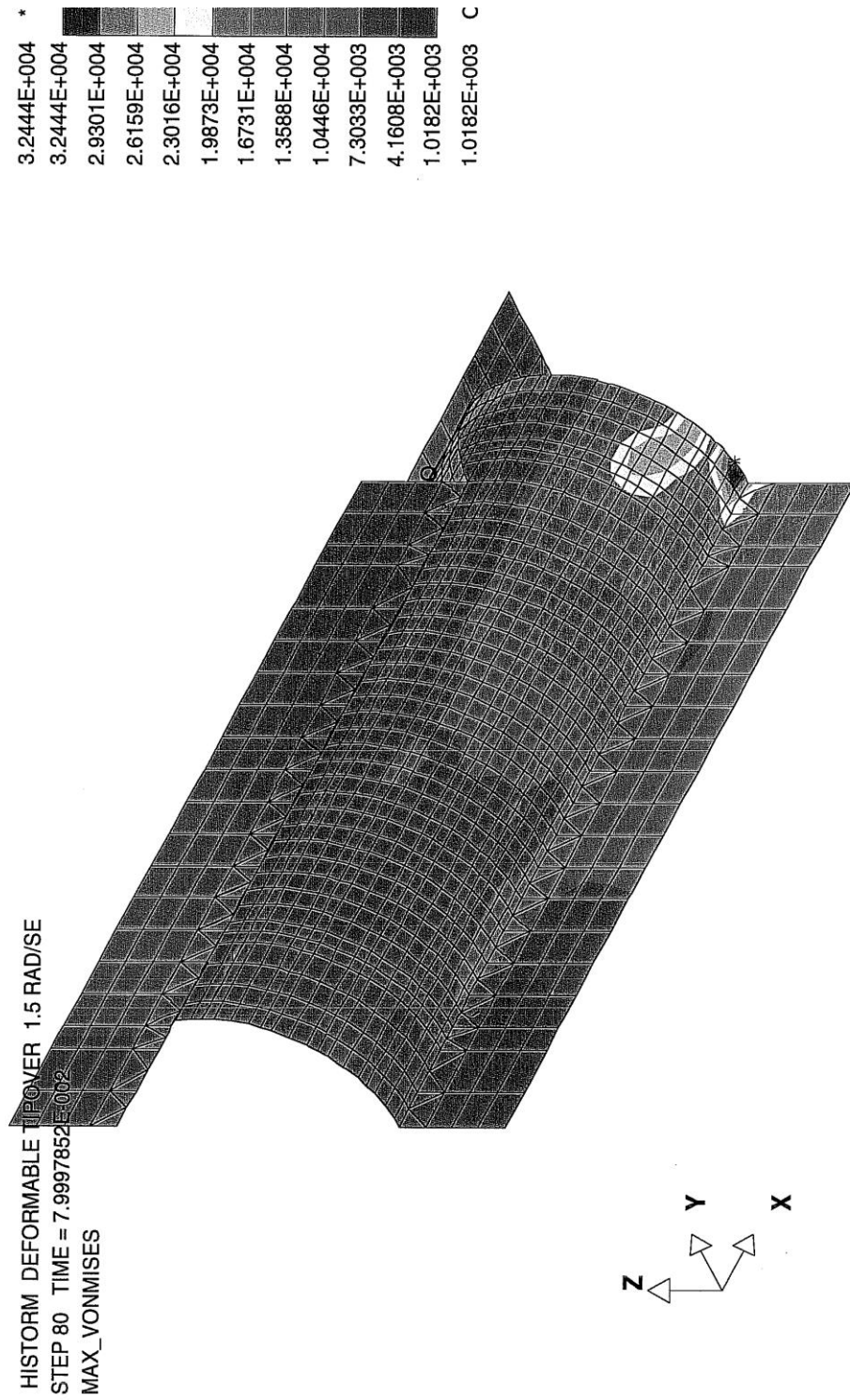




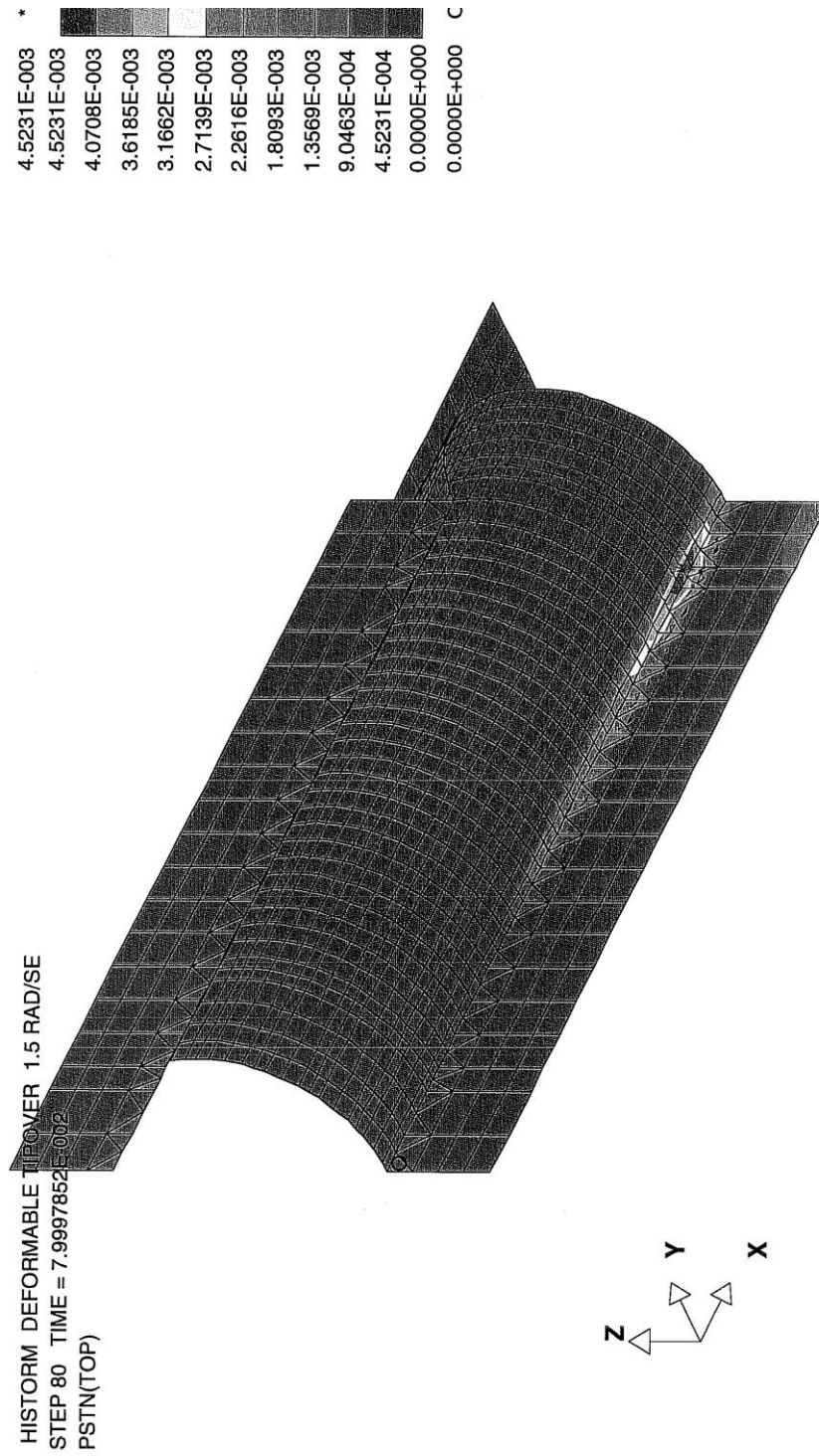
**FIGURE 3.4.13: Von Mises Stress Outer Shell**



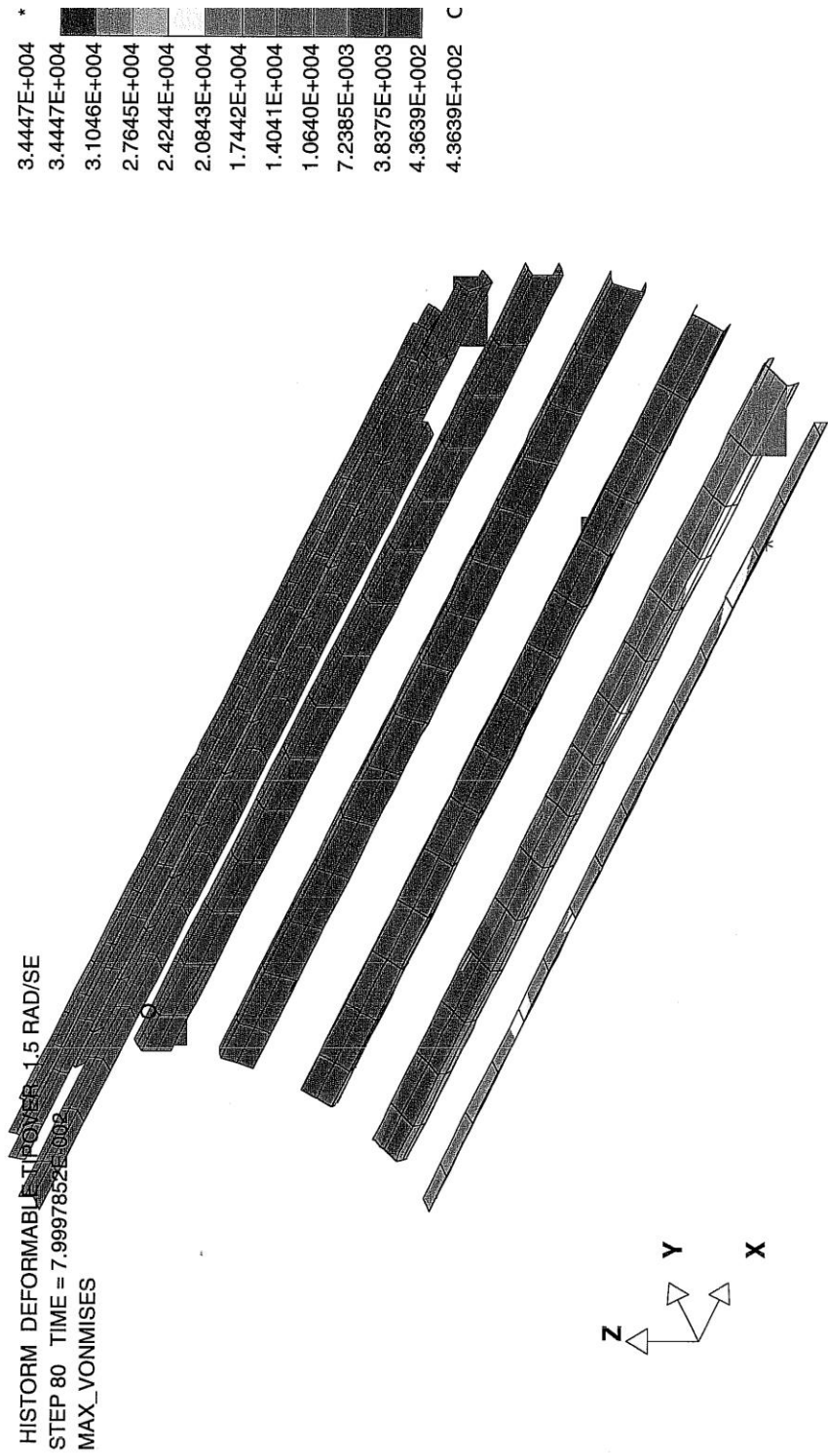
**FIGURE 3.4.14: Plastic Strain Outer Shell**



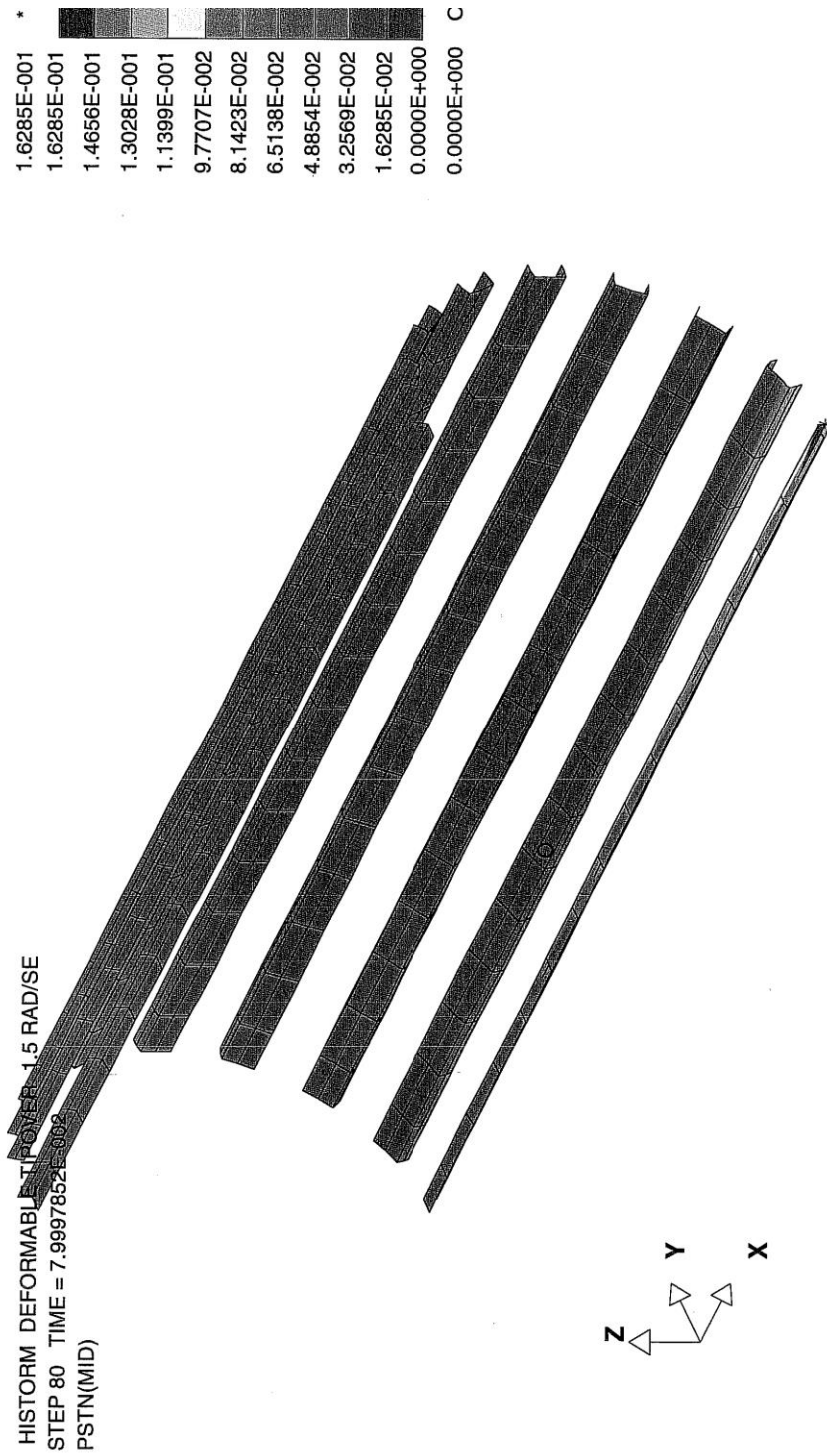
**FIGURE 3.4.15: Von Mises Stress - Inner Shell**



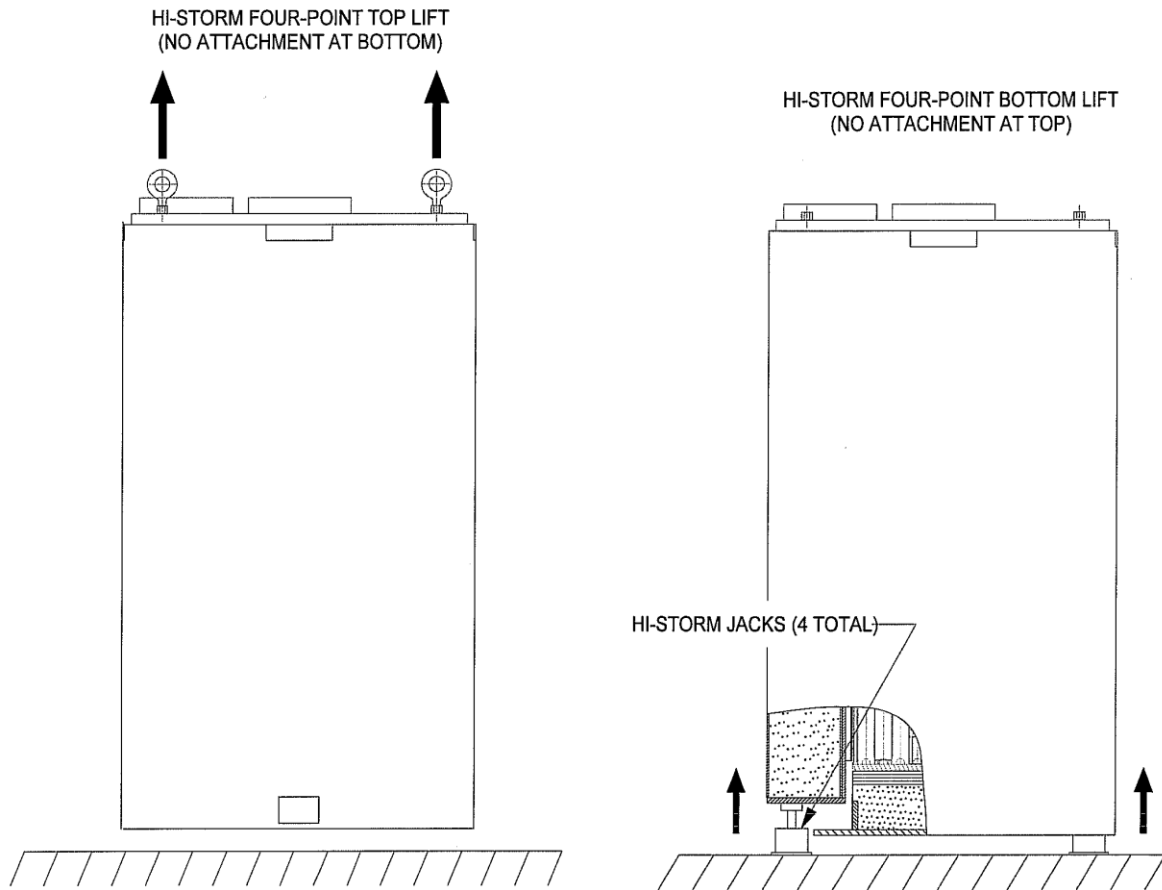
**FIGURE 3.4.16: Plastic Strain Inner Shell**



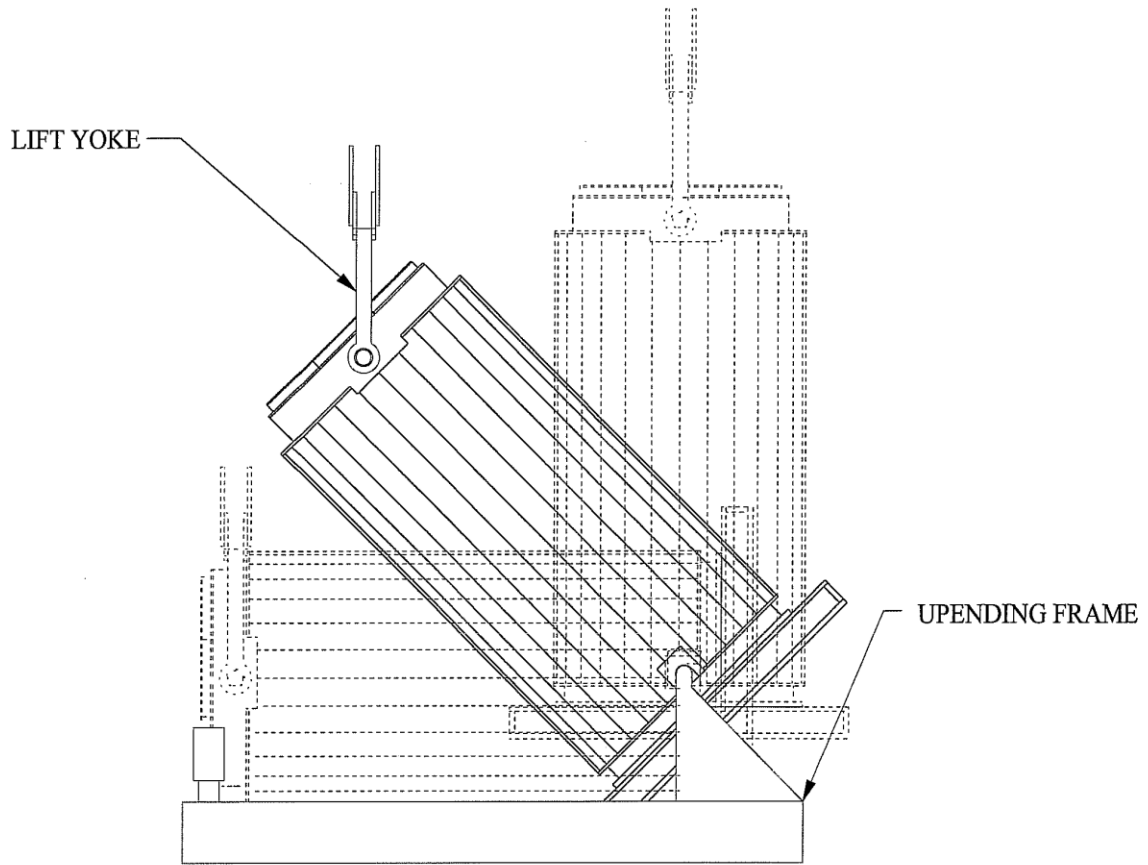
**FIGURE 3.4.16a: Von Mises Stress - Channel**



**FIGURE 3.4.16b: Plastic Strain – Channel**

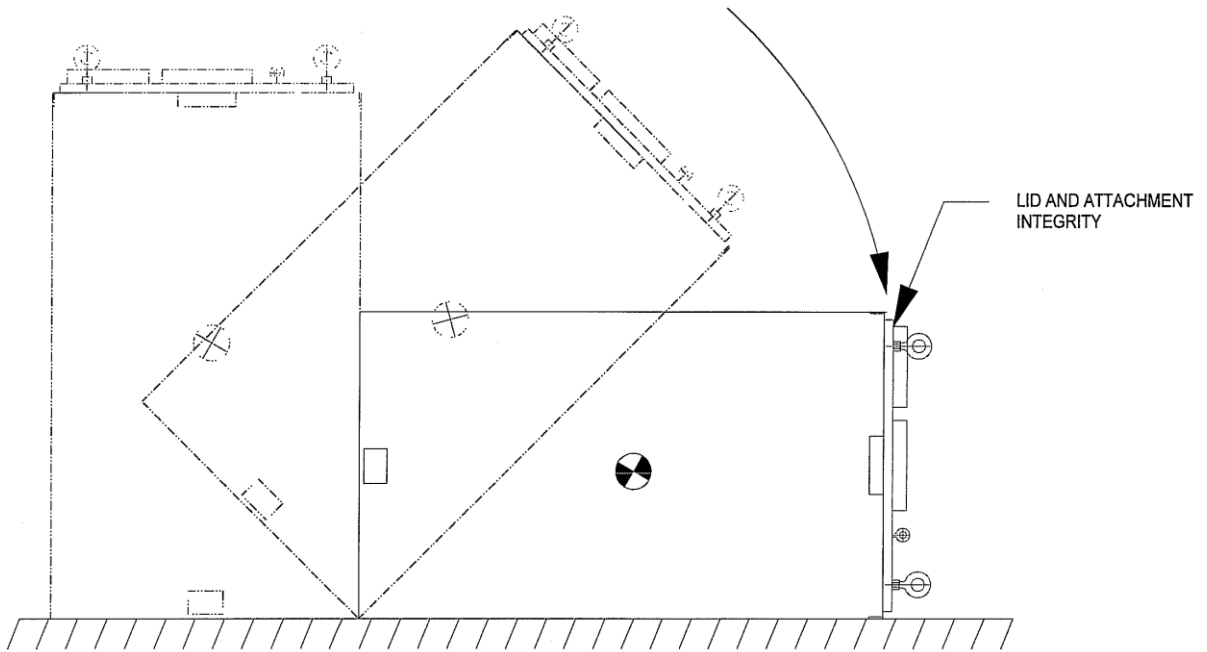


**FIGURE 3.4.17: Top and Bottom Lifting of the Loaded HI-STORM 100**

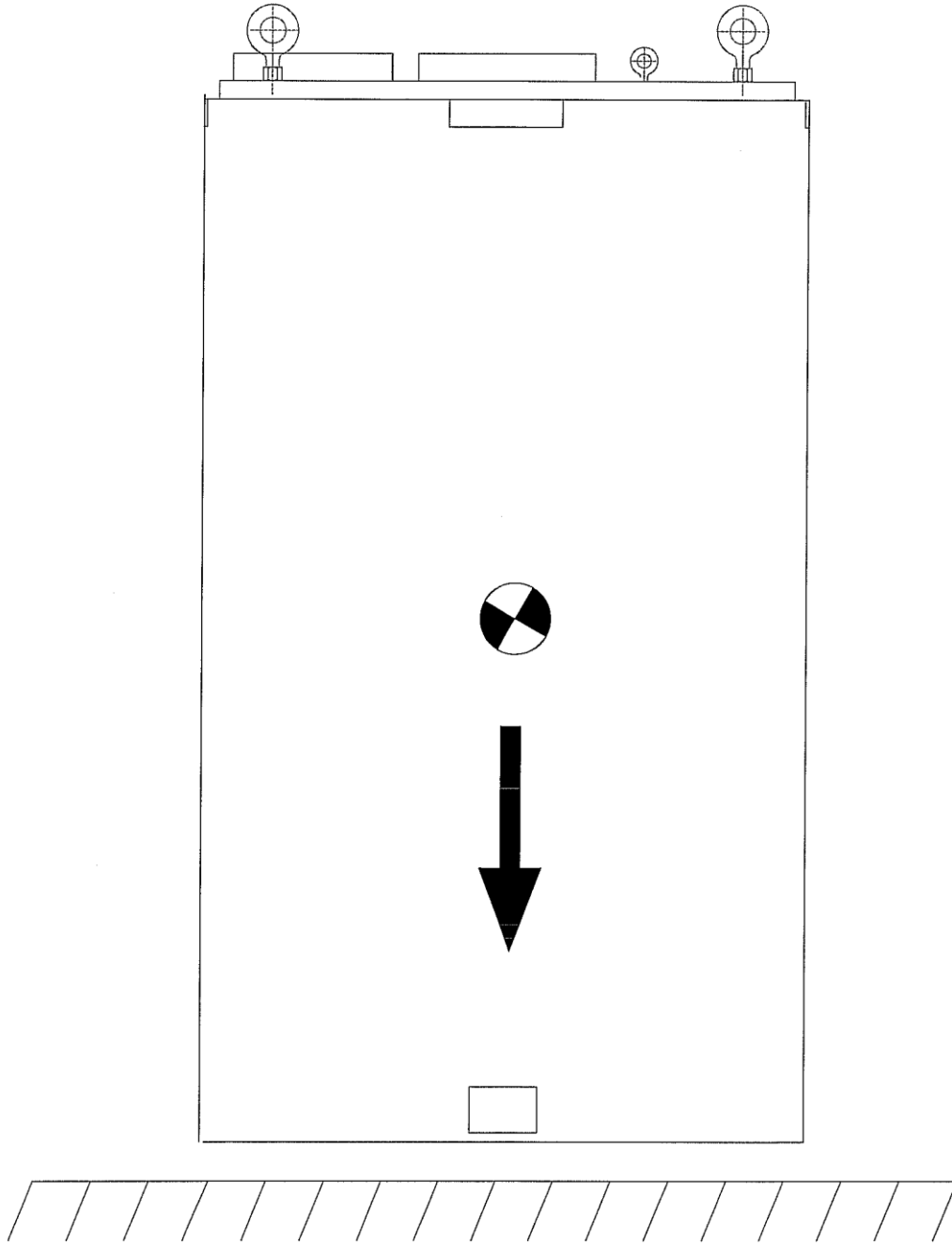


**FIGURE 3.4.18: HI-TRAC Upending in the Upending Frame**

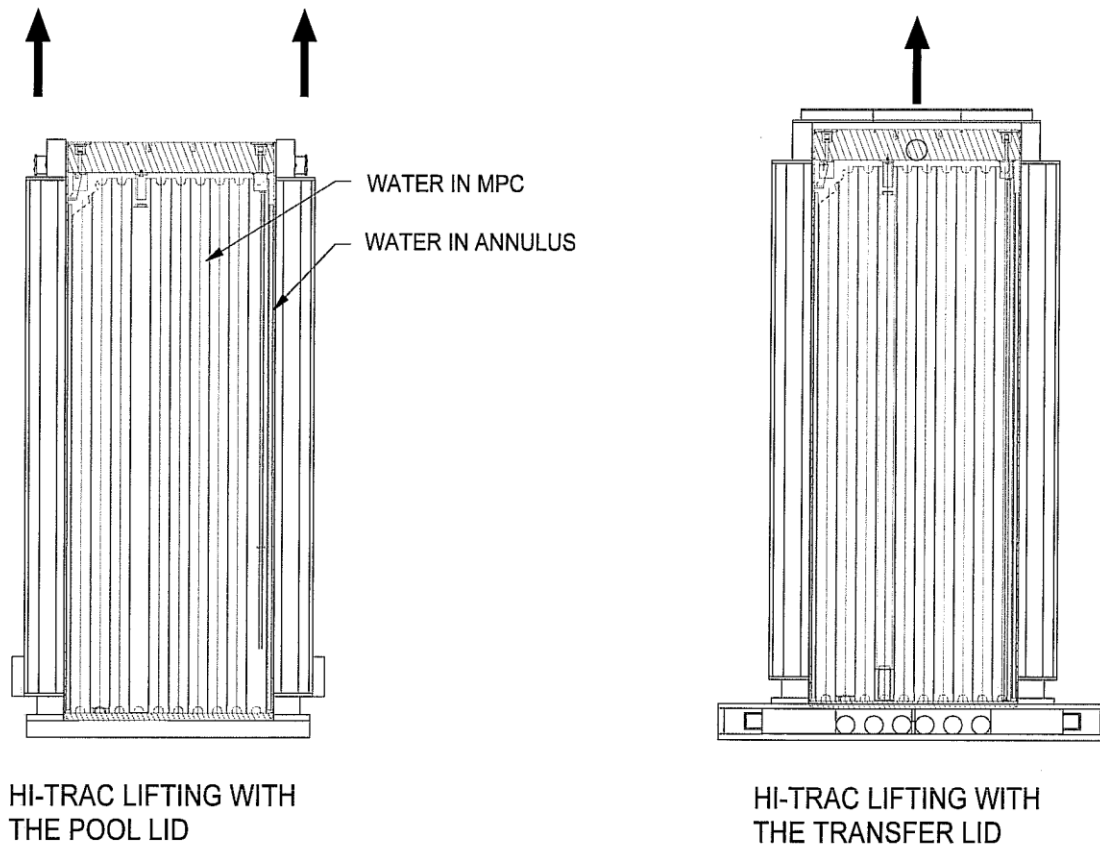




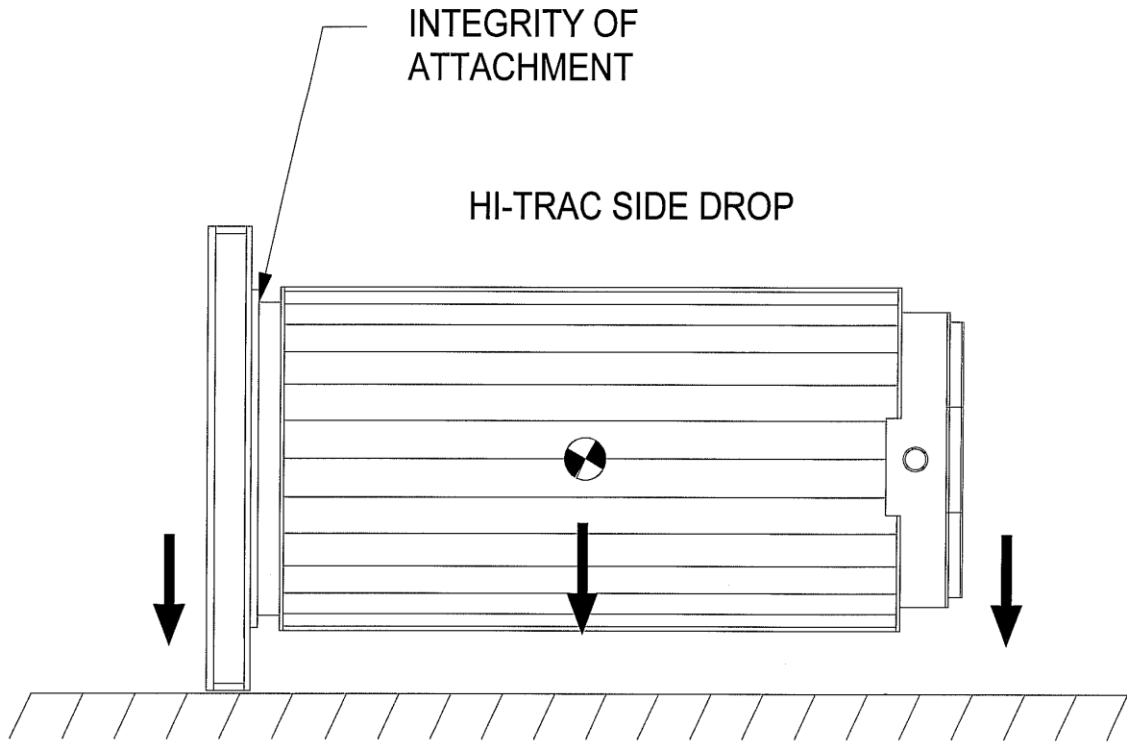
**FIGURE 3.4.19: HI-STORM 100 Tip-Over Event**



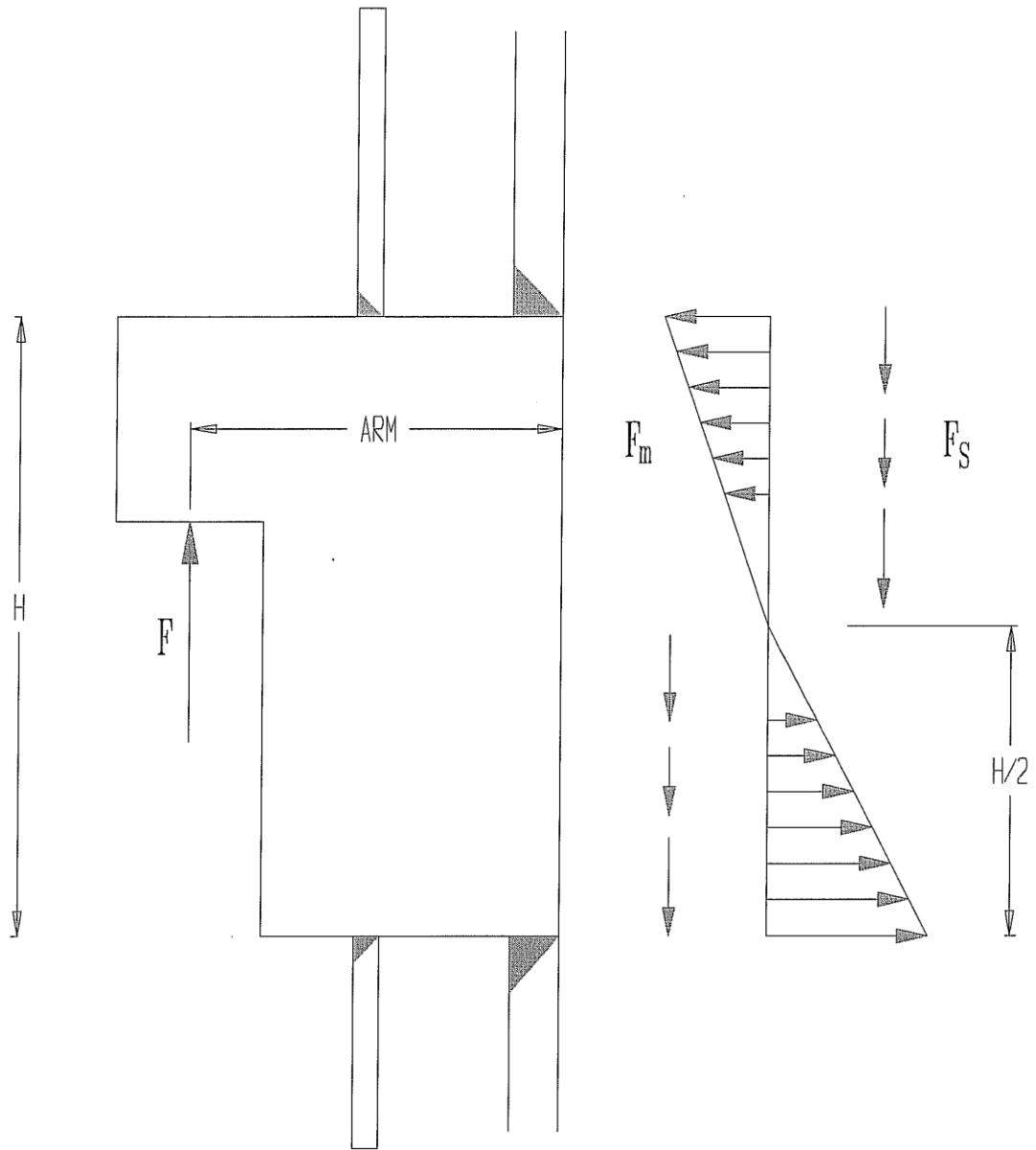
**FIGURE 3.4.20: HI-STORM 100 End Drop Event**



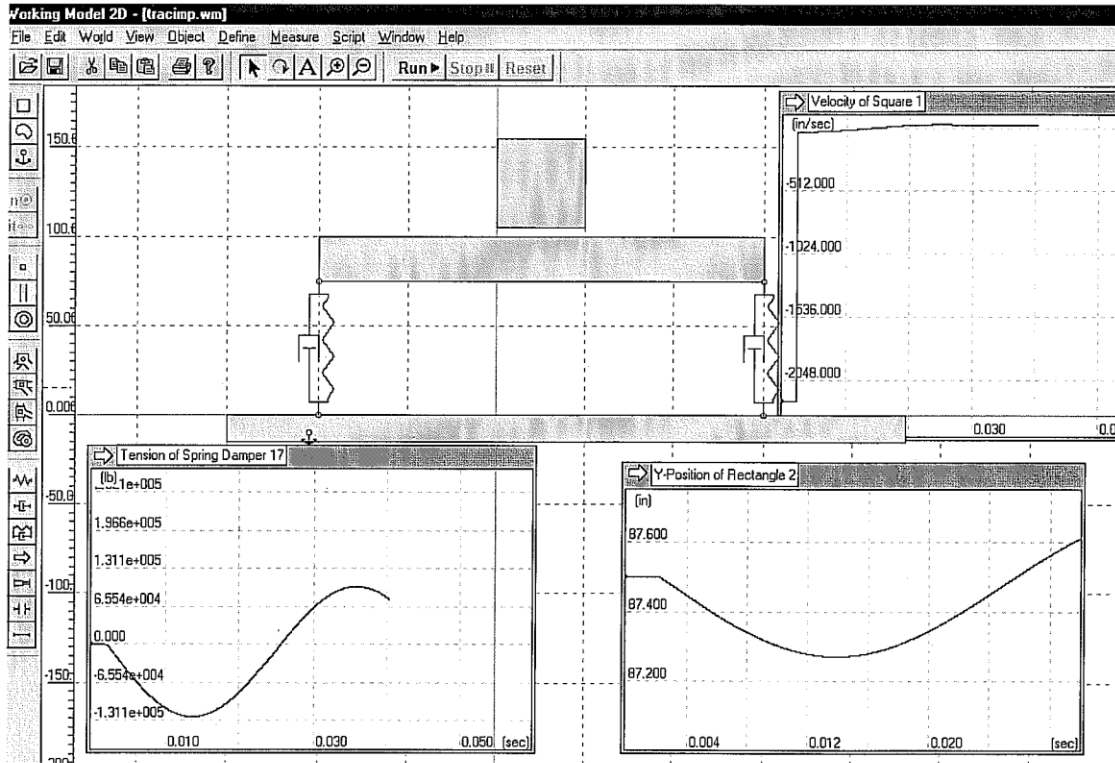
**FIGURE 3.4.21: HI-TRAC Lifting with the Pool and Transfer Lids**



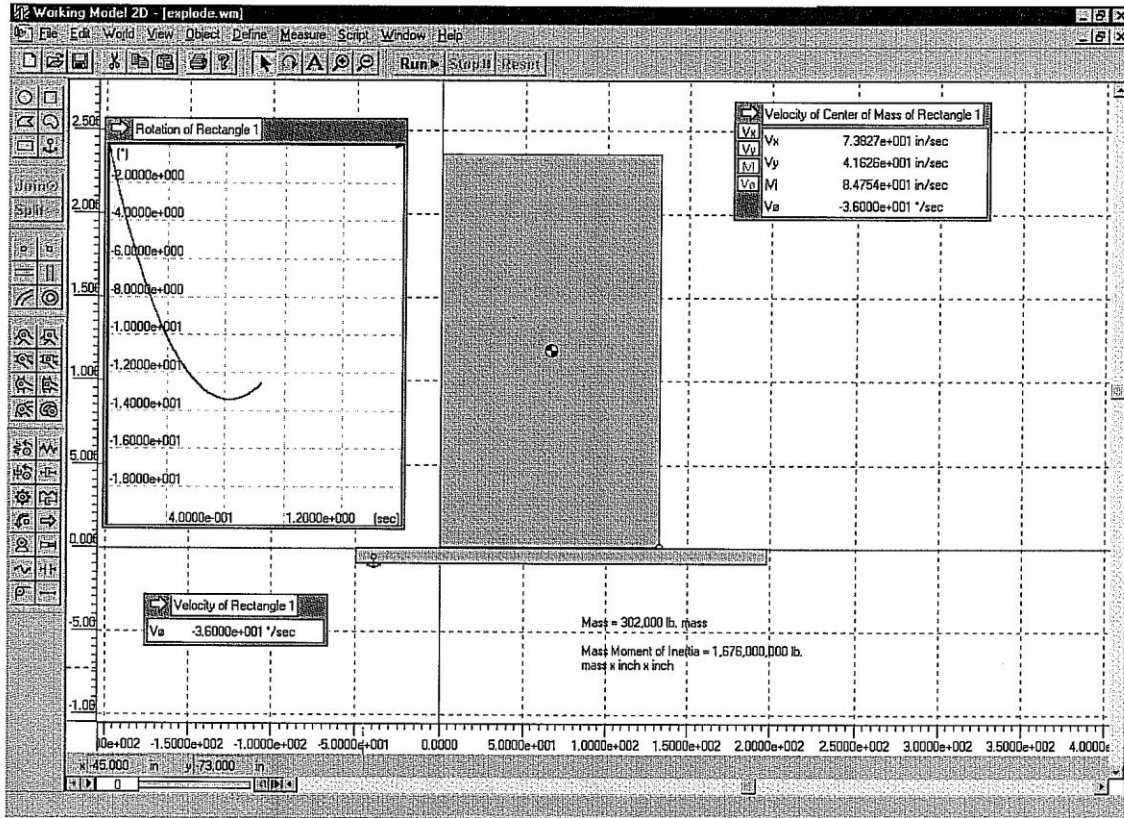
**FIGURE 3.4.22: HI-TRAC Side Drop Event**



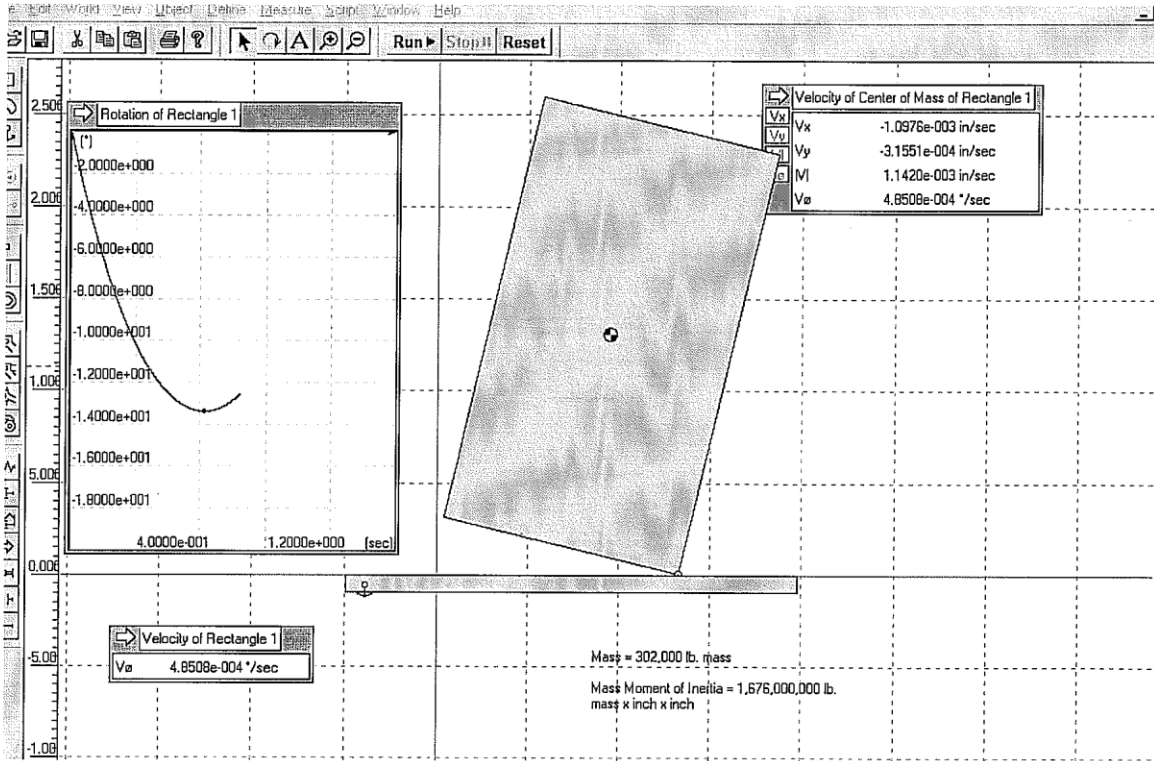
**FIGURE 3.4.23: FORCES AND MOMENTS ON 125 TON ROTATION TRUNNION WELD**



**FIGURE 3.4.24: WORKING MODEL SOLUTION FOR IMPACT FORCES ON HI-TRAC 100 TRANSFER CASK OUTER SHELL**

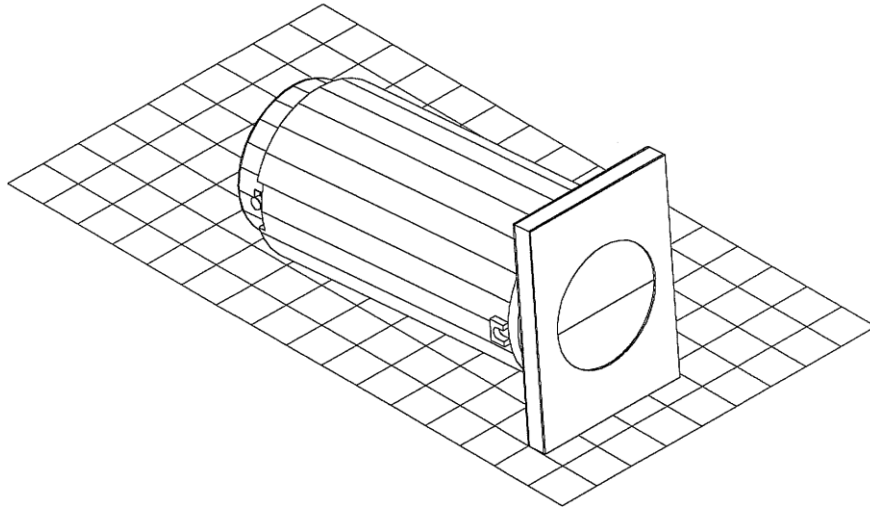


**FIGURE 3.4.25: HI-STORM 100 OVERTURNING SCENARIO – INITIAL ANGULAR VELOCITY = 0.628 RAD/SECOND ASSUMED CAUSED BY A PRESSURE PULSE**

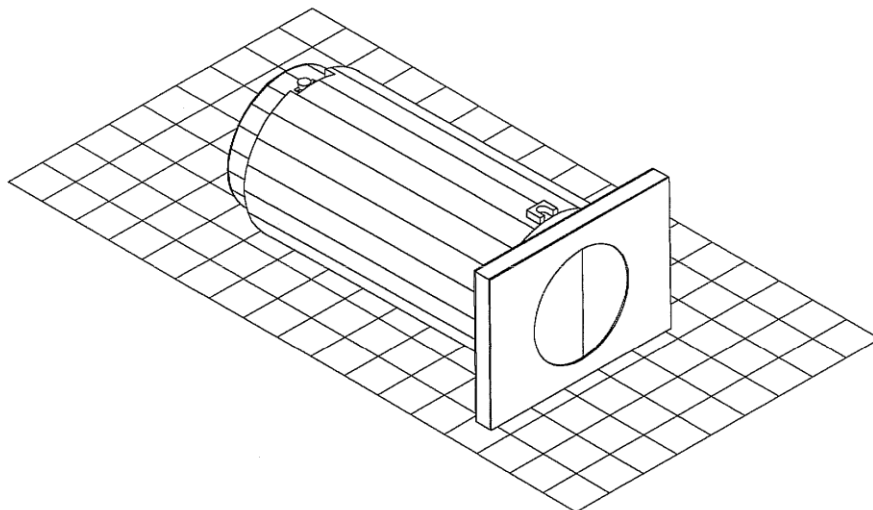


**FIGURE 3.4.26: HI-STORM 100 OVERTURNING SCENARIO – INITIAL ANGULAR VELOCITY = 0.628 RADIANS/SECOND MAXIMUM ANGULAR EXCURSION**

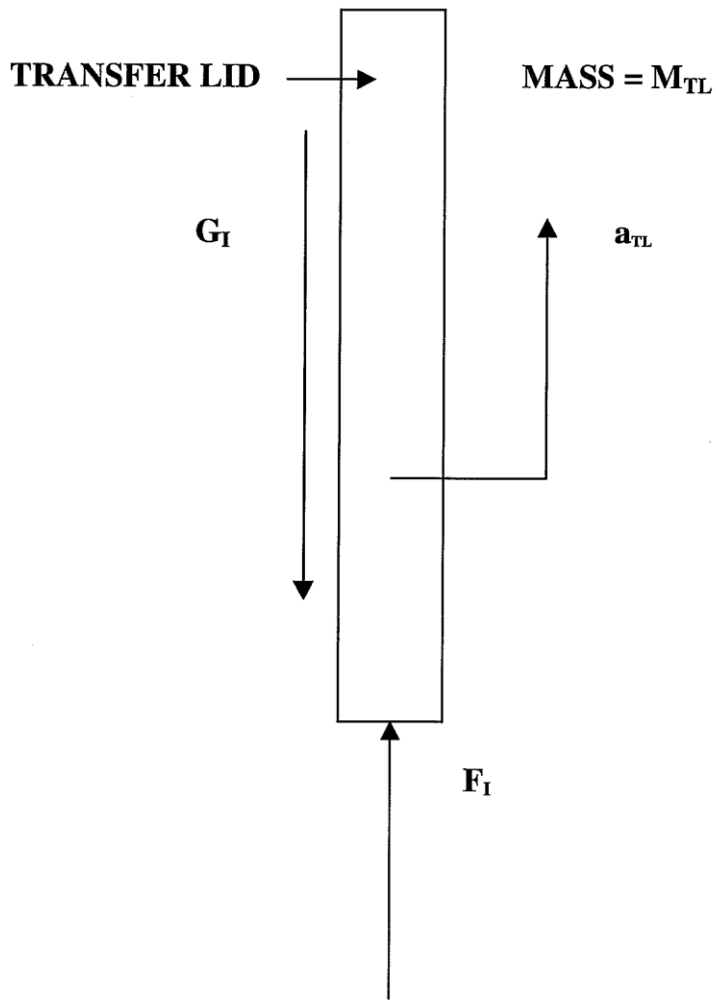




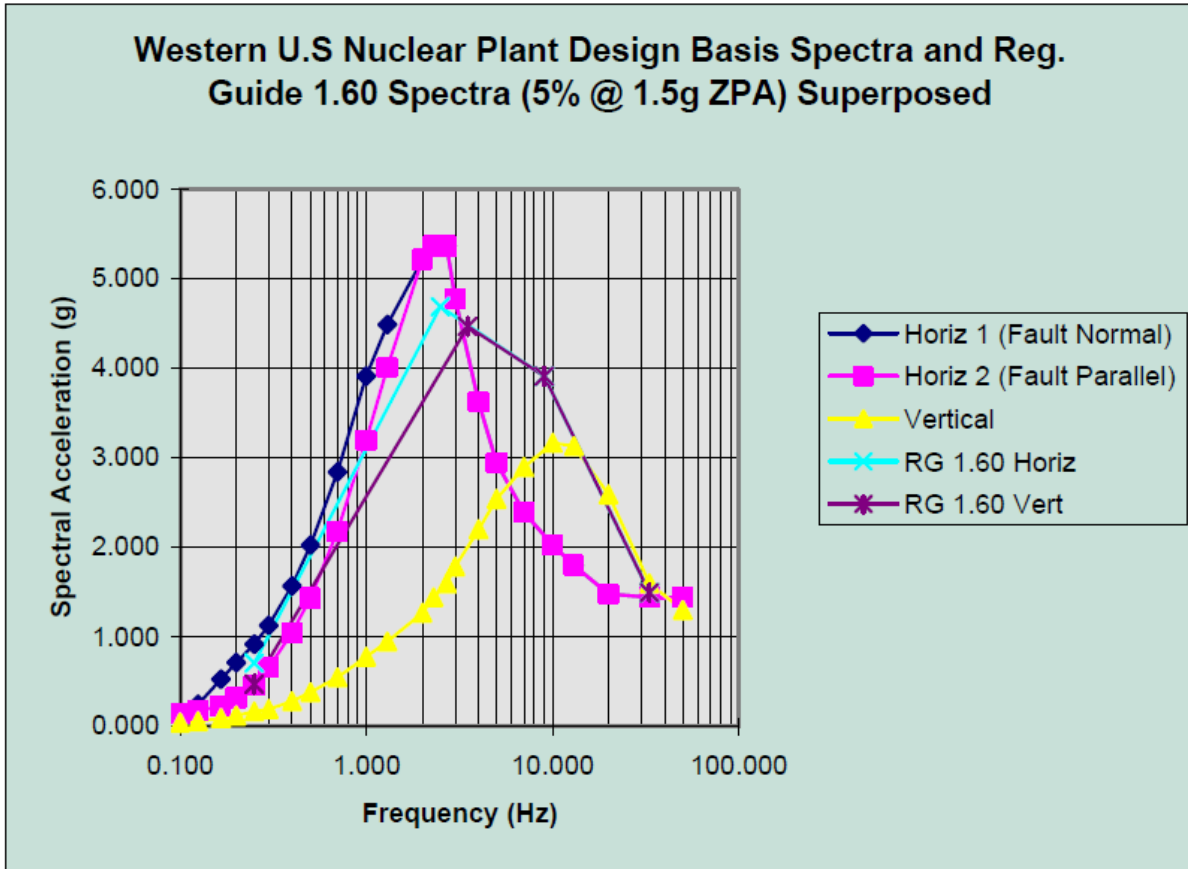
**FIGURE 3.4.27: HI-TRAC TRANSFER CASK IN SHORT-SIDE IMPACT  
(CASK RESTS AT A POSITION OF  $-5^{\circ}$  FROM HORIZONTAL)**



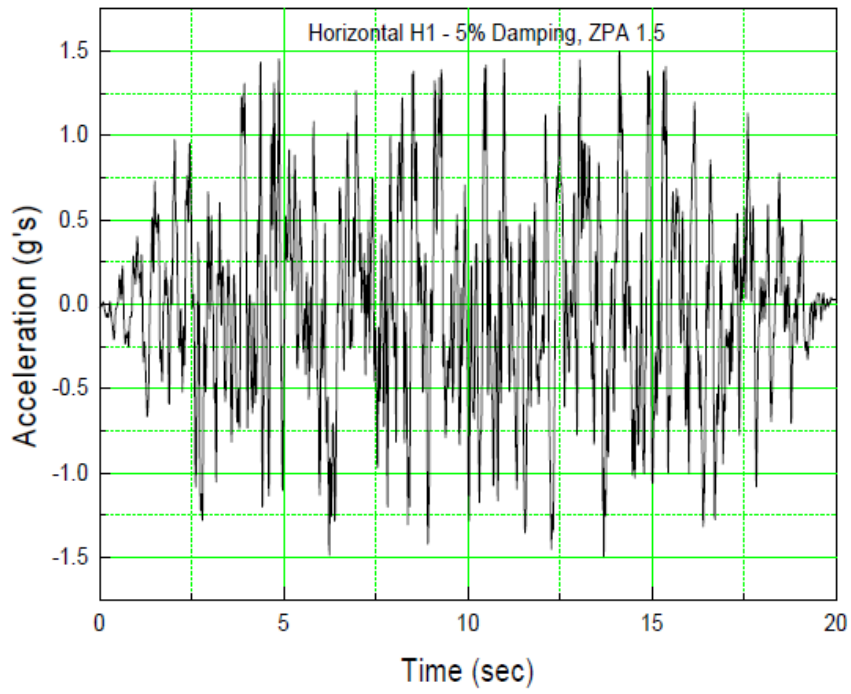
**FIGURE 3.4.28: HI-TRAC TRANSFER CASK IN LONG-SIDE IMPACT  
(CASK RESTS AT A POSITION OF  $-1^{\circ}$  FROM HORIZONTAL)**



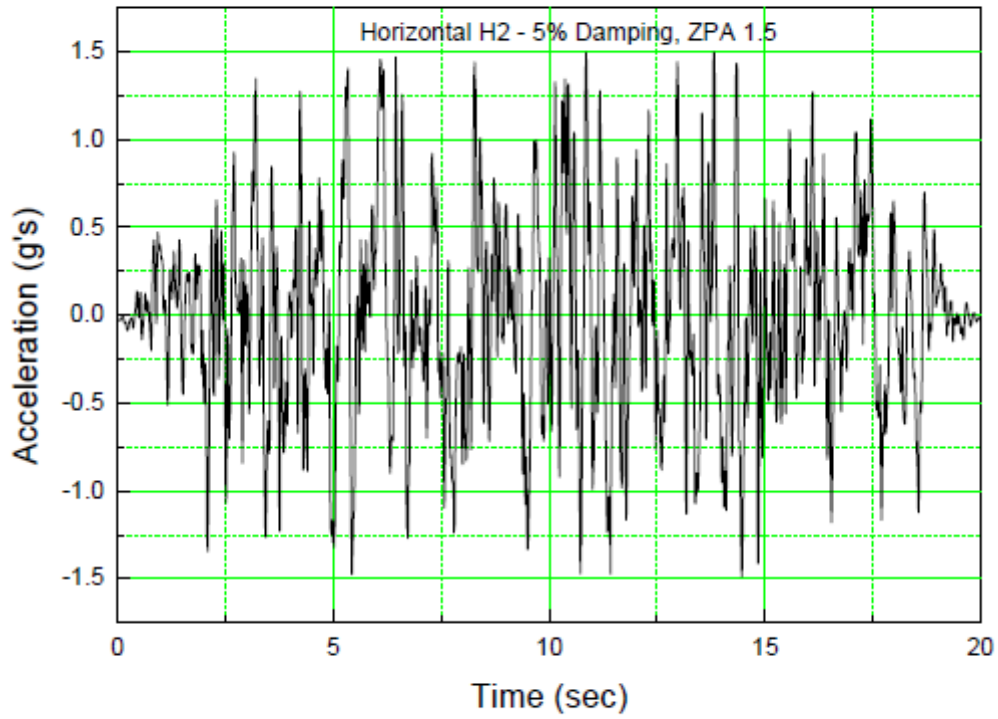
**FIGURE 3.4.29: FREE-BODY OF TRANSFER LID DURING PRIMARY IMPACT WITH TARGET**



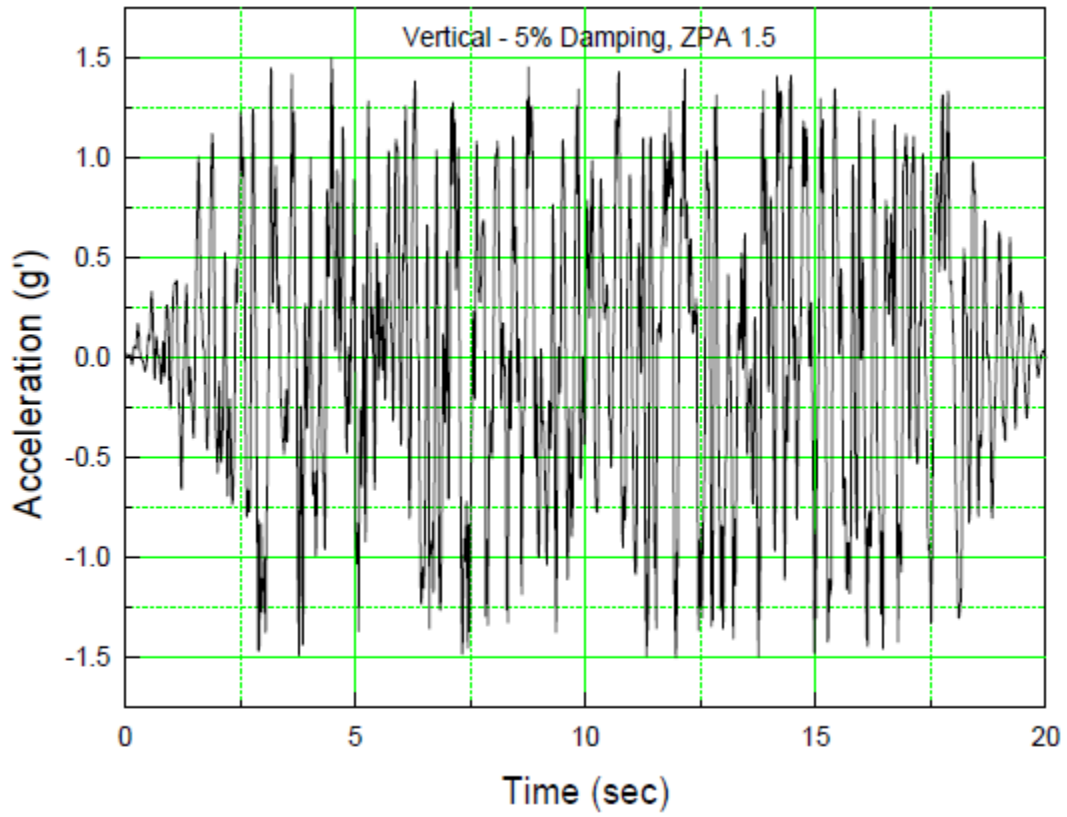
**FIGURE 3.4.30: SEISMIC SPECTRA SETS USED FOR TIME HISTORY ANALYSIS OF HI-STORM 100A ON ISFSI PAD**



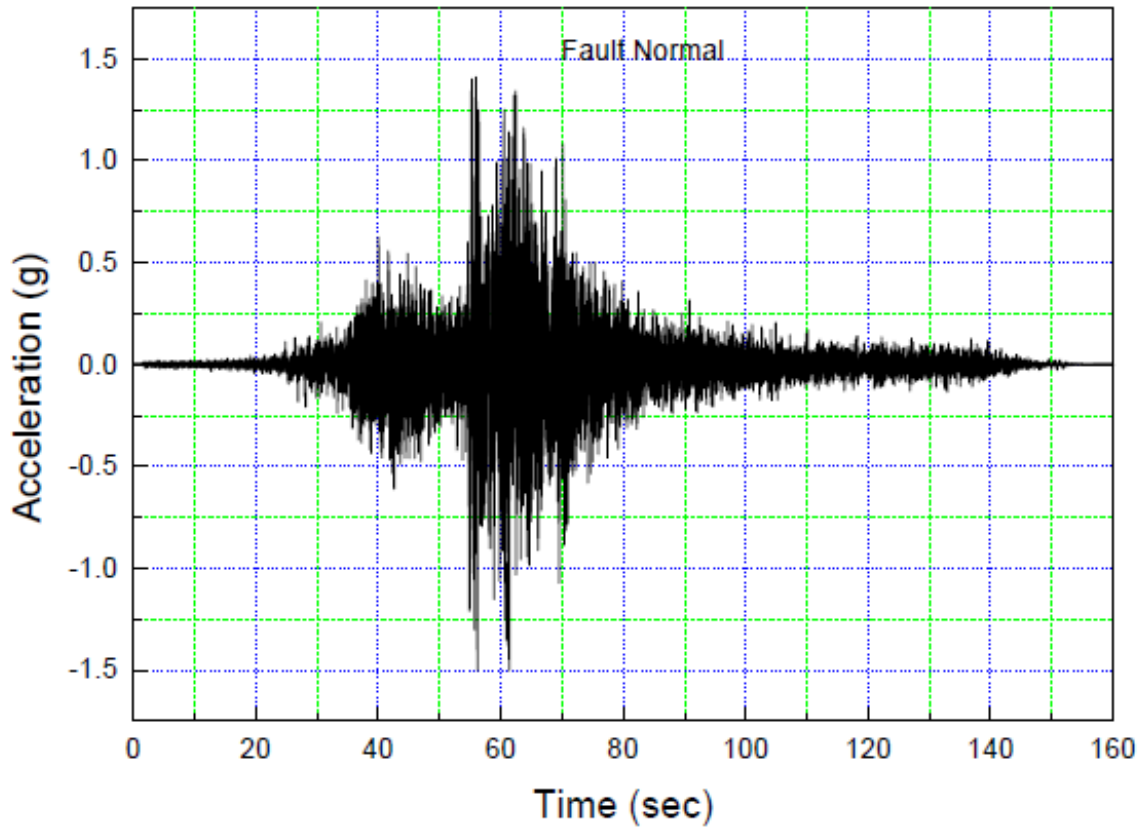
**FIGURE 3.4.31 - RG 1.60 "H1"**



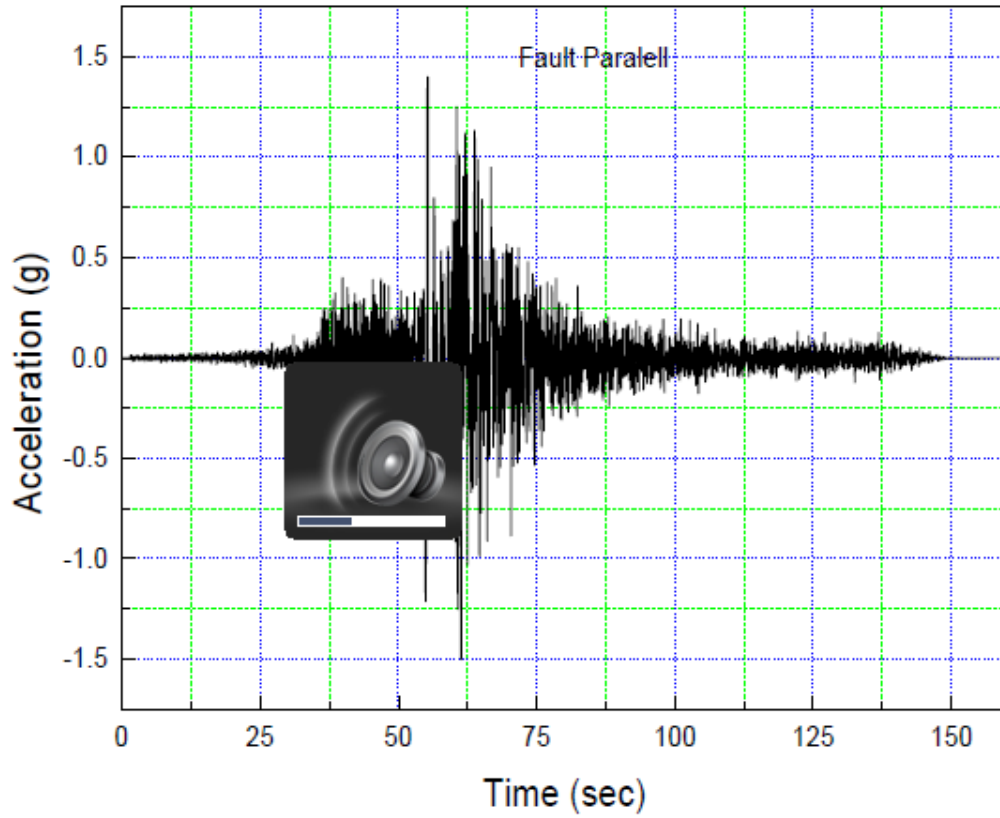
**FIGURE 3.4.32 - RG 1.60 "H2"**



**FIGURE 3.4.33 - RG 1.60 "VT"**

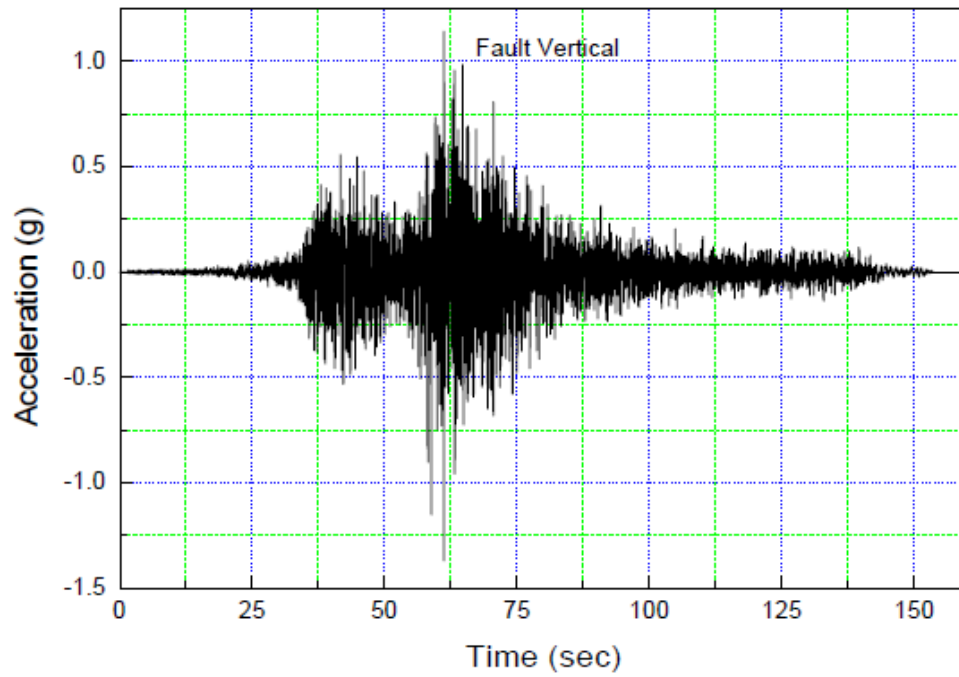


**FIGURE 3.4.34 Horizontal Acceleration Time history “FN”**

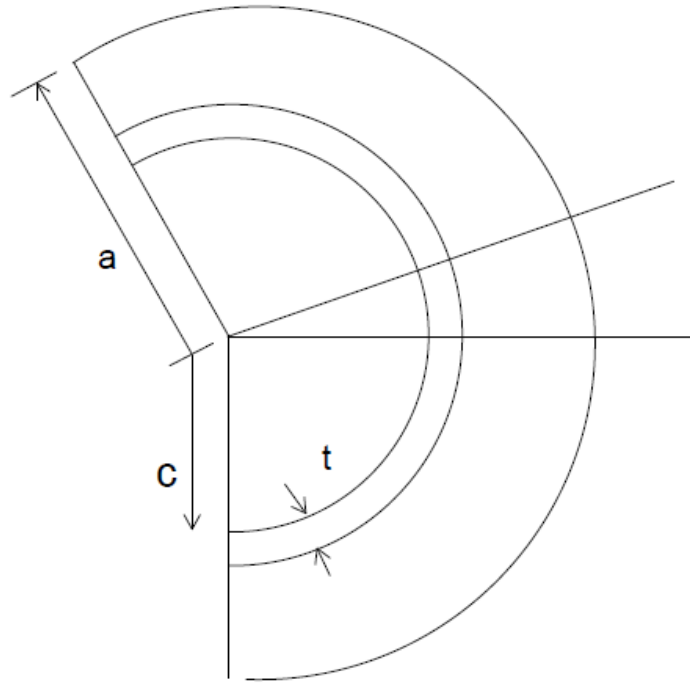


**FIGURE 3.4.35 Horizontal Acceleration Time history “FP”**

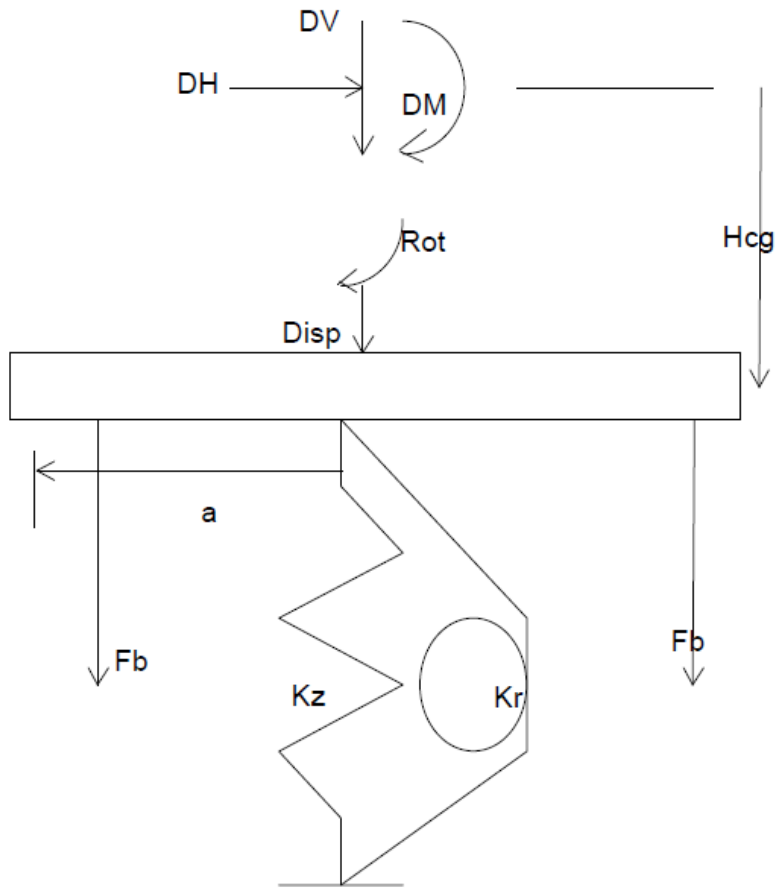




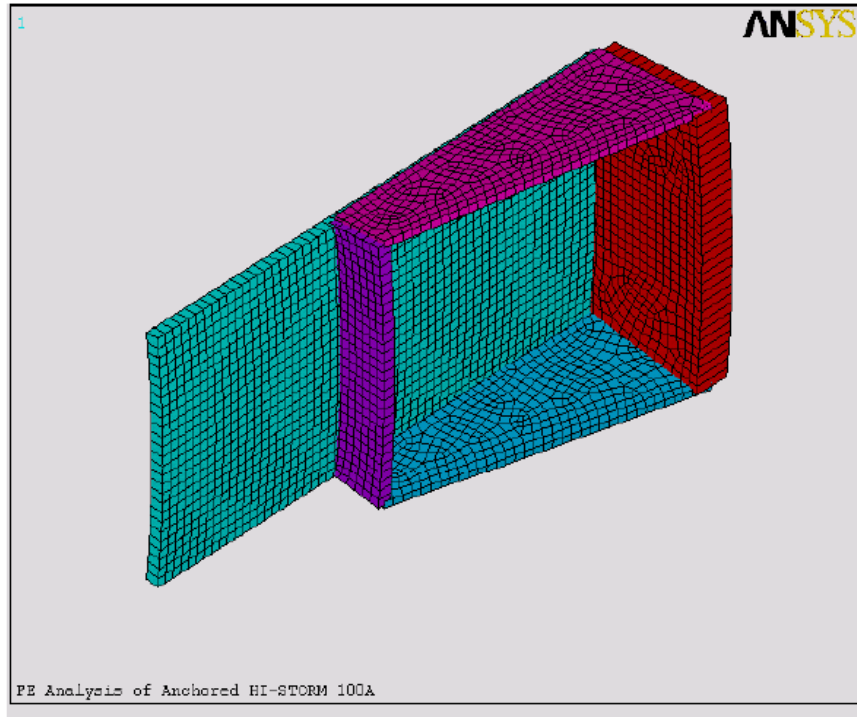
**FIGURE 3.4.36 Vertical Acceleration Time history “FV”**



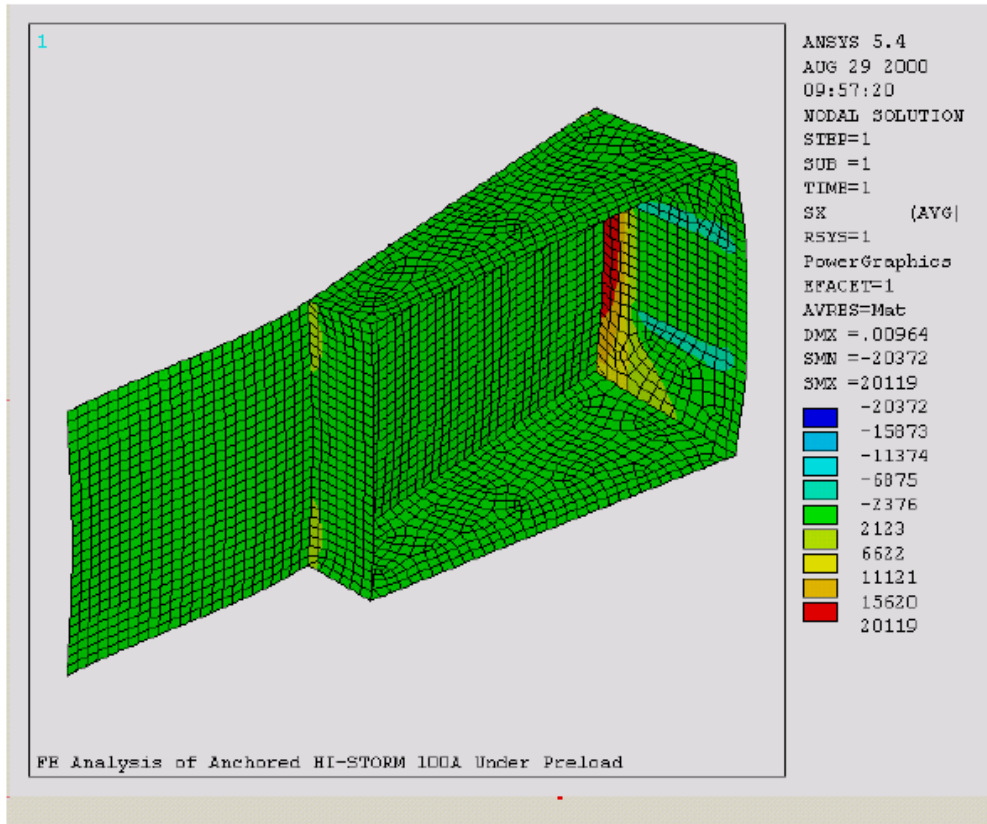
**FIGURE 3.4.37 GEOMETRY FOR QUASI-STATIC ANALYSIS**



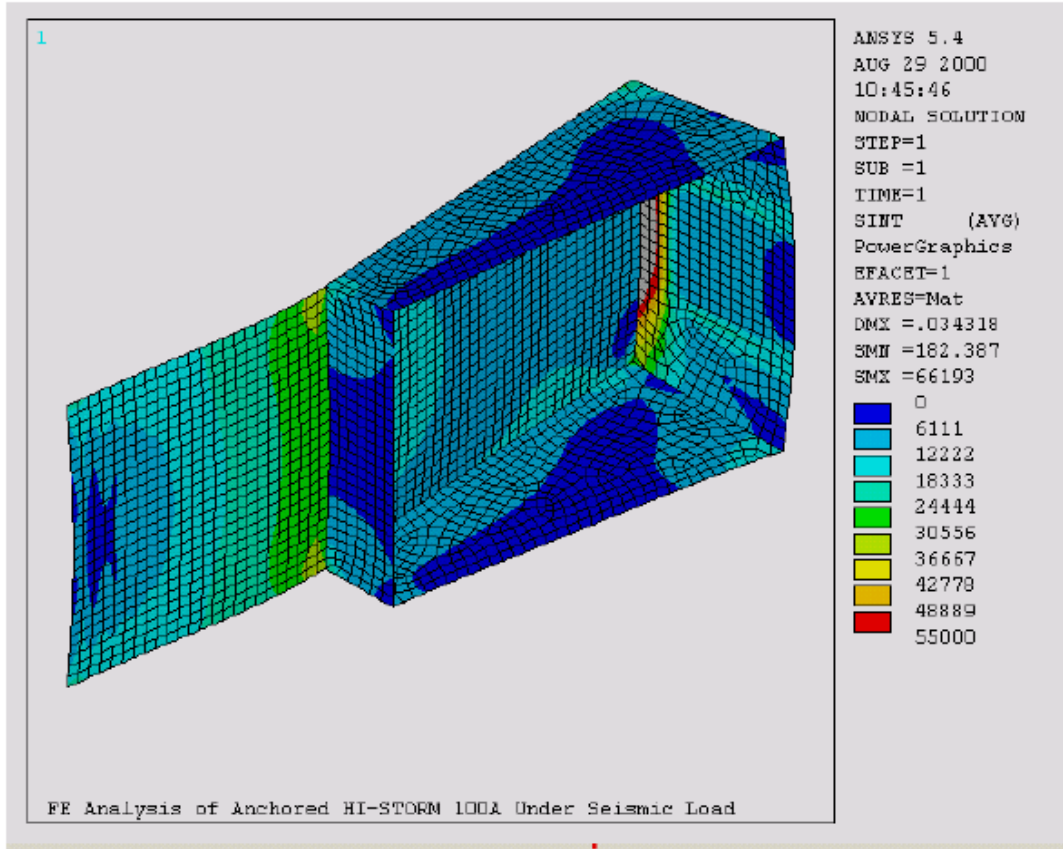
**FIGURE 3.4.38 FREE BODY FOR QUASI-STATIC ANALYSIS**



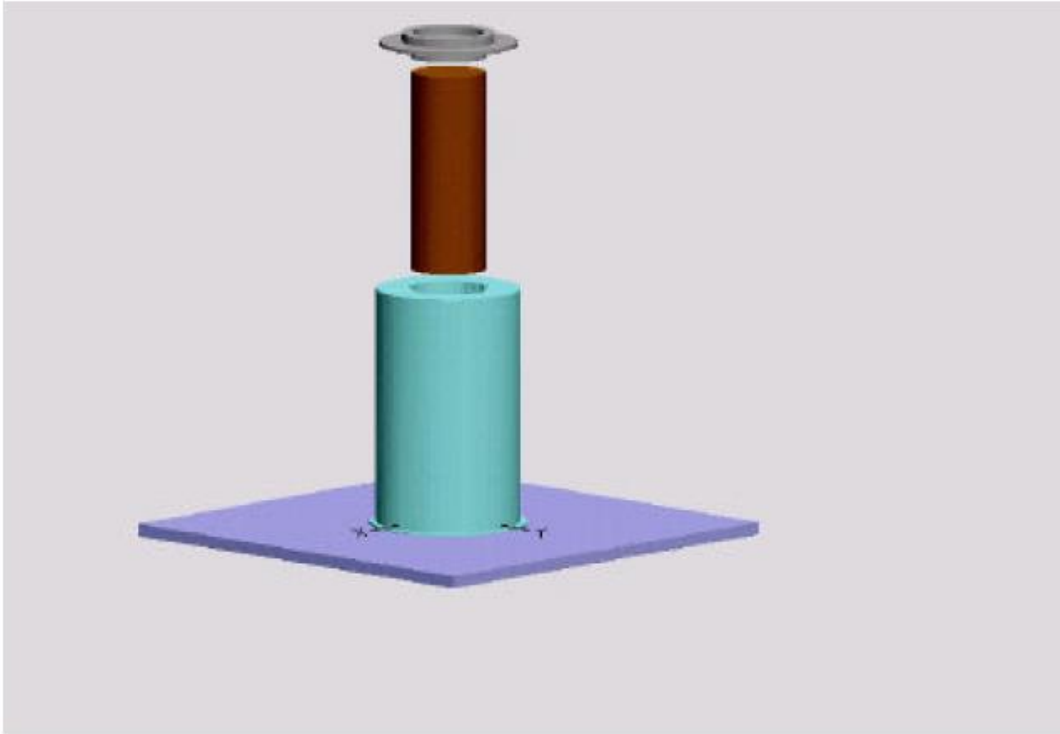
**FIGURE 3.4.39 SECTOR LUG FINITE ELEMENT MESH**



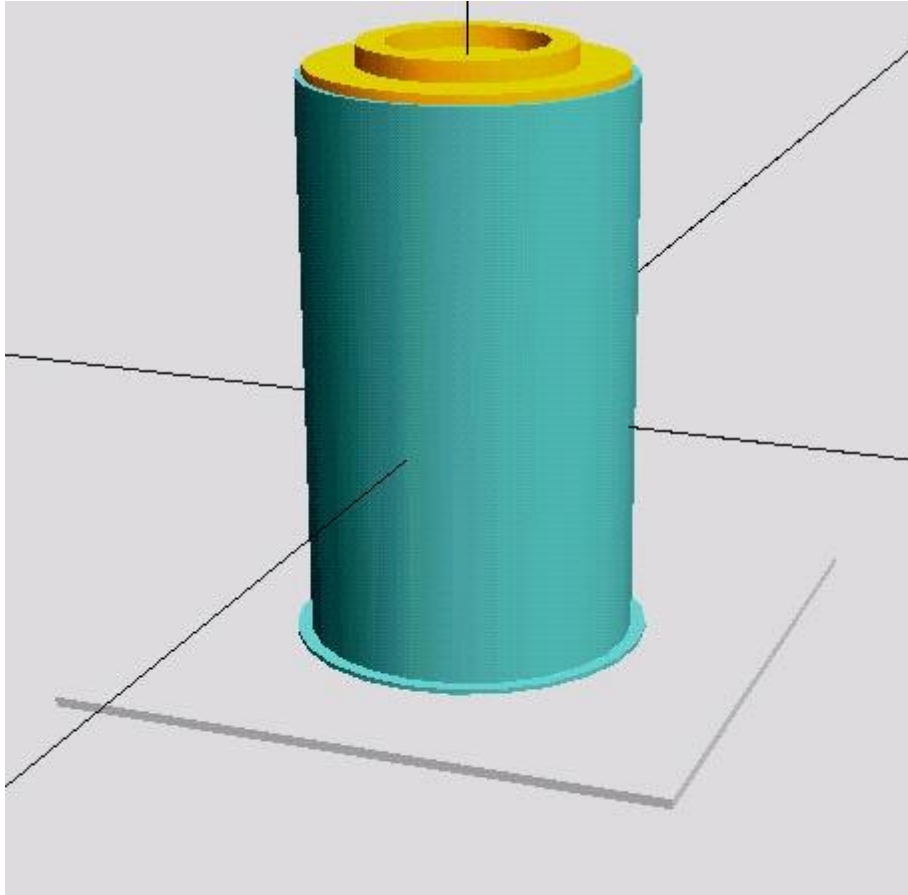
**FIGURE 3.4.40: SECTOR LUG STRESS – CASE 1 PRELOAD**



**FIGURE 3.4.41: SECTOR LUG STRESS INTENSITY – CASE 2 PRELOAD + SEISMIC**

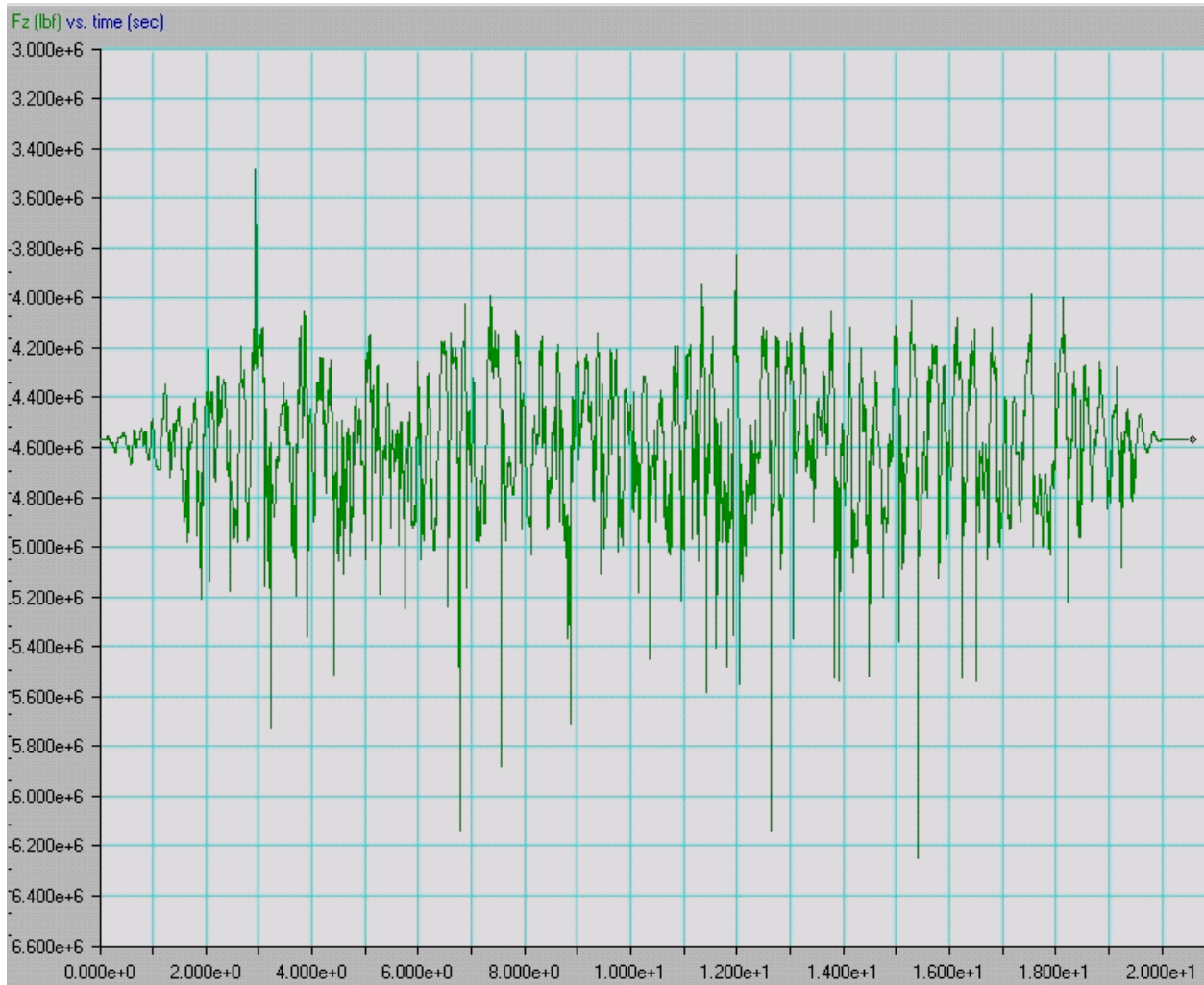


**FIGURE 3.4.42: EXPLODED VIEW SHOWING GROUND PLANE, OVERPACK, MPC, AND OVERPACK TOP LID**

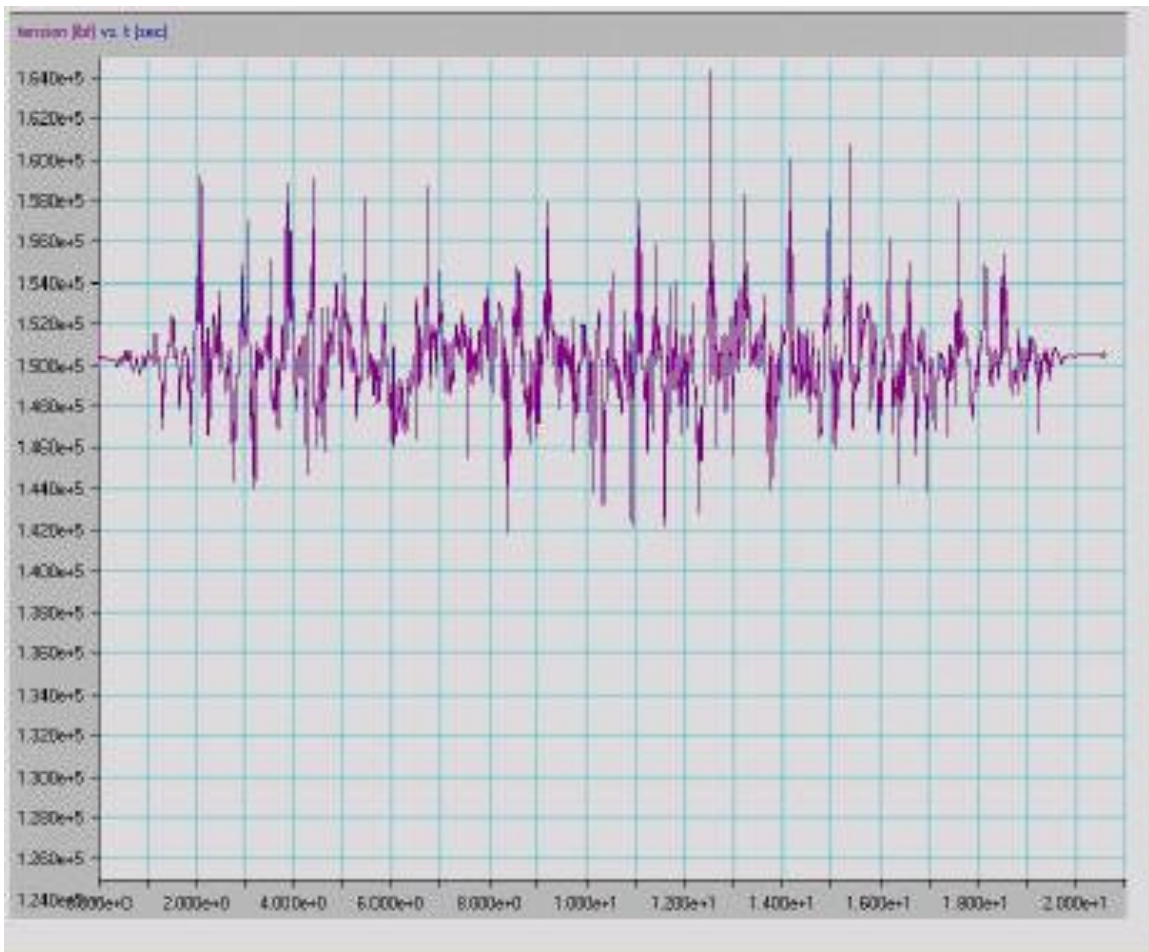


**FIGURE 3.4.43: VIEW OF ASSEMBLED HI-STORM ON PAD – MPC INSIDE AND TOP LID ATTACHED (Note Extended Baseplate for Anchor Connections)**

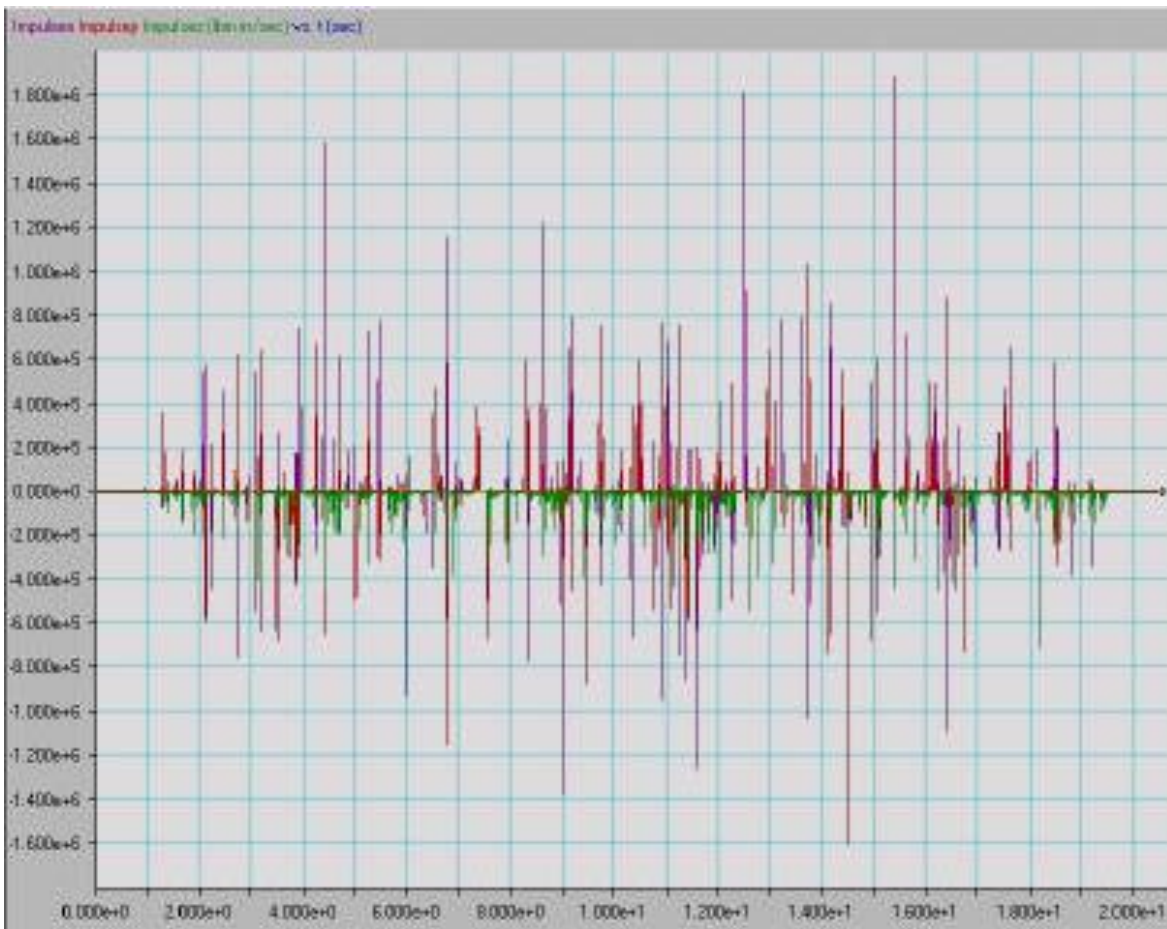




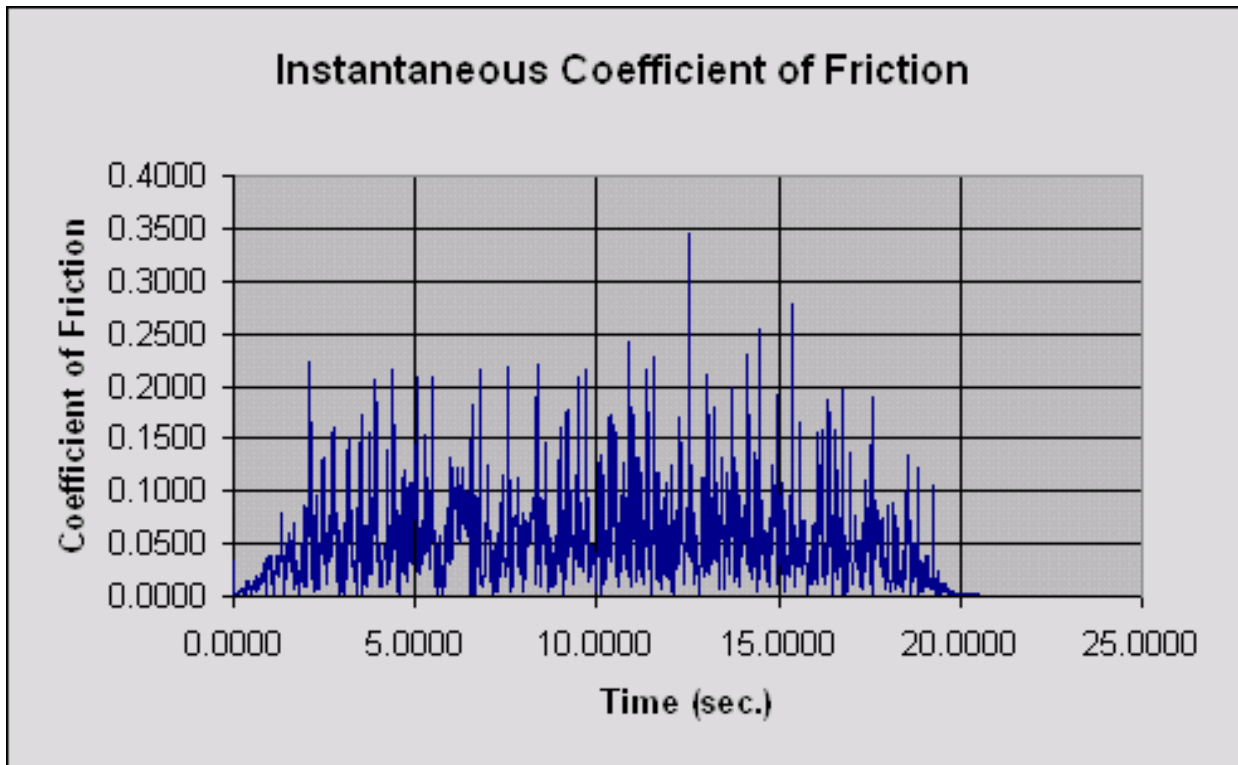
**FIGURE 3.4.44: VARIATION OF FOUNDATION RESISTANCE FORCE VS. TIME FOR REG. GUIDE 1.60 SEISMIC INPUT**



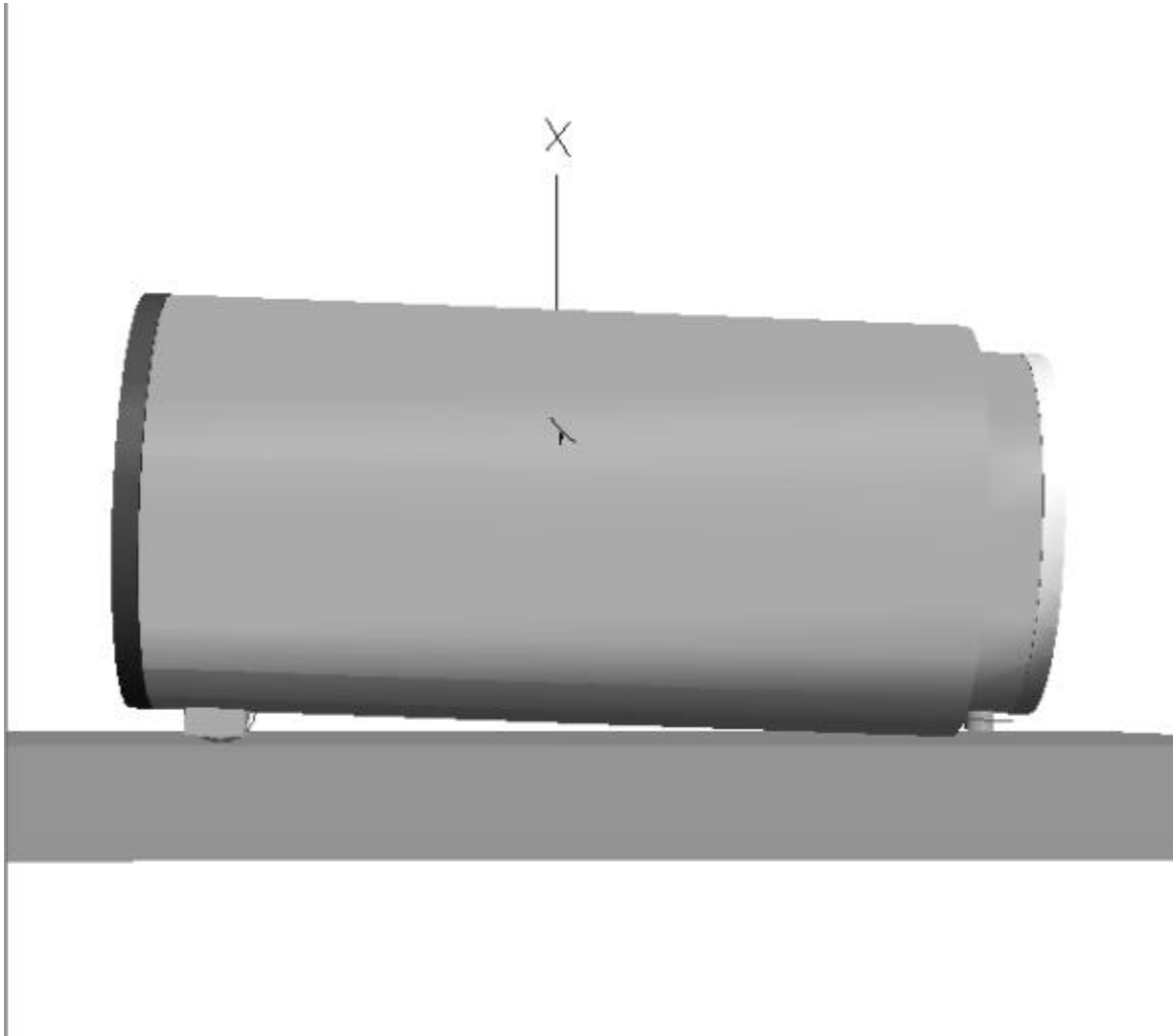
**FIGURE 3.4.45: VARIATION OF REPRESENTATIVE STUD TENSILE FORCE VS. TIME FOR REG. GUIDE 1.60 SEISMIC INPUT**



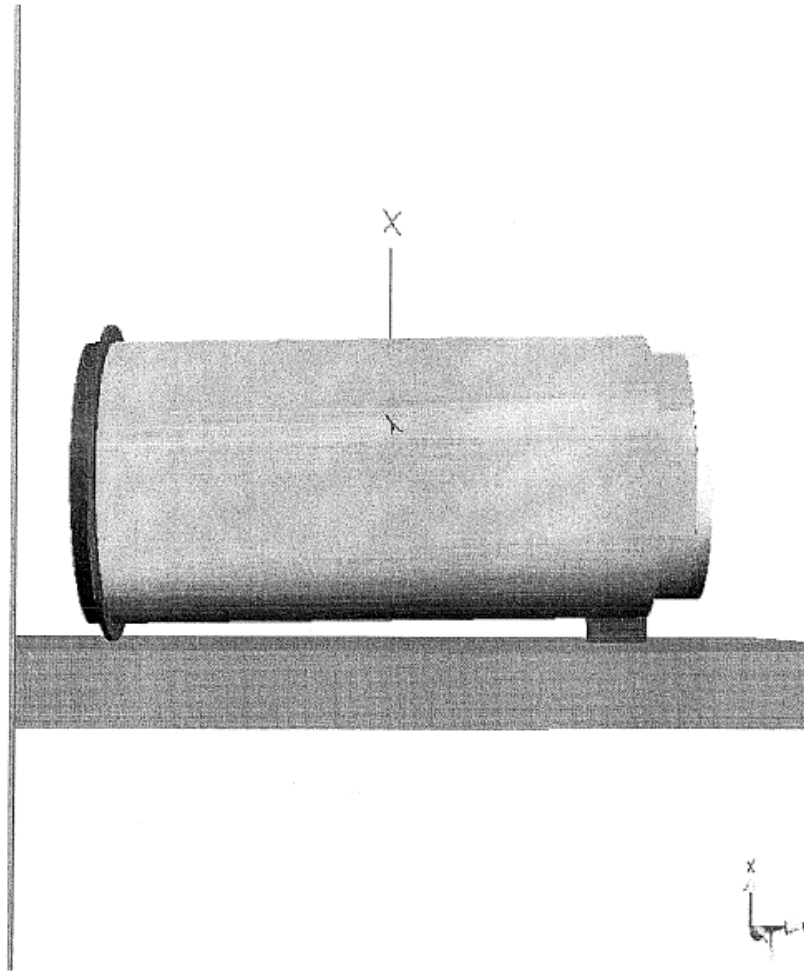
**FIGURE 3.4.46: MPC/Hi-STORM 100A IMPULSE VS. TIME – REG. GUIDE 1.60  
EVENT**



**FIGURE 3.4.47: INSTANTANEOUS CALCULATED COEFFICIENT OF FRICTION – REG. GUIDE 1.60 EVENT**



**FIGURE 3.4.48: HI-TRAC 125 BENCHMARK SIMULATION OF DROP SCENARIO A**



**FIGURE 3.4.49: SIMULATION OF HI-TRAC 125D 42" HORIZONTAL DROP WITH PRIMARY IMPACT AT TOP END RADIAL SUPPORT TAB**

### 3.5 FUEL RODS

The regulations governing spent fuel storage cask approval and fabrication (10 CFR 72.236) require that a storage cask system “will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions” (§72.236(l)). Per Regulatory Guide 3.61, Section 3.5, “When fuel cladding is considered in the design criteria for confinement of radioactive material under normal or accident condition, provide an analysis or test results showing that the cladding will maintain its integrity.” Although the cladding of intact fuel rods does provide a barrier against the release of radioactive fission products, the confinement evaluation for the HI-STORM System (Chapter 7) takes no credit for fuel cladding integrity in satisfying the regulatory confinement requirement.

As described in Section 7.1, the confinement boundary in the HI-STORM System consists of the MPC Enclosure Vessel. The Enclosure Vessel is designed and, to the extent practicable, manufactured in accordance with the most stringent ASME B&PV Code (Section III, Subsection NB). As required by NB, all materials are 100% UT inspected and all butt welds are subjected to 100% volumetric inspection. The field closure features redundant barriers (the MPC lid and port cover plates are the primary barriers, the closure ring is the secondary barrier). Section 7.1 further describes that the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 [7.1.2] such that leakage from the confinement boundary may be considered non-credible. Section 7.2 addresses confinement for normal and off-normal conditions, and states “Since the MPC confinement vessel remains intact, and the design basis temperatures and pressure are not exceeded, leakage from the MPC confinement boundary is not credible”. Confinement for accident conditions is addressed in Section 7.3, which states “there is no mechanistic failure that results in a breach of, and associated leakage of radioactive material from the MPC confinement boundary”.

The assured integrity of the MPC Confinement Boundary eliminates the reliance on the fuel cladding to prevent release of radiological matter to the environment. Since the cladding is not considered as part of the confinement during normal, off-normal, or accident conditions, there is no need for providing an analysis for computing the allowable g-load for the fuel rods to demonstrate cladding integrity and none is included in this FSAR.

### 3.6 SUPPLEMENTAL DATA

#### 3.6.1 Additional Codes and Standards Referenced in HI-STORM 100 System Design and Fabrication

The following additional codes, standards and practices were used as aids in developing the design, manufacturing, quality control and testing methods for HI-STORM 100 System:

##### a. Design Codes

- (1) AISC Manual of Steel Construction, 1964 Edition and later.
- (2) ANSI N210-1976, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations".
- (3) American Concrete Institute Building Code Requirements for Structural Concrete, ACI-318-95.
- (4) Code Requirements for Nuclear Safety Related Concrete Structures, ACI349-85/ACI349R-85, ACI349.1R-80, and ACI 349-97.
- (5) ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities.
- (6) ASME NQA-2-1989, Quality Assurance Requirements for Nuclear Facility Applications.
- (7) ANSI Y14.5M, Dimensioning and Tolerancing for Engineering Drawings and Related Documentation Practices.
- (8) ACI Detailing Manual - 1980.
- (9) Crane Manufacturer's Association of America, Inc., CMAA Specification #70, Specifications for Electric Overhead Traveling Cranes, Revised 1988.

##### b. Material Codes - Standards of ASTM

- (1) E165 - Standard Methods for Liquid Penetrant Inspection.
- (2) A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.



- (3) A262 - Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.
  - (4) A276 - Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.
  - (5) A479 - Steel Bars for Boilers & Pressure Vessels.
  - (6) A564, Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes.
  - (7) C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.
  - (8) A380 - Recommended Practice for Descaling, Cleaning and Marking Stainless Steel Parts and Equipment.
  - (9) C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.
  - (10) E3, Preparation of Metallographic Specimens.
  - (11) E190, Guided Bend Test for Ductility of Welds.
  - (12) NCA3800 - Metallic Material Manufacturer's and Material Supplier's Quality System Program.
- c. Welding Codes: ASME Boiler and Pressure Vessel Code, Section IX - Welding and Brazing Qualifications, 1995 Edition.
- d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling Requirements
- (1) ANSI 45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants.
  - (2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase).
  - (3) ANSI - N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants (Regulatory Guide 1.58).

- (4) ANSI-N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants.
- (5) ANSI - N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants.
- (6) ANSI-N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants.
- (7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Equipment Materials and Services for Nuclear Power Plants (Regulatory Guide 1.123).
- (8) ANSI N45.2.15-18 - Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants.
- (9) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Regulatory Guide 1.146).
- (10) ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1995 Edition.
- (11) ANSI - N16.9-75 Validation of Calculation Methods for Nuclear Criticality Safety.

e. Reference NRC Design Documents

- (1) NUREG-0800, Radiological Consequences of Fuel Handling Accidents.
- (2) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.
- (3) NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, January 1997, Final Report.

f. Other ANSI Standards (not listed in the preceding)

- (1) 8.1 (N16.1) - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.
- (2) 8.17, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.
- (3) N45.2 - Quality Assurance Program Requirements for Nuclear Facilities - 1971.

- (4) N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.
- (5) N45.2.10 - Quality Assurance Terms and Definitions - 1973.
- (6) 57.2 (N210) - Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.
- (7) N14.6 (1993) - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials.
- (8) N626-3, Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code Section III, Div. 1, Certifying Activities.

g. Code of Federal Regulations

- (1) 10CFR20 - Standards for Protection Against Radiation.
- (2) 10CFR21 - Reporting of Defects and Non-compliance.
- (3) 10CFR50 - Appendix A - General Design Criteria for Nuclear Power Plants.
- (4) 10CFR50 - Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.
- (5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Material.
- (6) 10CFR71 - Packaging and Transportation of Radioactive Material.

h. Regulatory Guides

- (1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).
- (2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.
- (3) RG 1.28 - (ANSI N45.2) - Quality Assurance Program Requirements.
- (4) RG 1.29 - Seismic Design Classification (Rev. 3).
- (5) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Material.

- (6) RG 1.38 - (ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants.
- (7) RG 1.44 - Control of the Use of Sensitized Stainless Steel.
- (8) RG 1.58 - (ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel.
- (9) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.
- (10) RG 1.64 - (ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants.
- (11) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility.
- (12) RG 1.74 - (ANSI N45.2.10) Quality Assurance Terms and Definitions.
- (13) RG 1.85 - Materials Code Case Acceptability - ASME Section 3, Div. 1.
- (14) RG 1.88 - (ANSI N45.2.9) Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records.
- (15) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis.
- (16) RG 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.
- (17) RG 1.123 - (ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.
- (18) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, 1978.
- (19) Reg. Guide 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.
- (20) RG 3.41 - Validation of Calculational Methods for Nuclear Criticality Safety, Revision 1, 1977.
- (21) Reg. Guide 8.8 - Information Relative to Ensuring that Occupational Radiation

Exposure at Nuclear Power Plants will be as Low as Reasonably Achievable (ALARA).

- (22) DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants".

i. Branch Technical Position

- (1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.
- (2) ASB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

j. Standard Review Plan (NUREG-0800)

- (1) SRP 3.2.1 - Seismic Classification.
- (2) SRP 3.2.2 - System Quality Group Classification.
- (3) SRP 3.7.1 - Seismic Design Parameters.
- (4) SRP 3.7.2 - Seismic System Analysis.
- (5) SRP 3.7.3 - Seismic Subsystem Analysis.
- (6) SRP 3.8.4 - Other Seismic Category I Structures (including Appendix D), Technical Position on Spent Fuel Rack.
- (7) SRP 3.8.5 - Foundations
- (8) SRP 9.1.2 - Spent Fuel Storage, Revision 3, 1981.
- (9) SRP 9.1.3 - Spent Fuel Pool Cooling and Cleanup System.
- (10) SRP 9.1.4 - Light Load Handling System.
- (11) SRP 9.1.5 - Overhead Heavy Load Handling System.
- (12) SRP 15.7.4 - Radiological Consequences of Fuel Handling Accidents.

k. AWS Standards

- (1) D1.1 - Structural Welding Code, Steel.

- (2) A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.
- (3) A3.0 - Standard Welding Terms and Definitions.
- (4) A5.12 - Tungsten Arc-welding Electrodes.
- (5) QC1 - Standards and Guide for Qualification and Certification of Welding Inspectors.

1. Others

- (1) ASNT-TC-1A - Recommended Practice for Nondestructive Personnel Qualification and Certification.
- (2) SSPC SP-2 - Surface Preparation Specification No. 2 Hand Tool Cleaning.
- (3) SSPC SP-3 - Surface Preparation Specification No. 3 Power Tool Cleaning.
- (4) SSPC SP-10 - Near-White Blast Cleaning.

3.6.2 Computer Programs

Three computer programs, all with a well established history of usage in the nuclear industry, have been utilized to perform structural and mechanical analyses documented in this report. These codes are ANSYS, DYNA3D, and WORKING MODEL. ANSYS is a public domain code which utilizes the finite element method for structural analyses.

WORKING MODEL, Version V.3.0/V.4.0

This code is used in this 10CFR72 submittal to compute the dynamic load resulting from intermediate missile impact on the overpack closure and to evaluate the maximum elastic spring rate associated with the target during a HI-TRAC handling accident event.

WORKING MODEL has been previously utilized in similar dynamic analyses of the HI-STAR 100 system (Docket No. 72-1008).

"WORKING MODEL" (V3.0/V4.0) is a Computer Aided Engineering (CAE) tool with an integrated user interface that merges modeling, simulation, viewing, and measuring. The program includes a dynamics algorithm that provides automatic collision and contact handling, including detection, response, restitution, and friction.

Numerical integration is performed using the Kutta-Merson integrator which offers options for variable or fixed time-step and error bounding.

The Working Model Code is commercially available. Holtec has performed independent QA validation of the code (in accordance with Holtec's QA requirements) by comparing the solution of several classical dynamics problems with the numerical results predicted by Working Model. Agreement in all cases is excellent.

Additional theoretical material is available in the manual: "Users Manual, Working Model, Version 3", Knowledge Revolution, 66 Bovet Road, Suite 200, San Mateo, CA, 94402.

This code has been acquired by MSC Software and has now been designated "VisualNastran Desktop". The most current version, which has been used in this revision, is VN 2003. The descriptions given above are still valid.

### DYNA3D

"DYNA3D" is a nonlinear, explicit, three-dimensional finite element code for solid and structural mechanics. It was originally developed at Lawrence Livermore Laboratories and is ideally suited for study of short-time duration, highly nonlinear impact problems in solid mechanics. DYNA3D is commercially available for both UNIX work stations and Pentium class PCs running Windows 95 or Windows NT. The PC version has been fully validated at Holtec following Holtec's QA procedures for commercial computer codes. This code is used to analyze the drop accidents and the tip-over scenario for the HI-STORM 100. Benchmarking of DYNA3D for these storage analyses is discussed and documented in Appendix 3.A. DYNA3D is also known as LS-DYNA and is currently supported and distributed by Livermore Software. Each update is independently subject to QA validation.

#### 3.6.3 Appendices Included in Chapter 3

##### 3.A HI-STORM Deceleration Under Postulated Vertical Drop Event and Tipover

#### 3.6.4 Calculation Packages

In addition to the calculations presented in Chapter 3, supporting calculation packages have been prepared to document other information pertinent to the analyses. As new components are added (e.g., the HI-STORM 100S versions and additional MPC's), supporting calculation packages back up the summary results reported herein.

The calculation packages contain additional details on component weights, supporting calculations for some results summarized in the chapter, and miscellaneous supporting data that supplements the results summarized in Chapter 3 of the FSAR. All of the finite element tabular data, node and element data, supporting figures, and numerical output for all fuel baskets are contained in the calculation package supplement supporting Revision 1 of the FSAR.

### 3.7 COMPLIANCE WITH NUREG-1536

Supporting information to provide reasonable assurance with respect to the adequacy of the HI-STORM 100 System to store spent nuclear fuel in accordance with the stipulations of the Technical Specifications (Chapter 12) is provided throughout this FSAR. An itemized table (Table 3.0.1 at the beginning of this chapter) has been provided to locate and collate the substantiating material to support the technical evaluation findings listed in NUREG-1536 Chapter 3, Article VI.

The following statements are germane to an affirmative safety evaluation:

- The design and structural analysis of the HI-STORM 100 System is in full compliance with the provisions of Chapter 3 of NUREG-1536 except as listed in the Table 1.0.2 (list of code compliance exceptions).
- The list of Regulatory Guides, Codes, and standards presented in Section 3.6 herein is in full compliance with the provisions of NUREG-1536.
- All HI-STORM 100 structures, systems, and components (SSC) that are important to safety (ITS) are identified in Table 2.2.6. Section 1.5 contains the design drawings that describe the HI-STORM 100 SSCs in complete detail. Explanatory narrations in Subsections 3.4.3 and 3.4.4 provide sufficient textual details to allow an independent evaluation of their structural effectiveness.
- The requirements of 10CFR72.24 with regard to information pertinent to structural evaluation is provided in Chapters 2, 3, and 11.
- Technical Specifications pertaining to the structures of the HI-STORM 100 System have been provided in Section 12.3 herein pursuant to the requirements of 10CFR72.26.
- A series of analyses to demonstrate compliance with the requirements of 10CFR72.122(b) and (c), and 10CFR72.24(c)(3) have been performed which show that SSCs designated as ITS possess an adequate margin of safety with respect to all load combinations applicable to normal, off-normal, accident, and natural phenomenon events. In particular, the following information is provided:
  - i. Load combinations for the fuel basket, enclosure vessel, and the HI-STORM 100/HI-TRAC overpacks for normal, off-normal, accident, and natural phenomenon events are compiled in Tables 2.2.14, 3.1.1, and 3.1.3 through 3.1.5, respectively.
  - ii. Stress limits applicable to the materials are found in Subsection 3.3.
  - iii. Stresses at various locations in the fuel basket, the enclosure vessel, and the HI-STORM 100/HI-TRAC overpacks have been computed by analysis.



Descriptions of stress analyses are presented in Sections 3.4.3 and 3.4.4.

- iv. Factors of safety in the components of the HI-STORM 100 System are reported as below:
- |    |                                   |                                       |
|----|-----------------------------------|---------------------------------------|
| a. | Fuel basket                       | Tables 3.4.3 and 3.4.6                |
| b. | Enclosure vessel                  | Tables 3.4.4, 3.4.6, 3.4.7, and 3.4.8 |
| c. | HI-STORM 100 overpack/<br>HI-TRAC | Table 3.4.5                           |
| d. | Miscellaneous<br>components       | Table 3.4.9                           |
| e. | Lifting devices                   | Subsection 3.4.3                      |
- The structural design and fabrication details of the fuel baskets whose safety function in the HI-STORM 100 System is to maintain nuclear criticality safety, have been carried out to comply with the provisions of Subsection NG of the ASME Code (loc. cit.) Section III. The structural factors of safety, summarized in Tables 3.4.3 and 3.4.6 for all credible load combinations under normal, off-normal, accident, and natural phenomenon events demonstrate that the Code limits are satisfied in all cases. As the stress analyses have been performed using linear elastic methods and the computed stresses are well within the respective ASME Code limits, it follows that the physical geometry of the fuel basket will not be altered under any load combination to create a condition adverse to criticality safety. This conclusion satisfies the requirement of 10CFR72.124(a), with respect to structural margins of safety for SSCs important to nuclear criticality safety.
  - Structural margins of safety during handling, packaging, and transfer operations, mandated by the provisions of 10CFR Part 72.236(b), require that the lifting and handling devices are engineered to comply with the stipulations of ANSI N14.6, NUREG-0612, Regulatory Guide 3.61, and NUREG-1536, and that the components being handled meet the applicable ASME Code service condition stress limits. The requirements of the governing codes for handling operations are summarized in Subsection 3.4.3 herein. A summary table of factors of safety for all ITS components under lifting and handling operations, presented in Subsection 3.4.3, shows that adequate structural margins exist in all cases.
  - Consistent with the requirements of 10CFR72.236(i), the confinement boundary for the HI-STORM 100 System has been engineered to maintain confinement of radioactive materials under normal, off-normal, and postulated accident conditions. This assertion of confinement integrity is made on the strength of the following

information provided in this FSAR.

- i. The MPC Enclosure Vessel which constitutes the confinement boundary is designed and fabricated in accordance with Section III, Subsection NB (Class 1 nuclear components) of the ASME Code to the maximum extent practicable.
  - ii. The MPC lid of the MPC Enclosure Vessel is welded using a strength groove weld and is subjected to volumetric examination or multiple liquid penetrant examinations, pressure testing, and liquid penetrant (root and final) testing to establish a maximum confidence in weld joint integrity.
  - iii. The closure of the MPC Enclosure Vessel consists of *two* independent isolation barriers.
  - iv. The confinement boundary is constructed from stainless steel alloys with a proven history of material integrity under environmental conditions.
  - v. The load combinations for normal, off-normal, accident, and natural phenomena events have been compiled (Table 2.2.14) and applied on the MPC Enclosure Vessel (confinement boundary). The results, summarized in Tables 3.4.4 through 3.4.9, show that the factor of safety (with respect to the appropriate ASME Code limits) is greater than one in all cases. Design Basis natural phenomena events such as tornado-borne missiles (large, intermediate, or small) have also been analyzed to evaluate their potential for breaching the confinement boundary. Analyses presented in Subsection 3.4.8, and summarized in unnumbered tables in Subsection 3.4.8, show that the integrity of the confinement boundary is preserved under all design basis projectile impact scenarios.
- The information on structural design included in this FSAR complies with the requirements of 10CFR72.120 and 10CFR72.122, and can be ascertained from the information contained in Table 3.7.1.
  - The provisions of features in the HI-STORM 100 structural design, listed in Table 3.7.2, demonstrate compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

Item	Compliance	Location of Supporting Information in This Document
i. Design and fabrication to acceptable quality standards	<p>All ITS components designed and fabricated to recognized Codes and Standards:</p> <ul style="list-style-type: none"> <li>• Basket: Subsection NG, Section III</li> <li>• Enclosure Vessel: Subsection NB, loc. cit.</li> <li>• HI-STORM 100 Subsection NF, loc. cit.</li> </ul> <p>Structure:</p> <ul style="list-style-type: none"> <li>• HI-TRAC Structure: Subsection NF, loc. cit.</li> </ul>	<p>Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7</p> <p>Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7</p> <p>Subsections 2.0.2 and 3.1.1</p> <p>Subsections 2.0.3 and 3.1.1</p>
ii. Erection to acceptable quality standards	<ul style="list-style-type: none"> <li>• Concrete in HI-STORM 100 meets requirements of : ACI –349(85)</li> </ul>	<p>Appendix 1.D Subsection 3.3.2</p>
iii. Testing to acceptable quality standards	<ul style="list-style-type: none"> <li>• All non-destructive examination of ASME Code components for provisions in the Code (see exceptions in Table 2.2.15).</li> <li>• Pressure test of pressure vessel per the Code.</li> <li>• Testing for radiation containment per provisions of NUREG-1536</li> </ul>	<p>Section 9.1</p> <p>Section 9.1</p> <p>Sections 7.1 and 9.1</p>

Item	Compliance	Location of Supporting Information in This Document
	<ul style="list-style-type: none"> <li>• Concrete testing in accordance with ACI-349(85)</li> </ul>	Appendix 1.D
iv. Adequate structural protection against environmental conditions and natural phenomena.	Analyses presented in Chapter 3 demonstrate that the confinement boundary will preserve its integrity under all postulated off-normal and natural phenomena events listed in Chapters 2.	Section 2.2 Chapter 11
v. Adequate protection against fires and explosions	<ul style="list-style-type: none"> <li>• The extent of combustible (exothermic) material in the vicinity of the cask system is procedurally controlled (the sole source of hydrocarbon energy is diesel in the tow vehicle).</li> <li>• Analyses show that the heat energy released from the postulated fire accident condition surrounding the cask will not result in impairment of the confinement boundary and will not lead to structural failure of the overpack. The effect on shielding will be localized to the external surfaces directly exposed to the fire which will result in a loss of the water in the water jacket for the HI-TRAC, and no significant change in the HI-STORM 100 overpack.</li> <li>• Explosion effects are shown to be bounded by the Code external pressure design basis and there is no adverse effect on ready retrievability of the MPC.</li> </ul>	Subsections 12.3.20 and 12.3.21  Subsection 11.2.4  Subsection 11.2.11 and Subsection 3.1.2.1.1.4; 3.4.7
vi. Appropriate inspection, maintenance, and testing	Inspection, maintenance, and testing requirements set forth in this FSAR are in full compliance with the governing regulations and established industry practice.	Sections 9.1 and 9.2 Chapter 12
vii. Adequate accessibility in emergencies.	<p>The HI-STORM 100 overpack lid can be removed to gain access to the multi-purpose canister.</p> <p>The HI-TRAC transfer cask has removable bottom and top lids.</p>	Chapter 8  Chapter 8

Item	Compliance	Location of Supporting Information in This Document
viii. A confinement barrier that acceptably protects the spent fuel cladding during storage.	<p>The peak temperature of the fuel cladding at design basis heat duty of each MPC has been demonstrated to be maintained below the limits specified in ISG-11 [4.1.4].</p> <p>The confinement barriers consist of highly ductile stainless steel alloys. The multi-purpose canister is housed in the overpack, built from a steel structure whose materials are selected and examined to maintain protection against brittle fracture under off-normal ambient (cold) temperatures (minimum of -40°F).</p>	<p>Section 4.4</p> <p>Subsection 3.1.1 Subsection 3.1.2.3</p>
ix. The structures are compatible with the appropriate monitoring systems.	The HI-STORM 100 overpack is a thick, upright cylindrical structure with large ventilation openings near the top and bottom. These openings are designed to prevent radiation streaming while enabling complete access to temperature monitoring probes.	Section 1.5, Subsection 2.3.3.2
x. Structural designs that are compatible with ready retrievability of fuel.	<p>The fuel basket is designed to be an extremely stiff honeycomb structure such that the storage cavity dimensions will remain unchanged under all postulated normal and accident events. Therefore, the retrievability of the spent nuclear fuel from the basket will not be jeopardized.</p> <p>The MPC canister lid is attached to the shell with a groove weld which is made using an automated welding device. A similar device is available to remove the weld. Thus, access to the fuel basket can be realized.</p> <p>The storage overpack and the transfer casks are designed to withstand accident loads without suffering permanent deformations of their structures that would prevent retrievability of the MPC by normal means. It is demonstrated by analysis that there is no physical interference between the MPC and the enveloping HI-STORM storage overpack or HI-TRAC transfer cask.</p>	<p>Subsection 3.1.1</p> <p>Sections 8.1 and 8.3</p> <p>Section 3.4</p>

Item	Compliance	Location of Supporting Information in This Document
i. Redundant sealing of confinement systems.	Two physically independent lids, each separately welded to the MPC shell (Enclosure Vessel shell) provide a redundant confinement system.	Section 1.5, Drawings Section 7.1.
ii. Adequate heat removal without active cooling systems.	Thermal analyses presented in Chapter 4 show that the HI-STORM 100 System will remove the decay heat generated from the stored spent fuel by strictly passive means and maintain the system temperature within prescribed limits.	Sections 4.4 and Sections 9.1 and 9.2
iii. Storage of spent fuel for a minimum of 20 years.	The service life of the MPC, storage overpack, and HI-TRAC are engineered to be in excess of 20 years.	Subsections 3.4.11 and 3.4.12
iv. Compatibility with wet or dry spent fuel loading and unloading facilities.	<ul style="list-style-type: none"> <li>• The system is designed to eliminate any material significant interactions in the wet (spent fuel pool) environment.</li> <li>• The HI-TRAC transfer cask is engineered for full compatibility with the MPCs, and standard loading and unloading facilities.</li> <li>• The HI-TRAC System is engineered for MPC transfer on the ISFSI pad with full consideration of ALARA and handling equipment compatibility.</li> </ul>	Subsection 3.4.1 Subsection 8.1.1 Subsection 8.1.1
v. Ease of decontamination.	<ul style="list-style-type: none"> <li>• The external surface of the multi-purpose canister is protected from contamination during fuel loading through a custom designed sealing device.</li> <li>• The HI-STORM storage overpack is not exposed to contamination</li> <li>• All exposed surfaces of the HI-TRAC transfer cask are coated to aid in decontamination</li> </ul>	Figures 8.1.13 and 8.1.14 Chapter 8 Section 1.5, Drawings

Item	Compliance	Location of Supporting Information in This Document
vi. Inspection of defects that might reduce confinement effectiveness.	<ul style="list-style-type: none"> <li>• The MPC enclosure vessel is designed and fabricated in accordance with ASME Code, Section III, Subsection NB, to the maximum extent practical.</li> <li>• Pressure testing and NDE of the closure welds verify containment effectiveness.</li> </ul>	Section 9.1
vii. Conspicuous and durable marking.	<p>The stainless steel lid of each MPC will have model number and serial number engraved for ready identification.</p> <p>The exterior envelope of the cask (the storage overpack) is marked in a conspicuous manner as required by 10CFR 72.236(k).</p>	N/A
viii. Compatibility with removal of the stored fuel from the site, transportation, and ultimate disposal by the U.S. Department of Energy.	The MPC is designed to be in full compliance with the DOE's draft specification for transportability and disposal published under the now dormant "MPC" program.	Section 2.4 Subsection 1.2.1.1

### 3.8 REFERENCES

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## APPENDIX 3.A: HI-STORM DECELERATION UNDER POSTULATED VERTICAL DROP EVENT AND TIPOVER

### 3.A.1 INTRODUCTION

Handling accidents with a HI-STORM overpack containing a loaded MPC are credible events (Section 2.2.3). The stress analyses carried out in Chapter 3 of this safety analysis report assume that the inertial loading on the load bearing members of the MPC, fuel basket, and the overpack due to a handling accident are limited by the Table 3.1.2 decelerations. The maximum deceleration experienced by a structural component is the product of the rigid body deceleration sustained by the structure and the dynamic load factor (DLF) applicable to that structural component. The DLF is a function of the contact impulse and the structural characteristics of the component.

The rigid body deceleration is a strong function of the load-deformation characteristics of the impact interface, weight of the cask, and the drop height or angle of free rotation. For the HI-STORM 100 System, the weight of the structure and its surface compliance characteristics are known. However, the contact stiffness of the ISFSI pad (and other surfaces over which the HI-STORM 100 may be carried during its movement to the ISFSI) is site-dependent. The contact resistance of the collision interface, which is composed of the HI-STORM 100 and the impacted surface compliance, therefore, is not known a priori for a specific site. Analyses for the rigid body decelerations are, therefore, presented here using a reference ISFSI pad (which is the pad used in a recent Lawrence Livermore National Laboratory report and is the same reference pad used in the HI-STAR 100 FSAR). The finite element model (grid size, extent of model, soil properties, etc.) follows the LLNL report.

An in-depth investigation by the Lawrence Livermore Laboratory (LLNL) into the mechanics of impact between a cask-like impactor on a reinforced concrete slab founded on a soil-like subgrade has identified three key parameters, namely, the thickness of the concrete slab,  $t_p$ , compressive strength of the concrete  $f_c'$  and equivalent Young's Modulus of the subgrade  $E$ . These three parameters are key variables in establishing the stiffness of the pad under impact scenarios. The LLNL reference pad parameters, which we hereafter denote as Set A, provide one set of values of  $t_p$ ,  $f_c'$ , and  $E$  that are found to satisfy the deceleration criteria applicable to the HI-STORM 100 cask. Another set of parameters, referred to as Set B herein, is also shown to satisfy the g-load limit requirements. In fact, an infinite number of combinations of  $t_p$ ,  $f_c'$ , and  $E$  can be compiled that would meet the g-load limit qualification. However, in addition to satisfying the g-limit criterion, the pad must be demonstrated to possess sufficient flexural and shear stiffness to meet the ACI 318-95 strength limits under factored load combinations. The minimum strength requirement to comply with ACI 318-95 provisions places a restriction on the lower bound values of  $t_p$ ,  $f_c'$ , and  $E$  that must be met in an ISFSI pad design.

Our focus in this appendix, however, is to quantify the peak decelerations that would be experienced by a loaded HI-STORM 100 cask under the postulated impact scenarios for the two pad designs defined by parameter Sets A and B, respectively. The information presented in this appendix also serves to further authenticate the veracity of the Holtec DYNA3D model described in the 1997 benchmark report [3.A.4.]

### 3.A.2 Purpose

The purpose of this appendix is to demonstrate that the rigid body deceleration experienced by the HI-STORM 100 System during a handling accident or non-mechanistic tip-over are below the design basis deceleration of 45g's (Table 3.1.2). Two accidental drop scenarios of a loaded HI-STORM 100 cask on the ISFSI pad are considered in this appendix. They are:

- i. Tipover: A loaded HI-STORM 100 is assumed to undergo a non-mechanistic tipover event and impacting the ISFSI pad with an incipient impact angular velocity, which is readily calculated from elementary dynamics.
- ii. End drop: The loaded HI-STORM 100 is assumed to drop from a specified height  $h$ , with its longitudinal axis in the vertical orientation, such that its bottom plate impacts the ISFSI pad.

The dynamic load factors are a function of the predominant natural frequency of vibration of the component for a given input load pulse shape. Dynamic load factors are applied, as necessary, to the results of specific component analyses performed using the loading from the design basis rigid body decelerations. Therefore, for the purposes of this appendix, it is desired to demonstrate that the rigid body deceleration experienced in each of the drop scenarios is below the HI-STORM 100 45g design basis.

### 3.A.3 Background and Methodology

In 1997 Lawrence Livermore National Laboratory (LLNL) published the experimentally obtained results of the so-called fourth series billet tests [3.A.1] together with a companion report [3.A.2] documenting a numerical solution that simulated the drop test results with reasonable accuracy. Subsequently, USNRC personnel published a paper [3.A.3] affirming the NRC's endorsement of the LLNL methodology. The LLNL simulation used modeling and simulation algorithms contained within the commercial computer code DYNA3D [3.A.6].

The LLNL cask drop model is not completely set forth in the above-mentioned LLNL reports. Using the essential information provided by the LLNL [3.A.2] report, however, Holtec is able to develop a finite element model for implementation on LS-DYNA3D [3.A.5] which is fully consistent with LLNL's (including the use of the Butterworth filter for discerning rigid body deceleration from "noisy" impact data). The details of the LS-DYNA3D dynamic model, henceforth referred to as the Holtec model, are contained in the proprietary benchmark report [3.A.4] wherein it is shown that the peak deceleration in every case of billet drop analyzed by LLNL is replicated within a small tolerance by the Holtec model. The case of the so-called

"generic" cask, for which LLNL provided predicted response under side drop and tipover events, is also bounded by the Holtec model. In summary, the benchmarking effort documented in [3.A.4] is in full compliance with the guidance of the Commission [3.A.3].

Having developed and benchmarked an LLNL-consistent cask impact model, a very similar model is developed and used to prognosticate the HI-STORM drop scenarios. The reference elasto-plastic-damage characteristics of the target concrete continuum used by LLNL, and used in the HI-STAR 100 FSAR are replicated herein. The HI-STORM 100 target model is identical in all aspects to the reference pad approved for the HI-STAR 100 FSAR.

In the tipover scenario the cask surface structure must be sufficiently pliable to cushion the impact and limit the rigid body deceleration. The angular velocity at the contact time is readily calculated using planar rigid body dynamics and is used as an initial condition in the LS-DYNA3D simulation.

The end drop event produces a circular impact patch equal to the diameter of the overpack baseplate. The elasto-plastic-damage characteristics of the concrete target and the drop height determine the maximum deceleration. A maximum allowable height "h" is determined to limit the deceleration to a value below the design basis.

A description of the work effort and a summary of the results are presented in the following sections. In all cases, the reported decelerations are below the design basis of 45g's at the top of the MPC fuel basket.

### 3.A.4 Assumptions and Input Data

#### 3.A.4.1 Assumptions

The assumptions used to create the model are completely described in Reference [3.A.4] and are shown there to be consistent with the LLNL simulation. There are key aspects, however, that are restated here:

The maximum deceleration experienced by the cask during a collision event is a direct function of the structural rigidity (or conversely, compliance) of the impact surface. The compliance of the ISFSI pad is quite obviously dependent on the thickness of the pad,  $t_p$ , the compressive strength of the concrete,  $f_c'$  and stiffness of the sub-grade (expressed by its effective Young's modulus, E). The structural rigidity of the ISFSI pad will increase if any of the three above-mentioned parameters ( $t_p$ ,  $f_c'$  or E) is increased. For the reference pad, the governing parameters (i.e.,  $t_p$ ,  $f_c'$  and E) are assumed to be identical to the pad defined by LLNL [3.A.2], which is also the same as the pad utilized in the benchmark report [3.A.4]. We refer to the LLNL ISFSI pad parameters as Set A. (Table 3.A.1).

As can be seen from Table 3.A.1, the nominal compressive strength  $f_c'$  in Set A is limited to 4200 psi. However, experience has shown that ISFSI owners have considerable practical difficulty in limiting the 28 day strength of poured concrete to 4200 psi, chiefly because a principal element of

progress in reinforced concrete materials technology has been in realizing ever increasing concrete nominal strength. Inasmuch as a key objective of the ISFSI pad is to limit its structural rigidity (and not  $f_c'$  per se), and limiting  $f_c'$  to 4200 psi may be problematic in certain cases, an alternative set of reference pad parameters is defined (Set B in Table 3.A.1), which permits a higher value of  $f_c'$  but much smaller values of pad thickness,  $t_p$  and sub-grade Young's modulus, E.

The ISFSI owner has the option of constructing the pad to comply with the limits of Set A or Set B without performing site-specific cask impact analyses. It is recognized that, for a specific ISFSI site, the reinforced concrete, as well as the underlying engineered fill properties, may be different at different locations on the pad or may be uniform, but non-compliant with either Set A or Set B. In that case, the site-specific conditions must be performed to demonstrate compliance with the design limits of the HI-STORM system (e.g., maximum rigid body g-load less than 45 g's). The essential data which define the pad (Set A and Set B) used to qualify the HI-STORM 100 are provided in Table 3.A.1.

The HI-STORM 100 steel structural elements (outer shell, inner shell, radial plates, lid, etc.), are fabricated from SA-516 Grade 70. The steel is described as a bi-linear elastic-plastic material with limited strain failure by five material parameters (E,  $S_y$ ,  $S_u$ ,  $\epsilon_u$ , and  $\nu$ ). The numerical values used in the finite element model are shown in Table 3.A.2. The concrete located inside of the overpack for this dynamic analysis is defined to be identical with the concrete pad. This is conservative since the concrete assumed in the reference pad is reinforced. Therefore, the strength of the concrete inside the HI-STORM 100 absorbs less energy if it is also assumed to be reinforced.

#### 3.A.4.2 Input Data

Table 3.A.1 characterizes the properties of the full-scale reference target pad used in the analysis of the full size HI-STORM 100 System. The principal strength parameters that define the stiffness of the pad, namely,  $t_p$ , E and  $f_c'$  are input in the manner described in [3.A.2] and [3.A.4].

Table 3.A.2 contains the material description parameters for the steel types; SA-516-70 used in the numerical investigation.

Table 3.A.3 details the geometry of the HI-STORM 100 used in the drop simulations. This data is taken from applicable HI-STORM 100 drawings.

#### 3.A.5 Finite Element Model

The finite-element model of the Holtec HI-STORM 100 overpack (baseplate, shells, radial plates, lid, concrete, etc.), concrete pad and a portion of the subgrade soil is constructed using the pre-processor integrated with the LS-DYNA3D software [3.A.5]. The deformation field for all postulated drop events (the end-drop and the tipover) exhibits symmetry with the vertical plane passing through the cask diameter and the concrete pad length. Using this symmetry condition of the deformation field only a half finite-element model is constructed. The finite-element model is organized into nineteen independent parts (the baseplate components, the outer shell, the inner

shell, the radial plates, the channels, the lid components, the basket steel plates, the basket fuel zone, the concrete pad and the soil). The final model contains 30351 nodes, 24288 solid type finite-elements, 1531 shell type finite-elements, seven (7) materials, ten (10) properties and twenty-four (24) interfaces. The finite-element model used for the tipover-drop event is depicted in Figures 3.A.1 through 3.A.4. Figures 3.A.5 through 3.A.8 show the end-drop finite-element model.

The soil grid, shown in Figure 3.A.9, is a rectangular prism (800 inches long, 375 inches wide and 470 inches deep), is constructed from 13294 solid type finite-elements. The material defining this part is an elastic isotropic material. The central portion of the soil (400 inches long, 150 inches wide and 170 inches deep) where the stress concentration is expected to appear is discretized with a finer mesh.

The concrete pad is 320 inches long, 100 inches wide and is 36 inches thick. This part contains 8208 solid finite-elements. A uniform sized finite-element mesh, shown in Figure 3.A.10, is used to model the concrete pad. The concrete behavior is described using a special constitutive law and yielding surface (MAT\_PSEUDO\_TENSOR) contained within LS-DYNA3D. The geometry, the material properties, and the material behavior are identical to the LLNL reference pad (Material 16 IIB).

The half portion of the steel cylindrical overpack contains 1531 shell finite-elements. The steel material description (SA-516-70) is realized using a bi-linear elasto-plastic constitutive model (MAT\_PIECEWISE\_LINEAR\_PLASTICITY). Figure 3.A.11 depicts details of the steel components of the cask finite-element mesh, with the exception of the inner shell, channels and lid components, which are shown in Figures 3.A.12 and 3.A.13. The concrete filled between the inner and the outer shells, and contained in the baseplate and lid components is modeled using 1664 solid finite-elements and is depicted in Figure 3.A.14. The concrete material is defined identical to the pad concrete.

The MPC and the contained fuel are modeled in two parts that represent the lid and baseplate, and the fuel area. An elastic material is used for both parts. The finite-element mesh pertinent to the MPC contains 1122 solid finite-elements and is shown in Figure 3.A.15. The mass density is appropriate to match a representative weight of 356,521 lb. that is approximately mid-way between the upper and lower weight estimates for a loaded HI-STORM 100.

The total weight used in the analysis is approximately 2,000 lb. lighter than the HI-STORM 100 containing the lightest weight MPC.

Analysis of a single mass impacting a spring with a given initial velocity shows that both the maximum deceleration " $a_M$ " of the mass and the time duration of contact with the spring " $t_c$ " are related to the dropped weight " $w$ " and drop height " $h$ " as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}; t_c \sim \sqrt{w}$$

Therefore, the most conservatism is introduced into the results by using the minimum weight. It is emphasized that the finite element model described in the foregoing is identical in its approach to the "Holtec model" described in the benchmark report [3.A.4]. Gaps between the MPC and the overpack are included in the model.

### 3.A.6 Impact Velocity

#### a. Linear Velocity: Vertical Drops

For the vertical drop event, the impact velocity,  $v$ , is readily calculated from the Newtonian formula:

$$v = \sqrt{(2gh)}$$

where

$$\begin{aligned} g &= \text{acceleration due to gravity} \\ h &= \text{free-fall height} \end{aligned}$$

#### b. Angular Velocity: Tip-Over

The tipover event is an artificial construct wherein the HI-STORM 100 overpack is assumed to be perched on its edge with its C.G. directly over the pivot point A (Figure 3.A.16). In this orientation, the overpack begins its downward rotation with zero initial velocity. Towards the end of the tip-over, the overpack is horizontal with its downward velocity ranging from zero at the pivot point (point A) to a maximum at the farthest point of impact (point E in Figure 3.A.17). The angular velocity at the instant of impact defines the downward velocity distribution along the contact line.

In the following, an explicit expression for calculating the angular velocity of the cask at the instant when it impacts on the ISFSI pad is derived. Referring to Figure 3.A.16, let  $r$  be the length AC where C is the cask centroid. Therefore,

$$r = \left( \frac{d^2}{4} + h^2 \right)^{1/2}$$

The mass moment of inertia of the HI-STORM 100 System, considered as a rigid body, can be written about an axis through point A, as

$$I_A = I_c + \frac{W}{g} r^2$$

where  $I_c$  is the mass moment of inertia about a parallel axis through the cask centroid C and W is the weight of the cask ( $W = Mg$ ).

Let  $\theta_1(t)$  be the rotation angle between a vertical line and the line AC. The equation of motion for rotation of the cask around point A, during the time interval prior to contact with the ISFSI pad, is

$$I_A \frac{d^2 \theta_1}{dt^2} = Mg r \sin \theta_1$$

This equation can be rewritten in the form

$$\frac{I_A}{2} \frac{d(\dot{\theta}_1)^2}{d\theta_1} = Mg r \sin \theta_1$$

which can be integrated over the limits  $\theta_1 = 0$  to  $\theta_1 = \theta_{2f}$  (See Figure 3.A.17).

The final angular velocity  $\dot{\theta}_1$  at the time instant just prior to contact with the ISFSI pad is given by the expression

$$\dot{\theta}_1(t_B) = \sqrt{\frac{2Mg r}{I_A} (1 - \cos \theta_{2f})}$$

where, from Figure 3.A.17

$$\theta_{2f} = \cos^{-1} \left( \frac{d}{2r_1} \right)$$

This equation establishes the initial conditions for the final phase of the tip-over analysis; namely, the portion of the motion when the cask is decelerated by the resistive force at the ISFSI pad interface.

Using the data germane to HI-STORM 100 (Table 3.A.3), and the above equations, the angular velocity of impact is calculated as 1.49 rad/sec.

### 3.A.7 Results

#### 3.A.7.1 Set A Pad Parameters

It has been previously demonstrated in the benchmark report [3.A.4] that bounding rigid body decelerations are achieved if the cask is assumed to be rigid with only the target (ISFSI pad) considered as an energy absorbing media. Therefore, for the determination of the bounding



decelerations reported in this appendix, the HI-STORM storage overpack was conservatively made rigid except for the radial channels that position the MPC inside of the overpack. The MPC material behavior was characterized in the identical manner used in the Livermore Laboratory analysis as was the target ISFSI pad and underlying soil. The LS-DYNA3D time-history results are processed using the Butterworth filter (in conformance with the LLNL methodology) to establish the rigid body motion time-history of the cask. The material points on the cask where the acceleration displacement and velocity are computed for each of the drop scenarios are shown in Figure 3.A.18.

Node 82533 (Channel A1), which is located at the center of the outer surface of the baseplate, serves as the reference point for end-drop scenarios.

Node 84392 (Channel A2), which is located at the center of the cask top lid outer surface, serves as the reference point for the tipover scenario with the pivot point indicated as Point 0 in Figure 3.A.18.

The final results are shown in Table 3.A.4.

i. Tipover:

The time-histories of the impact force, the displacement and velocity time-histories of Channel A2, and the average vertical deceleration of the overpack lid top plate have been determined for this event [3.A.7].

The deceleration at the top of the fuel basket is obtained by ratioing the average deceleration of the overpack lid top plate. The maximum filtered deceleration at the top of the fuel basket is 42.85g's, which is below the design basis limit.

ii. End Drop:

The drop height  $h = 11$ " is considered in the numerical analysis. This is considered as an acceptable maximum carry height for the HI-STORM 100 System if lifted above a surface with design values of  $t_p$ ,  $f_c'$ , and  $E$  equal to those presented in Table 3.A.1 for Parameter Set "A". The maximum filtered deceleration at the top of the fuel basket is 43.98g's, which is below the design basis limit.

The computer code utilized in this analysis is LS-DYNA3D [3.A.5] validated under Holtec's QA system. Table 3.A.4 summarizes the key results from all impact simulations for the Set A parameters discussed in the foregoing.

The filter frequencies (to remove unwanted high-frequency contributions) for the Holtec cask analyses analyzed in this TSAR is the same as used for the corresponding problem analyzed in [3.A.2] and [3.A.4]. To verify the Butterworth filter parameters (350 Hz cutoff frequency, etc.) used in processing the numerical data, a Fourier power decomposition was generated.

### 3.A.7.2 Set B Parameters

As stated previously, Set B parameters produce a much more compliant pad than the LLNL reference pad (Set A). This fact is borne out by the tipover and end drop analyses performed on the pad defined by the Set B parameters. Table 3.A.4 provides the filtered results for the two impact scenarios. In every case, the peak decelerations corresponding to Set B parameters are less than those for Set A (also provided in Table 3.A.4).

Impact force and acceleration time history curves for Set B have the same general shape as those for Set A and are contained in the calculation package [3.A.7]. All significant results are summarized in Table 3.A.4.

### 3.A.8 Computer Codes and Archival Information

The input and output files created to perform the analyses reported in this appendix are archived in Holtec International calculation package [3.A.7].

### 3.A.9 Conclusion

The DYNA3D analysis of HI-STORM 100 reported in this appendix leads to the following conclusion:

- a. If a loaded HI-STORM undergoes a free fall for a height of 11 inches in a vertical orientation on to a reference pad defined by Table 3.A.1, the maximum rigid body deceleration is less than 45g's for both Set A and Set B pad parameters.
- b. If a loaded HI-STORM 100 overpack pivots about its bottom edge and tips over on to a reference pad defined by Table 3.A.1, then the maximum rigid body deceleration of the cask centerline at the plane of the top of the MPC fuel basket cellular region is less than 45g's for both Set A and Set B parameters.

Table 3.A.4 provides key results for all drop cases studied herein for both pad parameter sets (A and B). If the pad designer maintains each of the three significant parameters ( $t_p$ ,  $f_c'$ , and E) below the limit for the specific set selected (Set A or Set B), then the stiffness of the pad at any ISFSI site will be lower and the computed decelerations at the ISFSI site will also be lower. Furthermore, it is recognized that a refinement of the cask dynamic model will accrue further reduction in the computed peak deceleration. For example, incorporation of the structural flexibility in the MPC enclosure vessel, fuel basket, etc., would lead to additional reductions in the computed values of the peak deceleration. These refinements, however, add to the computational complexity. Because g-limits are met without the above-mentioned and other refinements in the cask dynamic model, the simplified dynamic model described in this appendix was retained to reduce the overall computational effort.

3.A.10 References

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- [3.A.7] Siromani, D. "Analysis of the Loaded HI-STORM 100 System Under Drop and Tip-Over Scenarios", Holtec Report HI-2002474, March 2016.

Table 3.A.1: Essential Variables to Characterize the ISFSI Pad (Set A and Set B)

Item	Parameter Set A	Parameter Set B
Thickness of concrete, (inches)	36	28
Nominal compressive strength of concrete at 28 days, (psi)	4,200	6,000
Max. modulus of elasticity of the subgrade (psi)	28,000	16,000

## Notes:

1. The concrete Young's Modulus is derived from the American Concrete Institute recommended formula  $57,000\sqrt{f}$  where  $f$  is the nominal compressive strength of the concrete (psi).
2. The effective modulus of elasticity of the subgrade will be measured by the classical "plate test" or other appropriate means before pouring of the concrete to construct the ISFSI pad.
3. The pad thickness, concrete compressive strength, and the subgrade soil effective modulus are the upper bound values to ensure that the deceleration limits under the postulated events set forth in Table 3.1.2 are satisfied.

Table 3.A.2: Essential Steel Material Properties for HI-STORM 100 Overpack

Steel Type	Parameter	Value
SA-516-70 at T = 350 deg. F	E	2.800E + 07
	S <sub>y</sub>	3.315E+04 psi
	S <sub>u</sub>	7.000E+04 psi
	ε <sub>u</sub>	0.21
	ν	0.30

Note that the properties of the steel components, except for the radial channels used to position the MPC, do not affect the results reported herein since the HI-STORM 100 is eventually assumed to behave as a rigid body (by internal constraint equations automatically computed by DYNA3D upon issue of a “make rigid” command). In Section 3.4, however, stress and strain results for an additional tip-over analysis, performed using the actual material behavior ascribed to the storage overpack, are presented for the sole purpose of demonstrating ready retrievability of the MPC after the tip-over. As an option, the radial channels may be fabricated from SA240-304 material. The difference in material properties, however, has a negligible effect on the end results.

Table 3.A.3: Key Input Data in Drop Analyses

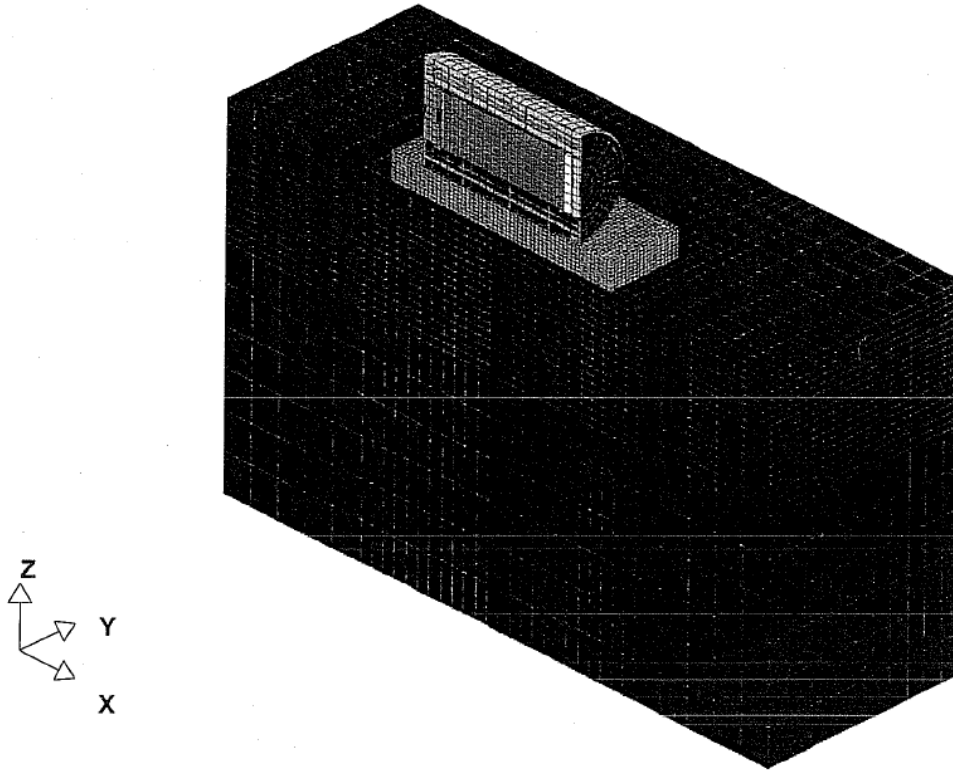
Overpack weight	267,664 lb
Radial Concrete weight	163,673 lb
Length of the cask	231.25 inches
Diameter of the bottom plate	132.50 inches
Inside diameter of the cask shell	72.50 inches
Outside diameter of the cask shells	132.50 inches
MPC weight (including fuel)	88,857 lb
MPC height	190.5 inches
MPC diameter	68.375 inches
MPC bottom plate thickness	2.5 inches
MPC top plate thickness	9.5 inches

Table 3.A.4: Filtered Results for Drop and Tip-Over Scenarios for HI-STORM 100<sup>†</sup>

Drop Event	Max. Displacement (inch)		Impact Velocity (in/sec)	Max. Deceleration <sup>††</sup> at the Top of the (g's) Basket		Duration of Deceleration Pulse (msec)	
	Set A	Set B		Set A	Set B	Set A	Set B
End Drop for 11 inches	0.65	0.81	92.2	43.98	41.53	3.3	3.0
Non-Mechanistic Tip-over	4.25	5.61	304.03	42.85	39.91	2.3	2.0

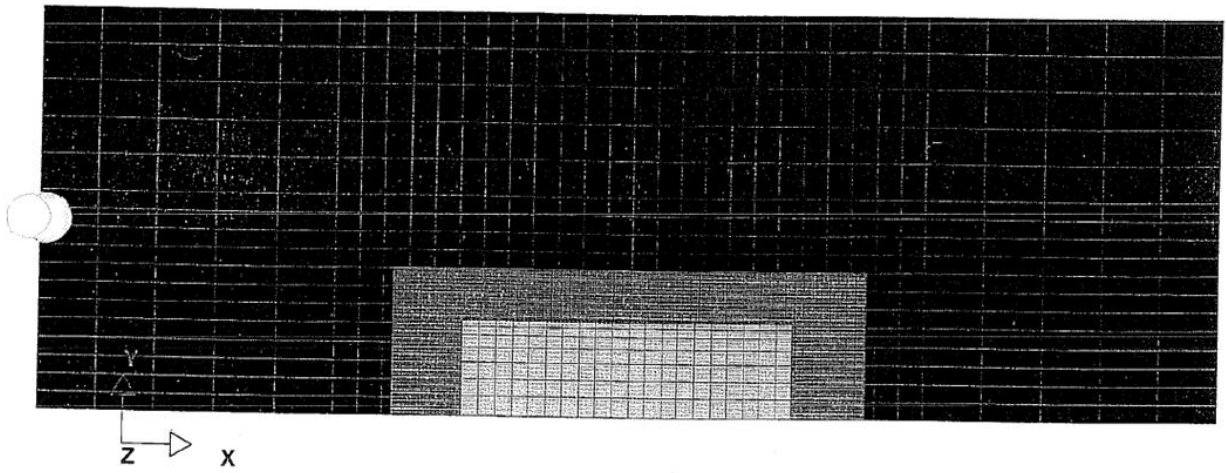
<sup>†</sup> The passband frequency of the Butterworth filter is 350 Hz.

<sup>††</sup> The distance of the top of the fuel basket is 206" from the pivot point. The distance of the top of the cask is 231.25" from the pivot point. Therefore, all displacements, velocities, and accelerations at the top of the fuel basket are 89.08% of those at the cask top (206"/231.25").

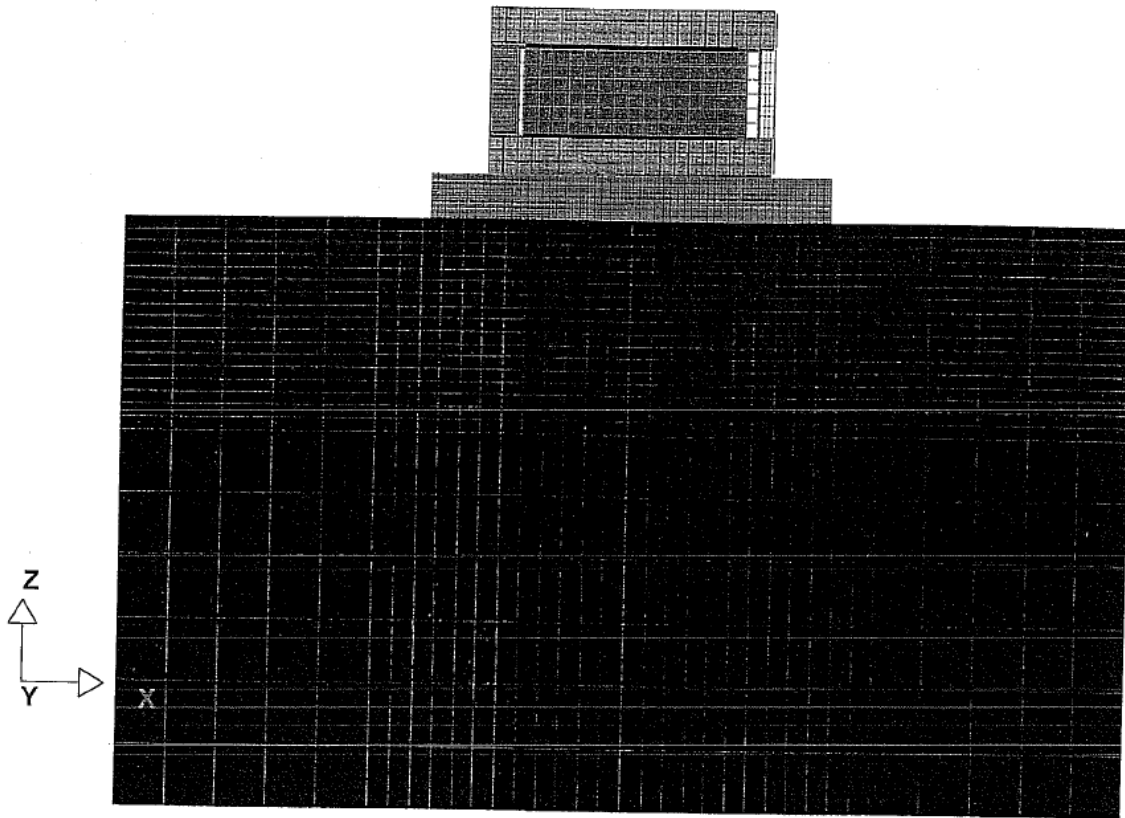


**Fig 3.A.1 TIPOVER FINITE-ELEMENT MODEL (3-D VIEW)**

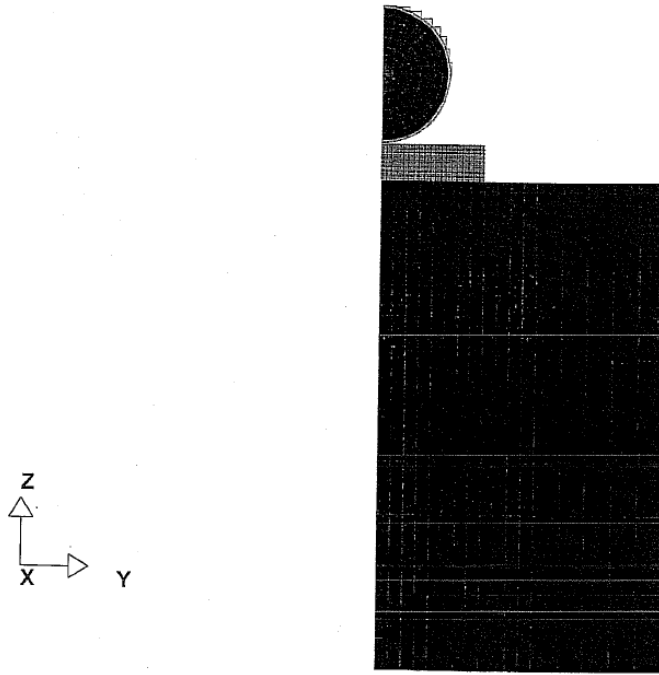




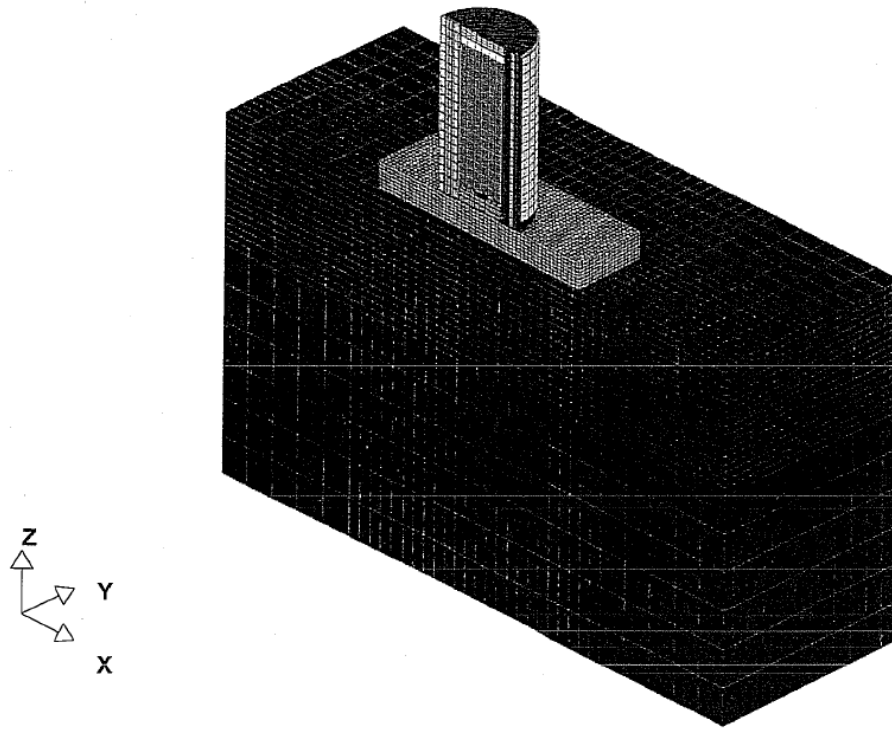
**Fig 3.A.2 TIPOVER FINITE-ELEMENT MODEL (PLAN)**



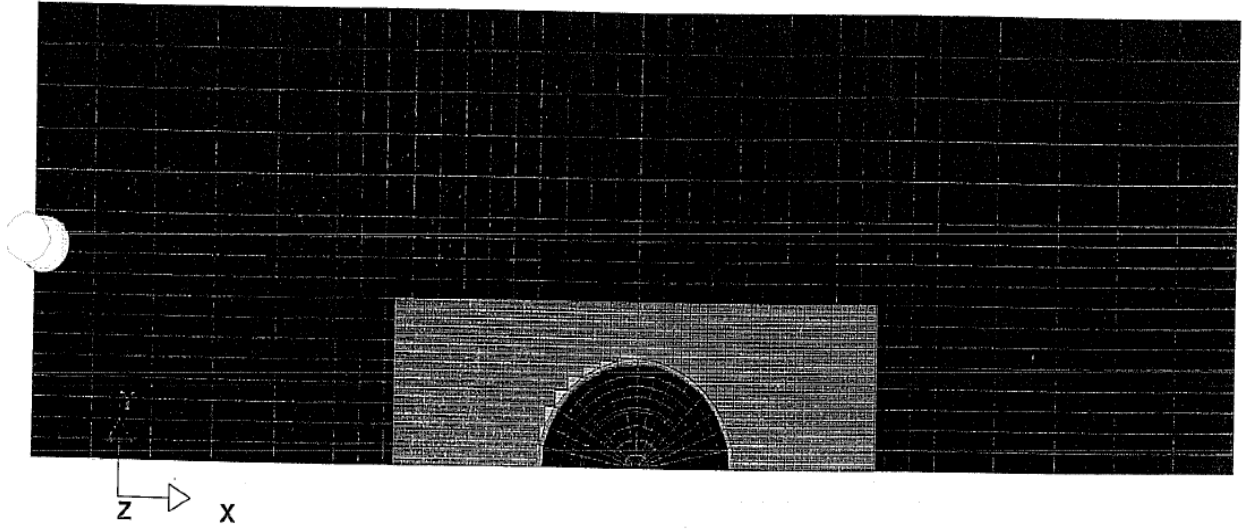
**Fig 3.A.3 TIPOVER FINITE-ELEMENT MODEL (XZ VIEW)**



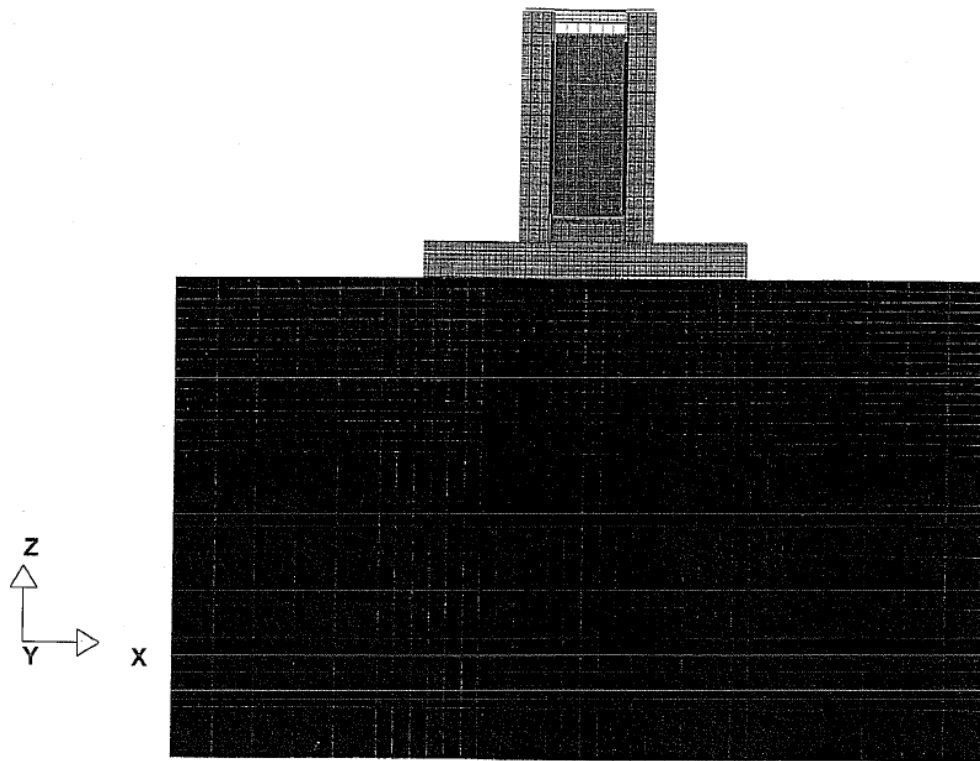
**Fig 3.A.4 TIPOVER FINITE-ELEMENT MODEL (YZ VIEW)**



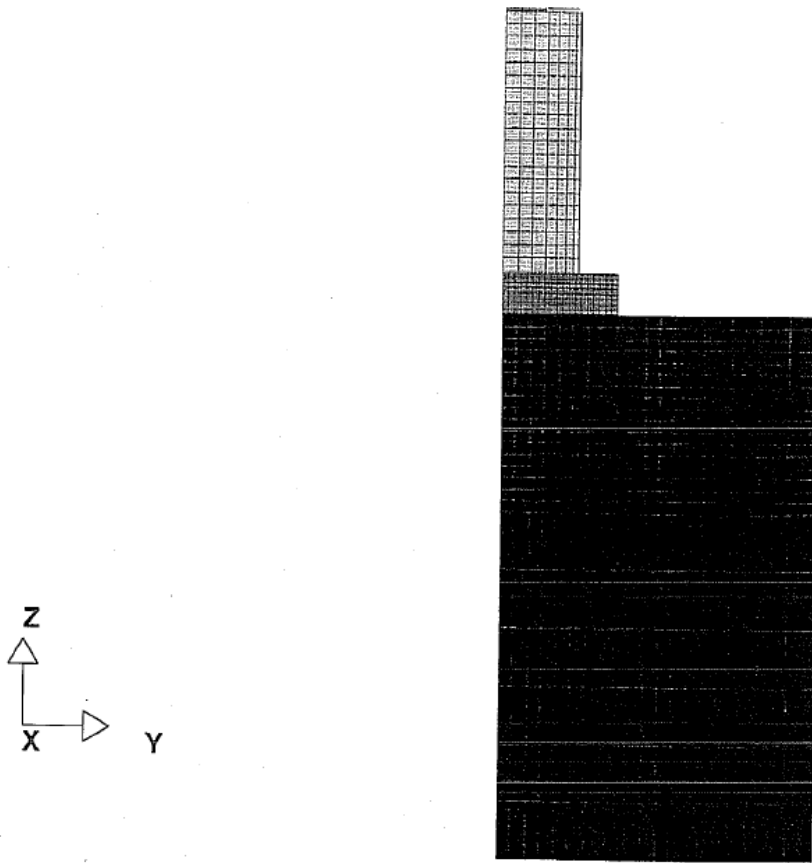
**Fig 3.A.5 END-DROP FINITE-ELEMENT MODEL (3-D VIEW)**



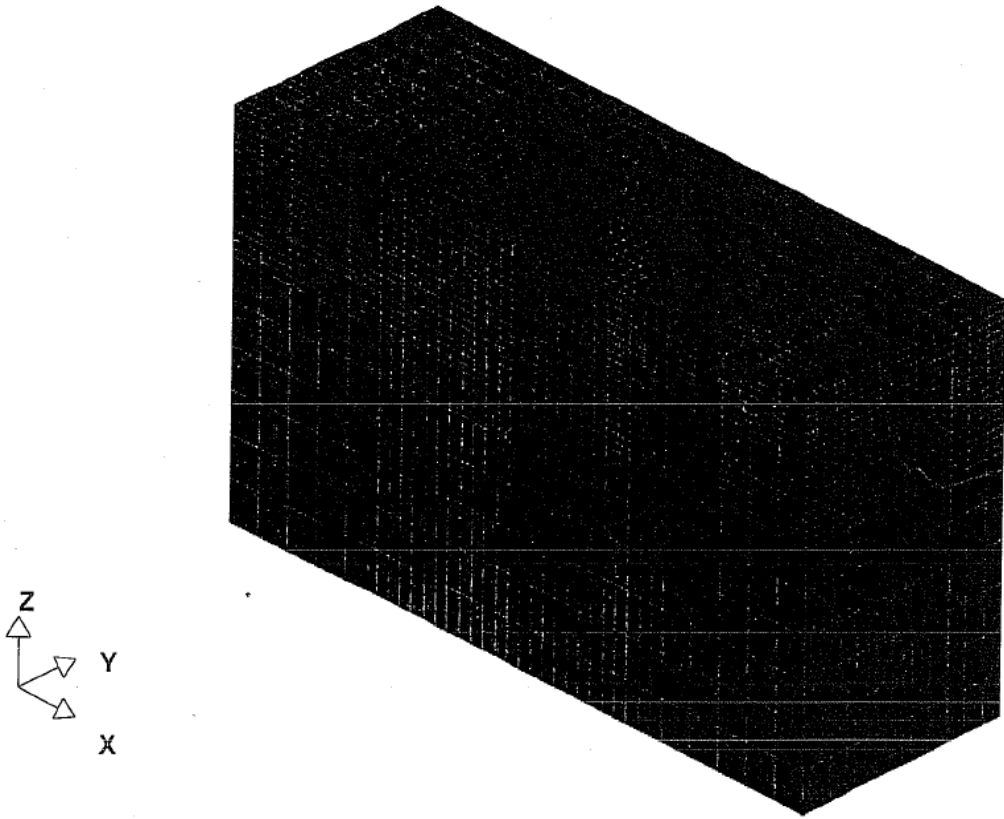
**Fig 3.A.6 END-DROP FINITE-ELEMENT MODEL (PLAN)**



**Fig 3.A.7 END-DROP FINITE-ELEMENT MODEL (XZ VIEW)**

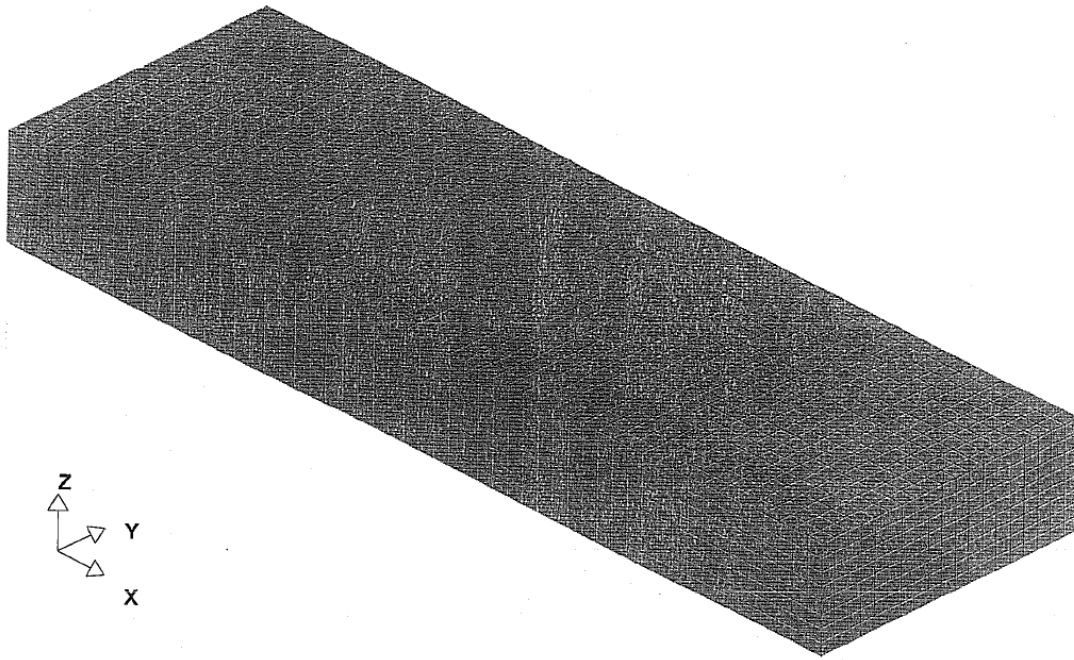


**Fig 3.A.8 END-DROP FINITE-ELEMENT MODEL (YZ VIEW)**

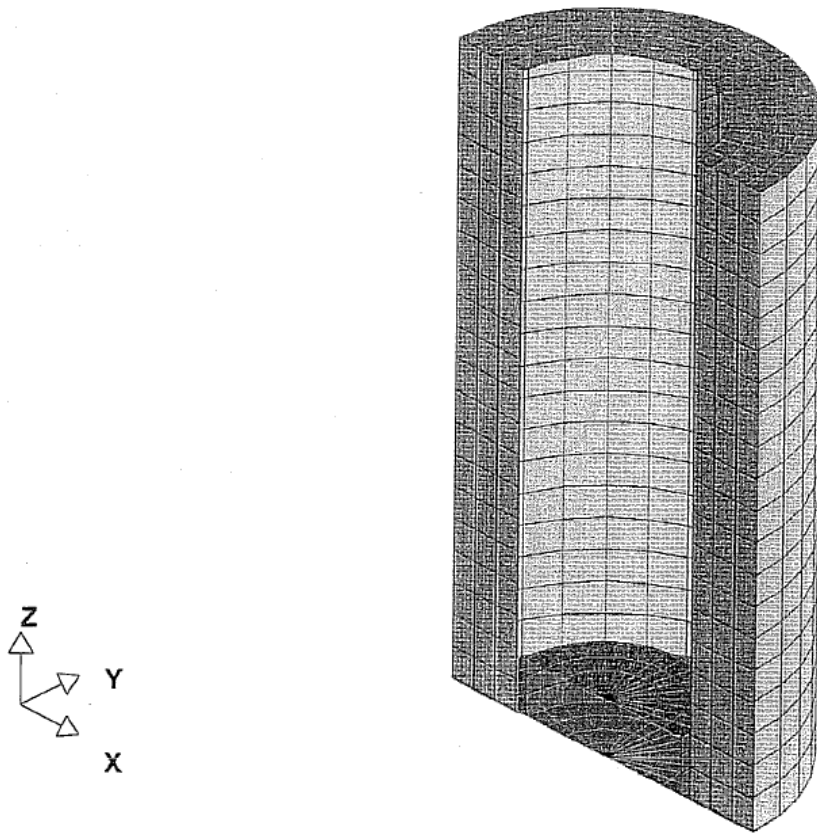


**Fig 3.A.9 SOIL FINITE-ELEMENT MODEL (3-D VIEW)**

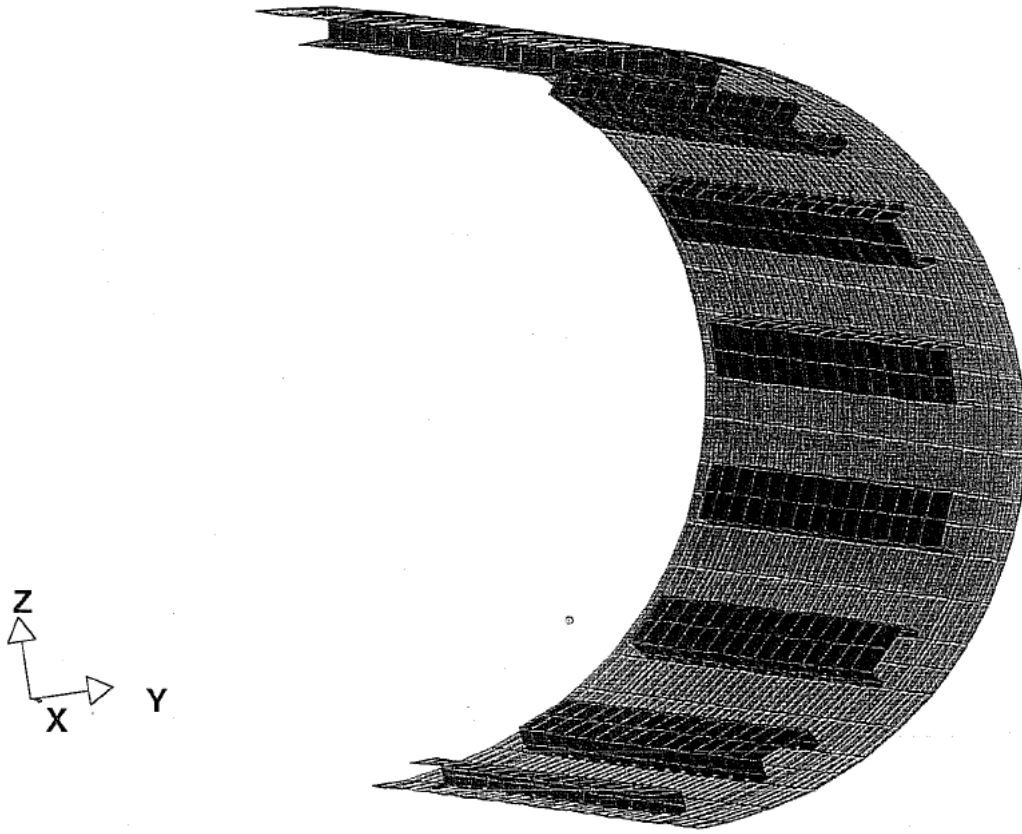




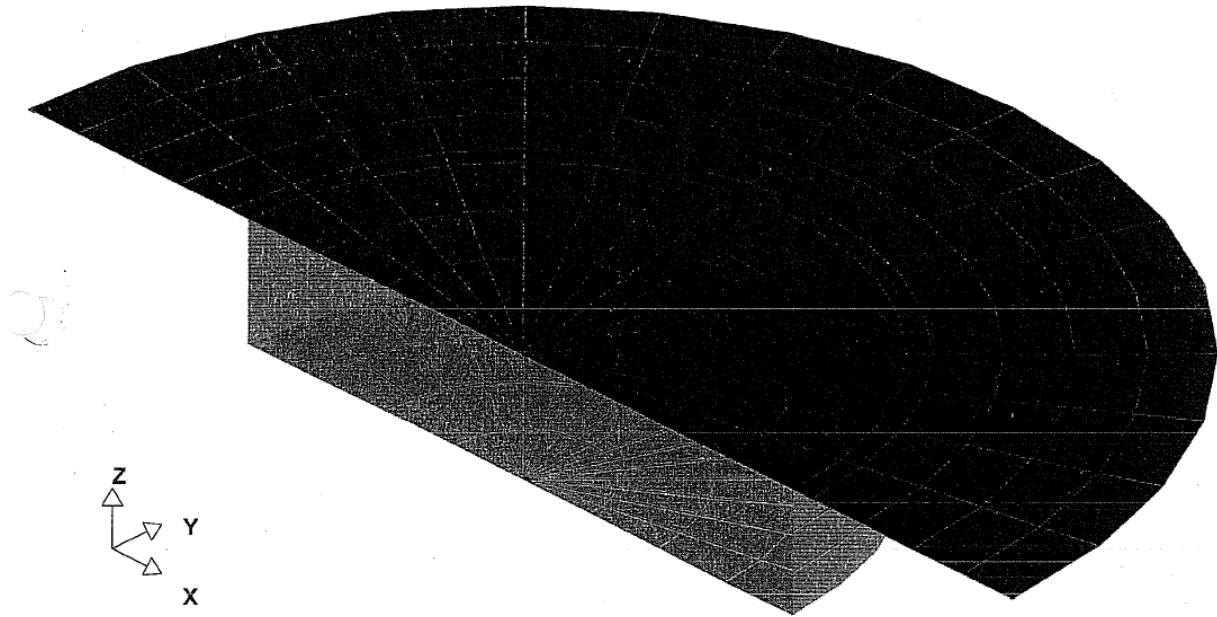
**Fig 3.A.10 CONCRETE PAD FINITE-ELEMENT MODEL (3-D VIEW)**



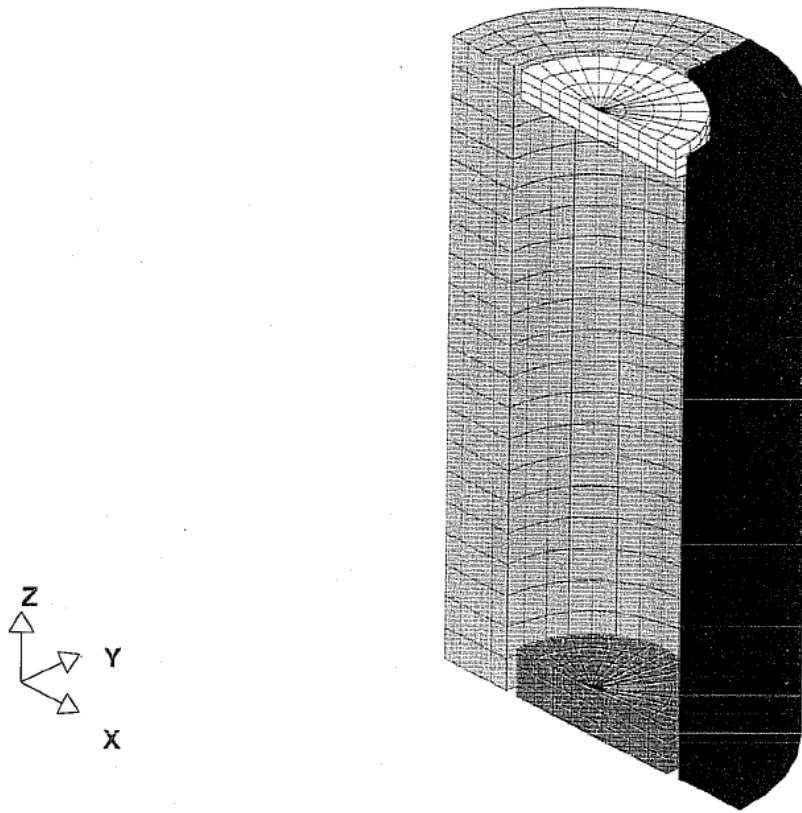
**Fig 3.A.11 OVERPACK STEEL STRUCTURE FINITE-ELEMENT MODEL (3-D VIEW)**



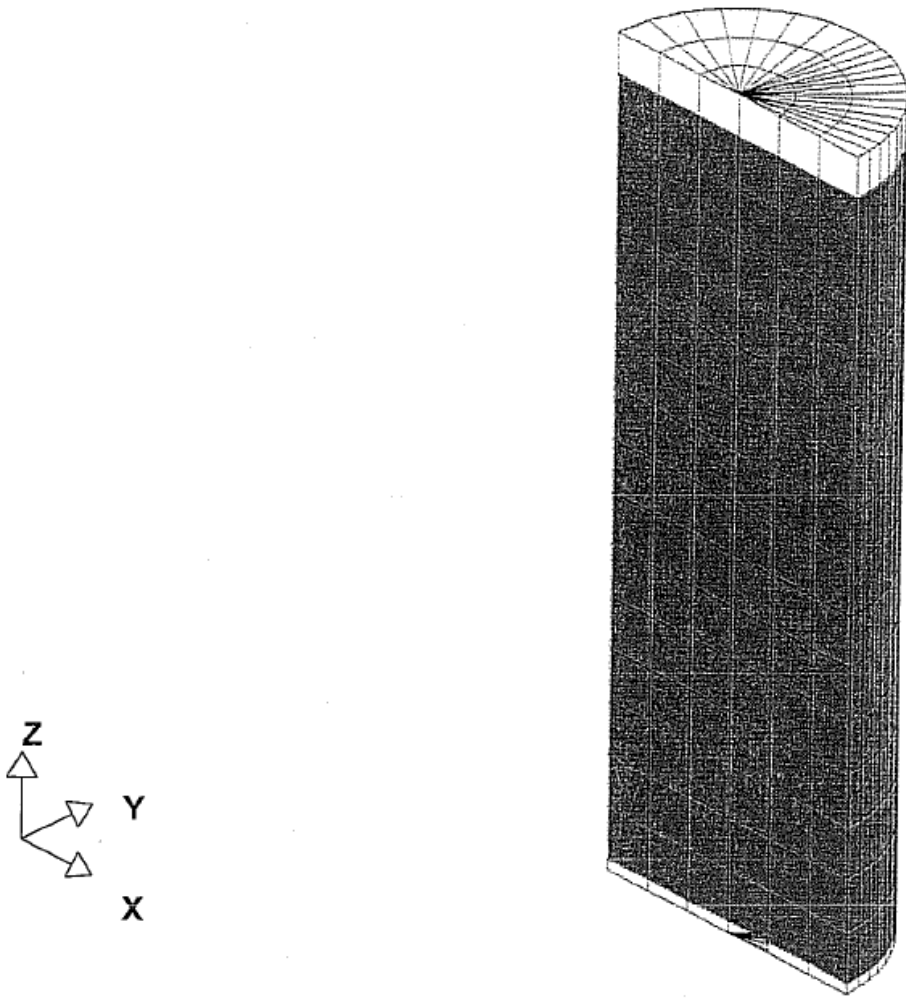
**FIGURE 3.A.12 INNER SHELL AND CHANNELS FINITE-ELEMENT MODEL (3-D VIEW)**



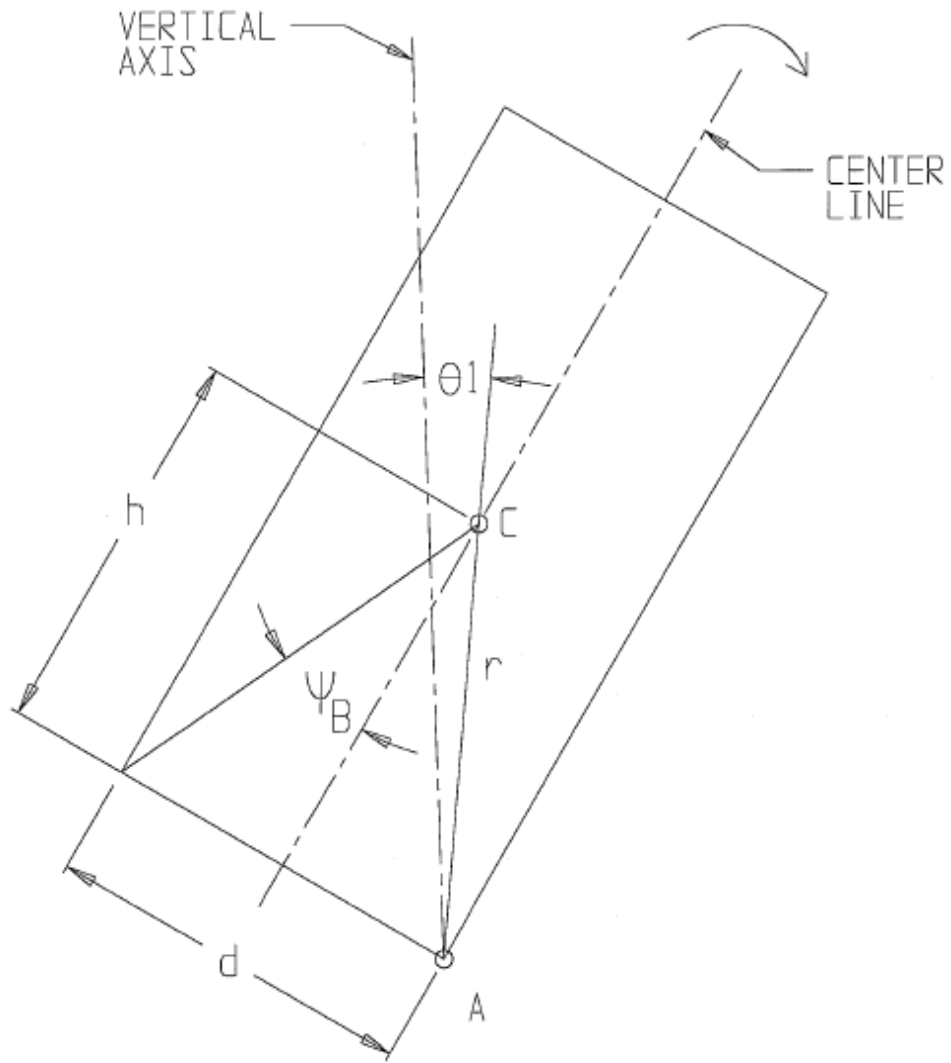
**FIGURE 3.A.13 LID STEEL FINITE-ELEMENT MODEL (3-D VIEW)**



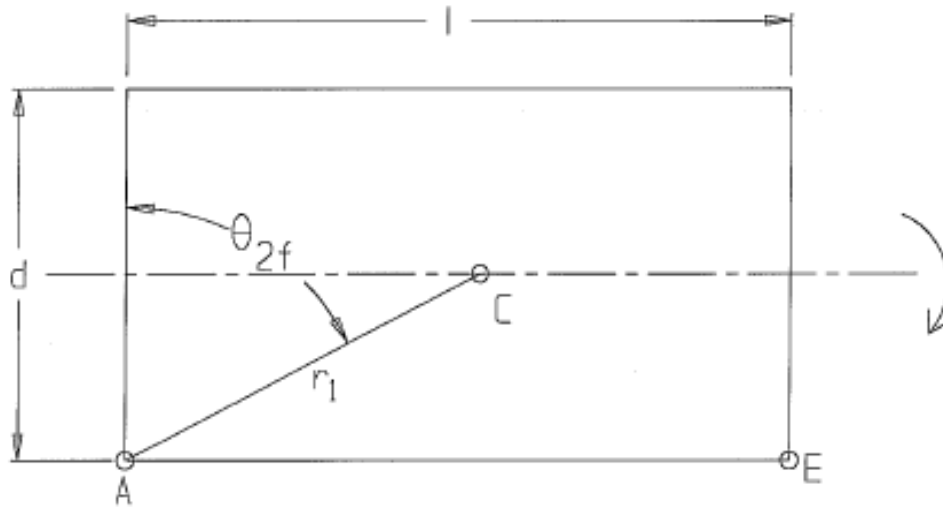
**FIGURE 3.A.14 OVERPACK CONCRETE COMPONENTS FINITE-ELEMENT MODEL  
(3-D VIEW)**



**FIGURE 3.A.15 MPC FINITE-ELEMENT MODEL (3-D VIEW)**

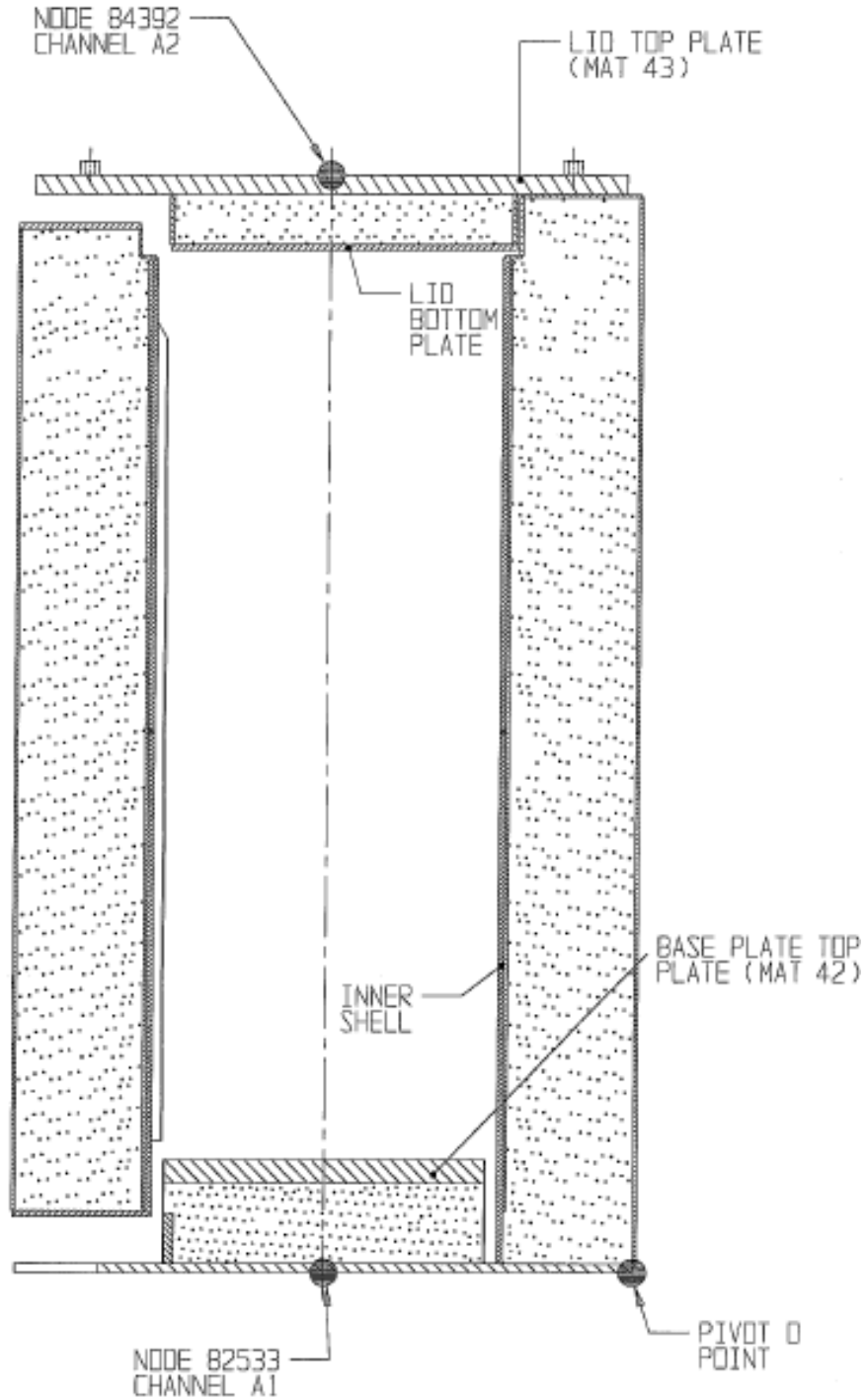


**FIGURE 3.A.16: PIVOT POINT DURING TIP-OVER CONDITION**



**FIGURE 3.A.17: TIP-OVER EVENT OVERPACK SLAMS AGAINST THE FOUNDATION DEVELOPING A RESISTIVE FORCE**





**FIGURE 3.A.18: MEASUREMENT POINTS AND CORRESPONDING FINITE-ELEMENT MODEL NODES**

## SUPPLEMENT 3.I

### STRUCTURAL EVALUATION FOR THE HI-STORM 100U SYSTEM

#### 3.I.0 OVERVIEW

In this supplement, the structural adequacy of the HI-STORM 100U Vertical Ventilated Module (VVM) is evaluated pursuant to the guidelines of NUREG-1536. The organization of technical information in this supplement mirrors the format and content of Chapter 3 except that it only contains material directly pertinent to the HI-STORM 100U VVM.

The HI-STORM 100U VVM serves as the storage space for the loaded MPC and consists of the CEC (the Container Shell, the Divider Shell and MPC Guides, and a welded Bottom Plate), and a lid consisting of plain concrete encased in structural steel arranged to provide appropriate inlet and outlet air passages (the Closure Lid). These individual components are collectively referred to as VVM Components. Interfacing SSCs that surround and support the VVM, as well as proximate structures, collectively referred to as ISFSI Structures, are explained in Supplement 2.I. Section 1.I.2 contains a complete description of the VVM Components and the ISFSI Structures (accompanied by appropriate figures) and their respective functions within the HI-STORM 100U ISFSI. The essential design details of both the VVM Components and the ISFSI Structures are set down in the licensing drawing in Supplement 1.I. The design basis loadings for the facility are provided in Supplement 2.I.

The applicable codes, standards, and practices governing the structural analysis of the HI-STORM 100U module as well as the design criteria, are also presented in Supplement 2.I. Throughout this supplement, in the context of the VVM components, the term “*safety factor*” is defined as the *ratio of the allowable stress (load) or displacement for the applicable load combination to the maximum computed stress (load) or displacement*.

For the ISFSI Structures made of reinforced concrete, the safety factor is defined as the ratio of the ultimate moment (or shear) capacity to the actual maximum moment (or shear) developed under the factored load combination.

MPC structural integrity has been evaluated in Chapter 3. In this supplement, the integrity of the MPC, due to its rattling motion inside the VVM storage cavity during a seismic event (a new loading condition in the underground storage configuration) is considered.

#### 3.I.1 STRUCTURAL DESIGN

##### 3.I.1.1 Discussion

The HI-STORM 100U system consists of three principal components: the Multi-Purpose Canister (MPC), the HI-STORM 100U storage module, herein denoted as the Vertical Ventilated Module (VVM) (includes the Cavity Enclosure Container (CEC) and the Closure Lid), and the HI-TRAC transfer cask. This supplement to Chapter 3 presents the structural evaluation of a VVM for the

applicable load cases summarized in Supplement 2.I (Table 2.I.5). In Section 3.I.4, the safety factors for each load case for the VVM Components are quantified. In addition, the safety evaluation of the ISFSI Structures is carried out using the factored load combinations from ACI-318 (2005) (see Table 2.I.11). Summary tables of bounding safety factors are provided for governing load combination for the ISFSI Structures. A licensing drawing for the HI-STORM 100U VVM is provided in Section 1.I.5. Table 2.I.1 provides a listing of the applicable regulations, codes and standards for the VVM Components and the ISFSI structures. The design of the VVM components and the ISFSI Structures is fully articulated in the licensing drawing and Table 2.I.2. The applicable Design Basis Earthquake is defined by the free field spectra shown in Figure 2.I.4.

### 3.I.1.2 Design Criteria

Design (and acceptance) criteria for the HI-STORM 100U VVM Components and the ISFSI structures are summarized in Tables 2.I.1 and 2.I.6.

### 3.I.1.3 Loads

Individual loads, applicable to the HI-STORM 100U System, are defined in Sections 2.I.4, 2.I.5, and 2.I.6, and load cases applicable to the VVM Components are summarized in Table 2.I.5. Table 2.I.11 contains load combinations applicable to the ISFSI Structures (reinforced concrete structures) in the HI-STORM 100U ISFSI.

### 3.I.1.4 Allowables

Allowable stresses for carbon steel and Alloy X used in the structural components of the HI-STORM 100U and the stored MPC are provided in Sections 3.1 and 3.3. The relevant data from those sections is reproduced here, as Tables 3.I.3 (a)-(d) to make the supplement self-contained.

### 3.I.1.5 Brittle Fracture

Brittle fracture considerations for HI-STORM 100U are bounded by HI-STORM 100 and 100S because of the VVM's underground configuration, and the use of the same material types and thicknesses as in the aboveground overpacks.

### 3.I.1.6 Fatigue

The HI-STORM 100U system is not subject to significant long-term cyclic loads. Therefore, failure due to fatigue is not a concern for the HI-STORM 100U system.

### 3.I.1.7 Buckling

The CEC Container Shell is the only component of the VVM subject to axial compression. However, since the shell is backed by a substrate, welded to a Bottom Plate at its base, and surrounded by the ISFSI Pad at the top, instability is not considered credible. The Divider Shell

does not experience any axial compressive stress that might induce buckling.

### 3.I.2 WEIGHTS AND CENTERS OF GRAVITY

Table 3.I.1 provides bounding weights of the individual HI-STORM 100U components.

The locations of the calculated centers of gravity (C.G's) are presented in Table 3.I.2 and are computed using the bounding weights. All centers of gravity are located on the VVM centerline.

Bounding weight values for the CEC and the Closure Lid include an overage on the weight generated by the CAD drawing package.

### 3.I.3 MECHANICAL PROPERTIES OF MATERIALS

Tables 2.I.3 and 2.I.8 list applicable codes, materials of construction, and ITS designations for all functional parts in the HI-STORM 100U system except for the MPC and its internals, which remain unchanged (listed in Table 2.2.6).

#### 3.I.3.1 VVM Steel Properties

Applicable material property and allowable stress tables in Chapter 3 for the VVM are reproduced in Tables 3.I.3 (a)-(c) for convenience.

#### 3.I.3.2 Unreinforced Concrete

The primary function of the unreinforced concrete in the HI-STORM 100U VVM Closure Lid is shielding. Unreinforced concrete is not considered as a primary load-bearing (structural) member. However, its ability to withstand compressive, bearing and penetrant loads under the design basis and various service conditions is analyzed. The allowable bearing strength of plain concrete for normal loading conditions is calculated in accordance with ACI 318 (2005) [2.I.5]. Table 3.I.4 provides a bearing limit consistent with the concrete compressive strength in the same table. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. Unless specifically called out in Table 3.I.4, Appendix 1.D provides requirements on unreinforced concrete.

#### 3.I.3.3 Reinforced Concrete

Reinforced concrete is used in the construction of the ISFSI Structures, namely, the retaining wall, the TSP, the VIP, and the SFP. All reinforced concrete in the HI-STORM 100U ISFSI will conform to stress criteria of ACI-318 (2005).

### 3.I.4 GENERAL STANDARDS FOR CASKS

In this section, new or additional material applicable to the HI-STORM 100U system is included.

Section 3.4 contains all required information associated with the MPCs and with the HI-TRAC transfer cask and is not repeated here. Results reported in this supplement section are generally applicable only to the HI-STORM 100U VVM.

### 3.I.4.1 Chemical and Galvanic Reactions

In order to provide reasonable assurance that the VVM will meet its intended Design Life of 40 years (the License Life is 20 years) and perform its intended safety function(s), chemical and galvanic reactions and other potentially degrading mechanisms must be accounted for in its design and construction.

The HI-STORM 100U VVM is a buried structure and as such chemical and galvanic reactions and other potentially degrading factors are, in some respects, more challenging than for aboveground models. Although the CEC is not a part of the MPC containment boundary, it should not corrode to the extent where localized in-leakage of water occurs or where gross general corrosion prevents the component from performing its primary safety function. In the following, considerations in the VVM's design and construction consistent with the applicable guidance provided in ISG-15 [3.I.3] are summarized.

All VVM components are galvanically compatible. Except for the CEC exterior surfaces, all steel surfaces of the VVM are lined and coated with the same surface preservative that is used in the aboveground HI-STORM overpacks. (The surface preservative used to protect HI-STORM 100S steel surfaces is a proven zinc rich inorganic/metallic material that protects galvanically and has self healing characteristics for added assurance). All exposed surfaces interior to the VVM, as stated in Supplement 1.I, are accessible for the reapplication of surface preservative, if necessary.

The steel Divider Shell requires insulation to perform its primary thermal function. The insulation selected shall be suitable for high temperature and high humidity operation and shall be foil faced, jacketed or otherwise made water resistant to ensure the required thermal resistance is maintained in accordance with Supplement 4.I. The high zinc content in the coating of the Divider Shell provides protection for both the Divider Shell and the jacketing or foil from any potential galvanic corrosion concerns. With respect to radiation resistance, the insulation blanket does not contain any organic binders. The damage threshold for ceramics is known to be approximately  $1 \times 10^{10}$  Rads. Chloride corrosion is not a concern since chloride leachables are limited and sufficiently low and the Divider Shell is not made from stainless steel [3.I.20]. Stress corrosion cracking of the foil or jacketing, whether made from stainless steel or other material is not an applicable corrosion mechanism due to minimal stresses derived from self-weight. The foil or jacketing and attachment hardware shall either have sufficient corrosion resistance (e.g. stainless steel, aluminum or galvanized steel) or shall be protected with a suitable surface preservative. The insulation is adequately secured to prevent significant blockage of the ventilation passages in case of failure of a single attachment (strap, clamp, bolt or other attachment hardware). The following table provides the acceptance criteria for the selection of insulation material for the Divider Shell and ranks them in order of importance.

<b>Acceptance Criteria for the Selection of the Insulation Material</b>	
<b>Rank</b>	<b>Criteria</b>
1	Adequate thermal resistance
2	Adequate high temperature resistance
3	Adequate humidity resistance
4	Adequate radiation resistance
5	Adequate resistance to the ambient environment
6	Sufficiently low chloride leachables
7	Adequate integrity and resistance to degradation and corrosion during long-term storage

Kaowool<sup>®</sup> ceramic fiber insulation [3.I.20] is selected as one that satisfies the acceptance criteria to the maximum degree. The Kaowool<sup>®</sup> insulation material provides excellent resistance to chemical attack and is not degraded by oil or water. Alternatively, a Holtec approved equivalent that meets the acceptance criteria set forth in the table above may be used.

The CEC Container Shell, which is exposed to the substrate, requires additional pre-emptive measures to prevent corrosion, if the substrate is of aggressive chemistry. This subsection provides a description of corrosion mitigation measures required to be implemented to protect the HI-STORM 100 VVM. Because the guiding principle in the HI-STORM Systems is to target a service life of 100 years so as to guarantee a design life of 40 years, these corrosion prevention measures are in addition to the preemptively incorporated standard corrosion allowance of 1/8-inch applied to the subterranean parts of the CEC in direct contact with the surrounding substrate. Calculation of the required CEC Container Shell and Bottom Plate thicknesses on a site-specific basis may indicate the availability of an additional corrosion reserve.

### **Soil Corrosivity and Corrosion Mitigation Measures for the Exterior of the CEC**

Corrosion mitigation of the exterior of the CEC warrants special consideration for the following reasons: (i) inaccessibility of the exterior coated surface after installation (ii) potential for a highly aggressive (i.e., corrosive) soil environment at certain sites, and (iii) potential for a high radiation field. Since the buried configuration will not allow for the reapplication of surface preservative, corrosion mitigation measures shall be determined after careful evaluation of the soil's corrosivity at the user's ISFSI site.

To evaluate soil corrosivity, a "10 point" soil-test evaluation procedure, in accordance with the guidelines of Appendix A of ANSI/AWWA C105/A21 [3.I.4], will be utilized. The classical soil evaluation criteria in the aforementioned standard focuses on parameters such as: 1) resistivity, 2) pH, 3) redox (oxidation-reduction) potential, 4) sulfides, 5) moisture content, 6) potential for stray current, and 7) experience with existing installations in the area. Using the procedure outlined in ref. [3.I.4], the ISFSI soil environment corrosivity is categorized as either "mild" for a soil test evaluation resulting in 9 points or less or "aggressive" for a soil test evaluation resulting in 10

points or greater. The following table details the corrosion mitigation measures that shall be implemented based on soil environment corrosivity:

<b>Implementation of Corrosion Mitigation Measures</b>			
<b>Soil Environment Corrosivity</b>	<b>Corrosion Mitigation Measures</b>		
	Coating (see note i)	Concrete Encasement (see note ii)	Cathodic Protection (see note iii)
<b>Mild</b>	Required	Choice of either concrete encasement or cathodic protection; or both	
<b>Aggressive</b>	Required	Optional	Required
Notes:			
i. An acceptable exterior surface preservative (coating) applied on the CEC.			
ii. Concrete encasement of the CEC external surfaces to establish a high pH buffer around the metal mass.			
iii. A suitably engineered impressed current cathodic protection system (ICCPS)			

The corrosion mitigation measures tabulated above are further detailed in the following subsections:

i. Coating

In addition to the corrosion allowance, the CEC shall be coated with a radiation resistant surface preservative designed for below-grade and/or immersion service. Inorganic and/or metallic coatings are sufficiently radiation resistant for this application; therefore, radiation testing is not required [3.I.5]. Organic coatings such as epoxy, however, must have proven radiation resistance [3.I.5] or must be tested without failure to at least  $10^7$  Rad. Radiation resistance to lower radiation levels is acceptable on a site-specific basis. Radiation testing shall be performed in accordance with ASTM D 4082 [3.I.6] or equivalent. The coating should be conservatively treated as a Service Level II coating as described in Reg. Guide 1.54 [3.I.7]. As such, the coating shall be subjected to appropriate quality assurance in accordance with the applicable guidance provided by ASTM D 3843-00 [3.I.8]. The coating should preferably be shop applied in accordance with manufacturer's instructions and, if appropriate, applicable guidance from ANSI C 210-03 [3.I.9]. The Keeler & Long polyamide-epoxy coating, according to the manufacturer's product data sheet [3.I.10], is pre-tested to radiation levels up to  $1 \times 10^9$  Rads without failure. The following table provides the acceptance criteria for the selection of coatings for the exterior surfaces of the CEC and ranks them in order of importance.

<b>Acceptance Criteria for the Selection of Coatings</b>	
<b>Rank</b>	<b>Criteria</b>
1	suitable for immersion and/or below grade service

Acceptance Criteria for the Selection of Coatings	
Rank	Criteria
2a	compatible with the ICCPS (if used) <ul style="list-style-type: none"> <li>• adequate dielectric strength</li> <li>• adequate resistance to cathodic disbondment</li> </ul>
2b	compatible with concrete encasement (if used) <ul style="list-style-type: none"> <li>• adequate resistance to high alkalinity</li> </ul>
3	adequate radiation resistance
4	adequate adhesion to steel
5	adequate bendability/ductility/cracking resistance/abrasion resistance
6	adequate strength to resist handling abuse and substrate stress

The Keeler & Long polyamide-epoxy coating is selected as one that satisfies the acceptance criteria to the maximum degree. Alternatively, a Holtec approved equivalent that meets the acceptance criteria set forth in the table above may be used.

ii. Concrete Encasement

The CEC concrete encasement shall provide a minimum of 5 inches of cover to provide a pH buffering effect for additional corrosion mitigation. The above concrete cover thickness has been conservatively determined for a 100-year service life in a strongly aggressive environment based on the concrete corrosion/degradation data provided in the literature [3.I.12, Table 5.3] (1.2 mm/yr surface depth failure rate). The required 5 inch minimum thickness is more conservative than that recommended in ACI Codes, such as ACI 318 [3.3.2], which call for up to 3 inches of concrete cover over steel reinforcement in aggressive environments. Considering that the concrete encasement is restricted to mild soil environments (unless used in conjunction with cathodic protection) and has a non-structural role, the 5 inch concrete encasement thickness is considered more than sufficient to provide reasonable assurance that a 40 year service life can be achieved. The lowest part of the CEC sits in a recessed region of the Support Foundation with an annular gap normally filled with substrate. If present, the CEC concrete encasement slurry will fill this annular gap during construction.

The function of the concrete encasement is for corrosion mitigation only; however, cracks larger than hairline cracks may significantly reduce its effectiveness. To control size and population of cracks, concrete reinforcement is included. The following reinforcement methods may be applied:

- a. Fiber reinforcement: Fiber reinforcement may be of several materials, including steel, glass and plastic (polypropylene). The selection of the fiber reinforcement material shall be such that adequate resistance to radiation and high alkalinity is maintained. If using steel fibers, adequate damage protection of the CEC coating shall be ensured during concrete placement per written procedures. Steel fiber shall be implemented using written procedures and the applicable guidance from ACI 544.3R [3.I.25] or a similar consensus code or standard.



Fiber reinforcement materials other than steel shall be implemented using written procedures, manufacturer recommendations and applicable guidance from ACI, ASCE and/or ASTM. One such document is ASTM C1116-03 [3.I.26].

- b. Steel wire reinforcement: Steel wire reinforcement shall be implemented in accordance with written procedures and the guidance from ACI 318 [3.3.2] or more recent version. For corrosion protection, the steel wire reinforcement shall have a concrete cover of approximately 2 to 3 inches from the interfacing substrate.

Regardless of reinforcement method, the material selected shall be corrosion resistant or otherwise appropriately coated (e.g. epoxy coated steel wire) for corrosion resistance.

The concrete encasement shall be installed in accordance with Holtec approved procedures following applicable guidance from the ACI code (e.g. ACI 318 [3.3.2]), as appropriate, for commercial concrete. Installation procedures shall address mix designs (incorporating Portland cement), testing, mixing, placement, and reinforcement, with the aim to enhance concrete durability and minimize voids and micro-cracks.

iii. Impressed Current Cathodic Protection System (ICCPS)

For a particular ISFSI site, the user may choose to either extend an existing ICCPS to protect the installed ISFSI, or to establish an autonomous ICCPS. The initial startup of the ICCPS must occur within one year after installation of the VVM to ensure timely corrosion mitigation. In addition, the ICCPS should be maintained operable at all times after initial startup except for system shutdowns due to power outages, repair or preventive maintenance and testing, or system modifications. Because there are a multitude of ISFSI variables that will bear upon the design of the ICCPS for a particular site, the essential criteria for its performance and operational characteristics are set down in this FSAR, which the detailed design work for each ISFSI site must follow.

<b>Design Criteria for the Impressed Current Cathodic Protection System</b>
---

- |   |
|---|
| <p>a. The cathodic protection system shall be capable of maintaining the CEC at a minimum (cathodic) potential as required by NACE Standard RP0285-2002 [3.I.21].</p> <p>b. The ICCPS shall include provisions to infer its proper operation and effectiveness on a periodic basis.</p> <p>c. The system shall be designed to mitigate corrosion of the CEC for its design life.</p> <p>d. The cathodic protection system design, installation, operation, testing, and maintenance shall follow the applicable guidelines of:</p> <ul style="list-style-type: none"> <li>- 49CFR195 Subpart H “Corrosion Control”, Oct. 1, 2004 edition [3.I.13]</li> <li>- NACE Standard RP0285-2002 “Corrosion Control of Underground Storage Tank Systems by Cathodic Protection” [3.I.21]</li> </ul> |
|---|

The following standards and/or publications may also be utilized for additional guidance in the design, installation, operation, testing, and maintenance of the ICCPS as needed (in case of conflict, the guidelines of item d above shall prevail):

- API RP1632, “Cathodic Protection of Underground Petroleum Storage Tanks and Piping Systems” [3.I.22]
- NACE RP0169-96, “Control of External Corrosion on Underground or Submerged Piping Systems” [3.I.23]
- 49CFR192 Subpart I, “Requirements for Corrosion Control”, Oct. 1, 2004 edition [3.I.24]
- Other standards or publications referenced by any of the above three standards and publications.

Records of system operating data necessary to adequately track the operable status of the ICCPS shall be maintained in accordance with the user’s quality assurance program.

Finally, the surface preservative used to coat the CEC must meet the requirements described in (i) above but must also be compatible with cathodic protection and resistant to the alkaline conditions created by cathodic protection and/or concrete encasement. Organic coatings, such as the Keeler & Long coating selected for (i) above, are inherently compatible with both cathodic protection [3.I.11] and concrete [3.I.10].

#### 3.I.4.2 Positive Closure

There are no quick-connect/disconnect ports in the confinement boundary of the HI-STORM 100U system. Because the only access to the MPC is through the VVM Closure Lid, which weighs well over 10 tons, inadvertent opening of the VVM cavity is not feasible.

### 3.I.4.3 Lifting Devices

As required by Reg. Guide 3.61, lifting operations applicable to the VVM lid are analyzed. Because of the nature of the HI-STORM 100U system, lid placement or removal may occur with a loaded MPC inside the VVM cavity; these are the sole operations requiring analysis in accordance with Reg. Guide 3.61 and are examined in this supplement.

As discussed in Subsection 3.4.3, the lifting component itself (the four lift lugs) must meet the primary stress limits prescribed by NUREG-0612 and Regulatory Guide 3.61; the welds in the load path, near the lifting holes, are required to meet the condition that stresses remain below yield under three times the lifted load (per Reg. Guide 3.61). Further, for additional conservatism, away from the lifting location, the ASME Code limit for the Level A service condition applies.

The lifting analysis results summarized below include a 15% inertia amplifier.

#### HI-STORM 100U VVM Closure Lid Lifting Analysis (Load Case 05 in Table 2.I.5)

The four lifting lugs are conservatively analyzed to ANSI N14.6 stress limits using simple strength of materials calculations. Each of four lugs is considered as a cantilever beam attached to the lid and carries 25% of the lid weight. The bending moment and shear force at the root of the cantilever (where it is attached to the lid) is computed and the maximum stress is conservatively compared with the minimum of the yield strength/6 or the ultimate strength/10. As required, increasing the lid weight by 15% includes inertia effects. Using the calculated bending moment and shear force at the root of the lug, the structural evaluation of the weld attaching the lug to the lid is performed and compared with the requirements of Regulatory Guide 3.61. The results from these two calculations demonstrate that the required safety factors are substantially greater than 1.0 (exceeding the requirements of NUREG-0612 and Reg. Guide 3.61). The details of the calculations are presented in the calculation package supporting this submittal [3.I.27]. Lifting slings that attach to the lugs shall be sized to meet the safety factors set forth in ANSI B30.3.

To evaluate the global state of stress in the lid body, a finite element model of the lid, which includes contact interfaces between steel and concrete, is constructed to evaluate the state of stress under lifting conditions. Figure 3.I.1 shows the constructed ANSYS finite element model. The lifted scenario is simulated by fixing the four lifting locations at the lift lug sling attachment location, and applying an appropriate weight density to match the lifted weight. The results are evaluated for satisfaction of normal condition (ASME Level A) limits at the appropriate locations.

The table below summarizes key results obtained from the lifting analyses for the HI-STORM 100U VVM Closure Lid for a bounding set of input design loads.

HI-STORM 100U VVM Lid Lifting Analyses (Load Case 05 in Table 2.I.5)			
Item	Calculated Value	Allowable	Safety Factor
Bending of Lift Lugs (kip)(ANSI N14.6)	4.000	5.275	1.32 (see Note 1)
Shear in Lift Lugs (kip)(ANSI N14.6)	1.609	3.165	1.97 (see Note 1)
Load in Welds Near Lifting Lugs (kip) (Reg. Guide 3.61)	5.657	6.33	1.12 (see Note 2)
Primary Stress in Lid (ksi)(ASME Level A Limit)	< 10	26.25	> 2.63
Note 1: Computed safety factors represent the margin over that required by NUREG-0612 (0.1 x ultimate load).			
Note 2: Computed safety factor is based on 60% of yield strength for base metal and represents margin over limit set by Reg. Guide 3.61.			

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100U VVM Closure Lid. All factors of safety, using applicable criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports, from USNRC Regulatory Guide 3.61, and from NUREG-0612, are greater than 1.0.

#### 3.I.4.4 Heat

##### i. Summary of Pressures and Temperatures

Tables 2.I.1 and 2.I.2 present applicable design inputs for the HI-STORM 100U VVM. No new inputs are required for the HI-TRAC and the MPC.

##### ii. Differential Thermal Expansion

All clearances between the MPC and the HI-STORM 100U VVM are equal to or larger than the corresponding clearances in the aboveground HI-STORM 100 systems (see Section 4.4). Therefore, no interferences between the MPC and the VVM will occur due to thermal expansion of the loaded MPC. The Divider Shell is insulated on one surface and is exposed to heated air on the other shell surface. Therefore an analysis to demonstrate that free axial thermal expansion of the Divider Shell will not close the initial gap between the top end of the Divider Shell and the base of the Closure Lid is provided. The Divider Shell is considered as a heated member, subject to an average temperature increase over its entire length. The actual axial absolute temperature profile can be integrated over the length of the Divider Shell to define the average absolute temperature. Once the average absolute temperature is known, the free thermal growth is computed and compared with the provided gap between the Divider Shell and the Closure Lid.

The average temperature rise above ambient is bounded by DT (ambient is 80°F per Table 2.I.1,

and average metal temperature over the length of the Divider Shell is from Table 4.I.3, footnote):

$$DT = (300^{\circ}\text{F} - 80^{\circ}\text{F}) = 220^{\circ}\text{F}$$

From Table 3.I.3 (a), a bounding coefficient of thermal expansion, appropriate to DT, is:

$$\alpha = 6.27 \times 10^{-6} \text{ in./in.} \cdot ^{\circ}\text{F}$$

The nominal length of the divider shell is:

$$L = 221.5625''$$

Therefore, the free thermal expansion, based on the nominal length is  $\alpha \times L \times DT$ , and is computed and compared against the nominal gap provided (as shown in the licensing drawing).

Key Result from Free Thermal Growth Analysis of Divider Shell

Item	Bounding Value	Allowable Value*	Safety Factor
Thermal Growth (inch)	< 0.4	0.5	>1.25 (against contact)
*This is the nominal gap provided between the top end of the Divider Shell and the Closure Lid Surface (see Dwg. 4501, sheet 4 in Subsection 1.I.5).			

iii. Stress Calculations – VVM Components

a. HI-STORM 100U VVM Stresses Under Transporter Loading and Substrate Overburden (Load Case 07 in Table 2.I.5)

During HI-STORM 100U system loading, a HI-TRAC transfer cask with a fully loaded MPC is placed over a HI-STORM 100U VVM using a specially designed transporter and a lifting device meeting “single-failure proof” requirements, as applicable. The transfer cask is connected to the CEC using an ancillary mating device (see Figure 3.I.4). Although a handling accident is not credible, the CEC must possess the capacity to support any transporter loads imposed at and below the substrate surface during the short time when the transporter is positioned over a VVM cavity and carrying the weight of the loaded HI-TRAC (i.e., before the HI-TRAC is placed on the mating device). This loading condition leads to a maximum sub-surface lateral pressure on the CEC shell which may potentially cause its ovalization. This configuration also includes the loaded transporter traveling over a previously loaded VVM on its way to an empty CEC.

Table 3.I.1 gives the essential data on the representative transporter including its loaded weight and its track length and width (i.e., size of the load patch (Figure 3.I.5)). The average normal pressure, at the transporter track and TSP interface is computed by dividing the weight of the loaded transporter by the total area of the two load patches.

To determine the stress and displacement field in the CEC due to the combined action of the loaded transporter and the soil overburden, a 3-D ANSYS model of a VVM (see Figure 3.I.2) is prepared.

The finite element model has the following attributes:

- The soil is modeled as an elastic continuum with properties specified in Tables 2.I.2 and 3.I.5. The VVM Interface Pad (VIP), which is separated from the Top Surface Pad (TSP) by a construction joint, is unaffected by the deflection of the TSP under the transporter weight. The VIP essentially is a dead weight on the soil column below and is appropriately incorporated in the model. To appropriately model the VIP within the confines of a linearly elastic construct, it is represented by a material with a very low Young's Modulus, but the correct weight density. This modeling assumption provides the appropriate weight on the substrate from the VIP but provides no additional strength to the TSP or to the CEC.
- The minimum CEC pitch from the licensing drawing is used.
- The TSP, shown in the licensing drawings, is represented by its appropriate elastic properties (Table 3.I.4).
- The soil mass surrounding the ISFSI is assumed to be constrained from expansion across the planes of symmetry (so as to maximize the Poisson compression load on the CEC). The bottom of the soil continuum extends to the SFP.
- The CEC shell is assumed to have its nominal un-corroded thickness; the stress and strain results are subsequently adjusted to reflect the postulated corrosion allowance (see Table 2.I.1).
- To linearize the problem, the soil is assumed to be bonded to all interfacing surfaces.

Table 3.I.10 provides the input data used in the analysis.

The results of the stress analysis are pictorially shown in Figure 3.I.11, where stress intensity in the CEC is plotted. As can be seen from the figure, the maximum primary stress intensity value is 1,390 psi based on the nominal shell thickness of 1 in. Accounting for the corrosion allowance in the CEC shell, the maximum stress intensity (essentially bending in nature) is appropriately adjusted to 1,816 psi  $((1 \text{ in}/0.875 \text{ in})^2 \times 1390 \text{ psi})$ . When compared with the Level A stress limit from ASME code Section III, Subsection NF (per Table 2.I.5), the maximum computed stress intensity provides a factor of safety:

$$SF = \frac{\textit{allowable}}{\textit{actual}} = \frac{26.25}{1.82} = 14.4$$

Because the stresses in the CEC shell remain elastic, no reduction in the diametrical opening of the CEC due to plastic deformation is indicated. Therefore, the retrievability of the MPC is assured.

- b. HI-STORM 100U Lid Integrity Evaluation for Normal plus Explosion Loads, CEC Container Shell Evaluation Under Bounding Vertical Load (Load Case 02 in Table 2.I.5), and Design Basis Fire (Load Case 06 in Table 2.I.5)

The VVM Closure Lid rests on the CEC and resists vertical loads, arising from dead weight, and from induced loadings from explosions, from seismic accelerations, and from tornado missile impact. In this subsection, the analysis considers only the normal loading condition plus the steady pressure bounding the explosion pressure (see Table 2.I.1). The finite element model shown in Figure 3.I.1 is used to obtain this solution; the Closure Lid vertical support is now all around and is provided by the CEC Container Shell Flange (instead of by the lift lugs). The stresses from the solution are compared, per the criteria in Table 2.I.5, with allowable stress values for plate and shell structures as provided in ASME Section III Code, Subsection NF. The allowable stress intensity is per Table 3.I.3 (c) for Level D conditions at a bounding temperature of 350 °F.

The vertical load on the Container Shell ring flange, which can be computed from equilibrium, does not bound the vertical load under normal conditions when the Closure Lid is removed and replaced by a loaded HI-TRAC plus a Mating Device. The bounding vertical load during the transfer operation is an input for the evaluation of the Container Shell for this load case using Strength of Materials methodology. Key results from the analysis of the Closure Lid under the normal loading condition plus the steady pressure, and the follow-on analysis of the corroded Container Shell under the bounding vertical load (during the MPC transfer operation) are summarized in the following table:

Stress Analysis of the Closure Lid and CEC Container Shell Under Bounding Vertical Load During Normal Operations (Load Case 02 in Table 2.I.5)			
Item	Bounding Value from calculations	Allowable Limit	Safety Factor
Maximum Primary Principal Stress Anywhere in Lid (ksi)	< 12.0	59.65(Level D Stress Intensity Limit) 26.25 (Level A Stress Limit)	> 4.97* > 2.19*
CEC Container Ring Flange Weld (kips)	< 300	3,018	> 10.06
Compression Stress in CEC Container Shell Under Bounding Vertical Load (ksi)	< 1.425**	17.5	> 12.28

Stress Analysis of the Closure Lid and CEC Container Shell Under Bounding Vertical Load During Normal Operations (Load Case 02 in Table 2.I.5)			
Item	Bounding Value from calculations	Allowable Limit	Safety Factor
* The results from the analysis are presented in terms of principal stresses for simplicity. Safety factors are determined by comparison with the Level D stress intensity limits (Table 3.I.3(c)), or with Level A stress limits (Table 3.I.3 (b)). Regardless of the measure used, the safety factors are large.			
** The bounding compressive stress is based on a fully corroded shell thickness and also conservatively includes the full weight of the CEC in addition to the bounding load at the top.			

From the above results, it is concluded that there is minimum structural demand on the HI-STORM 100U Closure Lid and CEC Container Shell during normal operation (even if the explosion pressure is conservatively considered as a normal condition).

With respect to the fire event (Load Case 06 in Table 2.I.5), where the Closure Lid steel temperature rises to the limit set in Table 2.I.5, it is noted from Tables 3.I.3 (a) and (b) that the Level A stress limit is reduced to 0.68 of the room temperature value, the yield strength is reduced to 0.66 of its room temperature value, and the ultimate strength is reduced to 0.92 of its room temperature value. From the stress values obtained in the lid (even with the explosion 10 psi surface pressure load included), it is evident that a total collapse of the lid due to reduction of the ultimate strength is not credible.

Seismic loading on the VVM is considered in Subsection 3.I.4.7 (Load Case 04 in Table 2.I.5). Subsection 3.I.4.8 considers tornado missile impact (Load Case 03 in Table 2.I.5).

iv. Stress Calculations – ISFSI Structures

The 100U ISFSI consists of plate-type reinforced concrete structures whose minimum section strength properties are defined by Table 2.I.2 and the licensing drawings. The ISFSI is supported by the subgrade underneath the SFP, which may include pilings, if required, to meet the effective stress wave velocity in Table 2.I.2. The loadings on the ISFSI are:

- a. Dead load of the VVM and the concomitant effect of settlement over the Design Life of the system. (D in Table 2.I.11). The method to incorporate the effect of long-term settlement of the subgrade underneath the SFP (may also be referred to as the undergrade), described in Subsection 2.I.2, is used. This method essentially consists of using the deflection properties of the different layers to define equivalent elastic properties of the subgrade underneath the SFP. In the finite element analysis of the SFP, the degraded elastic properties of the subgrade underneath the SFP are utilized to account for the effect of long-term settlement. The long-term settlement of the subgrade underneath the TSP and VIP is also considered in a similar manner.



The Dead load on the SFP from the weight of the loaded VVM's nearly equals the weight of the earth removed. Therefore, the long-term settlement of the SFP is expected to be quite small. Likewise, the dead load on the TSP and the VIP is relatively small (from self-weight of the pads).

The retaining wall under excavated condition (see Subsection 2.I.2) supports the soil overburden pressure (classified herein as Dead load).

- b. Live load from the loaded transporter acts directly on the TSP (see Figure 3.I.4 and 3.I.5). This load also adds to the overall load on the SFP (L in Table 2.I.11). The load from the transporter is the sole live load applicable to the ISFSI structures. For structural qualification, the loaded transporter (live load) is assumed to be situated over the centrally located cavity.
- c. Seismic load is computed using the methodology presented in Subsection 3.I.4.7. This load, denoted as E in Table 2.I.11, is the aggregate of the peak dynamic load exerted on the ISFSI less the dead weight. For conservatism, the load E is applied as a static load in the stress analysis of ISFSI structures even though it is impulsive in nature.

Paragraph 3.I.4.7.3 contains details on the stress analysis of the ISFSI structures to demonstrate ACI code compliance.

#### 3.I.4.5 Cold

Due to its subterranean configuration, the structural components of the VVM are relatively protected from extremes in the ambient temperature in comparison to the HI-STORM 100 or 100S overpacks. Therefore, no new analyses are identified for the HI-STORM 100U system.

#### 3.I.4.6 Flood

The buried configuration of the HI-STORM 100U system renders it immune from sliding under the action of a design basis flood. No new analyses are needed for an actual extreme environmental event.

Although the condition does not necessarily arise due to a flood, a limiting uplift scenario where the VVM CEC is in place and the surrounding substrate produces a buoyant force by unspecified means is considered. For this condition (Load Case 01 in Table 2.I.5), the limiting uplift condition determines the minimum weight that needs to be in place to prevent uplift during construction. This could be in the form of a temporary cover. The upward directed buoyant force exerted on the CEC cavity is computed assuming a weight density of water and compared with the dead weight of the CEC. Under the postulated condition, the net uplift load (Buoyant Force – Weight of CEC) can be calculated. The required temporary weight that is needed to produce a net downward force is calculated in [3.I.27] and specified in Table 2.I.5.

For the case of a loaded VVM with the Closure Lid in place, or for an empty CEC with the Closure Lid in-place, the buoyant force is less than the vertical download, so there is no uplift.

Should the full buoyant force develop from any means, a lateral pressure load is imposed on the CEC bottom plate. Conservatively assuming an empty VVM, the full buoyant force provides a pressure causing bending of the CEC Bottom Plate, which is partially restrained against rotation by the CEC shells (note that in a loaded VVM, the MPC also helps to support the Bottom Plate of the CEC as its weight causes the central shim to act as a support for the Bottom Plate of the CEC). The stress intensity resulting from CEC Bottom Plate bending is compared to the Level D allowable stress intensity. Using the solutions for maximum stress in a clamped and simply supported plate, and averaging the results from the two solutions to approximately account for the rotational restraint provided by the CEC Container Shell, gives the following bounding safety factor for stress in the bottom plate under the postulated buoyancy loading:

Allowable Stress = 66,875 psi (Table 3.I.3(c) @ 125 °F per Table 2.I.5). Safety Factor is calculated to be greater than 4.0.

### 3.I.4.7 Seismic Event - HI-STORM 100U (Load Case 04 in Table 2.I.5)

The HI-STORM 100U system, plus its contents, may be subject to the Design Basis Earthquake (DBE) defined by the response spectra in Figure 2.I.4. As mentioned in supplement 2.I and further explained in this subsection, the DBE has been defined for the 100U ISFSI to insure that the operative spectra (Figure 2.I.4) essentially envelope the corresponding site DBE spectra at virtually all US sites. Because the VVM is buried in the substrate, tipover of the VVM is not credible. The entire VVM can move laterally with the surrounding and supporting substrate.

Under the action of lateral seismic loads, the CEC Container Shell globally acts as a beam-like structure supported on a foundation driven by the site seismic accelerations. During a seismic event, the lateral loading on the CEC consists of:

- i) Inertia force from CEC self-weight
- ii) Inertia forces from the Closure Lid self-weight
- iii) Inertia forces from the self weight of the VIP
- iv) Interface forces from the rattling of the MPC within its confines of the CEC and the rattling of the contents inside the MPC
- v) Interface forces from the subgrade and from the SFP

The CEC Container Shell develops longitudinal stresses as it bends like a beam to resist the input seismic loads. In addition, the CEC Container Shell tends to ovalize under the loads. Both effects are captured in the seismic analysis.

The Design Basis Seismic Model (DBSM) used to perform the safety analysis of the 100U ISFSI under the Design Basis Earthquake (DBE) defined by Figure 2.I.4 is described in the following.

#### 3.I.4.7.1 Design Basis Seismic Analysis Model

Parametric studies were performed to support the initial certification of the HI-STORM 100U

VVM. These studies defined the Design Basis Seismic Model. In particular, a non-linear dynamic model on LS-DYNA was found to produce much greater response and internal stresses than a linear analysis on SASSI. Further, a 5x5 VVM array model was standardized for dynamic analysis purposes. Accordingly, LS-DYNA is used for all required dynamic analysis of the VVM array. The DBSM consists of three discrete models, namely:

1. A VVM Array Model used to characterize the interaction of the ISFSI with the surrounding soil continuum. This is performed using a 5x5 VVM array (see Figure 3.I.3-B).
2. A VVM Array Model for the optional 100U design where retaining walls are in place (see Figure 3.I.3-C). The lateral subgrade beyond the retaining wall is assumed to be removed all the way down to the bottom of the SFP, which conservatively represents an excavation configuration.
3. A single VVM model with a detailed simulation of the internal parts of the VVM to obtain an accurate characterization of the stress/displacement field (see Figure 3.I.3-D).

The seismic analysis consists of three discrete steps, namely:

- A. Soil-structure model development.
- B. Use of the VVM Array Model to determine the bounding dynamic loads applied to the ISFSI Structures.
- C. Use of the Single VVM Model to compute stresses in the VVM Components.

#### A. Soil–Structure Model Development

- i. Based on the lower bound shear wave velocity profile of US nuclear power plants (Figure 2.I.6), a two-step earthquake response analysis using the computer code SHAKE2000 and LS-DYNA is performed to establish a bounding seismic loading condition for the 100U underground fuel storage system. The Design Basis Earthquake for the HI-STORM 100U system thus obtained is defined by the seismic response spectra at both the ground surface and the ISFSI foundation surface elevations as shown in Figure 2.I.4. The input seismic acceleration time history used in the first step (SHAKE) analysis is derived from the Regulatory Guide 1.60 seismic response spectrum and designated as the rock outcrop motion. The input acceleration time history is scaled to yield ground surface ZPAs (at the top of grade elevation) specified in Table 2.I.2. The 1-D SHAKE analysis model consists of 21 native soil layers of the 100U ISFSI site with a total thickness of 101 ft; the top of the 6<sup>th</sup> soil layer is aligned with the bottom of the SFP. The total soil depth of the SSI Model is about five times the height of the underground ISFSI (Due to the limitation of the linear code, a further increase of the soil depth in the SHAKE model leads to questionable seismic response results in the case of a strong seismic motion and weak soil properties). The averaged strain compatible shear wave velocity is 450 ft/s for the soil layers above the SFP and is 485 ft/s for the layers below the SFP, which has been set as the lower-bound soil design data in Table 2.I.2 for a candidate 100U ISFSI site. The finite

element soil model in the second step (LS-DYNA seismic response analysis) uses the average strain-compatible wave velocities obtained from the SHAKE analysis to represent the soil layers above and below the SFP elevation. The acceleration time history at the soil column bottom surface, also obtained from the SHAKE analysis in the first step, is used as the input seismic motion for the LS-DYNA seismic response analysis. The response spectrum plots shown in Figure 2.I.4 are the results of the LS-DYNA seismic response analysis.

Figure 3.I.3-A shows the LS-DYNA soil model for the seismic response analysis. Note that the lateral dimension of the ISFSI soil model is significantly greater than that of the ISFSI. The periphery nodes of the soil model space at the same elevation are constrained to move together to simulate the seismic response of the semi-infinite space of soil. According to the numerical study on various lateral boundary conditions of the finite element soil model [2.I.10], this lateral boundary condition, also known as a “slave boundary condition”, is appropriate to predict the soil response in a seismic event. The same soil model and input seismic motion used in the LS-DYNA seismic response analysis will be used for the LS-DYNA soil-structure interaction analysis for the 100U ISFSI loaded with VVMs. The boundaries of the soil model are sufficiently away from the ISFSI pads to ensure that structural response of the ISFSI will not be significantly affected.

- ii. The spectra in Figure 2.I.4 define the seismic input against which the spectra at a candidate ISFSI site should be compared to determine whether the generic analysis in this FSAR is bounding or additional site specific analysis set down per sub-section 2.I.6 are required.
- iii. Consistent with the sketch in Figure 2.I.5, the 100U soil-structure LS-DYNA model consists of loaded VVMs, concrete pads, and soil spaces with properties as defined in Tables 2.I.2 and 3.I.4. The ISFSI model is developed based on a 5×5 VVM configuration, which has previously been approved under LAR 1014-6 and is considered to be appropriate for capturing the effect of the ISFSI size on the structural analysis results. Depending on the purpose of the analysis, the 100U soil-structure model may include 5x5 fully loaded VVMs or just one loaded VVM. Similarly, a loaded Vertical Cask Transporter (VCT) may be considered in the model to obtain the bounding load applied to the TSP and to demonstrate the seismic stability of the loaded VCT. For the optional ISFSI design including a retaining wall, the soil-structure model is developed based on the governing configuration where the subgrade outside the retaining wall is excavated all the way to the depth of the SFP elevation. Therefore, a total of three 100U soil-structure LS-DYNA models (see Figures 3.I.3-B to 3.I.3-D) are developed to perform the design basis earthquake analysis.
- iv. The corrosion of the CEC is considered by using a reduced thickness (i.e., 1/8” thinner than the nominal thickness) in the soil-structure LS-DYNA models.
- v. Proper element size and time step controls in the dynamic model are implemented

following the guidance in references [3.I.28] and [3.I.29].

## B. VVM Array Model

The object of the VVM Array model is to obtain conservative values of the loads on the ISFSI structures under the Design Basis Earthquake (Figure 2.I.4). The VVM Array model has the following essential attributes:

- i. The MPC is represented by a solid rigid cylinder of mass equal to its total mass. This means that all internal masses will move in unison and the inertia forces of the MPC are maximized, which will conservatively result in greater impact loads applied to MPC guides and the CEC base plate.
- ii. The Divider Shell and the CEC shell are modeled as elastic shells but the Closure Lid and the Lid Ring are simulated as rigid bodies. Note that the combination of elastic shells and rigid lid ring used in the finite element model has little effect on the load path between the Divider Shell and the CEC flange during the seismic event.
- iii. The ISFSI pads (i.e., TSP, SFP, etc.) are simulated as a flexible plate-type structure, as is the retaining wall, if used. The retaining wall is added to the finite element model in the optional ISFSI design case (see Table 3.I.6).
- iv. The SFP is fully loaded with a 5×5 VVM array.
- v. A loaded VCT is assumed to be located at the center of the fully populated ISFSI except for the case with retaining walls. The VCT, along with the carried transfer cask, is modeled as a freestanding rigid body.
- vi. The elastic material model is used for all ISFSI concrete structures except for the TSP, which is characterized by an inelastic concrete model to account for energy dissipation in the concrete due to the impact loading from the loaded VCT. For the case where cracking of the concrete needs to be considered, the Young's Modulus of the SFP is reduced to 50% of its nominal value per the guidance in Section 3.4 of [3.I.29].

## C. Single VVM Model

The Single VVM model is used to perform the safety evaluation of the VVM components and the stored MPC under the Design Basis Earthquake. The applicable acceptance criteria are provided in Table 2.I.6. To conservatively evaluate the structural integrity of the VVM components, the Young's Modulus for the SFP is assumed to be equal to 50% of its nominal value. This is prompted by the results of VVM Array Model runs (see Table 3.I.7), which indicate that the VVM Components experience amplified responses if the reduced modulus is used for the SFP.

The Single VVM model complies with the provisions set forth in the following:

- i. The SFP is loaded with only one VVM at the edge of the SFP. A loaded VCT, modeled as a freestanding rigid body, is conservatively assumed to be located above the center of the loaded VVM.

- ii. The Cavity Enclosure Container (CEC) is discretized by an appropriate finite element grid to simulate its Container Shell and Bottom Plate, the Divider Shell, and the MPC guides in an explicit manner. The true stress-strain relationship of the material is used to obtain the realistic deformation of these structural members.
- iii. The MPC shell, baseplate, and top lid are modeled using sufficient element discretization so that the peak primary stresses of the MPC components under the seismic loading condition can be captured for structural evaluation.
- iv. The fuel basket is modeled with thin shell finite elements arrayed to simulate inter-cell connectivity in an explicit manner.
- v. Nominal small gaps between the fuel basket and the MPC are explicitly modeled, as is the nominal gap between the MPC and the CEC at the upper and lower MPC guide locations.
- vi. Each fuel assembly is represented by an equivalent homogeneous, isotropic prismatic beam of an equivalent elastic modulus whose fundamental lateral natural frequency accords with that of the actual fuel assembly. A bounding fuel assembly weight is used and the fuel basket is assumed to be fully populated with fuel assemblies.
- vii. The seismic responses of MPC structural components are simulated using the elastic material model so that the stress results can be directly compared with the corresponding ASME NB stress limits.

#### 3.I.4.7.2 Qualification of VVM Components

The CEC Components and parts of the MPC subject to significant loadings during the DBE event are:

- a. CEC shell and Divider Shell (subject to ovalization)
- b. MPC shell (bending of the shell as a beam, resulting in axial membrane stress in the shell)
- c. MPC top and bottom guides
- d. Lateral loading on the fuel basket panels.
- e. Localized strain in the MPC shell (due to impact of the MPC with the MPC guides attached to the Divider Shell)

The safety analysis of each component under the DBE event is summarized below:

- a. CEC shell and Divider Shell: Maximum radial deformation of the two shells is tracked for the single VVM simulation scenario in Table 3.I.6. The ratio of the original ovalization to the actual ovalization gives the safety factor:

$$\begin{aligned}
 \text{Safety Factor} &= \frac{\text{Permissible radial displacement}}{\text{Maximum computed radial displacement from Figure 3.I.23}} \\
 &= \frac{2.5''}{0.1325''} = 18.86
 \end{aligned}$$

- b. Primary stress in the MPC shell: The maximum stress intensity in the MPC shell is computed under the single VVM simulation scenario. The allowable stress intensity for this case corresponds to the Level D condition. The safety factor is computed as:

$$\begin{aligned} \text{Safety Factor} &= \frac{\text{Level D allowable Stress Intensity from Table 3.I.3(d)}}{\text{Maximum computed primary Stress Intensity from Figure 3.I.24}} \\ &= \frac{42,000 \text{ psi}}{12,860 \text{ psi}} = 3.26 \end{aligned}$$

- c. Top and Bottom MPC Guides: The maximum lateral load bearing capacity of the top and bottom plate guides is computed in Supplement 4 of Reference [3.I.27]. The maximum dynamic impact loads from the single VVM model can be extracted from the impact load time history results shown in Figure 3.I.25. The safety factor is calculated as:

$$\begin{aligned} \text{Safety Factor} &= \frac{\text{MPC Guides Lateral Load Bearing Capacity}}{\text{Maximum MPC to MPC Guides Contact Force}} \\ &= \frac{4.41 \times 10^5 \text{ lb}}{108,826 \text{ lb}} = 4.05 \end{aligned}$$

For the tubular MPC top guide design, the MPC impact analysis documented in Supplement 11 of Reference [3.I.27] demonstrates that the tube guide would not experience any global plastic deformation under the Design Basis Earthquake condition. This means that there is no risk of progressive flattening of the guide tubes from repetitive impacts during the seismic event.

- d. Loading on the Fuel Basket panel: The fuel basket panels are qualified to withstand 45 g's of lateral acceleration (during the non-mechanistic tip-over event). The maximum fuel g-load predicted by the LS-DYNA simulation is 2.5 g's as shown in Figure 3.I.26. The factor of safety, therefore, will be equal to the ratio of the two. Hence,

$$\text{Safety Factor} = \frac{45}{2.5} = 18$$

- e. Maximum Local Strain in the Confinement Boundary in the Impact Region:

The small clearance between the MPC and the MPC guides can lead to a high localized strain in the region of the shell where the impact from rattling of the canister under a seismic event occurs. The extent of local strain from impact is minimized by locating

the MPC guide in the vertical direction such that the mid-height of the impact footprint is aligned with the bottom surface of the closure lid. Thus the location of impact is removed from the lid-to-shell weld junction. It is necessary to insure that the maximum value of the local (true) strain in the shell (confinement boundary) region of impact is well below the failure strain. For this purpose, the recommendation in [3.I.31] is used. The methodology for computing the local strain is presented in the following and applied to the seismic problem analyzed in this subsection.

A finite element model of the MPC suitable for implementation in LS-DYNA is prepared with special emphasis on the top region of the canister where a very fine grid is employed. All elements have elasto-plastic and large strain capability. The solid elements in the lid and the lid-to-shell weld are of type 2 (fully integrated) and those in the shell are type 16 (fully integrated). The integration across the shell wall employs the maximum number of points available in LS-DYNA (10 points). A mesh sensitivity study has been performed using a finer grid size for the MPC shell to verify that the results are converged.

The MPC contents, namely the fuel basket and the SNF, are modeled exactly as set forth in the DBSM in the foregoing (articles (iii.), (iv.), and (v.) in Subsection 3.I.4.7.1 C Single VVM Model). To define a conservative scenario of MPC/MPC guide impact, the velocity time history of the top of the MPC is surveyed from the dynamic analysis of the VVM using the DBSM. The maximum velocity thus obtained is assumed to exist as the initial condition in the LS-DYNA simulation. This assumption is most conservative because it assumes that the cyclic motion transmitted by the earthquake does not detract from the canister's momentum before impact occurs (observations show that the canister slows down by the earthquake's cyclic energy input, thus significantly lessening the severity of the impact). In addition, the MPC guide is fixed at its base, which conservatively ignores the deformation of the divider shell and therefore maximizes the impact. The finite element model is shown in Figure 3.I.12. To implement the above model, the search for the maximum velocity in the dynamic solution yielded less than 24.7 in/sec as shown in Figure 3.I.27. Applying an initial velocity of 26.0 in/sec as the initial condition to the above model provided the strain field shown in Figure 3.I.13 for the tubular guide design. The impact between the MPC and the MPC top guides results in an MPC shell maximum plastic (true) strain of less than  $1.52 \times 10^{-2}$  in/in for the tubular guide design and  $3.1 \times 10^{-2}$  in/in for the optional plate guide design (see Calculation 11 of [3.I.27]), respectively, which are only a small fraction of the acceptable value (0.1) per [3.I.31]. Therefore the integrity of the confinement boundary is assured.

### 3.I.4.7.3 Strength Qualification of the ISFSI Structure

Under the Design Basis Earthquake (Figure 2.I.4), the loads exerted on the Support Foundation Pad and the Top Surface Pad (as illustrated in Figure 3.I.4) are obtained from the LS-DYNA SSI simulations listed in Table 3.I.6. Table 3.I.7 lists the peak ISFSI interface loads obtained from



various LS-DYNA runs listed in Table 3.I.6. In order to incorporate an additional margin of safety in the ISFSI structural analysis, the structural evaluation of ISFSI components uses input loads (see Table 3.I.8) that bound the peak ISFSI interface loads from the LS-DYNA SSI simulations in Table 3.I.7. The use of the bounding loads is in keeping with a similarly bounding value of settlement specified for the strength analysis of the SFP and the TSP (see Table 2.I.2).

The SFP and TSP shall meet the minimum structural requirements set down in Table 2.I.2 and the licensing drawings. The SFP and TSP are required to satisfy ACI-318 (2005) strength limits under all applicable load combinations (Table 2.I.11).

Likewise, the retaining wall, if used, shall meet the minimum concrete and rebar requirements provided in Table 2.I.2 and the licensing drawings. The site specific design may utilize a thicker and more heavily reinforced wall, if necessary, at user's option.

Table 3.I.8 provides the loading data used in the strength analysis of the ISFSI structures. The following discrete analyses are required:

- (i) Compute the long-term settlement of the undergrade supporting the SFP assuming all VVM locations are loaded for the entire Design Life: Determine the "effective" elastic modulus of the subgrade under the SFP to simulate the effect of differential settlement in the structural analyses model. As discussed in Section 2.I.4, the long-term settlement of the undergrade from the loaded VVMs and the dead weight of the SFP is very small because the combined equivalent density of the loaded VVM's and the SFP is nearly equal to the density of the excavated subgrade.
- (ii) Compute the long-term settlement of the subgrade under the TSP/VIP relative to the SFP from subgrade weight in addition to the dead weight of the TSP and VIP. Determine the "effective" elastic modulus of the subgrade between the TSP/VIP and the SFP to simulate the effect of long term differential settlement in the structural analyses model. As discussed in Section 2.I.4, the long-term settlement of the well conditioned subgrade under the TSP is appreciably small because of the small long-term loadings acting on the TSP.
- (iii) Prepare a finite element model of the pads in ANSYS and determine the stress field under the factored Dead and Live loads with the settlement based "degraded" elastic moduli for the soil regions directly beneath the TSP and SFP (Spaces A and C in Figure 2.I.5). For the lateral subgrade adjacent to the HI-STORM 100U ISFSI (Spaces B and D in Figure 2.I.5), the dynamic (strain compatible) elastic moduli from Table 3.I.4 are conservatively used as input to maximize the flexural moments due to differential settlement.
- (iv) Compute the stress field in the pads under factored seismic loads using dynamic elastic modulus corresponding to the minimum shear wave velocity of the subgrade specified in Table 2.I.2.
- (v) Use the bounding loads listed in Table 3.I.8 to compute the stress fields in the pads (SFP and TSP) from the DBE.
- (vi) Combine the factored loads and determine the total stress resultants. Compare with the respective section strengths to establish the factors of safety for the SFP and TSP.

- (vii) Compute the bearing stress (or load) on the subgrade under the TSP using the combined factored loads from the transporter and the TSP/VIP and compare with the corresponding allowable limit to establish the safety factor for the subgrade under the TSP.

A comprehensive summary of the analyses and the associated margins of safety are discussed below:

The structural evaluation of the HI-STORM 100U ISFSI is performed using the commercial computer code ANSYS [3.I.33]. The constituents of the ISFSI namely the Support Foundation Pad (SFP), the subgrade under the SFP (the undergrade), the Top Surface Pad (TSP) and the subgrade lateral to the CEC under the TSP are all modeled using linear elastic SOLID45 elements. The VVM interface pad (VIP), which carries no load except for its self-weight, is conservatively omitted in the model. The lateral subgrade adjacent to the HI-STORM 100U ISFSI (Spaces B and D in Figure 2.I.5) is included in the FE model and extends laterally a distance that exceeds the overall depth of the FE model considered for structural analysis. The element mesh is intentionally kept fine in the areas of load application on the SFP and the TSP. For convenience of load application, the footprint of the CEC base on the SFP is carefully articulated in the finite element model. The substrate under the SFP is terminated at approximately 101 ft below the TSP, which is consistent with the Design Basis Seismic Model discussed in Subsection 3.I.4.7.1. The “base” model (Simulation Model I) considers that all the storage locations in ISFSI are populated and experience identical peak vertical seismic loading from Table 3.I.8, which bounds the peak result obtained from the LS-DYNA SSI solution as discussed previously. Because of the symmetric geometry and loading, a quarter symmetric finite element model is sufficient to represent the fully loaded ISFSI. Figure 3.I.14 shows the finite element model of HI-STORM 100U ISFSI. For a non-symmetric model as in case of Simulation Model II, a full FE model of the HI-STORM 100U ISFSI as shown in Figure 3.I.15 is used. The “degraded” elastic moduli of the subgrade under the SFP and the subgrade between the TSP and SFP is appropriately computed to account for the long-term differential settlement effects as described in Subsection 2.I.4. The long-term settlement and the “effective” subgrade elastic moduli are derived using the governing soil characteristics following guidelines from [2.I.6]. Table 3.I.5 lists the bounding subgrade characteristics and the concomitant elastic moduli effective under dynamic loading. To address different loading patterns on the ISFSI and for completeness, additional partially loaded ISFSI configurations are considered in the evaluations. The partial loaded configurations include a two row loaded ISFSI (two rows of VVM locations adjacent to the symmetry line are loaded), a single row loaded ISFSI (the middle row of VVM locations is loaded) and a single VVM loaded ISFSI (a single VVM location centered near the periphery of the ISFSI is loaded). Figures 3.I.19, 3.I.20 and 3.I.21 illustrate the partial loading configurations for the ISFSI. These loading configurations are hereinafter referred to as Simulation Models II, III, and IV, respectively. Note that the terms “Loading Configuration” and “Simulation Model” are intermittently referred throughout the calculation. For Simulation Models I through IV, the optional retaining walls are not included in the finite element model.

The effects of the retaining wall are evaluated in a fifth Simulation Model (Simulation Model V), which is shown in Figure 3.I.16. In this model, the lateral subgrade surrounding the retaining walls is completely removed to bound any future excavation activities associated with the construction

of a new underground ISFSI, and for consistency with the LS-DYNA SSI simulation model (see Figure 3.I.3-C). For Simulation Model V, the middle row of VVM locations is loaded similar to Simulation Model III (see Figure 3.I.22) which excludes the transporter load as discussed previously.

To simulate the material continuity at the extreme boundary surfaces of the model, translations are constrained at the lateral face of the subgrade. The extreme bottom surface of the model is fixed representing the bedrock (or competent soil) elevation.

The following individual load steps are considered in the analysis:

1. Bounding load transmitted by the VVM, as determined from the LS-DYNA SSI analysis and summarized in Table 3.I.8, is applied as an effective pressure on the footprints of the CEC base at all VVM locations.
2. The load from the transporter is applied as a normal pressure (see Figures 3.I.17, 3.I.19, 3.I.20 and 3.I.21) over the transporter load patch on the TSP. For Simulation Model V, the transporter load is not applied to the TSP since no VVM loading/unloading operations are expected to occur during excavation activities associated with the construction on a new underground ISFSI.
3. The dead weight from VIP is applied as normal pressures on the substrate elements directly beneath the VIP.
4. In-plane tensile loads on the SFP and TSP from the retaining wall are applied as lateral pressures on the SFP and TSP boundaries for Simulation Model V.
5. To simulate the self weight of the modeled portion of the ISFSI, a 1g gravity load is applied. The densities of the various constituents are appropriately input in the model to accurately reflect the individual component weights.

It must be noted that the structural analysis of the ISFSI conservatively considers the peak dynamic loads from the LS-DYNA SSI analysis. However, it shall be permitted to use equivalent static loads obtained by removing high frequency components that would not contribute to the structural response using appropriate filters.

Since the peak loads from the LS-DYNA SSI analyses are substantially larger in comparison to the dead and live loads, the load combination LC-3 from Table 2.I.11 governs for the ISFSI structural evaluation. However, the analyses are carried out for load combinations LC-2 and LC-3, and the corresponding results substantiate that the load combination LC-3 is governing.

Figures 3.I.18a through 3.I.22c depict the maximum in-plane stresses in the ISFSI concrete structures (viz. SFP and TSP) for the governing load combination LC-3 for all the ISFSI configurations analyzed. The in-plane axial and bending stress on the SFP and the TSP elements are post-processed to compute the equivalent moments. The induced moments are compared to the respective moment capacities to determine the corresponding factor of safety. Table 3.I.10 summarizes the results for the SFP and the TSP respectively for all ISFSI configurations analyzed.

The minimum flexure safety factor is produced by Simulation Model IV, and it is associated with the TSP. In the Simulation Models I, II, III and IV, the peak load from the LS-DYNA SSI analysis acting on one transporter track (bearing on the TSP) is conservatively applied as a static load on both transporter tracks simultaneously, thereby significantly overestimating the load on the TSP. As mentioned previously, the peak dynamic loads obtained from the LS-DYNA SSI analyses from a DBE event are of impulsive nature. Use of the peak loads for static structural evaluations of the ISFSI is evidently conservative. Furthermore, no credit is taken for the Dynamic Increase Factor of 25% for flexure and 10% for shear permitted by [3.I.32] in the strength qualification of reinforced concrete.

The Table 3.I.11 summarizes the punching shear safety factor for the SFP and TSP. The minimum punching shear safety factor is associated with the TSP under the transporter seismic load, and it is well above 1.0.

The peak transporter load on the TSP from the LS-DYNA SSI analyses plus the load from the TSP are used to compute the maximum bearing stress in the substrate surface under the TSP. According to ACI-360 [2.I.8], the bearing stress can be calculated by uniformly distributing the load over the entire bearing area of the pad. For conservatism, the bearing stress calculation for the 100U subgrade is performed using a bearing area significantly less than that of the smallest TSP (i.e., the TSP of one-VVM ISFSI). The maximum bearing stress in the sub-grade (Table 3.I.12) is smaller than the presumptive bearing stress limit, resulting in minimum safety factor above 2.0 imposed by the ACI code [2.I.8].

The evaluation of the CEC shell under the loads from the transporter load in addition to the subgrade overburden is presented in Subsection 3.I.4.4.

Finally, the structural integrity of the retaining wall is evaluated for the Design Basis Earthquake loading condition; the structural demand to the wall under normal operational conditions is small and therefore not structurally governing. Since the retaining wall is connected with the TSP and SFP through a shear key at the top and dowels at the bottom (see licensing drawing in Section 1.I.5), it can be treated as a simply supported plate (along its top and bottom edges) in the structural analysis. Therefore, the wall essentially experiences bending stress in the DBE event due to lateral soil pressure. The maximum bending moment of the retaining wall, which can be determined based on Figure 3.5-1 of Reference [3.I.28] or based on the retaining wall stress results obtained from the LS-DYNA SSI analysis for Case 3 in Table 3.I.6 (both approaches yield approximately the same result), is shown in Table 3.I.10 to be well below the bending capacity of the wall. The shear connections at the top and bottom of the retaining wall have also been evaluated for the loads induced during a Design Basis Earthquake. The results of the strength evaluation are provided in Table 3.I.13.

### 3.I.4.8 Tornado Missile Evaluation

#### 3.I.4.8.1 HI-STORM 100U Lid Integrity Evaluation for Tornado Missile Strike (Load Case 03 in Table 2.I.5)

Design basis tornado missiles are specified in Table 2.2.5. The Closure Lid is the only above ground component of the VVM; therefore, missile impact analyses focus on this component. Large and intermediate tornado missiles are assumed to strike the center top surface of the lid at the design basis speed (see Table 2.2.5). For both missile analyses, a finite element model of the Closure Lid is employed (using dimensions from licensing drawings and applicable material properties), and includes contact between concrete and steel (see Figure 3.I.1). LS-DYNA is used to perform dynamic simulations of the impacts to demonstrate that neither missile completely penetrates the composite structure. The ANSYS model shown in Figure 3.I.1 is simplified to develop an input file for the LS-DYNA simulation. Elastic-Plastic Material 24 is used for the steel and Material 72 is used for the concrete. For a conservative result, engineering stress relations for the lid steel work are used with an assumed ultimate strain of 21% (per ASME Code, Sec. II, Part A). As LS-DYNA expects that true stress-strain data is input, the use of true stress-strain data, to obtain a more realistic result, is permitted (if appropriate justification is provided for the true stress-strain relation). The solution obtained using engineering stress strain data is clearly conservative in that material failure is set at the engineering ultimate strain limit rather than reflecting the true strain at failure, which will be considerably larger. A strain rate effect is incorporated by increasing the yield and ultimate strengths by a maximum of 50% (depending on the rate) as suggested by data for SA-36 steel [3.I.19]. This is the same strain rate increase used in the evaluations to assess the performance of the aboveground HI-STORM when impacted by a jet fighter aircraft [3.I.16]. A time history normal pressure loading is applied over the metal annular region around the outlet opening to simulate the large missile, and the global deformation damage to the lid is assessed. The formula from “Topical Report – Design of Structures for Missile Impact”, BC-TOP-9A, Rev. 2, 9/74 [3.I.17] is used to establish appropriate pressure-time data. For the speed and mass associated with the large missile, the impact force-time curve has the form

$$F(t) = 0.625 \text{ sec/ft} \times 184.8 \text{ ft/sec} \times 4000 \text{ lb} \times \sin(20t) = 462,000 \text{ lb} \times \sin(20t) \text{ for } t < 0.0785 \text{ sec.} \\ = 0 \text{ for } t \geq 0.0785 \text{ sec.}$$

This representation of the large missile impact load is appropriate as recent full-scale impact testing of a modern passenger vehicle demonstrates. Figure 3.I.6 shows the force-time history from the full-scale test of a full-size Ford passenger vehicle [3.I.18]. The test was performed at an impact speed of 35 mph and the vehicle had approximately the same weight as the design basis large deformable missile. Since the force is directly proportional to the pre-impact momentum, an estimate of the peak force at 126 mph for the vehicle is obtained by a simple ratio of the impact velocities and missile mass. Estimating the peak value from the plot produces a resulting peak force of 496,000 lb, which is the same order of magnitude as the peak value predicted from the Bechtel Topical Report, although the shape and duration of the curve is different. The results from the analysis using the load-time function from the Bechtel formula show no significant lid damage from the large missile strike on the lid because of the concrete backing. Inspection of the result concludes that the deformed shape after the event does not preclude lid removal, the lid remains in-place, and the MPC has not been impacted. The maximum lid vertical deflection during the strike is less than 0.1 inch and there are a few local regions of permanent effective plastic strain. The details of this calculation are found in [3.I.27]. The large missile impact is not the bounding strike because of the large area of impact and significant energy loss that occurs when the vehicle

is crushed upon impact; the rigid, intermediate missile imparts more local and global damage to the Closure Lid.

The impact of the intermediate missile is conservatively simulated as a rigid 8" diameter cylindrical steel bar weighing 275 lb. (per Table 2.2.5), traveling at 126 mph and striking the Closure Lid at the most vulnerable location, which is through the top vent opening. The strike can be either at the center of the inner shield dome or slightly off-center so as to miss the central steel connecting bar. In order to strike the MPC top lid, the intermediate missile must penetrate the steel weldment and encased concrete (see licensing drawings in Section 1.I.5). Figures 3.I.7 and 3.I.8 show the intermediate impact scenarios considered. Figures 3.I.9 and 3.I.10 show the lid state at the time of maximum bottom plate vertical displacement. For both cases, no dislodgement of the lid is indicated and plastic strains occur only in the immediate vicinity of the strike. A summary of results that bound the computed results for the two intermediate missile strikes is presented in Table 3.I.9.

Next, consider that the intermediate or large missile is traveling horizontally and strikes the side of the Closure Lid. A large missile strike at this location with a horizontal orientation is most likely not credible because of the low profile of the lid. The large missile would rotate as it broke up, resulting only in a glancing blow to the lid. However, an evaluation of the Closure Lid flange ring in either missile side strike is needed to ensure that the Closure Lid will not be driven sideways under the impact and separate from the CEC. A key structural element is the weld connecting the Closure Lid restraint ring to the Closure Lid. The capacity of the welds in the load path that resist the lateral impact load is:

Closure Lid Weld Capacity = 8,381,000 lb.

This capacity is computed assuming a limiting weld stress of 60% of the ultimate tensile strength of the base material. In any of the evaluated missile strikes from above, the peak impact load (filtered at 350 Hz (see similar filtering in the HI-STAR 100 transport license)) does not exceed 1,200,000 lb. Interface loads from top impacts are expected to bound impact loads from side impacts because of the geometry involved; therefore, the safety factor on the CEC Container Shell flange ring, acting to hold the lid in-place, is:

SF (flange ring) = Closure Lid Weld Capacity/ Filtered Peak Impact Load > 6.9

Finally, a small missile entering the outlet duct will not damage the MPC because there is no direct line-of-sight to the MPC, and even if it arrives at the MPC, it will have undergone multiple impacts with the duct walls, and can only impact the thick MPC lid. Therefore, MPC damage from the small missile is not credible.

An assessment of all simulation results concludes that the postulated missile strikes will not preclude MPC retrievability, will not cause loss of confinement, and will not affect criticality. In no scenario, does the lid become dislodged.

### 3.I.4.8.2 Tornado Missile Protection during Construction

The number of VVMs in a HI-STORM 100U ISFSI may vary depending on a user's need. While there is a minimum spacing (pitch) requirement (see licensing drawing in Subsection 1.I.5), there is no limitation on the maximum spacing. Furthermore, a module array may have a non-rectangular external contour such as shown in the licensing drawing with a trapezoidal contour. Finally, an ISFSI may be constructed in multiple campaigns to allow the user to align the VVM cavity construction schedule with the plant's fuel storage needs. Any ISFSI constructed in one campaign shall have the following mandatory perimeter protection features:

- i. The Radiation Protection Space (RPS) shall extend to an appropriate distance beyond the outer surface of the CEC shell (see licensing drawing in Subsection 1.I.5). Calculations have been performed [3.I.27] that confirm that a 10' distance beyond the outer surface of the CEC shell is sufficient to prevent the 8" diameter rigid cylindrical missile (defined in Table 2.I.1 and is the most penetrating of the missile types considered in this FSAR) from contacting the CEC shell should this missile strike the exposed cut from the adjacent construction. The penetration analysis conservatively assumed a subgrade with minimum resistance to missile penetration and the formulation described in [3.I.30].
- ii. Unless a retaining wall (see licensing drawing) has been built to confine and retain the subgrade at the boundary of the RPS (or beyond) in the particular direction of excavation, an Excavation Exclusion Zone (EEZ) shall be defined within which any excavation activity during an operating ISFSI is prohibited (see Subsection 2.I.2). The retaining wall is the EEZ boundary if the retaining wall is located at or beyond the RPS.

### 3.I.4.9 HI-STORM 100U VVM Service Life

The VVM is engineered for 40 years of design life, while satisfying the conservative design requirements defined in Supplement 2.I. For information supporting the 40 year design life addressing chemical and galvanic reactions as well as other potentially degrading factors see Subsection 3.I.4.1. Requirements for periodic inspection and maintenance of the HI-STORM 100U VVM throughout the 40-year design life are defined in Supplement 9.I. The VVM is designed, fabricated, and inspected under the comprehensive Quality Assurance Program discussed in Chapter 13.

### 3.I.5 FUEL RODS

No new analysis of fuel rods is required for storage of an MPC in a HI-STORM 100U VVM.

### 3.I.6 SUPPLEMENTAL DATA

#### 3.I.6.1 Additional Codes and Standards Referenced in HI-STORM 100 System Design and Fabrication

No additional Codes and Standards are added for the HI-STORM 100U system.

### 3.I.6.2 Computer Programs

ANSYS 5.7, 7.0, 9.0, 11.0, and LS-DYNA (previously known as DYNA3D) [3.I.2] are used for the finite element analyses prepared by Holtec and summarized in this supplement.

#### ANSYS

ANSYS is a public domain code, well benchmarked code, which utilizes the finite element method for structural analyses. It can simulate both linear and non-linear material and geometric behavior. It includes contact algorithms to simulate surfaces making and breaking contact, and can be used for both static and dynamic simulations. ANSYS has been independently QA validated at Holtec International. In this FSAR submittal, ANSYS is used within [3.1.27] and the element size used in the application follows the recommendation of the code developers.

#### LS-DYNA

LS-DYNA is a nonlinear, explicit, three-dimensional finite element code for solid and structural mechanics. It was originally developed at Lawrence Livermore Laboratories and is ideally suited for study of short-time duration, highly nonlinear impact problems in solid mechanics. LS-DYNA is commercially available and has been independently validated at Holtec following Holtec's QA procedures for commercial computer codes. This code has been used to analyze the Non-Mechanistic Storage tip-over for the HI-STORM 100 Part 72 general license. In this supplement, the code is used to establish the performance of the HI-STORM 100U under a design basis seismic event, and to evaluate the response to a design basis missile.

LS-DYNA is currently supported and distributed by Livermore Software. Each update is independently subject to QA validation at Holtec.

### 3.I.6.3 Appendices Included in Supplement 3.I

None.

### 3.I.6.4 Calculation Packages

A calculation package [3.1.27] containing the structural calculations supporting Supplement 3.I has been prepared and archived according to Holtec International's Quality Assurance program (see Chapter 13), and submitted with this application. A second calculation report [3.I.14], documenting the SASSI analyses, has been prepared by a Holtec subcontractor under the subcontractor's QA program.

### 3.I.7 COMPLIANCE WITH NUREG-1536

The material in this supplement for the HI-STORM 100U system provides the same information



as previously provided for the aboveground HI-STORM 100 systems. Therefore, to the extent applicable, the information provided is in compliance with NUREG-1536.

### 3.I.8 REFERENCES

The references in Section 3.8 apply to the VVM to the extent that they are appropriate for use with an underground system. The additional references below are specific to Supplement 3.I.

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- [3.I.3] USNRC Interim Staff Guidance (ISG-15), “Materials Evaluation”, Revision 0, January 2001.
- [3.I.4] ANSI/AWWA C105/A21.5-99, “American National Standard (ANSI) for Polyethylene Encasement for Ductile-Iron Pipe Systems”.
- [3.I.5] M. B. Bruce and M. V. Davis, “Radiation Effects on Organic Materials in Nuclear Plants”, Final Report, 1981. (Prepared by Georgia Institute of Technology for EPRI)
- [3.I.6] ANSI D 4082-02, “American National Standard (ANSI) Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Light Water Nuclear Power Plants”.
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- [3.I.8] ANSI D 3843-00, “American National Standard (ANSI) Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities”.
- [3.I.9] ANSI C 210-03, “American National Standard (ANSI) Standard Practice for Liquid-Epoxy Coating Systems for the Interior and Exterior of Steel Water Pipelines”.
- [3.I.10] Keeler & Long Inc. Product Data Sheet for Kolor-Proxy™ Primer KL3200 Series, Product Code KL3200.
- [3.I.11] Samuel A. Bradford, “Practical Handbook of Corrosion Control in Soils”, ASM International and CASTI Publishing Inc., 2004.
- [3.I.12] L. M. Poukhonto, “Durability of Concrete Structures and Constructions – Silos,

Bunkers, Reservoirs, Water Towers, Retaining Walls”, A. A. Balkema Publishers, 2003.

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- [3.I.15] S. Stojko, Application of DYNA3D to Non-Linear Soil Structure Interaction (SSI) Analysis of Retaining Wall Structures, International LS-DYNA3D Conference, March 1993.
- [3.I.16] ASLB Hearings, Private Fuel Storage, LLC, Docket # 72-22-ISFSI, ASLBP 97-732-02-ISFSI, February 2005.
- [3.I.17] Topical Report – Design of Structures for Missile Impact”, BC-TOP-9A, Rev. 2, Bechtel Corporation, 9/74
- [3.I.18] SAE Technical Paper 2000-01-0627, Development and Validation of High Fidelity Vehicle Crash Simulation Models, S.W. Kirkpatrick, Applied Research Associates, Inc.
- [3.I.19] H. Boyer, Atlas of Stress Strain Curves, ASM International, 1987, p.189.
- [3.I.20] Thermal Ceramics Inc., Product Data Sheet for Blanket Products (Kaowool® Blanket).
- [3.I.21] NACE Standard RP0285-2002 “Corrosion Control of Underground Storage Tank Systems by Cathodic Protection”, NACE International.
- [3.I.22] API RP1632, “Cathodic Protection of Underground Petroleum Storage Tanks and Piping Systems”, American Petroleum Institute.
- [3.I.23] NACE RP0169-96, “Control of External Corrosion on Underground or Submerged Piping Systems”, NACE International.
- [3.I.24] 49CFR Part 192 Subpart I “Requirements for Corrosion Control, Title 49 of the Code of Federal Regulations, Oct, 1 2004 Edition, Office of the Federal Register, Washington, D.C.
- [3.I.25] ACI 544.3R-93 (or latest), Guide for Specifying, Proportioning, Mixing, Placing, and Finishing Steel Fiber Reinforced Concrete.

- [3.I.26] ASTM C1116-03 (or latest) Standard Specification for Fiber-Reinforced Concrete and Shotcrete.
- [3.I.27] HI-2053389, Calculation Package Supporting Structural Evaluation of HI-STORM 100U, Revision 9, September 2010 (Holtec Proprietary).
- [3.I.28] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 2000.
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- [3.I.33] ANSYS 11.0, ANSYS Inc., 2007 and ANSYS 13.0, Ansys Inc., 2010, Copyright SAS IP, Inc.

**TABLE 3.I.1****HI-STORM 100U BOUNDING WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>MPCs</b>	
<ul style="list-style-type: none"> <li>• Without SNF</li> <li>• Fully loaded with SNF and Fuel Spacers</li> </ul>	See Table 3.2.1 90,000
<b>HI-STORM 100U VVM</b>	
<ul style="list-style-type: none"> <li>• Closure Lid (with shielding concrete)</li> <li>• CEC (empty without Closure Lid)</li> <li>• Maximum Loaded Weight (with bounding MPC)</li> </ul>	24,000 33,000 147,000
<b>Loaded Transporter (Typical)</b>	
<ul style="list-style-type: none"> <li>• Carrying a loaded HI-TRAC</li> <li>• Empty</li> <li>• Length &amp; width of each load patch (2 load patches per transporter)</li> <li>• Computed average normal pressure on two load patches</li> </ul>	400,000 160,000 197.1875 inch by 29.5 inch 34.4 psi
Loaded HI-TRAC and Mating Device	275,000
Note 1: CEC and Closure Lid include an overage up to 5%	
Note 2: Transporter weight is based on representative units used in the industry.	

**TABLE 3.I.2****CENTER OF GRAVITY DATA FOR THE HI-STORM 100U SYSTEM**

<b>Component</b>	<b>Height of CG Above Datum (in)</b>
MPC	See Table 3.2.3
HI-STORM 100U VVM CEC (empty without Closure Lid)	108.7
HI-STORM 100U VVM Closure Lid	20.26
Note: Datum for CEC is at the top surface of the foundation; datum for Closure Lid is at bottom surface of baseplate of lid.	

**TABLE 3.I.3 (a)\***  
**RELEVANT MATERIAL PROPERTIES FOR THE HI-STORM 100U**  
**Yield, Ultimate, Linear Thermal Expansion, Young's Modulus**

Temp. (Deg. F)	SA516 and SA515, Grade 70			
	S <sub>y</sub>	S <sub>u</sub>	α	E
-40	38.0	70.0	---	29.95
100	38.0	70.0	5.53 (5.73)	29.34
150	36.3	70.0	5.71 (5.91)	29.1
200	34.6	70.0	5.89 (6.09)	28.8
250	34.15	70.0	6.09 (6.27)	28.6
300	33.7	70.0	6.26 (6.43)	28.3
350	33.15	70.0	6.43 (6.59)	28.0
400	32.6	70.0	6.61 (6.74)	27.7
450	31.65	70.0	6.77 (6.89)	27.5
500	30.7	70.0	6.91 (7.06)	27.3
550	29.4	70.0	7.06 (7.18)	27.0
600	28.1	70.0	7.17 (7.28)	26.7
650	27.6	70.0	7.30 (7.40)	26.1
700	27.4	70.0	7.41 (7.51)	25.5
750	26.5	69.3	7.50 (7.61)	24.85
800	25.3	64.3	7.59 (7.71)	24.2
* Footnotes in corresponding table in Section 3.3 apply to the values in parentheses.				

**TABLE 3.I.3 (b)**  
**DESIGN AND LEVEL A: ALLOWABLE STRESS FROM ASME NF**  
**Material : SA516 Grade 70, SA515 Grade 70**  
**Service Conditions: Design and Level A Stress**  
**Item: Stress**

Temp. (Deg. F)	Classification and Value (ksi)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 650	17.5	17.5	26.3
700	16.6	16.6	24.9
750	14.8	14.8	22.2
800	12.0	12.0	18.0

**TABLE 3.I.3 (c)**  
**LEVEL D: STRESS INTENSITY**

**Code: ASME NF**  
**Material: SA516, Grade 70**  
**Service Conditions: Level D**  
**Item: Stress Intensity**

Temp. (Deg. F)	Classification and Value (ksi)		
	S <sub>m</sub>	P <sub>m</sub>	P <sub>m</sub> + P <sub>b</sub>
-20 to 100	23.3	45.6	68.4
200	23.1	41.5	62.3
300	22.5	40.4	60.6
400	21.7	39.1	58.7
500	20.5	36.8	55.3
600	18.7	33.7	50.6
650	18.4	33.1	49.7
700	18.3	32.9	49.3

**TABLE 3.I.3 (d)**

**Code:** ASME NB  
**Material:** Alloy X  
**Service Conditions:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>P<sub>m</sub></b>	<b>P<sub>L</sub></b>	<b>P<sub>L</sub> + P<sub>b</sub></b>
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7



**TABLE 3.I.4**  
**REFERENCE AND DERIVED PROPERTIES OF ISFSI REINFORCED CONCRETE,**  
**SUBGRADE, AND UNDERGRADE**

<b>Property</b>	<b>Value</b>
Concrete Compressive Strength (psi)	4,500
Concrete Rupture Strength (psi)	335.4
Allowable Bearing Stress (psi)	4,972.5*
Mean Coefficient of Thermal Expansion (in/in-deg. F)	5.5E-06
Modulus of Elasticity (psi)	$57,000 \times (\text{Concrete Compressive strength (in psi)})^{1/2}$
Subgrade Yield Stress (psi)	$25^{\text{‡}}$
Subgrade Strain Compatible Modulus of Elasticity (ksi) (see Figure 2.I.5)	Space A: 18.8 Space B: 14.0 Spaces C and D: 17.7

\* Per ACI-318 (2005), Sec. 10.17.1 and Sec. 9.3.2.4. Since the ISFSI concrete is always confined, the allowable value is doubled.

‡ Only applied to Space A.

**TABLE 3.I.5**  
**SOIL PROPERTIES, COMPUTED SETTLEMENT, AND CORRESPONDING ELASTIC**  
**MODULI FOR THE SUBGRADE**

Item		Value
1.	Characteristics for Subgrade Below SFP: Water Content 'w <sub>n</sub> ' Soil Parameter 'a' Soil Parameter 'b' Poisson's Ratio	14% 0.18 0.13 0.45
2.	Derived Properties for the Subgrade Below SFP (Note 1): Computed Long-Term Settlement (in) (Note 2) Computed Elastic Modulus (psi)	0.189 5,377
3.	Values used in the Structural Analyses Model for Subgrade Below SFP: Limiting Long-Term Settlement (in) Corresponding Elastic Modulus (psi)	From Table 2.I.2 5,081
4.	Soil Characteristics for Subgrade Above SFP: Water Content 'w <sub>n</sub> ' Soil Parameter 'a' Soil Parameter 'b' Poisson's Ratio	14% 0.09 0.13 0.45
5.	Derived Properties for the Subgrade Above SFP (Note 3): Computed Long-Term Settlement (in) (Note 2) Computed Elastic Modulus (psi)	0.39 5,073
6.	Values used in the Structural Analyses Model for Subgrade Above SFP: Long-Term Settlement (in) Corresponding Elastic Modulus (psi)	From Table 2.I.2 4,946
<p>Note 1: The substrate characteristics are obtained using the density data from Table 2-3 and Table 5-1 of reference [2.I.7]. The soil compaction index 'Cc' is a direct function of soil parameters w<sub>n</sub>, a, and b per [2.I.7]. The long-term settlement and the elastic modulus are derived using the relationships in [2.I.6].</p> <p>Note 2: See Table 2.I.2 for the values of settlement (greater than those computed here for conservatism) used as the Design Basis data for qualification of the ISFSI structures.</p> <p>Note 3: The Design Basis settlement has been set at a higher value than that computed for the TSP and SFP to allow for the variation in the soil parameters at a host site.</p>		

**TABLE 3.I.6**  
**MENU OF LS-DYNA RUNS (SSI ANALYSES)**

<b>No.</b>	<b>Case</b>	<b>Comment</b>
1.	VVM array model (5x5 array) with 100% concrete modulus for the SFP	To obtain interface load for the ISFSI structures
2.	VVM array model (5x5 array) with 50% concrete modulus for the SFP	To obtain interface load for the ISFSI structures
3.	VVM array model (5x5 array) for the optional ISFSI design with retaining walls	To obtain interface load for the ISFSI structures
4.	Design Basis Single VVM seismic model	To qualify VVM components.

**TABLE 3.I.7**  
**ISFSI INTERFACE LOADS OBTAINED FROM LS-DYNA SSI SIMULATIONS**

Interface Load	Case 1	Case 2	Case 3	Case 4
CEC to SFP Impact Load, lb	$6.499 \times 10^5$	$6.267 \times 10^5$	$6.590 \times 10^5$	$6.433 \times 10^5$
Transporter to TSP Contact Load per Track, lb	$1.078 \times 10^6$	$1.109 \times 10^6$	N/A	$1.148 \times 10^6$
Soil Compressive Load on the Retaining Wall, lb	N/A	N/A	$3.290 \times 10^6$	N/A
In-Plane Tensile Load on TSP from Retaining Wall, lb	N/A	N/A	$9.672 \times 10^5$	N/A
In-Plane Tensile Load on SFP from Retaining Wall, lb	N/A	N/A	$2.287 \times 10^6$	N/A

**TABLE 3.I.8**  
**LOADS APPLIED IN THE ISFSI STRUCTURAL EVALUATION†**

Load on ISFSI	Simulation Models I, II and III	Simulation Model IV
Load on SFP at each VVM location ‡, lbf	660,000	660,000
Total Load on TSP due to Transporter ‡, lbf	$5.6 \times 400000 = 2.24 \times 10^6$	N/A
In-Plane Tensile Load on TSP Extreme Face, lbf	N/A	$9.68 \times 10^5$
In-Plane Tensile Load on SFP Extreme Face, lbf	N/A	$2.29 \times 10^6$
<p>Notes:</p> <p>† For conservatism, the loads used for ISFSI structural evaluation bound the peak loads obtained from SSI simulations (see Table 3.I.7).</p> <p>‡ The listed load is a sum of dead and seismic components. These loads are appropriately divided as dead and seismic in ANSYS prior to applying the appropriate load factors and combinations per Table 2.I.11.</p>		

**TABLE 3.I.9\***

<b>RESULTS FROM TORNADO MISSILE ANALYSIS (LOAD CASE 03 OF TABLE 2.I.5)</b>			
<b>Item</b>	<b>Bounding Value, inch</b>	<b>Allowable Value, inch</b>	<b>Safety Factor</b>
Maximum Vertical Displacement of lid (inch) (inclined impact)	< 3	12**	> 4
Perforation of Inner Shield Dome Steel	Yes (see Fig. 3.I.7)	N/A	N/A
Maximum Peak Impact Force (kips)	< 1,000	1,849	> <b>1</b>
* Details of the calculations can be found in [3.I.27].			
** This is the minimum distance between the Closure Lid bottom plate and the top lid of the MPC.			

**TABLE 3.I.10**  
**MOMENT RESULTS AND CORRESPONDING MINIMUM SAFETY FACTORS FOR THE ISFSI STRUCTURES**

<b>Support Foundation Pad (SFP) ‡</b>				
<b>ISFSI Load Configuration</b>	<b>Moment Induced (lbf-in/in)</b>	<b>Axial Force (lbf/in)</b>	<b>Corresponding Moment Capacity (lbf-in/in)</b>	<b>Minimum Safety Factor (Limiting Element No.)</b>
Simulation Model I	192,000	-1529.8	263,270	1.371 (1766)
Simulation Model II	192,990	-2965.3	276,665	1.434 (3329)
Simulation Model III	230,920	-2541.3	272,710	1.181 (1766)
Simulation Model IV	168,140	-2158.8	269,141	1.601 (1520)
Simulation Model V	170,170	5,140	180,583	1.061 (1766)
<b>Top Surface Pad (TSP) ‡</b>				
Simulation Model I	318,070	-15713	347,755	1.093 (67115)
Simulation Model II	186,960	-4707.9	258,978	1.385 (147092)
Simulation Model III	293,240	-11386	312,850	1.067 (67115)
Simulation Model IV	277,670	-8660.1	290,860	1.048 (66996)
Simulation Model V	82,161	-966	228,791	2.785 (97531)
<b>Retaining Wall †</b>				
Fully Loaded (Case 3 of Table 3.I.6)	80,000	N/A	175,000	2.19

**TABLE 3.I.10 (continued)**  
**MOMENT RESULTS AND CORRESPONDING MINIMUM SAFETY FACTORS FOR THE ISFSI STRUCTURES**

‡ The moment capacities for the SFP and TSP are calculated using axial-force-moment interaction diagram corresponding to the axial force and moment induced in the limiting element. An example showing the moment and axial force calculation for the limiting element, using the in-plane stresses, is included under Figure 3.I.18.b. Figures 3.I.18a through 3.I.22c also capture the stress plots for the governing load combination (LC-3 in Table 2.I.11) for all the ISFSI loading configurations analyzed. These figures also identify the limiting element (i.e., maximum loaded element) that corresponds with the results presented above.

Note that the flexural safety factors calculated above are based on the maximum moment induced in a single element, which is very conservative. Averaging over a width of the loaded section would result in much higher safety factors.

† The moment capacity for the Retaining Wall is based on the pure bending.



**TABLE 3.I.11**  
**PUNCHING SHEAR SAFETY FACTORS FOR ISFSI STRUCTURES**

ISFSI Structure	Punching Safety Factor
SFP†	2.1
TSP	1.32
† Note that the punching shear calculation for the SFP is conservatively based on a bounding load of 950 kips.	

**TABLE 3.I.12  
PRESUMPTIVE SOIL BEARING**

Computed Bearing Stress (psi)	Allowable Bearing Stress (psi)	Safety Factor	Minimum Safety Factor Required per [2.I.8]
42.8	90	2.1	2.0

**TABLE 3.I.13**  
**RESULTS OF STRENGTH EVALUATION FOR RETAINING WALL SHEAR CONNECTIONS**

Component	Minimum Safety Factor
Top Shear Key	9.41
Bottom Dowels	1.56

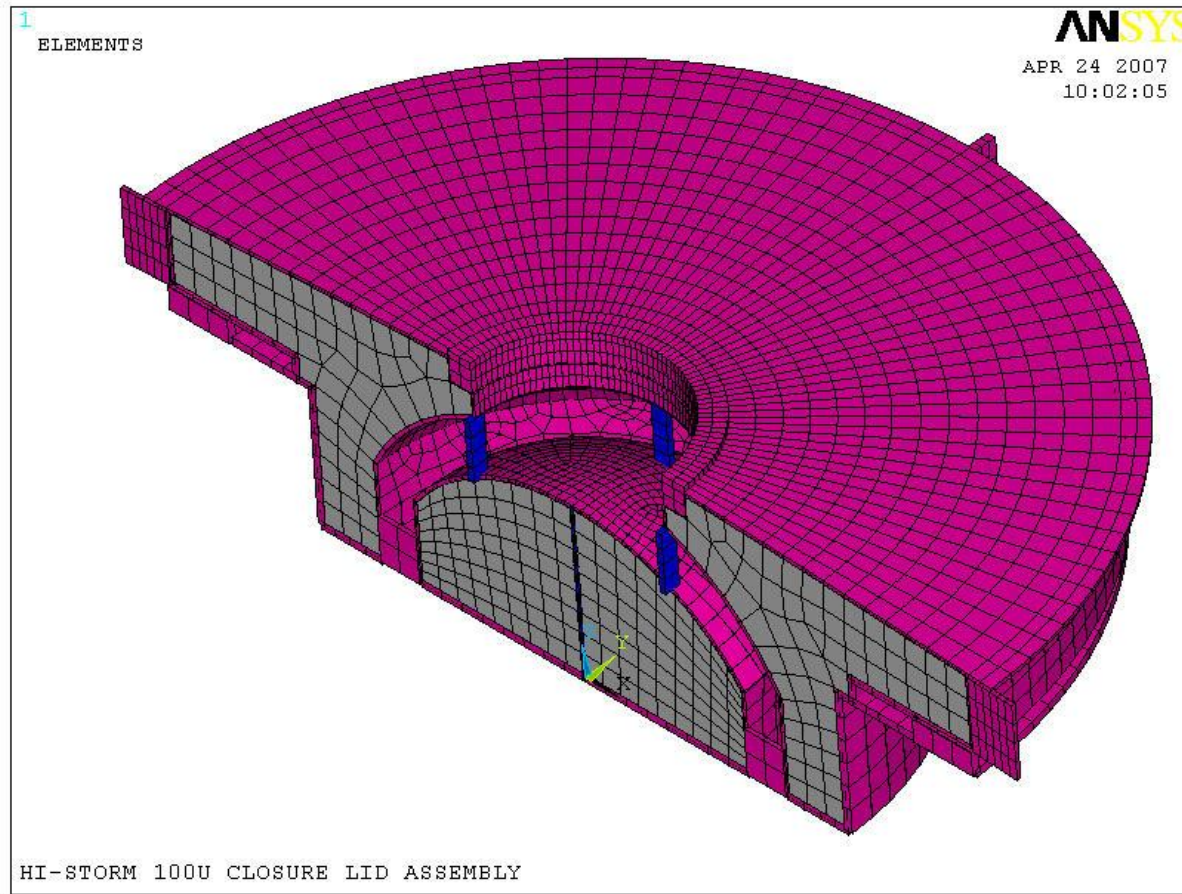


Figure 3.I.1; 3-D ANSYS/LS DYNA Finite Element Model of Closure Lid (Current Configuration)

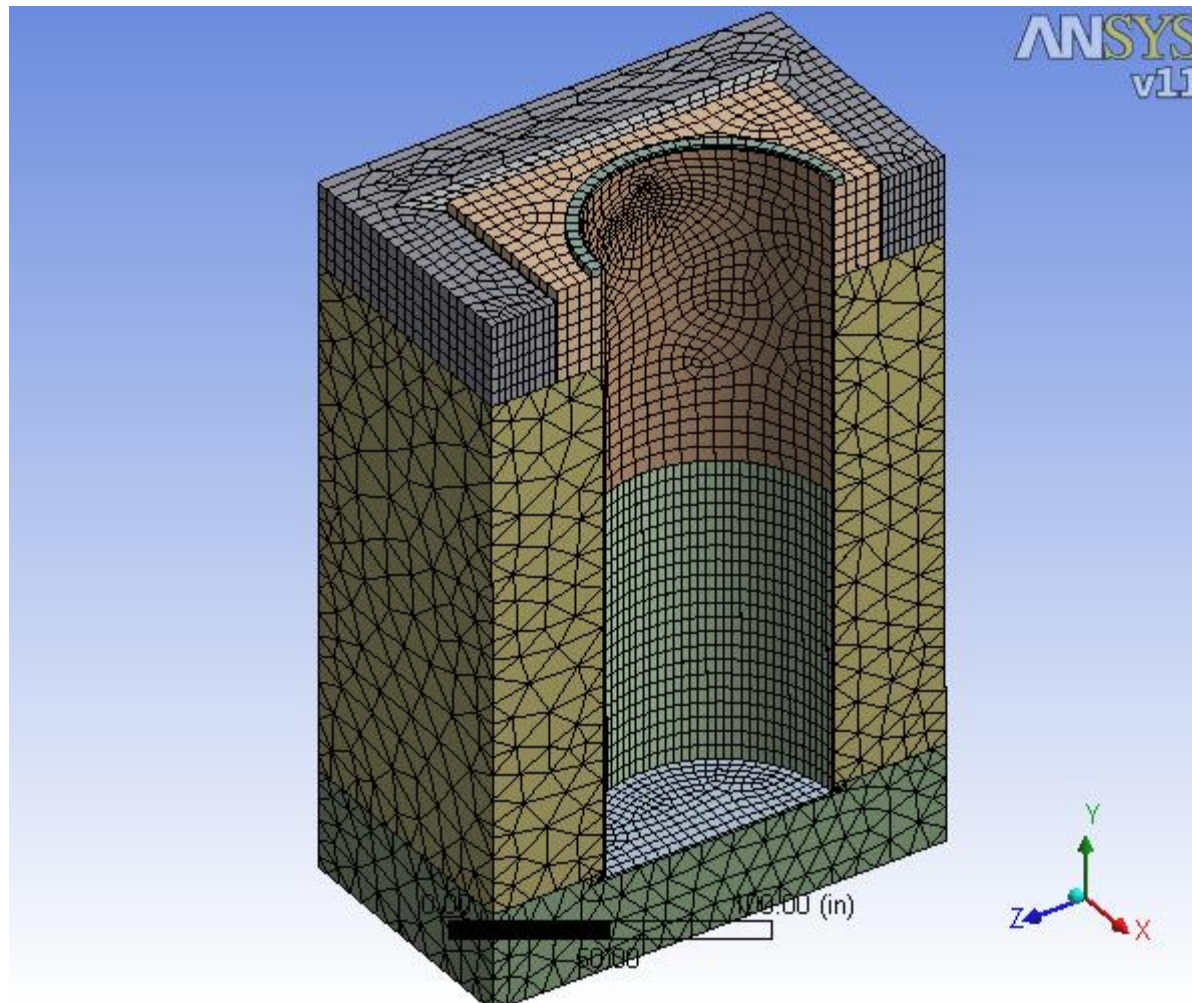


Figure 3.I.2; 3-D ANSYS Finite Element One-Half Model of Substrate Surrounding VVM, CEC Container Shell, TSP, and VIP

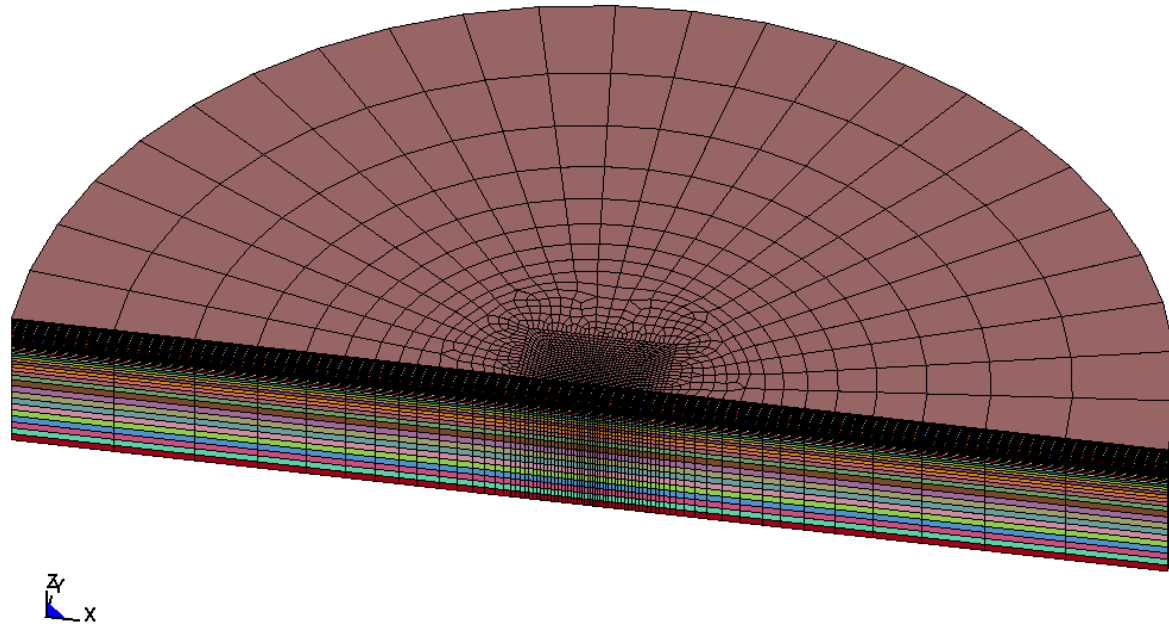


Figure 3.I.3-A; 3-D LSDYNA Soil Model for Design Basis Seismic Response Analysis

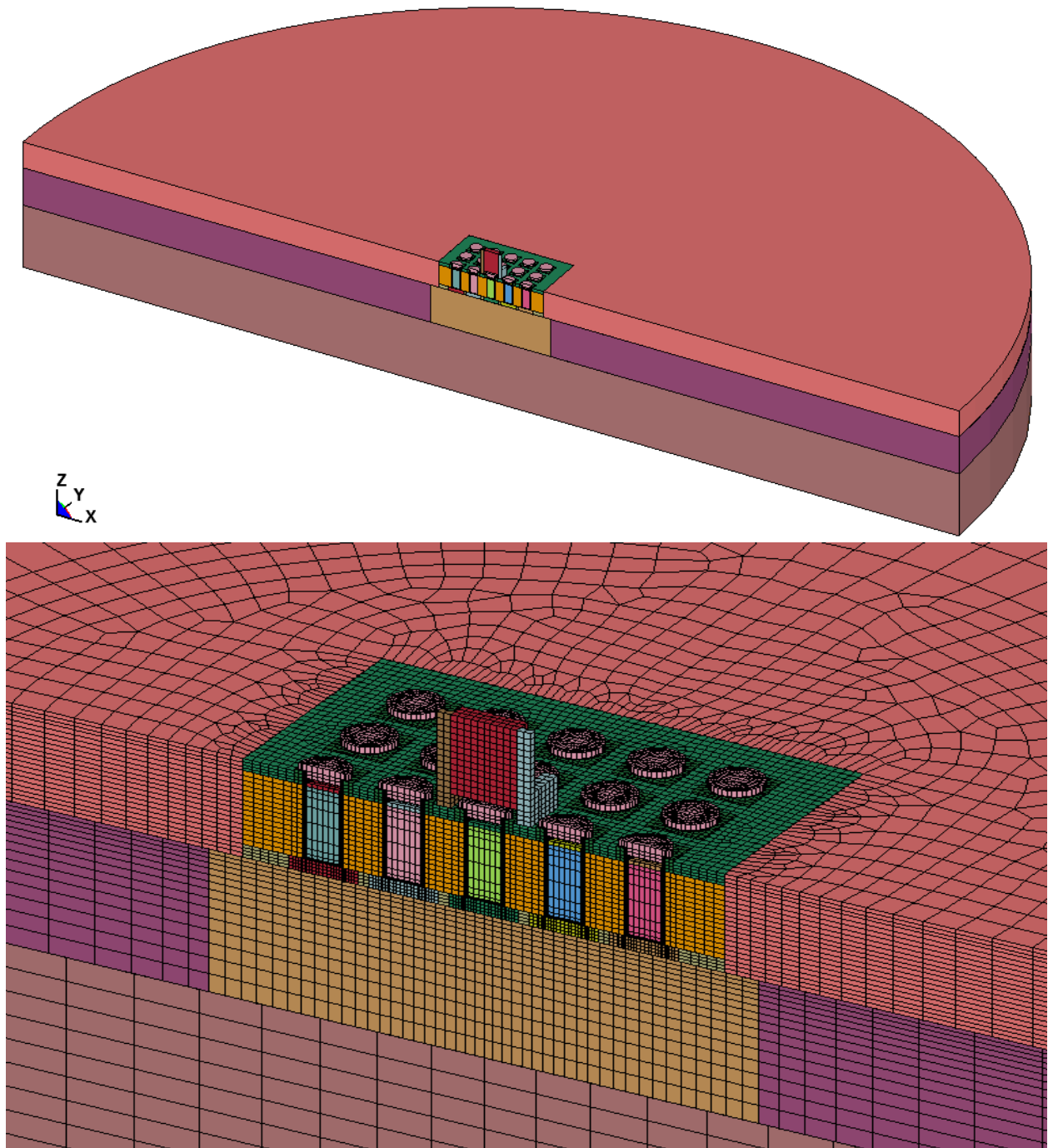


Figure 3.I.3-B; 3-D LSDYNA Model for Non-Linear SSI Analysis of 5x5 loaded VVMs on the Support Foundation



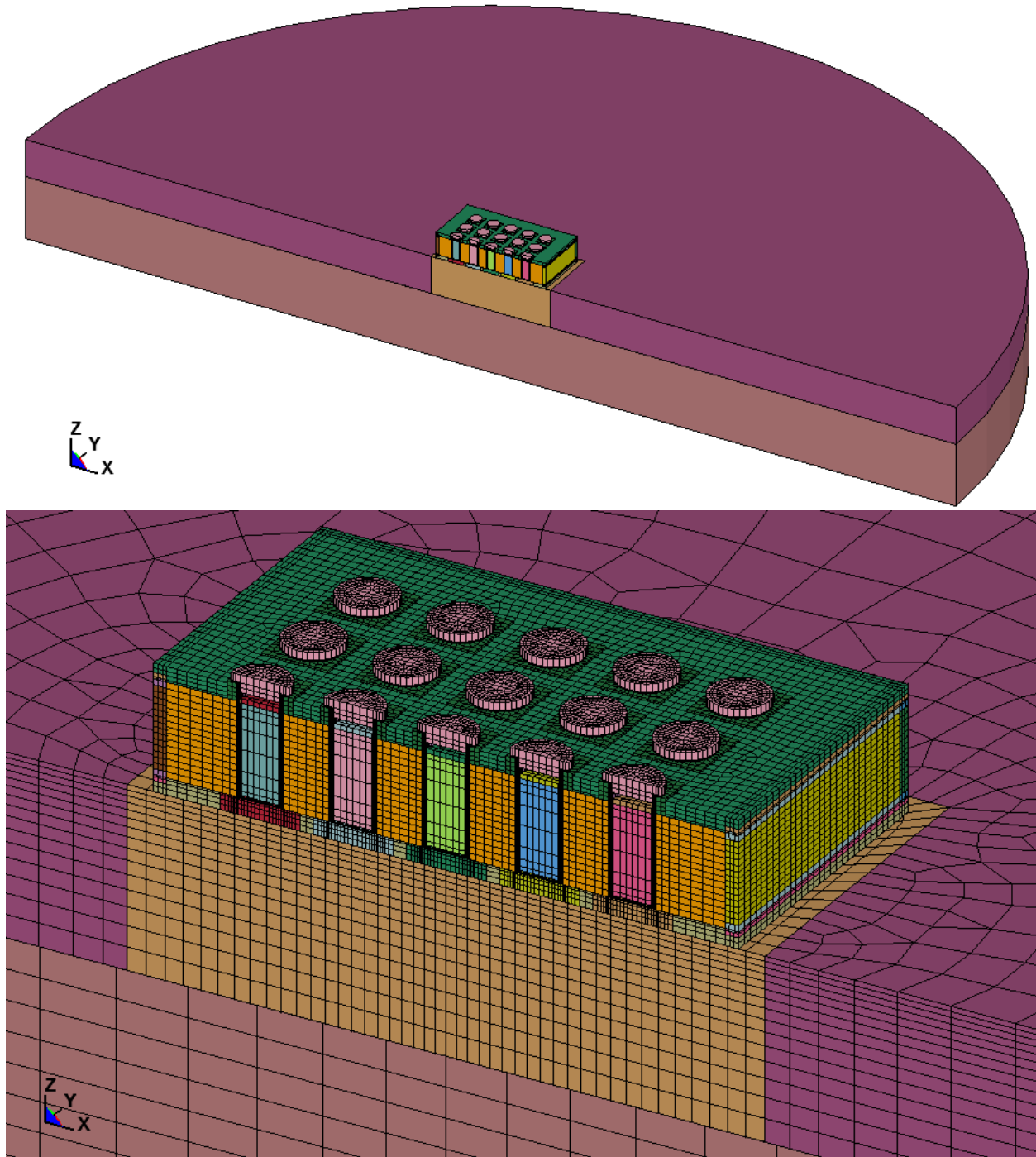


Figure 3.I.3-C; 3-D LSDYNA Model for Non-Linear SSI Analysis of 5x5 loaded VVMs on the Support Foundation with Retaining Walls



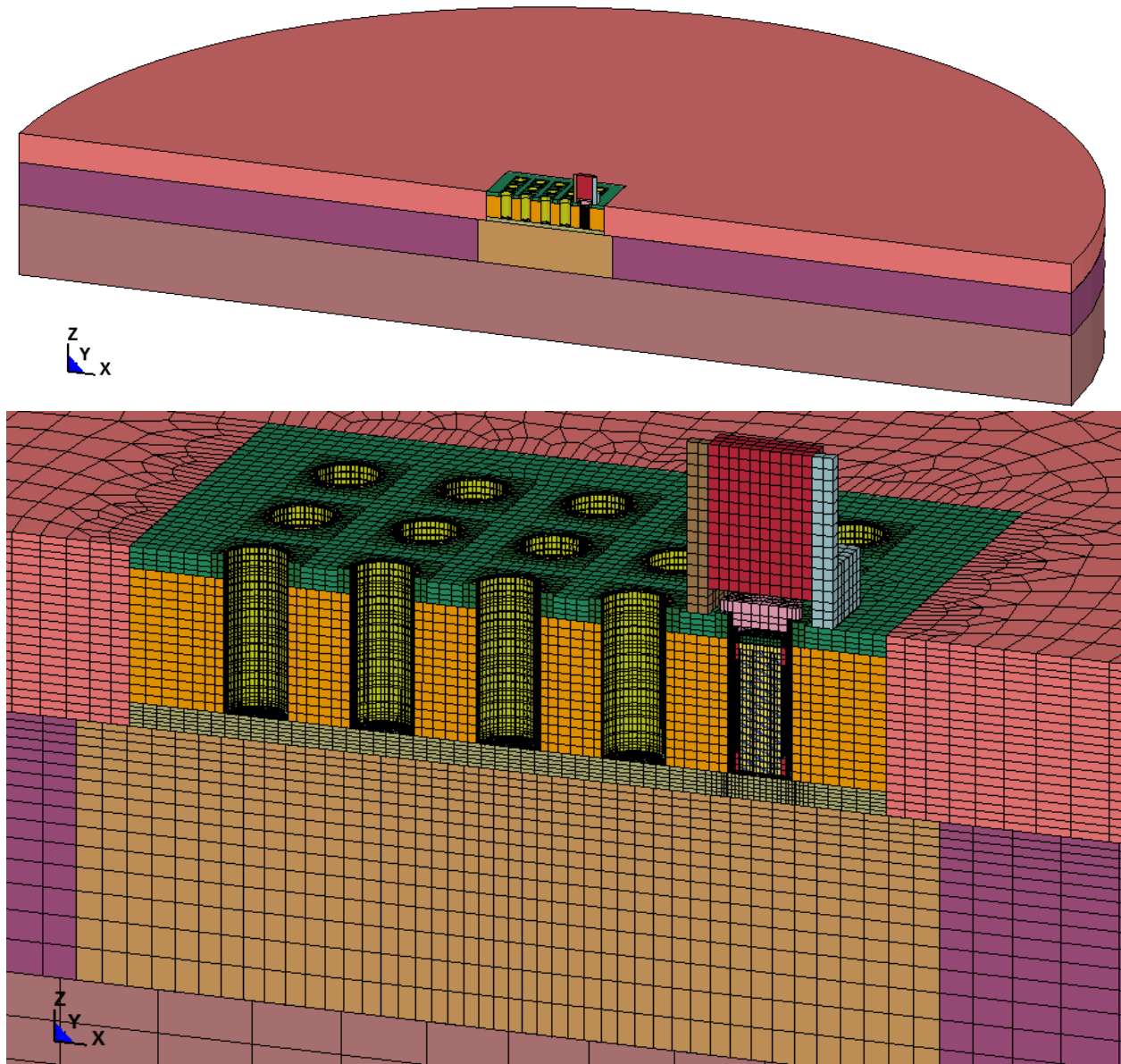


Figure 3.I.3-D; 3-D LSDYNA Model for Non-Linear SSI Analysis of a single VVM on Support Foundation

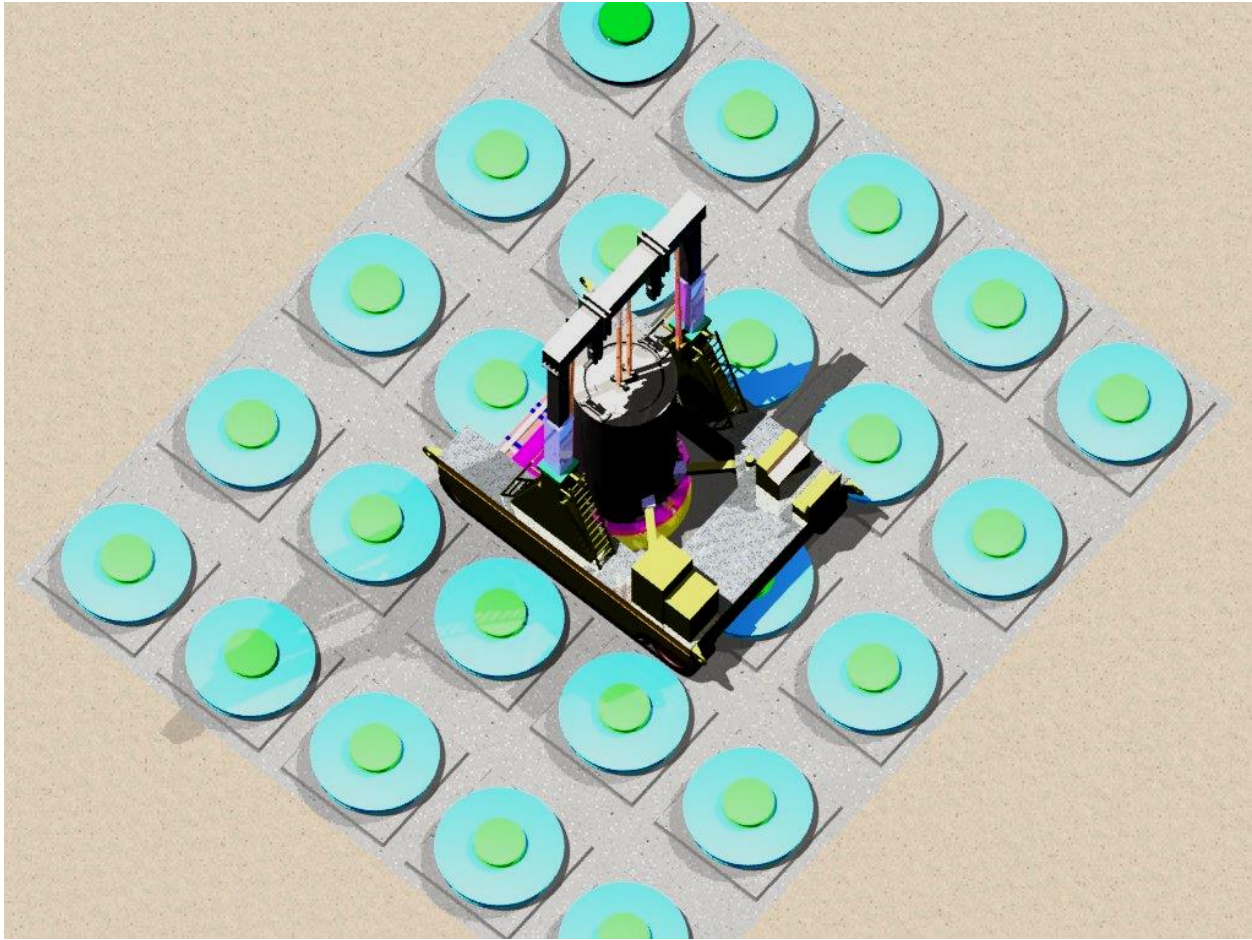


Figure 3.I.4; Cask Transporter on the ISFSI Positioned to Transfer MPC in the Central Cavity in the 5x5 VVM Array (illustrative analysis case)

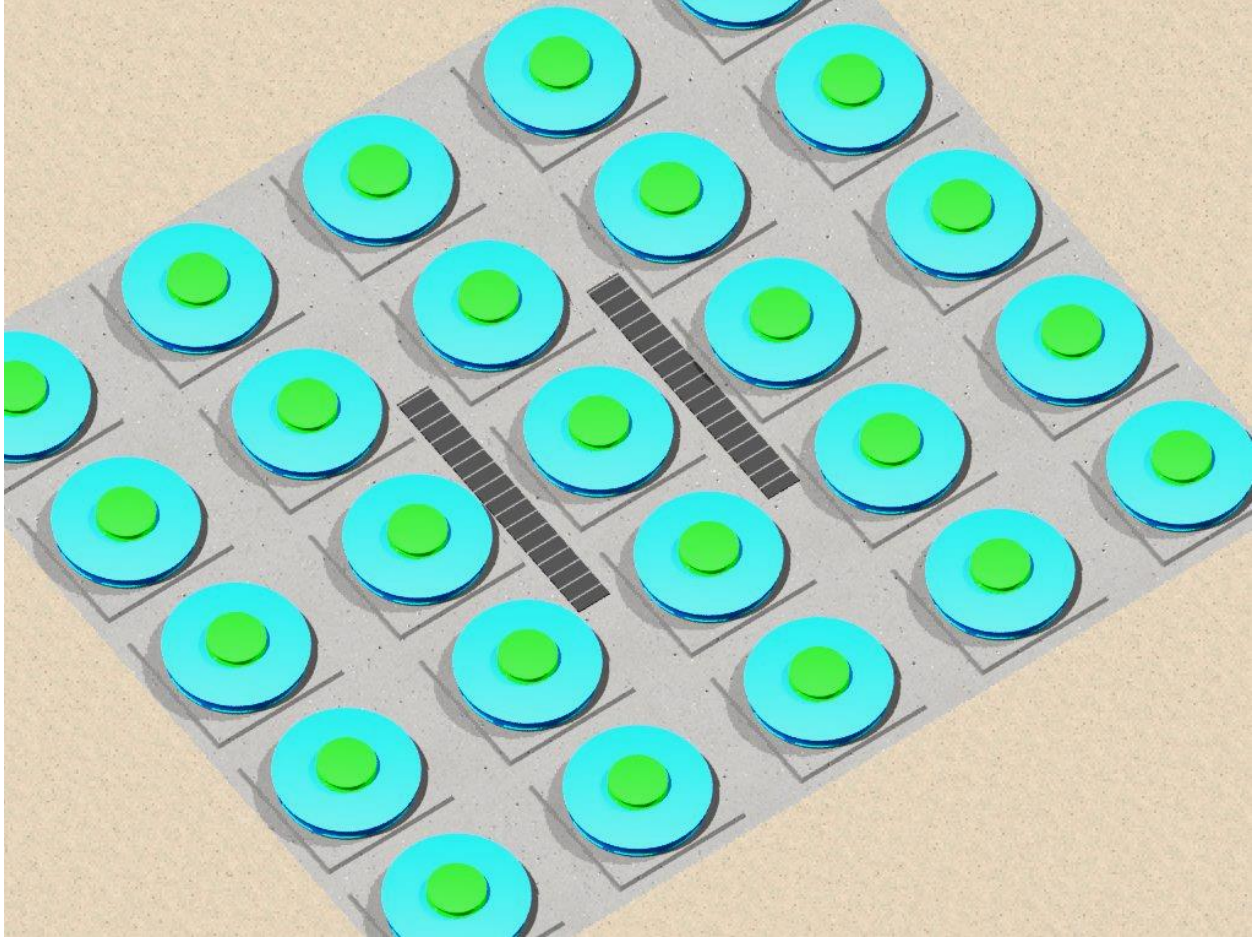


Figure 3.I.5; Load patch from the loaded Transporter in Figure 3.I.19  
(Illustrative analysis case)

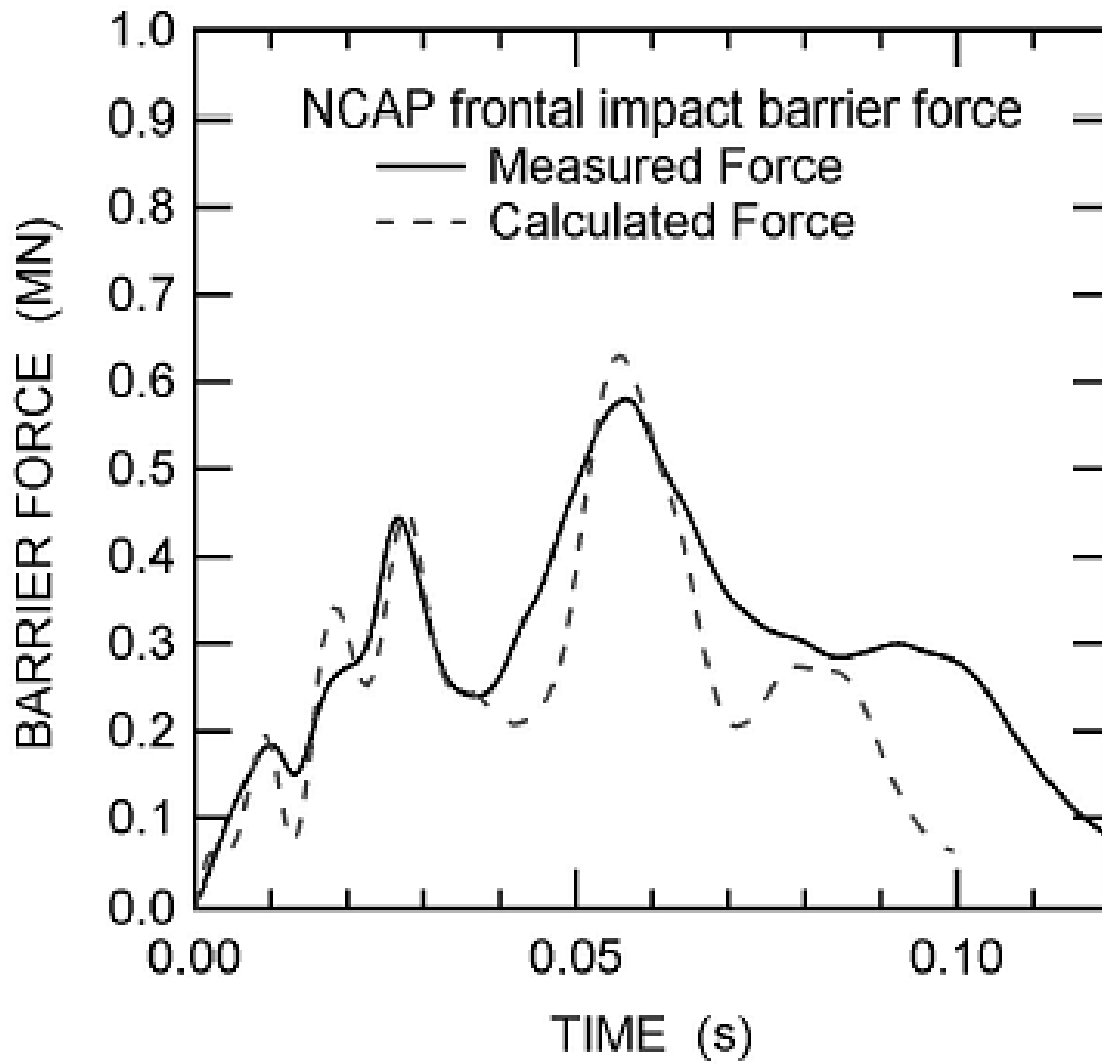


Figure 3.I.6; Test Results from 35mph Impact of a Ford (1705 Kg) Against a Rigid Wall



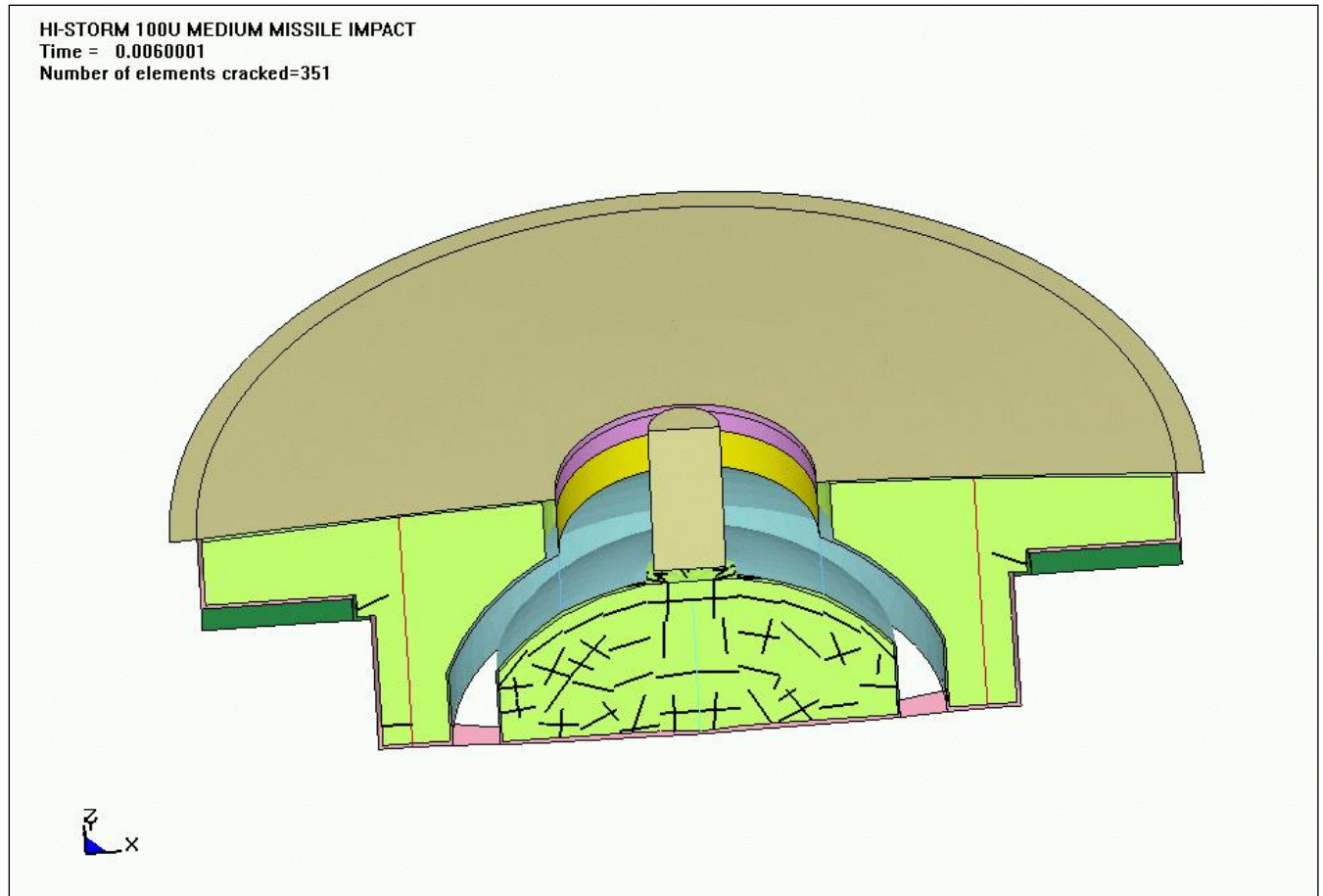


Figure 3.I.7; LSDYNA Model Section for Central Intermediate Missile Strike (subsequent to impact)

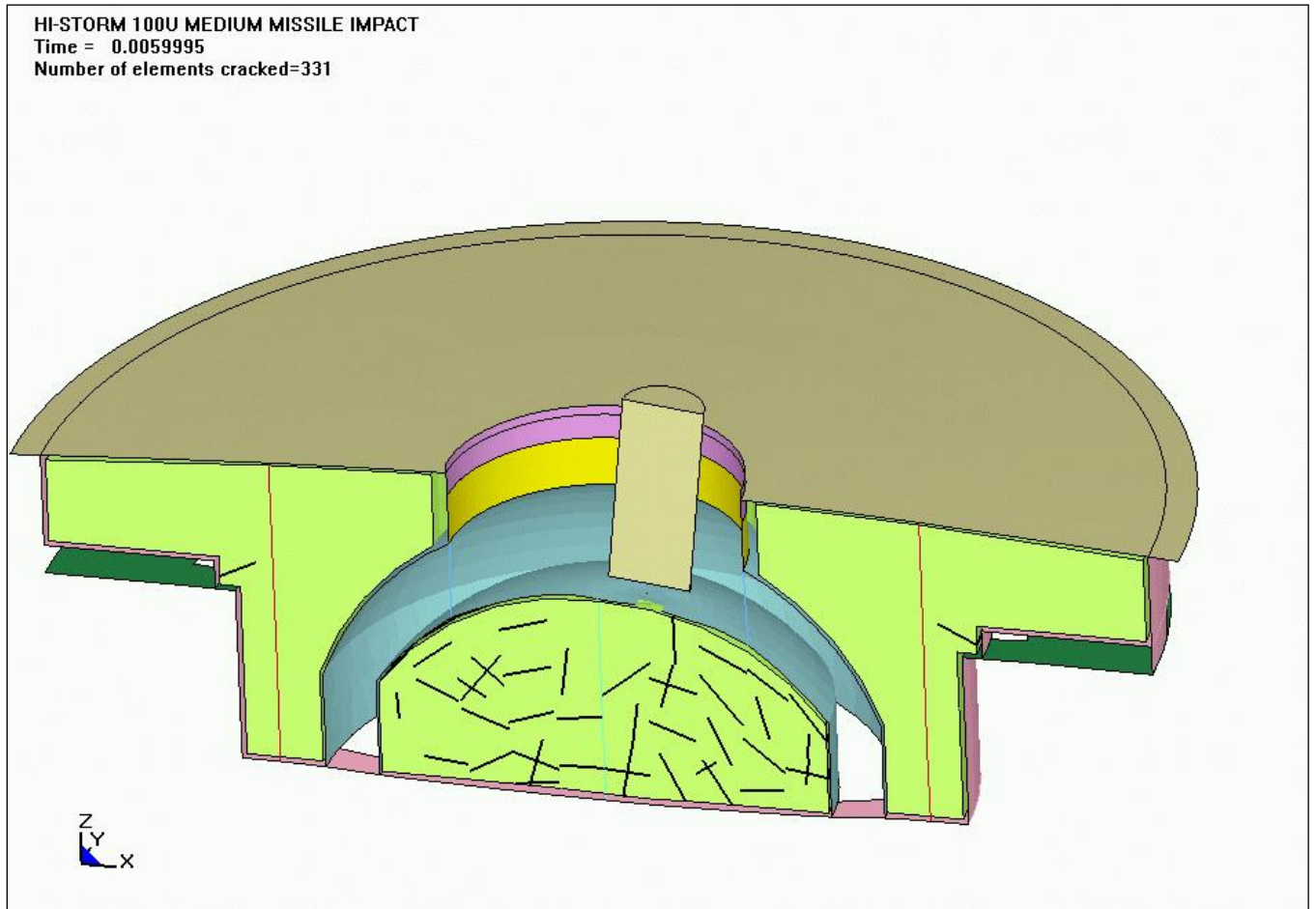


Figure 3.I.8; LSDYNA Model Section for Inclined Intermediate Missile Strike (subsequent to impact)

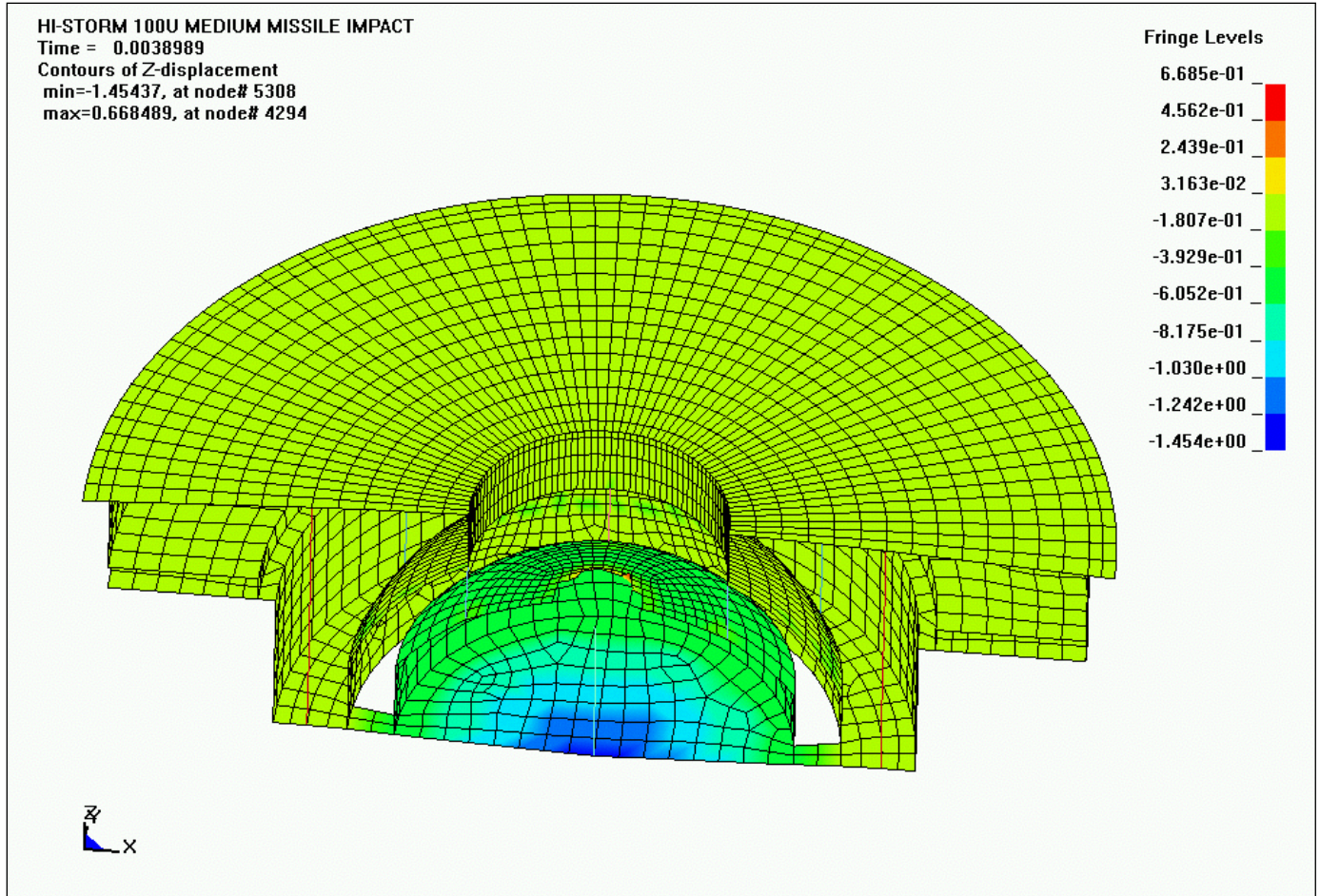


Figure 3.I.9; Deformation Profile at Time of Maximum Deformation – Central Strike

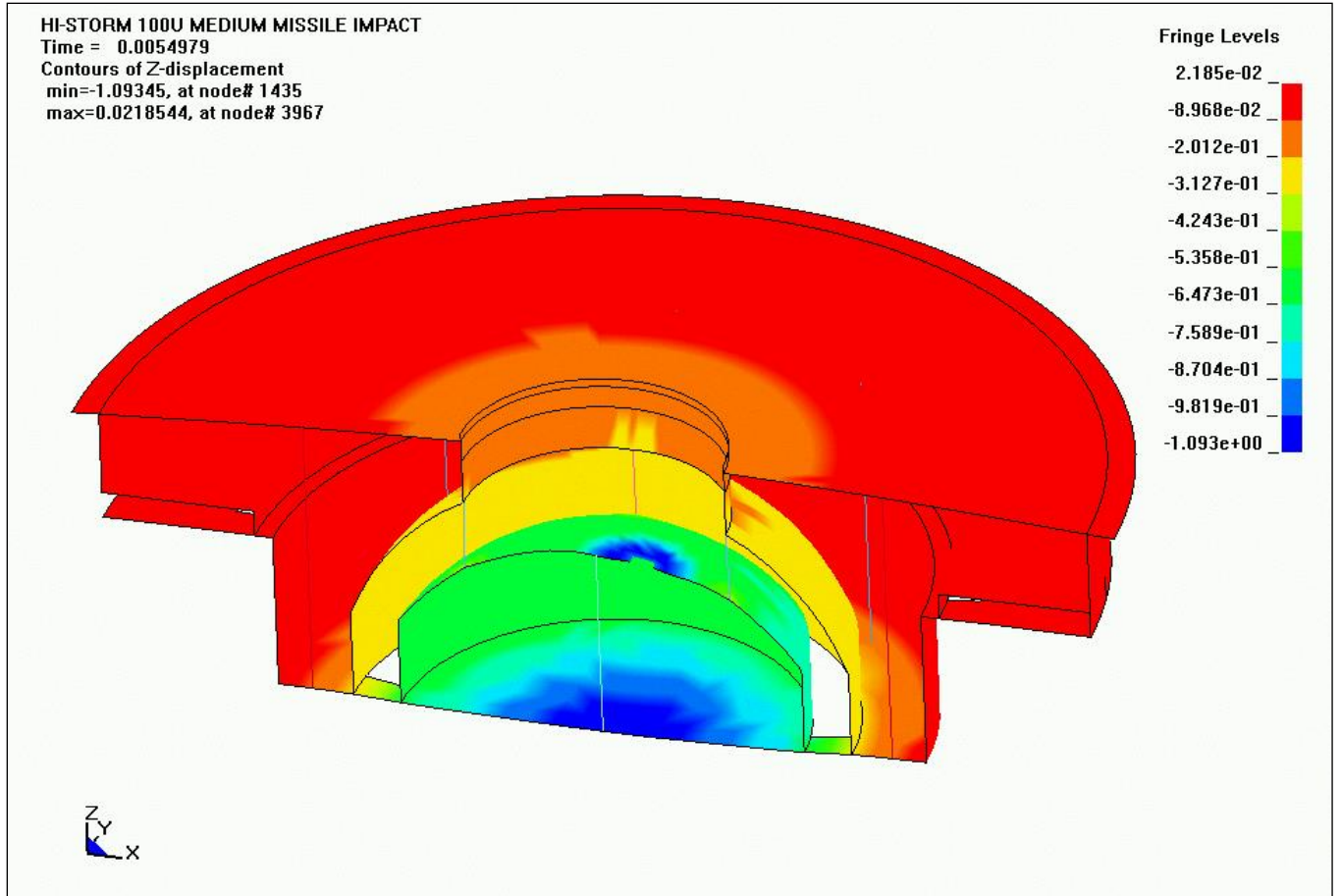


Figure 3.I.10; Deformation Profile at Time of Maximum Deformation – Inclined Strike



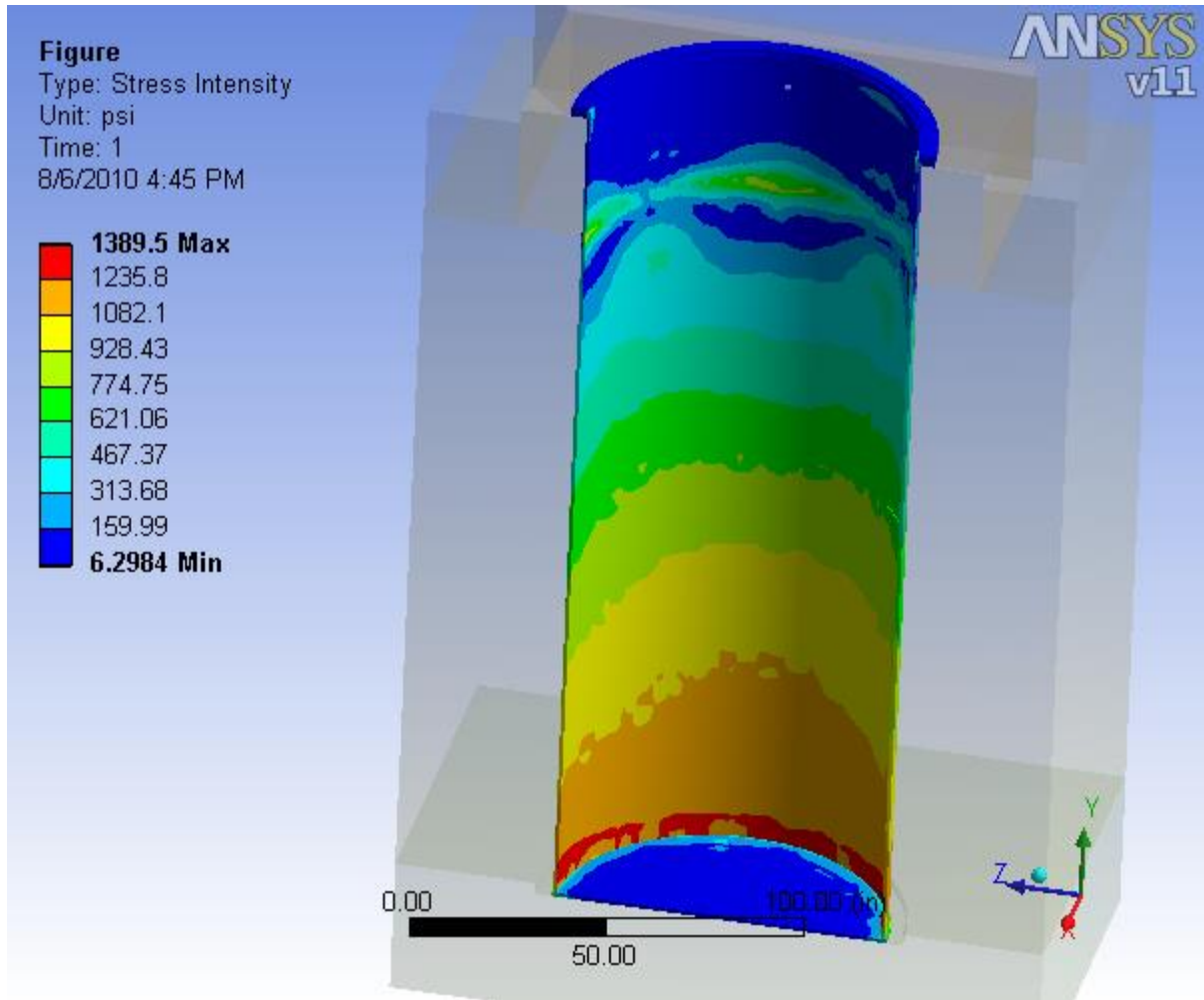


Figure 3.I.11; Stress Distribution in CEC Shell from Transporter and Substrate (Load Case 07)

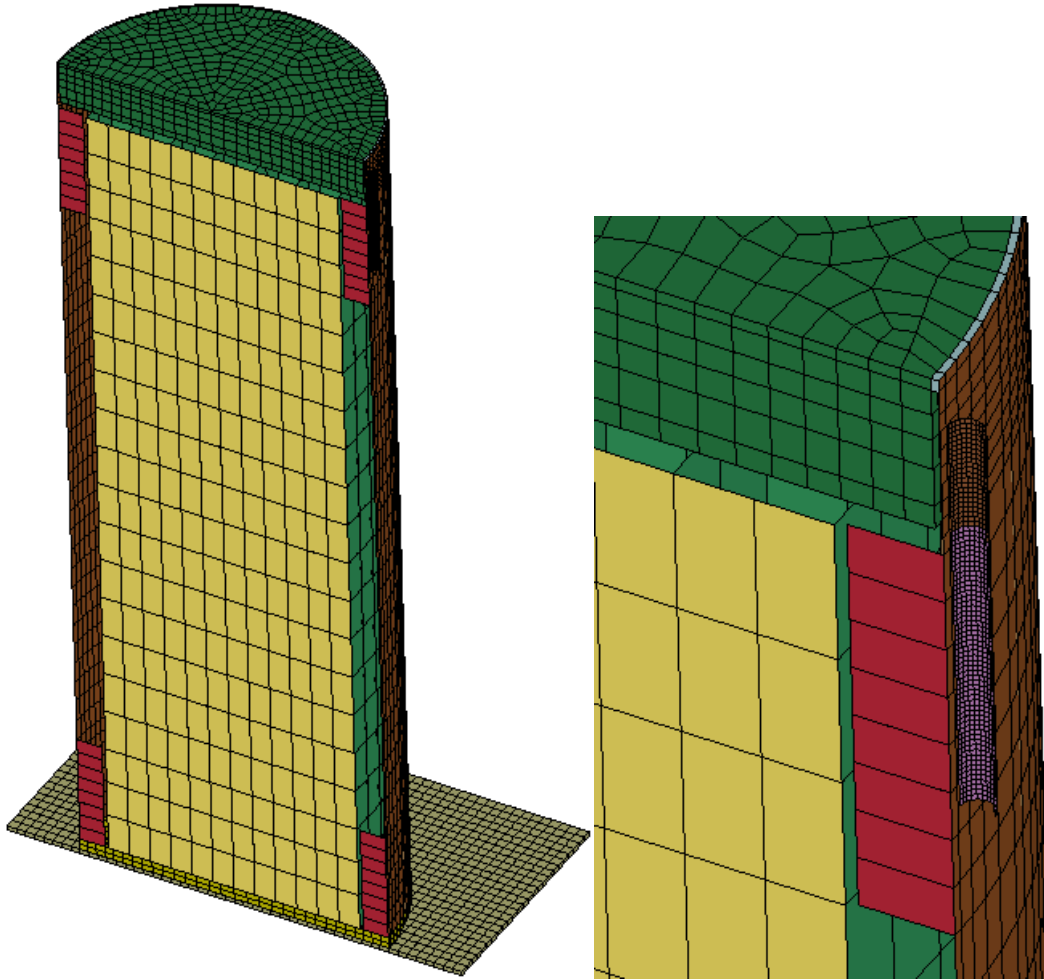


Figure 3.I.12; MPC Guide/MPC Impact LS-DYNA Model

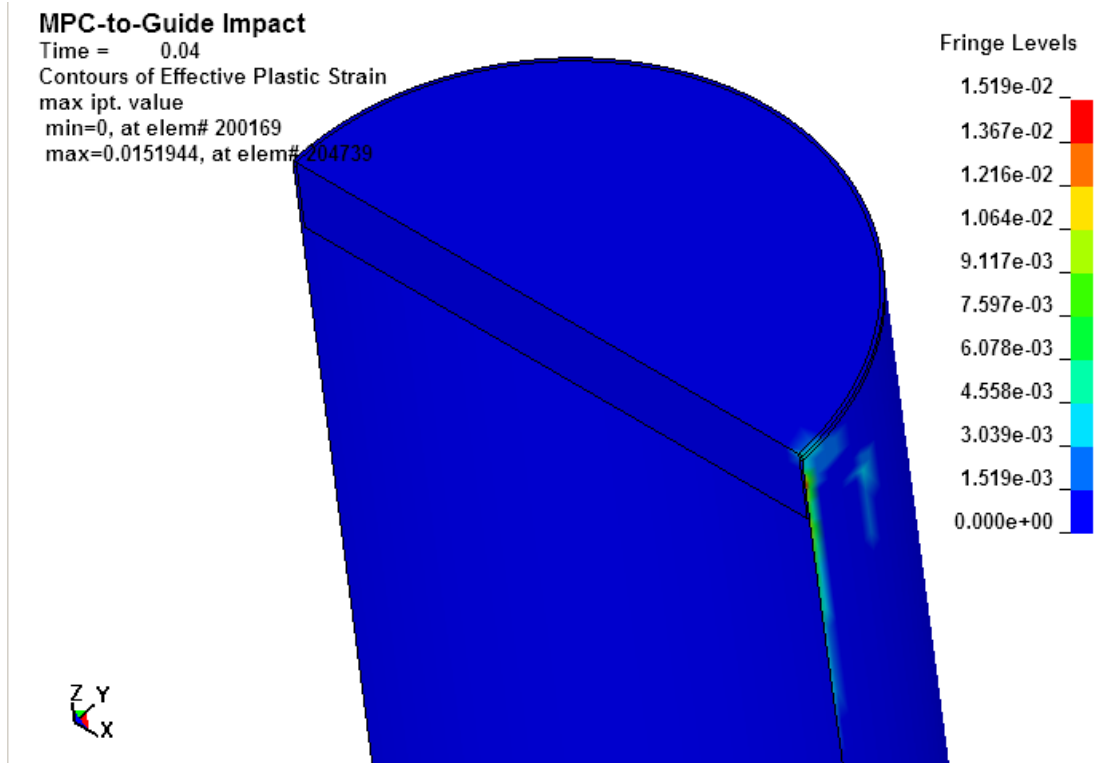


Figure 3.I.13; Maximum Plastic Strain of the MPC Enclosure Members in the Impact Region

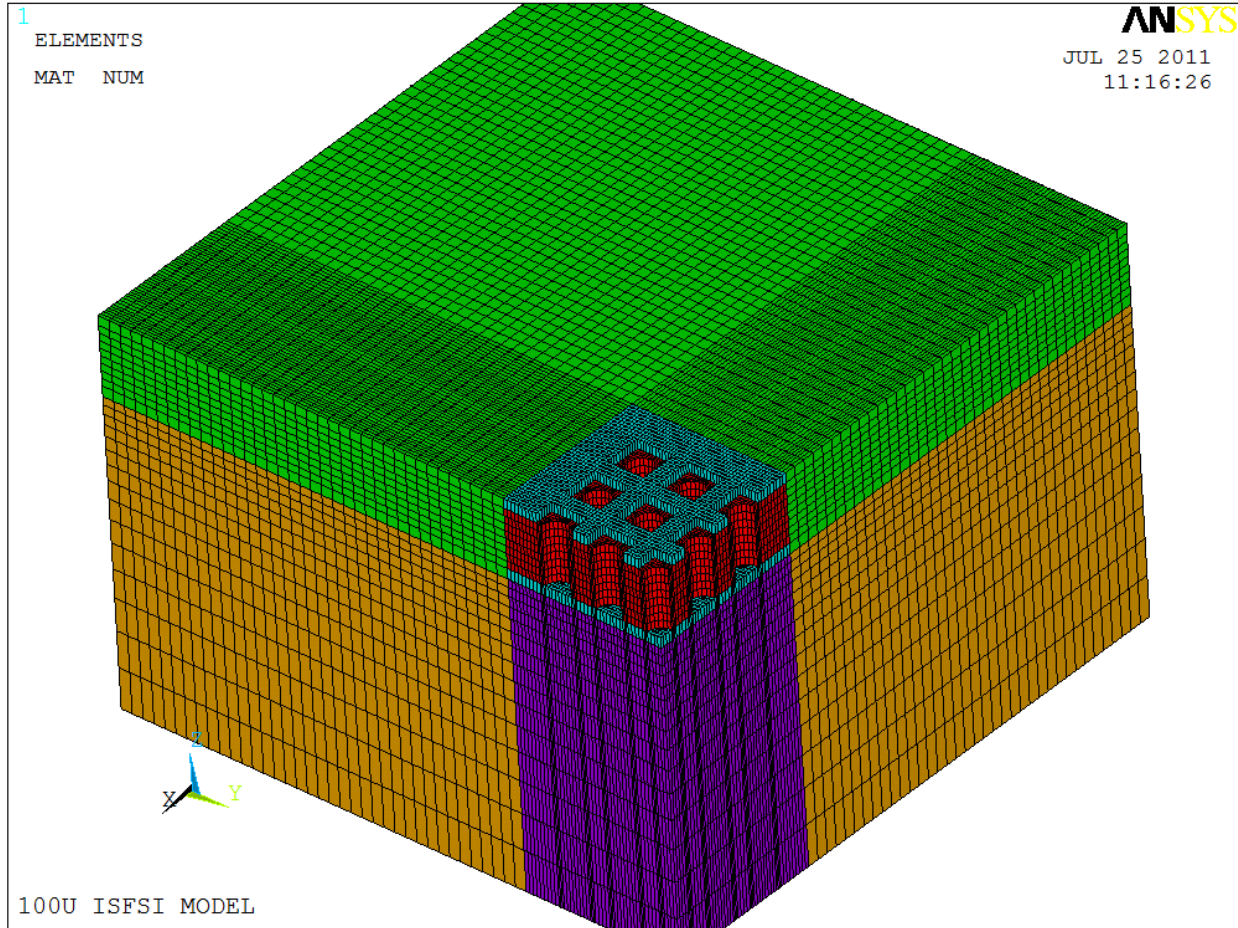
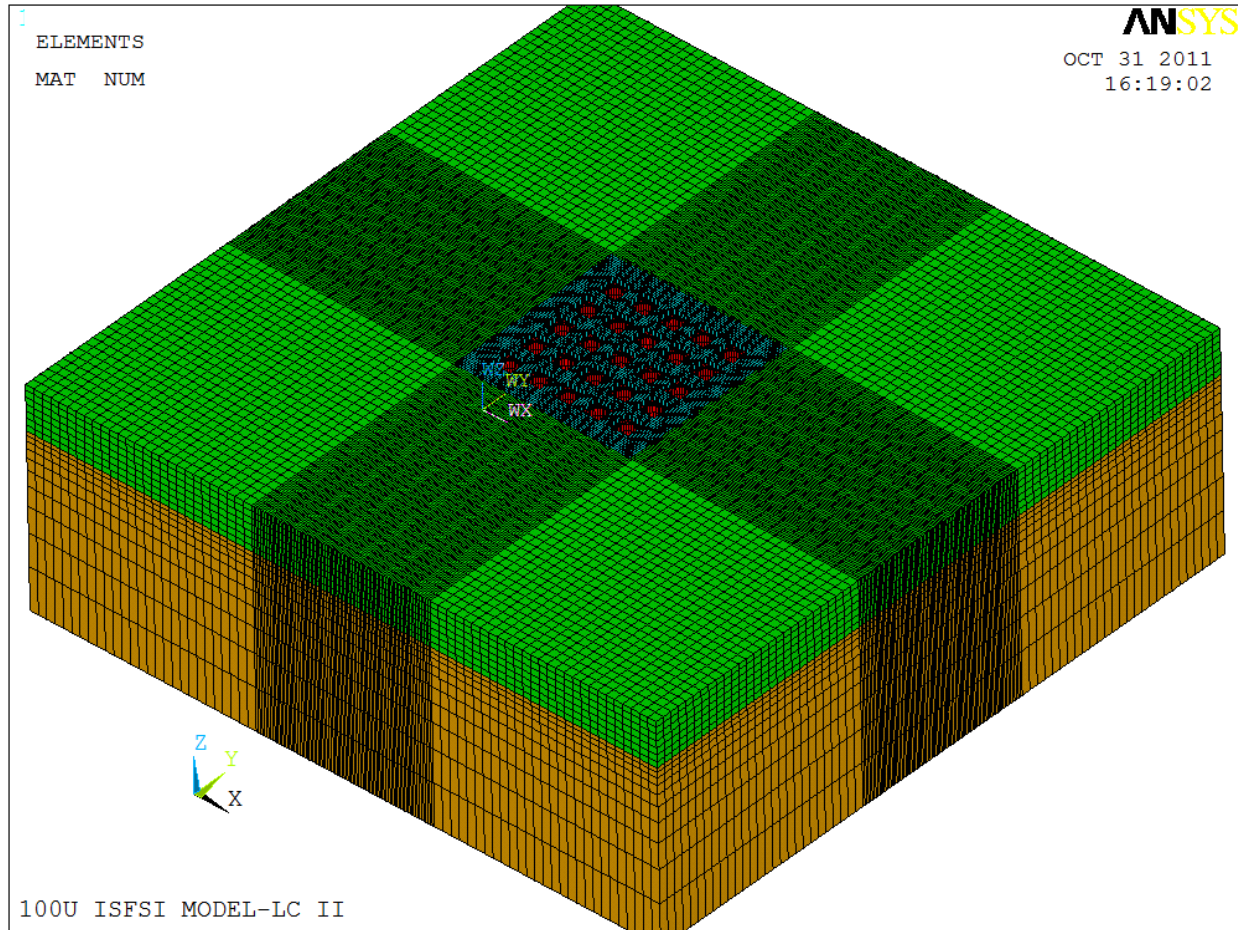


Figure 3.I.14; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Models I, III and IV



Note that Simulation Model II is identical to Models I, III and IV in all respects, except that this is a full FE model with coarser element mesh for the soil components (soil elements). Also note that the symmetric boundary constraint does not apply to the full model.

Figure 3.I.15; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Model II

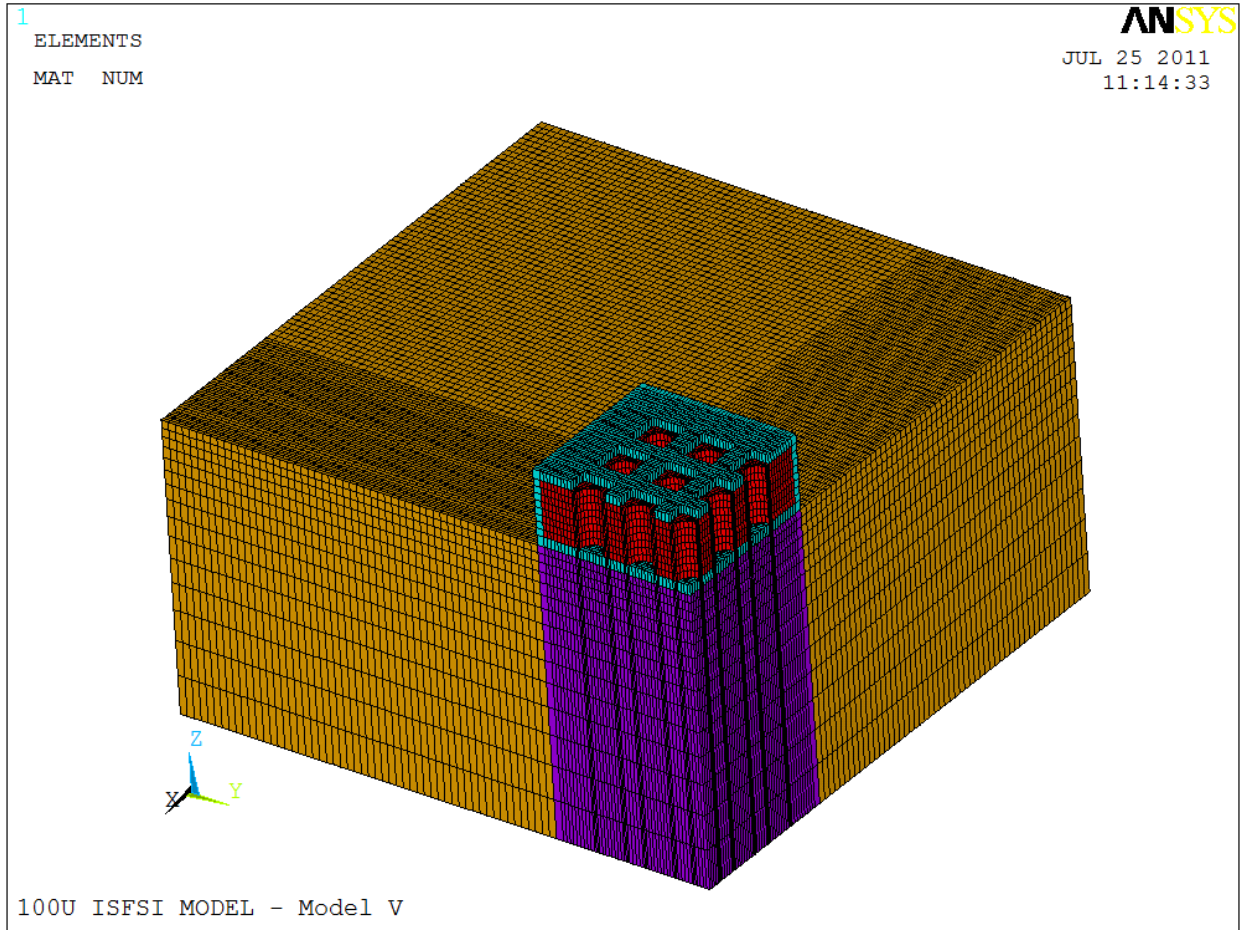
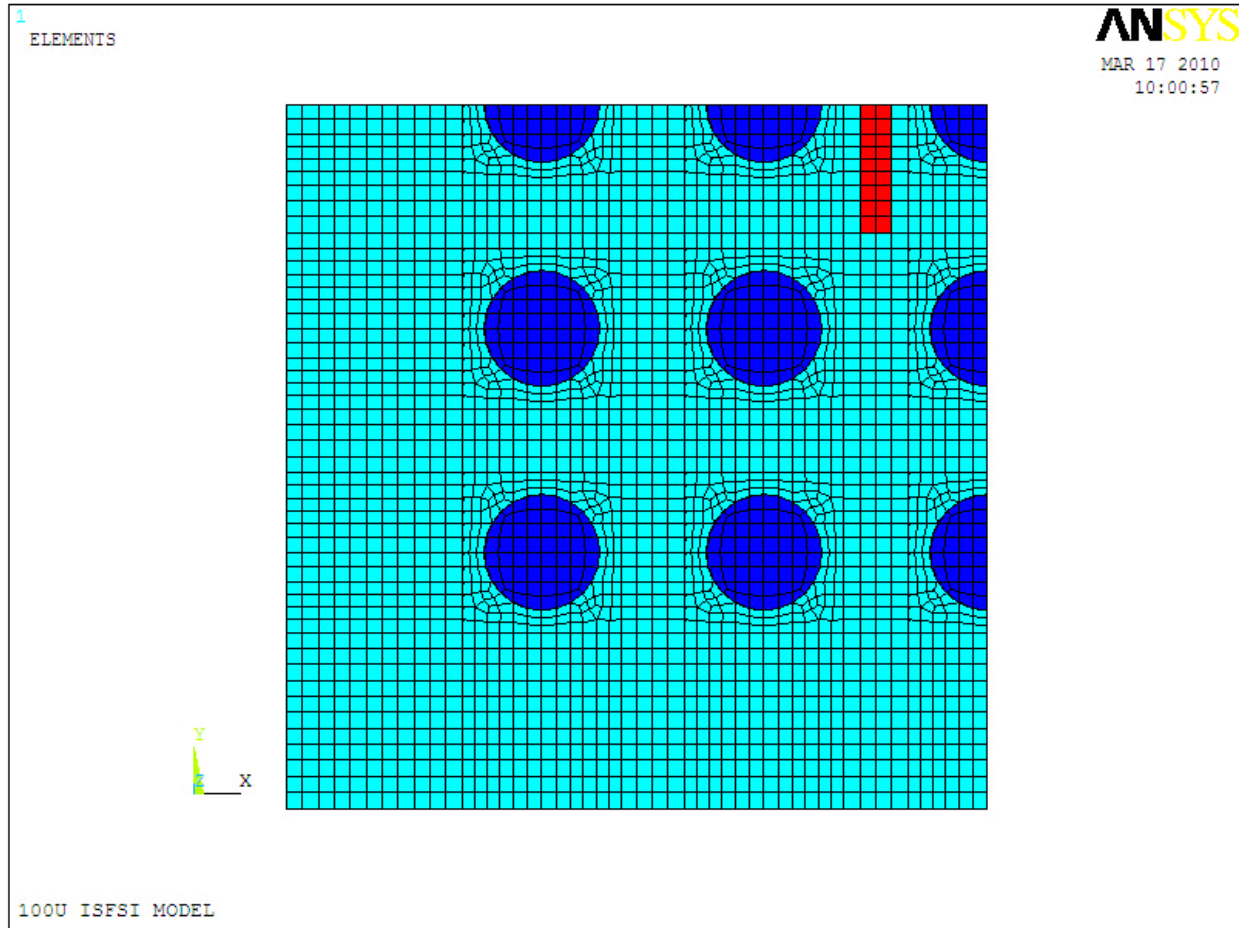


Figure 3.I.16; Finite Element Model of the ISFSI Reinforced Concrete Structures for Simulation Model V (with Optional Retaining Wall)





Note: The blue footprints show the SFP area loaded with the SSC's and the red footprint represents the loaded TSP area with the transporter (VCT). The soil extending beyond the SFP boundary is not shown in the above plot for clarity.

Figure 3.I.17; ANSYS Finite Element Model of ISFSI Showing the Fully Loaded Configuration (Simulation Model I)

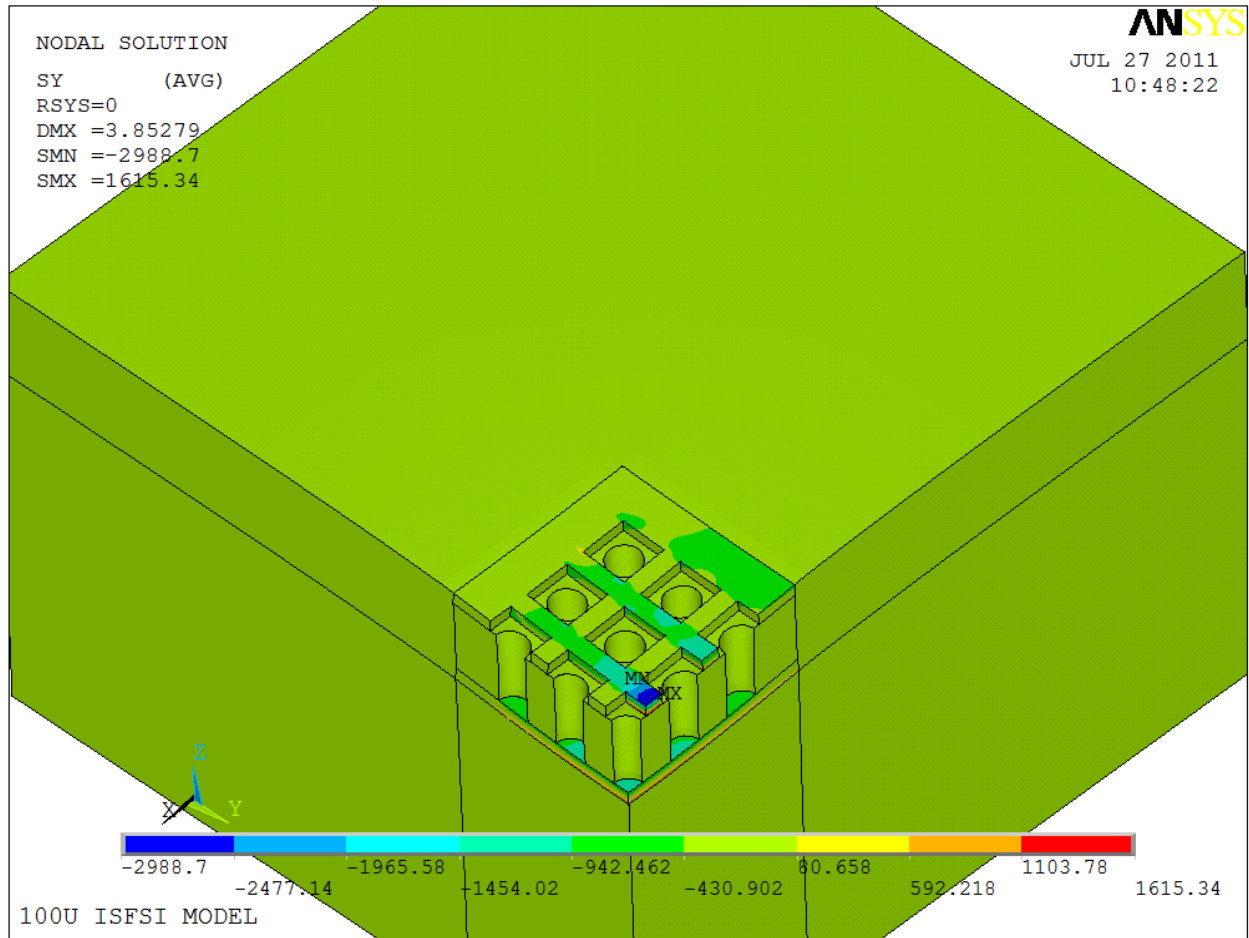
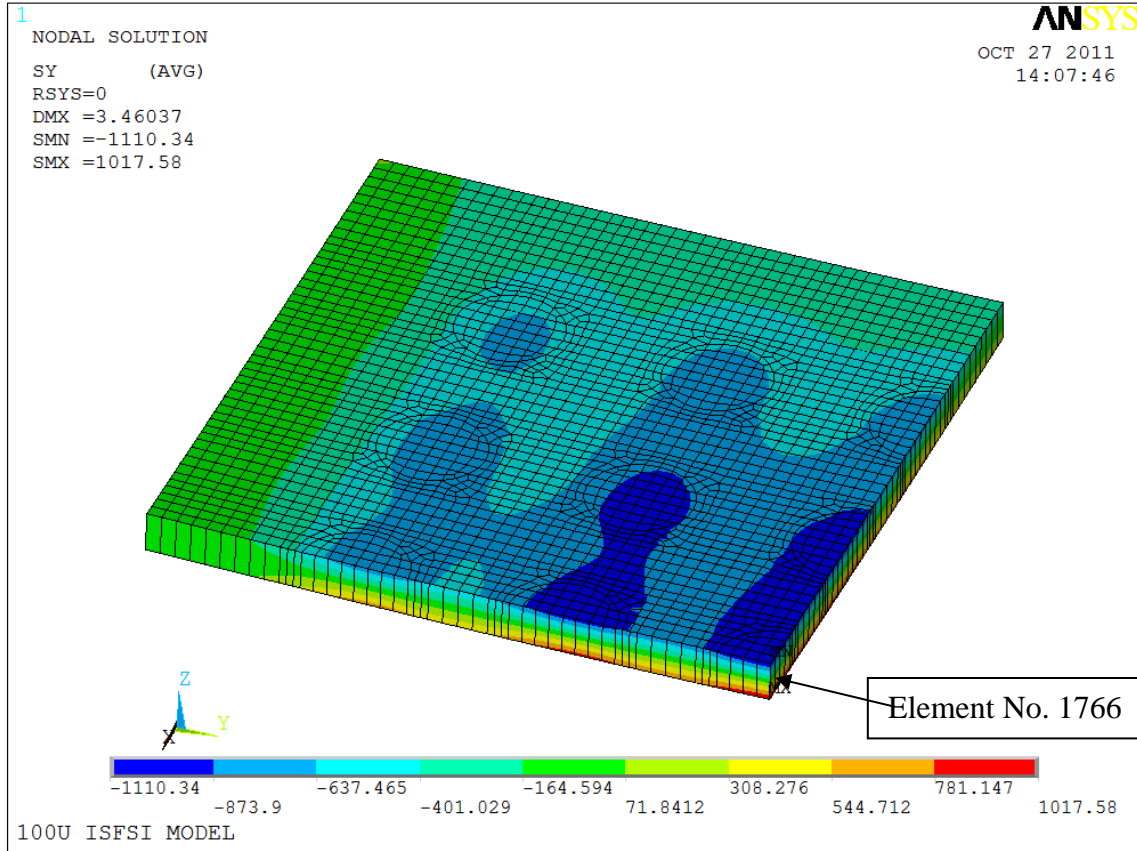


Figure 3.I.18a; Normal Stress ( $S_y$ ) in the ISFSI in the Direction of the Transporter Path for Simulation Model I – Load Combination LC-3 from Table 2.I.11



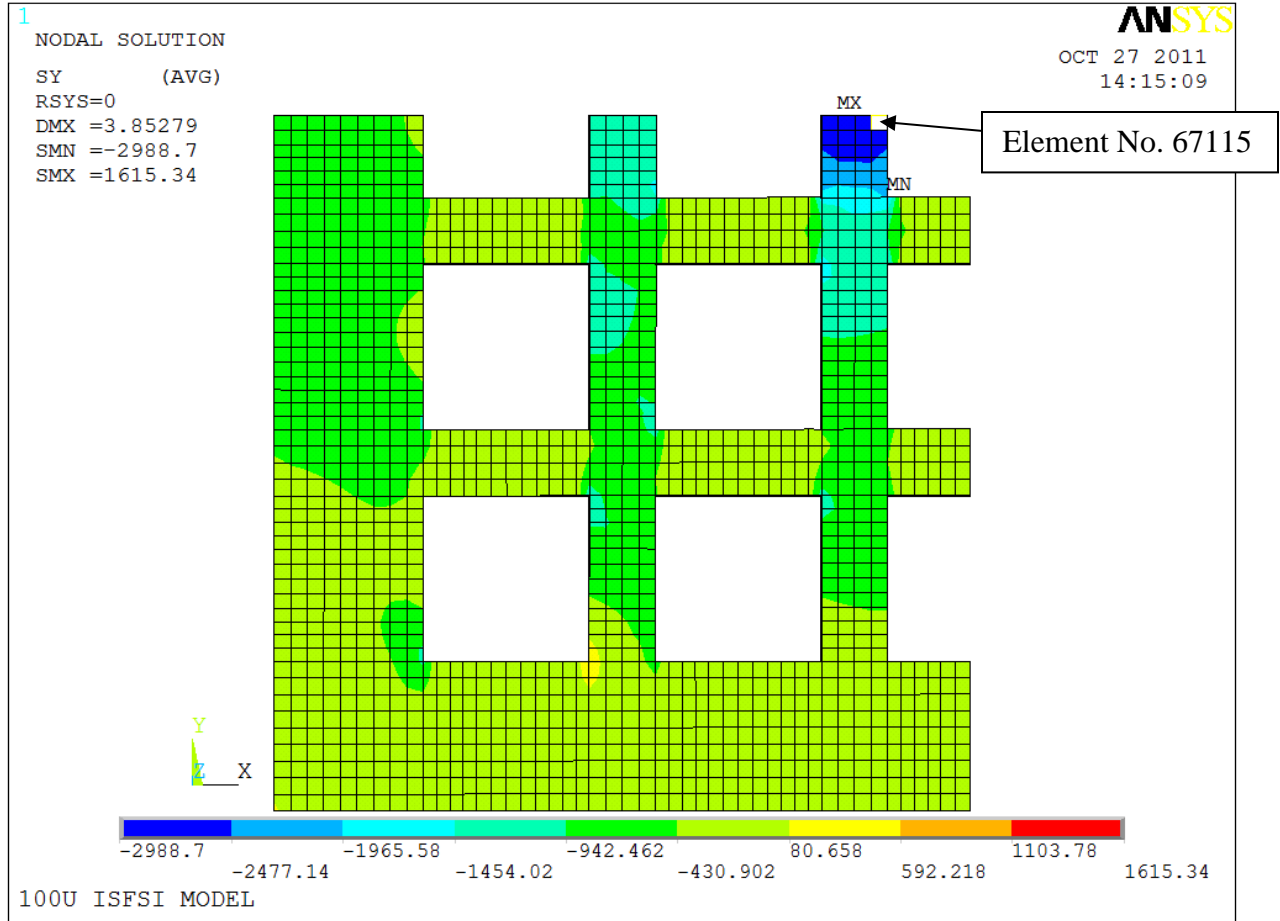


Notes:

Element 1766 realizes the maximum loading and produces the limiting safety factor.

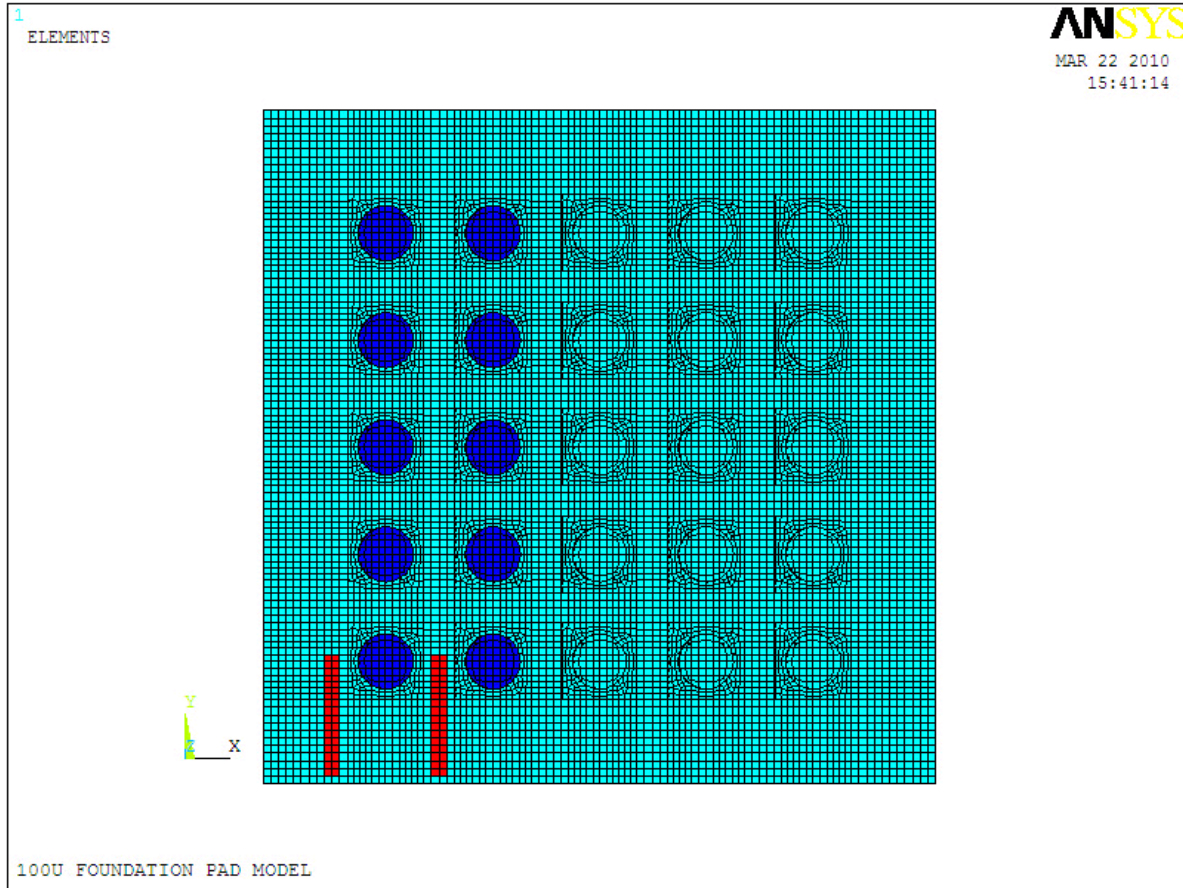
Moment (Mx) in the limiting element is calculated as =  $\{1017.6 - (-1110.3)\} \times 33^2 / 12 = 193,106.9$  lbf-in/in, Axial Force (Fx) in the limiting element is calculated as =  $\{(-1110.3) + 1017.6\} \times 33 / 2 = -1529.6$  lbf/in, where 33 represents the thickness of the SFP in inches. The axial force and moment values calculated here correspond well with the results shown in Table 3.I.10 for the SFP for Simulation Model I. The small deviation (less than 0.6%) from the results shown in Table 3.I.10 is due to the fact that the results in Table 3.I.10 are computed based on the average surface stresses (top and bottom) considering all 4 nodes that define each element surface. In the calculations above, the maximum and minimum stresses at a single node location (top and bottom) are considered for simplicity.

Figure 3.I.18b; Normal Stress (Sy) in SFP, Simulation Model I – Load Combination LC-3 from Table 2.I.11



Note: Element 67115 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.18c; Normal Stress (Sy) in TSP Simulation Model I – Load Combination LC-3 from Table 2.I.11



Note: The blue footprints show the SFP area loaded with the SSC's and the red footprint represents the loaded TSP area with the transporter (VCT). The soil extending beyond the pad (SFP/TSP) boundary is not shown in the above plot for clarity.

Figure 3.I.19; ANSYS Finite Element Model of ISFSI Showing the Partially Loaded Configuration (Simulation Model II)

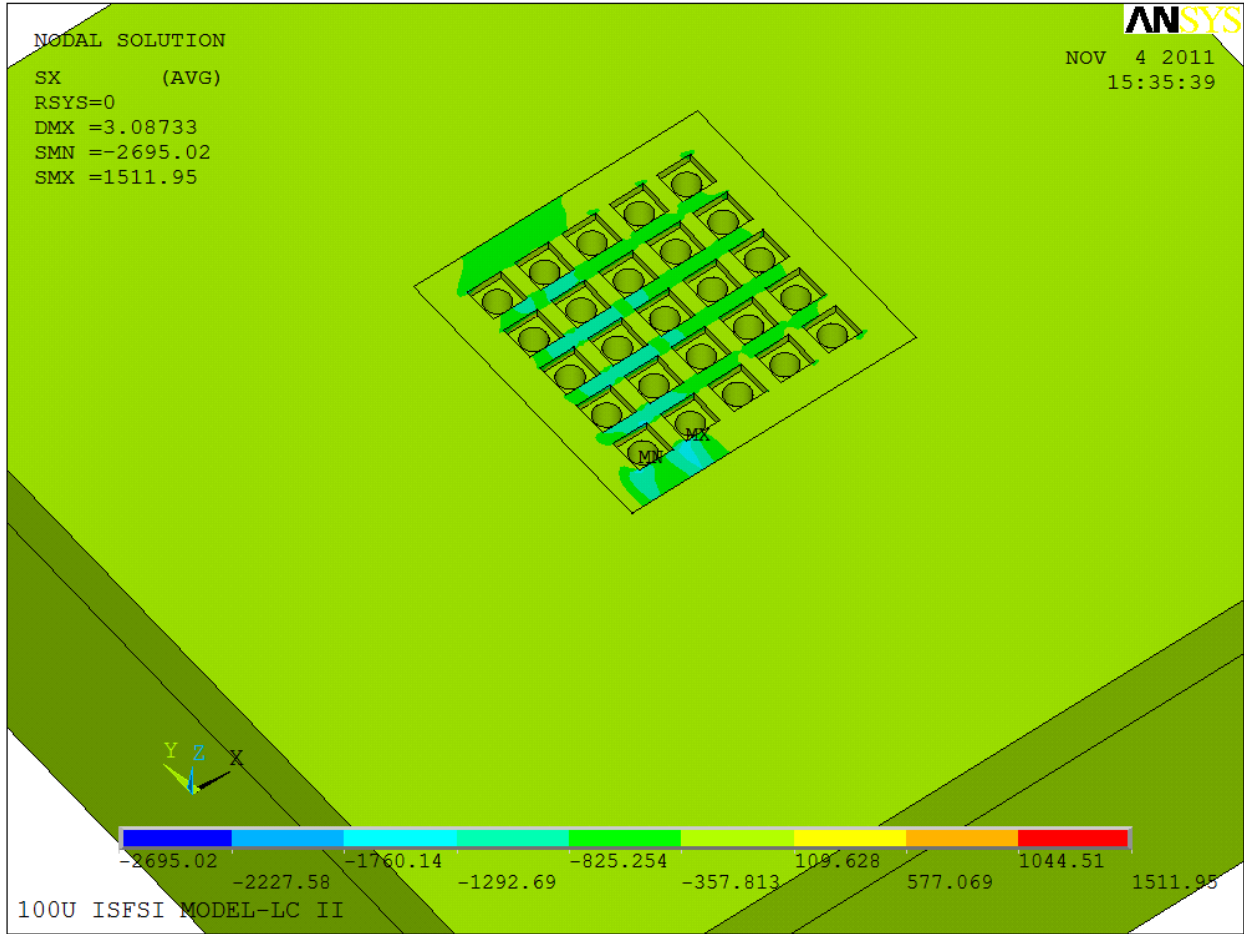
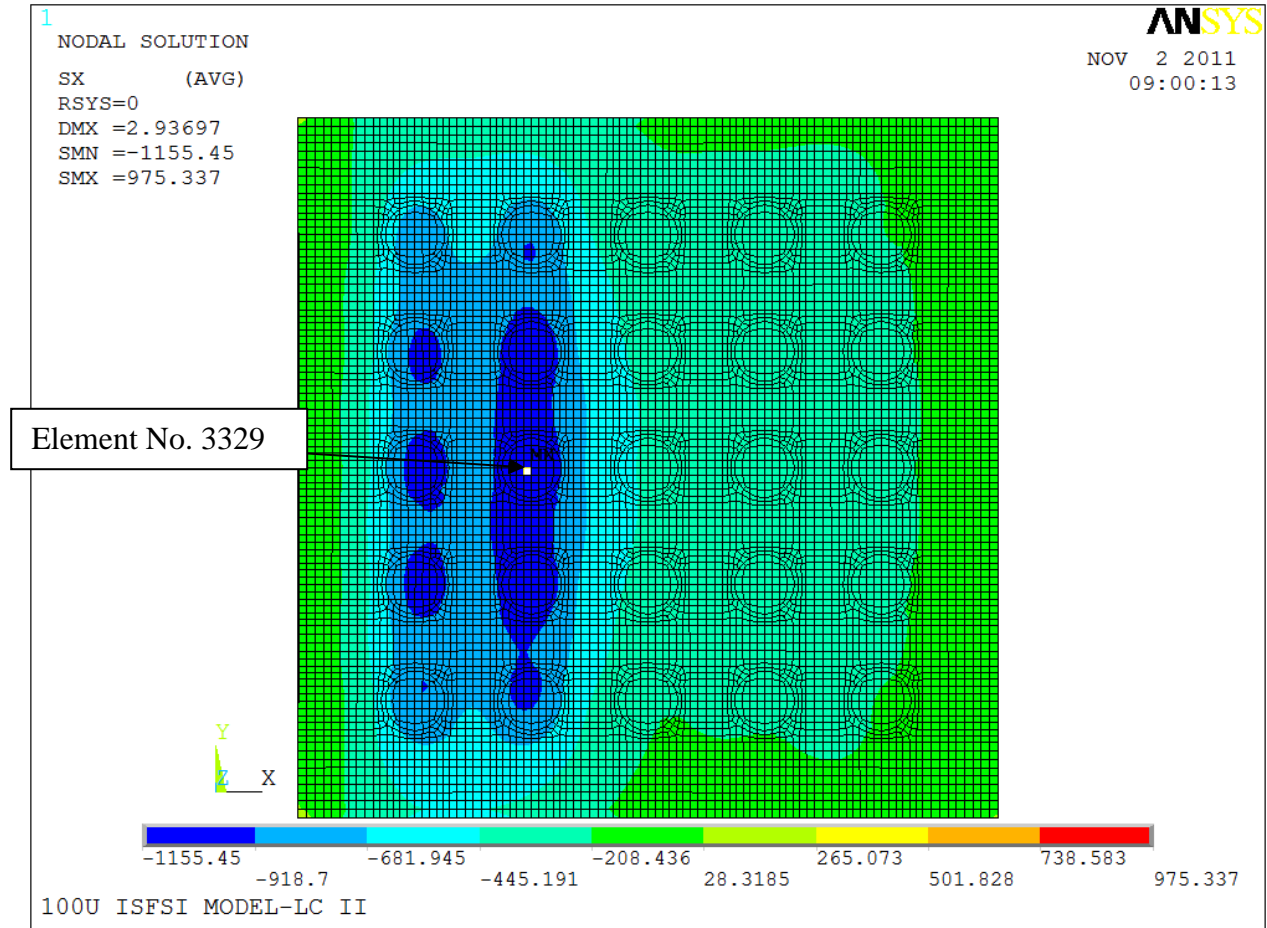
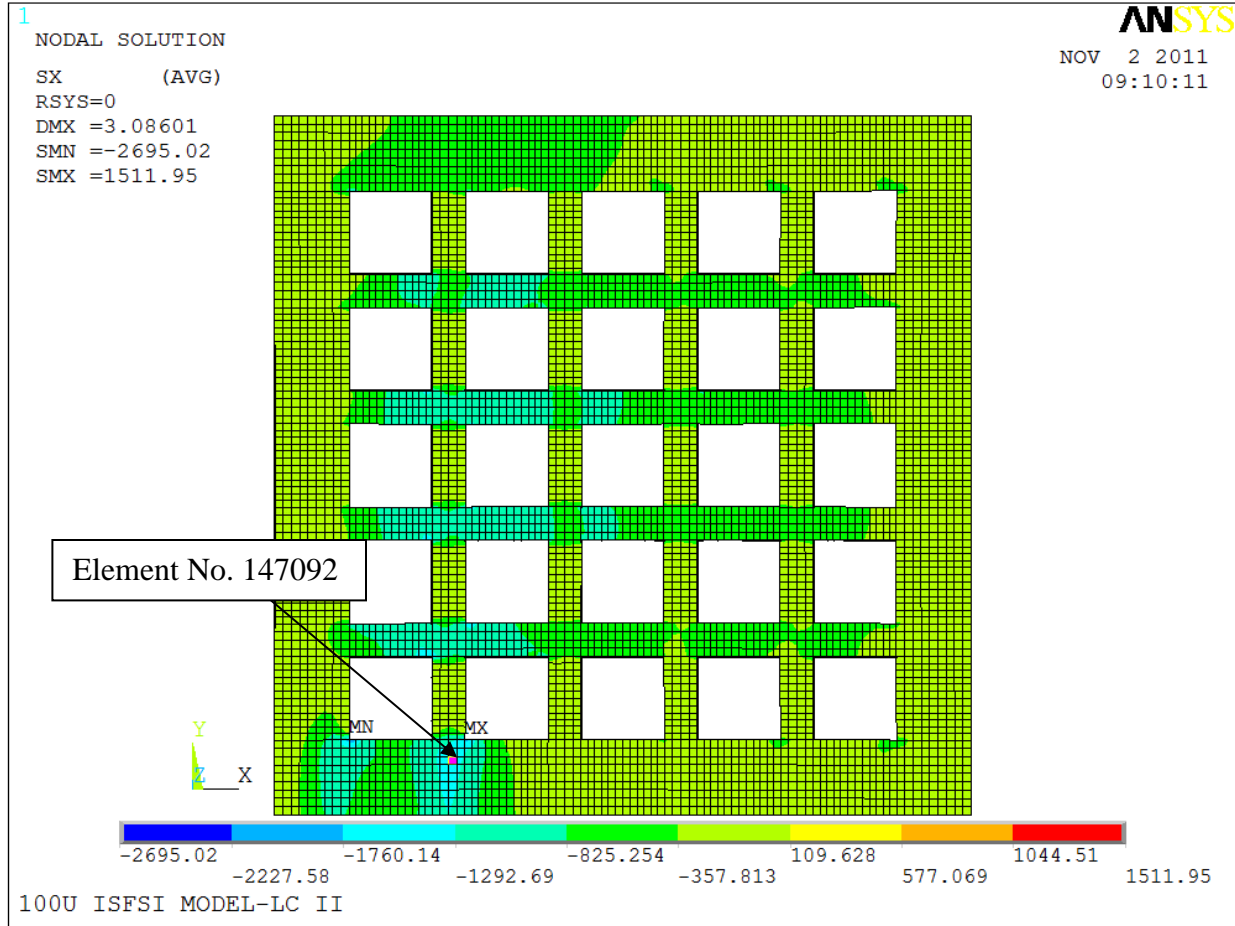


Figure 3.I.19a; Normal Stress in the ISFSI in the Direction of the Transporter Path for Simulation Model II – Load Combination LC-3 from Table 2.I.11



Note: Element 3329 realizes the maximum loading and produces the limiting safety factor.

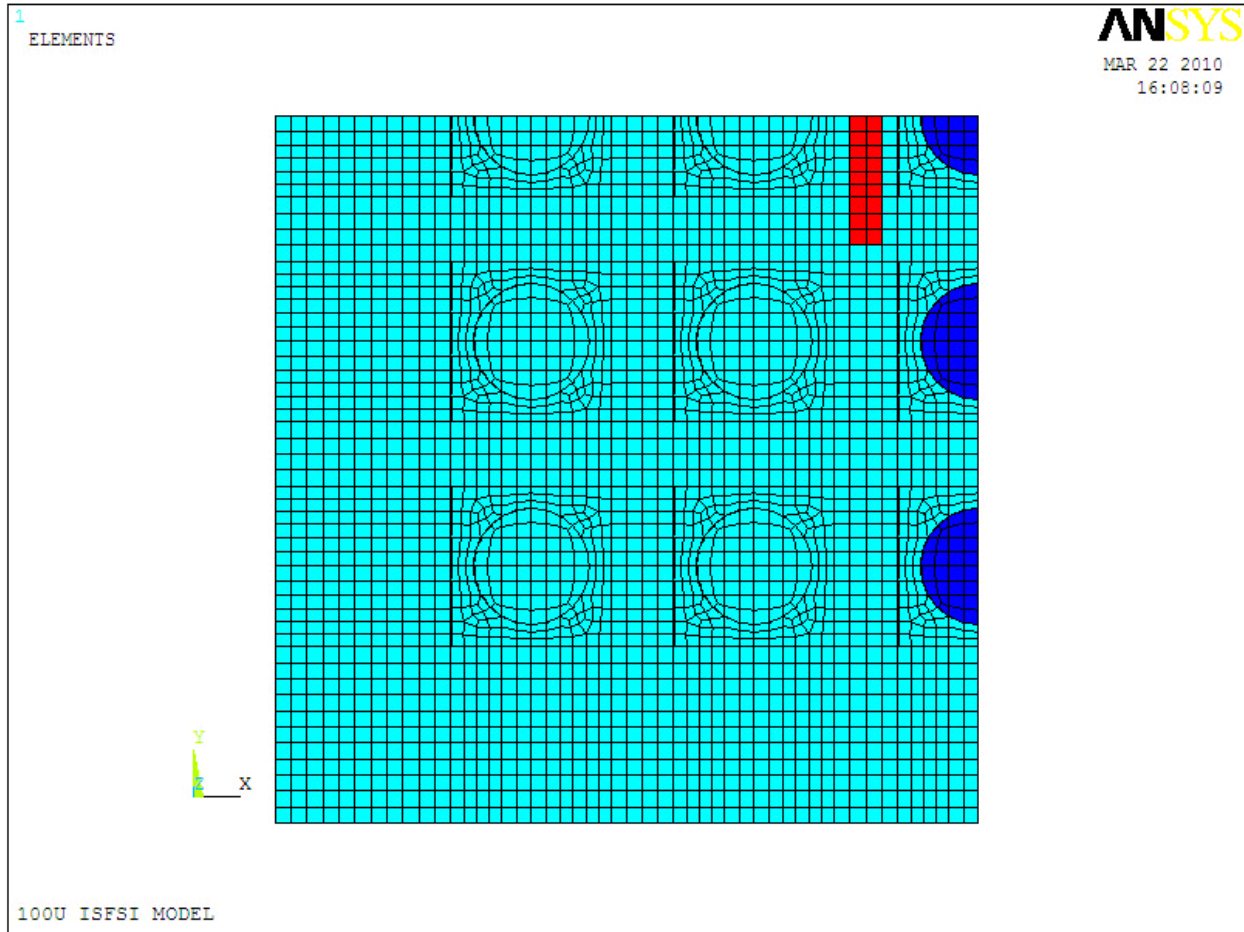
Figure 3.I.19b; Normal Stress ( $S_x$ ) in SFP, Simulation Model II – Load Combination LC-3 from Table 2.I.11



Note: Element 147092 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.19c; Normal Stress ( $S_x$ ) in TSP, Simulation Model II – Load Combination LC-3 from Table 2.I.11





Note: The blue footprints show the loaded VVM locations on the SFP and the red footprint represents the loaded TSP area with the transporter.

Figure 3.I.20; ANSYS Finite Element of ISFSI Showing the Center Row Loading (Simulation Model III)

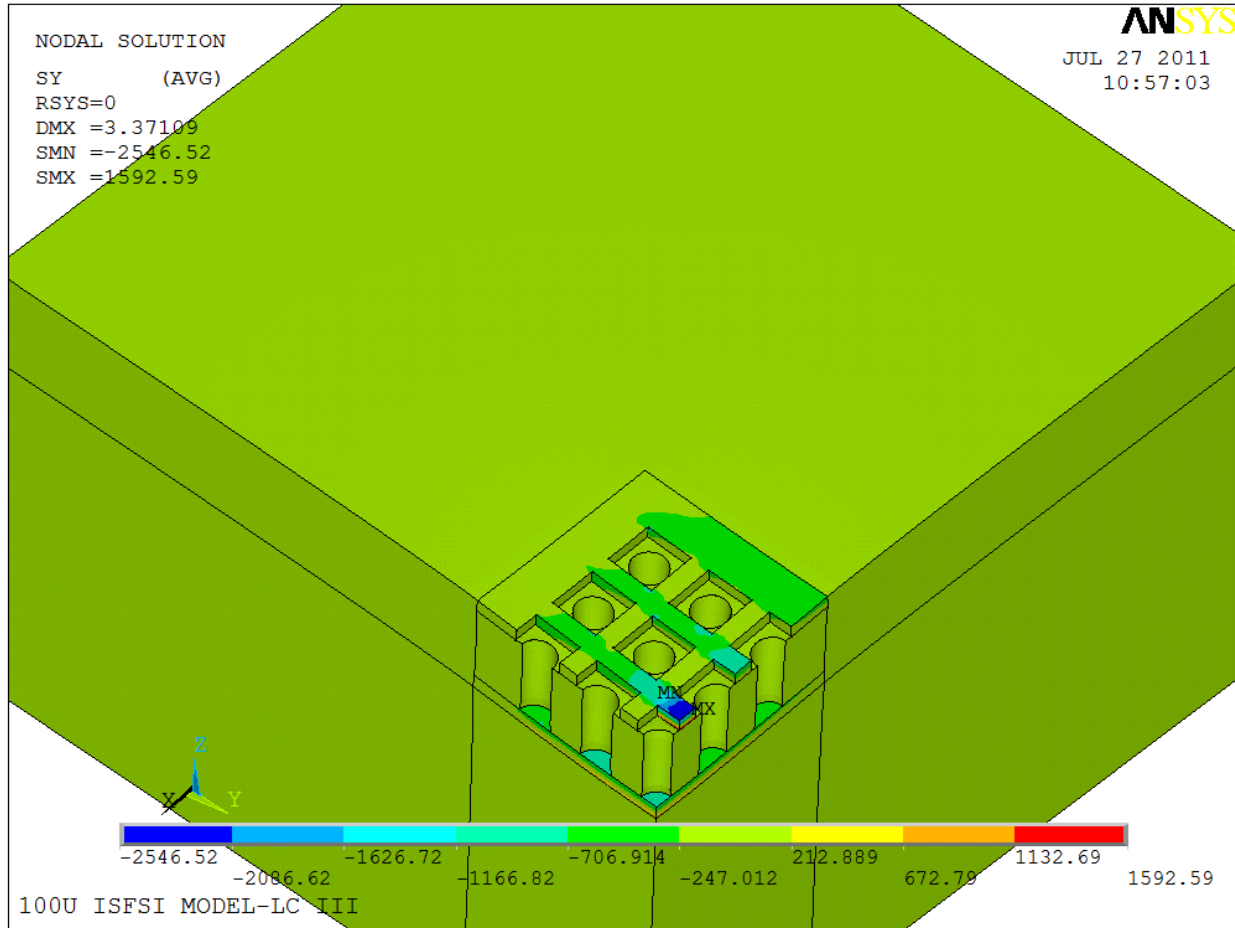
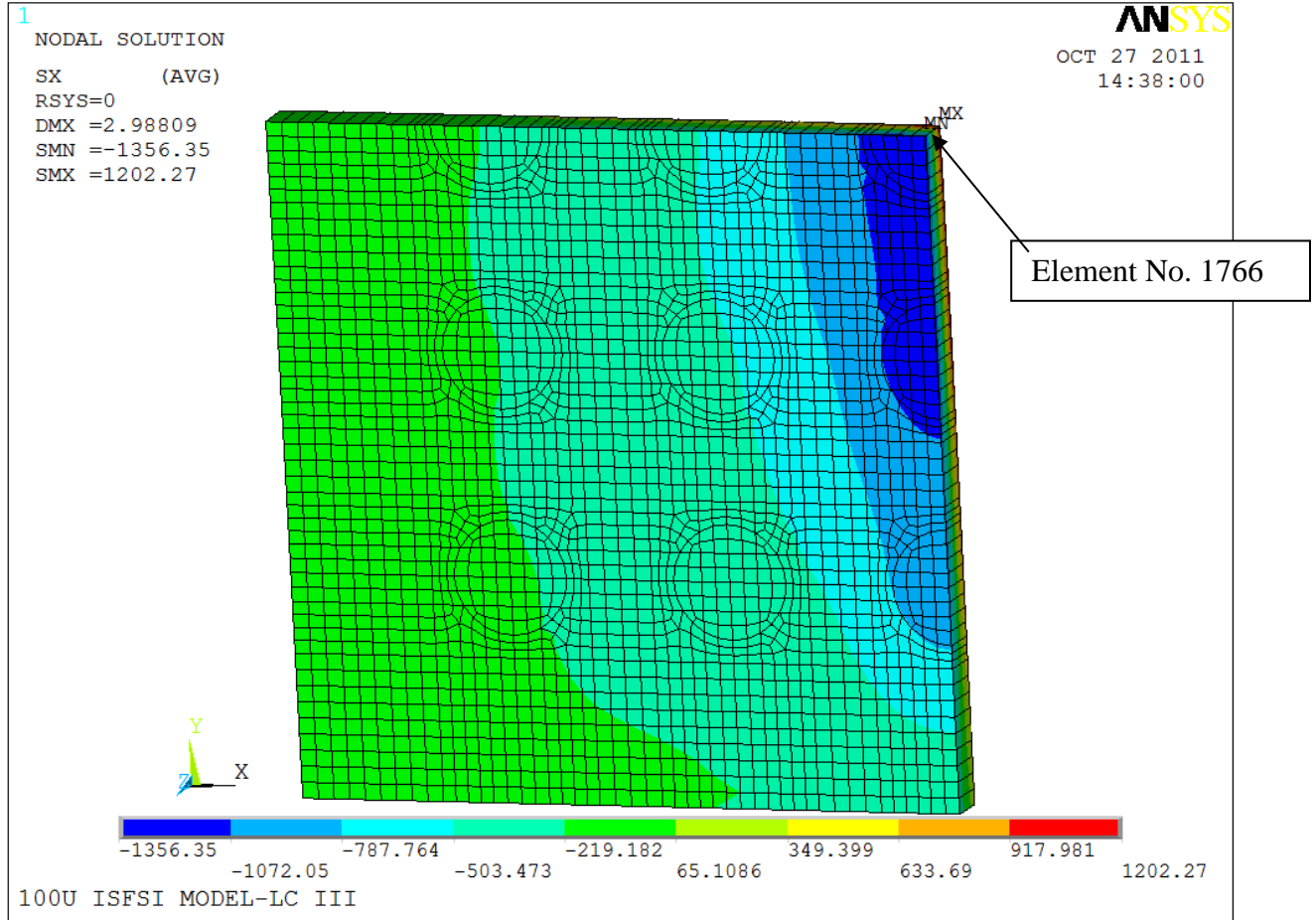


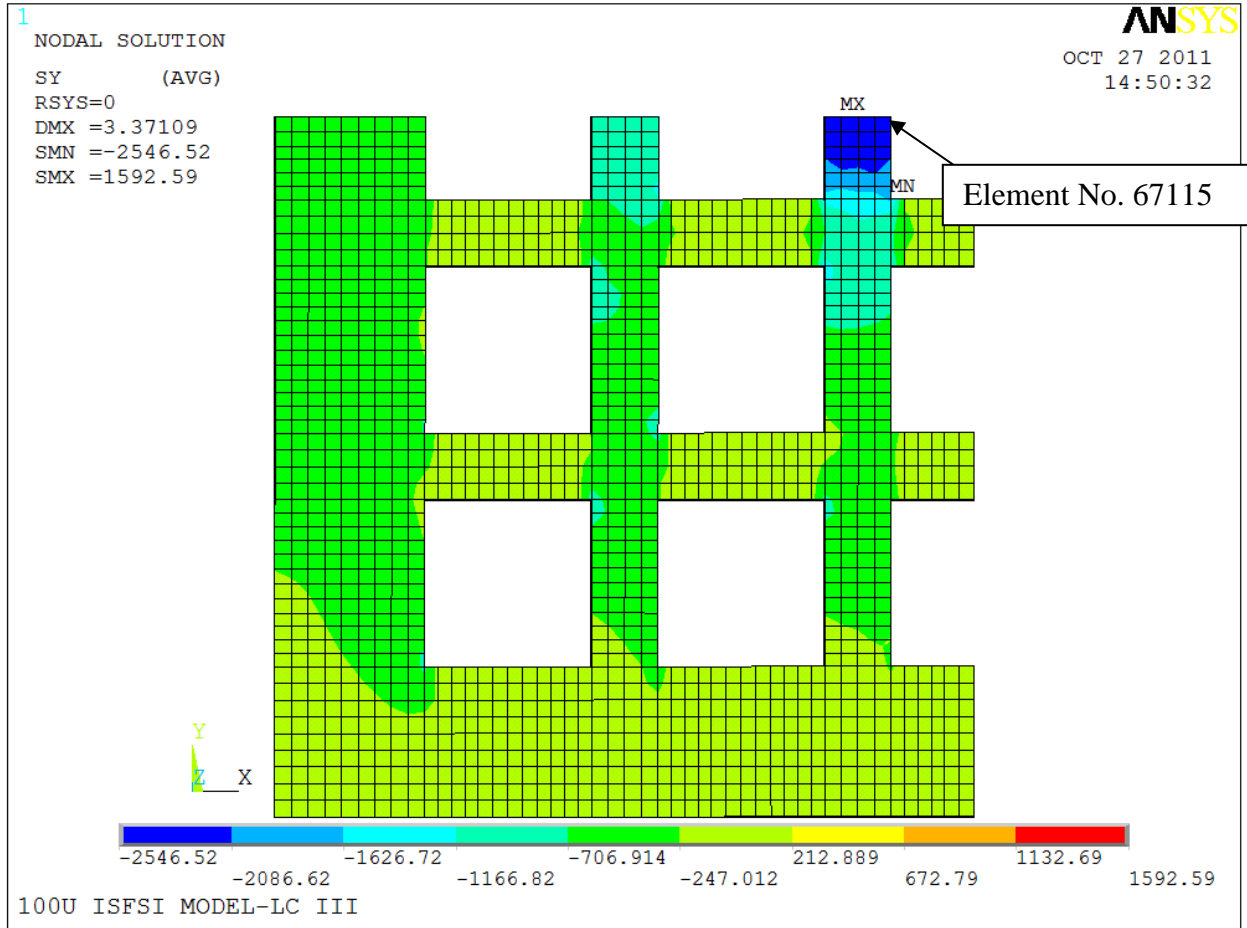
Figure 3.I.20a; Normal Stress in the ISFSI in the Direction of the Transporter Path for Simulation Model III – Load Combination LC-3 from Table 2.I.11





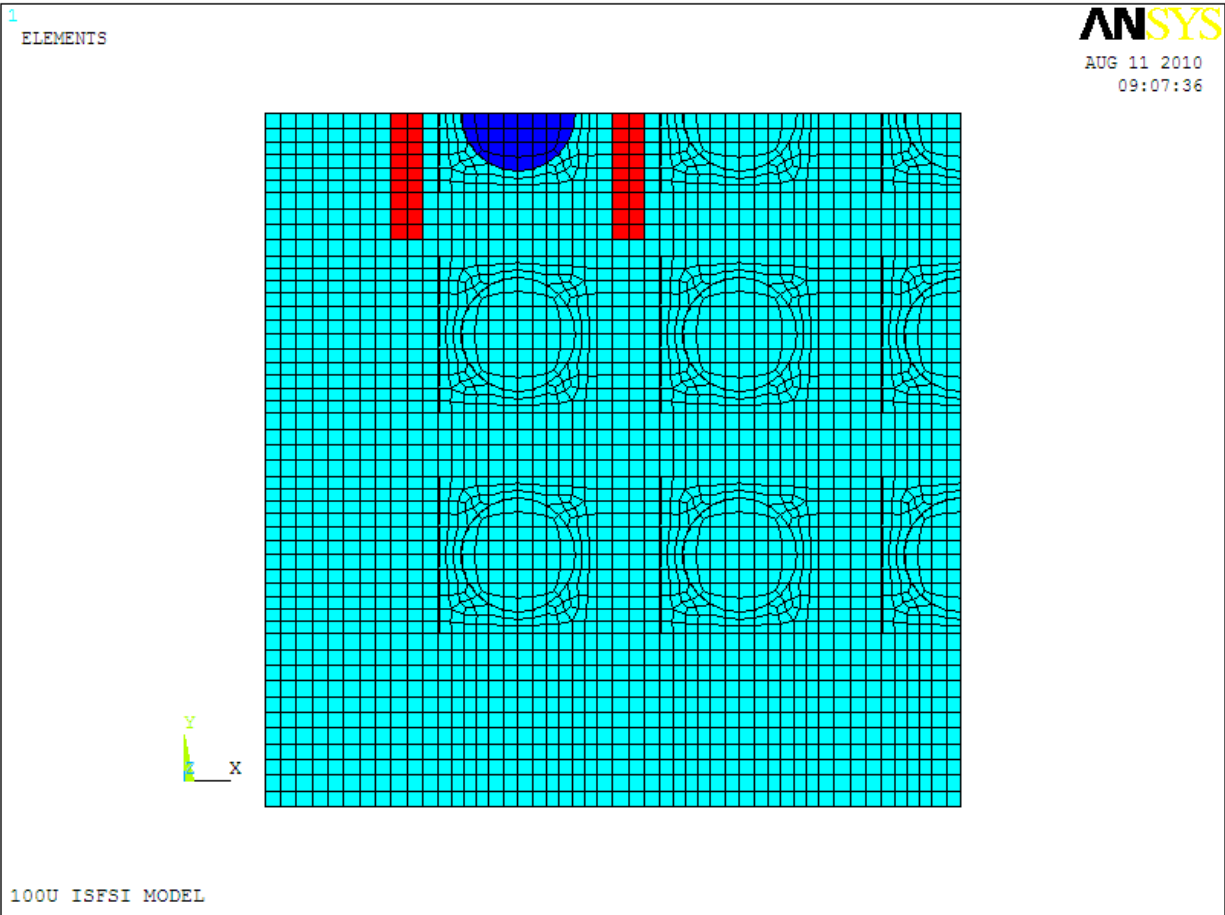
Note: Element 1766 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.20b; Normal Stress (Sx) in SFP, Simulation Model III – Load Combination LC-3 from Table 2.I.11



Note: Element 67115 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.20c; Normal Stress (Sy) in TSP, Simulation Model III – Load Combination LC-3 from Table 2.I.11



Note: The blue footprints show the loaded VVM locations on the SFP and the red footprint represents the loaded TSP area with the transporter.

Figure 3.I.21; ANSYS Finite Element of ISFSI Showing the Single VVM Loaded (Simulation Model IV)

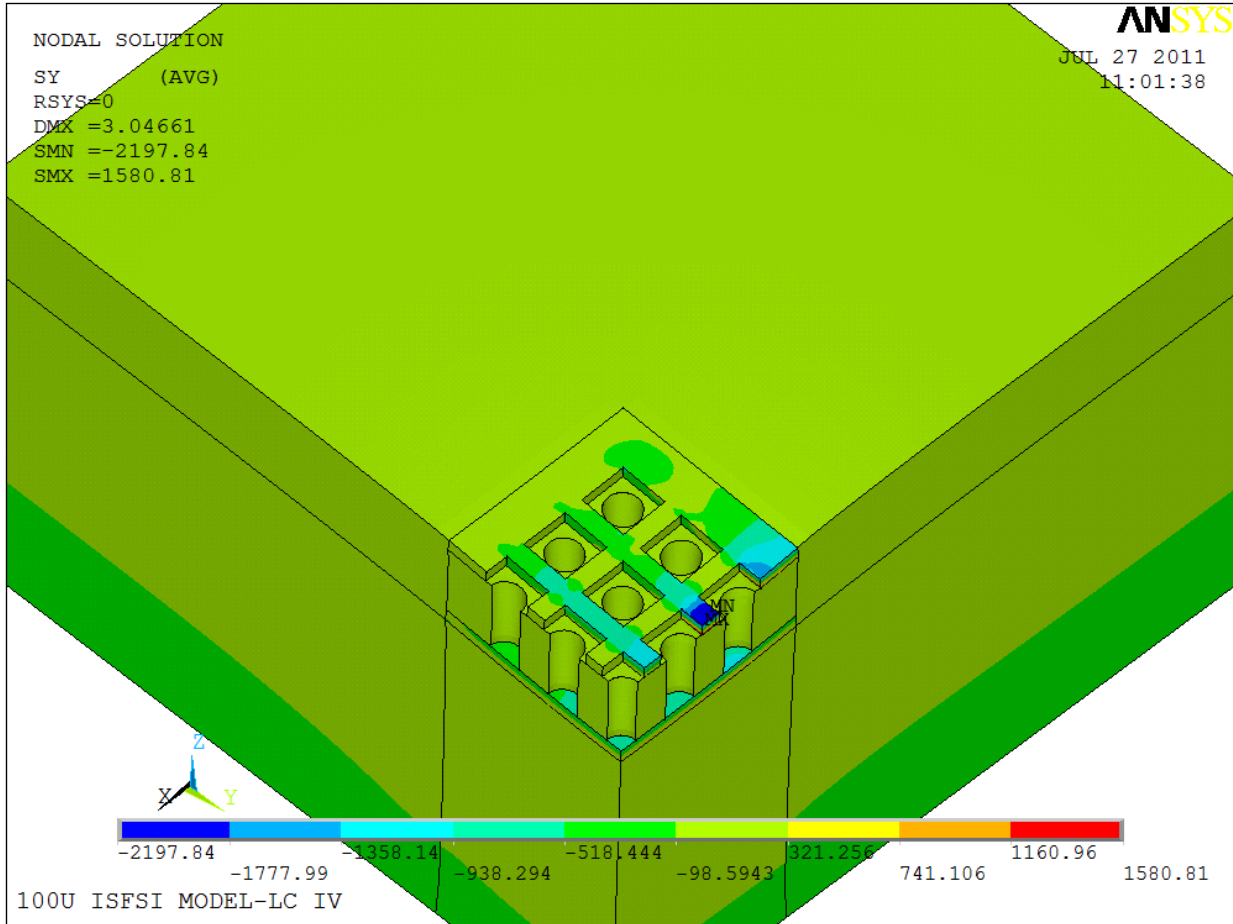
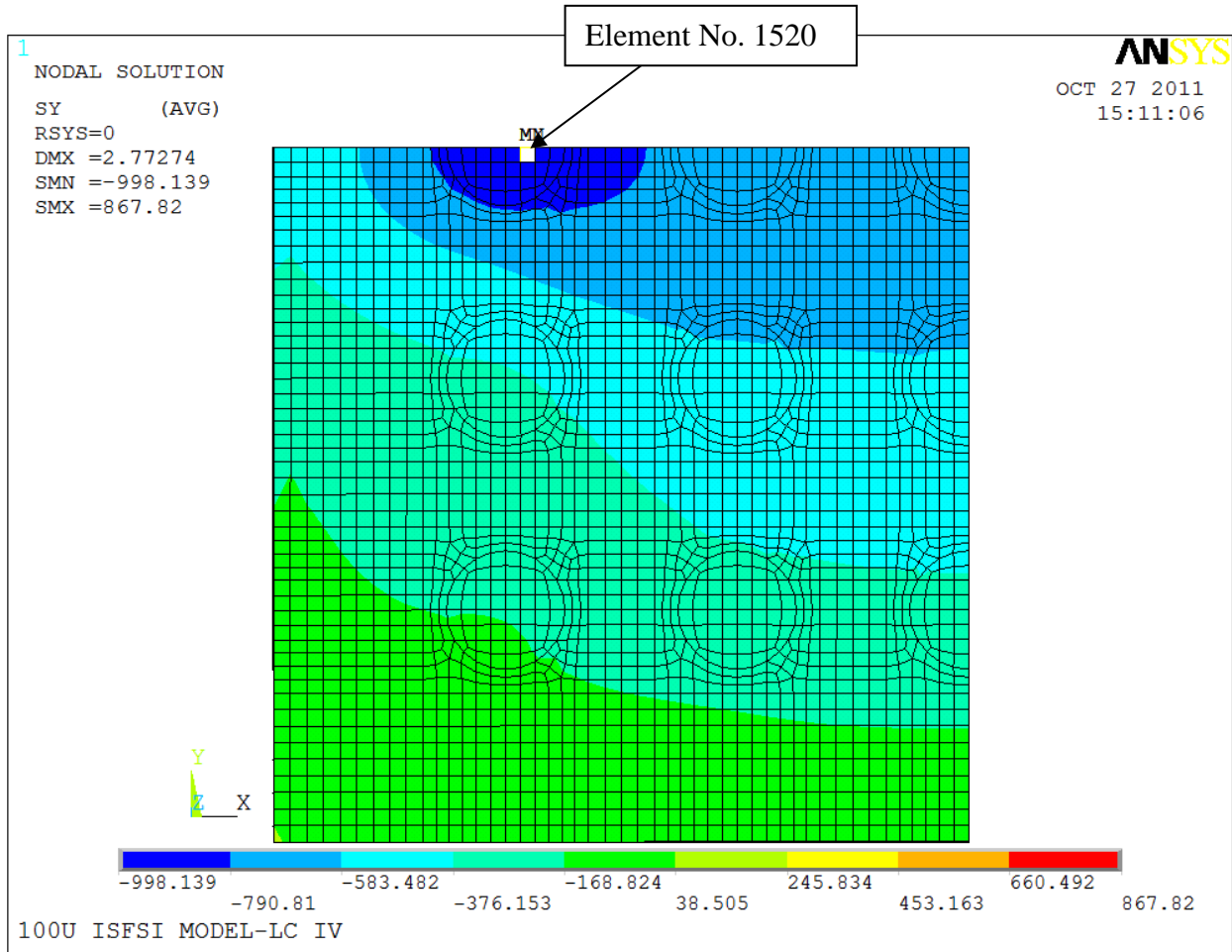
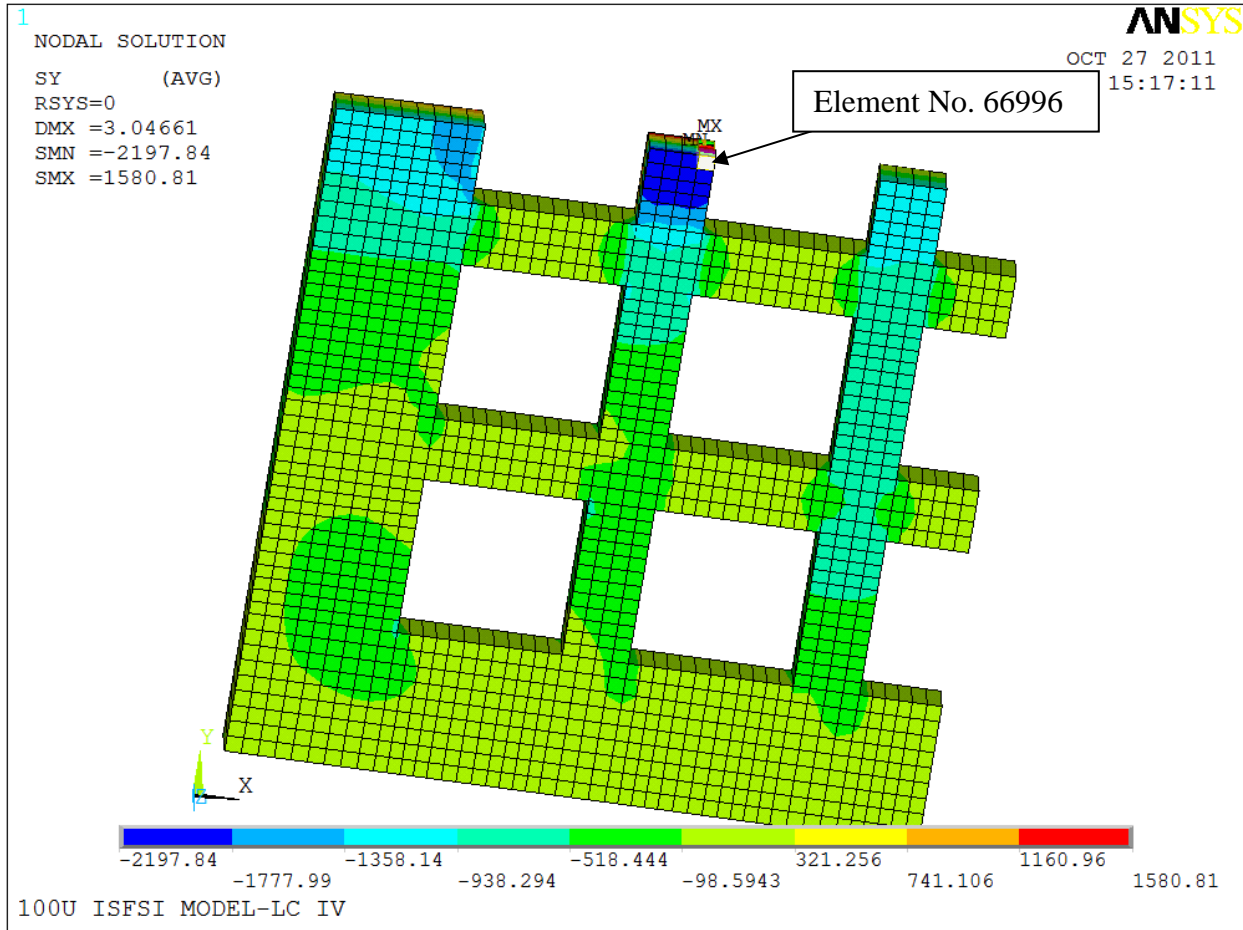


Figure 3.I.21a; Normal Stress in the ISFSI in the Direction of the Transporter Path for Simulation Model IV – Load Combination LC-3 from Table 2.I.11



Note: Element 1520 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.21b; Normal Stress (Sy) in SFP, Simulation Model IV – Load Combination LC-3 from Table 2.I.11



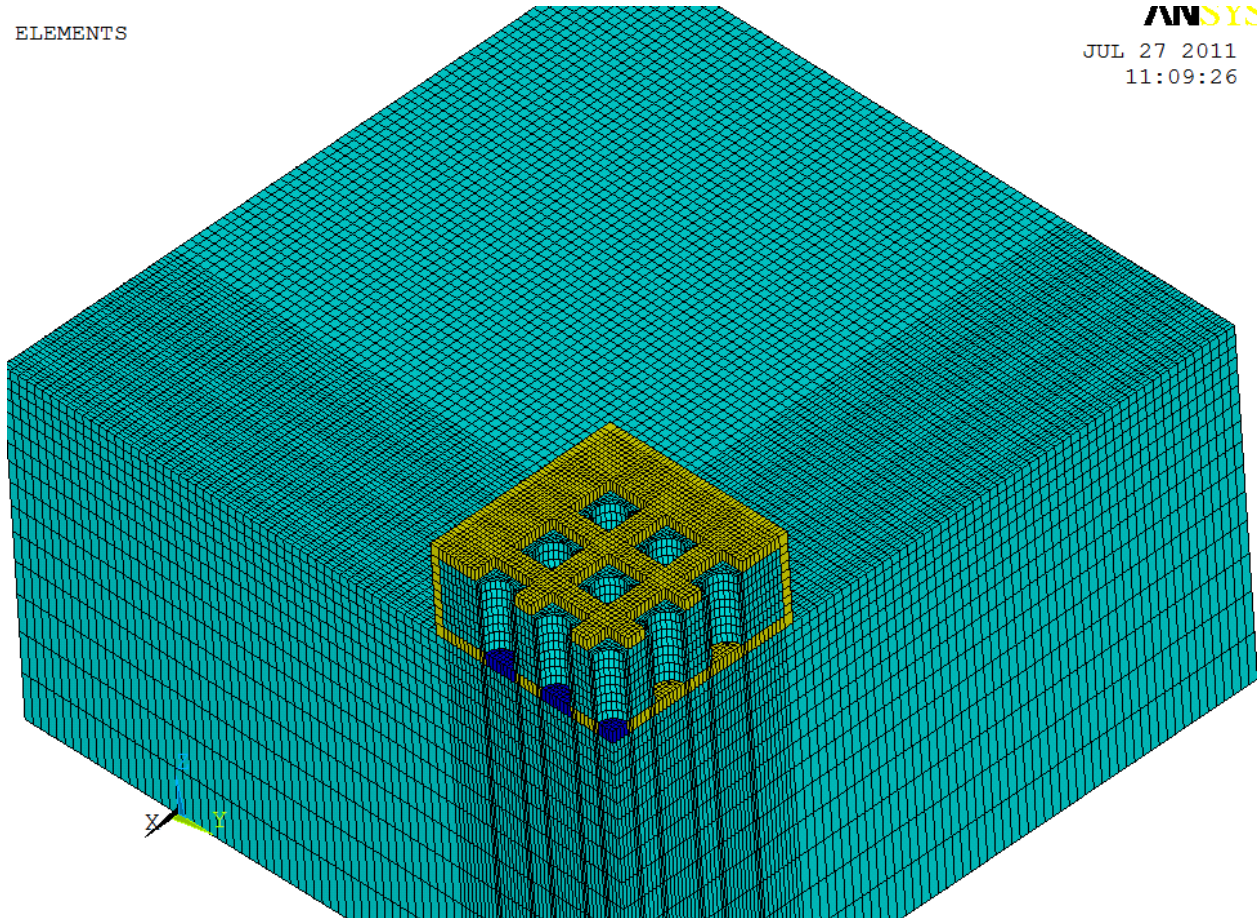
Note: Element 66996 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.21c; Normal Stress (Sy) in TSP, Simulation Model IV – Load Combination LC-3 from Table 2.I.11

ELEMENTS

ANSYS

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Note: The blue footprints show the loaded VVM locations on the SFP and there is no transporter load.

Figure 3.I.22; ANSYS Finite Element of ISFSI with Retaining Wall Optional Design (Simulation Model V)



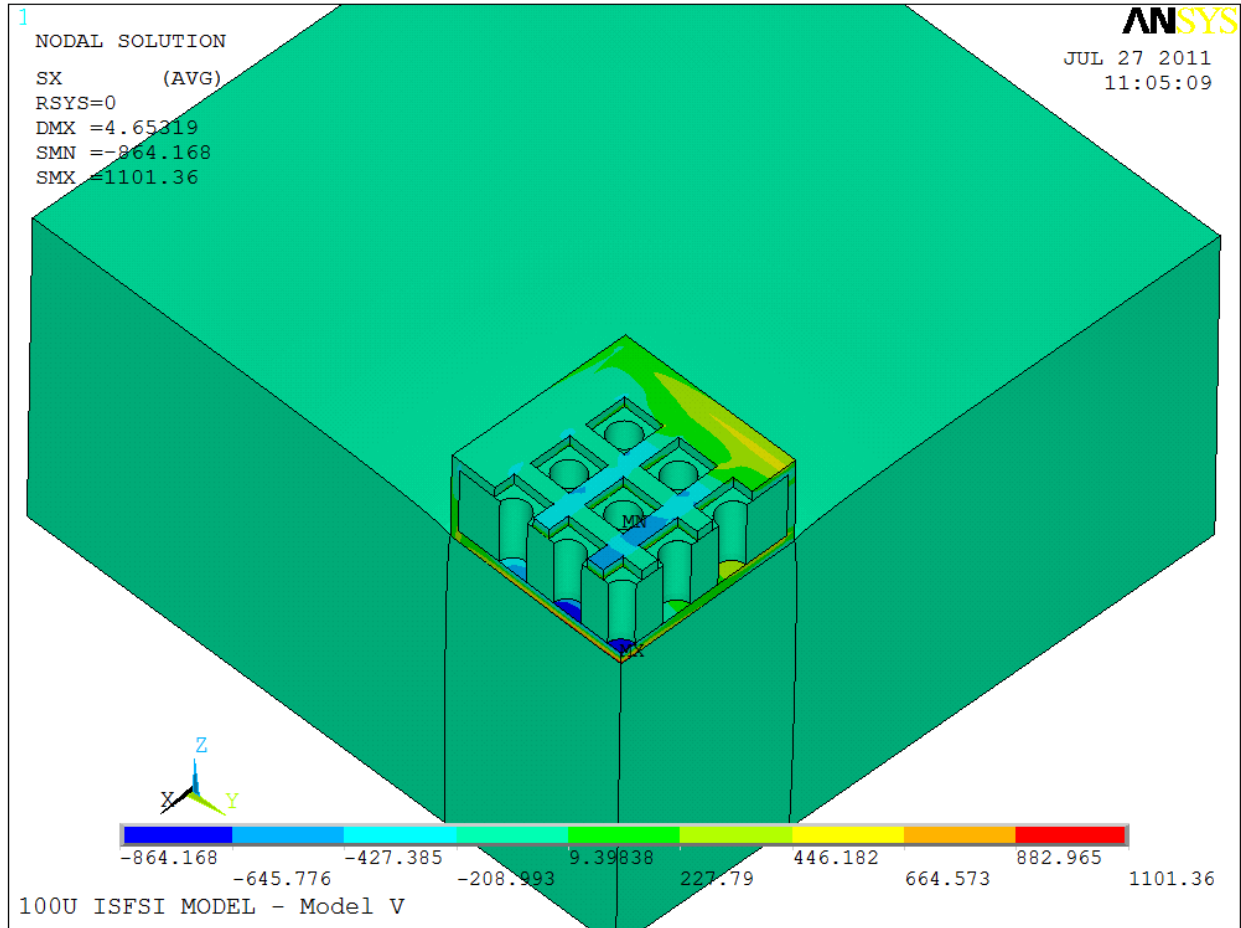
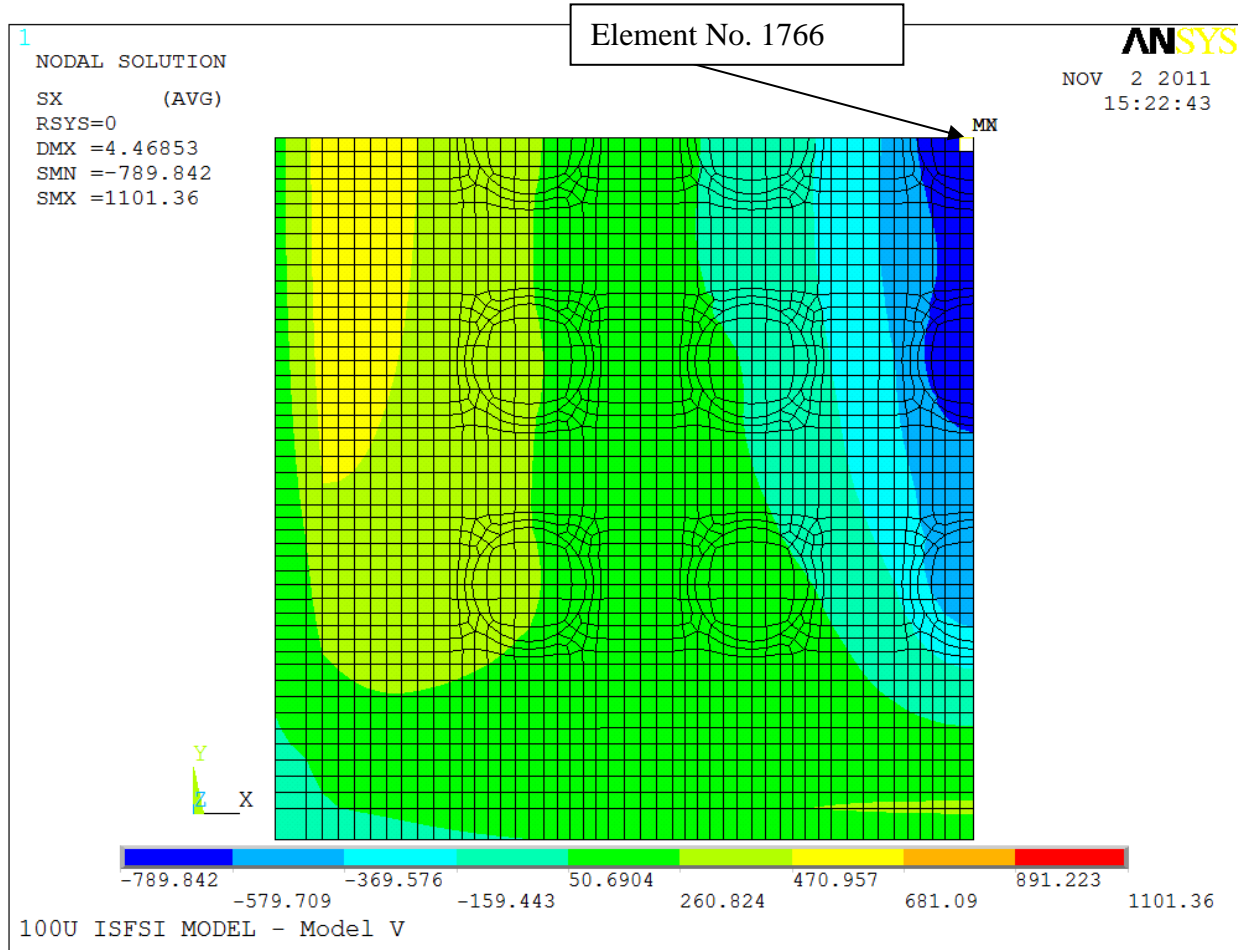


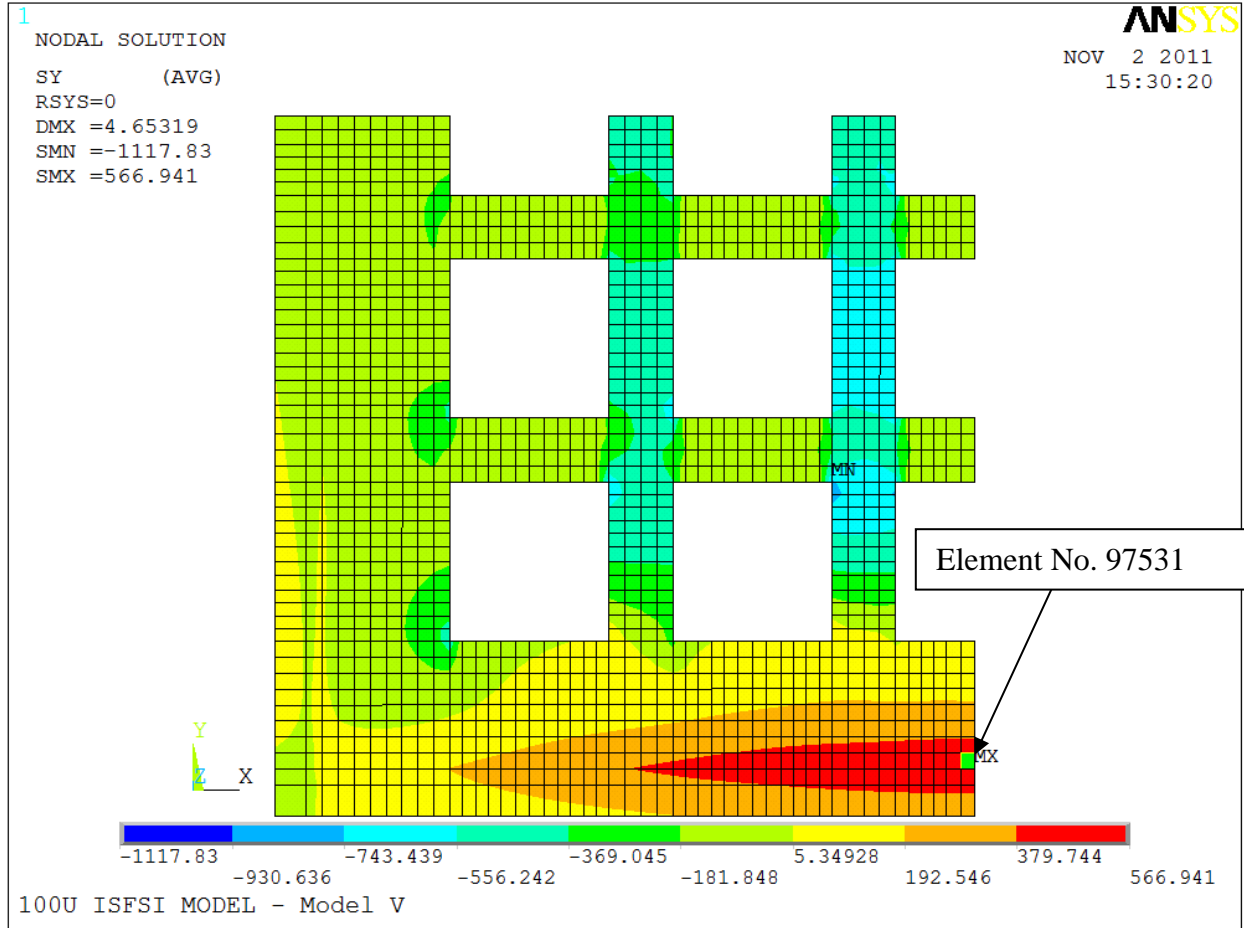
Figure 3.I.22a; Normal Stress in the ISFSI in the Direction for Simulation Model V – Load Combination LC-3 from Table 2.I.11





Note: Element 1766 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.22b; Normal Stress (Sx) in SFP, Simulation Model V – Load Combination LC-3 from Table 2.I.11



Note: Element 97531 realizes the maximum loading and produces the limiting safety factor.

Figure 3.I.22c; Normal Stress (Sy) in TSP, Simulation Model V – Load Combination LC-3 from Table 2.I.11

**SSI ANALYSIS OF HI-STORM 100U**

Time = 20.5

Contours of X-displacement

min=-1.78424, at node# 300310

max=-1.34641, at node# 401100

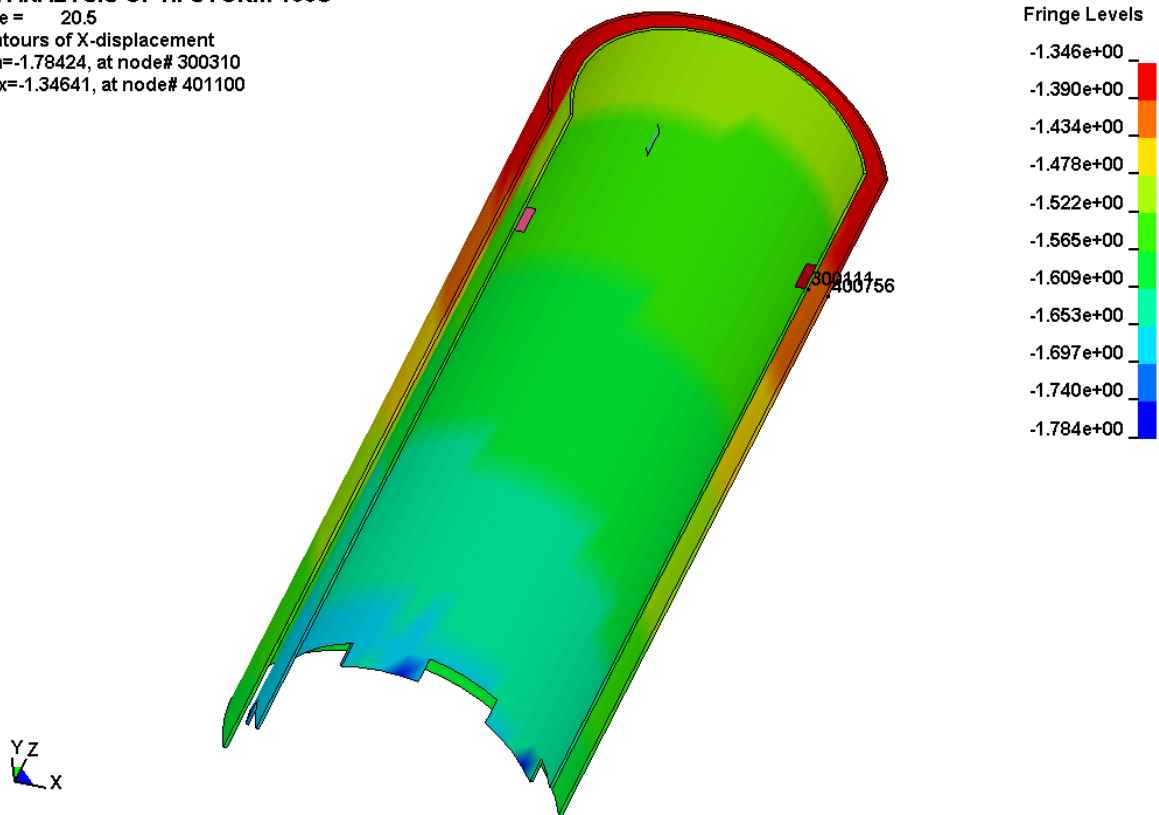


Figure 3.I.23-A; Divider and CEC Shell Displacement Distribution at the End of the Earthquake

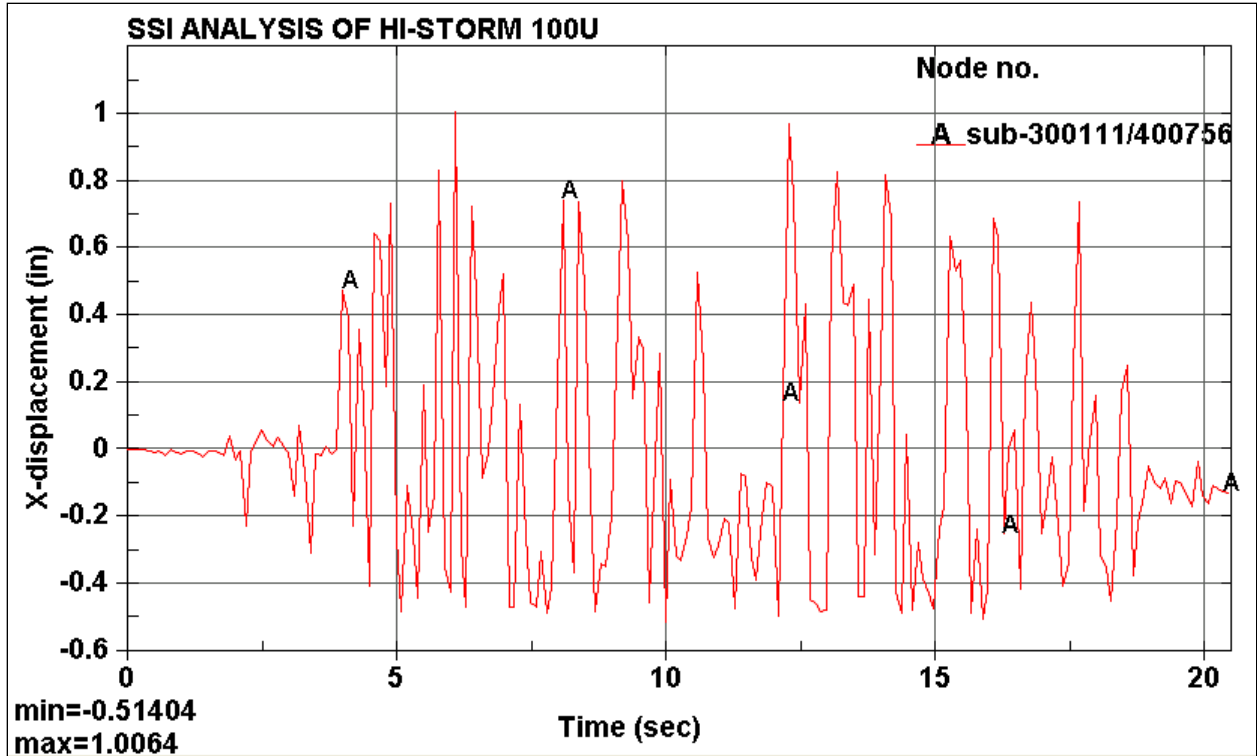


Figure 3.I.23-B; Changes of the Radial Gap between CEC Shell and Divider Shell Measured at the Top Guide Elevation  
(Radial gap change at the end of earthquake = 0.1325 inches)

**SSI ANALYSIS OF HI-STORM 100U**

Time = 6.1

Contours of Maximum Shear Stress

max ipt. value

min=313.138, at elem# 203894

max=17789.9, at elem# 204265

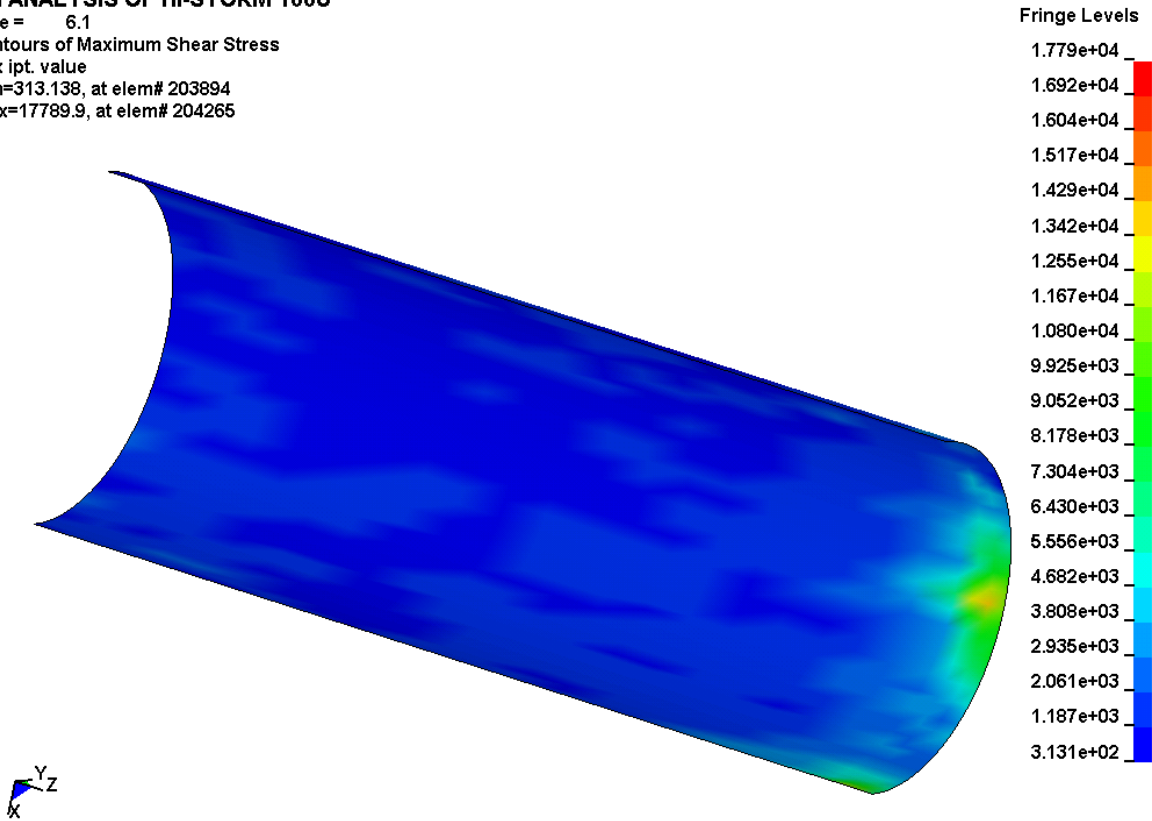


Figure 3.I.24; Maximum Shear Stress of the MPC Shell  
(Maximum Primary Stress Intensity =  $2 \times 6,430$  psi = 12,860 psi)

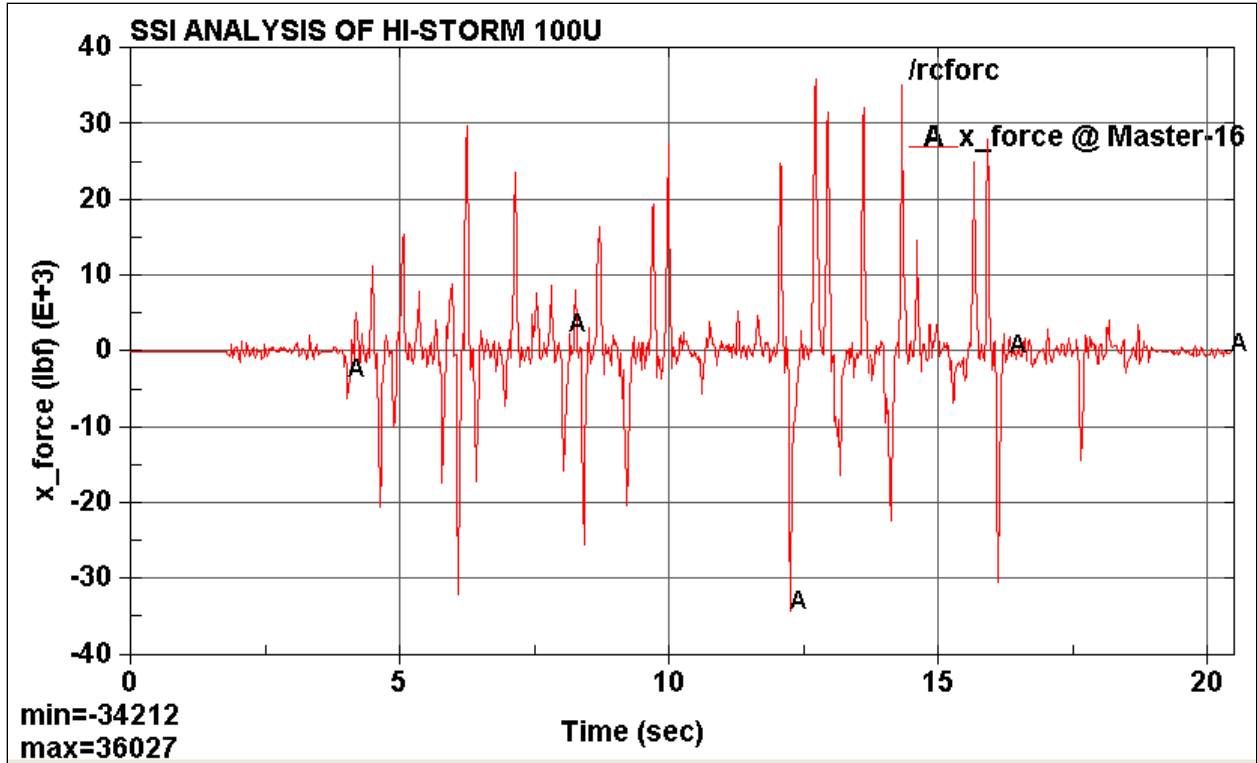


Figure 3.I.25-A; Impact Force between the MPC and MPC Top Guides  
(Maximum Impact Force =  $2 \times 36,027 \text{ lb} = 72,054 \text{ lb}$  to account for half-symmetric model)

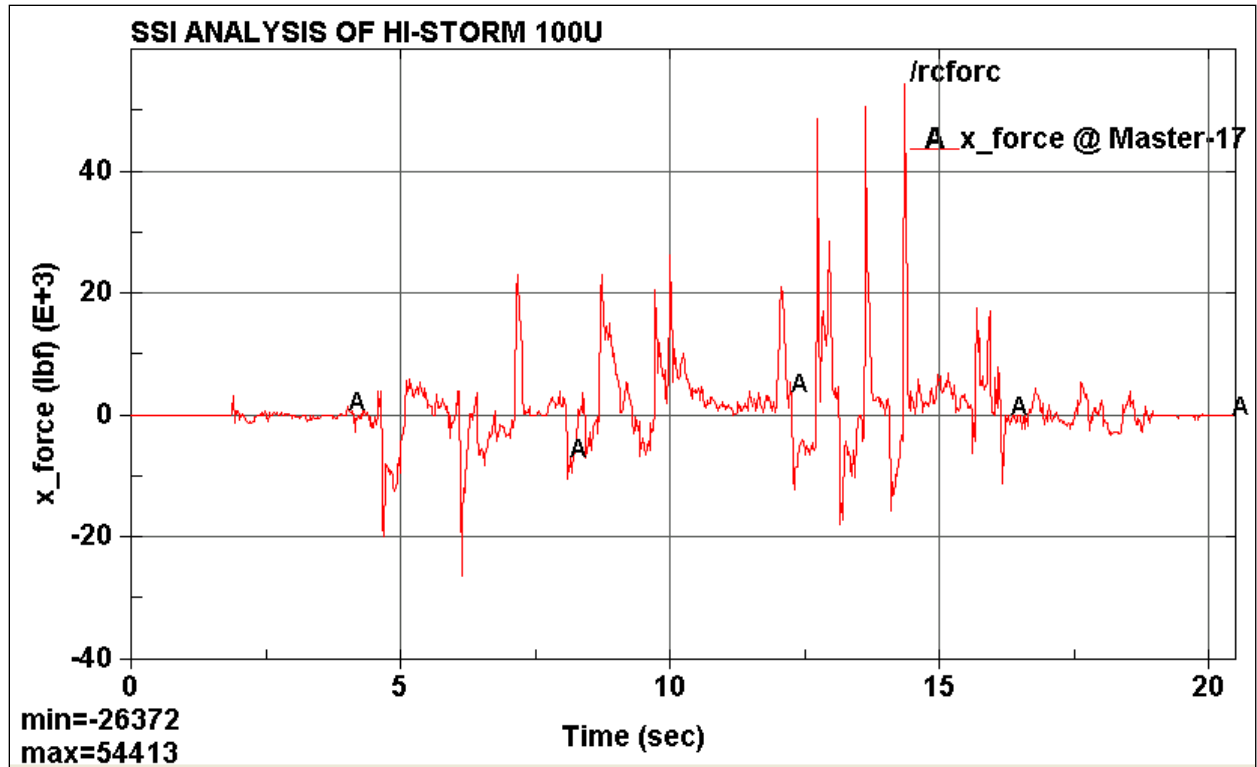


Figure 3.I.25-B; Impact Force between the MPC and MPC Bottom Guides  
(Maximum Impact Force =  $2 \times 54,413 \text{ lb} = 108,826 \text{ lb}$  to account for half-symmetric model)

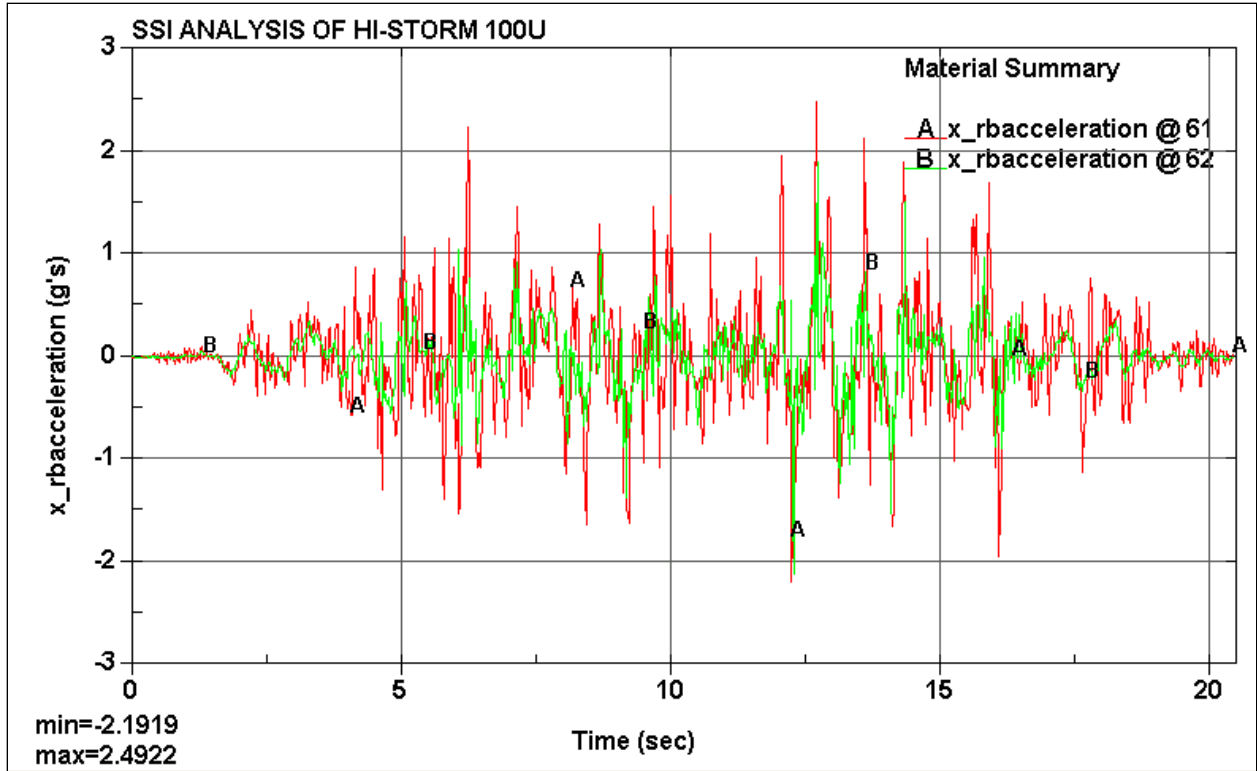


Figure 3.I.26; MPC Lid and Baseplate Lateral Acceleration Time Histories  
(A - MPC Lid; B - MPC Baseplate)



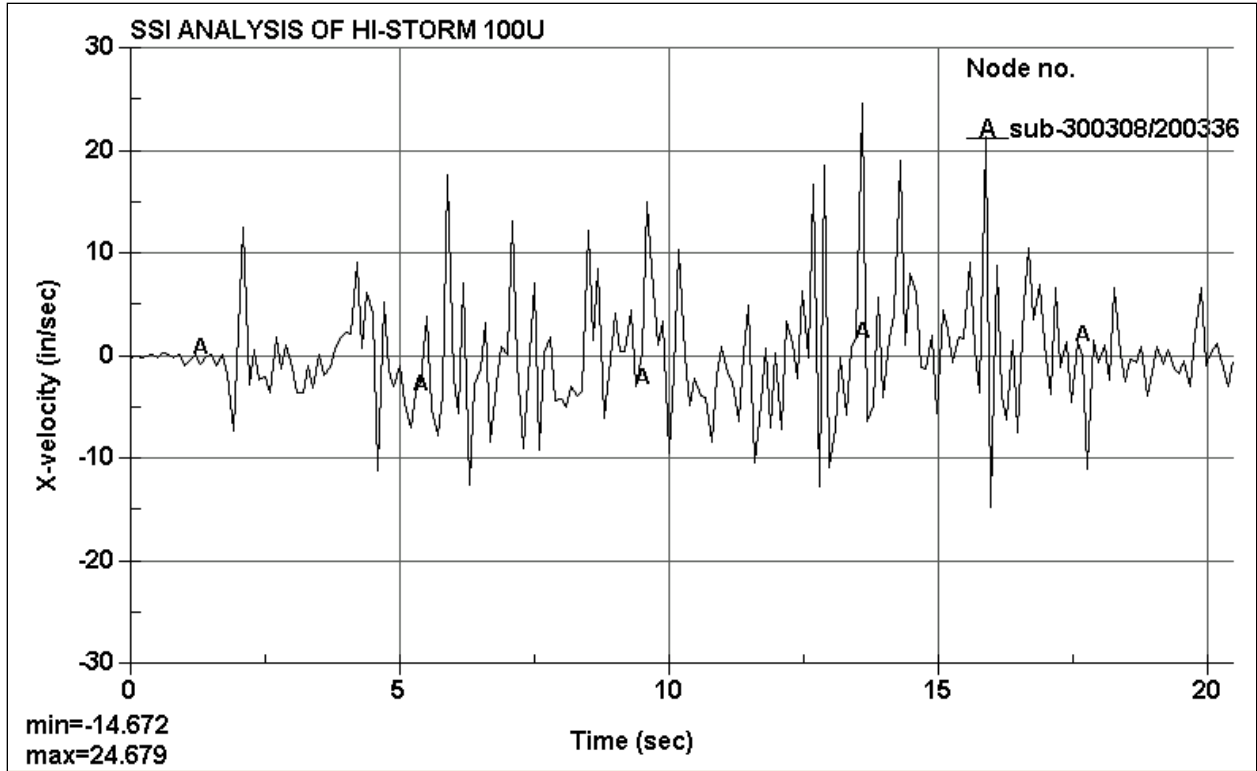


Figure 3.I.27; MPC Lid to MPC Top Guide Approaching Velocity Time History

## SUPPLEMENT 3.III

### STRUCTURAL EVALUATION OF THE MPC-68M

#### 3.III.0 OVERVIEW

In this supplement, the structural adequacy of the MPC-68M is evaluated pursuant to the guidelines of NUREG-1536.

The organization of technical information in this supplement mirrors the format and content of Chapter 3 except that it only contains material directly pertinent to the MPC-68M.

The MPC-68M consists of a stainless steel (Alloy X) Enclosure Vessel, which is identical to that of the MPC-68, a BWR fuel basket made from Metamic-HT, and aluminum basket shims. Section 1.III.2 contains a complete description of the MPC-68M components.

The applicable codes, standards, and practices governing the structural analysis of the MPC-68M as well as the design criteria, are presented in Supplement 2.III. Throughout this supplement, the term “*safety factor*” is defined as the *ratio of the allowable stress (load) or displacement for the applicable load combination to the maximum computed stress (load) or displacement*. Where applicable, bounding safety factors are computed using values that bound the calculated results.

#### 3.III.1 STRUCTURAL DESIGN

##### 3.III.1.1 Discussion

A general discussion of the structural features of the MPC is provided in Subsection 3.1.1, and in general it applies to the MPC-68M with one notable exception. The MPC-68M fuel basket is qualified using a deflection-based acceptance criterion (see Subsection 2.III.0.1) as opposed to a stress-based criterion. The drawings of the MPC-68M fuel basket and MPC Enclosure Vessel are provided in Section 1.5.

##### 3.III.1.2 Design Criteria

Same as in Subsection 3.1.2, including all of its paragraphs, except as modified in Subsection 2.III.0.1 for the MPC-68M fuel basket.

#### 3.III.2 WEIGHTS AND CENTERS OF GRAVITY

Since the weight density of Metamic-HT is significantly less than that of Alloy X, the MPC-68M weighs less than the MPCs listed in Table 3.2.1. The bounding weights for the MPC-68M are provided in Table 3.III.1.

The center of gravity (CG) height of the empty MPC-68M, and various other configurations involving the MPC-68M, is provided in Table 3.III.2.

### 3.III.3 MECHANICAL PROPERTIES OF MATERIALS

The strength properties of Metamic-HT have been characterized through a comprehensive test program, and Minimum Guaranteed Values suitable for structural design are provided in Supplement 1.III.

The fuel basket shims are made of an aluminum alloy (ASTM B221 2219-T851). Representative mechanical properties for the fuel basket shims are tabulated in Table 3.III.3. Strictly speaking, the shim is not a structural material because it does not withstand any tensile loads and is located in a confined space which would prevent its uncontrolled deformation under load. The simulation of the shim in the basket's structural model, however, utilizes its mechanical properties of which only the Yield Strength has a meaningful (but secondary) role. Accordingly, in this FSAR, the nominal value of the Yield Strength specified in Table 3.III.3 herein, is set down as a "critical characteristic" for the shim material. The minimum value of the Yield Strength reported in the material supplier's CoC must be at least 90% of the nominal value in the above referenced table to ensure that the non-mechanistic tip-over analysis will not have to be revisited.

The mechanical properties for all other materials of construction are the same as in Section 3.3 (including all subsections and tables).

### 3.III.4 GENERAL STANDARDS FOR CASKS

#### 3.III.4.1 Chemical and Galvanic Reactions

The materials used in the HI-STORM 100 System are examined in Subsection 3.4.1 to establish that they do not participate in any chemical or galvanic reactions when exposed to the various environments during all normal operating conditions and off-normal and accident events. The only new materials that are introduced in the MPC-68M are Metamic-HT (for the fuel basket) and aluminum (for the basket shims). The environmental compatibility of these materials is examined below.

The MPC-68M is principally constructed of stainless steel shell, aluminum basket shims, and Metamic-HT. Borated aluminum and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from Metamic (classic) and stainless steel materials. Not one case of chemical or galvanic degradation has been found in such fuel racks. This experience provides a sound basis to conclude that chemical and galvanic corrosion of these materials will be negligible. For further protection, both Metamic-HT and aluminum basket shims are installed in the anodized state in the MPC.

Furthermore, galvanic corrosion is not an applicable corrosion mechanism during long-term

storage since the interior of the MPC during normal operation is both inerted with helium and essentially devoid of any moisture while the MPC shell surfaces are expected to be practically free from condensation and gross environmental contaminants. The cleanliness requirements and inspections during fabrication and fuel loading operations also ensure that the MPC has minimal surface debris and impurities.

#### Tests on Metamic-HT

Extensive tests [1.III] have been conducted to establish material properties of Metamic-HT including its corrosion-resistance characteristics. The Metamic-HT specimens were used for corrosion testing in demineralized water and in 2000 ppm boric acid solution. The tests concluded that the Metamic-HT panels will sustain no discernible degradation due to corrosion when subjected to the severe thermal and aqueous environment that exists around a fuel basket during fuel loading or unloading conditions.

#### Aluminum Alloy

Aluminum alloy used in the fuel basket shims are hard anodized to achieve the desired emissivity specified in Supplement 4.III. The anodizing is an electrolytic passivation process used to increase the thickness of the natural oxide layer on the surface of metal parts. Anodizing increases corrosion resistance and wear resistance of the material surface. There is no mechanistic process for the basket shims with hard anodized surface to react with borated water or demineralized water during fuel loading operation. Under the long-term storage condition, the basket shims are exposed to dry and inert helium with no potential for reaction.

Finally, to ensure safe fuel loading operation, the operating procedure described in Chapter 8 provides for the monitoring of hydrogen gas in the area around the MPC lid prior to and during welding or cutting activities. Although the aluminum surfaces (Metamic-HT fuel basket and aluminum basket shims) are anodized, there is still a potential for generation of hydrogen in minute amounts when immersed in spent fuel pool water for an extended period. Accordingly, as a defense-in-depth measure, the lid welding procedure requires purging the space below the MPC lid prior to and during welding or cutting operation to eliminate any potential for formation of any combustible mixture of hydrogen and oxygen. Following the completion of the MPC lid welding and hydrostatic testing, the MPC-68M is drained and dried. After the completion of the drying operation, there is no credible mechanism for any combustible gases to be generated within the MPC-68M.

#### 3.III.4.2                      Positive Closure

Same as in Subsection 3.4.2.

### 3.III.4.3 Lifting Attachment Points

The structural analyses of the lifting attachment points in Subsection 3.4.3 (including all paragraphs) are bounding for the MPC-68M for the following reasons:

- i. the MPC-68M does not require any changes to the HI-STORM overpacks or the HI-TRAC transfer casks for loading operations or long-term storage;
- ii. the MPC-68M utilizes the same MPC Enclosure Vessel design as all MPCs;
- iii. the fully loaded weight of the MPC-68M (Table 3.III.1) is less than bounding MPC weight analyzed in Chapter 3 (Table 3.2.1).

### 3.III.4.4 Heat

The thermal evaluation of the MPC-68M is reported in Supplement 4.III.

#### 3.III.4.4.1 Summary of Pressures and Temperatures

The design pressures and design temperatures listed in Tables 2.2.1 and 2.2.3, respectively, are applicable to the MPC-68M. Temperature limits of MPC-68M fuel basket and basket shim materials are specified in Table 4.III.2.

#### 3.III.4.4.2 Differential Thermal Expansion

The material presented in Supplement 4.III demonstrates that a physical interference between discrete components of the MPC-68M (e.g., fuel basket and enclosure vessel) will not develop due to differential thermal expansion during any operating condition.

#### 3.III.4.4.3 Stress Calculations

The majority of the stress calculations reported in Paragraph 3.4.4.3 are unaffected by or bound the addition of the MPC-68M to the HI-STORM 100 System for the following reasons:

- i. the MPC-68M does not require any changes to the HI-STORM overpacks or the HI-TRAC transfer casks for loading operations or long-term storage;
- ii. the MPC-68M utilizes the same MPC Enclosure Vessel design as all MPCs;
- iii. the fully loaded weight of the MPC-68M (Table 3.III.1) is less than the bounding MPC weight analyzed in Chapter 3 (Table 3.2.1);

Therefore, the stress calculations reported in Paragraph 3.4.4.3 are not repeated here unless material, geometry, or load changes warrant new analysis or discussion. In other words, unless a

new analysis is presented in this subsection, the results in Paragraph 3.4.4.3 for the HI-STORM 100 System are also valid for the MPC-68M either inside the HI-STORM overpack or the HI-TRAC transfer cask.

#### 3.III.4.4.3.1 Analysis of Load Cases F.3.b and F.3.c (Table 3.1.3)

During a non-mechanistic tip-over event, the fuel assemblies exert a lateral force on the fuel basket panels as the overpack impacts the ground and decelerates. The lateral force causes the fuel basket panels to deflect potentially affecting the spacing between stored fuel assemblies. To maintain the fuel in a subcritical configuration, a deflection limit for the fuel basket panels is set in Subsection 2.III.0.1, which is supported by the criticality safety analysis in Supplement 6.III. Here a finite element analysis is performed using ANSYS to demonstrate that the maximum lateral deflection in the fuel basket panels under a bounding deceleration of 60g is less than the limit specified in Section 2.III.0.1. The 60g input deceleration is bounding because it exceeds the design basis deceleration limit of 45g for the non-mechanistic tip over of the HI-STORM storage overpack (see Subsection 3.III.4.10) and for the horizontal drop of the HI-TRAC transfer cask (see Subsection 3.4.9), and is equal to the design basis lateral deceleration limit of 60g for the HI-STAR transport cask [1.1.3] for future considerations. The analysis methodology presented in this subsection is identical to the methodology used in [2.III.6.2] to qualify the F-37 fuel basket.

As shown in Figure 3.III.1, a representative slice of the MPC-68M fuel basket, consisting of a smaller end section and a full section, is modeled in detail including the contained fuel assemblies and supporting basket shims. The fuel basket panels are modeled with SOLSH190 solid shell elements. The basket shims and each fuel assembly are modeled with SOLID45 solid elements. The mass density assigned to the fuel assemblies corresponds to the maximum BWR fuel assembly weight per Table 2.1.22, except at the 16 cell locations along the basket perimeter where Damaged Fuel Containers are permitted. At these 16 locations, the mass density corresponds to the maximum weight of a BWR fuel assembly plus DFC per Table 2.1.22. Standard contact pairs using CONTA173/TARGE170 elements are defined at the interfaces of fuel assembly/basket panel, shim/basket panel, and between stacked basket panels including all the intersecting slot locations. At the perimeter corners, the intersecting basket panels are bonded together in the finite element model, and the strength properties of the corner most elements are then adjusted depending on whether there is a full length weld at that location. At corner locations that are not welded full length (see licensing drawing in Section 1.5), the elastic modulus of the corner elements is reduced to 1% of the MGW in Table 1.III.2 to effectively eliminate the joint's shear and moment carrying capacity. The fuel basket material model is implemented with true stress-true strain multi-linear isotropic hardening plasticity model. An elastic material model is used for the basket shims since no plastic deformation is expected. To accommodate large plastic deformation in the fuel basket panels, sufficiently small element sizes ( $< 0.40$  in) are used and 9 integration points through the thickness are specified. A sensitivity study was performed in [2.III.6.2] to confirm that the panel stresses and displacements obtained using solid shell elements are converged and comparable to those obtained using 5 solid elements through the thickness of the panel.

The 60g deceleration is applied to the model with the basket in the so-called 0° orientation (see Figure 3.III.5). This orientation is chosen for analysis because it maximizes the lateral load on a single basket panel, which in turn maximizes the lateral deflection of the panel. In the 0° orientation, the amplified weight of each stored fuel assembly (during the 60g impact event) bears entirely on one basket panel. Conversely, in the 45° orientation, the amplified weight of each stored fuel assembly is equally supported by two basket panels. The difference in loading between these two basket orientations is pictorially shown in Figure 3.III.5, where “m” denotes the fuel assembly mass, “a” denotes the maximum lateral deceleration, and “d” denotes the enveloping size of the fuel assembly. For comparison purposes, the pressure loads on the basket panels are defined as “p” and “q”, respectively, for the 0° and 45° orientations. From the figure, the pressure load p that develops in the 0° orientation is 41% greater than the pressure load q that develops in the 45° orientation. Hence, the lateral deflection of a basket panel is much greater for the 0° orientation (which is why it is chosen for detailed analysis). It is also noted that the 90° corners where the basket panels intersect do not provide any additional moment resistance because of the slotted joint construction (see Figure 1.III.1); therefore, the 45° orientation (or any other orientation between 0° and 45°) does not give rise to any prying loads at the cell corners. Finally, to ensure that the analysis for the 0° orientation is conservative and bounds all other basket orientations, the analysis is performed based on a lateral impact deceleration of 60g even though, according to the results presented in Section 3.III.4.10, the maximum impact deceleration due to the non-mechanistic tip over event (measured at the top of the overpack lid) is less than 45g.

The stress and strain distributions in the fuel basket panels at 60g are shown in Figures 3.III.2 and 3.III.3, respectively. These figures show that the state of stress in the fuel basket panels is primarily elastic. The fuel basket displacements are plotted in Figure 3.III.4. Table 3.III.4 compares the maximum lateral displacement in a fuel basket panel (relative to its end supports) with the deflection limit specified in Subsection 2.III.0.1.

Per the licensing drawing, the nominal width of fuel basket panels in the vertical direction may be increased or decreased provided that the length of the panel slots is increased or decreased proportionally. This means that the fixed-height fuel basket may be assembled using more (or fewer) panels than the number depicted on the licensing drawing. The results of the ANSYS static analysis for the fuel basket presented herein are valid for any panel width since (a) the lateral load on the fuel basket per unit (vertical) length remains the same and (b) the length of the slots measured as a percentage of the panel width remains the same.

Finally, to evaluate the potential for crack propagation and growth for the MPC-68M fuel basket under the non-mechanistic tipover event, a bounding crack propagation analysis is carried out in Attachment D of [1.III.A.3]. The analysis demonstrates that a through-thickness linear flaw measuring 1/32 inch in length (i.e., maximum undetectable flaw size per inspection criteria) remains stable under the most severe accident loading conditions.

### 3.III.4.4.3.2 Elastic Stability and Yielding of the MPC-68M Fuel Basket under Compression Loads (Load Case F3 in Table 3.1.3)

Under certain conditions, the fuel basket plates may be under direct compressive load. Although the finite element simulations can predict the onset of an instability and post-instability behavior, the computation in this subsection uses (the more conservative) classical instability formulations to demonstrate that an elastic instability of the basket plates is not credible.

A solution for the stability of the fuel basket plate is obtained using the classical formula for buckling of a wide bar [3.III.1]. Material properties are selected corresponding to a metal temperature of 375°C, which bounds the computed metal temperatures anywhere in the fuel basket (see Table 4.III.3). The critical buckling stress for a pin-ended bar is:

$$\sigma_{cr} = \left( \pi \right)^2 \frac{E}{12(1-\nu^2)} \left( \frac{h}{a} \right)^2$$

where h is the plate thickness, a is the unsupported plate length, E is the Young's Modulus of Metamic-HT at 375°C,  $\nu$  is Poisson's Ratio (use 0.3 for this calculation)

From the drawings in Section 1.5, h = 0.40 in, a = 6.05 in, and E = 6,125 ksi (Table 1.III.2). Then, the classical critical buckling stress is computed as 24.199 ksi, which exceeds the yield strength of the material (9.425 ksi) at 375°C. This demonstrates that basket plate instability by elastic buckling is not possible.

### 3.III.4.5 Cold

Same as in Subsection 3.4.5.

### 3.III.4.6 HI-STORM 100 Kinematic Stability under Flood Condition (Load Case A in Table 3.1.1)

The stability evaluation of the HI-STORM 100 overpack under flood conditions in Subsection 3.4.6 bounds the scenario of a loaded MPC-68M inside a HI-STORM overpack. The previous analysis is bounding because it uses as input the empty weight of the HI-STORM overpack (i.e., no MPC inside) combined with the maximum CG height from Table 3.2.3.

### 3.III.4.7 Seismic Event and Explosion

Since there are no physical changes to the HI-STORM overpacks and the MPC-68M reduces the CG height of the loaded HI-STORM overpacks, relative to those analyzed in Chapter 3, the seismic event and explosion analyses presented in Subsection 3.4.7 (including all paragraphs) bound the scenario of a loaded MPC-68M inside a HI-STORM overpack.



#### 3.III.4.8 Tornado Wind and Missile Impact (Load Case B in Table 3.1.1 and Load Case 04 in Table 3.1.5)

The results for the post-impact response of the HI-STORM 100 overpack in Subsection 3.4.8 for the combination of tornado missile plus either steady tornado wind or instantaneous tornado pressure drop bound the results for a loaded MPC-68M inside a HI-STORM overpack. The results are bounding because they are calculated assuming a lower bound weight for the loaded HI-STORM and an upper bound CG height (as compared to a loaded MPC-68M inside a HI-STORM).

In addition, since the MPC-68M does not require any physical changes to the HI-STORM overpacks or the HI-TRAC transfer casks for MPC loading, the missile penetration analyses presented in Subsection 3.4.8 remain valid.

#### 3.III.4.9 HI-TRAC Drop Events

The HI-TRAC drop analyses presented in Subsection 3.4.9 (including all paragraphs) are valid for a loaded MPC-68M inside a HI-TRAC for the following reasons:

- i. the MPC-68M does not require any changes to the HI-TRAC transfer casks for MPC loading;
- ii. the MPC and its contents are modeled as a solid body (i.e., no explicit modeling of MPC fuel basket);
- iii. the difference in weight between a fully loaded MPC-68M and the MPC analyzed in Subsection 3.4.9 is less than 5% of the total drop weight.

#### 3.III.4.10 HI-STORM 100 Non-Mechanistic Tip-over and Vertical Drop Event (Load Cases 02.a and 02.c in Table 3.1.5)

Pursuant to the provision in NUREG-1536, a non-mechanistic tip-over of a loaded MPC-68M inside a HI-STORM overpack on to the ISFSI pad is considered in this supplement. Calculations are also performed to determine the maximum vertical carry height limit such that the deceleration sustained by a vertical free fall of a HI-STORM overpack carrying a loaded MPC-68M onto the ISFSI pad is less than design basis deceleration limit specified in Table 3.1.2.

The tip-over analysis performed in Appendix 3.A is based on the HI-STORM 100 geometry and a bounding weight. Since the MPC-68M has a slightly higher center of gravity and weighs less than the MPC modeled in Appendix 3.A, it is not a foregone conclusion that the maximum rigid body deceleration level is, in fact, reduced if a HI-STORM 100, with a loaded MPC-68M inside, suffers a non-mechanistic tip-over onto the identical target. In what follows, we present a summary of the analysis undertaken to demonstrate conclusively that the result for maximum deceleration level is less than design basis deceleration limit specified in Table 3.1.2 when the MPC-68M is stored

inside the HI-STORM 100 overpack. The analysis employs the methodology previously established in Subsection 3.4.10 for analyzing the HI-STORM 100S overpack.

Appendix 3.A presents a result for the angular velocity of the cylindrical body representing a HI-STORM 100 just prior to impact with the defined target. The result is expressed in Subsection 3.A.6 in terms of the cask geometry, and the ratio of the mass divided by the mass moment of inertia about the corner point that serves as the rotation origin. Since the mass moment of inertia is also linearly related to the mass, the angular velocity at the instant just prior to target contact is independent of the cask mass. Subsequent to target impact, we investigate post-impact response by considering the cask as a cylinder rotating into a target that provides a resistance force that varies linearly with distance from the rotation point. We measure “time” as starting at the instant of impact, and develop a one-degree-of freedom equation for the post-impact response (for the rotation angle into the target) as:

$$\ddot{\theta} + \omega^2\theta = 0$$

where

$$\omega^2 = \frac{kL^3}{3I_A}$$

The initial conditions at time zero are: the initial angle is zero and the initial angular velocity is equal to the rigid body angular velocity acquired by the tip-over from the center-of-gravity over corner position. In the above relation, L is the length of the overpack, I is the mass moment of inertia defined in Appendix 3.A, and k is a “spring constant” associated with the target resistance. If we solve for the maximum angular acceleration subsequent to time zero, we obtain the result in terms of the initial angular velocity as:

$$\ddot{\theta}_{\max} = \omega\dot{\theta}_0$$

If we form the maximum linear acceleration at the top of the overpack lid, we can finally relate the decelerations of the HI-STORM 100 configuration analyzed in Appendix 3.A and the HI-STORM 100/MPC-68M configuration solely in terms of their geometry properties and their mass ratio. The value of “k”, the target spring rate is the same for both overpacks so it does not appear in the relationship between the two decelerations. After substituting the appropriate geometry and calculated masses, we determine that the ratio of maximum rigid body decelerations at the top surface of the lids is:

$$A_{\text{HI-STORM 100-68M}}/A_{\text{HI-STORM 100}} = 1.01$$

The fact that the calculated ratio is only marginally above 1.0 indicates that the MPC-68M has a minor effect on the non-mechanistic tip-over analysis performed in Appendix 3.A. The maximum

rigid body deceleration for the HI-STORM 100/MPC-68M configuration is determined by scaling the calculated result from Appendix 3.A as follows:

$$A_{\text{HI-STORM 100-68M}} = 1.01 \times A_{\text{HI-STORM 100}} = 1.01 \times 42.98g = 43.42g$$

This demonstrates that when the MPC-68M is stored inside the HI-STORM 100 overpack the result for maximum deceleration level is less than the design basis deceleration limit specified in Table 3.1.2. Based on the comparative evaluations in Subsection 3.4.10, the HI-STORM 100 overpack is the limiting overpack for the non-mechanistic tip-over event. Therefore, when the MPC-68M is inside the HI-STORM 100S or the HI-STORM 100S Version B overpack, the maximum rigid body deceleration at the top surface of the lid is less than the deceleration above.

Next we demonstrate that the deceleration sustained by a vertical free fall of a HI-STORM overpack carrying a loaded MPC-68M onto the ISFSI pad is less than the design basis deceleration limit specified in Table 3.1.2. According to Appendix 3.A, analysis of a single mass impacting a spring with a given initial velocity shows that the maximum deceleration “ $a_M$ ” of the mass is related to the dropped weight “ $w$ ” and the drop height “ $h$ ” as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}$$

In other words for a fixed drop height, as the dropped weight decreases, the maximum deceleration of the mass increases. Since the MPC-68M weighs less than the MPC analyzed in Appendix 3.A, the maximum deceleration calculated in Appendix 3.A is not bounding. From the above relationship, the maximum deceleration for the HI-STORM 100/MPC-68M configuration is determined as:

$$a_{100-68M} = \sqrt{\frac{w_{100}}{w_{100-68M}}} a_{100}$$

where  $w_{100-68M}$  is the weight of a HI-STORM 100 carrying a loaded MPC-68M,  $w_{100}$  is the weight of a loaded HI-STORM 100 overpack as analyzed in Appendix 3.A, and  $a_{100}$  is the maximum deceleration of the HI-STORM 100 calculated in Appendix 3.A for an 11” vertical drop. The above equation yields the following result:

$$a_{100-68M} = 44.39g$$

Although the result is higher than the maximum deceleration calculated in Appendix 3.A, it is still less than the design basis vertical deceleration limit specified in Table 3.1.2. Therefore, the previously established lift height limit of 11 inches for a loaded HI-STORM overpack is also applicable to HI-STORM overpacks carrying the MPC-68M. Note that depending on site-specific conditions, a lower lift height limit may be required. See Section 2.2.3.1.

Finally, Subsection 3.4.10 provides the results of a simple elastic strength of materials calculation, which demonstrates that the cylindrical storage overpack will not permanently deform to the extent that the MPC cannot be removed by normal means after a tip-over event. Those results are valid for the MPC-68M since:

- i. there are no changes to the HI-STORM overpack stemming from the MPC-68M;
- ii. the external dimensions of the MPC-68M are the same as all other MPC types;
- iii. the results are calculated using upper bound impact decelerations.

#### 3.III.4.11 Storage Overpack and HI-TRAC Transfer Cask Service Life

Same as in Subsection 3.4.11 (including all paragraphs).

#### 3.III.4.12 MPC Service Life

Same as in Subsection 3.4.12 and with the following supplementary information provided herein.

##### 3.III.4.12.1 Metamic-HT Considerations

Metamic-HT has been extensively tested as indicated in Supplement 1.III. Testing has included extensive tests for creep, irradiation and corrosion to ensure long-term fuel basket performance under normal conditions of storage. The Metamic-HT is also not susceptible to structural fatigue and brittle fracture under long term conditions of storage. Corrosion is discussed further in Subsection 3.III.4.1. Creep and boron depletion are further discussed below.

##### i) Fuel Basket Creep

The Metamic sourcebook contains data on the testing to determine the creep characteristics of the Metamic-HT under both unirradiated and irradiated conditions. A creep equation to estimate a bounding estimate of total creep as a function of stress and temperature is also provided. The creep equation developed from this test provides a conservative prediction of accumulated creep strain by direct comparison to measured creep in unirradiated and irradiated coupons.

The creep equation for Metamic-HT that bounds *all* measured data (tests run for 20,000 hours) is of the classical exponential form in stress and temperature [1.III.3], which is written symbolically as  $\epsilon = f(\sigma, T)$ .

Creep in the MPC-68M fuel basket will not be a reactivity modifier because the basket is arrayed in the vertical orientation. The lateral loading of the fuel basket walls is insignificant and hence no mechanistic means for the basket panels to undergo lateral deformation from creep exists, even if the panel material were susceptible to creep.

The creep effect would tend to shorten the fuel basket under the self-weight of the basket. An illustrative calculation of the cumulative reduction of the basket length is presented below to demonstrate the insignificant role of creep in the MPC-68M fuel basket.

The in-plane compressive stress,  $\sigma$ , at height  $x$  in the basket panel is given by

$$\sigma = \rho(H-x) \quad (3.III.1)$$

where:

$\rho$  = weight density of Metamic-HT

$H$  = height of the fuel basket

Using the above stress equation, the total creep shrinkage,  $\delta$ , is given by

$$\delta = \int_0^H f(\sigma, T) dx \quad (3.III.2)$$

where:

$T$  = panel's metal temperature (conservatively assumed to be 375°C for a period of 60 years)

$H$  = height of the basket (conservatively assumed to be 200 inches)

Using the creep equation (provided in [1.III.3]) and performing the above integration numerically yields  $\delta = 0.138$  inch. In other words, the computed shrinkage of the basket is less than 0.069% of its original length. Therefore, it is concluded that for the vertical storage configuration the creep effects of the MPC-68M fuel basket are insignificant due to absence of any meaningful loads on the panels. Therefore, creep in the Metamic-HT fuel basket is not a matter of safety concern.

## ii) Fuel Basket Boron Depletion

The similarities between Metamic-HT and Metamic (classic) neutron absorbers and their exposure to the same long-term conditions of storage in the HI-STORM 100 system provide a logical basis to expect negligible neutron absorber boron depletion in Metamic-HT. However, to assure criticality safety during worst case design basis conditions over the 40-year design life, the analysis discussed in Subsection 6.III demonstrates that the boron depletion in the Metamic-HT is negligible over a 50-year duration. Thus, sufficient levels of boron are present in the fuel basket to maintain criticality safety over the 40-year design life of the MPC.

### 3.III.4.12.2 Basket Shim Considerations:

#### i) Basket Shim Creep

Like the fuel basket, the basket shims are not subject to any significant loading during storage. The ability of the basket shims (made of a creep resistant aluminum alloy) has been evaluated and qualified in Docket No. 71-9325 [2.III.6.2] for transport applications where the stress level (in

horizontal configuration) is significant. Therefore, in light of the minuscule stress levels from self-weight in long-term storage, creep is ruled out as a viable concern for the basket shims.

## ii) Basket Shim Corrosion

Basket shim corrosion is discussed in Subsection 3.III.4.1.

### 3.III.4.13 Design and Service Life

Same as in Subsection 3.4.13.

### 3.III.5 FUEL RODS

Same as in Section 3.5.

### 3.III.6 SUPPLEMENTAL DATA

#### 3.III.6.1 Additional Codes and Standards Referenced in HI-STORM 100 System Design and Fabrication

Same as in Subsection 3.6.1.

#### 3.III.6.2 Computer Programs

ANSYS 11.0, which is a public domain finite element code, has been utilized to perform structural analyses documented in this supplement.

#### 3.III.6.3 Appendices Included in Supplement 3.III

None.

#### 3.III.6.4 Calculation Packages

A calculation package containing the structural calculations supporting Supplement 3.III has been prepared, reviewed, and archived according to Holtec International's quality assurance program (see Chapter 13).

### 3.III.7 COMPLIANCE WITH NUREG-1536

The material in this supplement for the MPC-68M provides the same information as previously provided for the other MPC types in Chapter 3. Therefore, to the extent applicable, the information provided is in compliance with NUREG-1536.

3.III.8        REFERENCES

- [3.III.1]        Buckling of Bars, Plates, and Shells, D.O. Brush and B.O. Almroth, McGraw-Hill, 1975, p.22.
- [3.III.2]        Properties of Aluminum Alloys, Tensile, Creep, and Fatigue Data at High and Low Temperatures, ASM International, November 2006.
- [3.III.3]        ASME Boiler & Pressure Vessel Code, Section II, Parts A and D, American Society of Mechanical Engineers, 2007.
- [3.III.4]        Deleted.
- [3.III.5]        ASTM Specification B221M-07, “Standard Specification for Aluminum and Aluminum-Alloy Extruded Bars, Rods, Wire, Profiles, and Tubes (Metric)”.

**TABLE 3.III.1**  
**WEIGHT DATA FOR MPC-68M**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>MPC-68M</b> <ul style="list-style-type: none"><li data-bbox="250 428 472 457">. Without SNF</li><li data-bbox="250 472 818 501">. Fully loaded with SNF and Fuel Spacers</li></ul>	30,000 90,000



**TABLE 3.III.2**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**  
**INVOLVING MPC-68M**

<b>Component</b>	<b>Height of CG Above Datum (in)</b>
MPC-68M (empty)	114.9
HI-STORM 100 Overpack w/ fully loaded MPC-68M	118.4
HI-STORM 100S(232) Overpack w/ fully loaded MPC-68M	113.3
HI-STORM 100S(243) Overpack w/ fully loaded MPC-68M	117.7
HI-STORM 100S Version B(218) Overpack w/ fully loaded MPC-68M	108.5
HI-STORM 100S Version B(229) Overpack w/ fully loaded MPC-68M	112.8
HI-TRAC 125 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68M (water jacket filled)	97.4
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68M (water jacket filled)	96.5
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68M (water jacket filled)	96.9
HI-TRAC 100D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68M (water jacket filled)	94.3

## Notes:

1. The datum used for calculations involving the HI-STORM is the bottom of the overpack baseplate. The datum used for calculations involving the HI-TRAC is the bottom of the pool lid.
2. The datum used for calculations involving only the MPC is the bottom of the MPC baseplate.
3. The CG height of the HI-STORM overpack is calculated based on standard density concrete (i.e., 166 pcf dry) in the radial cavity. At higher densities, the CG height is slightly lower, which makes the HI-STORM overpack less prone to tipping.

**TABLE 3.III.3  
FUEL BASKET SHIMS – NOMINAL MECHANICAL PROPERTIES**

<b>Aluminum Alloy (B221 2219-T8511)</b>					
Temp. °C (°F)	S <sub>y</sub>	S <sub>u</sub>	E	α	% Elongation
25 (75)	290 (42)	400 (58)	7.2 (10.5)	–	5
150 (300)	243 (35)	307 (44)	6.8 (9.8)	23.9 (13.3)	6.4
204 (400)	188 (27)	231 (34)	6.3 (9.1)	24.5 (13.6)	8.2
230 (450)	171 (25)	209 (30)	6.1 (8.8)	24.8 (13.8)	8.6
260 (500)	154 (22)	182 (26)	5.9 (8.5)	25.0 (13.9)	8.6
290 (550)	98 (14)	116 (17)	5.5 (8.0)	25.4 (14.1)	10.5

## Definitions:

S<sub>y</sub> = Yield Stress, MPa (ksi)

α = Mean Coefficient of thermal expansion, cm/cm-°C x 10<sup>-6</sup> (in/in-°F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress, MPa (ksi)

E = Young's Modulus, MPa x 10<sup>4</sup> (psi x 10<sup>6</sup>)

## Notes:

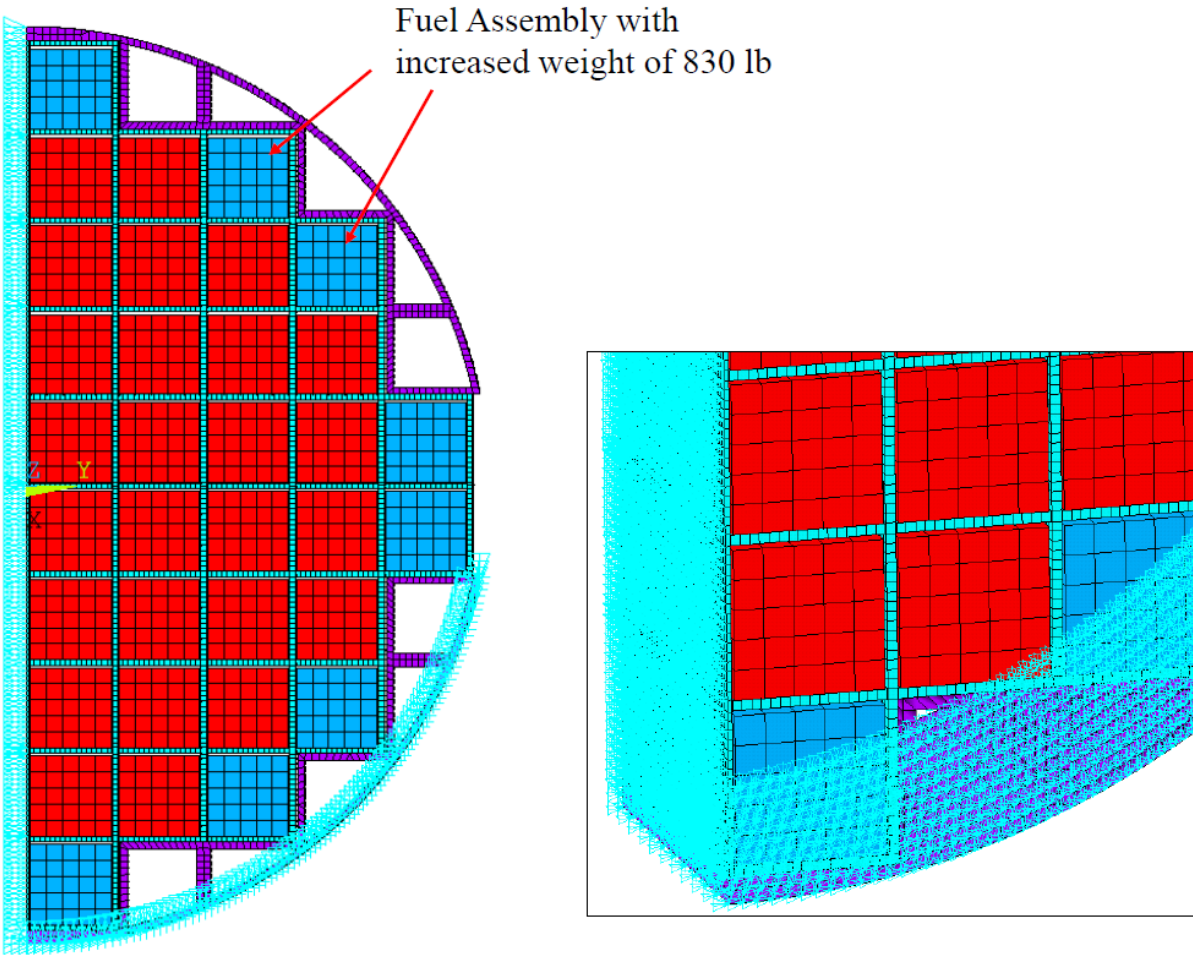
1. Source for E values is “Properties of Aluminum Alloys”, page 82 [3.III.2] (properties listed in the table above are not affected by time at temperature).
2. Source for α is Table TE-2 of [3.III.3] (values listed in TE-2 are also considered representative of Aluminum Alloy (2219-T8511) (UNS No. A92219)).
3. Source for S<sub>y</sub>, S<sub>u</sub>, and % Elongation values at room temperature is ASTM Specification B221M [3.III.5]. Values at elevated temperatures are obtained by scaling the room temperature values using the data from [3.III.3].

**TABLE 3.III.4**  
**MAXIMUM DISPLACEMENT IN MPC-68M FUEL BASKET**

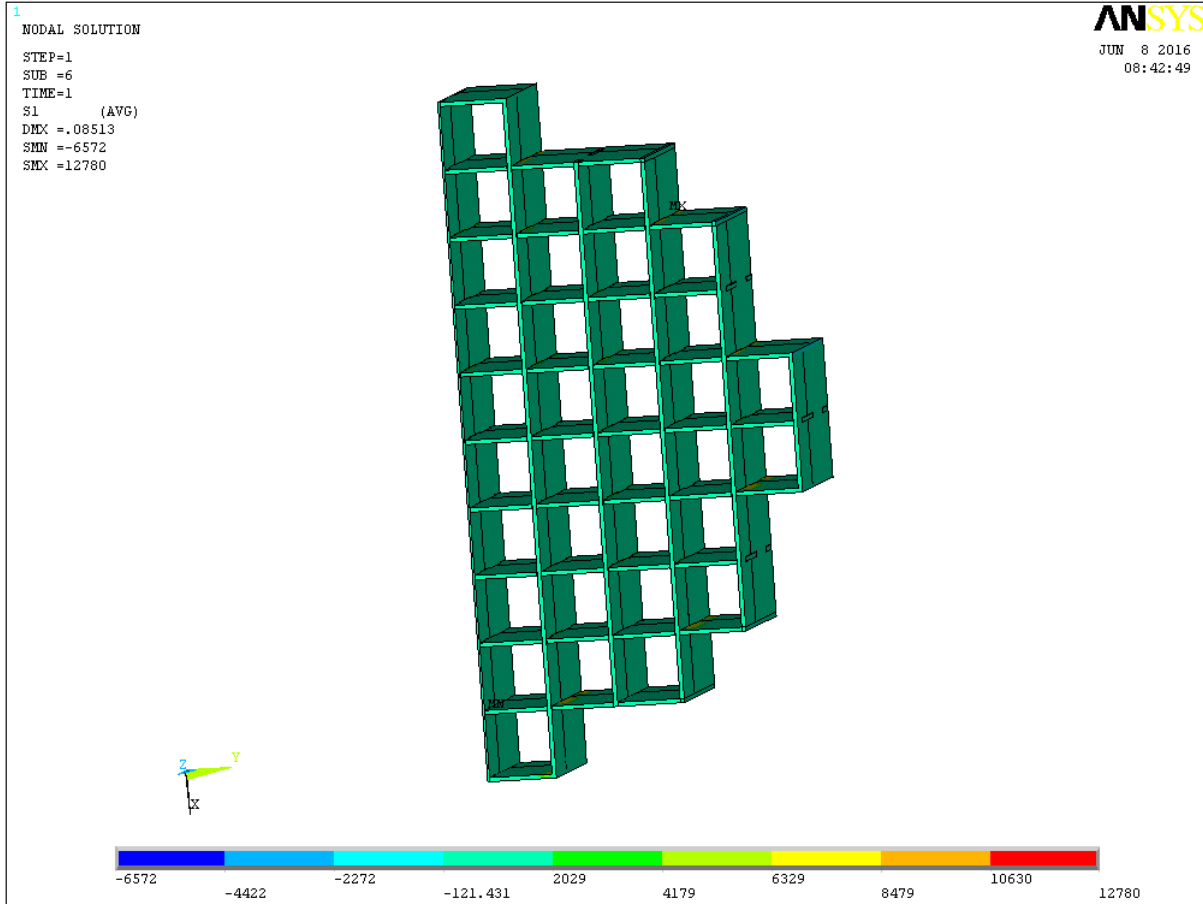
<b>Maximum Lateral Displacement in Fuel Basket Panel, <math>\theta</math> (dimensionless) (Note 1)</b>	<b>Maximum Allowable Value of <math>\theta</math> (from Table 2.III.4)</b>	<b>Safety Factor</b>
$1.008 \times 10^{-3}$	0.005	4.96

Notes:

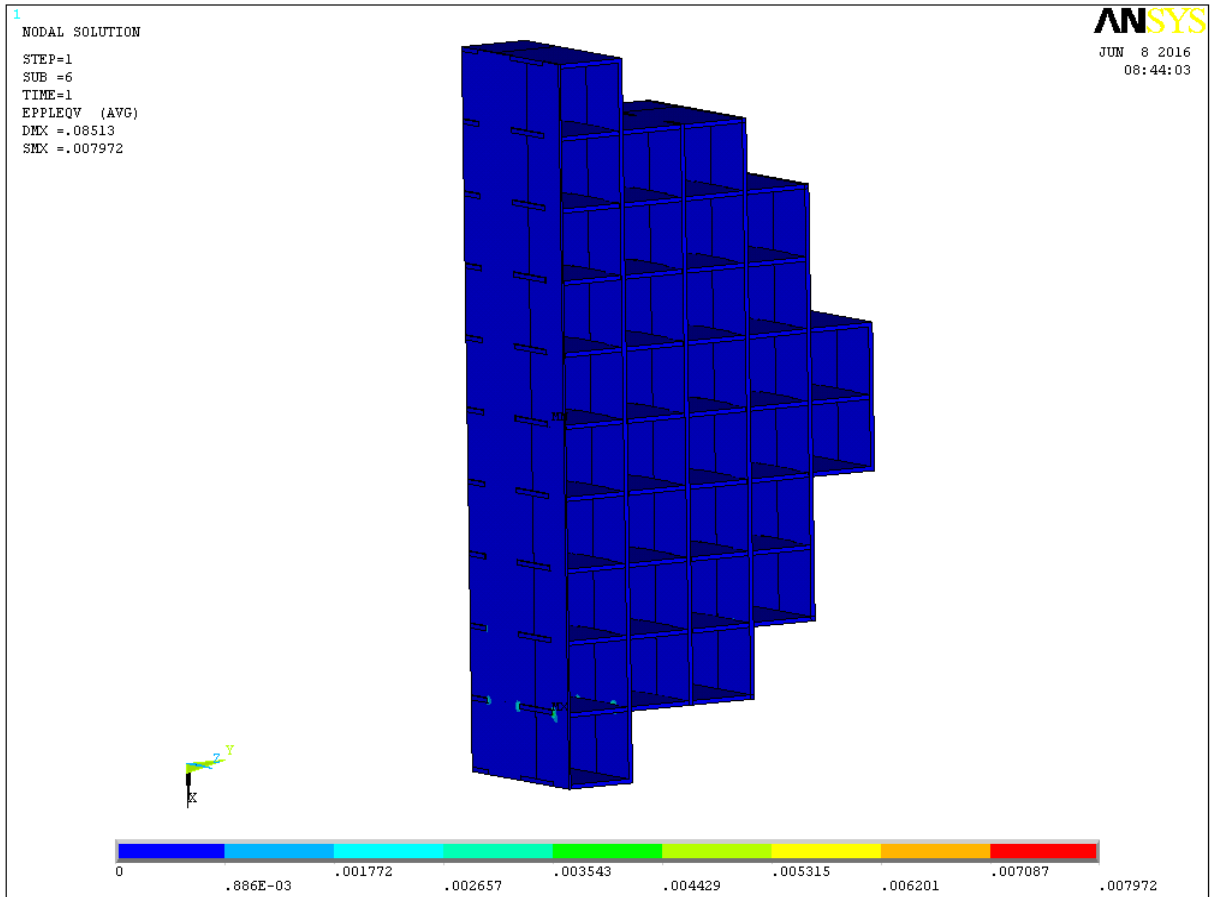
1. See Subsection 2.III.0.1 for definition of  $\theta$ .



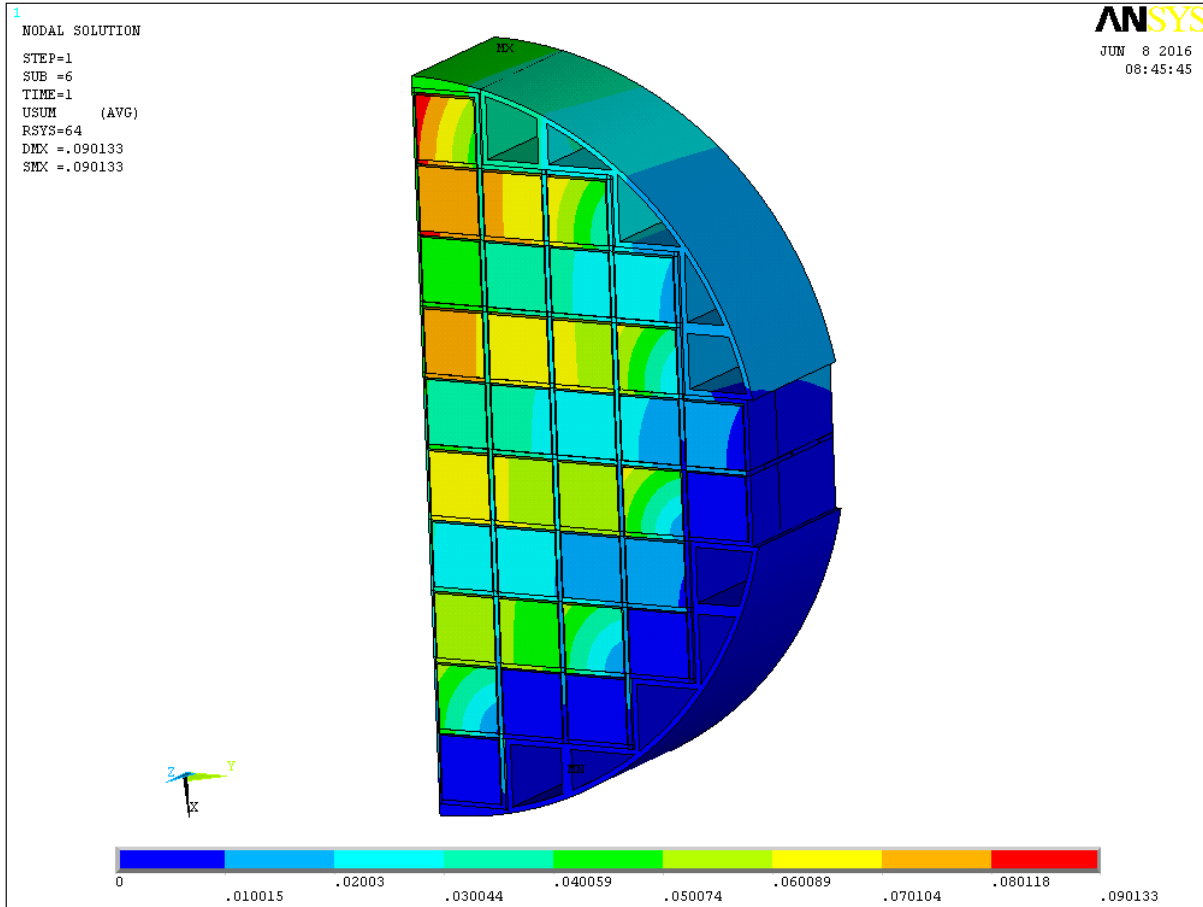
**FIGURE 3.III.1: FINITE ELEMENT MODEL OF MPC-68M FUEL BASKET**



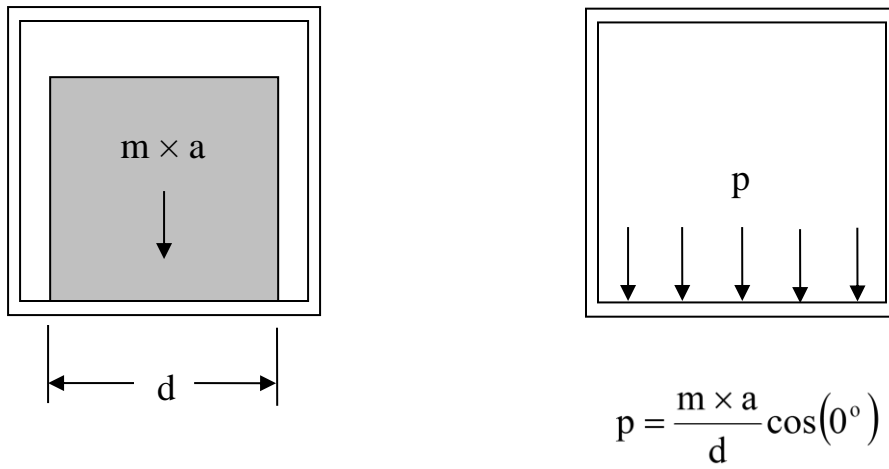
**FIGURE 3.III.2: TRUE STRESS DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60G LOAD**



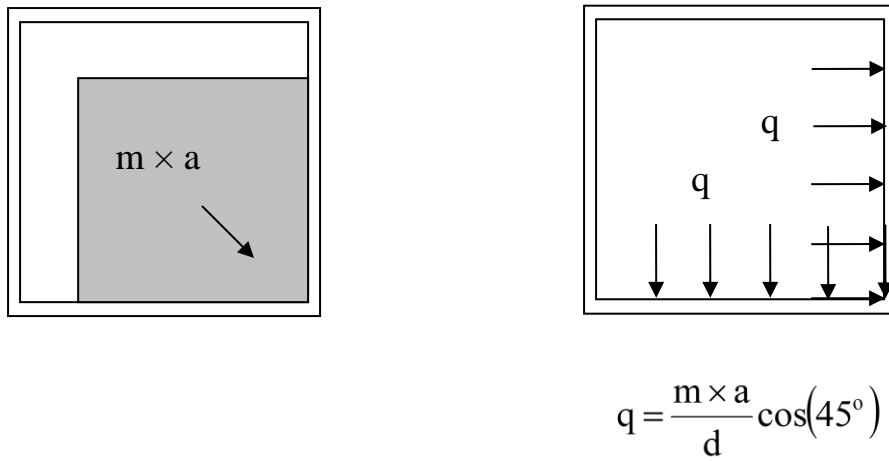
**FIGURE 3.III.3: PLASTIC STRAIN DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60g LOAD**



**FIGURE 3.III.4: DISPLACEMENT CONTOURS IN MPC-68M FUEL BASKET UNDER 60g LOAD**



(a) 0° Orientation



(b) 45° Orientation

**FIGURE 3.III.5: FUEL LOADING FOR 0° AND 45° BASKET ORIENTATIONS**



## CHAPTER 4<sup>1</sup> THERMAL EVALUATION

### 4.0 OVERVIEW

The HI-STORM System is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. An array of HI-STORM Systems laid out in a rectilinear pattern will be stored on a concrete ISFSI pad in an open environment. In this section, compliance of the HI-STORM thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI is established. The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. Effects of incident solar radiation (insolation) and partial radiation blockage due to the presence of neighboring casks at an ISFSI site are included in the analyses. Finally, the thermal margins of safety for long-term storage of both moderate burnup (up to 45,000 MWD/MTU) and high burnup spent nuclear fuel (greater than 45,000 MWD/MTU) in the HI-STORM 100 system are quantified. Safe thermal performance during on-site loading, unloading and transfer operations utilizing the HI-TRAC transfer cask is also demonstrated.

The HI-STORM thermal evaluation follows the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4] to demonstrate thermal compliance of the HI-STORM system. . These guidelines provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF)<sup>2</sup> and other confinement boundary components, and on the maximum permissible pressure in the confinement space under certain operating scenarios. Specifically, the requirements are:

1. The fuel cladding temperature for long-term storage shall be limited to 752°F (400°C).
2. The fuel cladding temperature for short-term operations shall be limited to 752°F (400°C) for high burnup fuel and 1058°F (570°C) for moderate burnup fuel.
3. The fuel cladding temperature should be maintained below 1058°F (570°C) for accident and off-normal event conditions.
4. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions.

- 1 This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary .
- 2 Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See glossary).

5. The cask materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.
6. For fuel assemblies proposed for storage, the cask system should ensure a very low probability of cladding breach during long-term storage.
7. The HI-STORM System should be passively cooled.
8. The thermal performance of the cask shall be in compliance with the design criteria specified in FSAR Chapters 1 and 2 for normal, off-normal, and accident conditions.

As demonstrated in this chapter, the HI-STORM System is designed to comply with all of the criteria listed above. Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions. All thermal analyses to evaluate normal conditions of storage in a HI-STORM storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC transfer cask are described in Section 4.5. All thermal analyses to evaluate off-normal and accident conditions are described in Section 4.6. This FSAR chapter is in full compliance with ISG-11 and with NUREG-1536 guidelines, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.2.

The HI-STORM thermal evaluations for CSF are grouped in two categories of fuel assemblies. The two groups are classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE group of fuel assemblies are characterized by low burnup, long cooling time, and short active fuel lengths. Consequently, their heat loads are dwarfed by the DB group of fuel assemblies. All Dresden-1 (6x6 and 8x8 and a thoria rod canister constituted as part of an 8x8 fuel assembly), Quad+, Humboldt Bay (7x7 and 6x6), Indian Point, Haddam Neck and all stainless-steel clad fuel assemblies are classified as LHE fuel. The low heat emitting characteristics of these fuel assemblies render them non-governing for thermal evaluation. The HI-STORM System temperatures for MPCs loaded with LHE fuel are bounded by design basis evaluations reported in this chapter.

The HI-STORM System is evaluated for two fuel storage scenarios. In one scenario, designated as uniform loading, every basket cell is assumed to be occupied with fuel producing heat at the maximum rate. As discussed in Chapter 2, this storage specification is extremely conservative, and virtually impossible to realize in actual practice. A less unrealistic, yet conservative idealization of storage scenario, designated as regionalized loading, involves defining two discrete regions within the basket. The two regions are designated as Region 1 (inner region) and Region 2 (outer region). Regionalized storage is designed to recognize storage of fuel assemblies having wide disparity in heat emission rates. For further discussion of regionalized storage, Section 2.1 of Chapter 2 should be consulted.

The HI-STORM System is designed for one reference storage condition defined in Table 4.0.1. This condition establishes the required helium backfill pressures computed later in this chapter (See Subsection 4.4.5.1). Having defined the helium backfill pressures an array of analyses are performed to evaluate the range of storage configurations specified in Chapter 2 and results reported in Section 4.4.

Table 4.0.1

## REFERENCE HI-STORM OPERATING CONDITIONS

<b>Condition</b>	<b>Value</b>
MPC Decay Heat	Table 2.1.26
MPC Operating Pressure	7 atm (absolute)
Normal Ambient Temperature	Table 2.2.2

## 4.1 DISCUSSION

The HI-STORM FSAR seeks to establish complete compliance with the provisions of ISG-11 [4.1.4]. For this purpose the HI-STORM normal storage fuel cladding temperatures are required to meet the 752°F (400°C) temperature limit for all CSF (See Section 4.3). Additionally, when the MPCs are deployed for storing High Burnup Fuel (HBF) further restrictions during certain fuel loading activities (vacuum drying) are set forth to preclude fuel temperatures from exceeding the normal temperature limits. To ensure explicit compliance, a specific term “short term operations” is defined in Chapter 2 to cover all fuel loading activities. ISG-11 fuel cladding temperature limits are applied for short-term operations (see Table 4.3.1).

Potential thermally challenging states for the spent fuel arise if the fuel drying process utilizes the pressure reduction process (i.e., vacuum drying). The short-term evolutions that may be thermally limiting and warrant analysis are:

- i. Vacuum Drying
- ii. Loaded MPC in HI-TRAC in the Vertical Orientation

The threshold MPC heat generation rate at which the HI-STORM peak cladding temperature reaches a steady state equilibrium value approaching the normal storage peak clad temperature limit is computed in this chapter. Likewise, the MPC heat generation rates that produce the steady state equilibrium temperature approaching the normal storage peak clad temperature limit for the MPC in HI-TRAC are computed in this chapter. These computed heat generation rates directly bear upon the compliance of the system with ISG-11 [4.1.4] and are, accordingly, adopted in the system Technical Specifications for high burnup fuel (HBF).

The aboveground HI-STORM system consists of a sealed MPC situated inside a vertically-oriented, ventilated storage overpack. Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top, respectively, of the cylindrical overpack. The SNF assemblies reside inside the MPC, which is sealed with a welded lid to form the confinement boundary. The MPC contains a stainless-steel honeycomb fuel basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. Each fuel basket panel, with the exception of exterior panels on the MPC-68 and MPC-32, is equipped with a thermal neutron absorber panel sandwiched between an Alloy X steel sheathing plate and the fuel basket panel, along the entire length of the active fuel region. The MPC is backfilled with helium up to the design-basis initial fill level (Table 1.2.2). This provides a stable, inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM System to the environment by passive heat transport mechanisms only.

The helium backfill gas plays an important role in the MPC’s thermal performance. The helium fills all the spaces between solid components and provides an improved conduction medium (compared to air) for dissipating decay heat in the MPC. Within the MPC the pressurized helium environment sustains a closed loop thermosiphon action, removing SNF heat by an upward flow of helium through the storage cells. This MPC internal convection heat dissipation mechanism is illustrated in

Figure 4.1.1. On the outside of the MPC a ducted overpack construction with a vertical annulus facilitates an upward flow of air by buoyancy forces. The annulus ventilation flow cools the hot MPC surfaces and safely transports heat to the outside environment. The annulus ventilation cooling mechanism is illustrated in Figure 4.1.2. To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC confinement boundary is designed and fabricated in accordance with the ASME B&PV Code Section III, Subsection NB as an all-seal-welded pressure vessel with redundant closures. It is demonstrated in Section 11.1.3 that the failure of one field-welded pressure boundary seal will not result in a breach of the pressure boundary. The helium gas is therefore assumed to be retained in an undiluted state, and may be credited in the thermal analyses.

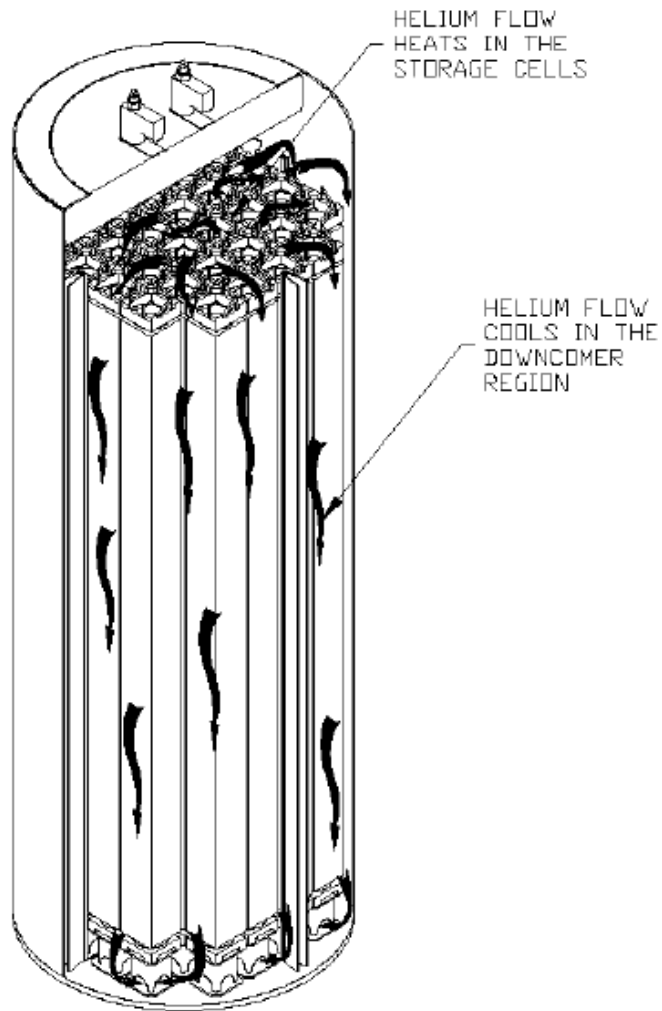
An important thermal design criterion imposed on the HI-STORM System is to limit the maximum fuel cladding temperature to within design basis limits (Table 4.3.1) for long-term storage of design basis SNF assemblies. An equally important requirement is to minimize temperature gradients in the MPC so as to minimize thermal stresses. In order to meet these design objectives, the MPC baskets are designed to possess certain distinctive characteristics, which are summarized in the following.

The MPC design minimizes resistance to heat transfer within the basket and basket periphery regions. This is ensured by an uninterrupted panel-to-panel connectivity realized in the all-welded honeycomb basket structure. The MPC design incorporates top and bottom plenums with interconnected downcomer paths. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket, and by elongated semicircular holes in each basket cell wall. The bottom plenum is formed by large elongated semicircular holes at the base of all cell walls. The MPC basket is designed to eliminate structural discontinuities (i.e., gaps) which introduce added thermal resistances to heat flow. Consequently, temperature gradients are minimized in the design, which results in lower thermal stresses within the basket. Low thermal stresses are also ensured by an MPC design that permits unrestrained axial and radial growth of the basket. The possibility of stresses due to restraint on basket periphery thermal growth is eliminated by providing adequate basket-to-canister shell gaps to allow for basket thermal growth during all operational modes.

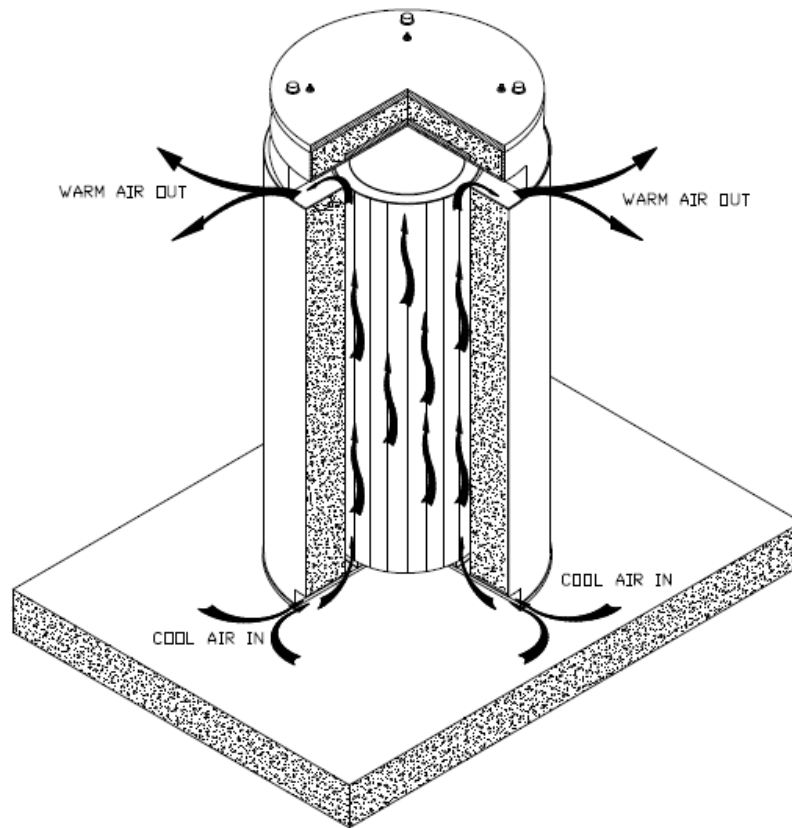
The MPCs design maximum decay heat loads for storage of zircaloy clad fuel are listed in Table 4.0.1. Storage of stainless steel clad fuel is permitted for a low decay heat limit set forth in Chapter 2 (Tables 2.1.17 through 2.1.24). Storage of zircaloy clad fuel with stainless steel clad fuel in an MPC is permitted. In this scenario, the zircaloy clad fuel must meet the lower decay heat limits for stainless steel clad fuel. The axial heat distribution in each fuel assembly is conservatively assumed to be non-uniformly distributed with peaking in the active fuel mid-height region (See axial burnup Table 2.1.11).

The HI-STORM System (i.e., HI-STORM overpack, HI-TRAC transfer cask and MPC) is evaluated under normal storage (HI-STORM overpack), during off-normal and accident events and during short term operations in a HI-TRAC. Results of HI-STORM thermal analysis during normal (long-term) storage are obtained and reported in Section 4.4. Results of off-normal and accident events are reported in Section 4.6. Results of HI-TRAC short term operations (fuel loading, vacuum drying)

are reported in Section 4.5.



**FIGURE 4.1.1: MPC INTERNAL HELIUM CIRCULATION**



**FIGURE 4.1.2: VENTILATION COOLING OF A HI-STORM SYSTEM**



## 4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Materials present in the MPCs include stainless steels (Alloy X), neutron absorber (Boral or METAMIC) and helium. Materials present in the HI-STORM storage overpack include carbon steels and concrete. Materials present in the HI-TRAC transfer cask include carbon steel, lead, Holtite-A neutron shield, paints (See Appendix 1.C) and demineralized water. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Individual thermal conductivities of the alloys that comprise the Alloy X materials and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2 and 4.2.3 provide numerical thermal conductivity data of materials at several representative temperatures. The currently approved neutron absorber materials, (Boral™ and Metamic™) are both made of aluminum powder and boron carbide powder. Although their manufacturing processes differ, from a thermal standpoint, their ability to conduct heat is virtually identical. Therefore, the values of conductivity of the original neutron absorber (Boral) continue to be used in the thermal calculations.

For the HI-STORM overpack, the emissivity/absorptivity of painted surfaces is particularly important. Recognizing the considerable variations in reported values for thermal conductivity and emissivity, the values that are conservative with respect to both authoritative references and values used in analyses on previously licensed cask dockets have been selected. Specific discussions of the conservatism of the selected values are included in the following paragraphs.

As specified in Table 4.2.1, the concrete thermal conductivity is taken from Marks' Standard Handbook for Mechanical Engineers, which is conservative compared to a variety of recognized concrete codes and references. Neville, in his book "Properties of Concrete" (4<sup>th</sup> Edition, 1996), gives concrete conductivity values as high as 2.1 Btu/(hr×ft×°F). For concrete with siliceous aggregates, the type to be used in HI-STORM overpacks, Neville reports conductivities of at least 1.2 Btu/(hr×ft×°F). Data from Loudon and Stacey, extracted from Neville, reports conductivities of 0.980 to 1.310 Btu/(hr×ft×°F) for normal weight concrete protected from the weather. ACI-207.1R provides thermal conductivity values for seventeen structures (mostly dams) at temperatures from 50-150°F. Every thermal conductivity value reported in ACI-207.1R is greater than the value used in the HI-STORM thermal analyses. Additionally, the NRC has previously approved analyses that use higher conductivity values than those applied in the HI-STORM thermal analysis. For example, thermal calculations for the NRC approved Vectra NUHOMS cask system (June 1996, Rev. 4A) used thermal conductivities as high as 1.17 Btu/(hr×ft×°F) at 100°F. Based on these considerations, the concrete thermal conductivity value chosen for HI-STORM thermal analyses is considered to be conservative.

Holtite-A is a composite material consisting of approximately 37 wt% epoxy polymer, 1 wt% B<sub>4</sub>C and 62 wt% aluminum trihydrate. While polymers are generally characterized by a low conductivity (0.05 to 0.2 Btu/ft-hr-°F), the addition of fillers in substantial amounts can raise the mixture conductivity by up to a factor of ten. The thermal conductivity of epoxy filled resins with alumina is

reported in the technical literature<sup>3</sup> as approximately 0.5 Btu/ft-hr-°F and higher. A conservatively postulated conductivity of 0.3 Btu/ft-hr-°F is used in the thermal models for the neutron shield region<sup>4</sup> (in the HI-TRAC transfer cask). As the thermal inertia of the neutron shield is not credited in the analyses, the density and heat capacity properties are not reported herein.

Surface emissivity data for key materials of construction are provided in Table 4.2.4. The emissivity properties of painted external surfaces are generally excellent. Kern [4.2.5] reports an emissivity range of 0.8 to 0.98 for a wide variety of paints. In the HI-STORM thermal analysis, an emissivity of 0.85<sup>5</sup> is applied to painted surfaces. A conservative solar absorptivity coefficient of 1.0 is applied to all exposed overpack surfaces.

In Table 4.2.5, the heat capacity and density of the MPC, overpack and CSF materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. The temperature-dependent values of the viscosities of helium and air are provided in Table 4.2.6.

The heat transfer coefficient for exposed surfaces is calculated by accounting for both natural convection and thermal radiation heat transfer. The natural convection coefficient depends upon the product of Grashof (Gr) and Prandtl (Pr) numbers. Following the approach developed by Jakob and Hawkins [4.2.9], the product Gr×Pr is expressed as  $L^3\Delta T Z$ , where L is height of the overpack,  $\Delta T$  is overpack surface temperature differential and Z is a parameter based on air properties, which are known functions of temperature, evaluated at the average film temperature. The temperature-dependent values of Z are provided in Table 4.2.7.

3 “Principles of Polymer Systems”, F. Rodriguez, Hemisphere Publishing Company (Chapter 10).

4 The thermal conductivity value used in the thermal models for the neutron shield region is confirmed to be bounded by the Holtite-A test data [4.2.13] with a margin.

5 This is conservative with respect to prior cask industry practice, which has historically utilized higher emissivities [4.2.16].

Table 4.2.1

**SUMMARY OF HI-STORM SYSTEM MATERIALS  
THERMAL PROPERTY REFERENCES**

<b>Material</b>	<b>Emissivity</b>	<b>Conductivity</b>	<b>Density</b>	<b>Heat Capacity</b>
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	[4.2.3], [4.2.17], [4.2.18], [4.2.7]	NUREG [4.2.6]	Rust [4.2.4]	Rust [4.2.4]
UO <sub>2</sub>	Note 1	NUREG [4.2.6]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel (machined forgings) <sup>6</sup>	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Stainless Steel Plates <sup>7</sup>	ORNL [4.2.11], [4.2.12]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Carbon Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Boral	Note 1	Test Data (Note 2)	Test Data (Note 2)	Test Data (Note 2)
Holtite-A	Note 1	[4.2.13]	Not Used	Not Used
Concrete	Note 1	Marks' [4.2.1]	Appendix 1.D	Handbook [4.2.2]
Lead	Note 1	Handbook [4.2.2]	Handbook [4.2.2]	Handbook [4.2.2]
Water	Note 1	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
METAMIC	Note 1	Test Data [4.2.14], [4.2.15]	Test Data [4.2.14], [4.2.15]	Test Data [4.2.14], [4.2.15]

Note 1: Emissivity not reported as radiation heat dissipation from these surfaces is conservatively neglected.

Note 2: AAR Structures Boral thermophysical test data.

<sup>6</sup> Used in the top lid of the MPC.

<sup>7</sup> Used in the basket panels, neutron absorber sheathing, MPC shell, and MPC baseplate.

Table 4.2.2

**SUMMARY OF HI-STORM SYSTEM MATERIALS  
THERMAL CONDUCTIVITY DATA**

<b>Material</b>	<b>At 200°F (Btu/ft-hr-°F)</b>	<b>At 450°F (Btu/ft-hr-°F)</b>	<b>At 700°F (Btu/ft-hr-°F)</b>	<b>At 1000°F (Btu/ft-hr-°F)</b>
Helium	0.0976	0.1289	0.1575	0.1890
Air*	0.0173	0.0225	0.0272	0.0336
Alloy X***	8.4	9.8	11.0	12.4
Carbon Steel	24.4	23.9	22.4	20.0
Concrete**	1.05	1.05	1.05	1.05
Lead	19.4	17.9	16.9	N/A
Water	0.392	0.368	N/A	N/A
<p>* At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F at 32°F and 0.0176 Btu/ft-hr-°F at 212°F.</p> <p>** Conservatively assumed to be constant for the entire range of temperatures.</p> <p>*** Individual thermal conductivities of the alloys that comprise the Alloy X materials are reported in Appendix 1.A. Lowerbound Alloy X thermal conductivity is tabulated herein.</p>				

Table 4.2.3

**SUMMARY OF FUEL ELEMENT COMPONENTS  
THERMAL CONDUCTIVITY DATA**

<b>Zircaloy Cladding</b>		<b>Fuel (UO<sub>2</sub>)</b>	
<b>Temperature (°F)</b>	<b>Conductivity (Btu/ft-hr-°F)</b>	<b>Temperature (°F)</b>	<b>Conductivity (Btu/ft-hr-°F)</b>
392	8.28*	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28*
* Lowest values of conductivity used in the thermal analyses for conservatism.			

Table 4.2.4

## SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA\*

<b>Material</b>	<b>Emissivity</b>
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel (machined forgings)	0.36
Stainless Steel Plates	0.587**
Carbon Steel	0.66
* See Table 4.2.1 for cited references.	
** Lowerbound value from the cited references in Table 4.2.1.	

Table 4.2.5

## DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY\*

<b>Material</b>	<b>Density (lbm/ft<sup>3</sup>)</b>	<b>Heat Capacity (Btu/lbm-°F)</b>
Helium	(Ideal Gas Law)	1.24
Zircaloy	409	0.0728
Fuel (UO <sub>2</sub> )	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Boral	154.7	0.13
Concrete	140**	0.156
Lead	710	0.031
Water	62.4	0.999
METAMIC	163.4**	0.22**
* See Table 4.2.1 for cited references.		
** Lowerbound values reported for conservatism.		

Table 4.2.6

## GASES VISCOSITY\* VARIATION WITH TEMPERATURE

Temperature (°F)	Helium Viscosity (Micropoise)	Temperature (°F)	Air Viscosity (Micropoise)
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	338.4	246.3
463.0	288.7	567.1	293.0
537.8	299.8	701.6	316.7
737.6	338.8	1078.2	377.6
921.2	373.0	-	-
1126.4	409.3	-	-
* Obtained from Rohsenow and Hartnett [4.2.2].			

Table 4.2.7

VARIATION OF NATURAL CONVECTION PROPERTIES  
PARAMETER "Z" FOR AIR WITH TEMPERATURE

Temperature (°F)	Z (ft <sup>-3</sup> °F <sup>-1</sup> )*
40	2.1×10 <sup>6</sup>
140	9.0×10 <sup>5</sup>
240	4.6×10 <sup>5</sup>
340	2.6×10 <sup>5</sup>
440	1.5×10 <sup>5</sup>
* Obtained from Jakob and Hawkins [4.2.9]	

### 4.3 SPECIFICATIONS FOR COMPONENTS

HI-STORM System materials and components designated as “Important to Safety” (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) which warrant special attention are summarized in Table 4.3.1. The neutron shielding ability of Holtite-A neutron shield material used in the HI-TRAC transfer cask is ensured by demonstrating that the material exposure temperatures are maintained below the maximum allowable limit. Long-term integrity of SNF is ensured by the HI-STORM System thermal evaluation which demonstrates that fuel cladding temperatures are maintained below design basis limits. Neutron absorber materials used in MPC baskets for criticality control (made from B<sub>4</sub>C and aluminum) are stable in excess of temperatures specified in Table 2.2.3<sup>1</sup>. Accordingly this temperature is conservatively adopted as the short-term temperature limit for neutron absorber materials. The overpack concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that American Concrete Institute (ACI) guidance on temperature limits (see Appendix 1.D) is followed.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term off-normal and severe hypothetical accident conditions. The inherent mechanical characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of HI-STORM System thermal performance, material temperature limits for long-term normal, short-term operations, and off-normal and accident conditions are provided in Table 4.3.1. In Table 4.3.1, ISG-11 [4.1.4] temperature limits are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

#### 4.3.1 Evaluation of Moderate Burnup Fuel

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.1]. Upon cooling below a threshold temperature ( $T_p$ ), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature  $T_p$  are excessive. For moderate burnup fuel,  $T_p$  is conservatively estimated as 350°C [4.3.1]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.1]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted in the HI-STORM FSAR for short-term operations for MBF fueled MPCs (see Table 4.3.1).

<sup>1</sup> B<sub>4</sub>C is a refractory material that is unaffected by high temperature (on the order of 1000°F) and aluminum is solid at temperatures up to 1080°F [4.2.19].

Table 4.3.1

HI-STORM SYSTEM MATERIAL TEMPERATURE LIMITS<sup>9</sup>

<b>Material</b>	<b>Normal Long-Term Temperature Limits [°F]</b>	<b>Short-Term Temperature Limits [°F]</b>
CSF cladding (zirconium alloys and stainless steel)	752	Short-Term Operations 752 (HBF) 1058 (MBF) Off-Normal and Accident 1058
Neutron Absorber	Table 2.2.3	Table 2.2.3
Holtite-A <sup>10</sup>	N/A (Not Used)	Table 2.2.3
Concrete <sup>11</sup>	Table 2.2.3	Table 2.2.3
Water	N/A	Table 2.2.3 <sup>12</sup>

<sup>9</sup> This table specifies temperature limits for non-ASME Code materials. Temperature limits of ASME Code materials (structural steels) are specified in Table 2.2.3.

<sup>10</sup> See Chapter 1, Appendix 1.B.

<sup>11</sup> These values are applicable for concrete in the overpack body, overpack lid and overpack pedestal..

<sup>12</sup> Saturation temperature at HI-TRAC water jacket design pressure specified in Table 2.2.1.



#### 4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

The HI-STORM System (i.e., HI-STORM overpack, HI-TRAC transfer cask and MPC) thermal evaluation is performed in accordance with the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4]. To ensure a high level of confidence in the thermal evaluation, 3-Dimensional models of the MPC, HI-STORM overpack and HI-TRAC transfer cask are constructed to evaluate fuel integrity under normal (long-term storage), off-normal and accident conditions and in the HI-TRAC transfer cask under short-term operation and hypothetical accidents. The thermal models incorporate an array of conservatisms to ensure robustly bounding thermal solutions. The principal features of these models are described in this section for HI-STORM and Section 4.5 for HI-TRAC. Thermal analysis results for the long-term storage scenarios are obtained and reported in this section.

##### 4.4.1 Overview of the Thermal Model

The MPC basket design consists of four distinct geometries to hold 24 or 32 PWR, or 68 BWR fuel assemblies. The basket is a matrix of interconnected square compartments designed to hold the fuel assemblies in a vertical position under long term storage conditions. The basket is a honeycomb structure of stainless steel (Alloy X) plates with full-length edge-welded intersections to form an integral basket configuration. All individual cell walls, except outer periphery cell walls in the MPC-68 and MPC-32, are provided with neutron absorber plates sandwiched between the box wall and a stainless steel sheathing plate over the full length of the active fuel region. The neutron absorber plates used in all MPCs are made of an aluminum-based, boron carbide-containing material to provide criticality control, while maximizing heat conduction capabilities.

Thermal analysis of the HI-STORM System is performed for an array of limiting heat load scenarios defined in Chapter 2 for uniform and regionalized fuel loading (wherein each fuel assembly in a region is assumed to be generating heat at the maximum permissible rate). While the assumption of limiting heat generation in each storage cell imputes a certain symmetry to the cask thermal problem, it grossly overstates the total heat duty of the system in most cases because it is unlikely that any basket would be loaded with fuel emitting heat at their limiting values (see for example a fuel loading scenario discussed in Section 2.1). The principal attributes of the thermal model are described in the following:

- i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange is a highly nonlinear function of surface temperatures.
- ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profiles in the fuel assemblies.
- iii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the MPC is spatially distributed with the maximum values reached in the central core region.

#### 4.4.1.1 Description of the 3-D Thermal Model

##### i. Introduction

The interior of the MPC is a 3-D array of square shaped cells inside an irregularly shaped basket outline confined inside the cylindrical space of the MPC cavity. To ensure an adequate representation of these features, a 3-D geometric model of the MPC is constructed using the FLUENT CFD code pre-processor [4.1.2]. Other than representing the composite cell walls (made up of Alloy X panels, neutron absorber panels and Alloy X sheathing) by a homogeneous panel with equivalent orthotropic (thru-thickness and parallel plates direction) thermal conductivities, the 3-D model requires no idealizations of the fuel basket structure. Further, since it is impractical to model every fuel rod in every stored fuel assembly explicitly, the cross section bounded by the inside of the storage cell (inside of the fuel channel in the case of BWR MPCs), which surrounds the assemblage of fuel rods and the interstitial helium gas (also called the “rodded region”), is replaced with an “equivalent” square homogeneous section characterized by an effective thermal conductivity. Homogenization of the storage cell cross-section is illustrated in Figure 4.4.1. As the effective conductivity of the rodded region includes radiation heat transfer the conductivities will be a strong function of temperature because radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) rises as the fourth power of absolute temperature. Therefore, in effect, the effective conductivity of the equivalent square section (depending on the coincident temperature) will be different throughout the basket. For thermal-hydraulic simulation, each fuel assembly in its storage cell is represented by an equivalent porous medium. For BWR fuel, the presence of the fuel channel divides the storage cell space into two distinct axial flow regions, namely, the in-channel (rodded) region and the square prismatic annulus region (in the case of PWR fuel this modeling complication does not exist).

##### ii. Details of the 3-D Model

The 3-D model implemented to analyze the HI-STORM system has the following key attributes:

- a. As mentioned above, the composite walls in the fuel basket consisting of the Alloy X structural panels, the aluminum-based neutron absorber, and the Alloy X sheathing, are represented by an orthotropic homogeneous panel of equivalent thermal conductivity in the three principal directions. The in-plane and thru-thickness thermal conductivities of the composite wall are computed using a standard procedure for such shapes with certain conservatisms, as described below.

During fabrication, a uniform normal pressure is applied to each “Box Wall - Neutron Absorber - Sheathing” sandwich in the assembly fixture during welding of the sheathing periphery on the box wall. This ensures adequate surface-to-surface contact between the neutron absorber and the adjacent Alloy X surfaces. The mean coefficient of linear expansion of the neutron absorber is higher than the thermal expansion coefficients of the basket and sheathing materials. Consequently, basket heat-up from the stored SNF will further ensure a

tight fit of the neutron absorber plate in the sheathing-to-box pocket. Nevertheless the possible presence of small microscopic gaps due to less than perfect surface-to-surface contact requires consideration of an interfacial contact resistance between the neutron absorber and box-sheathing surfaces. In the thermal analysis a 2 mil neutron absorber to pocket gap has been used. This is conservative as the sandwich is engineered to ensure an essentially no-gap fitup and assembly of the neutron-absorber panels. Furthermore, no credit is taken for radiative heat exchange across the neutron absorber to sheathing or neutron absorber to box wall gaps.

The heat conduction properties of the composite “Box Wall - Neutron Absorber - Sheathing” sandwich panels in the two principal basket cross sectional directions (i.e., thru-thickness and parallel plates direction) are unequal. In the thru-thickness direction, heat is transported across layers of sheathing, helium-gap, neutron absorber and box wall resistances that are essentially in series. Heat conduction in the parallel plates direction, in contrast, is through an array of essentially parallel resistances comprised of these several layers listed above. In this manner the composite walls of the fuel basket storage cells are replaced with a solid wall of equivalent through thickness and parallel plates direction conductivities. Table 4.4.1 provides the values of the conductivities as a function of temperature for the different MPC types.

- b. In the case of a BWR CSF, the fuel bundle and the small surrounding spaces inside the fuel “channel” are replaced by an equivalent porous media having the flow impedance properties computed using a conservatively articulated 3-D CFD model [4.4.2]. The space between the BWR fuel channel and the storage cell is represented as an open flow annulus. The fuel channel is also explicitly modeled. The porous medium within the channel space is also referred to as the “rodded region”. The fuel assembly is assumed to be positioned coaxially with respect to its storage cell. The 3-D model of an MPC-68 storage cell occupied with channeled BWR fuel is shown in Figure 4.4.4.

In the case of the PWR CSF, the porous medium extends to the entire cross-section of the storage cell. As described in [4.4.2], the CFD model for both the BWR and PWR case is prepared for the Design Basis fuel in comprehensive detail, which includes grid straps, BWR water rods and PWR guide and instrument tubes (assumed to be plugged for conservatism).

- c. Every MPC fuel storage cell is assumed to be occupied by design basis PWR or BWR fuel assemblies specified in Chapter 2 (Table 2.1.5). The in-plane thermal conductivity of the design basis fuel assemblies are obtained using ANSYS [4.1.1] finite element models of an array of fuel rods enclosed by a square box. Radiation heat transfer from solid surfaces (cladding and box walls) are enabled in these models. Using these models the effective conduction-radiation conductivities are obtained and reported in Table 4.4.2. For heat transfer in the axial direction an area weighted mean of cladding and helium conductivities are computed (see Table 4.4.2). Axial conduction heat transfer in the fuel pellets and radiation heat dissipation in the axial direction are conservatively ignored. Thus, the thermal

conductivity of the rodded region, like the porous media simulation for helium flow, is represented by a 3-D continuum having effective planar and axial conductivities.

- d. The internals of the MPC, including the basket cross section, bottom mouse holes, top plenum, and circumferentially irregular downcomer are modeled explicitly. For simplicity, the mouse holes are modeled as rectangular openings with understated flow area.
- e. The inlet and outlet vents in the HI-STORM overpack are modeled explicitly to incorporate any effects of non-axisymmetry of inlet air passages on the system's thermal performance.
- f. The air flow in the HI-STORM/MPC annulus is simulated by a  $k-\omega$  turbulence model with the transitional option enabled.

The 3-D model described above is illustrated in the cross section for the MPC-68 in Figure 4.4.3. A closeup of the fuel cell spaces which explicitly include the channel-to-cell gap in the 3-D model is shown in Figure 4.4.4. The principal 3-D modeling conservatisms are listed below:

- 1) The storage cell spaces are loaded with design basis fuel having the highest axial flow resistance (See Table 2.1.5 ).
- 2) Each storage cell is generating heat at its limiting value under uniform or regionalized storage scenarios as defined in Chapter 2, Section 2.1.
- 3) Axial dissipation of heat by the fuel pellets is neglected.
- 4) Axial dissipation of heat by radiation in the fuel bundle is neglected.
- 5) The fuel assembly channel length for BWR fuel is overstated.
- 6) The most severe environmental factors for long-term normal storage - ambient temperature of 80°F and 10CFR71 insolation levels - were coincidentally imposed on the system.
- 7) The absorbtivity of the external surfaces of the HI-STORM is conservatively assumed to be unity.
- 8) To understate MPC internal convection heat transfer, the helium pressure is understated.
- 9) No credit is taken for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports or between the MPC basket wall and the basket shims.
- 10) Heat dissipation by fuel basket peripheral supports is neglected.
- 11) Fuel basket and MPC shell emissivities are understated (see Table 4.2.4).
- 12) The  $k-\omega$  model used for simulating the HI-STORM annulus flow yields uniformly conservative results [4.1.6].

The effect of crud resistance on fuel cladding surfaces has been evaluated and found to be negligible. The evaluation assumes a thick crud layer (130  $\mu\text{m}$ ) with a bounding low conductivity (conductivity of helium). The crud resistance increases the clad temperature by a very small amount ( $\sim 0.1^\circ\text{F}$ ). Accordingly this effect is neglected in the thermal evaluations.

#### 4.4.1.2 Fuel Assembly 3-Zone Flow Resistance Model

The HI-STORM System is evaluated for storage of bounding PWR (W-17x17) and BWR (GE-10x10) fuel assemblies. During fuel storage helium enters the MPC fuel cells from the bottom plenum and flows upwards through the open spaces in the fuel storage cells and exits in the top plenum. Because of the low flow velocities the helium flow in the fuel storage cells and MPC spaces is in the deep laminar regime ( $Re < 100$ ). The bottom and top plenums are essentially open spaces engineered in the fuel basket ends to facilitate helium circulation. In the case of BWR fuel storage, a channel enveloping the fuel bundle divides the flow in two parallel paths. One flow path is through the in-channel or rodded region of the storage cell and the other flow path is in the square annulus area outside the channel. In the global thermal modeling of the HI-STORM System the following approach is adopted:

- (i) In BWR fueled MPCs an explicit channel-to-cell gap is modeled.
- (ii) The fuel assembly enclosed in a square envelope (fuel channel for BWR fuel or fuel storage cell for PWR fuel) is replaced by porous media with equivalent flow resistance.

The above modeling approach is illustrated in Figure 4.4.4.

In the FLUENT program, porous media flow resistance is modeled as follows:

$$\Delta P = D\mu VL \quad (\text{Eq. 1})$$

where  $\Delta P$  is the hydraulic pressure loss,  $D$  is the flow resistance coefficient,  $\mu$  is the fluid viscosity,  $V$  is the superficial fluid velocity and  $L$  is the porous media length. In the HI-STORM thermal models the fuel storage cell length between the bottom and top plenums<sup>12</sup> is replaced by porous media. As discussed below the porous media length is partitioned in three zones with discrete flow resistances.

To characterize the flow resistance of fuel assemblies inside square envelopes (fuel channel for BWR fuel or fuel storage cell for PWR fuel) 3D models of W-17x17 and GE-10x10 fuel assemblies are constructed using the FLUENT CFD program. These models are embedded with several pessimistic assumptions to overstate flow resistance. These are:

- (a) Water rods (BWR fuel) and guide tubes (PWR fuel) are assumed to be completely blocked
- (b) Fuel rods assumed to be full length
- (c) Channel length (BWR fuel) overstated
- (d) Bounding grid thickness used
- (e) Bottom fittings resistance overstated
- (f) Bottom nozzle lateral flow holes (BWR fuel) assumed to be blocked

<sup>12</sup> These are the mousehole openings at the ends of the fuel basket to facilitate helium circulation. The mouseholes are explicitly included in the 3D thermal models with an understated flow area.

Using the 3D fuel assembly models flow solutions under an impressed pressure differential between the two extremities of the fuel storage cell are computed at reference conditions (7 atmosphere absolute pressure and 450°F temperature). The results of the 3D flow solutions are post-processed as described next and equivalent porous media flow resistances obtained.

Because of the narrow flow passages in the bare rods and gridded regions of the fuel assembly the flow resistance of the fueled length to axial helium flow is greater than the flow resistance from the fuel assembly ends (bottom nozzle, top fitting, handle etc.). This physical fact is duly recognized by defining three distinct axial zones as follows:

- Zone 1: Length below the active fuel region
- Zone 2: Active fuel region
- Zone 3: Length above the active fuel region

In the 3-Zone flow resistance modeling, the flow resistance of each zone is characterized by post-processing the 3D fuel flow model solutions. For this purpose two approaches to flow resistance characterization are adopted. The first approach is the pressure drop method. This method is suitable when a zone is characterized by irregular geometries and the objective is to obtain a lumped resistance to duplicate the pressure drop. The second method is the shear stress method, which is suitable for flow zones characterized by regular geometries. For the 3-Zone flow resistance modeling the pressure drop method is adopted for the inactive regions (Zone 1 and Zone 3). The flow resistance coefficients are computed by post-processing the fuel assemblies 3D model flow solutions as follows:

- Step 1: Obtain the helium volumetric flow  $Q$  under the impressed pressure differential.
- Step 2: Compute helium superficial velocity,  $V = Q/A$  where  $A$  is the square envelope cross-sectional area.
- Step 3: Obtain the individual Zone 1 and Zone 3 lengths ( $L_1$  and  $L_3$ ) and pressure drops ( $\Delta P_1$  and  $\Delta P_3$ ) from the FLUENT solutions.
- Step 4: Compute Zone 1 and Zone 3 resistance coefficients  $D_1$  and  $D_3$  using Eq. 1,  $V$ ,  $L_1$ ,  $L_3$ ,  $\Delta P_1$  and  $\Delta P_3$  from above steps.

The shear stress method is suitable for the active fuel region (Zone 2) as this region is characterized by an ordered array of entities (rods and grids). This method uses area averaged wall shear stresses post-processed from the active region (Zone 2) of the fuel assembly. Using hydraulic flow principles the wall shear stresses are mapped to flow resistance coefficients. To account for geometric discontinuities the active fuel region is sliced in a suitable number of constant geometry (bare rods and grids) sub-regions. Based on the fuel bundle layout, a total of 17 slices are identified for GE-10x10 fuel and 20 slices for W-17x17 fuel. In each sub-region an area averaged shear stress over all wetted surfaces (fuel rods, non-fuel rods, square envelope and grids) is post-processed and flow resistance coefficients of each slice are computed. The flow resistance of Zone 2 is obtained by computing the length-weighted average of the slice resistance coefficients.

#### 4.4.2 [deleted]

#### 4.4.3 Test Model

The HI-STORM thermal analysis is performed on the FLUENT [4.1.2] Computational Fluid Dynamics (CFD) program. To ensure a high degree of confidence in the HI-STORM thermal evaluations, the FLUENT code is benchmarked using data from tests conducted with casks loaded with irradiated SNF ([4.1.3],[4.1.7]). The benchmark work is archived in QA validated Holtec reports ([4.1.5],[4.1.6]). These evaluations show that the FLUENT solutions are conservative in all cases. In view of these considerations, additional experimental verification of the thermal design is not necessary.

#### 4.4.4 Maximum and Minimum Temperatures

##### 4.4.4.1 Maximum Temperatures

The 3-D model from the previous subsection is used to determine temperature distributions under long-term normal storage conditions for an array of cases covering PWR and BWR fuel storage in uniform and regionalized loading configurations. For this purpose one bounding MPC design in each of the two fuel classes – MPC-68 for BWR and MPC-32 for PWR – are analyzed and results obtained and summarized in this subsection. For a bounding evaluation the MPCs are assumed to be emplaced in a limiting overpack (HI-STORM 100S Version B).

The HI-STORM 100S Version B is the limiting overpack by virtue of the inlet and outlet vents design. Compared to two other overpack designs (i.e., HI-STORM 100 and HI-STORM 100S), the HI-STORM 100S Version B has smaller inlet and outlet vents. Thus Version B vent airflow resistances are bounding. Also, the HI-STORM 100S Version B is the shortest of the overpacks. This reduces the chimney height which minimizes the driving head for air flow. Because the HI-STORM 100S Version B will have the least cooling air flow, it will yield bounding results.

A cross-reference of HI-STORM thermal analyses is provided in Table 4.4.5. Under regionalized loading, an array of runs covering a range of regionalized storage configurations specified in Chapter 2 ( $X=0.5$  to  $X=3$ ) are analyzed. The results are graphed in Figures 4.4.6 and 4.4.7 for PWR and BWR fuel storage respectively. Based on this array of runs the fuel storage condition corresponding to  $X = 0.5$  is determined to be limiting for both PWR and BWR MPCs. Accordingly HI-STORM MPC and overpack temperatures are reported for this storage condition in Tables 4.4.6 and 4.4.7.

Damaged fuel is canestirezed in damaged fuel containers (DFCs) before long-term storage. Each MPC type has designated locations for placement of DFCs, as described in Chapter 2. Particularly,

the DFCs are placed for storage in basket peripheral locations. The presence of DFCs impedes helium flow through it. However, since the DFCs are placed in the cold peripheral locations, they do not control the peak cladding temperature. Moreover, as a substantial fraction of basket cells are occupied by intact fuel, the overall effect of DFC fuel storage on the heat dissipation from the basket is small. To account for its impact, a heat load penalty described in Tables 4.4.3 and 4.4.4 is placed on the DFC locations under both uniform and regionalized heat load patterns.

It should be noted that the 3-D FLUENT cask model incorporates the effective conductivity of the fuel assembly submodel. Therefore the FLUENT models report the peak temperature in the fuel storage cells. Thus, as the fuel assembly models include the fuel pellets, the FLUENT calculated peak temperatures are actually peak pellet centerline temperatures which bound the peak cladding temperatures with a margin.

The following observations can be derived by inspecting the temperature field obtained from the thermal models:

- The fuel cladding temperatures are below the regulatory limit (ISG-11 [4.1.4]) under all storage scenarios (uniform and regionalized) in all MPCs.
- The maximum temperature of the basket structural materials are within their design limits.
- The maximum temperature of the neutron absorbers are below their design limits.
- The maximum temperatures of the MPC pressure boundary materials are below their design limits.
- The maximum temperatures of concrete is within the guidance of the governing ACI Code (see Table 4.3.1).

The above observations lead us to conclude that the temperature field in the HI-STORM System with a loaded MPC containing heat emitting SNF complies with all regulatory temperature limits. In other words, the thermal environment in the HI-STORM System is in compliance with Chapter 2 Design Criteria.

As discussed in Chapter 1, the HI-STORM 100S and HI-STORM 100S Version B have an optional configuration with cylindrical guides. The optional stainless steel guide tubes are much shorter than the channels and the total surface area of the tubular guides is smaller than that of the channels. This reduces friction loss in the air flow and improves the natural convection heat dissipation by air through the annular gap between MPC and HI-STORM. Therefore, the HI-STORM 100S and HI-STORM 100S Version B with the optional cylindrical guide configuration complies with all regulatory temperature limits.

#### 4.4.4.2 Minimum Temperatures



In Table 2.2.2 of this report, the minimum ambient temperature condition for the HI-STORM storage overpack and MPC is specified to be  $-40^{\circ}\text{F}$ . If, conservatively, a zero decay heat load with no solar input is applied to the stored fuel assemblies, then every component of the system at steady state would be at a temperature of  $-40^{\circ}\text{F}$ . Low service temperature ( $-40^{\circ}\text{F}$ ) evaluation of the HI-STORM is provided in Chapter 3. All HI-STORM storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode at this minimum temperature condition.

#### 4.4.4.3 Effects of Elevation

The reduced ambient pressure at site elevations significantly above the sea level will act to reduce the ventilation air mass flow, resulting in a net elevation of the peak cladding temperature. However, the ambient temperature (i.e., temperature of the feed air entering the overpack) also drops with the increase in elevation. Because the peak cladding temperature also depends on the feed air temperature (the effect is one-for-one within a small range, i.e.,  $1^{\circ}\text{F}$  drop in the feed air temperature results in  $\sim 1^{\circ}\text{F}$  drop in the peak cladding temperature), the adverse ambient pressure effect of increased elevation is partially offset by the ambient air temperature decrease. The table below illustrates the variation of air pressure and corresponding ambient temperature as a function of elevation.

Elevation (ft)	Pressure (psia)	Ambient Temperature Reduction versus Sea Level
Sea Level (0)	14.70	$0^{\circ}\text{F}$
2000	13.66	$7.1^{\circ}\text{F}$
4000	12.69	$14.3^{\circ}\text{F}$

A survey of the elevation of nuclear plants in the U.S. shows that nuclear plants are situated near about sea level or elevated slightly ( $\sim 1000$  ft). The effect of the elevation on peak fuel cladding temperatures is evaluated by performing calculations for a HI-STORM 100 System situated at an elevation of 1500 feet. At this elevation the ambient temperature would decrease by approximately  $5^{\circ}\text{F}$  (See Table above). The peak cladding temperatures are calculated for a bounding configuration (non-uniform storage at  $X = 0.5$ ), and conservatively assuming no reduction in ambient temperature using the 3D model described in Subsection 4.4.1.1 and compared to the sea level conditions. The results are given in the following table.

MPC Design	PCT at Sea Level	PCT at 1500 feet
MPC-32 PWR	$711.4^{\circ}\text{F}$	$723.8^{\circ}\text{F}$
MPC-68 BWR	$697.1^{\circ}\text{F}$	$718.2^{\circ}\text{F}$

These results show that the PCT, including the effects of site elevation, continues to be well below the regulatory cladding temperature limit of  $752^{\circ}\text{F}$ . In light of the above evaluation, it is not necessary to place any ISFSI elevation constraints for HI-STORM deployment at elevations up to

1500 feet. If, however, an ISFSI is sited at an elevation greater than 1500 feet, the effect of altitude on the PCT shall be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient conditions.

#### 4.4.5 Maximum Internal Pressure

##### 4.4.5.1 MPC Helium Backfill Pressure

For design basis heat load, the helium backfill shall be sufficient to produce the required operating pressure of 7 atmospheres (absolute) during normal storage at reference conditions (See Table 4.0.1). Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain helium backfill pressure specified at a reference temperature (70°F). The minimum backfill pressure to attain this operating pressure for each MPC type is provided in Table 4.4.11. An upper limit on the helium backfill pressure corresponds to the design pressure of the MPC vessel (Table 2.2.1). The upper limit on the backfill pressure is also reported in Table 4.4.11. To bound the minimum and maximum backfill pressures listed in Table 4.4.11 with margin, a helium backfill specification is set forth in Table 4.4.12. These values support the technical specification of the system for the design basis heat load of the MPC.

In addition the technical specifications allow for using a wider range on the backfill pressure if the heat load of the MPC is less than 28.74 kW. The minimum of this range corresponds to an operating pressure of 5 atm. The heat loads for this condition are provided in Table 2.1.31. If the MPC is loaded such that Table 2.1.31 is satisfied, the lower pressure range in the technical specifications may be used.

It is conservative to backfill the MPC to the higher pressure range regardless of MPC heat load.

Two methods are available for ensuring that the appropriate quantity of helium has been placed in an MPC:

- i. By pressure measurement
- ii. By measurement of helium backfill volume (in standard cubic feet)

The direct pressure measurement approach is more convenient if FHD is used to dry the MPC cavity. In this case, a certain quantity of helium is already in the MPC. Because the helium is mixed inside the MPC during the FHD operation, the temperature of the helium gas at the exit of the MPC, along with the pressure provides a reliable means to compute the inventory of helium. A shortfall or excess of helium is adjusted by a calculated raising or lowering of the MPC pressure such that the MPC backfill pressure is within the specifications.

When vacuum drying is used to dry the MPC cavity it is more convenient to fill the MPC by introducing a known quantity of helium by measuring the quantity of helium introduced using a calibrated mass flow meter or other measuring apparatus. The required quantity of helium is

computed by the product of net free volume and helium specific volume at the reference temperature (70°F) and a target pressure that lies in the mid-range of the specifications.

The net free volume of the MPC is obtained by subtracting B from A, where

A = MPC cavity volume in the absence of fuel and any DFC and/or NFH as computed from nominal design dimensions.

B = Total volume of the fuel and any DFC and/or NFH as computed from nominal design dimensions

#### 4.4.5.2 MPC Pressure Calculations

The MPC pressure calculations are performed using the reference conditions in Table 4.0.1 since this condition always results in the highest pressure in the MPC and bounds the allowed pressure range for the lower MPC heat load as described above.

During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature. The gas pressure inside the MPC will also increase with rising temperature to its maximum operating pressure. The pressure rise is determined using the ideal gas law.

Table 4.4.8 presents a summary of the minimum MPC free volumes determined for each MPC type (MPC-24, MPC-68, MPC-32, and MPC-24E). The MPC maximum gas pressure is computed for a postulated release of fission product gases from fuel rods into this free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. Based on fission gases release fractions in NUREG-1536 [4.4.1] and the net free volume and initial fill gas pressure of the fuel rods, maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are calculated and provided in Table 4.4.9. The maximum computed gas pressures reported in Table 4.4.9 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Table 2.2.1.

#### Evaluation of Non-Fuel Hardware

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the PWR baskets influences the MPC internal pressure through two distinct effects. The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures as well as the temperature of the gas filling the space between fuel rods. The gas volume displaced by the mass of non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influence on the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STORM thermal analysis, the computed temperature field (with non-fuel hardware excluded) has been determined to provide a conservatively bounding temperature field for the PWR baskets (MPC-

24, MPC-24E, and MPC-32). The MPC cavity free space is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included. This approach ensures conservative bounding pressures.

During in-core irradiation of BPRAs, neutron capture by the B-10 isotope in the neutron absorbing material produces helium. Two different forms of the neutron absorbing material are used in BPRAs: Borosilicate glass and B<sub>4</sub>C in a refractory solid matrix (Al<sub>2</sub>O<sub>3</sub>). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin-walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operation conditions (2,300 psia and approximately 600°F coolant temperature). The B<sub>4</sub>C- Al<sub>2</sub>O<sub>3</sub> neutron absorber material is principally used in B&W and CE fuel BPA designs. The relatively low temperature of the poison material in BPA rods (relative to fuel pellets) favor the entrapment of helium atoms in the solid matrix.

Several BPA designs are used in PWR fuel that differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and W-15x15) has used 6, 12, 16, and 20 rods per assembly BPRAs and the later (W-17x17) fuel uses up to 24 rods per BPA. The BPA rods in the older fuel are much larger than the later fuel and, therefore, the B-10 isotope inventory in the 20-rod BPRAs bounds the newer W-17x17 fuel. Based on bounding BPA rods internal pressure, a large hypothetical quantity of helium (7.2 g-moles/BPA) is assumed to be available for release into the MPC cavity from each fuel assembly in the PWR baskets. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.9.

#### 4.4.6 Engineered Clearances to Eliminate Thermal Interferences

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal stresses in load bearing members, the HI-STORM System is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansion calculations are performed to demonstrate that engineered gaps in the HI-STORM System are adequate to accommodate thermal expansion of the fuel basket and MPC.

The HI-STORM System is engineered with gaps for the fuel basket and MPC to expand thermally without restraint of free end expansion. Differential thermal expansion of the following gaps are evaluated:

- a. Fuel Basket-to-MPC Radial Gap
- b. Fuel Basket to MPC Axial Gap
- c. MPC-to-Overpack Radial Gap
- d. MPC-to-Overpack Axial Gap

To demonstrate that the fuel basket and MPC are free to expand without restraint, it is required to show that differential thermal expansion from fuel heatup is less than the as-built gaps that exist in the HI-STORM System. For this purpose a suitably bounding temperature profile ( $T(r)$ ) for the fuel basket is established in Figure 4.4.5 wherein the center temperature (TC) is set at the limit ( $752^{\circ}\text{F}$ ) for fuel cladding (conservatively bounding assumption) and the basket periphery (TP) conservatively postulated at an upperbound of  $610^{\circ}\text{F}$  (see Table 4.4.6 for the maximum fuel and basket periphery temperatures). To maximize the fuel basket differential thermal expansion, the basket periphery-to-MPC shell temperature difference is conservatively maximized ( $\Delta T = 175^{\circ}\text{F}$ ). From the bounding temperature profile  $T(r)$  and  $\Delta T$ , the mean fuel basket temperature ( $T1$ ) and MPC shell temperature ( $T2$ ) are computed as follows:

$$T1 = \frac{\int_0^1 rT(r)dr}{\int_0^1 rdr} = 676^{\circ}\text{F}$$

$$T2 = TP - \Delta T = 425^{\circ}\text{F}$$

The differential radial growth of the fuel basket ( $Y1$ ) from an initial reference temperature ( $T_0 = 70^{\circ}\text{F}$ ) is computed as:

$$Y1 = R \times [A1 \times (T1 - T_0) - A2 \times (T2 - T_0)]$$

where:

$R$  = Basket radius (conservatively assumed to be the MPC radius)

$A1, A2$  = Coefficients of thermal expansion for fuel basket and MPC shell at  $T1$  and  $T2$  respectively for Alloy X (Chapter 1 and Table 3.3.1)

For computing the relative axial growth of the fuel basket in the MPC, bounding temperatures for the fuel basket (TC) and MPC shell temperature  $T2$  utilized above are adopted. The differential expansion is computed by a formula similar to the one for radial growth after replacing  $R$  with basket height ( $H$ ), which is conservatively assumed to be that of the MPC cavity.

For computing the radial and axial MPC-to-overpack differential expansions, the MPC shell is postulated at its design temperature (Chapter 2, Table 2.2.3) and thermal expansion of the overpack is ignored. Even with the conservative computation of the differential expansions in the manner of the foregoing, it is evident from the data presented in Table 4.4.10 that the differential expansions are a fraction of the nominal cold gaps presented on the corresponding drawings in Section 1.5.

#### 4.4.7 Evaluation of System Performance for Normal Conditions of Storage

The HI-STORM System thermal analysis is based on a detailed and complete heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and

overpack. The thermal model incorporates conservative features that render the results for long-term storage to be extremely conservative.

Temperature distribution results obtained from this highly conservative thermal model show that the maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be much greater due to the conservative assumptions incorporated in the analysis. The long-term impact of decay heat induced temperature levels on the HI-STORM System structural and neutron shielding materials is considered to be negligible. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep-induced degradation. Furthermore, stresses induced due to imposed temperature gradients are within Code limits (See Structural Evaluation Chapter 3). Therefore, it is concluded that the HI-STORM System thermal design is in compliance with 10CFR72 requirements.

Table 4.4.1

**EFFECTIVE CONDUCTIVITY OF THE COMPOSITE FUEL BASKET WALLS**  
(Btu/hr-ft-°F)

Temperature (°F)	MPC-32		MPC-24/MPC-24E*		MPC-68	
	Thru-Thickness Direction	Parallel Plates Direction	Thru-Thickness Direction	Parallel Plates Direction	Thru-Thickness Direction	Parallel Plates Direction
200	6.000	14.65	5.676 4.800**	13.85 11.17**	5.544	12.06
450	7.260	16.12	6.864 5.808**	15.32 12.54**	6.708	13.45
700	8.316	17.20	7.884 6.672**	16.44 13.62**	7.680	14.52

\* Lowerbound values reported.  
\*\* Effective conductivities of basket peripheral panels.

Table 4.4.2

**LIMITING EFFECTIVE CONDUCTIVITIES OF THE RODDED REGION**  
(Btu/hr-ft-°F)

Temperature (°F)	PWR Fuel		BWR FUEL	
	Planar	Axial	Planar	Axial
200	0.257	0.753	0.282	0.897
450	0.406	0.833	0.425	0.988
700	0.604	0.934	0.606	1.104

Table 4.4.3

MAXIMUM PERMISSIBLE HEAT LOAD FOR LOCATIONS WITH DFCs UNDER  
UNIFORM LOADING

MPC Type	Decay Heat for Locations with Damaged Fuel Assemblies and Fuel Debris (kW)
MPC-24E/24EF	1.114
MPC-32/32F	0.718
MPC-68/68FF/68M	0.393

Table 4.4.4

MAXIMUM PERMISSIBLE HEAT LOAD FOR LOCATIONS WITH DFCs UNDER  
REGIONALIZED LOADING

MPC Type	Decay Heat for Locations with Damaged Fuel Assemblies and Fuel Debris (kW)
MPC-24E/24EF	$0.75 \cdot q_2$ (Note 1)
MPC-32/32F	$0.65 \cdot q_2$ (Note 1)
MPC-68/68FF/68M	$0.75 \cdot q_2$ (Note 1)
Note 1: $q_2$ is the maximum permissible heat load allowed for intact fuel in Region 2.	



Table 4.4.5

## MATRIX OF HI-STORM SYSTEM THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Long Term Normal	Ambient	SS	N <sub>T</sub> , Q <sub>D</sub> , ST, SC, I <sub>O</sub>	4.4.4
2	Off-Normal Environment	Ambient	SS(B)	O <sub>T</sub> , Q <sub>D</sub> , ST, SC, I <sub>O</sub>	4.6.1
3	Extreme Environment	Ambient	SS(B)	E <sub>T</sub> , Q <sub>D</sub> , ST, SC, I <sub>O</sub>	4.6.2
4	Partial Ducts Blockage	Ambient	SS(B)	N <sub>T</sub> , Q <sub>D</sub> , ST, SC, I <sub>1/2</sub>	4.6.1
5	All Inlets Ducts Blocked	Overpack	TA	N <sub>T</sub> , Q <sub>D</sub> , ST, SC, I <sub>C</sub>	4.6.2
6	Fire Accident	Overpack	TA	Q <sub>D</sub> , F	4.6.2
7	Burial Under Debris	Overpack	AH	Q <sub>D</sub>	4.6.2

Legend:N<sub>T</sub> - Maximum Annual Average (Normal) Temperature (80°F)O<sub>T</sub> - Off-Normal Temperature (100°F)E<sub>T</sub> - Extreme Hot Temperature (125°F)Q<sub>D</sub> - Design Basis Maximum Heat Load

SS - Steady State

SS(B) - Bounding Steady State

TA - Transient Analysis

AH - Adiabatic Heating

I<sub>O</sub> - All Inlet Ducts OpenI<sub>1/2</sub> - Half of Inlet Ducts OpenI<sub>C</sub> - All Inlet Ducts Closed

ST - Insolation Heating (Top)

SC - Insolation Heating (Curved)

F - Fire Heating (1475°F)

Table 4.4.6

MAXIMUM MPC TEMPERATURES FOR LONG-TERM NORMAL STORAGE  
CONDITION<sup>13</sup>

Component	Temperature, °F	
	MPC-32	MPC-68
Fuel Cladding	<b>711</b>	697
MPC Basket	<b>708</b>	692
Basket Periphery	<b>604</b>	566
MPC Shell	<b>469</b>	452

Table 4.4.7

BOUNDING HI-STORM OVERPACK TEMPERATURES FOR LONG-TERM NORMAL  
STORAGE<sup>14</sup>

Component	Local Section Temperature <sup>15</sup> , °F
Inner shell	322
Outer shell	174
Lid bottom plate	302
Lid top plate	190
Overpack Body Concrete	248
Overpack Lid Concrete	246
Area Averaged Air outlet <sup>16</sup>	235

- 13 The temperatures reported in this table for the bounding fuel storage configuration (regionalized storage at  $X = 0.5$ ) are below the design temperatures specified in Chapter 2, Table 2.2.3. Results of the bounding canister (MPC-32) are highlighted in bold.
- 14 The temperatures reported in this table (all for MPC-32 at  $X = 0.5$ ) are below the design temperatures specified in Chapter 2, Table 2.2.3.
- 15 Section temperature is defined as the through-thickness average temperature.
- 16 Reported herein for the option of temperature measurement surveillance of outlet ducts air temperature as set forth in the Technical Specifications.

Table 4.4.8

## SUMMARY OF MPC FREE VOLUME CALCULATIONS

Item	Volume (MPC-24) [ft <sup>3</sup> ]	Volume (MPC-24E) [ft <sup>3</sup> ]	Volume (MPC-32) [ft <sup>3</sup> ]	Volume (MPC-68) [ft <sup>3</sup> ]
Cavity Volume	367.9	367.9	367.9	367.3
Basket Metal Volume	44.3	51.4	24.9	34.8
Bounding Fuel Assemblies Volume	78.8	78.8	105.0	93.0
Basket Supports (or Basket shims) and Fuel Spacers Volume	6.1	6.1	9.0	11.3
Net Free Volume*	238.7 (6,759 liters)	231.6 (6,558 liters)	229 (6,484 liters)	228.2 (6,462 liters)
<p>* Net free volumes are obtained by subtracting basket, fuel, supports and spacers metal volume from cavity volume. The free volumes used for MPC internal pressure calculations are conservatively understated.</p>				

Table 4.4.9

## SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE\*

Condition	MPC-24*** (psig)	MPC-24E*** (psig)	MPC-32 (psig)	MPC-68 (psig)
Initial backfill** (at 70°F)	48.5	48.5	48.5	48.5
Normal: intact rods	99.0	99.0	99.0	96.8
1% rods rupture	100.0	99.7	99.7	97.2
Off-Normal (10% rods rupture)	106.0	106.2	108.7	101.2
Accident (100% rods rupture)	169.3	171.5	196.4	141.1
<p>* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.</p> <p>** Conservatively assumed at the Tech. Spec. maximum value (See Table 4.4.12).</p> <p>*** Pressure calculations use the bounding MPC-32 temperature field.</p>				

Table 4.4.10

## SUMMARY OF HI-STORM DIFFERENTIAL THERMAL EXPANSIONS

Gap Description	Differential Expansion* (in)
Fuel Basket-to-MPC Radial Gap	0.095
Fuel Basket-to-MPC Axial Gap	0.487
MPC-to-Overpack Radial Gap	0.139
MPC-to-Overpack Minimum Axial Gap	0.771
* The differential expansion values reported in this table are bounded by the nominal cold gaps presented on the drawings in Section 1.5.	

Table 4.4.11

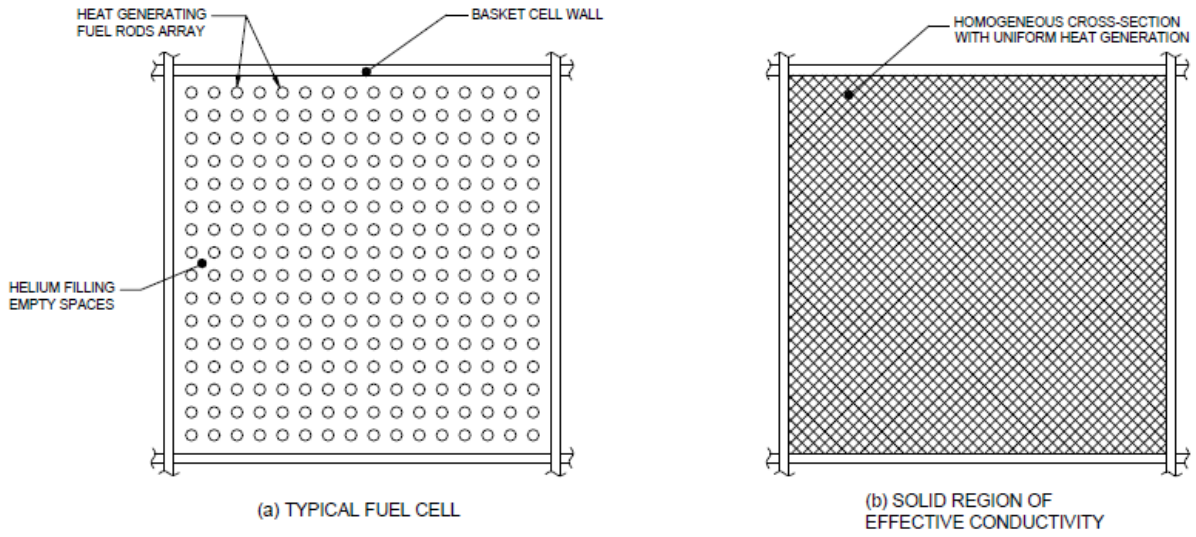
## THEORETICAL LIMITS\* OF MPC HELIUM BACKFILL PRESSURE\*\*

MPC	Minimum Backfill Pressure (psig)	Maximum Backfill Pressure (psig)
MPC-32/24/24E	44.1	49.1
MPC-68	45.2	50.3
* The helium backfill pressures are set forth in the Technical Specifications with a margin (See Table 4.4.12).		
** The pressures tabulated herein are at a reference gas temperature of 70°F.		

Table 4.4.12

## MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS

Item	Specification
Minimum Pressure	45.5 psig @ 70°F Reference Temperature
Maximum Pressure	48.5 psig @ 70°F Reference Temperature

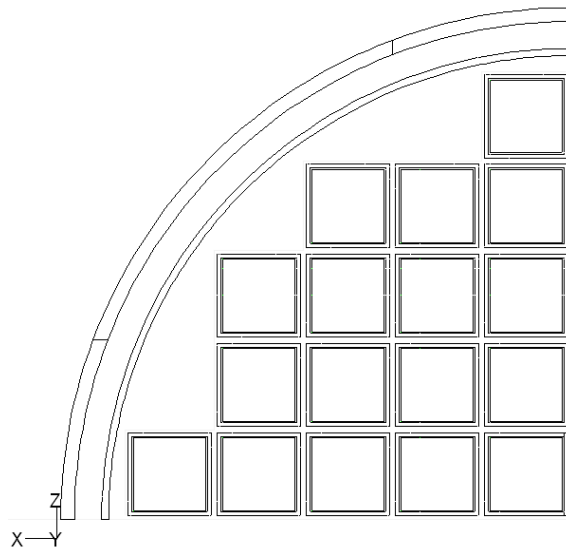


**FIGURE 4.4.1: HOMOGENIZATION OF THE STORAGE CELL CROSS-SECTION**

FIGURE 4.4.2

[Intentionally Deleted]

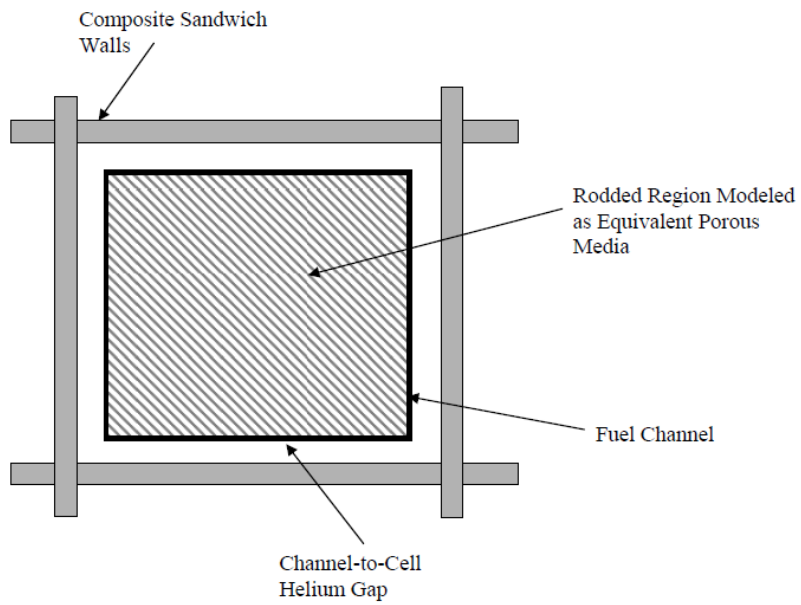




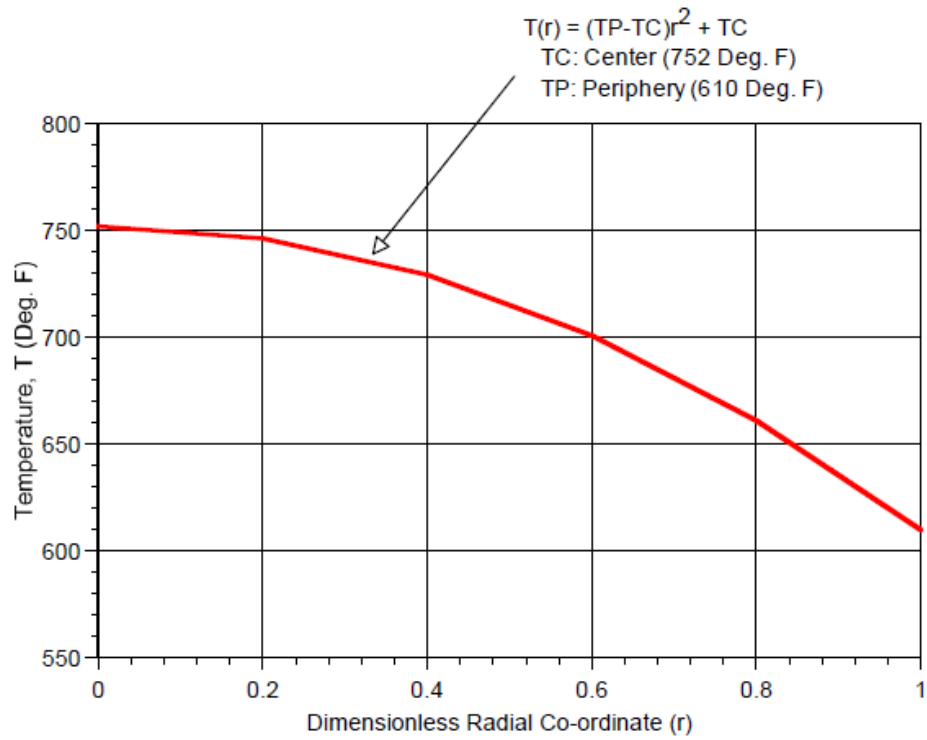
Grid

Dec 23, 2004  
FLUENT 6.1 (3d, dp, segregated, ske)

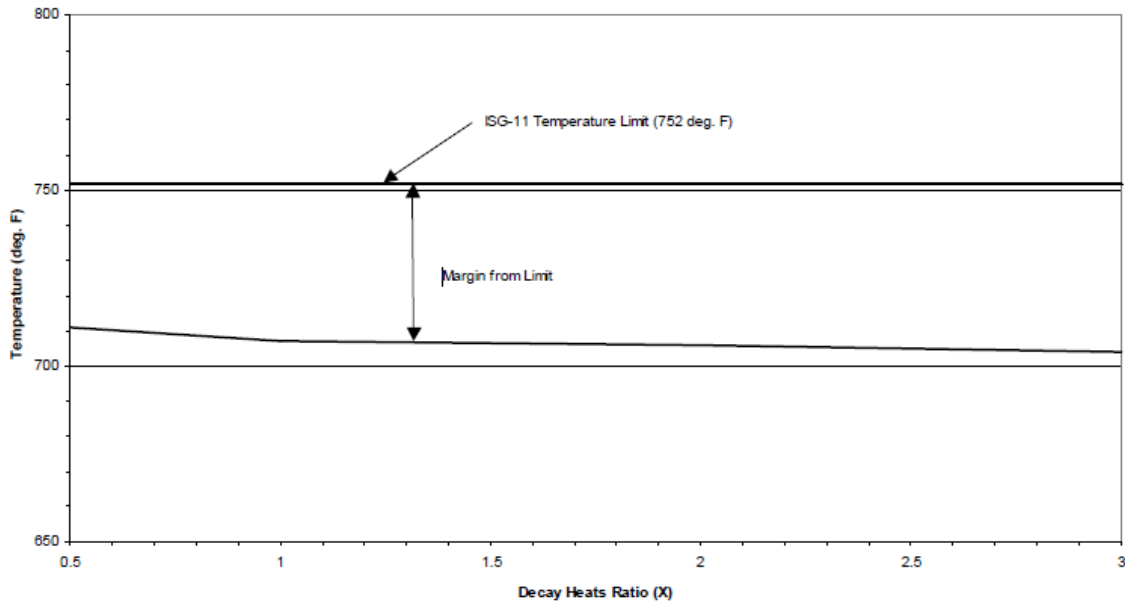
**FIGURE 4.4.3: PLANAR VIEW OF HI-STORM MPC-68 QUARTER SYMMETRIC 3-D MODEL**



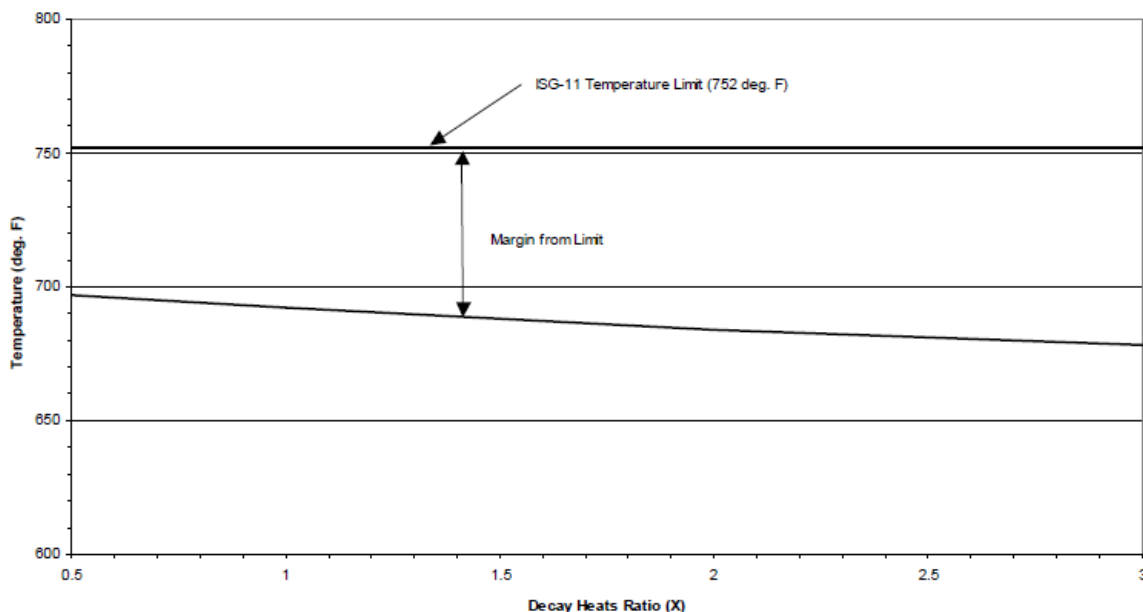
**FIGURE 4.4.4: CLOSEUP VIEW OF THE MPC-68 CHANNELED FUEL CELL SPACES**



**FIGURE 4.4.5: BOUNDING BASKET TEMPERATURE PROFILE FOR DIFFERENTIAL EXPANSION**



**FIGURE 4.4.6: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-32)**



**FIGURE 4.4.7: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-68)**

#### 4.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

Prior to placement in a HI-STORM overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transported to the HI-STORM module. In the unlikely event that the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM overpacks or between a HI-STAR transport overpack and a HI-STORM storage overpack must be carried out in an assuredly safe manner. All of the above operations, henceforth referred to as “short term operations”, are short duration events that would likely occur no more than once or twice for an individual MPC.

The device central to all of the above operations is the HI-TRAC transfer cask that, as stated in Chapter 1, is available in two anatomically similar weight ratings (100- and 125-ton). Two different versions of the 100 ton and the 125 ton HI-TRAC, the classical version and the version D, are available for use during fuel transfer operations. The HI-TRAC transfer cask is a short-term host for the MPC; therefore it is necessary to establish that, during all thermally challenging operation events involving either the 100-ton or 125-ton versions of the HI-TRAC, the permissible temperature limits presented in Section 4.3 are not exceeded. The following discrete thermal scenarios, all of short duration, involving the HI-TRAC transfer cask, have been identified as warranting thermal analysis.

- i. Post-Loading Wet Transfer Operations
- ii. MPC Cavity Vacuum Drying
- iii. Normal Onsite Transport in a Vertical Orientation
- iv. MPC Cooldown and Reflood for Unloading Operations

Onsite transport of the MPC occurs with the HI-TRAC in the vertical orientation, which preserves the thermosiphon action within the MPC. To avoid excessive temperatures, transport with the HI-TRAC in the horizontal condition is generally not permitted. However, it is recognized that an occasional downending of a HI-TRAC may become necessary to clear an obstruction such as a low egress bay door opening. In such a case the operational imperative for HI-TRAC downending must be ascertained and the permissible duration of horizontal configuration must be established on a site-specific basis and compliance with the thermal limits of ISG-11 [4.1.4] must be demonstrated as a part of the site-specific safety evaluation.

The fuel handling operations listed above place a certain level of constraint on the dissipation of heat from the MPC relative to the normal storage condition. Consequently, for some scenarios, it is necessary to provide additional cooling when decay heat loads are such that short-term cladding temperature limits would be exceeded. For such situations, the Supplemental Cooling System (SCS) is required to provide additional cooling during short term operations. The SCS is required by the CoC for any MPC when the MPC heat load is such that short-term cladding temperature limits would be exceeded. The specific design of an SCS must accord with site-specific needs and resources, including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the SCS are satisfied by plant-specific designs are set forth in Appendix 2.C.

#### 4.5.1 HI-TRAC Thermal Model

The HI-TRAC transfer cask is used to load and unload the HI-STORM concrete storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI pad. Section views of the HI-TRAC have been presented in Chapter 1. Within a loaded HI-TRAC, heat generated in the MPC is transported from the contained fuel assemblies to the MPC in the manner described in Section 4.4. From the outer surface of the MPC to the ambient air, heat is transported by a combination of conduction, thermal radiation and natural convection. For evaluation of the thermal state of a loaded canister during all short-term operations, the three dimensional (3D) thermal model of the MPC described in Section 4.4 is utilized.

All FLUENT thermal analyses to establish margins of safety are carried out for the MPC model that yields the highest peak cladding temperature and MPC cavity pressure under the long term storage condition as discussed in Section 4.4. The above criterion identifies MPC-32 under regionalized fuel loading with  $X = 0.5$  and  $X=3$  as the governing cases.

Two HI-TRAC transfer cask designs, namely, the 125-ton and the 100-ton versions, are developed for onsite handling and transport, as discussed in Chapter 1. The two designs are principally different in terms of lead thickness and the thickness and number of the heat dissipating ribs (radial connectors) in the water jacket region. The aggregate heat dissipation by the ribs is defined by the

product of the number of radial ribs,  $N$  and thickness,  $t_r$ . The numerical model developed for HI-TRAC thermal characterization conservatively accounts for these differences by applying the higher lead thickness and constructing the water jacket region having the lowest product of  $N$  and  $t_r$ . In this manner, the HI-TRAC thru-wall resistance to heat transfer is overestimated, yielding higher MPC internal and fuel cladding temperatures.

Transport of heat within HI-TRAC occurs through multiple concentric layers of air, steel and shielding materials. A small gap exists between the outer surface of the MPC and the inner surface of the HI-TRAC overpack. Heat is transported across this gap by the parallel mechanisms of natural convection, conduction and thermal radiation. Assuming that the MPC is centered and does not contact the transfer cask walls conservatively minimizes heat transport across this gap. Heat is transported through the cylindrical wall of the HI-TRAC transfer cask by conduction through successive layers of steel, lead, and steel. A water jacket, which provides neutron shielding for the HI-TRAC transfer cask, surrounds the cylindrical steel wall. The water jacket is essentially an array of carbon steel radial ribs with welded, connecting enclosure plates. Heat is dissipated by conduction and natural convection in the water cavities and by conduction in the radial ribs. Heat is passively rejected to the ambient from the outer surface of the HI-TRAC transfer cask by natural convection and thermal radiation.

The HI-TRAC transfer cask thermal analysis is based on a 3D FLUENT model that incorporates several conservative features, namely:

- i. A constant solar flux is assumed with maximum permissible heat load and asymptotic steady state conditions to yield the most adverse temperature field in the cask. A theoretically bounding solar absorptivity of 1.0 is applied to all exposed surfaces.
- ii. Air motion in the HI-TRAC annulus is explicitly modeled. The MPC is assumed to be concentrically aligned with the cask cavity and the annulus is filled with air.
- iii. Although the HI-TRAC transfer cask baseplate is in contact with supporting surfaces, for conservatism, an insulated boundary condition is applied to the HI-TRAC baseplate.
- iv. The HI-TRAC transfer cask fluid columns in the water jacket and the open air volume above the MPC are conservatively assumed to remain in the laminar flow regime.
- v. The annular gap between the HI-TRAC transfer cask and MPC shrinks under heat up to operating temperatures. The enhancement of heat transfer due to the gap reduction is conservatively neglected.
- vi. Buoyancy driven motion of air above the MPC is included in the thermal model.
- vii. Radiation heat transfer is simulated by the more robust Discrete Ordinates (DO) model deployed in the HI-STAR 180 (Docket 71-9325) and HI-STORM FW (72-1032) in lieu of the DTRM model.

- viii. The rodded zone, which contains the spent fuel assemblies, is modeled as a homogeneous porous media using the flow resistance properties based on extensive CFD simulations [4.4.2] performed in the HI-STORM 100 docket and used in subsequent safety evaluations in both HI-STORM 100(Docket 72-1014) and HI-STORM FW dockets (Docket 72-1032). A recent experimental work by the Sandia National Laboratory (SNL), however, indicates that the axial flow resistance in the PWR fuel may be somewhat greater [4.5.1]. Although the SNL experimental data is not yet published by the NRC as of this writing (March, 2014), in the interest of conservatism, the higher flow resistance,  $1 \times 10^6 \text{ m}^{-2}$  for all three zones of fuel assembly, indicated by their work has been used in all of the thermal simulations for HI-TRAC [4.5.1].
- ix. The HI-TRAC external surfaces are modeled as surfaces with convection, radiative heat exchange with air and a constant maximum incident solar heat flux load. Insolation on cylindrical surfaces is conservatively based on 12-hour levels prescribed in 10CFR71 averaged on a 24-hour basis. It is considered appropriate to use the 24-hour averaged insolation level due to the large thermal inertia of HI-TRAC which precludes it from reaching a steady-state thermal condition during a twelve-hour period.

The computational fluid dynamics model of the HI-TRAC transfer cask captures all essential details of the cask body including the radial ribs, lead, steel shells and the water jacket. Figures 4.5.1 show the discretization of the cask and its enclosed MPC for FLUENT implementation.

#### 4.5.2 Time-to-Boil for a Water-Filled MPC

Fuel loading operations are conducted with the HI-TRAC transfer cask and its contents submerged in pool water. Under these conditions, the HI-TRAC transfer cask is essentially at the pool water temperature. When the HI-TRAC transfer cask and the loaded MPC under water-flooded conditions is removed from the pool and staged in an ambient air environment, the water, MPC, and HI-TRAC transfer cask metal absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the HI-TRAC transfer cask with time, starting from an initial pool water temperature. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC transfer cask.

The available time before the water in the MPC would reach boiling is computed under a conservative set of assumptions summarized below:

- i. Heat loss by natural convection and radiation from the exposed HI-TRAC surfaces to ambient air is neglected (i.e., an adiabatic heat-up calculation is performed).
- ii. The smaller of the two (i.e., 100-ton and 125-ton) HI-TRAC transfer cask metal mass is credited in the analysis. The 100-ton design has a significantly smaller quantity of metal mass, which will result in a higher rate of temperature rise.



- iii. The water mass in the MPC cavity is understated.

Table 4.5.2 summarizes the lower bound weights and thermal inertias of the constituent components in the loaded HI-TRAC transfer cask. The rate of temperature rise of the HI-TRAC transfer cask and contents during an adiabatic heat-up is given by the ratio  $Q/C$  where:

$$Q = \text{Coincident fuel decay heat in the canister (Btu/hr)}$$

$$C = \text{Thermal inertia of a loaded HI-TRAC (Btu/}^\circ\text{F) (See Table 4.5.2)}$$

Therefore, the time-to-boil,  $\tau$  is given by the simple algebraic formula  $\tau = C(212-T)/Q$  where 212°F has been set as the boiling temperature and  $T$  represents the temperature of the pool water under fuel loading operations. The time-to-boil clock starts when the HI-TRAC is no longer submerged in the pool water. Table 4.5.3 provides a summary of  $\tau$  at several representative heat loads and initial pool water temperatures. The calculation of time-to-boil for a loaded canister shall be made using the above formula. An alternate method using the FLUENT thermal model described in Section 4.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions. Principal modeling steps and acceptance criteria are defined in Table 4.5.10a.

As set forth in the HI-STORM operating procedures, in the unlikely event that the maximum allowable time provided in Table 4.5.3 is found to be insufficient to complete wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water will enter via the MPC lid drain port connection and heated water will exit from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

$$M_w = \frac{Q}{C_{pw}(T_{max} - T_{in})}$$

where:

- $M_w$  = minimum water flow rate (lb/hr)
- $C_{pw}$  = water heat capacity (Btu/lb-°F)
- $T_{max}$  = maximum MPC cavity water mass temperature (must be less than 212°F)
- $T_{in}$  = MPC water inlet temperature
- $Q$  = Coincident fuel decay heat in the canister (Btu/hr)

For example, the MPC cavity water temperature limited to 150°F, MPC water inlet temperature at 125°F and design basis maximum heat load (36.9 kW, approximately 125,908 Btu/hr), the water flow rate computes as 5038 lb/hr (10.1 gpm).

### 4.5.3 MPC Temperatures During Moisture Removal Operations

#### 4.5.3.1 Vacuum Drying Operation

The initial loading of SNF in the MPC requires that the water within the MPC be drained, fuel dried and the water replaced with helium. Vacuum drying of fuel is conducted by evacuating the MPC after completion of MPC draining operation. For MPCs containing Moderate Burnup Fuel (MBF) assemblies only, this operation may be carried out using the vacuum drying method up to the threshold heat loads defined in Table 4.5.1. In this Table Q1 is the threshold heat load for vacuum drying operations without time limits. The requirements and limits for moisture removal are provided in LCO 3.1.1 of the HI-STORM 100 CoC and are specific to the amendment to which the HI-STORM 100 System is being loaded. To minimize fuel temperatures during vacuum drying operations the HI-TRAC annulus must be water filled.

At heat loads greater than the threshold heat load defined above, the peak cladding temperature cannot be maintained below the ISG-11, Revision 3 limit of 570°C for MBF under a vacuum condition of infinite duration. Under this scenario, cycles of vacuum drying resulting in heatup followed with cooling by helium are performed until drying criteria is achieved. Similarly, vacuum drying of MPCs containing one or more High Burnup Fuel (HBF) assemblies is also permitted under time limits. The peak cladding temperature for drying HBF must be maintained below ISG-11 Rev 3 limit of 400°C. It must be noted that the permissible time for heatup/cooldown cycles is a function of canister specific heat loads. At lower heat loads the duration of vacuum drying cycles is higher. The thermal model defined below must be used for heatup/cooldown cycles for site-specific canister heat load maps. It must be ensured per ISG-11 Rev 3 that the repeated thermal cycling is limited to less than 10 cycles, with cladding temperature variations less than 65°C (117°F) each cycle. Fuel drying may also be conducted by using forced helium drying (FHD) process as discussed in Section 4.5.3.2.

A 3-D FLUENT thermal model of the MPC is constructed in the same manner as described in Section 4.4. The principal input to this model is the effective conductivity of fuel under vacuum drying operations. To reasonably bound vacuum drying operations the effective conductivity of fuel is computed assuming the MPC is filled with water vapor at a very low pressure (1 torr) for the entire duration of vacuum drying<sup>17</sup>. The methodology for computing the effective conductivity is given in Section 4.4.1. To ensure a conservative evaluation the thermal model is incorporated with the following assumptions:

- i. Threshold heat load Q1, defined in Table 4.5.1, is assumed and steady-state condition reached under Q1 results in vacuum drying of only MBF without time limits.
- ii. The external surface of the MPC shell is postulated to vary linearly from 100°C (212°F) normal boiling temperature of water at the top to 111°C (231°F) elevated pressure boiling temperature at the bottom to account for the hydrostatic head.
- iii. The bottom surface of the MPC is insulated.

<sup>17</sup> This is conservative as the MPC pressure is progressively lowered below ambient pressure to facilitate moisture removal. Near the end of the vacuum drying operation the pressure is substantially lowered to approximately 1 torr to facilitate the 30-minute 3-torr vacuum rebound test followed by backfilling of the MPC with helium.

- iv. MPC internal convection heat transfer is suppressed.
- v. Top surface of the MPC is in communicative contact with air (Table 2.2.2). Natural convection and radiation cooling from the MPC top is included in the thermal model.

The principle objective of the vacuum drying analysis is to ensure that fuel temperatures are below ISG-11, Rev. 3 temperature limits (See Table 4.3.1). Under threshold heat load Q1 the results and margins are tabulated in Table 4.5.5. The results show that ISG-11, Rev. 3 limits are met with ample margins.

#### 4.5.3.2 Forced Helium Dehydration

To dry the MPC cavity using a Forced Helium Dehydration (FHD) system, a conventional, closed loop dehumidification system consisting of a condenser, a demister, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulence. A vapor pressure of 3 torr or less is assured by verifying that the helium temperature exiting the demister is maintained at or below the psychrometric threshold of 21°F for a minimum of 30 minutes. See Appendix 2.B for detailed discussion of the design criteria and operation of the FHD system.

The FHD system provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHD system ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit for normal conditions of storage, which is well below the high burnup cladding temperature limit 752°F (400°C) for all combinations of SNF type, burnup, decay heat, and cooling time. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode of natural convection in long term storage), it is readily concluded that the peak fuel cladding temperature under the latter condition will be greater than that during the FHD operation phase. In the event that the FHD system malfunctions, the forced convection state will degenerate to natural convection, which corresponds to the conditions of normal onsite transport. As a result, the peak fuel cladding temperatures will approximate the values reached during normal onsite transport as described elsewhere in this chapter.

In the event of a FHD malfunction, the MPC cavity must be filled with helium to achieve an MPC cavity pressure of 72 psig, as described in Chapter 11. A thermal analysis is performed to demonstrate that the fuel is under a safe condition in the event of FHD malfunction. The licensing basis thermal model with most limiting decay heat i.e. maximum decay heat load with X=3 (Table 2.1.30), is adopted for thermal analysis. Since this cask operation is performed inside a building, no solar insolation is applied on the cask external surfaces. The predicted steady state temperature results for this scenario are summarized in Table 4.5.10. The results demonstrate that the fuel cladding and all other component temperatures are well below their off-normal temperature limits specified in Table 2.2.3 of the FSAR. Therefore, the fuel and MPC are in a safe condition when the MPC pressure is maintained at least at 72 psig in the event of FHD malfunction.

#### 4.5.4 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

NUREG-1536 requires an evaluation of cask cooldown and reflood procedures to support fuel unloading from a dry condition. Two options are available for cask cooldown: Direct Water Quenching and Helium Cooldown followed by Direct Water Quenching. It is not mandatory to perform helium cooldown prior to quenching but some users may choose to perform helium cooldown for high heat load MPCs.

#### 4.5.4.1 Option 1: Direct Water Quenching

Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. Direct MPC cooldown is effectuated by introducing water through the lid drain line. From the drain line, water enters the MPC cavity near the MPC baseplate. Steam produced during the direct quenching process will be vented from the MPC cavity through the lid vent port. To maximize venting capacity, both vent port RVOA connections must remain open for the duration of the fuel unloading operations. As direct water quenching of hot fuel results in steam generation, it is necessary to limit the rate of water addition to avoid MPC overpressurization. For example, steam flow calculations using bounding assumptions (100% steam production and MPC at design pressure) show that the MPC is adequately protected upto a reflood rate of 3715 lb/hr. Limiting the water reflood rate to this amount or less would prevent exceeding the MPC design pressure.

During direct reflood operations the fuel cladding is subject to high temperature gradients and concomitant thermal stresses. The integrity of fuel under direct quenching is evaluated in a generic manner in the HI-STORM FW SAR (Docket No. 72-1032, Ref. [4.5.2]). To define a bounding scenario at time  $t = 0$  sec, a uniformly bounding temperature throughout the entire fuel rod is set at 752°F (400°C), which is the temperature limit of fuel cladding. At time  $t = 0.1$  sec, a reasonably bounding 80°F quench water temperature is assigned to the lower half of the fuel rod to simulate a thermal shock with a large step change in the cladding temperature. The resulting transient stress and strain distributions in the fuel rod are evaluated with finite element ANSYS models. The results show that the maximum stress and strain values remain within the elastic range and remain well within failure strain limit (a factor of 6 against failure strain). This safety analysis documented in Section 3.4.4.1.11 of the HI-STORM FW FSAR provides the assurance that the MPC reflood event will not cause a breach of fuel cladding.

#### 4.5.4.2 Option 2: Helium Cooldown followed by Direct Water Quenching

Under a closed-loop forced helium circulation condition, the helium gas is cooled, via an external chiller. The chilled helium is then introduced into the MPC cavity from connections at the top of the MPC lid. The helium gas enters the MPC basket and moves through the fuel basket cells, removing heat from the fuel assemblies and MPC internals. The heated helium gas exits the MPC from the lid connection to the helium recirculation and cooling system. Because of the turbulence and mixing of the helium contents in the MPC cavity by the forced circulation, the MPC exiting temperature is a reliable measure of the thermal condition inside the MPC cavity. The objective of the cooldown system is to lower the bulk helium temperature in the MPC cavity to below the normal boiling

temperature of water (212°F). For this purpose, the rate of helium circulation shall be sufficient to ensure that the helium exit gas temperature is below this threshold limit with a margin.

An example calculation for the required helium circulation rate is provided below to limit the helium temperature to 200°F. The calculation assumes no heat loss from the MPC boundaries and a conservatively bounding heat load (38 kW (1.3x10<sup>5</sup> Btu/hr)). Under these assumptions, the MPC helium is heated adiabatically by the MPC decay heat from a given inlet temperature (T<sub>1</sub>) to a temperature (T<sub>2</sub>). The required circulation rate to limit T<sub>2</sub> to 200°F is computed as follows:

$$m = \frac{Q_d}{C_p(T_2 - T_1)}$$

where:

- Q<sub>d</sub> = Design maximum decay heat load (Btu/hr)
- m = Minimum helium circulation rate (lb/hr)
- C<sub>p</sub> = Heat capacity of helium (1.24 Btu/lb-°F (Table 4.2.5))
- T<sub>1</sub> = Helium supply temperature (assumed 15°F in this example)

Substituting the values for the parameters in the equation above, m is computed as 567 lb/hr.

#### 4.5.5 Maximum Temperatures under Onsite Transport Conditions

##### 4.5.5.1 HI-TRAC Transport outside a Building

The requirements of utilizing SCS system in the onsite transfer operation are listed in Table 4.5.4. Condition 3 of Table 4.5.4 mandates the use of SCS for an MPC containing one or more HBF above the threshold heat load. This will assure that cladding temperature limits are met for HBF at heat loads higher than threshold heat load. See Appendix 2.C for the SCS requirements.

A 3-D FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was constructed as described in Subsection 4.5.1 to evaluate temperature distributions under onsite transport. In the normal onsite transport mode, the annular region between the canister and the cask has air and the cask is subject to heat input from insolation. The ambient temperature when HI-TRAC is placed in the outdoor environment corresponds to the maximum outdoor ambient temperature specified in Table 2.2.2 under short term operations. Even though the duration of onsite transport is typically short enough to preclude the MPC and HI-TRAC from reaching steady-state, a steady-state thermal analysis is conservatively performed. The results summarized herein are when steady state conditions have been reached.

The safety analysis of the onsite transport scenario requires the computation of the margins of safety with respect to the peak fuel cladding temperature of moderate and high burnup fuel<sup>18</sup>, MPC internal

<sup>18</sup> The cladding temperature limit for the high burn up fuel is more restrictive (See Table 4.3.1).

pressure, fuel basket metal temperature, hydraulic pressure in the water jacket and the temperature of the HI-TRAC body parts.

The water in the water jacket surrounding the HI-TRAC transfer cask body provides necessary neutron shielding. During normal handling and onsite transport operations this shielding water is contained within the water jacket at an elevated pressure. The water jacket is equipped with two pressure relief devices to prevent overpressure.

The steady state analyses are first performed for the design basis heat load under the two extreme allowable regionalized storage scenarios ( $Q = 36.9$  kW,  $X = 0.5$  and  $Q = 30.17$  kW,  $X = 3$  in MPC-32), which are the maximum permissible heat load allowed to be stored in HI-STORM system defined in Subsection 2.1.9.1. The computed fuel temperatures in this scenario remain below the cladding temperature limit of moderate burnup fuel, but exceed the temperature limit of high burnup fuel (Table 4.3.1). For both regionalized storage scenarios, the MPC internal pressure, fuel basket and the HI-TRAC parts temperatures are presented in Table 4.5.6, and their corresponding allowable limits show positive margins of safety. As these are bounding steady state temperatures, the results support onsite transport of fuel in the HI-TRAC without the aid of any supplemental cooling for MPC containing only moderate fuel burnup and cooling times up to the maximum design basis heat load of the HI-STORM System.

The steady state analyses are then performed for the threshold heat load which requires SCS for MPC containing HBF, which is defined in Table 4.5.4. In this scenario, the heat generation rate in the MPC is reduced to 90% of maximum design basis rate. Under the two extreme allowable regionalized storage scenarios ( $Q = 33.21$  kW,  $X = 0.5$  and  $Q = 27.15$  kW,  $X = 3$  in MPC-32), the computed fuel temperatures remain below the cladding temperature limits of high and moderate burnup fuel (Table 4.3.1). The MPC components, fuel basket and the HI-TRAC parts temperatures are presented in Table 4.5.7, and their corresponding allowable limits show positive margins of safety. Therefore, the results support onsite transport of fuel in the HI-TRAC without the aid of any supplemental cooling for any fuel burnup and cooling times up to 90% of maximum design basis heat load<sup>†</sup> of the HI-STORM System.

#### 4.5.5.2 Evaluation of HI-TRAC inside a Building

When HI-TRAC is located inside the fuel loading building, the ambient air temperature inside a building may be higher than the outdoor environment evaluated in Subsection 4.5.5.1. Since ambient air temperature may be higher inside the building, a maximum indoor air temperature specified in Table 2.2.2 under short term operations is used in the analysis. Since the cask is inside a building, no solar insolation is applied to the cask. A steady state analysis is performed for the 90% of design basis heat load under the regionalized storage scenario ( $Q = 27.15$  kW,  $X = 3$  in MPC-32), which results in the lowest PCT safety margin for high burnup fuel. The peak cladding, fuel basket and the HI-TRAC parts temperatures are presented in Table 4.5.8. The predicted component temperatures are below their corresponding temperatures for outdoor environment presented in Table 4.5.7. Therefore,

<sup>†</sup> The individual storage cell heat loads are also restricted to 90% of their design maximum heat loads.

the normal on-site transfer of a HI-TRAC outside the building and with solar insolation as evaluated in Subsection 4.5.5.1 is the limiting thermal condition.

#### 4.5.5.3 Evaluation of SCS Failure

Table 4.5.4 mandates the use of the SCS for MPC containing one or more HBF above the threshold heat load to ensure fuel remains below the short-term operation temperature limits mandated by ISG-11, Rev. 3. If the SCS fails during operation, an accident condition defined in Section 11.2, the thermal state of the fuel would asymptotically approach steady state maximum conditions corresponding to the coincident thermal payload in the HI-TRAC transfer cask. The results of steady state analysis are provided in Table 4.5.6 for the design basis heat load. It is shown that the fuel remains well below its accident limit (Table 4.3.1).

#### 4.5.5.4 Evaluation of SCS for Lower Backfill Pressure Limit

As stated in Table 1.2.2, MPC is allowed to be backfilled with a lower pressure limit than that specified in Table 4.4.12, if the MPC heat load meets the threshold heat load requirement in Table 1.2.2. The analyses that support the threshold heat load scenario and lower helium backfill pressure limits are reported in the Revision 6 of HI-STORM 100 FSAR. To evaluate the requirement of SCS system (Table 4.5.4) for the threshold heat load with lower backfill pressure limit, a steady state analysis is performed for the bounding heat load scenario, i.e. MPC-32 with heat load of uniform 28.74 kW under an ambient environment outside the building. The computed fuel temperatures in this scenario remain below the cladding temperature limit of moderate burnup fuel, but exceed the temperature limit of high burnup fuel (Table 4.3.1). The fuel basket and the HI-TRAC parts temperatures are presented in Table 4.5.9, and their corresponding allowable limits show positive margins of safety. As these are bounding steady state temperatures, the results support onsite transport of fuel in the HI-TRAC without the aid of any supplemental cooling for MPC containing only moderate fuel burnup and cooling times up to the threshold heat load in Table 1.2.2.

The steady state analyses are then performed for 90% of threshold heat load<sup>†</sup>, i.e.  $Q = 25.86$  kW uniform in MPC-32, the computed fuel temperatures remain below the cladding temperature limits of high and moderate burnup fuel (Table 4.3.1). The MPC components, fuel basket and the HI-TRAC parts temperatures are presented in Table 4.5.9, and their corresponding allowable limits show positive margins of safety. Therefore, the results support onsite transport of high burnup fuel in the HI-TRAC without the aid of any supplemental cooling up to the 90% of threshold heat load<sup>†</sup> in Table 1.2.2.

#### 4.5.5.5 Environmental Temperature Requirements

Short term operations involving the HI-TRAC transfer cask can be carried out on the basis of the safety evaluation herein if the reference ambient temperature (three day average around the cask) is below the “Threshold Temperature” defined in Table 2.2.2 as 110 deg. F for operations inside the

<sup>†</sup> The individual storage cell heat loads are also restricted to 90% of their threshold maximum heat loads.

part 50 structural boundary and 90 deg. F outside of it. The determination of the Threshold Temperature compliance shall be made based on the best available thermal data for the site.

If the reference ambient temperature exceeds the corresponding Threshold Temperature then a site specific analysis using the methodology set down in Section 4.5 shall be performed using the actual heat load and reference ambient temperature equal to the three day average to ensure that the steady state peak fuel cladding temperature will remain below the Table 2.2.3 limit. If the peak fuel cladding temperature exceeds Table 2.2.3 limit then the use of a Supplemental Cooling System (SCS) is mandatory.

#### 4.5.6 Maximum Internal Pressure

After fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling and on-site transfer operations in the HI-TRAC transfer cask, the gas temperature within the MPC rises to its maximum operating temperature as determined by on the thermal analysis methodology described previously. In Table 4.5.6, the MPC internal pressure co-incident with the MPC temperature is reported and compared with the short term (off-normal) pressure limit specified in Table 2.2.1 to show compliance with design limit. The MPC gas pressure listed in Table 4.5.6 is below the MPC design internal pressure listed in Table 2.2.1.



Table 4.5.1

## THRESHOLD HEAT LOADS FOR MOISTURE REMOVAL OPERATIONS

Drying Method	Fuel Burnup	Threshold Heat Load <sup>Note 1</sup>	Time Limits
Vacuum Drying	MBF	Q1	None
Vacuum Drying	MBF	MPC Heat Load > Q1	Yes (Note 2)
Vacuum Drying	HBF	36.9 kW	Yes (Note 2)
FHD	MBF and/or HBF	36.9 kW	None

Note 1: Threshold heat loads is defined as  $Q1 = 26 \text{ kW}$  (Uniform)

Note 2: Vacuum drying of the MPC must be performed using cycles of the drying system, according to the guidance contained in ISG-11 Revision 3 and as described in Paragraph 4.5.3.1. The time limit for these cycles shall be determined based on site specific conditions

Table 4.5.2

## HI-TRAC TRANSFER CASK LOWERBOUND WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
Water Jacket	7,000	1.0	7,000
Lead	52,000	0.031	1,612
Carbon Steel	40,000	0.1	4,000
Alloy-X MPC (empty)	39,000	0.12	4,680
Fuel	40,000	0.056	2,240
MPC Cavity Water*	6,500	1.0	6,500
			26,032 (Total)
* Conservative lower bound water mass.			

Table 4.5.3

## TIME-TO-BOIL FOR WATER IN THE MPC CAVITY

Initial Temperature (°F)	Time (hrs) @ Q = 36.9 kw	Time (hrs) @ Q = 30 kw	Time (hrs) @ Q = 25 kw	Time (hrs) @ Q = 20 kw
80	27.3	33.6	40.3	50.4
90	25.2	31.0	37.2	46.5
100	23.2	28.5	34.2	42.7
110	21.1	25.9	31.1	38.9
120	19.0	23.4	28.1	35.1
125	18.0	22.1	26.5	33.2

Table 4.5.4

## THRESHOLD HEAT LOADS FOR SUPPLEMENT COOLING SYSTEM REQUIREMENT

Condition*	Fuel in MPC	Heat Load Reduction Factor *	SCS Required
1	All MBF	100%	NO
2	One or More HBF	$\leq 90\%$	NO
3	One or More HBF	$> 90\%$	YES
* The threshold heat load is obtained by multiplying the design basis heat load per storage cell defined in Subsection 2.1.9.1 by the reduction factor listed in this table.			

Table 4.5.5

## MAXIMUM FUEL TEMPERATURES UNDER VACUUM DRYING OPERATIONS

Threshold Heat Load <sup>Note 1</sup>	Time Limit	Temperature (°F)	Temperature Limit <sup>Note 2</sup>	Margin (°F)
Q1	None	1046	1058	12
Notes:				
1) Threshold heat load is defined in Table 4.5.1.				
2) Temperature limit of moderate burnup fuel shown. Vacuum drying of high burn-up fuel is discussed in Subsection 4.5.3.				

Table 4.5.6

**HI-TRAC ONSITE TRANSFER- TEMPERATURE AND PRESSURE  
FOR DESIGN BASIS HEAT LOAD**

<b>Component</b>	<b>Maximum Temperatures (°F)</b>	
	X=0.5	X=3
Fuel Cladding	748	784
MPC Basket	745	779
Basket Peripheral Panels	612	567
MPC Shell	473	459
HI-TRAC Inner Shell	282	268
Radial Lead	279	266
HI-TRAC Water Jacket Shell	250	232
Axial Neutron Shield <sup>Note1</sup>	280	279
Water Jacket Bulk Water	244	225
<b>Pressure (psig)</b> <sup>Note 2</sup>		
Normal Condition	103.0	96.6
Note 1: Maximum section average temperature. Note 2: The MPC pressure is computed under the maximum backfill pressure specified in Table 4.4.12.		

Table 4.5.7

**HI-TRAC ONSITE TRANSFER- TEMPERATURE  
FOR 90% OF DESIGN BASIS HEAT LOAD**

Component	Maximum Temperatures (°F)	
	X=0.5	X=3
Fuel Cladding	702	734
MPC Basket	698	729
Basket Peripheral Panels	577	534
MPC Shell	446	437
HI-TRAC Inner Shell	268	257
Radial Lead	266	253
HI-TRAC Water Jacket Shell	239	223
Axial Neutron Shield <sup>Note1</sup>	279	266
Water Jacket Bulk Water	234	217
Note 1: Maximum section average temperature.		

Table 4.5.8

HI-TRAC ONSITE TRANSFER- TEMPERATURE  
FOR 110°F AMBIENT TEMPERATURE AND WITHOUT INSOLATION <sup>Note 1</sup>

Component	Maximum Temperatures (°F)
Fuel Cladding	730
MPC Basket	723
Basket Peripheral Panels	532
MPC Shell	430
HI-TRAC Inner Shell	246
Radial Lead	244
HI-TRAC Water Jacket Shell	216
Axial Neutron Shield <sup>Note2</sup>	262
Water Jacket Bulk Water	210
Note 1: The results presented in this table are for 90% of design basis heat load with X=3.	
Note 2: Maximum section average temperature.	

Table 4.5.9

**HI-TRAC ONSITE TRANSFER- TEMPERATURE  
FOR THRESHOLD HEAT LOAD AND LOWER BACKFILL PRESSURE IN TABLE 1.2.2**

Component	Maximum Temperatures (°F)	
	Threshold Heat Load	90% Threshold Heat Load
Fuel Cladding	774	721
MPC Basket	768	716
Basket Peripheral Panels	558	527
MPC Shell	426	401
HI-TRAC Inner Shell	257	246
Radial Lead	253	244
HI-TRAC Water Jacket Shell	228	219
Axial Neutron Shield <sup>Note1</sup>	255	250
Water Jacket Bulk Water	223	214
Note 1: Maximum section average temperature.		

Table 4.5.10  
 MAXIMUM TEMPERATURES DURING FHD MALFUNCTION\*

<b>Component</b>	<b>Temperature† °C (°F)</b>
Fuel Cladding	455 (851)
MPC Basket	450 (842)
MPC Shell	235 (455)
HI-TRAC Inner Shell	126 (259)
HI-TRAC Radial Lead Gamma Shield	124 (255)
Water Bulk Temperature inside Water Jacket	102 (216)

\* The MPC Cavity pressure is set at 72 psig during FHD malfunction.

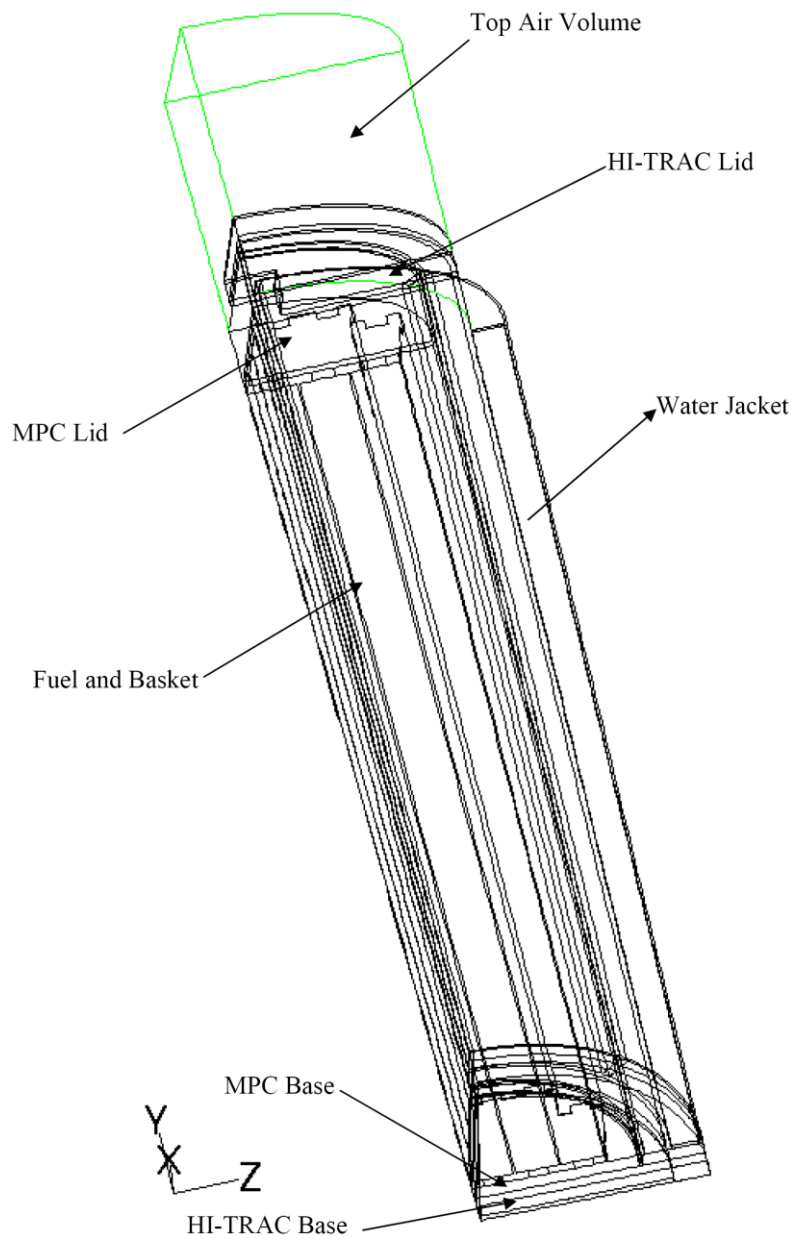
† All the predicted temperatures are below the off-normal temperature limits set forth in Table 2.2.3



Table 4.5.10a

## PRINCIPAL SITE-SPECIFIC TIME-TO-BOIL MODELING STEPS

<p>Step 1: Site Specific Conditions</p>	<p><u>Heat Loads</u> Site Specific heat load map</p> <p><u>Ambient Temperature</u> – Fuel handling building air temperature</p> <p><u>Initial Water Temperature</u> – Candidate temperature defined by cask user</p> <p><u>HI-TRAC Insolation</u> – None</p>
<p>Step 2: FLUENT Thermal Model</p>	<p>Incorporate HI-TRAC thermal methodologies (ii) thru (ix) defined in Section 4.5.1 and use the licensing basis HI-TRAC thermal model presented in [4.5.1]</p>
<p>Step 3: Run FLUENT Model</p>	<p>Apply thermal loads defined in Step 1 and compute the time dependent temperature field starting from the initial temperature defined in Step 1.</p>
<p>Step 4: Post-Process Results</p>	<p>Post-process FLUENT solution and obtain bulk water temperature <math>T_b(\tau)</math> as a function of time <math>\tau</math>. Interpolate <math>T_b(\tau)</math> to compute maximum permissible time-to-boil <math>\tau^*</math> meeting <math>T_b(\tau^*) &lt; 212^\circ\text{F}</math>.</p>



**Figure 4.5.1: 3D QUARTER SYMMETRIC THERMAL MODEL OF THE HI-TRAC TRANSFER CASK**

**Figure 4.5.2: INTENTIONALLY DELETED**

## 4.6 OFF-NORMAL AND ACCIDENT EVENTS<sup>19</sup>

In accordance with NUREG 1536 the HI-STORM 100 System is evaluated for the effects of off-normal and accident events. The design basis off-normal and accident events are defined in Chapter 2. For each event, the cause of the event, means of detection, consequences, and corrective actions are discussed and evaluated in Chapter 11. To support the Chapter 11 evaluations, thermal analyses of limiting off-normal and accident events are provided in the following.

To ensure a bounding evaluation for the array of fuel storage configurations permitted in Section 2.1, a limiting storage condition is evaluated in this section. The limiting storage condition is previously determined in the Section 4.5 and adopted herein for all off-normal and accident evaluations.

### 4.6.1 Off-Normal Events

#### 4.6.1.1 Off-Normal Pressure

This event is defined as a combination of (a) maximum helium backfill pressure (Table 4.4.12), (b) 10% fuel rods rupture, and (c) limiting fuel storage configuration. The principal objective of the analysis is to demonstrate that the MPC off-normal design pressure (Table 2.2.1) is not exceeded. The MPC off-normal pressures are reported in Table 4.4.9. The result<sup>20</sup> is confirmed to be below the off-normal design pressure (Table 2.2.1).

#### 4.6.1.2 Off-Normal Environmental Temperature

This event is defined by a time averaged ambient temperature of 100°F for a 3-day period (Table 2.2.2). The results of this event (maximum temperatures and pressures) are provided in Table 4.6.1 and 4.6.2. The results are below the off-normal condition temperature and pressure limits (Tables 2.2.1 and 2.2.3).

#### 4.6.1.3 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with debris screens installed on the inlet and outlet openings. These screens ensure the air passages are protected from entry and blockage by foreign objects. As required by the design criteria presented in Chapter 2, it is postulated that the HI-STORM air inlet vents are 50% blocked. The resulting decrease in flow area increases the flow resistance of the inlet ducts. The effect of the increased flow resistance on fuel temperature is analyzed for the normal ambient temperature (Table 2.2.2) and a limiting fuel storage configuration. The computed temperatures are reported in Table 4.6.1 and the corresponding MPC internal pressure in Table 4.6.2.

<sup>19</sup> A new standalone Section 4.6 is added in CoC Amendment 3 to address thermal analysis of off-normal and accident events. The results are evaluated in Chapter 11.

<sup>20</sup> Pressures relative to 1 atm absolute pressure (i.e. gauge pressures) are reported throughout this section.

The results are confirmed to be below the temperature limits (Table 2.2.3) and pressure limit (Table 2.2.1) for off-normal conditions.

#### 4.6.2 Accident Events

##### 4.6.2.1 Fire Accidents

###### 4.6.2.1.1 Licensing-Basis (Postulated) Fire Accident

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low due to the lack of combustible materials at an ISFSI, a licensing-basis fire event has been assumed and analyzed. Under the postulated fire, the outer layers of HI-TRAC or HI-STORM overpacks are heated for the duration of fire by the incident thermal radiation and forced convection heat fluxes. The amount of combustible material for the postulated fire is limited to a volume of 50 gallons.

###### (a) HI-STORM Fire

The fuel tank fire is conservatively assumed to surround the HI-STORM Overpack. Accordingly, all exposed overpack surfaces are heated by radiation and convection heat transfer from the fire. Based on NUREG-1536 and 10 CFR 71 guidelines [4.6.1], the following fire parameters are assumed:

1. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.
2. The average flame temperature must be at least 1475°F (800°C). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces, which is very conservative considering the size of the HI-STORM cask. It is therefore conservative to use the 1475°F (800°C) temperature.
3. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration.
4. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal

measurements [4.6.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft<sup>2</sup>×°F) is applied to exposed overpack surfaces during the short-duration fire.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width [4.6.1], the fuel ring surrounding the overpack covers 147.6 ft<sup>2</sup> and has a depth of 0.54 in. From this depth and a constant fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.62 minutes. The fuel consumption rate of 0.15 in/min is a lowerbound value from a Sandia National Laboratories report [4.6.2]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

To evaluate the impact of fire heating of the HI-STORM overpack, a thermal model of the overpack cylinder was constructed using the ANSYS computer code. The initial temperature of the overpack was conservatively assumed to be the maximum temperature field during storage (Table 4.4.7).. In this model the outer surface and top surface of the overpack were subjected for the duration of fire (3.62 minutes) to the fire conditions defined in this subsection. In the post-fire phase, the ambient conditions preceding the fire were restored. The transient study was conducted for a period of 5 hours, which is sufficient to allow temperatures in the overpack to reach their maximum values and begin to recede.

Due to the severity of the fire condition radiative heat flux, heat flux from incident solar radiation is negligible and is not included. Furthermore, the smoke plume from the fire would block most of the solar radiation. It is recognized that the ventilation air in contact with the inner surface of the HI-STORM Overpack with design-basis decay heat and normal ambient temperature conditions varies between 80°F at the bottom and 220°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy a miniscule fraction of area of the cylindrical surface of the massive HI-STORM Overpack. Due to the short duration of the fire event and the relative isolation of the ventilation passages from the outside environment, the ventilation air is expected to experience little intrusion of the fire combustion products. As a result of these considerations, it is conservative to assume that the air in the HI-STORM Overpack ventilation passages is held constant at a temperature of 300°F during the entire duration of the fire event.

The thermal transient response of the storage overpack is determined using the ANSYS finite element program. Time-histories for points in the storage overpack are monitored for the duration of the fire and the subsequent post-fire equilibrium phase.

Heat input to the HI-STORM Overpack while it is subjected to the fire is from a combination of an incident radiation and convective heat fluxes to all external surfaces. This can be expressed by the following equation:

$$q_F = h_{fc} (T_A - T_S) + \sigma \epsilon [(T_A + C)^4 - (T_S + C)^4]$$

where:

$q_F$  = Surface Heat Input Flux (Btu/ft<sup>2</sup>-hr)

$h_{fc}$  = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft<sup>2</sup>-hr-°F)

$\sigma$  = Stefan-Boltzmann Constant  
 $T_A$  = Fire Temperature (1475°F)  
 $C$  = Conversion Constant (460 (°F to °R))  
 $T_S$  = Surface Temperature (°F)  
 $\varepsilon$  = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [4.6.2].

After the fire event, the ambient temperature is restored and the storage overpack cools down (post-fire temperature relaxation). Heat loss from the outer surfaces of the storage overpack is determined by the following equation:

$$q_s = h_s (T_s - T_A) + \varepsilon \sigma [(T_s + C)^4 - (T_A + C)^4]$$

where:

$q_s$  = Surface Heat Loss Flux (W/m<sup>2</sup> (Btu/ft<sup>2</sup>-hr))  
 $h_s$  = Natural Convection Heat Transfer Coefficient (Btu/ft<sup>2</sup>-hr-°F)  
 $T_s$  = Surface Temperature (°F)  
 $T_A$  = Ambient Temperature (°F)  
 $\sigma$  = Stefan-Boltzmann Constant  
 $\varepsilon$  = Surface Emissivity  
 $C$  = Conversion Constant (460 (°F to °R))

In the post-fire temperature relaxation phase,  $h_s$  is obtained using literature correlations for natural convection heat transfer from heated surfaces [4.2.9].

A two-dimensional, axisymmetric model was developed for this analysis. Material thermal properties used were taken from Section 4.2. The outer surface and top surface of the overpack are exposed to the ambient conditions (fire and post-fire), and the base of the overpack is insulated. The transient study is conducted for a period of 5 hours, which is sufficient to allow temperatures in the overpack to reach their maximum values and begin to recede.

During the fire the overpack external shell temperatures are substantially elevated (~550°F). However, all concrete remains below the short term temperature limit (Table 2.2.3).

These results demonstrate that the fire accident event analyzed in a most conservative manner is determined to have a minor affect on the HI-STORM Overpack. The concrete temperatures at all locations remain below the allowable temperature limit. The temperatures of steel structures are within the allowable temperature limits.

Having evaluated the effects of the fire on the overpack, we now evaluate the effects on the MPC and contained fuel assemblies. Guidance for the evaluation of the MPC and its internals during a fire event is provided by NUREG-1536 (4.0,V,5.b), which states:

“For a fire of very short duration (i.e., less than 10 percent of the thermal time constant of the cask body), the NRC finds it acceptable to calculate the fuel temperature increase by assuming that the cask inner wall is adiabatic. The fuel temperature increase should then be determined by dividing the decay energy released during the fire by the thermal capacity of the basket-fuel assembly combination.”

The time constant of the cask body (i.e., the overpack) can be determined using the formula:

$$\tau = \frac{c_p \times \rho \times L_c^2}{k}$$

where:

$c_p$  = Overpack Specific Heat Capacity (Btu/lb-°F)

$\rho$  = Overpack Density (lb/ft<sup>3</sup>)

$L_c$  = Overpack Characteristic Length (ft)

$k$  = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (142 lb/ft<sup>3</sup>) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM Overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 128 \text{ hrs}$$

One-tenth of this time constant is approximately 12.8 hours (768 minutes), substantially longer than the fire duration of 3.62 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The fuel temperature rise is computed next.

Table 4.5.2 lists lower-bound thermal inertia values for the MPC and the contained fuel assemblies. Applying a conservative upperbound decay heat load (38 kW (1.3x10<sup>5</sup> Btu/hr)) and adiabatic heating for the 3.62 minutes fire, the fuel temperature rise computes as:

$$\Delta T_{fuel} = \frac{\text{Decay heat} \times \text{Time duration}}{(\text{MPC} + \text{Fuel}) \text{ heat capacities}} = \frac{1.3 \times 10^5 \text{ Btu/hr} \times (3.62 / 60) \text{ hr}}{(2240 + 4680) \text{ Btu/}^\circ\text{F}} = 1.1^\circ\text{F}$$



This is a very small increase in fuel temperature. Consequently, the impact on the MPC internal helium pressure will be quite small. The peak temperatures of the fuel cladding, MPC fuel basket and MPC shell are summarized in Table 4.6.8. Based on a conservative analysis of the HI-STORM 100 System response to a hypothetical fire event, it is concluded that the fire event does not adversely affect the temperature of the MPC or contained fuel. We conclude that the ability of the HI-STORM 100 System to cool the spent nuclear fuel within design temperature limits during and after fire is not compromised.

An alternate method using the FLUENT thermal model described in Section 4.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FW FSAR. Principal modeling steps and acceptance criteria are defined in Table 4.6.10.

#### (b) HI-TRAC Fire

During the handling of the HI-TRAC transfer cask, the transporter fuel tank capacity must be limited to a 50 gallons. The duration of the 50-gallon fire under the conservatively postulated spill defined in the HI-STORM fire evaluation computes as 4.775 minutes. To demonstrate the fuel cladding and MPC pressure boundary integrity under exposure to this fire duration event during a fire accident analysis of the loaded 100-ton HI-TRAC is performed. In this analysis, the contents of the HI-TRAC are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat input and heat input from the short duration fire. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC. Using understated thermal inertia of the HI-TRAC and design maximum heat load (36.9 kW) the temperature rise rate computes as 5.553°F/min. Therefore, the temperature rise computed as the product of this rate and the fire duration reported above is 26.5°F. In this manner the maximum cladding temperature obtained by adding the temperature rise to the initial condition (See Table 4.5.6 for design basis heat load with X=3) computes as 811°F. The maximum fire temperature computed in the conservative manner above remains below the 1058°F accident temperature limit (Table 4.3.1) by substantial margins.

The elevated temperatures as a result of the fire accident will cause the pressure in the water jacket to increase and the overpressure relief valves to vent steam to the atmosphere. Based on the fire heat input to the water jacket, 11% of the water in the water jacket is boiled off. However, it is conservatively assumed, for dose calculations, that all the water in the water jacket is lost. In the 125-ton HI-TRAC, which uses Holtite in the lids for neutron shielding, the elevated fire temperatures would cause the Holtite to exceed its design accident temperature limits. This condition is conservatively addressed by ignoring neutron shield in the accident dose calculations.

Due to the increased temperatures of the MPC during fire accident the internal MPC pressure increases. The fire accident pressure is computed assuming the MPC cavity temperature rises by the fire accident temperature rise computed in this section. The result is tabulated in Table 4.6.2. The fire accident MPC pressure is substantially below the accident pressure limit (Table 2.2.1).

An alternate method using the FLUENT thermal model described in Section 4.5 can be adopted

to evaluate HI-TRAC site-specific fire accident event. Principal modeling steps and acceptance criteria are defined in Table 4.6.11.

#### 4.6.2.1.2 Site-Specific Postulated Fire Accident

All fire accident scenarios other than licensing-basis fire described above (from 50 gallons of combustible liquid) shall be considered as site-specific fire accidents. In such cases, a site-specific fire analysis must be performed using NRC endorsed acceptance criteria.

The following acceptance criteria apply to site-specific fire accidents:

- i. The peak cladding temperature during and after a fire accident shall not exceed the ISG-11 [4.1.4] permissible limit (see Table 2.2.3).
- ii. The steel structure of the overpack shall remain physically stable i.e., the maximum temperature shall be less than 50% of the component's melting temperature.
- iii. The confinement boundary shall continue to serve its design function. Confinement integrity is confirmed if the pressure in the MPC remains below the accident condition pressure limit (200 psig, Table 2.2.1); otherwise, an Enclosure Vessel failure analysis will be required.
- iv. The concrete temperature limit shall remain below its temperature limit specified in Table 2.2.2.

#### 4.6.2.2 Jacket Water Loss

In this subsection, the fuel cladding and MPC boundary integrity is evaluated for a postulated loss of water from the HI-TRAC water jacket. The HI-TRAC is equipped with an array of water compartments filled with water. For a bounding analysis, all water compartments are assumed to lose their water and be replaced with air. Heat dissipation by natural convection and radiation in the air space is included in the thermal model. The HI-TRAC is assumed to have the maximum thermal payload (design heat load) and assumed to have reached steady state (maximum) temperatures. Under these assumed set of adverse conditions, the maximum temperatures are computed using the 3D HI-TRAC thermal model constructed in Section 4.5 with the water in water jacket spaces replaced with air. The computed results are tabulated in Table 4.6.3 for the design basis heat load. The results of jacket water loss evaluation confirm that the cladding, MPC and HI-TRAC component temperatures are below the limits prescribed in Chapter 2 (Table 2.2.3). The co-incident MPC pressure is also computed and compared with the MPC accident design pressure (Table 2.2.1). The result (Table 4.6.2) is confirmed to be below the limit.

#### 4.6.2.3 Extreme Environmental Temperatures

To evaluate the effect of extreme weather conditions, an extreme ambient temperature (Table 2.2.2) is postulated to persist for a 3-day period. For a conservatively bounding evaluation the extreme temperature is assumed to last for a sufficient duration to allow the HI-STORM 100 System to reach steady state conditions. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative. Starting from a baseline condition evaluated in Section 4.4 (normal ambient temperature and limiting fuel storage configuration) the temperatures of the HI-STORM 100 System are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures (45°F). The HI-STORM extreme ambient temperatures computed in this manner are reported in Table 4.6.4. The co-incident MPC pressure is also computed (Table 4.6.2) and compared with the accident design pressure (Table 2.2.1). The result is confirmed to be below the accident limit.

#### 4.6.2.4 100% Blockage of Air Inlets

This event is defined as a complete blockage of all four bottom inlets. The immediate consequence of a complete blockage of the air inlets is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the considerable inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for extended durations. This accident condition is, however, a short duration event that is identified and corrected through scheduled periodic surveillance. Nevertheless, this event is conservatively analyzed assuming a substantial duration of blockage. The event is analyzed using the FLUENT CFD code. For MPC heat load up to the full design basis, the HI-STORM thermal model is the same 3-Dimensional model constructed for normal storage conditions (see Section 4.4) except for the bottom inlet ducts, which are assumed to be impervious to air. Using this model, a transient thermal solution of the HI-STORM 100 System starting from normal storage conditions is obtained. The results of the blocked ducts transient analysis that support the required action completion times for clearing the inlets are presented in Table 4.6.5 and confirmed to be below the accident temperature limits (Table 2.2.3). The co-incident MPC pressure is also computed and compared with the accident design pressure (Table 2.2.1). The result (Table 4.6.2) is confirmed to be below the limit.

For MPC heat loads which meet the values in Table 2.1.31, the results of the transient analysis that support the required action completion times for clearing the inlets are presented in Table 4.6.7 and confirm all temperatures are below the accident temperature limits (Table 2.2.3).

#### 30-Day 100% Vent Blockage Accident

As noted above, the fuel and component temperatures rise due to complete blockage of HI-STORM vents. This temperature rise is small for casks where heat loads are much lower than design basis heat loads. A threshold heat load is defined for all MPCs in Table 4.6.8a at or below which fuel and component temperatures remain below their respective 30-day accident temperature limits (Table 2.2.3) under steady state conditions. A steady state evaluation of a complete vent blockage at threshold heat loads is performed for both MPC-32 and MPC-68. Steady state temperature and MPC cavity pressure results are presented in Table 4.6.9. The results demonstrate that the fuel and component temperatures remain below their respective 30-day accident temperature limits defined in the Design Criteria Chapter 2 with robust margins. MPC cavity pressure is also below the accident design limit with robust margins. Thermal performance of MPC-68 bounds all types of MPC-68 and MPC-24. Therefore, the threshold total decay heat for MPC-68 is also adopted for all other variants of MPC-68 and MPC-24 canisters. To identify and clear any blockages mandatory surveillance is defined in Chapter 11.

Since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, the following evaluations are performed to demonstrate that the MPCs are safe at off-normal and accident conditions. Thermal off-normal and design basis events or accident conditions defined in Chapter 4 of the FSAR are concurrently evaluated with the 100% vent blockage event at threshold heat load:

- (a) Pressure (fuel rod rupture): There is no credible event to cause fuel rods to rupture during a 100% vent blockage event because of the following reasons:
- The computed PCT under 100% vent blockage accident condition (Table 4.6.9) is below the ISG-11 Rev 3 long-term normal temperature limit, and
  - there is no credible loading on the fuel assemblies to cause fuel rods to rupture during a 100% vent blockage event.

Accordingly, the computed cavity pressures under 30-day vent blockage event evaluated herein is not affected.

(b) Off-Normal Ambient Temperatures: This event is defined in Section 4.6.1.2 as an ambient temperature of 100°F for a 3-day period. The results of off-normal environmental temperatures coincident with 100% vent blockage event are summarized in Table 4.6.12. Component temperatures are obtained by adding the off-normal-to-normal ambient temperature difference of 20°F (11.1°C) to temperatures computed for MPCs at threshold decay heat (Table 4.6.9). The results are below the off-normal limits (Table 2.2.3) with substantial margins.

(c) Partial Blockage of Air Inlets: This condition is already covered by the postulated event wherein all the HI-STORM vents are assumed blocked.

(d) Fire: During transfer operations at the ISFSI, there is a possibility of a fire accident event to occur coincident with a 100% vent blockage event. The impact of fire on the MPC and fuel temperatures is extremely small (approximately 1°F). Overpack temperatures are primarily impacted due to heat

input from the fire which is considerably larger than the SNF decay heat. Therefore, as evaluated in Section 4.6.2.1(a), the overpack components and concrete temperatures remain below their respective accident temperature limits. Therefore, this accident event coincident with a 100% vent blockage event does not challenge the HI-STORM 100 System safety limits.

(e) Extreme Environment Temperature: This event is defined in Section 4.6.2.3 as an ambient temperature of 125°F for a 3-day period. The results of extreme environmental temperatures coincident with 100% vent blockage event are summarized in Table 4.6.13. Component temperatures are obtained by adding the extreme-to-normal ambient temperature difference of 45°F (25°C) to temperatures computed for MPCs at threshold decay heat (Table 4.6.9). The results are below the accident limits (Table 2.2.3) with substantial margins.

(f) Burial under Debris: This accident event is evaluated in Section 4.6.2.5. Since the threshold decay heat is substantially lower than the maximum design basis heat load and cask initial temperatures (Table 4.6.9) are similar for 100% vent blockage event and that evaluated in Table 4.6.6, the evaluation in Section 4.6.2.5 remains bounding.

In this manner the above evaluations reasonably assure that the HI-STORM 100 system containing MPCs are safe under off-normal and accident conditions coincident with 30-day 100% blocked vents under the threshold heat load.

#### 4.6.2.5 Burial Under Debris

Burial of the HI-STORM 100 System under debris is not a credible accident. During storage at the ISFSI there are no structures over the casks. Minimum regulatory distances from the ISFSI to the nearest ISFSI security fence precludes the close proximity of substantial amount of vegetation. There is no credible mechanism for the HI-STORM 100 System to become completely buried under debris. However, for conservatism, complete burial under debris is considered.

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 System will undergo a transient heat up under adiabatic conditions. The minimum available time ( $\Delta\tau$ ) for the fuel cladding to reach the accident limit depends on the following: (i) thermal inertia of the cask, (ii) the cask initial conditions, (iii) the spent nuclear fuel decay heat generation and (iv) the margin between the initial cladding temperature and the accident temperature limit. To obtain a lowerbound on  $\Delta\tau$ , the HI-STORM 100 Overpack thermal inertia (item i) is understated, the cask initial temperature (item ii) is maximized, decay heat overstated (item iii) and the cladding temperature margin (item iv) is understated. A set of conservatively postulated input parameters for items (i) through (iv) are summarized in Table 4.6.6. Using these parameters  $\Delta\tau$  is computed as follows:

$$\Delta\tau = \frac{m \times c_p \times \Delta T}{Q}$$

where:

- $\Delta\tau$  = Allowable burial time (hr)
- $m$  = Mass of HI-STORM System (lb)
- $c_p$  = Specific heat capacity (Btu/lb-°F)
- $\Delta T$  = Permissible temperature rise (°F)
- $Q$  = Decay heat load (Btu/hr)

Substituting the parameters in Table 4.6.6, a substantial burial time (34.6 hrs) is obtained. The co-incident MPC pressure is also computed and compared with the accident design pressure (Table 2.2.1). The result (Table 4.6.2) is confirmed to be below the limit.

Table 4.6.1  
OFF-NORMAL CONDITION MAXIMUM  
HI-STORM TEMPERATURES<sup>21</sup>

<b>Location<sup>22</sup></b>	<b>Off-Normal Ambient Temperature<sup>23</sup> (°F)</b>	<b>Partial Inlet Ducts Blockage (°F)</b>
Fuel Cladding	731	725
MPC Basket	728	721
MPC Shell	489	478
Overpack Inner Shell	342	339
Lid Concrete Bottom Plate	322	321
Overpack Body Local Temperature	340*	337
Lid Concrete Local Temperature	322*	321

\* Obtained by adding the off-normal-to-normal ambient temperature difference of 20°F to local maximum concrete temperatures during normal conditions presented in [4.5.1].

<sup>21</sup> The temperatures reported in this table are below the off-normal temperature limits specified in Chapter 2, Table 2.2.3.

<sup>22</sup> Temperatures of limiting components reported.

<sup>23</sup> Obtained by adding the off-normal-to-normal ambient temperature difference of 20°F (11.1°C) to normal condition HI-STORM temperatures reported in Section 4.4.

Table 4.6.2  
OFF-NORMAL AND ACCIDENT CONDITION MAXIMUM MPC PRESSURES  
FOR DESIGN BASIS HEAT LOAD

<b>Condition</b>	<b>Pressure (psig)</b>
Off-Normal Conditions	
Off-Normal Ambient	101.4
Partial Blockage of Inlet Ducts	100.4
Accident Conditions	
Extreme Ambient Temperature	104.4
100% Blockage of Air Inlets	118.1
Burial Under Debris	134.8
HI-TRAC Fire Accident	106.2
HI-TRAC Jacket Water Loss	108.6



Table 4.6.3  
 HI-TRAC JACKET WATER LOSS ACCIDENT MAXIMUM  
 TEMPERATURES FOR DESIGN BASIS HEAT LOAD

Component	Temperature (°F)	
	X = 0.5	X =3
Fuel Cladding	811	837
MPC Basket	808	829
MPC Shell	514	496
HI-TRAC Inner Shell	365	342
HI-TRAC Radial Lead Gamma Shield	363	342
HI-TRAC Water Jacket Shell	289	271
HI-TRAC Lid Neutron Shield Section Average	304	309

Table 4.6.4  
EXTREME ENVIRONMENTAL CONDITION MAXIMUM  
HI-STORM TEMPERATURES

Component	Temperature <sup>24</sup> (°F)
Fuel Cladding	756
MPC Basket	753
MPC Shell	514
Overpack Inner Shell	367
Lid Concrete Bottom Plate	347
Overpack Body Local Temperature	365*
Lid Concrete Local Temperature	347*

\* Obtained by adding the extreme ambient to normal ambient temperature difference of 45°F to local maximum concrete temperatures during normal conditions presented in [4.5.1].

24 Obtained by adding the extreme ambient to normal temperature difference (45°F) to normal condition temperatures reported in Section 4.4.

Table 4.6.5

100% BLOCKED INLET DUCTS MAXIMUM HI-STORM TEMPERATURES  
FOR DESIGN BASIS HEAT LOAD

Component	Temperatures@32 hrs (°F)	Temperatures@24 hrs (°F)
Fuel Cladding	890	860
MPC Basket	884	855
MPC Shell	583	561
MPC Lid (Note 1)	599	574
Overpack Inner Shell	480	454
Lid Concrete Bottom Plate	433	411
Overpack Body Local Temperature	477	450
Lid Concrete Local Temperature	433	411
Note 1: Maximum thru thickness section average temperature reported.		

Table 4.6.6

SUMMARY OF INPUTS FOR BURIAL UNDER DEBRIS ANALYSIS

Thermal Inertia Inputs:	
M (Lowerbound HI-STORM 100 Weight)	150000 lb
Cp (Carbon steel heat capacity) <sup>25</sup>	0.1 Btu/lb-°F
Cask initial temperature <sup>26</sup>	728°F
Q (Decay heat)	$1.3 \times 10^5$ Btu/hr
$\Delta T$ (clad temperature margin) <sup>27</sup>	300°F

<sup>25</sup> Carbon steel has the lowest heat capacity among the principal materials employed in MPC and overpack construction (carbon steel, stainless steel and concrete).

<sup>26</sup> Conservatively overstated.

<sup>27</sup> The clad temperature margin is conservatively understated in this table.

Table 4.6.7  
SUMMARY OF BLOCKED AIR INLET DUCT EVALUATION RESULTS FOR MPC HEAT  
LOAD UP TO 28.74 kW

	Max. Initial Steady-State Temp. <sup>†</sup> (°F)	Temperature Rise (°F)		Transient Temperature (°F)	
		at 33 hrs	at 72 hrs	at 33 hrs	at 72 hrs
Fuel Cladding	740	101	160	841	900
MPC Shell	351	184	250	535	601
Overpack Inner Shell #1 <sup>††</sup>	199	113	174	312	373
Overpack Inner Shell #2 <sup>†††</sup>	155	193	286	348	441
Overpack Outer Shell	145	14	40	159	185
Concrete Section Average*	172	79	141	251	313

<sup>†</sup> Conservatively bounding temperatures reported includes a hypothetical rupture of 10% of the fuel rods.

<sup>††</sup> Coincident with location of initial maximum temperature.

<sup>†††</sup> Coincident with active fuel axial mid-height.

\* Although section average concrete temperatures are tabulated herein, local temperature of concrete in HI-STORM overpack and lid are below its accident temperature limit specified in Table 2.2.3

Table 4.6.8\*

MPC TEMPERATURES AS A RESULT OF  
THE LICENSING-BASIS FIRE CONDITION

<b>Material/Component</b>	<b>Initial Condition (°F)</b>	<b>During Fire (°F)</b>	<b>Post-Fire Cooldown (°F)</b>
Fuel Cladding	731	732	732
MPC Fuel Basket	728	729	729
MPC Shell	489	490	490

Table 4.6.8a

THRESHOLD DECAY HEAT FOR 100% VENT BLOCKAGE

<b>MPC Type</b>	<b>Threshold Decay Heat, kW</b>	<b>Per Storage Cell Decay Heat Limit, kW</b>
MPC-24/24E/EF	18	0.75
MPC-68/68F/68FF/68M	18	0.264
MPC-32/32F	16	0.5

\* NUREG-1536 requires that the ambient temperature before a fire be 100°F, so the temperatures from the off-normal condition (which is an ambient temperature of 100°F) are taken as the initial condition for the fire event.

Table 4.6.9

**STEADY STATE MAXIMUM HI-STORM TEMPERATURES AND MPC CAVITY PRESSURE AT THRESHOLD HEAT LOAD UNDER 100% VENT BLOCKAGE**

<b>Component</b>	<b>MPC-32 Temperatures (°F)</b>	<b>MPC-68 Temperatures (°F)</b>
Fuel Cladding	714	730
MPC Basket	712	727
MPC Shell	471	502
MPC Lid (Note 1)	495	522
MPC Closure Ring	453	486
MPC Baseplate (Note 1)	327	342
Overpack Inner Shell (Note 2)	403	430
Overpack Concrete	401	426
Overpack Lid Concrete Bottom Plate	372	396
Overpack Lid Concrete Top Plate	221	225
Overpack Lid Concrete	372	396
<b>MPC Cavity Pressure, psig (Note 3)</b>		
No Rod Rupture	102.6	104.7
With 1% Rod Rupture	103.6	105.2
<p>Note 1: Thru-thickness section average temperature is reported.</p> <p>Note 2: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.</p> <p>Note 3: Although the CFD evaluations have been performed with an operating temperature corresponding to minimum initial helium backfill specification of 29.3 psig at 70°F reference temperature, maximum initial helium backfill pressure of 48.5 psig is adopted to compute MPC cavity pressure. In reality, the actual MPC cavity pressure will be lower than that reported above.</p>		

Table 4.6.10

## PRINCIPAL SITE-SPECIFIC HI-STORM FIRE ACCIDENT MODELING STEPS

Step 1: Site Specific Conditions	<p><u>Heat Loads</u> Site Specific heat load map.</p> <p><u>Ambient Temperature</u> – Normal storage temperature defined in Chapter 2.</p> <p><u>Fire Accident</u> – Compute fire duration <math>\tau_f</math> based on site specific fuel quantity in accordance with methodology defined in Sub-Section 4.6.2.1(a).</p>
Step 2: FLUENT Thermal Model	Incorporate HI-STORM thermal methodologies defined in Sub-Sections 4.4.1.1 and 4.4.1.2. Use the licensing basis HI-STORM thermal model presented in [4.5.1]. Apply heat loads and ambient temperature defined in Step 1 and obtain baseline initial temperature field.
Step 3: Fire Transient Solution	Apply fire parameters defined by fire temperature, fire emissivity and convection heat transfer coefficient specified in Sub-Section 4.6.2.1(a) to FLUENT Model and compute time dependent HI-STORM temperature field starting from initial temperature field obtained in Step 2 upto end of fire $\tau_f$ .
Step 4: Post-Fire Solution	Restore ambient temperature conditions as defined in Sub-Section 4.6.2.1(a) and compute time dependent temperature field under cooldown of HI-STORM cask by natural convection and radiation. Conservatively assume paint loss from all exterior surfaces. Continue solution until all component and fuel temperatures reach their maximum and begin to recede.
Step 5: Post-Process Results	Post-process FLUENT solution and evaluate compliance of maximum fuel, basket, MPC confinement boundary, HI-STORM concrete and enclosure shell temperatures to Chapter 2 accident temperature limits. Additionally, steel structure of the overpack shall remain physically stable, i.e., the maximum temperature shall be less than 50% of the component's melting temperature. Compute maximum MPC pressure in accordance with Sub-Section 4.4.5 methodology and evaluate compliance with Chapter 2 accident pressure limits.

Table 4.6.11

## PRINCIPAL SITE-SPECIFIC HI-TRAC FIRE ACCIDENT MODELING STEPS

Step 1: Site Specific Conditions	<p><u>Heat Loads</u> Site Specific heat load map.</p> <p><u>Ambient Temperature</u> – Short Term Operations temperature defined in Chapter 2.</p> <p><u>Fire Accident</u> – Compute fire duration <math>\tau_f</math> based on site specific fuel quantity in accordance with methodology defined in Sub-Section 4.6.2.1(b).</p>
Step 2: FLUENT Thermal Model	Incorporate HI-TRAC thermal methodologies (i) thru (ix) defined in Section 4.5.1. Use the licensing basis HI-TRAC thermal model presented in [4.5.1]. Apply heat loads and ambient temperature defined in Step 1 and obtain baseline initial temperature field.
Step 3: Fire Transient Solution	Apply fire parameters defined by fire temperature, fire emissivity and convection heat transfer coefficient specified in Sub-Section 4.6.2.1(a) to FLUENT Model and compute time dependent HI-TRAC temperature field starting from initial temperature field obtained in Step 2 upto end of fire $\tau_f$ .
Step 4: Post-Fire Solution	Restore ambient temperature conditions as defined in Sub-Section 4.6.2.1(a). Conservatively assume destruction of paint from exterior surfaces and complete Holtite loss. Compute time dependent temperature field under cooldown of HI-TRAC cask by natural convection and radiation. Continue solution until all component and fuel temperatures reach their maximum and begin to recede.
Step 5: Post-Process Results	Post-process FLUENT solution and evaluate compliance of maximum fuel, basket, MPC confinement boundary and HI-TRAC enclosure shell temperatures with Chapter 2 accident temperature limits. Compute maximum MPC pressure in accordance with Sub-Section 4.4.5 methodology and evaluate compliance with Chapter 2 accident pressure limits.



Table 4.6.12

**STEADY STATE HI-STORM TEMPERATURES WITH MPCs AT THRESHOLD HEAT  
LOAD UNDER 100% VENT BLOCKAGE AND COINCIDENT OFF-NORMAL  
ENVIRONMENTAL TEMPERATURE**

<b>Component</b>	<b>MPC-32 Temperatures<sup>Note 1</sup> (°F)</b>	<b>MPC-68 Temperatures<sup>Note 1</sup> (°F)</b>
Fuel Cladding	734	750
MPC Basket	732	747
MPC Shell	491	522
MPC Lid (Note 2)	515	542
MPC Closure Ring	473	506
MPC Baseplate (Note 2)	347	362
Overpack Inner Shell (Note 3)	423	450
Overpack Body Concrete	421	446
Overpack Lid Bottom Plate	392	416
Overpack Lid Top Plate	241	245
Overpack Lid Concrete	392	416
<b>MPC Cavity Pressure, psig</b>		
With 1% Rod Rupture	106.0	107.6
<p>Note 1: Unless otherwise specified, all the reported temperatures are peak maximum values.</p> <p>Note 2: Maximum through thickness average temperature at the hottest location is reported for structural thick components.</p> <p>Note 3: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.</p>		

Table 4.6.13

**STEADY STATE HI-STORM TEMPERATURES WITH MPCs AT THRESHOLD HEAT  
LOAD UNDER 100% VENT BLOCKAGE AND COINCIDENT EXTREME  
ENVIRONMENTAL TEMPERATURE**

<b>Component</b>	<b>MPC-32 Temperatures<sup>Note 1</sup> (°F)</b>	<b>MPC-68 Temperatures<sup>Note 1</sup> (°F)</b>
Fuel Cladding	759	775
MPC Basket	757	772
MPC Shell	516	547
MPC Lid (Note 2)	540	567
MPC Closure Ring	498	531
MPC Baseplate (Note 2)	372	387
Overpack Inner Shell (Note 3)	448	475
Overpack Body Concrete	446	471
Overpack Lid Bottom Plate	417	441
Overpack Lid Top Plate	266	270
Overpack Lid Concrete	417	441
<b>MPC Cavity Pressure, psig</b>		
With 1% Rod Rupture	109.0	110.6
<p>Note 1: Unless otherwise specified, all the reported temperatures are peak maximum values.</p> <p>Note 2: Maximum through thickness average temperature at the hottest location is reported for structural thick components.</p> <p>Note 3: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.</p>		

## 4.7 REGULATORY COMPLIANCE

### 4.7.1 Normal Conditions of Storage

NUREG-1536 [4.4.1] and ISG-11 [4.1.4] define several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 4.1 through 4.4. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by ISG-11 [4.1.4], the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System. Maximum clad temperatures for long-term storage conditions are reported in Section 4.4.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Sections 4.4, 4.5 and 4.6 for normal, short term operations, and off-normal & accident conditions. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for long-term storage conditions are reported in Section 4.4. Design temperature limits are summarized in Table 2.2.3. HI-STORM System components defined as important to safety are listed in Table 2.2.6.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 [4.1.4] limit of 400°C (752°F).

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal conditions. All thermal results reported in Section 4.4 are within the design criteria allowable ranges for all normal conditions of storage.

#### 4.7.2 Short Term Operations

Evaluation of short term operations is presented in Section 4.5. This section establishes complete compliance with the provisions of ISG-11 [4.1.4]. In particular, the ISG-11 requirement to ensure that maximum cladding temperatures under all fuel loading and short term operations be below 400°C (752°F) for high burnup fuel and below 570°C (1058°F) for moderate burnup fuel is demonstrated as stated below.

Specifically as required by ISG-11, the fuel cladding temperature is maintained below the applicable limits for HBF and MBF (Table 4.3.1) during short term operations.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal and off-normal conditions, assuming rupture of 1 percent and 10 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.

As required by NUREG-1536 (4.0,IV, 4), all cask and fuel materials are maintained within their minimum and maximum temperature for all short term operations in order to enable components to perform their intended safety functions.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for all short term operations.

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## SUPPLEMENT 4.I<sup>1</sup>

### THERMAL EVALUATION OF THE HI-STORM 100U SYSTEM

#### 4.I.0 OVERVIEW

The HI-STORM 100U is an underground vertical ventilated module (VVM) with openings for air ingress and egress and internal passages for ventilation cooling of loaded MPCs. The HI-STORM 100U construction is described in Supplement 1.I and illustrated in Figures 1.I.1 through 1.I.4. The HI-STORM 100U utilize the same MPCs used in the aboveground systems. The HI-STORM 100U inlets and outlets are 360° (axisymmetric) openings provided in the lid. The overall ventilation airflow path from inlet to outlet is illustrated in Figure 1.I.4. The licensing drawing package for the HI-STORM 100U VVM is provided in Section 1.I.5. This supplement provides a thermal evaluation of the HI-STORM 100U for normal, off-normal and accident conditions. The evaluations described herein parallel those of the aboveground HI-STORM cask contained in the main body of Chapter 4 of this FSAR. To ensure readability, the section in the main body of the chapter to which each section in this supplement corresponds is clearly identified. All tables in this supplement are labeled sequentially.

#### 4.I.1 INTRODUCTION

The information presented in this supplement is intended to serve as a complement to the information provided in the main body of Chapter 4. Thus, information in Chapter 4 that remains applicable to the HI-STORM 100U is not repeated herein. Specifically, the following information in the main body of Chapter 4 is not repeated:

1. The thermal properties of materials in Section 4.2 applicable to the HI-STORM 100U System.
2. The specifications for components in Section 4.3 applicable to the HI-STORM 100U System.
3. The descriptions of the thermal modeling of the MPC and its internals, including fuel assemblies, in Section 4.4 which are applicable in their entirety to the HI-STORM 100U.
4. The descriptions of the short-term loading operations, carried out using the HI-TRAC transfer cask, in Section 4.5 which remain applicable in their entirety to the HI-STORM 100U.

As confirmed by appropriate supporting analyses, the heat rejection capability of the HI-STORM 100U System is essentially equivalent to its aboveground counterparts for quiescent conditions (strictly speaking, slightly better, because of the larger intake and outlet passages located in the VVM

<sup>1</sup> For ease of supplement review the sections are numbered in parallel with the main Chapter 4.



lid). Further, its underground configuration renders its resistance to accident events such as fire greater than that of aboveground casks.

#### 4.I.2 THERMAL PROPERTIES OF MATERIALS<sup>1</sup>

The material properties compiled in Section 4.2 of the FSAR provide the required information, except for the material properties of thermal insulation and soil surrounding the HI-STORM 100U VVM, which is not present in the aboveground designs. The functional performance of insulation (applied on the cylindrical surface of the divider shell) is ensured by specifying a minimum thermal resistance. The thermal resistance is defined in conventional US units as the insulation temperature gradient (°F) per unit rate of heat loss (Btu/ft<sup>2</sup>-hr). To compute insulation resistance the insulation thickness is divided by the insulation thermal conductivity. The maximum acceptable conductivity of insulating material is obtained by dividing the insulation thickness by the specified thermal resistance. For use in transient evaluations, the density and specific heat capacity of the insulation are conservatively assumed to be the same as that of air. Property data on insulation and soil is provided in Table 4.I.1.

#### 4.I.3 SPECIFICATIONS FOR COMPONENTS<sup>2</sup>

All applicable material temperature limits in Section 4.3 of the FSAR continue to apply to the HI-STORM 100U. Temperature limits for insulation (used only in the HI-STORM 100U VVM) are specified in Table 2.I.8.

#### 4.I.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE<sup>3</sup>

##### 4.I.4.1 HI-STORM 100U Thermal Model

The HI-STORM 100U underground cask, like the aboveground overpack, is a vertical storage system designed to dissipate heat by ventilation cooling. The principal cask components - container shell, closure lid, support foundation and MPC - are shown in Figure 1.I.1. Internal cask details relevant to cask operation are shown in Figure 1.I.2. The cooling passages in the HI-STORM 100U cask is shown in Figure 1.I.4. As shown in this figure the cask lid is engineered with 360° air inlet and outlet openings. During storage, air enters the inlet opening and flows downwards in the outer annulus gap between the cask container shell and the divider shell. The divider shell is insulated to avoid heating of this incoming air. Near the bottom of the cask cavity, the air U-turns and flow upwards in the inner annulus gap between the MPC and divider shell. During its upward travel, the air extracts heat from the heat generating MPC cylinder. A concomitant effect of heat removal is the monotonic heating of air in the inner annulus. Near the top of the MPC the air enters the internal flow passages in the closure lid. Heated air exits from the vents openings in the top of the closure lid. To model the

<sup>1</sup> This section supplements Section 4.2.

<sup>2</sup> This section supplements Section 4.3.

<sup>3</sup> This section supplements Section 4.4.

ventilation cooling engineered in the underground cask, a 3-D thermal model of the HI-STORM 100U VVM is constructed as described next.

In this Supplement the HI-STORM 100U System consisting of the HI-STORM100U VVM and a loaded MPC is evaluated under normal, off-normal and accident conditions and during short-term operations. The thermal evaluations use the same aboveground MPC 3-D thermal models of the bounding PWR and BWR canisters (MPC-32 and MPC-68) situated in an underground 100U VVM. These models use the same 3-Zone porous media model used in the thermal analysis for the aboveground Overpack (HI-STORM 100S<sup>1</sup>) to represent the flow resistance of bounding BWR (GE-10x10) and PWR (W-17x17) fuel assemblies (See Chapter 4, Subsection 4.4.1.2).

The key attributes of MPC thermal model are as follows:

1. The MPC is modeled as a geometrically accurate 3D array of square shaped cells inside a cylindrical shell with bottom and top closures. The fuel basket bottom and top mouseholes are explicitly modeled as rectangular openings with understated flow area. The MPC model is identical to that described in the main body of Chapter 4.
2. The helium flow within the MPC is modeled as laminar. This is the same modeling approach used in the aboveground cask analyses.
3. The hydraulic resistance of the fuel assemblies stored within the MPC is represented in the 3D model by 3-Zone porous media flow resistances. This is the same as used in the HI-STORM aboveground modeling (See Chapter 4, Subsection 4.4.1.2).

Consistent with the aboveground HI-STORM 100S Overpack modeling a geometrically accurate 3D model of the HI-STORM 100U VVM is constructed for thermal analysis of the belowground casks. The VVM lid with its inlet and outlet vents and internal flow passages, the inner and outer annulus, the U-turn and the gas plenum above the MPC are explicitly modeled. Access to ambient air is artificially restricted in the model by erecting a vertical cylinder above the VVM. The cylinder is open at the top to allow air ingress and exit. In this manner lateral access to air is blocked and the potential for hot air mixing above the VVM is maximized.

The airflow through the cooling passages of the VVM is modeled as turbulent, using the  $k-\omega$  model with transitional option as recommended in the Holtec-proprietary benchmarking report [4.1.6]. This is the same modeling approach as used in the aboveground cask analyses. The underside of the VVM foundation pad (see Figure 1.I.1) is assumed to be supported on a subgrade at 77°F. This is the same boundary condition applied to the bottom of the ISFSI pad for the aboveground cask modeling in Section 4.4. For conservatism heat dissipation from the vertical surfaces of the VVM is suppressed

<sup>1</sup> The aboveground HI-STORM System includes a classical overpack design (HI-STORM 100) and a shortened version (HI-STORM 100S). The limiting design (HI-STORM 100S) is used in the aboveground thermal analysis.

by the assumption of zero heat flux model boundaries. The VVM thermal models are constructed using the same modeling platform used for aboveground analysis (FLUENT version 6.2).

#### 4.I.4.2 Thermal Analysis

The HI-STORM 100U System design has been designed with the objective of ensuring that it meets all temperature and pressure limits set forth in Chapter 2 and Supplement 2.I. In this supplement the HI-STORM 100U System is evaluated to demonstrate compliance with these limits.

The 100U is evaluated under an array of uniform and regionalized heat loads defined in Chapter 2 as a function of regionalization parameter X. To determine the most limiting heat load configuration an array of analyses are performed for the principal storage condition – long-term normal storage and fuel temperatures computed. The results of the analysis for the bounding PWR and BWR canisters (MPC-32 and MPC-68) are summarized in Table 4.I.2. The results show the following:

- i) The fuel temperatures in the underground VVM are essentially the same as or slightly better than fuel temperatures in the aboveground storage overpack (See Table 4.4.6).
- ii) The highest fuel temperatures are reached under regionalized storage at  $X = 0.5$  (same as in the case of the aboveground (HI-STORM 100S) overpack).
- iii) Higher fuel temperatures are reached in MPC-32 (same as in the case of the aboveground (HI-STORM 100S) Overpack).
- iv) Fuel storage in the MPC-32 under regionalized fuel storage at  $X=0.5$  is the limiting scenario for 100U System. This scenario is co-incident with the maximum permissible MPC heat load and therefore temperatures of other sub-systems (such as fuel basket, MPC and VVM) also reach their highest values. This result is also in keeping with the HI-STORM 100S solution presented in Section 4.4 in the main body of Chapter 4. Accordingly, this condition is adopted for evaluation of normal, off-normal and accident events and short-term operations.

Table 4.I.3 presents the HI-STORM 100U maximum temperatures and pressures for the limiting fuel storage scenario defined above. The results are below the Chapter 2 and Supplement 2.I temperature and pressure limits for normal storage.

#### 4.I.4.3 Effect of Elevation

An evaluation of the effects of elevation on thermal performance is performed for the HI-STORM 100U. The methodology described in Subsection 4.4.4.3 is applied for the HI-STORM 100U evaluation. The peak cladding temperatures are calculated for a bounding configuration (non-uniform storage at  $X = 0.5$ ) assuming no reduction of ambient temperature with elevation and compared to the sea level conditions. The results are given in the Table 4.I.4. The results show that the PCT, including the effects of site elevation, continues to be well below the regulatory cladding temperature limit of 752°F for quiescent conditions. In light of the above evaluation, it is not necessary to place any ISFSI elevation constraints for HI-STORM deployment at elevations up to 1500 feet that have quiescent conditions. If, however, an ISFSI is sited at an elevation greater than

sea level with non-quiet conditions, the effect of altitude on the PCT shall be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient and wind conditions.

#### 4.I.4.4 Wind Conditions

Non-quiet ambient conditions defined as a horizontal wind on an isolated HI-STORM 100U module is evaluated. These evaluations conservatively assume a unidirectional wind of sufficient duration for the HI-STORM 100U System to reach the asymptotic maximum (steady-state) temperatures. This event is evaluated using the 3D thermal model mentioned in the foregoing (in Section 4.I.4.1). To properly model wind effects, a half-symmetric model is constructed and non-quiet ambient conditions modeled as a horizontal wind blowing into the HI-STORM 100U inlet and outlet openings from one direction. Because the 100U ventilation openings are axisymmetric the maximum cladding temperature from a wind of a certain velocity is the same regardless of direction. However, under a wind condition, the fuel rod subject to the maximum cladding temperature migrates downstream from the canister's axis of symmetry. Thus, even though the spatial peak cladding temperature remains the same as the wind velocity vector is rotated, the location of the peak changes. Thus, for steady state conditions to be reached, the wind velocity vector (magnitude and direction, and sense of action) must remain constant for a long enough time to enable steady state conditions to be reached. The wind model is used to compute fuel temperatures at several wind speeds and results are tabulated in Table 4.I.7 for the case of  $X=3$ , which corresponds to 30.17 kW. The results show that, depending on magnitude of the wind velocity vector, the peak cladding temperature may be above or below the temperature corresponding to the quiet condition. In particular, the 5 mph wind is identified as being in the narrow range where the peak cladding temperature plateaus to its maximum value.

In recognition of the new design embodiment of HI-STORM 100U, the design basis heat load for the system is based on the assumption that the 5 mph horizontal wind velocity vector prevails for a sufficiently long time. Further, the inlet air temperature into the plenum is assumed to be elevated by 7°K (or 12.6°F) due to intermixing from the presence of other modules in the ISFSI array. This assumption has the effect of raising the design basis inlet air temperature to 92.6°F (from the quiet condition reference value of 80°F).

Furthermore, an inspection of the thermal results for the quiet condition cases in Table 4.I.2 indicates that the case of  $X = 0.5$  produces the largest peak cladding temperature (6°F more than other values of  $X$ ). Therefore  $X = 0.5$  case is used for the thermal analysis. Finally, because the MPC-32 case yields maximum value of the peak cladding temperature (see Table 4.I.2), MPC-32 is selected for computing the allowable heat load. Thus, the thermal problem posed herein assumes that:

- i. A 5 mph horizontal wind (constant speed and direction) is blowing.
- ii. The inlet air is at 92.6°F (not 80°F as assumed in the aboveground HI-STORM simulations), to factor the effect of limited mixing of the feed air with heated exhaust air streams.

- iii. The VVM contains a loaded MPC-32 with the most adverse regionalized storage condition (i.e.,  $X = 0.5$ ).
- iv. For the aboveground system, using Paragraph 2.I.9.1, the aggregate heat load and specific heat loads in Regions 1 and 2 are computed as follows:

$$\begin{aligned} Q_d &= 36.9 \text{ kW} \\ q_1 &= 0.709 \text{ kW} \\ q_2 &= 1.419 \text{ kW} \end{aligned}$$

Therefore, the thermal problem for the constant adverse wind velocity vector case is posed as follows:

Determine the penalty factor  $\alpha$  on  $Q_d$  such that the computed peak cladding temperature is bounded by  $400^\circ\text{C}$ .

The penalty factor  $\alpha$  is computed under the most punitive fuel loading scenario corresponding to  $X = 0.5$  wherein the highest peak cladding temperature is reached as discussed in subsection 4.I.4.2. It therefore follows by physical reasoning that  $\alpha$  computed in this manner is bounding under fuel loading scenarios  $X = 1, 2$  and  $3$  defined in Chapter 2, Paragraph 2.1.9.1. As described later in this section  $\alpha$  is conservatively adopted to penalize the maximum permissible heat loads under  $X = 1, 2$  and  $3$ .

The following peak cladding temperatures were calculated for the above problem for the two different heat loads:

$Q_d$ (kW)	Maximum Cladding Temperature, $^\circ\text{C}$
36.9	423.7
35.05	408.8

By examining the trend of the two cladding temperatures for  $Q$ , an approximately 1 kW additional penalty is taken to reduce the maximum allowable heat load to 33kW, which results in a temperature of approximately  $390^\circ\text{C}$  considering the trend in temperature change. This examination of temperature trend, including the 10 degree margin below  $400^\circ\text{C}$ , affirms the requirements for cladding integrity are satisfied for this design. Therefore, the penalty factor  $\alpha = 33/36.9 = 0.894$ .

The design basis heat load for the range of uniform and regionalized fuel storage scenarios defined in Chapter 2 Paragraph 2.1.9.1 are reduced as shown below:

$X$	Penalized Heat Load ( $Q_r = \alpha \cdot Q$ ) (kW)
0.5	$\alpha \cdot 36.9 \text{ kW} = 33.0 \text{ kW}$
1	$\alpha \cdot 34 \text{ kW} = 30.4 \text{ kW}$

2	$\alpha \cdot 31.48 \text{ kW} = 28.1 \text{ kW}$
3	$\alpha \cdot 30.17 \text{ kW} = 27.0 \text{ kW}$

This reduced heat load  $Q_r$  satisfies the requirement that the peak cladding temperature meet the 400°C under the constant adverse wind velocity vector case. This reduced value of the design basis heat load is conservatively adopted in the system Technical Specification for the “100U” model only.

#### 4.I.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

The short-term evaluations presented in Section 4.5 are applicable in their entirety for the underground VVM.

#### 4.I.6 THERMAL EVALUATION OF OFF-NORMAL AND ACCIDENT CONDITIONS<sup>1</sup>

##### 4.I.6.1 Off-Normal Conditions

###### (a) Elevated Ambient Air Temperature

The elevated ambient air temperature off-normal condition is defined in Table 2.I.1 as an ambient temperature of 100°F. This is 20°F higher than the normal condition ambient temperature of 80°F, also defined in Table 2.I.1 and used in the analyses described in Section 4.I.4.2 above. This condition is conservatively evaluated by adding 20°F to the calculated normal condition fuel cladding and component temperatures in Table 4.I.3. Results for this off-normal condition are presented in Table 4.I.5. The results are well below the permissible short-term temperature limits for fuel cladding, concrete, and structural steels.

###### (b) Partial Blockage of Air Inlets

In contrast to HI-STORM 100S, which features four inlet and four outlet vents, HI-STORM 100U has 360° (radially symmetric) vents making duct blockage in HI-STORM 100U more unlikely. Nevertheless, the case of 50% blockage of the HI-STORM 100U inlet vents is postulated. The partial air inlets blockage event is defined in Table 2.I.1 as the blockage of 50% of the air inlet flow area. This event is conservatively evaluated as a blockage of sufficient duration to reach the asymptotic maximum (steady-state) temperature field. Results for this off-normal condition are presented in Table 4.I.6. The results show large margins of safety with respect to the permissible short-term temperature limits for fuel cladding, concrete, and structural steels.

<sup>1</sup> This section supplements Section 4.6.

#### 4.I.6.2 Accident Conditions

##### (a) Fire

The fire accident is defined in Table 2.I.1 as a 1475°F fire lasting 217 seconds. This is the same intensity and duration as the fire accident evaluated in Section 4.6 of this FSAR for the aboveground overpack. The existing fire evaluation therein bounds the HI-STORM 100U fire event, for the following reasons:

1. Because the fire evaluated in Section 4.6 is an engulfing fire, the cask area exposed to the fire heat flux is maximized. The underground surfaces of the HI-STORM 100U VVM are not directly exposed to the fire heat flux, which significantly reduces the fire heat input to the VVM as compared to an aboveground overpack. The total heat input to the VVM during the fire event is therefore much lower than is evaluated in Section 4.6.
2. The openings of the inlet ducts and outlet ducts are both located near the top of the VVM. Because heated gases rise, a downward flow of combustion gases into the module cavity is not credible. The internal surfaces of the VVM cannot, therefore, be subjected to any significant temperature elevation due to fire.

The above considerations lead to the conclusion that the fire evaluation for the aboveground overpack bounds the HI-STORM 100U fire accident.

##### (b) Flood

The flood accident is defined in Table 2.I.1 as a deep submergence. The worst flood from a thermal perspective is a “smart flood” that just prevents all airflow with *no* MPC cooling by water. Although the HI-STORM 100U includes design features to prevent “smart flood” occurrence such a hypothetical condition is bounded by the 100% inlet ducts blocked accident evaluated in 4.I.6.2(d). As shown in the HI-STORM 100U licensing drawings, the bottom of the MPC is situated several inches below the top of the airflow cutouts in the bottom of the divider shell. Thus, even if the bottom cutouts are substantially covered by flood water the MPC baseplate heat dissipation ensures adequate cooling of the MPC and its stored fuel. This effect is significant, because the thermosiphon convective flow within the MPC is an efficient means of heat rejection to the thermal sink (wetted baseplate).

##### (c) Burial Under Debris

The burial under debris accident is defined in Table 2.I.1 as an adiabatic heat-up at the maximum decay heat load. The existing burial under debris evaluation in Section 4.6 bounds the HI-STORM 100U burial under debris event because the HI-STORM 100U System thermal inertia is greater than that of the aboveground systems. This results from the higher aggregate mass of the VVM as compared to the aboveground overpack. As such the existing burial under debris evaluation for the aboveground overpack bounds the HI-STORM 100U burial under debris accident.

## (d) 100% Blockage of Air Ducts

The 100% air ducts blockage accident is defined in Table 2.I.1 as the blockage of 100% of the air inlet duct flow area. This event is evaluated by blocking the entire inlet opening for a considerable duration (24 hours) and performing a transient calculation of VVM, MPC and cladding temperatures. The only difference between this evaluation and the evaluation described in Section 4.I.4.2 for normal storage is the blockage of the inlet vents and the inclusion of transient effects in the 3D HI-STORM 100U model. Numerical results for this accident are presented in Table 4.I.9. The results demonstrate that all fuel cladding and component temperatures remain below their respective short-term limits.

It should be noted that the increase in temperature would increase the MPC internal pressure. The calculation performed for this accident recognizes an increase in thermosiphon cooling within the MPC that would accompany from pressure increase in a conservative manner.

## (e) Extreme Environmental Temperature

The extreme environmental temperature accident condition is defined in Table 2.I.1 as an ambient temperature of 125°F. This is 45°F higher than the normal condition ambient temperature of 80°F, also defined in Table 2.I.1 and used in the analyses described in Section 4.I.4.2 above. This condition is conservatively evaluated by adding 45°F to the calculated normal condition fuel cladding and component temperatures in Table 4.I.3. Results for this off-normal condition are presented in Table 4.I.8. The results are confirmed to be less than accident temperature limits for fuel cladding, concrete, and ASME Code materials.

It should be noted that an increase in temperature is followed by a concomitant increase in MPC helium pressure. The bounding calculation performed for this accident does not credit the increase in thermosiphon cooling within the MPC that would accompany the pressure increase. As an increase in thermosiphon cooling would limit the temperature rise resulting from an elevated ambient temperature, the calculated temperatures and pressures are conservatively bounding for this event.

4.I.7 REGULATORY COMPLIANCE

As required by ISG-11, the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask materials and fuel cladding are maintained within their temperature limits for normal, off-normal and accident conditions in order to enable



components to perform their intended safety functions. Material temperature limits are summarized in Tables 2.2.3 and 2.I.7. HI-STORM 100U System components defined as important to safety are listed in Tables 2.2.6 and 2.I.7.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 limit of 400°C (752°F).

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this supplement, including conduction, natural convection, and thermal radiation, are passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in Chapters 2 and 3 for normal conditions. All thermal results are within the allowable limits for all normal conditions of storage.

Table 4.I.1

## Thermal Properties for HI-STORM 100U

Insulation	
Thermal Property	Specified Minimum Values
Divider Shell Insulation Thermal Resistance	4 ( $^{\circ}\text{F}\times\text{ft}^2\times\text{hr}$ )/Btu
Density <sup>1</sup>	0.075 lb/ft <sup>3</sup>
Specific Heat Capacity <sup>1</sup>	0.24 Btu/(lb $\times$ $^{\circ}\text{F}$ )
Emissivity	0.5
Soil	
Conductivity <sup>2</sup>	0.3 Btu/ft $\times$ hr $\times$ $^{\circ}\text{F}$

Table 4.I.2

## 100U Long-Term Normal Storage Maximum Fuel Temperatures (Quiescent Condition)

X <sup>3</sup>	Q (kW)	MPC-32 ( $^{\circ}\text{F}$ )	MPC-68 ( $^{\circ}\text{F}$ )
0.5	36.9	<b>711</b> <sup>4</sup>	658
1	34.0	705	656
2	31.48	705	652
3	30.17	705	648

1 Conservatively assumed to be that of air.

2 “Fundamentals of Heat and Mass Transfer”, Table A.3, 4<sup>th</sup> Edition, by F. Incropera and D. Dewitt, John Wiley and Sons.

3 X is defined as the ratio of maximum permissible assembly decay heat generation rates in the inner and outer regions.

4 Highest fuel cladding temperature is highlighted in bold. The co-incident storage scenario, MPC-32 under regionalized fuel storage at X = 0.5, is the limiting scenario for thermal evaluation.

Table 4.I.3

Maximum Normal Temperatures and Pressures Under the Limiting Fuel Storage Scenario  
(Quiescent Condition)

Component	Temperature (°F)	Permissible Limit (°F) (from Tables 2.2.3, 2.1.8 and 2.2.1)	Thermal Margin (°F)
Fuel Cladding	711	752	41
Fuel Basket	707	725	18
Fuel Basket Periphery	607	725	118
MPC Shell	476	500	24
MPC Lid	519	550	31
VVM Container Shell	122	800	678
VVM Lid Bottom Plate	308	800	492
Lid Concrete	270	300	30
Area Averaged Air Outlet Temperature	165	NA	NA
Divider Shell Insulation <sup>1</sup>	356	800	444
Pressure (psig)			
MPC	99.5	100	0.5

<sup>1</sup> To support thermal expansion evaluation in Section 3.I.4.4, the axially averaged divider shell temperature, 272°F, is reported in this footnote.

Table 4.I.4

## Effect of Site Elevation on Peak Cladding Temperature (Quiescent Condition)

MPC Design	PCT at Sea Level (°F)	PCT at 1500 feet (°F)
MPC-68 (BWR)	658	667
MPC-32 (PWR)	711	731

Table 4.I.5

## Maximum Temperatures Under Off-Normal Ambient Temperature (Quiescent Condition)

Component	Temperature (°F)	Permissible Limit (°F) (from Tables 2.2.3, 2.I.8 and 2.2.1)	Thermal Margin (°F)
Fuel Cladding	731	1058	327
Fuel Basket	727	1058	331
Fuel Basket Periphery	627	1058	431
MPC Shell	496	775	279
MPC Lid	539	775	236
VVM Container Shell	142	800	658
VVM Lid Bottom Plate	328	800	472
Lid Concrete Local Temperature	328*	572	244
Divider Shell Insulation	376	800	424
Pressure (psig)			
MPC	101.9	110	8.1

\* Obtained by adding the off-normal-to-normal ambient temperature difference of 20°F to local maximum concrete temperatures during normal conditions.

Table 4.I.6

## Maximum Temperatures Under Partial Blockage of Air Inlets (Quiescent Condition)

Component	Max. Temperature (°F)	Permissible Limit (°F) (from Tables 2.2.3, 2.I.8 and 2.2.1)	Thermal Margin (°F)
Fuel Cladding	744	1058	314
Fuel Basket	741	1058	317
Fuel Basket Periphery	631	1058	427
MPC Shell	494	775	281
MPC Lid	546	775	229
VVM Container Shell	130	800	670
VVM Lid Bottom Plate	338	800	462
Lid Concrete Local Temperature	339	572	233
Divider Shell Insulation	379	800	421
Pressure (psig)			
MPC	102.4	110	7.6

Table 4.I.7

## Effect of Wind on Peak Cladding Temperatures (X=3)

Wind Speed (mph)	Peak Clad Temperature (°F)
5	749
10	713
15	676

Table 4.I.8

## Results Under Extreme Environmental Temperature Accident (Quiescent Condition)

Component	Max. Temperature (°F)	Permissible Limit (°F) (from Tables 2.2.3, 2.I.8 and 2.2.1)	Thermal Margin (°F)
Fuel Cladding	756	1058	302
Fuel Basket	752	1058	306
Fuel Basket Periphery	652	1058	406
MPC Shell	521	775	254
MPC Lid	564	775	211
VVM Container Shell	167	800	633
VVM Lid Bottom Plate	353	800	447
Lid Concrete Local Temperature	353*	572	219
Divider Shell Insulation	401	800	399
Pressure (psig)			
MPC	104.9	200	95.1

\* Obtained by adding the extreme ambient to normal ambient temperature difference of 45°F to local maximum concrete temperatures during normal conditions.

Table 4.I.9

## Results Under 24-Hour 100% Air Inlets Blockage Accident (Quiescent Condition)

Component	Max. Temperature (°F)	Permissible Limit (°F) (from Tables 2.2.3, 2.I.8 and 2.2.1)	Thermal Margin (°F)
Fuel Cladding	942	1058	116
Fuel Basket	938	1058	120
Fuel Basket Periphery	805	1058	253
MPC Shell	681	775	94
MPC Lid	627	775	148
VVM Container Shell	254	800	546
VVM Lid Bottom Plate	475	800	325
Lid Concrete Local Temperature	477	572	95
Divider Shell Insulation	632	800	168
Pressure (psig)			
MPC	129.5	200	70.5

## SUPPLEMENT 4.III<sup>1</sup>

### THERMAL EVALUATION OF THE MPC-68M

#### 4.III.0 OVERVIEW

The MPC-68M is a 68 cell BWR canister engineered with a high B<sup>10</sup> containing Metamic-HT basket for enhanced criticality control. The MPC-68M is evaluated for storage in the aboveground family of HI-STORM overpacks. For a bounding evaluation an MPC-68M emplaced in the most flow resistive HI-STORM 100S Version B overpack<sup>2</sup> is analyzed under normal, off-normal and accident conditions. The evaluations described herein parallel those of the aboveground HI-STORM cask contained in the main body of Chapter 4 of this FSAR. In addition, a new heat load layout is added which is referred to as the "Quarter Symmetric Heat Load" (QSHL) pattern. In this pattern, the maximum permissible heat load in each storage cell, q, is specific to its location within the quadrant and is limited to a unique prescribed value given in Figure 2.III.1. This QSHL pattern seeks to minimize the large temperature differences between cladding temperatures in proximate fuel assemblies and is especially suited for canisterizing of fuel with widely varying specific heat loads such as at a plant undergoing decommissioning.

It should be noted that the QSHL pattern is a special case of regionalized loading, but is identified simply as "QSHL" to avoid confusion.

To ensure readability, the section in the main body of the chapter to which each section in this supplement corresponds is clearly identified. All tables in this supplement are labeled sequentially.

#### 4.III.1 INTRODUCTION

The information presented in this supplement is intended to serve as a complement to the information provided in the main body of Chapter 4. Except for the fuel basket, basket support, and basket shim materials, the information in Chapter 4 that remains applicable to the MPC-68M analysis is not repeated herein. Specifically the following information in the main body of Chapter 4 is not repeated:

1. The thermal properties of materials in Section 4.2 applicable to the MPC-68M.
2. The specifications for components in Section 4.3 applicable to the MPC-68M.
3. The descriptions of the thermal modeling of the MPC and its internals, including fuel assemblies, in Section 4.4 which are applicable in their entirety to the MPC-68M.

1 For ease of supplement review the sections are numbered in parallel with the main Chapter 4.

2 This approach is identical to the HI-STORM thermal analysis in Section 4.4.



4. The descriptions of the short-term loading operations, carried out using the HI-TRAC transfer cask, in Section 4.5 applicable to the MPC-68M.

As confirmed by appropriate supporting analyses, the heat rejection capacity of the MPC-68M<sup>1</sup> is equal to or better than its counterparts (strictly speaking, much better because of the highly conducting Metamic-HT fuel basket). This renders its resistance to accident events such as fire with greater margins of safety.

#### 4.III.2 THERMAL PROPERTIES OF MATERIALS<sup>2</sup>

The material properties compiled in Section 4.2 of the FSAR provide the required information, except for the material properties of Metamic-HT fuel basket, aluminum basket shims<sup>3</sup> and solid shims. The Metamic-HT and shims thermo-physical properties data is provided in Table 4.III.1.

#### 4.III.3 SPECIFICATIONS FOR COMPONENTS<sup>4</sup>

All applicable material temperature limits in Section 4.3 of the FSAR continue to apply to the MPC-68M. Temperature limits of MPC-68M fuel basket and basket shim materials is specified in Table 4.III.2.

#### 4.III.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE<sup>5</sup>

##### 4.III.4.1 Thermal Model

The MPC-68M thermal design is same as that of the currently licensed MPC-68. It features a 68 cells capacity fuel basket for storing BWR fuel. The basket is engineered with a bottom plenum by providing flow holes, a top open plenum by providing an engineered clearance and a peripheral downcomer to facilitate heat dissipation by thermosiphon action. The MPC-68M is helium pressurized to backfill specifications as discussed below:

- Case 1: Initial helium backfill pressure is defined in Chapter 4, Table 4.4.12 under uniform and regionalized loading based on regionalization parameter X (Section 2.1.9), and
- Case 2: Initial helium backfill pressure is defined in Table 1.III.1 under QSHL pattern defined in Supplement 2.III.

<sup>1</sup> Heat rejection capacity is defined as the amount of heat the storage system containing an MPC loaded with CSF stored in uniform storage will reject with the ambient environment at the normal temperature and the peak fuel cladding temperature at 400°C.

<sup>2</sup> This section supplements Section 4.2.

<sup>3</sup> The terms basket shims and extruded shims are interchangeably used in this chapter.

<sup>4</sup> This section supplements Section 4.3.

<sup>5</sup> This section supplements Section 4.4.

The principal differences are in the basket material of construction (Metamic-HT), the installation of aluminum basket shims in the basket peripheral spaces and replacement of the cell walls sandwich construction by monolithic (i.e. gaps free) basket panels. The design characteristics of the basket are as follows:

- i. The fuel basket is assembled from a rectilinear gridwork of thick plates having precision machined slots for facilitating snug-fit assembly and ensuring uninterrupted lateral dissipation of heat.
- ii. Aluminum basket shims conforming to the shapes of the fuel basket and MPC shell are installed in the peripheral spaces between the outside walls of the fuel basket and the inside walls of the Enclosure Vessel. The axial holes in the basket shims serve as the passageway for the downward flow of the helium gas under the thermosiphon action, which is intrinsic to the thermal design of all MPCs in the HI-STORM 100 system.
- iii. The fuel basket consists of adjacent square openings (cells) separated by one monolithic wall of the Metamic-HT neutron absorber.

In this supplement the MPC-68M placed in an above ground HI-STORM 100 System is evaluated under normal, off-normal and accident conditions and during short-term operations. The thermal evaluations use *the same aboveground MPC 3-D thermal modeling methodology* and the *same 3-Zone porous media model* used in the thermal analysis of the aboveground overpack (HI-STORM 100S<sup>1</sup>) to represent the flow resistance of bounding BWR (GE-10x10) fuel assemblies (See Chapter 4, Subsection 4.4.1.2).

The key attributes of MPC-68M thermal model are as follows:

1. The MPC-68M is modeled as a geometrically accurate 3D array of square shaped cells inside a cylindrical shell with bottom and top closures. The fuel basket bottom flow holes with understated flow area and top plenum are explicitly modeled.
2. The helium flow within the MPC is modeled as laminar. This is the same modeling approach used in the HI-STORM 100 cask analyses.
3. The hydraulic resistance of the fuel assemblies stored within the MPC is represented in the 3D model by 3-Zone porous media flow resistances. This is the same as used in the HI-STORM 100 modeling (See Chapter 4, Subsection 4.4.1.2).

Consistent with the HI-STORM 100 cask analyses a geometrically accurate 3D model of the HI-STORM 100 overpack is constructed for thermal analysis. The inlet and outlet vents and internal flow passages are explicitly modeled. The airflow through the cooling passages of the HI-STORM

<sup>1</sup> The aboveground HI-STORM System includes a classical overpack design (HI-STORM 100) and a shortened version (HI-STORM 100S). The limiting design (HI-STORM 100S) is used in the aboveground thermal analysis.

100 overpack is modeled as turbulent, using the  $k-\omega$  model with transitional option as recommended in the Holtec-proprietary benchmarking report [4.1.6]. This is the same modeling approach used in the HI-STORM 100 cask analyses. The underside of the HI-STORM 100 concrete pad is assumed to be supported on a subgrade at 77°F. This is the same boundary condition applied to the bottom of the ISFSI pad for the HI-STORM 100 modeling in Section 4.4.

#### 4.III.4.2 Thermal Analysis

The MPC-68M has been designed to permit storage under the array of uniform and regionalized heat loads defined in Chapter 2 as a function of the regionalization parameter  $X$ . As shown in Chapter 4 the highest cladding temperatures are reached under regionalized storage at  $X = 0.5$ . This scenario is co-incident with the maximum permissible MPC heat load and therefore temperatures of other sub-systems (such as fuel basket, MPC shell and overpack) also reach their highest values. The fuel cladding temperature under long term storage in HI-STORM is presented in Table 4.III.3.a.

The QSHL pattern has also been analyzed using the same FLUENT model previously used in this FSAR: no changes were made to the existing thermal model. The selected heat loads in Figure 2.III.1 are suitably limited to ensure that the peak cladding temperature in the MPC remains below that in the governing MPC analyzed in this FSAR (MPC-32) under all thermal scenarios. Thus the peak cladding temperature for the QSHL pattern is limited by a previously analyzed and licensed MPC.

Other important safety aspects of the QSHL pattern are:

1. The hottest fuel assemblies are located in-board of the peripheral locations in the basket so that the colder fuel in the peripheral cells helps block the radiation emitted by the hottest fuel assemblies.
2. The cell specific heat load,  $q$ , provided in Figure 2.III.1 is the maximum value permitted for that location. In virtually every case, the actual heat load in every cell will be lower than the allowed limit, thus resulting in a lower cladding temperature field overall than that computed herein.
3. The fuel cladding temperature for QSHL pattern under long term storage in HI-STORM is presented in Table 4.III.3.a. The predicted PCT is higher than that for the scenario with decay heat based on regionalized parameter  $X$  defined in Chapter 2. For this reason, QSHL pattern is adopted as the licensing basis pattern for MPC-68M.
4. The PCT and basket temperatures under the QSHL pattern is lower than that in the thermally governing case (MPC-32)

This QSHL scenario is adopted for demonstration of compliance with the temperature and pressure limits set forth in this Supplement and Chapter 2. The limiting scenario is analyzed and maximum temperatures and pressures under normal storage tabulated in Tables 4.III.3b and 4.III.4. The results

are below the Chapter 2 and Supplement 4.III normal temperature and pressure limits. In accordance with NUREG-1536 MPC-68M pressures are computed assuming 1% (normal), 10% (off-normal) and 100% (accident) rod ruptures with 100% rods fill gases and fission gases release in accordance with NUREG-1536 release fractions. The pressures are computed and tabulated in Table 4III.4. The 100% rods rupture pressure is below the accident design pressure (Table 2.2.1).

#### 4.III.4.3 Engineered Clearances to Eliminate Thermal Interferences

To minimize thermal stresses in load bearing members, the MPC-68M is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansions are evaluated to ensure the adequacy of engineered gaps. The following gaps are evaluated:

- a. Fuel Basket-to-MPC Radial Gap
- b. Fuel Basket-to-MPC Axial Gap
- c. MPC-to-Overpack Radial Gap
- d. MPC-to-Overpack Axial Gap

The FLUENT thermal model articulated above provides the temperature field in the HI-STORM overpack and MPC-68M from which the changes in the above gaps are directly computed. The nominal cold gaps are presented on the drawings in Section 1.5 and the corresponding differential expansions under normal storage conditions are presented in Table 4.III.8. The calculations show significant margins against restraint to free-end expansion are available in the design.

#### 4.III.4.4 Evaluation of Fuel Debris Storage

Fuel debris is permitted for storage in up to eight peripheral cells under the permitted heat load pattern shown in Figure 2.III.1. Although fuel debris is not required to meet cladding temperature limits, its effect on fuel stored in the interior cells must be assessed. Fuel debris in the canister is thermally conservatively evaluated assuming a bounding debris configuration and design heat load in all storage cells. The following assumptions are adopted to maximize the computed cladding temperatures:

1. The fuel debris is assumed to be completely pulverized and compacted into a square prismatic bar enclosed by the damaged fuel canister (DFC) with open helium space above it. In this manner the height of the prismatic bar emitting heat is minimized resulting in the maximization of lineal thermal loading (kw/ft) of the DFC and co-incident local heating of the fuel basket and neighboring storage cells.
2. Fuel debris assumed to be completely composed of UO<sub>2</sub>. As UO<sub>2</sub> has a lower conductivity relative to cladding, heat dissipation is understated.
3. The fuel debris is assumed to block through flow of helium inside the DFC.
4. All 16 peripheral storage locations (not just the 8 permitted by CoC) are assumed to contain fuel debris emitting maximum heat permitted by Technical Specifications (CoC)

Appendix B, Figure 2.4-1) and all interior cells are emitting design basis heat under the applicable heat load scenario.

5. The MPC operating pressure is understated to minimize internal convection heat transfer

The results of the analysis are tabulated in Table 4.III.11. The results support the following conclusions:

- Cladding temperature is substantially below the ISG-11, Rev. 3 limit.
- MPC basket is below the design limit (Table 4.III.2) by large margin.
- MPC shell and Overpack metal temperatures are below design limits (Table 2.2.3).
- Overpack body and lid concrete are well below design limits (Table 4.3.1).

#### 4.III.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

##### 4.III.5.1 HI-TRAC Thermal Model

The HI-TRAC thermal model presented in Section 4.5 is adopted for the evaluation of MPC-68M under short term operations.

##### 4.III.5.2 Maximum Time Limit During Wet Transfer Operations

Time-to-boil is calculated using the same methodology described in Section 4.5.2. Table 4.III.13 summarizes the thermal inertia of the constituent components in the loaded HI-TRAC transfer cask. Using the methodology described in Section 4.5.2, the time-to-boil is provided at representative initial temperatures for maximum QSHL in Table 4.III.14. This is an example calculation for the maximum design basis heat load. The same methodology can be adopted to determine the time-to-boil for canisters loaded at lower heat loads. An alternate method using the FLUENT thermal model described in Section 4.III.5.1 can be adopted to evaluate the time for water within the MPC to boil for site-specific conditions. Principal modeling steps and acceptance criteria are defined in Table 4.5.10a.

##### 4.III.5.3 MPC Temperature During Moisture Removal Operations

###### 4.III.5.3.1 Vacuum Drying

Prior to helium backfill the MPC-68M must be drained of water and demoisturized. At the start of draining operation, both the HI-TRAC annulus and the MPC are full of water. The presence of water in the MPC ensures that the fuel cladding temperatures are lower than design basis limits by large margins. As the heat generating region is uncovered during the draining operation, the fuel and basket mass will undergo a monotonic heat up from the initially cold conditions when the heated surfaces were submerged under water. To limit fuel temperatures demoisturization of the MPC-68M by the vacuum drying method is permitted provided the HI-TRAC annulus remains water filled

during vacuum drying operations. To support vacuum drying operations two limiting scenarios are defined below:

Scenario A: The MPC-68M is loaded with Moderate Burnup Fuel assemblies generating heat at the maximum permissible rate defined in Chapter 2 under the bounding regionalized storage scenario  $X = 0.5$ .

Scenario B: The MPC-68M is loaded with one or more High Burnup Fuel assemblies and the MPC-68M decay heat is less than a conservatively defined threshold heat load  $Q = 29 \text{ kW}^1$ .

Scenario C: The MPC-68M is loaded with Moderate Burnup Fuel assemblies generating heat at the maximum permissible rate defined under QSHL pattern (Figure 2.III.1).

To evaluate the above scenarios the vacuum drying analysis methodology presented in Section 4.5 is adopted and an MPC-68M specific thermal model constructed. The principal features of the thermal model are as follows:

- i. A bounding steady-state analysis is performed under the heat loads defined in the scenarios above.
- ii. The water in the HI-TRAC annulus is conservatively assumed to be boiling under the hydrostatic head of water at the annulus bottom (232°F).
- iii. The bottom surface of the MPC is insulated.

The thermal model articulated above is used to compute the maximum cladding temperature under the vacuum drying scenarios defined above. The results tabulated in Table 4.III.5 are in compliance with the ISG-11 temperature limits of Moderate Burnup Fuel (Scenario A) and High Burnup Fuel (Scenario B).

At heat loads greater than threshold heat load defined above, the peak cladding temperature cannot be maintained below the ISG-11, Revision 3 limit of 400°C for HBF under a vacuum condition of infinite duration. Under this scenario, cycles of vacuum drying resulting in heatup followed with cooling by helium are performed until drying criteria is achieved. The thermal model described above is used for heatup/cooldown cycles for site-specific canister heat load maps. It must be ensured per ISG-11 Rev 3 that the repeated thermal cycling is limited to less than 10 cycles, with cladding temperature variations less than 65°C (117°F) each.

#### 4.III.5.3.2 Forced Helium Dehydration

Evaluation of Forced Helium Dehydration in Section 4.5 is applicable to MPC-68M.

<sup>1</sup> Threshold heat load is defined as the product of maximum loaded assembly heat load  $r_{\text{max}}$  and the number of fuel storage cells ( $n=68$ ). Under this stipulation  $r_{\text{max}}$  must not exceed 0.426 kW.

#### 4.III.5.4 Cask Cooldown and Reflood During Fuel Unloading Operations

Evaluation of cask cooldown and reflood operation in Section 4.5 is applicable to MPC-68M.

#### 4.III.5.5 HI-TRAC Onsite Transfer Operation

A 3D FLUENT thermal model of an MPC-68M emplaced in a HI-TRAC transfer cask is constructed to evaluate the thermal state of fuel under onsite transport in the vertical orientation<sup>1</sup>. A bounding analysis is performed under the following conditions:

- (i) Steady state maximum temperatures have reached.
- (ii) The MPC-68M is loaded with fuel generating heat at the maximum permissible level under the limiting Quarter Symmetric Heat Load (QSHL) pattern.
- (iii) The HI-TRAC annulus is air filled.

The scenario defined above represents upper bound temperatures reached in the HI-TRAC without the aid of any auxiliary cooling such as the Supplemental Cooling System (SCS) defined in Section 4.5. The maximum cladding temperatures computed using the thermal model articulated above are tabulated in Table 4.III.6. As the cladding temperatures are below the limiting High Burnup Fuel temperature limits mandated by ISG-11 [4.1.4] SCS cooling is not necessary for ensuring cladding safety under onsite transfer operations for the MPC-68M canister. Accordingly SCS cooling is not mandated in the MPC-68M Technical Specifications. Additionally, the peak fuel cladding temperatures are bounded by MPC-32 (Section 4.5).

#### 4.III.5.6 Sensitivity Study

In lieu of anodization of the extruded shims used in the MPC-68M, they may be passivated in water to form a thin oxide layer. The emissivity of extruded shim surfaces is therefore reduced and requires a thermal evaluation.

In addition to the above, the radial gap between the basket, extruded shims and MPC shell is controlled using thin solid aluminum plates, which may be inserted between the basket and extruded shims to meet the gap criterion specified on the drawing (see Section 1.III.5). If the gap criterion on the drawing is met, solid shims are not required. These solid thin shim plates are made of aluminum and are supported by the extruded shims. A thermal analysis is performed in this subsection to determine the effect of these thin solid shim plates and low emissive extruded shims.

The following changes are made to the thermal model discussed in previous sub-sections to study the impact of the above mentioned design enhancements:

1. The panel notch gap on each side of the intersecting basket panels is increased to 0.8mm.

<sup>1</sup> In accordance with Section 4.5 onsite transfer in the horizontal orientation is not permitted.

2. The gap between the basket and extruded shims is modeled with an effective thermal conductivity. The effective thermal conductivity of the gap between the basket and extruded shims is calculated based on a two-dimensional CFD model. This 2-D model includes the solid shim placed between the basket and extruded shims. A schematic of the model is shown in Figure 4.III.1.
3. A conservatively lowerbound emissivity of 0.03 is used for the passivated extruded shim surfaces.
4. Emissivity of solid shims is shown in Table 4.III.1.

The solid shims are conservatively modeled to be equidistant from the basket wall and extruded shim wall. The effective thermal conductivity of the gap between the basket and extruded shims with the presence of solid shim plate bounds the scenario without the presence of solid shims. The sensitivity study documented herein therefore considers only the scenario with solid shim plate placed in the gap between the basket and extruded shim.

A sensitivity study is performed to evaluate the most limiting thermal scenario with least margins to fuel cladding temperature limit i.e. vacuum drying Scenario B defined in Section 4.III.5.3. The results of the sensitivity study to evaluate the effect of design enhancements made to the MPC and its contents are reported in Table 4.III.12. The results demonstrate that fuel temperature is well below its temperature limit and is also bounded by the results based on the licensing basis thermal model in Table 4.III.5.

Therefore, the design enhancements discussed in this subsection are bounded by the licensing basis thermal analysis documented in this chapter. No additional thermal analysis for other conditions (also considering the large temperature margins to limits) is therefore warranted.

#### 4.III.6 THERMAL EVALUATION OF OFF-NORMAL AND ACCIDENT CONDITIONS<sup>1</sup>

##### 4.III.6.1 Off-Normal Conditions

###### (a) Elevated Ambient Air Temperature

This off-normal event is defined in Paragraph 4.6.1.2. The principal effect of elevated ambient temperature is a rise of the HI-STORM 100 temperatures from the baseline normal storage temperatures by the difference between elevated ambient and normal ambient temperatures. The results of this event (maximum temperatures and pressures) are provided in Table 4.III.15. The results are below the off-normal condition temperature and pressure limits (Tables 2.2.1, 4.III.2 and 2.2.3).

<sup>1</sup> This section supplements Section 4.6.



## (b) Partial Blockage of Air Inlets

This off-normal event is defined in Paragraph 4.6.1.3. The principal effect of partial inlet vent blockage is a temperature rise in HI-STORM 100 System components from the baseline normal storage temperatures. Reasonably bounding evaluations in Paragraph 4.6.1.3 yield a certain rise in fuel cladding and component temperatures due to this off-normal event. This temperature adder is applied to HI-STORM MPC-68M storage temperature field in Table 4.III.3b and presented in Table 4.III.15. The results yield substantial margins for assuring safe storage of spent nuclear fuel, fuel basket and MPC confinement boundaries.

(c) Off-Normal Pressure

This event is defined as a combination of (a) maximum helium backfill pressure (Table 1.III.1), (b) 10% fuel rods rupture, and (c) limiting fuel storage configuration. The principal objective of the analysis is to demonstrate that the MPC off-normal design pressure (Table 2.2.1) is not exceeded. The MPC-68M off-normal pressure is reported in Table 4.III.4. The result<sup>1</sup> is below the off-normal design pressure (Table 2.2.1).

## 4.III.6.2 Accident Conditions

## (a) Fire

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low due to the lack of combustible materials at an ISFSI, a conservative fire event has been assumed and analyzed. The only credible concern is a fire from an on-site transport vehicle fuel tank. Under a postulated fuel tank fire, the outer layers of HI-TRAC or HI-STORM overpacks are heated for the duration of fire by the incident thermal radiation and forced convection heat fluxes. The amount of fuel in the on-site transporter is limited to a volume of 50 gallons.

(i) HI-STORM Fire<sup>2</sup>

The fuel tank fire is conservatively assumed to surround the HI-STORM Overpack. Accordingly, all exposed overpack surfaces are heated by radiation and convection heat transfer from the fire. Based on NUREG-1536 and 10 CFR 71 guidelines [4.III.2], the following fire parameters are assumed:

1. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively

1 Pressures relative to 1 atm absolute pressure (i.e. gauge pressures) are reported throughout this section.

2 The HI-STORM fire accident methodology is same as the generic methodology in Section 4.6 of the HI-STORM 100 FSAR.

assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.

2. The average flame temperature must be at least 1475°F (800°C). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces, which is very conservative considering the size of the HI-STORM cask. It is therefore conservative to use the 1475°F (800°C) temperature.
3. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration.
4. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal measurements [4.III.3], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft<sup>2</sup>×°F) is applied to exposed overpack surfaces during the short-duration fire.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width [4.III.2], the fuel ring surrounding the overpack covers 147.6 ft<sup>2</sup> and has a depth of 0.54 in. From this depth and fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.62 minutes. The fuel consumption rate of 0.15 in/min is a lowerbound value from a Sandia National Laboratories report [4.III.3]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

To evaluate the impact of fire heating of the HI-STORM overpack, a thermal model of the overpack cylinder was constructed and evaluated in Section 4.6 of the HI-STORM FSAR with overstated inputs. As justified below this overpack fire analysis remains conservative. It is recognized that the ventilation air in contact with the inner surface of the HI-STORM Overpack under design-basis decay heat varies between 80°F at the bottom and 275°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy a miniscule fraction of area of the cylindrical surface of the massive HI-STORM Overpack. Due to the short duration of the fire event and the relative isolation of the ventilation passages from the outside environment, the ventilation air is expected to experience little intrusion of the fire combustion products. However, as a conservative measure the air in the HI-STORM Overpack ventilation passages was held constant at a substantially elevated temperature (300°F) during the entire duration of the fire event.

During the fire the overpack external shell temperatures are substantially elevated (~550°F). All of concrete remains below its fire accident temperature limit (Table 2.2.3).

The results demonstrate that the fire accident event analyzed in a most conservative manner is determined to have a minor effect on the HI-STORM Overpack. The concrete temperatures at all locations remain below the short term temperature limit. The temperatures of steel structures are within allowable limits.

Having evaluated the effects of the fire on the overpack, we now evaluate the effects on the MPC-68M and contained fuel assemblies. Guidance for the evaluation of the MPC and its internals during a fire event is provided by NUREG-1536 (4.0,V,5.b), which states:

“For a fire of very short duration (i.e., less than 10 percent of the thermal time constant of the cask body), the NRC finds it acceptable to calculate the fuel temperature increase by assuming that the cask inner wall is adiabatic. The fuel temperature increase should then be determined by dividing the decay energy released during the fire by the thermal capacity of the basket-fuel assembly combination.”

The time constant of the cask body (i.e., the overpack) can be determined using the formula:

$$\tau = \frac{c_p \times \rho \times L_c^2}{k}$$

where:

$c_p$  = Overpack Specific Heat Capacity (Btu/lb-°F)

$\rho$  = Overpack Density (lb/ft<sup>3</sup>)

$L_c$  = Overpack Characteristic Length (ft)

$k$  = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (140 lb/ft<sup>3</sup>) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM Overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 140 \times 2.46^2}{1.05} = 126 \text{ hrs}$$

One-tenth of this time constant is approximately 12.6 hours (756 minutes), substantially longer than the fire duration of 3.62 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The fuel temperature rise is computed next.

Table 4.III.13 lists lower-bound thermal inertia values for the MPC-68M and the contained fuel assemblies. Applying design heat load (42.8 kW (1.46x10<sup>5</sup> Btu/hr)) and adiabatic heating for the 3.62 minutes fire, the fuel temperature rise computes as:

$$\Delta T_{fuel} = \frac{\text{Decay heat} \times \text{Time duration}}{(\text{MPC} + \text{Basket \& Shims} + \text{Fuel}) \text{ Thermal Inertia}} = \frac{1.46 \times 10^5 \frac{\text{Btu}}{\text{hr}} \times \left(\frac{3.62}{60}\right) \text{hr}}{(2184 + 1307 + 931 + 2665) \text{Btu}/^\circ\text{F}} = 1.25^\circ\text{F}$$

This is a very small increase in fuel temperature. Consequently, the impact on the MPC internal helium pressure will be quite small. Based on a conservative analysis of the HI-STORM 100 System response to a hypothetical fire event, it is concluded that the fire event does not adversely affect the temperature of the MPC or contained fuel. We conclude that the ability of the HI-STORM 100 System to cool the spent nuclear fuel within design temperature limits during and after fire is not compromised.

An alternate method using the FLUENT thermal model described in Section 4.III.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FSAR. Principal modeling steps and acceptance criteria are defined in Table 4.6.10.

#### (ii) HI-TRAC Fire<sup>1</sup>

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 100-ton HI-TRAC is performed. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC. In this analysis, the contents of the HI-TRAC are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat input and heat input from the short duration fire. The rate of temperature rise of the HI-TRAC depends on the thermal inertia of the cask, the cask initial conditions, the spent nuclear fuel decay heat generation, and the fire heat flux. Using conservatively bounding inputs – lowerbound thermal inertia, steady state maximum cask temperatures (Table 4.III.6) and design heat load (42.8 kW) - a bounding cask temperature rise of 5.21°F per minute is computed from the combined radiant and forced convection fire and decay heat inputs to the cask. During the handling of the HI-TRAC transfer cask, the transporter is limited to a maximum of 50 gallons. The duration of the 50-gallon fire using the methodology articulated above for HI-STORM fire is 4.775 minutes. Therefore, the temperature rise computed as the product of the rate of temperature rise and the fire duration is 24.9°F, and the co-incident fuel cladding temperature (734°F)<sup>2</sup> is below the 1058°F accident limit.

The elevated temperatures as a result of the fire accident will cause the pressure in the water jacket to increase and cause the overpressure relief valves to vent steam to the atmosphere. Based on the fire heat input to the water jacket, less than 11% of the water in the water jacket can be boiled off. However, it is conservatively assumed, for dose calculations, that all the water in the water jacket is lost. In the 125-ton HI-TRAC, which uses Holtite in the lids for neutron shielding, the elevated fire temperatures would cause the Holtite to exceed its design accident temperature limits. It is conservatively assumed, for dose calculations, that all the Holtite in the 125-ton HI-TRAC is lost.

<sup>1</sup> The HI-TRAC fire accident methodology is same as the generic methodology in Section 4.6 of the HI-STORM 100 FSAR.

<sup>2</sup> Computed by adding the fire temperature rise to initial fuel temperature (Table 4.III.6).

Due to the increased temperatures the MPC experiences as a result of the fire accident in the HI-TRAC transfer cask, the MPC internal pressure increases. The pressure rise is computed using the Ideal Gas Law and upperbound helium backfill pressure defined in Table 1.III.1 and results tabulated in Table 4.III.9. The computed MPC accident pressure is substantially below the accident design pressure (Table 2.2.1).

An alternate method using the FLUENT thermal model described in Section 4.III.5 can be adopted to evaluate HI-TRAC site-specific fire accident event. Principal modeling steps and acceptance criteria are defined in Table 4.6.11.

(b) Flood

The flood accident is defined in Chapter 2 as a deep submergence event. The worst flood from a thermal perspective is a “smart flood” that just rises to the top of the inlets to prevent airflow without the benefit of MPC cooling by water. This effect is bounded by the 100% inlets ducts blockage accident evaluated herein in Section 4.III.6.2(d).

(c) Burial Under Debris

This accident event is defined in Paragraph 4.6.2.5. The methodology for the burial under debris evaluation in Section 4.6 is employed to determine the minimum available time for the fuel cladding to reach the accident limit. Using the equation presented in Paragraph 4.6.2.5 and same clad temperature margin presented in Table 4.6.6, burial time is obtained and presented in Table 4.III.16. The coincident MPC pressure is also computed and compared with the accident design pressure (Table 2.2.1). The result (Table 4.III.16) is confirmed to be below the permissible limit.

(d) 100% Blockage of Air Ducts

This accident is defined in Section 4.6 as 100% blockage of the air inlet ducts. This event is evaluated by blocking the air inlets in the FLUENT thermal model and computing the temperature rise of the MPC and stored fuel with time. The results of the blocked ducts transient analysis that support the required action completion times for clearing the inlets are tabulated in Table 4.III.7. The results show that fuel cladding and component temperatures remain below their respective accident limits specified in Chapter 2 and Supplement 4.III. The increase in temperature results in a concomitant rise of the MPC pressure. The maximum accident pressure tabulated in Table 4.III.7 is below the design limit specified in Chapter 2.

Since the temperatures of MPC-68M are bounded by the MPCs evaluated in Chapter 4, threshold heat load defined in Table 4.6.8a can also be adopted for MPC-68M. A threshold heat load is defined in Table 4.6.8a at or below which periodic surveillance or vent blockage corrective actions defined in Sub-Section 11.2.13.2 are applicable.

## (e) Extreme Environmental Temperature

The accident event is defined in Paragraph 4.6.2.3. The principal effect of elevated ambient temperature is a rise of the HI-STORM 100 temperatures from the baseline normal storage temperatures by the difference between elevated ambient and normal ambient temperatures. The results of this event (maximum temperatures and pressures) are provided in Table 4.III.17. The results are below the accident condition temperature and pressure limits (Tables 2.2.1, 4.III.2 and 2.2.3).

## (f) 100% Rods Rupture Accident

In accordance with NUREG-1536 a 100% rods rupture accident is evaluated assuming 100% of the rods fill gases and fission gases release in accordance with NUREG-1536 release fractions. The MPC-68M pressure under this postulated accident is computed and tabulated in Table 4.III.4. The pressure is below the accident design pressure (Table 2.2.1).

## (g) Jacket Water Loss

The principal effect of jacket water loss accident is a temperature increment in the stored fuel and MPC from the baseline conditions under in a HI-TRAC. As the MPC-68M temperatures in the HI-TRAC are bounded by MPC-32 temperatures (see Table 4.5.6) the jacket water loss temperatures are likewise bounded by the HI-TRAC jacket water loss evaluation in Section 4.6.

4.III.7 REGULATORY COMPLIANCE

As required by ISG-11, the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions. Design pressures are specified in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask materials and fuel cladding are maintained within their temperature limits under normal, off-normal and accident conditions to enable them to perform their intended safety functions. Material temperature limits are specified in Tables 2.2.3 and 4.III.2.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 limit of 400°C (752°F).

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this supplement, including conduction, natural convection, and thermal radiation, are passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the normal storage design criteria specified in Chapters 2 and 4. All thermal results are within the limits under normal conditions of storage.

#### 4.III.8 REFERENCES

- [4.III.1] Aluminum Alloy 2219 Material Data Sheet, ASM Aerospace Specification Metals, Inc., Pompano Beach, FL.
- [4.III.2] United States Code of Federal Regulations, Title 10, Part 71.
- [4.III.3] Gregory, J.J. et. al., "Thermal Measurements in a Series of Large Pool Fires", SAND85-1096, Sandia National Laboratories, (August 1987).
- [4.III.4] Jakob, M. and Hawkins, G.A., "Elements of Heat Transfer," John Wiley & Sons, New York, (1957).

Table 4.III.1: Thermal Properties of Fuel Basket, Basket Extruded Shim and Solid Shim  
Materials

Property	Minimum Value	Reference
<b>Metamic-HT (fuel basket)</b>		
Conductivity	See Table 1.III.3	[1.III.3]
Emissivity	Note 1	[1.III.3]
Density	See Table 1.III.3	[1.III.3]
Heat Capacity	See Table 1.III.3	[1.III.3]
<b>Aluminum Alloy 2219 (extruded shims)</b>		
Conductivity	69.3 Btu/ft-hr-°F	[4.III.1]
Emissivity	Note 2	[1.III.3]
Density	177.3 lb/ft <sup>3</sup>	[4.III.1]
Heat Capacity	0.207 Btu/lb-°F	[4.III.1]
<b>Aluminum Alloy (solid shims)</b>		
Conductivity	86.6 Btu/ft-hr-°F	Section 1.III.5
Emissivity	Note 1	[1.III.3]
Density	177.3 lb/ft <sup>3</sup>	[4.III.1]
Heat Capacity	0.207 Btu/lb-°F	[4.III.1]
<p>Note 1: Fuel basket and solid shims are hard anodized to yield high emissivities. Lowerbound emissivity defined in Table 1.III.3 is adopted in safety evaluations.</p> <p>Note 2: Extruded shims are hard anodized to yield lowerbound emissivity defined in Note 2 of Table 1.III.3. Extruded shims may also be passivated to allow formation of a thin protective oxide layer that results in emissivity lower than that defined in Table 1.III.3, with a value of 0.1 [4.2.5]. A sensitivity study is performed in Paragraph 4.III.5.6 to demonstrate the acceptability of low emissivity extruded shims.</p>		



Table 4.III.2: Temperature Limits of Fuel Basket and Basket Shim Materials

<b>Metamic-HT (Note 1)</b>	
Normal storage	752°F
Short term operations, Off-Normal and Accident conditions	932°F
<b>Aluminum Alloy Shims (Note 2)</b>	
Normal storage	752°F
Short term operations, Off-normal and Accident conditions	932°F
Notes:	
<ol style="list-style-type: none"> <li>1. The B<sub>4</sub>C component in Metamic-HT is a refractory material that is unaffected by these temperatures and the aluminum component is solid at temperatures in excess of 1000°F.</li> <li>2. To preclude melting the temperature limits are set well below the melting temperature of Aluminum Alloys.</li> </ol>	

Table 4.III.3a: Fuel Loading Pattern Screening Evaluations

<b>Loading Pattern</b> <sup>Note 4</sup>	<b>Total Decay Heat, kW</b>	<b>Peak Cladding Temperature, °F</b>
Case 1: X=0.5 (Note 1)	36.9	598
Case 2: QSHL (Note 2)	42.8	<b>708</b>
<p>Note 1: The decay heat distribution is described in Section 2.1.9.</p> <p>Note 2: Quarter symmetric heat load pattern is defined in Figure 2.III.1</p> <p>Note 3: Since the highest PCT is reached for the QSHL pattern, it is adopted for all the licensing basis evaluations of fuel storage in MPC-68M.</p> <p>Note 4: Cases 1 and 2 are defined in Section 4.III.4.1.</p>		

Table 4.III.3b: Maximum Temperatures Under Normal Long-Term Storage

<b>Component</b>	<b>Temperature (°F)</b>
Fuel Cladding	708
Basket	674
Basket Shims	563
MPC Shell	499
Overpack Inner Shell	358 <sup>1</sup>
Overpack Body Concrete <sup>2</sup>	252
Overpack Lid Concrete <sup>2</sup>	257
Overpack Outer Shell	190
Area Averaged Air Outlet <sup>3</sup>	244

<sup>1</sup> Nominal exceedance of temperature limits has no risk on its structural integrity.

<sup>2</sup> Maximum thru thickness section average temperature reported.

<sup>3</sup> Reported herein for the option of outlet ducts air temperature surveillance set forth in the Technical Specifications.

Table 4.III.4: Maximum Pressures Under Normal Long Term Storage

<b>Condition</b>	<b>Pressure (psig)</b>
Initial maximum backfill* (at 70°F)	46.5
Normal: intact rods	98.7
1% rods rupture**	99.2
Off-Normal (10% rods rupture)	104.0
Accident (100% rods rupture)	152.0
* Conservatively assumed at the Tech. Spec. maximum value (see Table 4.4.12).	
** Per NUREG-1536, pressure analyses with ruptured fuel rods is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.	

Table 4.III.5: Maximum MPC-68M Temperatures Under Vacuum Drying Scenarios

<b>Component</b>	<b>Scenario A (°F)</b>	<b>Scenario B (°F)</b>	<b>Scenario C (°F)</b>
Fuel Cladding	754	732	896
Fuel Basket	729	698	854
Basket Shims	522	482	592
MPC Shell	325	307	343
Notes: (1) The peak cladding temperatures are below the ISG 11 temperature limits of Moderate Burnup Fuel (Scenarios A and C) and High Burnup Fuel (Scenario B). (2) The component temperatures are below the Chapter 2 and Supplement III temperature limits.			

Table 4.III.6: Maximum Steady State HI-TRAC Temperatures and Pressures During On-site Transfer Operations

<b>Component</b>	<b>Temperature [°F]</b>
Fuel Cladding	709 <sup>1</sup>
MPC Basket	676
Basket Periphery	606
MPC Shell	488
Aluminum Shims	555
HI-TRAC Inner Shell	286
HI-TRAC Outer Shell	274
Water Jacket Shell	263
Water Jacket Bulk Water	257
Top Lid Neutron Shield (Holtite) <sup>2</sup>	289
<b>Pressure (psig)</b>	
Initial Maximum Backfill	46.5
Operating Pressure	100.5

- 1 The calculated value is below the permissible limit for high-burnup fuel. Therefore auxiliary cooling of the HI-TRAC is not necessary to ensure cladding safety under onsite transfer operations involving the MPC-68M. Accordingly SCS cooling is not mandated in the MPC-68M Technical Specifications
- 2 Local neutron shield section temperature.

Table 4.III.7: Maximum Temperatures and Pressures Under 100% Air Inlets Blockage Accident

<b>Component</b>	<b>Case 2<sup>Note 2</sup> Temperature@32hrs (°F)</b>	<b>Case 2<sup>Note 2</sup> Temperature@16hrs (°F)</b>	<b>Case 1<sup>Note 2</sup> Temperature@32hrs (°F)</b>
Fuel Cladding	849	789	722
Fuel Basket	818	754	709
Basket Shims	702	645	626
MPC Shell	639	584	571
MPC Lid <sup>Note 1</sup>	599	549	521
Overpack Inner Shell	531	467	462
Body Concrete (Local Temperature)	525	461	456
Lid Concrete (Local Temperature)	447	397	375
<b>Pressure (psig)</b>			
MPC	116.3	109.5	111.6
Note 1: Maximum thru thickness section average temperature reported.			
Note 2: Cases 1 and 2 are defined in Section 4.III.4.1.			

Table 4.III.8: Differential Thermal Expansion

<b>Gap Description</b>	<b>Differential Expansion * mm (in)</b>
Fuel Basket-to-MPC Radial Gap	2.72 (0.107)
Fuel Basket-to-MPC Axial Gap	12.72 (0.501)
MPC-to-Overpack Radial Gap	3.55 (0.140)
MPC-to-Overpack Minimum Axial Gap	14.75 (0.581)
*The differential expansion values reported in this table are bounded by the nominal cold gaps presented on the drawings in Section 1.5.	

Table 4.III.9: MPC-68M Pressure Under HI-TRAC Fire Accident

Initial Operating Pressure	Table 4.III.6
Fire Pressure Rise	2.9 psig
Fire Accident Pressure	103.4 psig

Table 4.III.10: DELETED

Table 4.III.11: HI-STORM Temperatures Under Fuel Debris Storage

<b>Component</b>	<b>Temperature, °F</b>
Fuel Cladding	687 <sup>Note 1</sup>
Basket	656
Aluminum Shims	538
MPC Shell	482
Overpack Inner Shell	342
Overpack Outer Shell	189
Overpack Body Concrete <sup>Note 2</sup>	239
Overpack Lid Concrete <sup>Note 2</sup>	253
Average Air Outlet	244

Note 1: It is recognized that the assumption of all 16 DFC locations having fuel debris instead of permitted 8 cells has the effect of slightly understating the MPC heat load because of the lower per assembly heat permitted in DFC cells. However, because the effect is small (40.4 kW with all 16 cells versus 41.6 kW with permitted 8 cells) and the margins from limits are substantial, this has no adverse effect on the reported temperatures or conclusions. Moreover, the DFC is stored in the basket periphery cells. The effect of a slight change in the heat load in the periphery cells will have a second order effect on the peak cladding temperature which occurs in the inner cell locations.

Note 2: Maximum thru thickness section average temperature reported.

Table 4.III.12: Effect of Anodized Solid Shims and Passivated Extruded Shims on MPC-68M  
Temperatures Under Vacuum Drying Scenario B

<b>Component</b>	<b>Temperature (°F)</b>
Cladding	726
Fuel Basket	694
Basket Shims	420
MPC Shell	315



Table 4.III.13

## HI-TRAC Transfer Cask with MPC-68M: Lowerbound Weights and Thermal Inertias

<b>Component</b>	<b>Weight (lbs)</b>	<b>Heat Capacity (Btu/lb-°F)</b>	<b>Thermal Inertia (Btu/°F)</b>
Lead	52,000	0.031	1,612
Carbon Steel	40,000	0.1	4,000
Alloy-X MPC (empty)	18,200	0.12	2,184
Fuel	47,600	0.056	2,665
Metamic-HT	6,670	0.196	1,307
Basket Shims (Aluminum)	4,500	0.207	931
MPC Cavity Water*	6,170	1.0	6,170
			18,869 (Total)
* Conservative lower bound water mass.			

Table 4.III.14

## Time-to-Boil for Water in the MPC-68M Cavity at QSHL

<b>Initial Temperature (°F)</b>	<b>Time (hrs)</b>
80	17.0
90	15.7
100	14.4
110	13.1
120	11.8
125	11.2

Table 4.III.15: Off-Normal Condition Maximum HI-STORM Temperatures and MPC Cavity Pressures

<b>Component</b>	<b>Off-Normal Ambient Temperature<sup>1</sup> (°F)</b>	<b>Partial Inlet Ducts Blockage (°F)</b>
Fuel Cladding	728	722
Basket	694	687
MPC Shell	519	508
Overpack Inner Shell	378	375
Overpack Body Concrete (Local Temperature)	375 <sup>Note 1</sup>	372
Overpack Lid Concrete (Local Temperature)	328 <sup>Note 1</sup>	327
<b>MPC Cavity Pressure (psig)</b>		
MPC Pressure	101.0	100.1
Note 1: Obtained by adding the off-normal to ambient temperature difference of 20°F to the local maximum concrete temperatures during normal conditions.		

Table 4.III.16: Summary of Burial under Debris Accident Results

<b>Item</b>	<b>Results</b>
Burial Time	30.7 hours
MPC Cavity Pressure	133.3 psig

<sup>1</sup> Obtained by adding the off-normal-to-normal ambient temperature difference of 20°F to normal condition HI-STORM temperatures reported in Table 4.III.3b.

Table 4.III.17: Extreme Environmental Accident Condition Maximum HI-STORM  
Temperatures<sup>1</sup> and MPC Cavity Pressure

<b>Component</b>	<b>Temperature (°F)</b>
Fuel Cladding	753
Basket	719
Basket Shims	608
MPC Shell	544
Overpack Inner Shell	403
Overpack Body Concrete (Local Temperature)	400 <sup>Note 1</sup>
Overpack Lid Concrete (Local Temperature)	353 <sup>Note 1</sup>
Overpack Outer Shell	235
<b>MPC Cavity Pressure (psig)</b>	
MPC Pressure	103.9
Note 1: Obtained by adding the extreme ambient to normal ambient temperature difference of 45°F to the local maximum concrete temperatures during normal conditions.	

<sup>1</sup> Obtained by adding the extreme ambient to normal ambient temperature difference of 45°F to normal condition HI-STORM temperatures reported in Table 4.III.3b.

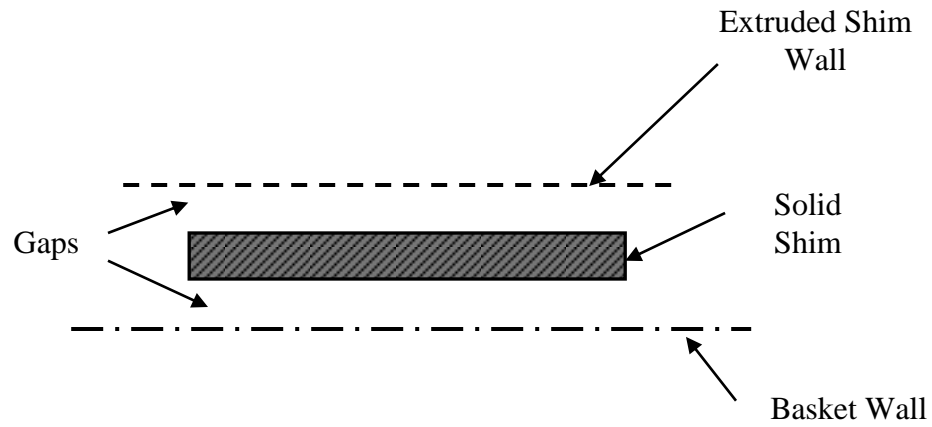


Figure 4.III.1: Schematic of 2-D Thermal Model of Gap between Basket and Extruded Shims with a Solid Shim

## CHAPTER 5†: SHIELDING EVALUATION

### 5.0 INTRODUCTION

The shielding analysis of the HI-STORM 100 System, including the HI-STORM 100 overpack, HI-STORM 100S overpack, HI-STORM 100S Version B overpack<sup>††</sup>, and the 100-ton (including the 100D) and 125-ton (including the 125D) HI-TRAC transfer casks, is presented in this chapter. The HI-STORM 100 System is designed to accommodate different MPCs within HI-STORM overpacks (the HI-STORM 100S overpack is a shorter version of the HI-STORM 100 overpack and the HI-STORM 100S Version B is shorter than both the HI-STORM 100 and 100S overpacks). The MPCs are designated as MPC-24, MPC-24E and MPC-24EF (24 PWR fuel assemblies), MPC-32 and MPC-32F (32 PWR fuel assemblies), and MPC-68, MPC-68F, and MPC-68FF (68 BWR fuel assemblies). The MPC-24E and MPC-24EF are essentially identical to the MPC-24 from a shielding perspective. Therefore only the MPC-24 is analyzed in this chapter. Likewise, the MPC-68, MPC-68F and MPC-68FF are identical from a shielding perspective as are the MPC-32 and MPC-32F and therefore only the MPC-68 and MPC-32 are analyzed. Throughout this chapter, unless stated otherwise, MPC-24 refers to either the MPC-24, MPC-24E, or MPC-24EF and MPC-32 refers to either the MPC-32 or MPC-32F and MPC-68 refers to the MPC-68, MPC-68F, and MPC-68FF.

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Sections 2.1.3 and 2.1.9. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

The MPC-68, MPC-68F, MPC-68FF, and MPC-68M are also capable of storing Dresden Unit 1 antimony-beryllium neutron sources and the single Thoria rod canister which contains 18 thoria rods that were irradiated in two separate fuel assemblies.

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

†† The HI-STORM 100S Version B was implemented in the HI-STORM FSAR (between Revisions 2 and 3) through the 10 CFR 72.48 process. The discussion of the HI-STORM 100S Version B and associated results were added to LAR 1014-2 at the end of the review cycle to support the NRC review of the radiation protection program proposed in the Certificate of Compliance in LAR 1014-2. The NRC did not review and approve any aspect of the design of the HI-STORM 100S Version B since it has been implemented under the provisions of 10 CFR 72.48.

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs) with any number of full-length rods and thimble plug rodlets in the locations without a full-length rod, thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs), neutron source assemblies (NSAs) or similarly named devices. These non-fuel hardware devices are an integral part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 System offers two different loading patterns in the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, and the MPC-68FF. These patterns are uniform and regionalized loading as described in Section 2.0.1 and 2.1.6. Since the different loading patterns have different allowable burnup and cooling times combinations, both loading patterns are discussed in this chapter.

The sections that follow will demonstrate that the design of the HI-STORM 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

#### Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The “controlled area” is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.
2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.
3. Dose rates from the cask must be consistent with a well established “as low as reasonably achievable” (ALARA) program for activities in and around the storage site.
4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR 72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- The HI-STORM 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STORM 100 System is acceptable during normal, off-normal, and accident conditions.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains a discussion on the release of radioactive materials from the HI-STORM 100 System. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM 100 System.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

## 5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM 100 System are:

- Gamma radiation originating from the following sources
  1. Decay of radioactive fission products
  2. Secondary photons from neutron capture in fissile and non-fissile nuclides
  3. Hardware activation products generated during core operations
  
- Neutron radiation originating from the following sources
  1. Spontaneous fission
  2.  $\alpha,n$  reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4.  $\gamma,n$  reactions (this source is negligible)
  5. Dresden Unit 1 antimony-beryllium neutron sources

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the steel structure of the MPC and the steel, lead, and water of the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations. Additionally, in the HI-TRAC 125 and 125D top lid and the transfer lid of the HI-TRAC 125, a solid neutron shielding material, Holtite-A is used to thermalize the neutrons. Boron carbide, dispersed in the solid neutron shield material utilizes the high neutron absorption cross section of  $^{10}\text{B}$  to absorb the thermalized neutrons.

The shielding analyses were performed with MCNP-4A [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.3 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6 and mixed oxide (MOX) fuel assemblies are the GE 6x6. The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes. Section 2.1.9 specifies the acceptable intact zircaloy clad fuel characteristics and the acceptable damaged fuel characteristics.



The design basis stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. Section 2.1.9 specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, and MPC-68FF are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. Section 2.1.9 specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in these MPCs. Section 2.1.9 also specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The burnup and cooling time values in Section 2.1.9, which differ by array class, were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC. Section 5.2 of this chapter describes the choice of the design basis fuel assembly based on a comparison of source terms and also provides a description of how the allowable burnup and cooling times were derived. Since for a given cooling time, different array classes have different allowable burnups in Section 2.1.9, burnup and cooling times that bound array classes 14x14A and 9x9G were used for the analysis in this chapter since these array class burnup and cooling time combinations bound the combinations from the other PWR and BWR array classes. Section 5.2.5 describes how this results in a conservative estimate of the maximum dose rates.

Section 2.1.9 specifies that the maximum assembly average burnup for PWR and BWR fuel is 68,200 and 65,000 MWD/MTU, respectively. The analysis in this chapter conservatively considers burnups up to 75,000 and 70,000 MWD/MTU for PWR and BWR fuel, respectively.

The burnup and cooling time combinations listed below bound all acceptable uniform and regionalized loading burnup levels and cooling times from Section 2.1.9. All combinations were analyzed in the HI-STORM overpack and HI-TRAC transfer casks.

Zircaloy Clad Fuel		
<b>MPC-24</b>	<b>MPC-32</b>	<b>MPC-68</b>
60,000 MWD/MTU 3 year cooling	45,000 MWD/MTU 3 year cooling	50,000 MWD/MTU 3 year cooling
69,000 MWD/MTU 4 year cooling	60,000 MWD/MTU 4 year cooling	62,000 MWD/MTU 4 year cooling
75,000 MWD/MTU 5 year cooling	69,000 MWD/MTU 5 year cooling	65,000 MWD/MTU 5 year cooling
		70,000 MWD/MTU 6 year cooling

Stainless Steel Clad Fuel		
MPC-24	MPC-32	MPC-68
40,000 MWD/MTU 8 year cooling	40,000 MWD/MTU 9 year cooling	22,500 MWD/MTU 10 year cooling

Results are presented in this chapter for the single burnup and cooling time combination for zircaloy clad fuel from the above table which produces the highest dose rate at 1 meter from the midplane of the HI-STORM overpack and HI-TRAC transfer casks. The burnup and cooling time combination may be different for normal and accident conditions and for the different overpacks. As mentioned earlier, there are different versions of the HI-STORM overpack: the HI-STORM 100, the HI-STORM 100S, and the HI-STORM 100S Version B. Section 5.3 describes all three overpacks. However, since the HI-STORM 100S Version B overpack has higher dose rates at the inlet vents and slightly higher offsite dose rates than the other overpacks, results are only presented for the HI-STORM 100S Version B overpack.

The 100-ton HI-TRAC with the MPC-24 has higher normal condition dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The MPC-32 results, MPC-68 results, and additional MPC-24 results are provided in Section 5.4 for comparison. The 100-ton HI-TRAC with the MPC-24 also has higher accident condition dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results for accident condition are presented in the section. Accident condition results for the MPC-32 and MPC-68 in the 100-ton HI-TRAC are not provided in this chapter. The HI-TRAC 100D is a variation on the 100-ton HI-TRAC with fewer radial ribs and a slightly different lower water jacket. Section 5.4 presents results for the HI-TRAC 100D with the MPC-32.

The HI-TRAC 100 and 100D dose rates bound the HI-TRAC 125 and 125D dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the HI-TRAC 125 and 125D. Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. Therefore, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.

As a general statement, the dose rates for uniform loading presented in this chapter bound the dose rates for regionalized loading therefore, dose rates for specific burnup and cooling time combinations in a regionalized loading pattern are not presented in this chapter. For regionalized loading where higher burned or shorter cooled assemblies are placed in the center of the cask, the dose rates would be substantially lower than the bounding dose rates presented here. For regionalized loading where the higher burned or shorter cooled assemblies are placed on the

periphery, the dose rates could be closer to the bounding dose rates presented here. Section 5.4.9 provides an additional brief discussion on regionalized loading.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

#### 5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.
2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM 100 System in Section 2.3.5.2 as: 300 mrem/hour on the radial surface of the overpack, 175 mrem/hour at the openings of the air vents, and 60 mrem/hour on the top of the overpack.

The HI-STORM overpack dose rates presented in this section are conservatively evaluated for the MPC-32, the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in Section 2.1.9.

Figure 5.1.13 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM 100S Version B overpack. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. The dose values reported at the locations shown on Figure 5.1.13 are averaged over a region that is approximately 1 foot in width.

The total dose rates presented in this chapter for the MPC-24 and MPC-32 are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Section 5.2.4.1. This is conservative because it is not expected that the cooling times for both the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values.

Tables 5.1.11, 5.1.12, and 5.1.13 provide the maximum dose rates adjacent to the HI-STORM 100S Version B overpack during normal conditions for the MPC-32, MPC-24, and MPC-68. Tables 5.1.14 through 5.1.16 provide the maximum dose rates at one meter from the HI-STORM 100S Version B overpack.

The HI-STORM 100S Version B overpack was analyzed for the dose rate at the controlled area boundary. Although the dose rates for the MPC-32 in HI-STORM 100S Version B are greater than those for the MPC-24 in HI-STORM 100S Version B at the ventilation ducts, as shown in Tables 5.1.11 and 5.1.12, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary for the HI-STORM 100S Version B overpack. This is acceptable because the vents are a small fraction of the radial surface area and the MPC-24 has higher dose rates at the radial midplane than the MPC-32 in the HI-STORM 100S Version B overpack. The MPC-24 was also chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the MPC-32 or the MPC-68 (see Section 2.1.9). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32. The controlled area boundary dose rates were also calculated including the BPRAs non-fuel hardware source. The controlled area boundary dose rates with the BPRAs bound the controlled area boundary dose rates with the TPDs. In the site specific dose analysis, users should perform an analysis which properly bounds the fuel to be stored including BPRAs if present. For guidance on fuel parameters for site-specific analyses see Subsection 5.2.5.4.

Table 5.1.7 provides dose rates adjacent to and one meter from the 100-ton HI-TRAC. Table 5.1.8 provides dose rates adjacent to and one meter from the 125-ton HI-TRACs. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125 and 100 transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point #5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed

burnup and cooling times specified in Section 2.1.9, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen to bound the allowable burnup and cooling times in Section 2.1.9. Results for other burnup and cooling times and for the MPC-68 and MPC-32 are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRACs or the HI-STORM overpack, it is important to understand the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. The figures discussed below were all calculated without the gamma source from BPRAs and were calculated for an earlier design of the HI-TRAC which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below.

Figure 5.1.5 shows the dose rate profile at 1 foot from the side of the 100-ton HI-TRAC transfer cask with the MPC-24 for 35,000 MWD/MTU and 5 year cooling. This figure clearly shows the behavior of the total dose rate and each of the dose components as a function of the cask height. To capture the effect of scattering off the concrete floor, the calculational model simulates the 100-ton HI-TRAC at a height of 6 inches (the typical cask carry height) above the concrete floor. As expected, the total dose rate on the side near the top and bottom is dominated by the Co-60 gamma dose component, while the center dose rate is dominated by the fuel gamma dose component.

The total dose rate and individual dose rate components on the surface of the pool lid on the HI-TRAC-100 when uniformly loaded are provided in Figure 5.1.6, illustrating the significant reduction in dose rate with increasing distance from the center of the pool lid. Specifically, the total dose rate is shown to drop by a factor of more than 20 from the center of the pool lid to the outer edge of the HI-TRAC. Therefore, even though the dose rate in Table 5.1.7 at the center of the pool lid is substantial, the dose rate contribution, from the pool lid, to the personnel exposure is minimal.

The behavior of the dose rate 1-foot from the transfer lid is shown in Figure 5.1.7. Similarly, the total dose rate and the individual dose rate components 1-foot from the top lid, as a function of distance from the axis of the 100-ton HI-TRAC, are shown in Figure 5.1.8. For both lids (transfer and top), the reduction in dose rate with increased distance from the cask axial centerline is substantial.

Note that regionalized loading can have a significant effect on the variation of the dose rate on the top and bottom as a function of the distance from the cask centerline. For a regionalized loading with higher burnup fuel in the center region, the dose rate profiles would be even more pronounced, i.e. the difference between the dose rate in the center and the dose rate near the edge

of the cask would be larger than shown in Figures 5.1.7 and 5.1.8. However, if a regionalized loading plan is selected where the higher burnup fuel is located in the outer region, then the difference would be less. In extreme cases, it would even be possible that the dose rate near the edge of the MPC is higher than the one at the center of the cask. This should be considered during loading operations in order to minimize occupational doses.

To reduce the dose rate above the water jacket, a localized temporary shield ring, described in Chapter 8, may be employed on the 125-ton HI-TRACs and on the 100-ton HI-TRAC. This temporary shielding, which is water, essentially extends the water jacket to the top of the HI-TRAC. The effect of the temporary shielding on the side dose rate above the water jacket (in the area around the lifting trunnions and the upper flange) is shown on Figure 5.1.9, which shows the dose profile on the side of the 100-ton HI-TRAC with the temporary shielding installed. For comparison, the total dose rate without temporary shielding installed is also shown on Figure 5.1.9. The results indicate that the temporary shielding reduces the dose rate by approximately a factor of 2 in the area above the water jacket.

To illustrate the reduction in dose rate with distance from the side of the 100-ton HI-TRAC, Figure 5.1.10 shows the total dose rate on the surface and at distances of 1-foot and 1-meter.

Figure 5.1.11 plots the total dose rate at various distances from the bottom of the transfer lid, including distances of 1, 5, 10, and 15 feet. Near the transfer lid, the total dose rate is shown to decrease significantly as a function of distance from the 100-ton HI-TRAC axial centerline. Near the axis of the HI-TRAC, the reduction in dose rate from the 1-foot distance to the 15-foot distance is approximately a factor of 15. The dose rate beyond the radial edge of the HI-TRAC is also shown to be relatively low at all distances from the HI-TRAC transfer lid. Thus, prudent transfer operating procedures will employ the use of distance to reduce personnel exposure. In addition, when the HI-TRAC is in the horizontal position and is being transported on site, a missile shield may be positioned in front of the HI-TRAC transfer lid or pool lid. If present, this shield would also serve as temporary gamma shielding which would greatly reduce the dose rate in the vicinity of the transfer lid or pool lid. For example, if the missile shield was a 2 inch thick steel plate, the gamma dose rate would be reduced by approximately 90%.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single HI-STORM 100S Version B cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. These values were conservatively calculated for a burnup of 60,000 MWD/MTU and a 3-year cooling time. In addition, the annual dose was calculated for a burnup of 45,000 MWD/MTU with a corresponding cooling time of 9 years. BPRAs were included in these dose estimates. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in

accordance with 10CFR72.212. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time). For guidance on fuel parameters for site-specific analyses see Subsection 5.2.5.4.

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM 100 cask array configurations provided in Table 5.1.9. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis fuels. Since the source strengths of the GE 6x6 intact and damaged fuel and the GE 6x6 MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the dose rates from the GE 6x6 fuels for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either GE 6x6 intact or damaged or GE 6x6 MOX fuel for normal conditions is required to demonstrate that the MPC-68 with GE 6x6 fuels will meet the normal condition regulatory requirements. Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E, MPC-32 and the MPC-68.

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM 100 System. Section 5.4.6 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 lists the gamma and neutron sources for the design basis stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.4.

The analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.104 limits and ALARA practices.

### 5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose

equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 Rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that may affect the shielding materials of the HI-STORM overpack. It is the 100% blockage of air inlets 30 day accident condition. To analyze this accident, it is conservatively assumed that all hydrogen and portion of oxygen are lost in a bounding volume of the neutron shield layer (concrete) that reaches a temperature of at least 350°F. The 100% blockage of air inlets 30 day accident affects the dose at 100 m from overpack. To illustrate the impact of this design basis accident, the dose rate at 91.44 m is provided in Table 5.1.17. The burnup and cooling time combination used in Table 5.1.17 correlates to the HI-TRAC accident condition discussed in this section and in Table 5.1.10. The heat load for this burnup and cooling time combination is significantly larger than the heat load threshold for the 30-day surveillance of the casks discussed in Section 4.6. Shielding analysis demonstrates that the HI-STORM 100% blockage of air inlets 30 day accident condition is bounded by the HI-TRAC loss of water in water jacket accident condition and therefore is in compliance with the 10CFR72.106 limits.

Other design basis accidents which may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.



The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10 for the 100-ton and 125-ton HI-TRACs at a distance of 1 meter and for the 100-ton HI-TRAC at a distance of 100 meters. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Based on the dose rate at 100 meters in Table 5.1.10, it would take 1608 hours (~67 days) for the dose at the controlled area boundary to reach 5 Rem. Assuming an accident duration of 30 days, the accumulated dose at the controlled area boundary would be 2.2Rem. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

The consequences of the design basis accident conditions for the MPC-68 and MPC-24E storing damaged fuel and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. Analyses in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition does not significantly affect the dose rates around the cask. Therefore, the damaged fuel post-accident dose rates are bounded by the intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.106 limits.

Table 5.1.1

[INTENTIONALLY DELETED]

Table 5.1.2

[INTENTIONALLY DELETED]

Table 5.1.3

[INTENTIONALLY DELETED]

Table 5.1.4

[INTENTIONALLY DELETED]

Table 5.1.5

[INTENTIONALLY DELETED]

Table 5.1.6

[INTENTIONALLY DELETED]

Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
60,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	124.94	33.16	958.00	469.38	1585.48	1594.02
2	3196.67 <sup>†</sup>	134.95	0.96	249.02	3581.60	3828.84
3	36.47	6.50	528.22	392.74	963.94	1112.46
3 (temp)	16.31	11.57	244.83	6.31	279.02	347.16
4	80.38	2.57	425.12	483.50	991.57	1116.07
4 (outer)	23.94	1.63	105.85	326.35	457.77	489.10
5 (pool lid)	623.98	47.34	4826.78	3152.66	8650.76	8715.53
5 (transfer)	1243.30	2.59	7192.64	1805.36	10243.89	10340.73
5(t-outer)	318.58	0.89	696.19	713.29	1728.95	1750.42
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	422.99	17.82	142.41	76.30	659.52	692.00
2	1400.26 <sup>†</sup>	41.25	11.27	93.37	1546.14	1655.62
3	177.50	9.93	118.30	36.64	342.37	391.80
3 (temp)	176.53	10.66	100.76	13.85	301.80	346.37
4	27.82	0.45	131.25	120.44	279.96	318.53
5 (transfer)	552.69	0.48	2938.22	503.83	3995.21	4034.34
5(t-outer)	76.44	1.54	264.85	144.64	487.47	491.37

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

<sup>†</sup> The cobalt activation of incore grid spacers accounts for approximately 6% of the surface and one-meter dose rates.



Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRACS FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
75,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 125-TON HI-TRACS</b>						
1	6.32	61.85	100.63	415.90	584.70	585.42
2	113.33 <sup>†</sup>	183.20	0.01	287.94	584.49	600.36
3	1.41	6.55	62.26	663.65	733.88	753.59
4	41.57	8.40	340.67	767.94	1158.58	1274.01
4 (outer)	4.84	6.00	42.31	16.11	69.26	83.45
5 (pool)	54.77	3.67	454.56	2883.53	3396.53	3404.24
5 (transfer)	65.81	4.78	601.40	440.29	1112.28	1117.76
<b>ONE METER FROM THE 125-TON HI-TRACS</b>						
1	14.93	24.68	12.90	68.44	120.95	122.99
2	50.47 <sup>†</sup>	59.39	0.52	98.23	208.61	215.68
3	5.66	13.95	12.58	61.07	93.26	98.17
4	11.54	2.03	82.02	79.09	174.68	202.33
5 (transfer)	25.98	0.92	290.76	76.26	393.92	396.85

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

<sup>†</sup> The cobalt activation of incore grid spacers accounts for 9.4% of the surface and one-meter dose rates.

Table 5.1.9

**DOSE RATES FOR ARRAYS OF MPC-24  
WITH DESIGN BASIS ZIRCALOY CLAD FUEL  
AT VARYING BURNUP AND COOLING TIMES**

Array Configuration	1 cask	2x2	2x3	2x4	2x5
<b>HI-STORM 100S Version B Overpack</b>					
<b>60,000 MWD/MTU AND 3-YEAR COOLING</b>					
Annual Dose (mrem/year) <sup>†</sup>	19.26	16.41	24.62	20.36	16.34
Distance to Controlled Area Boundary (meters) <sup>††,†††</sup>	350	450	450	500	550
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
Annual Dose (mrem/year) <sup>†</sup>	23.56	14.30	21.46	15.89	19.86
Distance to Controlled Area Boundary (meters) <sup>††</sup>	200	300	300	350	350

<sup>†</sup> 8760 hr. annual occupancy is assumed.

<sup>††</sup> Dose location is at the center of the long side of the array.

<sup>†††</sup> Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 3-year cooling, as specified in the Section 2.1.9, is lower than the burnup used for this analysis.

Table 5.1.10

DOSE RATES FROM HI-TRAC  
FOR ACCIDENT CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
AT BOUNDING BURNUP AND COOLING TIMES

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ONE METER FROM HI-TRAC</b>					
<b>125-TON HI-TRACs</b>					
<b>75,000 MWD/MTU AND 5-YEAR COOLING</b>					
2 (Accident Condition)	92.26	1.02	3476.98	3570.26	3583.16
2 (Normal Condition)	109.86	0.52	98.23	208.61	215.68
<b>100-TON HI-TRAC</b>					
<b>75,000 MWD/MTU AND 5-YEAR COOLING</b>					
2 (Accident Condition)	1354.67	17.88	4359.16	5731.72	5927.95
2 (Normal Condition)	829.09	9.90	168.82	1007.81	1117.29
<b>100 METERS FROM HI-TRAC</b>					
<b>100-TON HI-TRAC</b>					
<b>75,000 MWD/MTU AND 5-YEAR COOLING</b>					
2 (Accident Condition)	0.68	0.10	2.22	3.00	3.11

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.11

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	41.37	70.98	14.80	127.15	130.10
2	239.51	0.32	4.24	244.08	261.07
3	11.22	17.82	5.51	34.54	40.95
4	12.02	4.29	4.11	20.43	22.78

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.12

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
60,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	34.25	57.09	29.86	121.20	122.67
2	252.16	0.10	7.16	259.41	273.60
3	13.56	15.57	9.82	38.94	43.90
4	13.42	4.65	7.22	25.29	27.30

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.13

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
50,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	20.15	56.22	18.80	95.17
2	211.31	0.12	6.38	217.81
3	4.39	18.15	3.73	26.27
4	7.76	5.05	3.40	16.20

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.14

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	32.51	18.19	2.41	53.10	55.78
2	124.98	1.42	1.75	128.15	136.88
3	14.74	8.67	0.75	24.16	28.13
4	2.79	1.31	1.16	5.26	5.89

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.15

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
60,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	33.12	16.07	4.80	54.00	55.76
2	129.84	1.15	2.84	133.83	141.21
3	15.89	7.23	1.30	24.42	27.44
4	3.22	1.41	2.36	6.99	7.54

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.



Table 5.1.16

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK  
FOR NORMAL CONDITIONS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
50,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	25.76	15.91	3.21	44.88
2	107.43	0.95	2.46	110.84
3	7.78	8.96	0.73	17.47
4	1.78	1.65	0.84	4.27

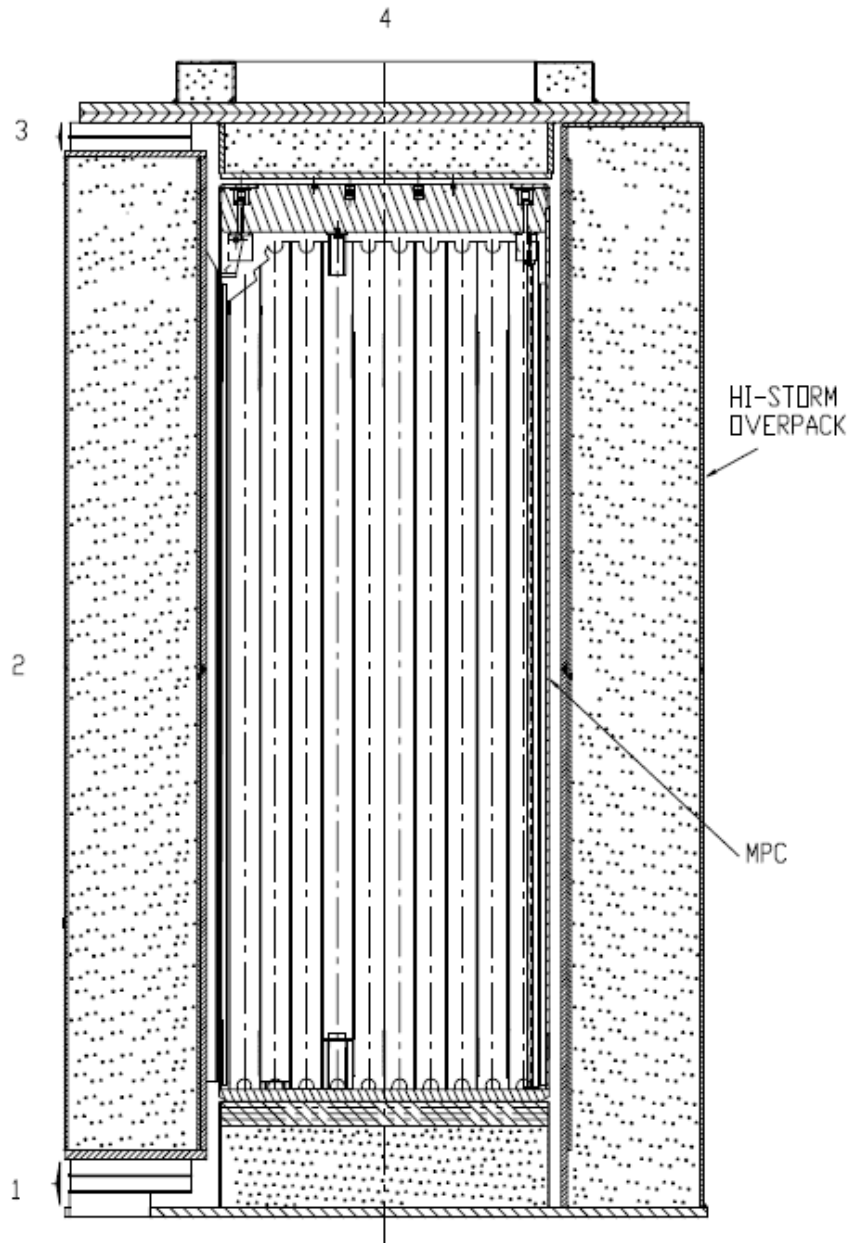
† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

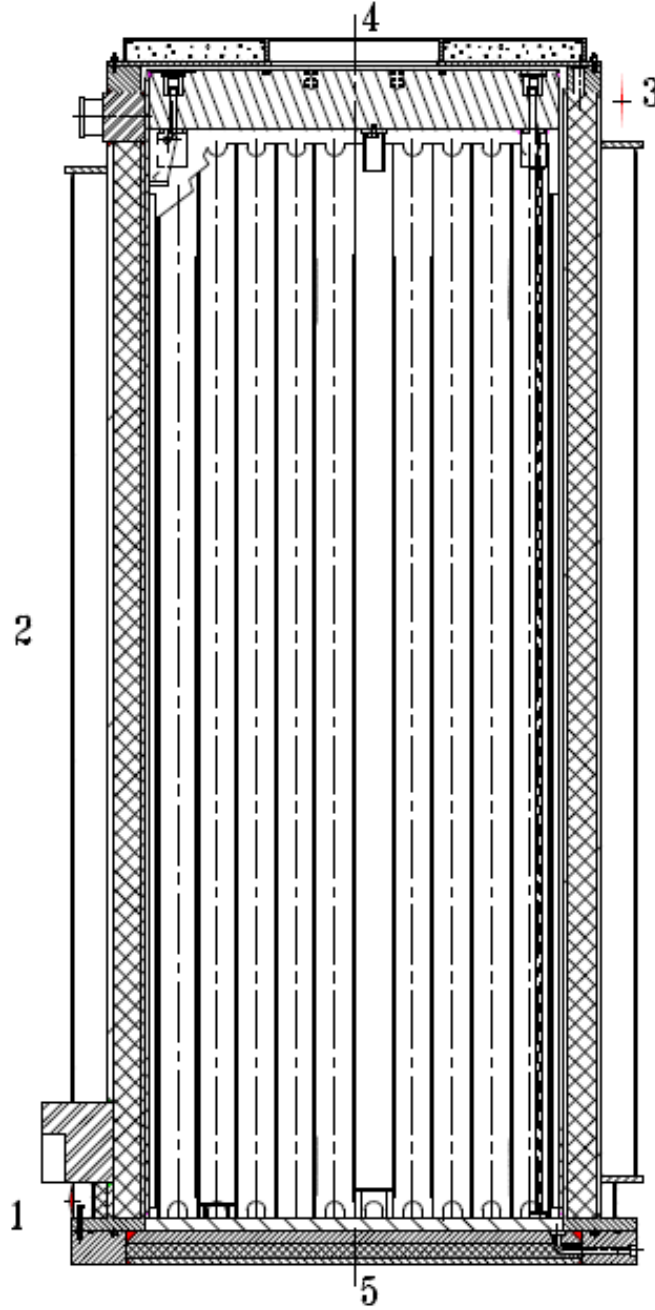
Table 5.1.17

DOSE RATES FROM HI-STORM FOR ACCIDENT CONDITIONS MPC-24 DESIGN BASIS  
ZIRCALOY CLAD FUEL WITH BPRAs AT BOUNDING BURNUP AND COOLING TIME

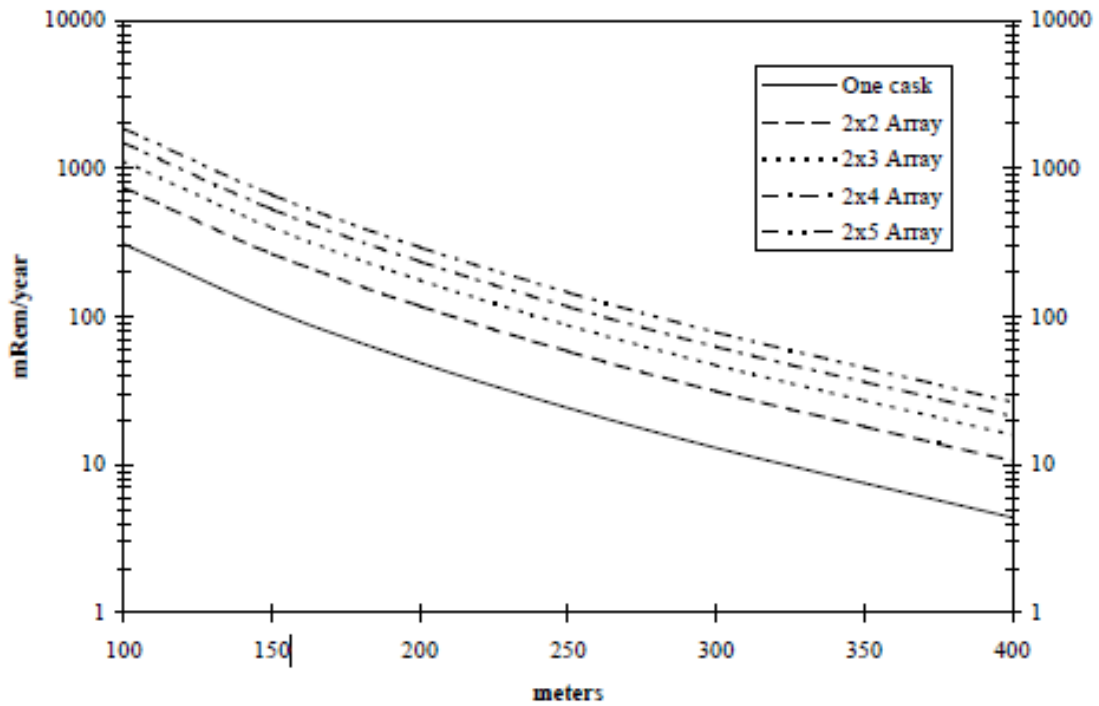
<b>91.44 METERS FROM HI-STORM</b>		
<b>75,000 MWD/MTU, 5.0% ENRICHMENT, AND 5-YEAR COOLING</b>		
<b>Top Dose Rate (mrem/hr)</b>	<b>Side Dose Rate (mrem/hr)</b>	<b>Total Dose Rate (mrem/hr)</b>
0.0020	0.0872	0.0891



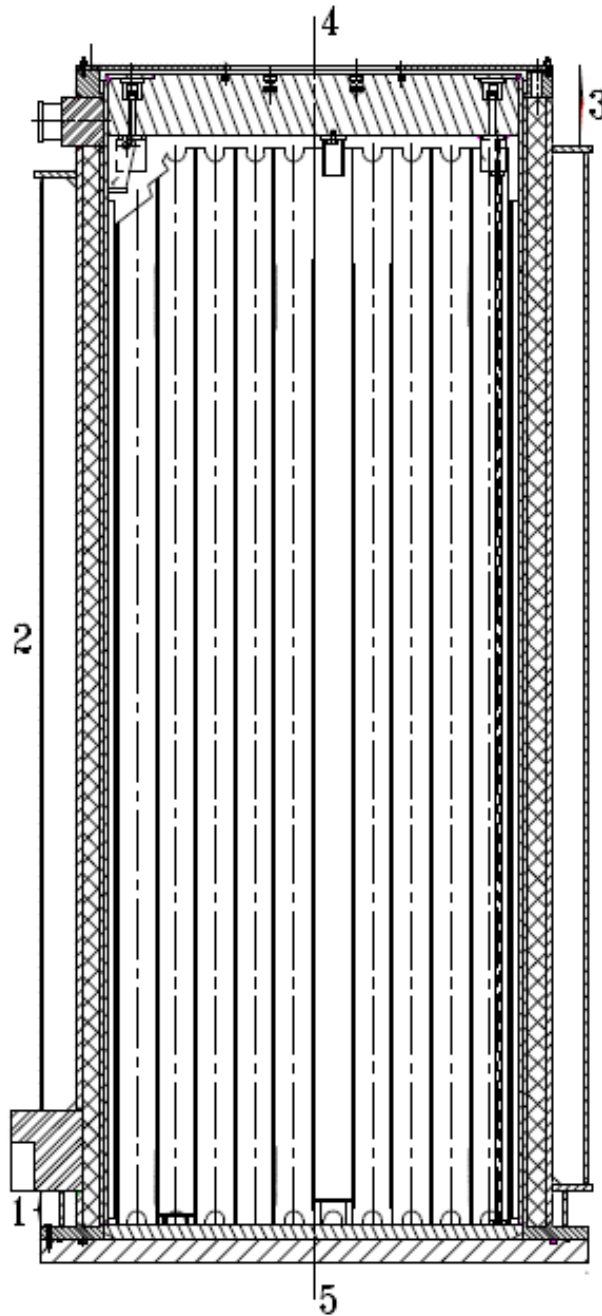
**FIGURE 5.1.1: CROSS SECTION ELEVATION VIEW OF HI-STORM 100 OVERPACK WITH DOSE POINT LOCATION**



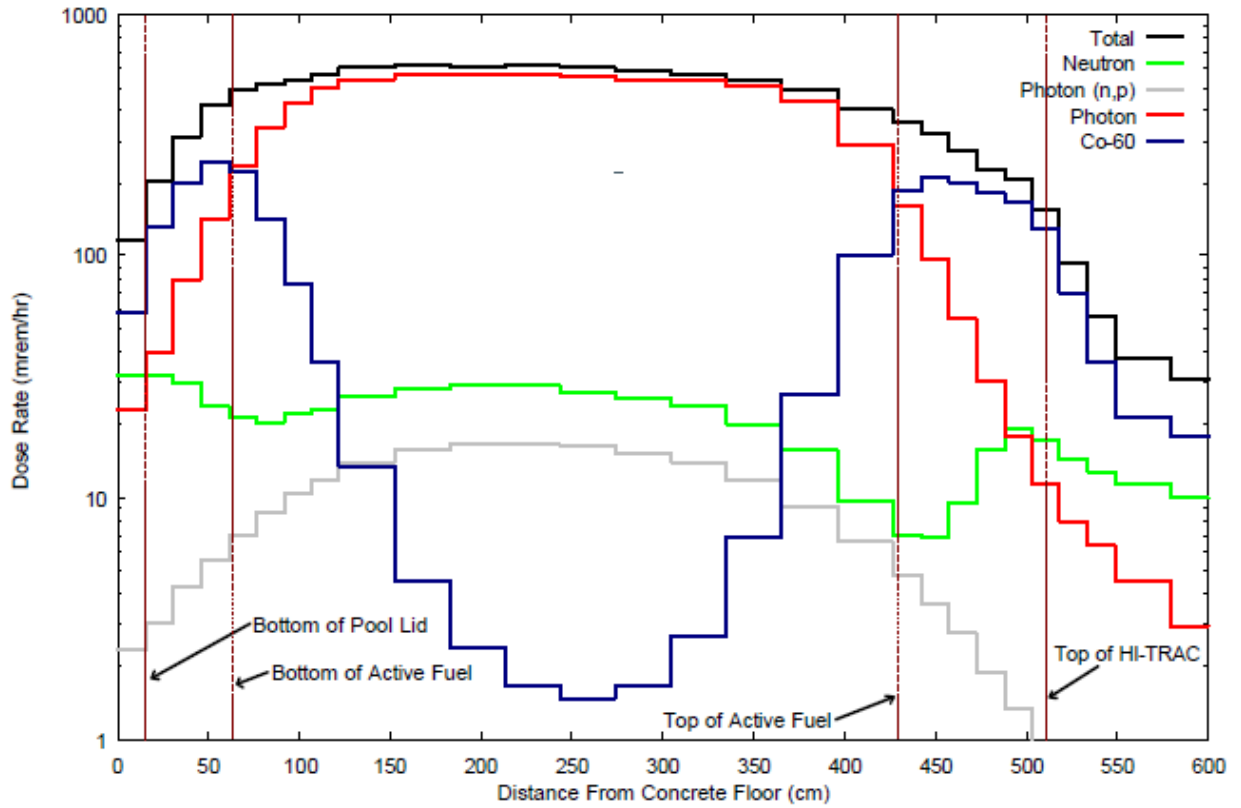
**FIGURE 5.1.2: CROSS SECTION ELEVATION VIEW OF 125 TON HI-TRAC TRANSFER CASK WITH DOSE POINT LOCATIONS**



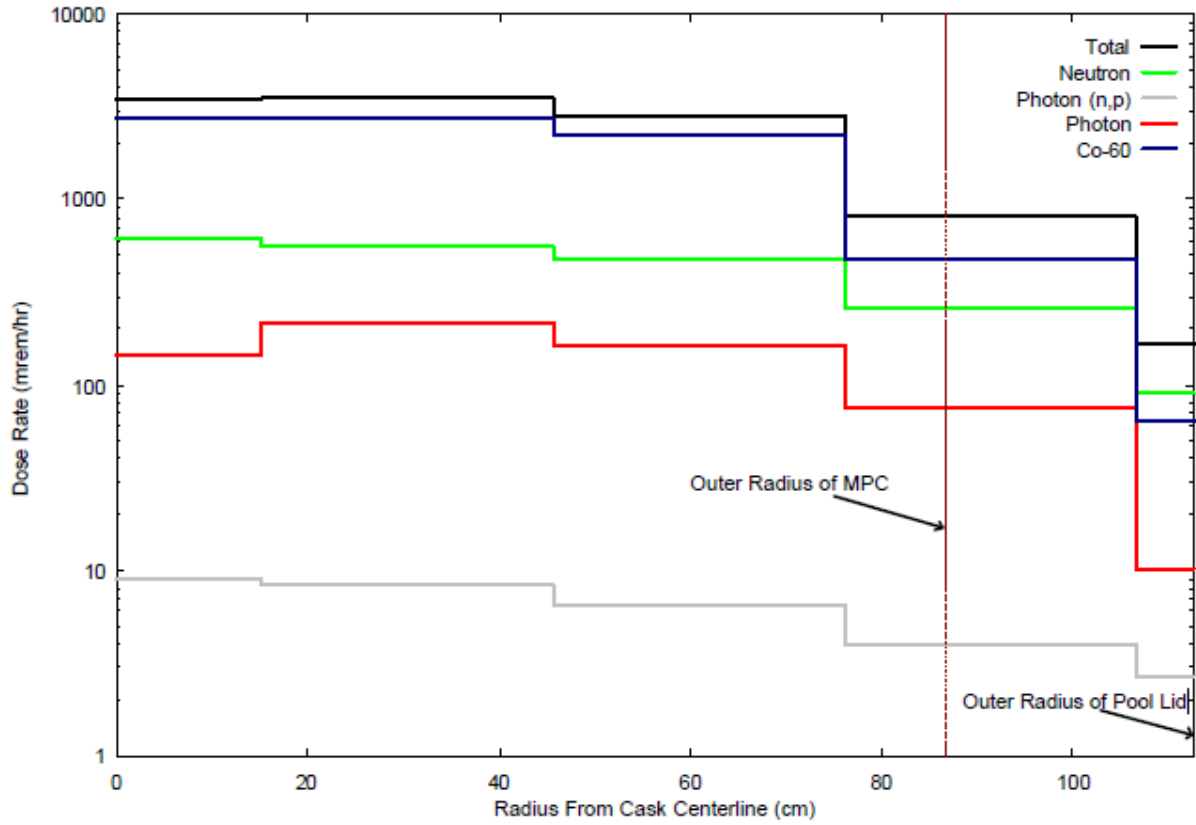
**FIGURE 5.1.3: ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 47,500 MWD/MTU AND 3-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)**



**FIGURE 5.1.4: CROSS SECTION ELEVATION VIEW OF 100 TON HI-TRAC TRANSFER CASK (WITH POOL LID) WITH DOSE POINT LOCATIONS**

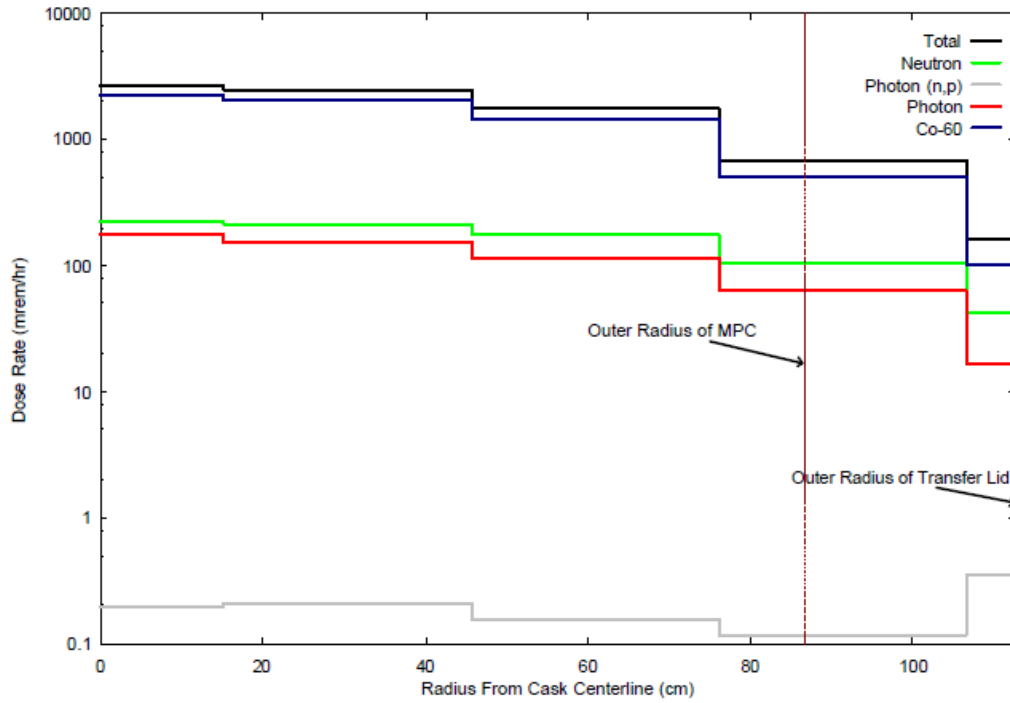


**FIGURE 5.1.5: DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**

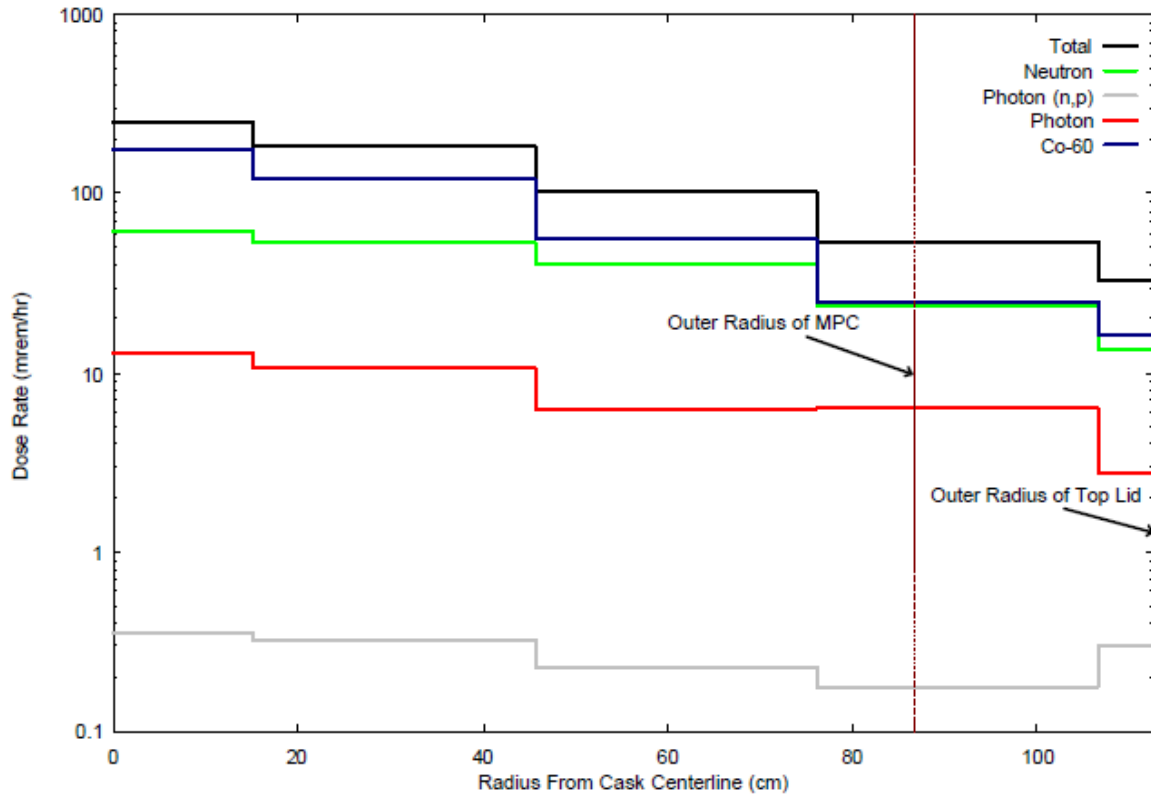


**FIGURE 5.1.6: DOSE RATE ON THE SURFACE OF THE POOL LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**

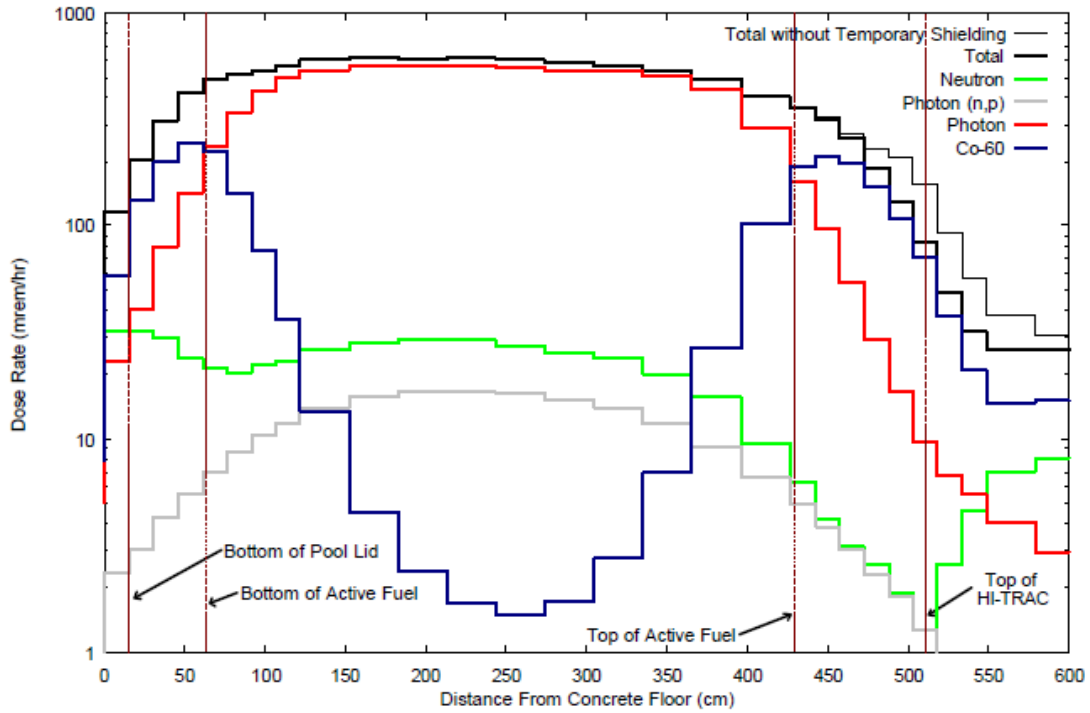




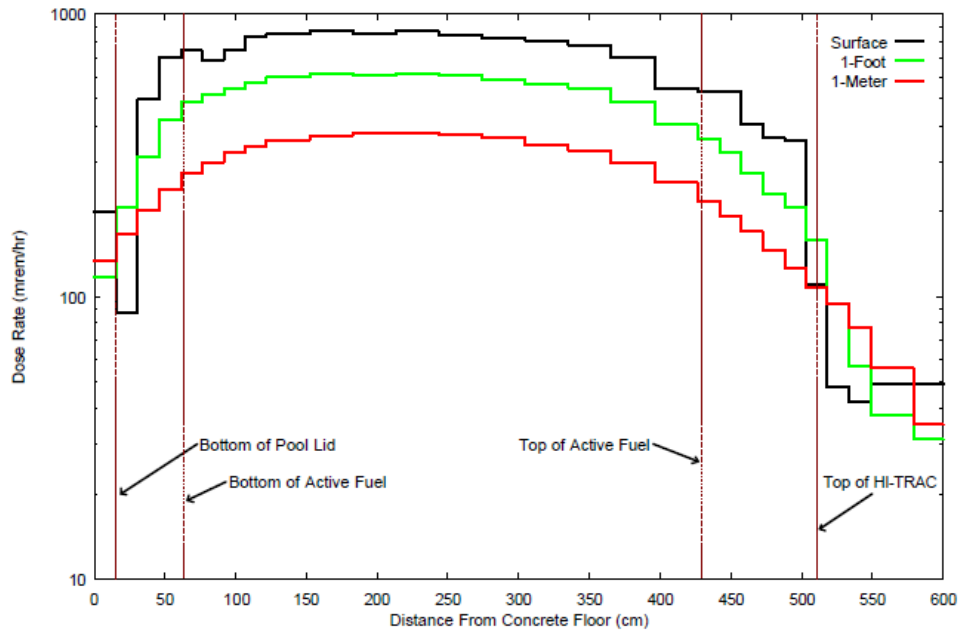
**FIGURE 5.1.7: DOSE RATE 1-FOOT FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**



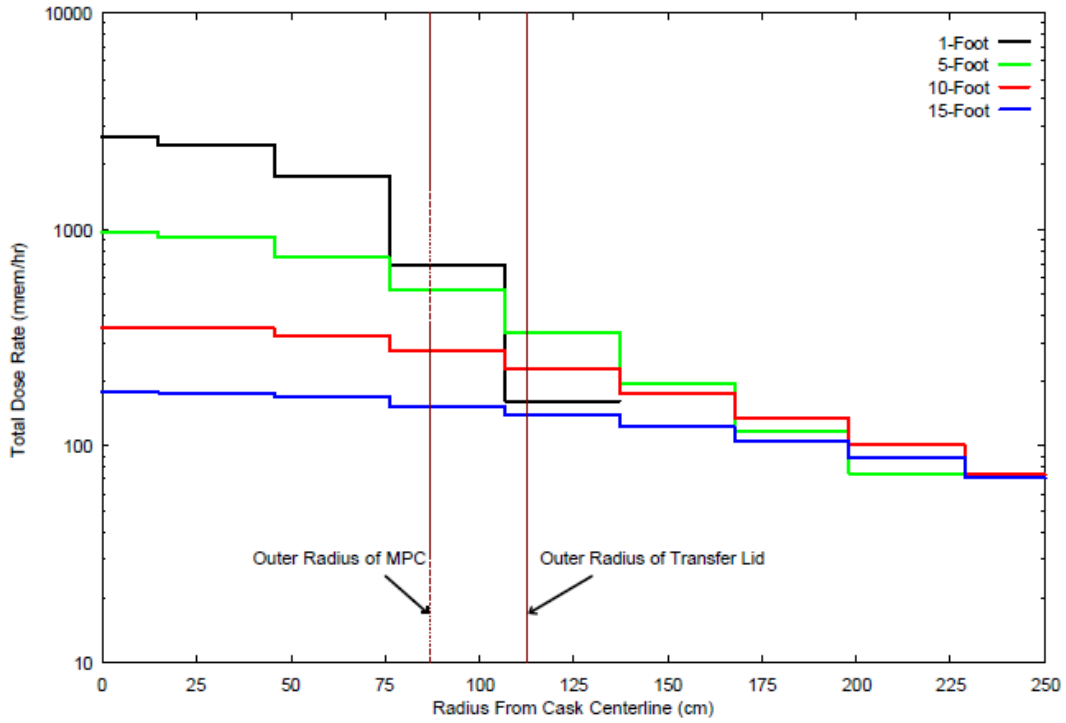
**FIGURE 5.1.8: DOSE RATE 1-FOOT FROM THE TOP OF TOP LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**



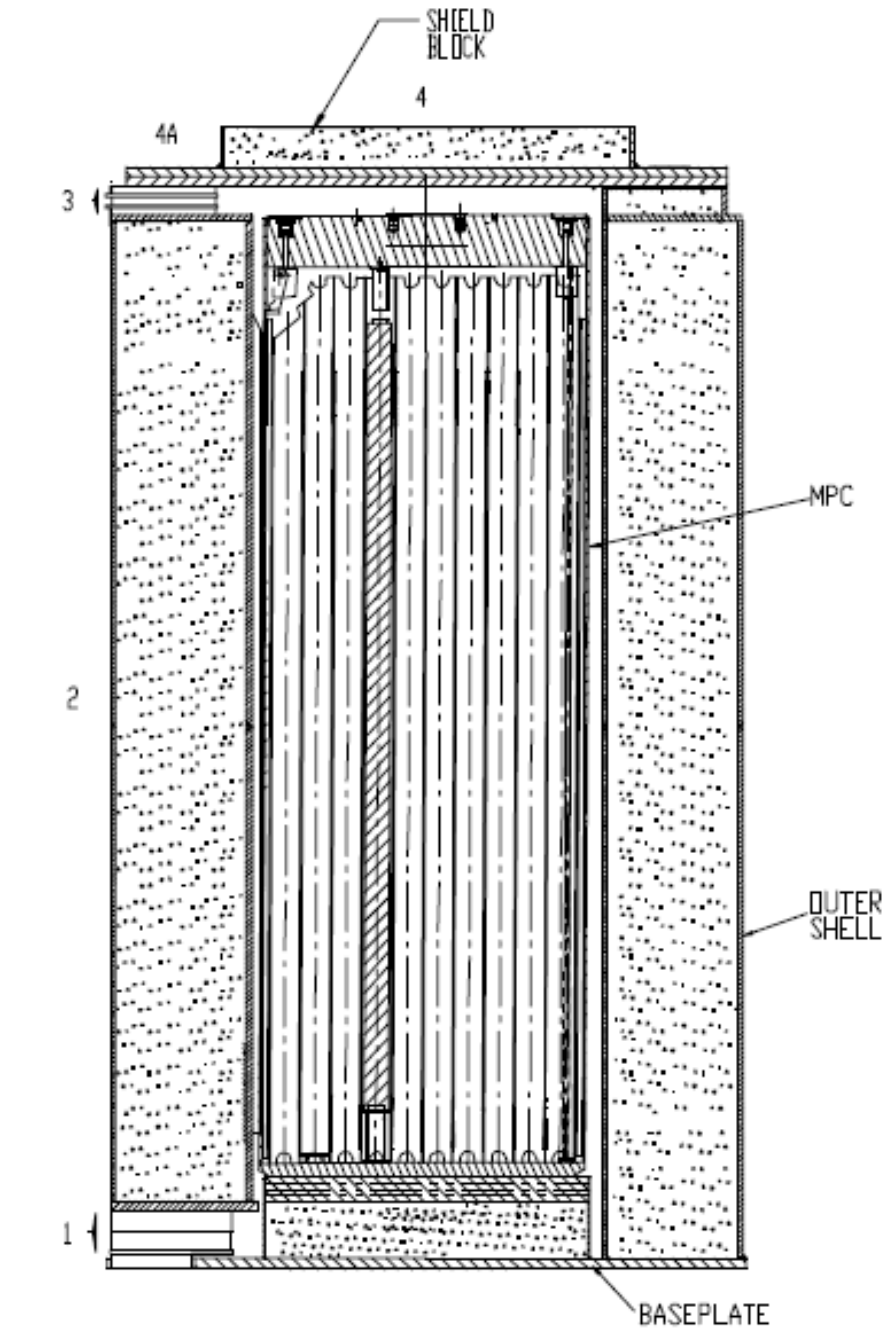
**FIGURE 5.1.9: DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH TEMPORARY SHIELDING INSTALLED, WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING (TOTAL DOSE WITHOUT TEMPORARY SHIELDING SHOWN FOR COMPARISON)**



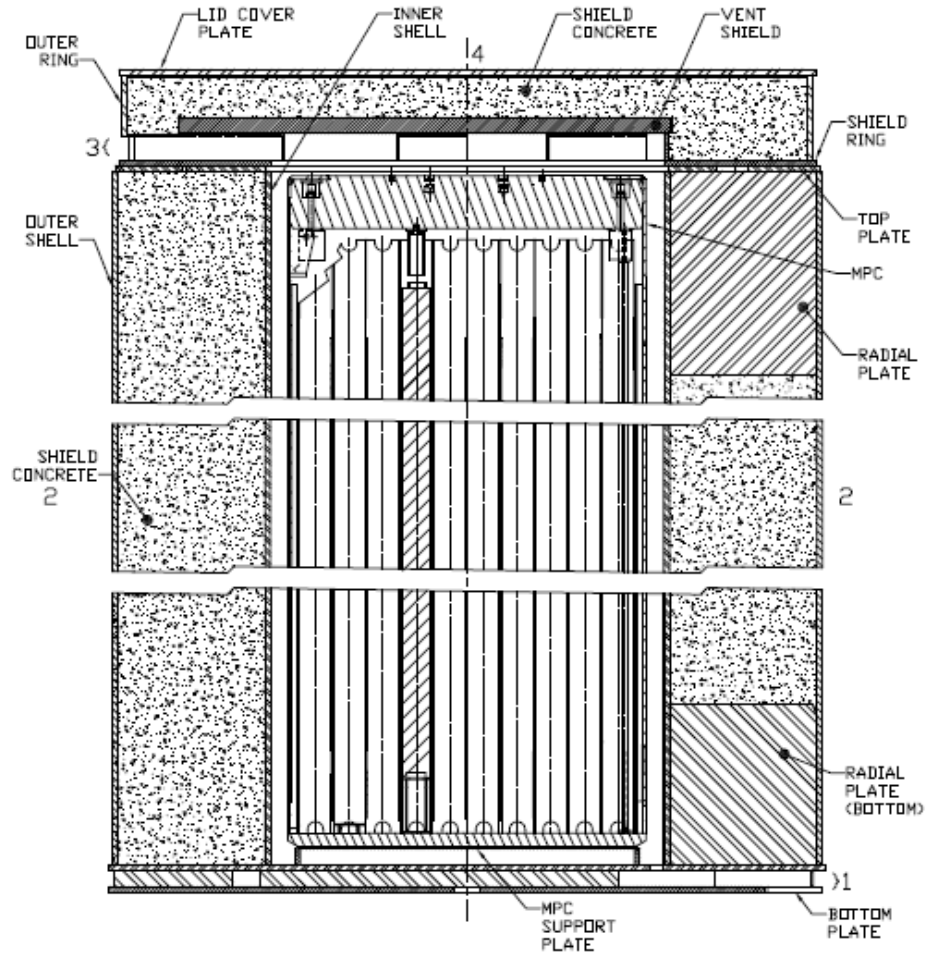
**FIGURE 5.1.10: DOSE RATE AT VARIOUS DISTANCES FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**



**FIGURE 5.1.11: DOSE RATE AT VARIOUS DISTANCES FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING**



**FIGURE 5.1.12: CROSS SECTION ELEVATION VIEW OF THE HI-STORM 100S OVERPACK WITH DOSE POINT LOCATION**



**FIGURE 5.1.13: HI-STORM 100S VERSION B OVERPACK CROSS SECTIONAL ELEVATION VIEW WITH DOSE POINT LOCATION**

## 5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

Sample input files for SAS2H and ORIGEN-S are provided in Appendices 5.A and 5.B, respectively. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from  $^{60}\text{Co}$  activity of the steel structural material in the fuel element above and below the active fuel region. The third source is from (n, $\gamma$ ) reactions described below.

A description of the design basis zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. The PWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load for a given burnup and cooling time from the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun. The BWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load for a given burnup and cooling time from the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest  $\text{UO}_2$  mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly is described in Table 5.2.2. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.21 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6 and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger  $^{60}\text{Co}$  activity.

The design basis stainless steel clad fuel assembly for the Indian Point 1, Haddam Neck, and San Onofre 1 assembly classes is described in Table 5.2.3. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.



The design basis assemblies mentioned above are the design basis assemblies for both intact and damaged fuel and fuel debris for their respective array classes. Analyses of damaged fuel are presented in Section 5.4.2.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

### 5.2.1 Gamma Source

Tables 5.2.4 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuels at varying burnups and cooling times. Tables 5.2.7 and 5.2.22 provides the gamma source in MeV/s and photons/s for the design basis 6x6 and MOX fuel, respectively.

Specific analysis for the HI-STORM 100 System, which includes the HI-STORM storage overpacks and the HI-TRAC transfer casks, was performed to determine the dose contribution from gammas as a function of energy. This analysis considered dose locations external to the 100-ton HI-TRAC transfer cask and the HI-STORM 100 overpack and vents. The results of this analysis have revealed that, due to the magnitude of the gamma source at lower energies, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose at all high dose locations). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Dose rate contributions from above and below this range were evaluated and found to be negligible. Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of  $^{59}\text{Co}$  to  $^{60}\text{Co}$ . The primary source of  $^{59}\text{Co}$  in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant  $^{59}\text{Co}$  impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Conservatively, the impurity level of  $^{59}\text{Co}$  was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

Holtec International has gathered information from utilities and vendors which shows that the 1.0 gm/kg impurity level is very conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. The typical Cobalt-59 impurity level for fuel since the late 1980s is less than 0.5 gm/kg. Based on this, fuel with a short cooling time, 5 to 9 years, would have a Cobalt-59 impurity level less than 0.5 gm/kg. Therefore, the use of a bounding Cobalt-59 impurity level of 1.0 gm/kg is very conservative, particularly for recently manufactured assemblies. Analysis in Reference [5.2.3] indicates that the cobalt impurity in steel and inconel for fuel manufactured in the 1970s ranged from approximately 0.2 gm/kg to 2.2 gm/kg. However, older fuel manufactured with higher cobalt impurity levels will also have a corresponding longer cooling time and therefore will be bounded by the analysis presented in this chapter. As confirmation of this statement, Appendix D presents a comparison of the dose rates around the 100-ton HI-TRAC and the HI-STORM with the MPC-24 for a short cooling time (5 years) using the 1.0 gm/kg mentioned above and for a long cooling time (9 years) using a higher cobalt impurity level of 4.7 gm/kg for inconel. These results confirm that the dose rates for the longer cooling time with the higher impurity level are essentially equivalent to (within 11%) or bounded by the dose rates for the shorter cooling time with the lower impurity level. Therefore, the analysis in this chapter is conservative.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM 100 system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 1 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. The masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative <sup>59</sup>Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a <sup>59</sup>Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a <sup>60</sup>Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the  $^{60}\text{Co}$  is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.
2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.10. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.11 through 5.2.13 provide the  $^{60}\text{Co}$  activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the MPC-32, MPC-24, and the MPC-68 for varying burnup and cooling times. The design basis 6x6 and MOX fuel assemblies are conservatively assumed to have the same  $^{60}\text{Co}$  source strength as the BWR design basis fuel. This is a conservative assumption as the design basis 6x6 fuel and MOX fuel assemblies are limited to a significantly lower burnup and longer cooling time than the design basis fuel.

In addition to the two sources already mentioned, a third source arises from  $(n,\gamma)$  reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cs-134 and Eu-154, two of the major contributors to the gamma source, range from 0.79 to 1.009 and 0.79 to 0.98, respectively. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

## 5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the  $^{235}\text{U}$  initial enrichments for various burnup ranges from 20,000 - 75,000 MWD/MTU for PWR and 20,000 - 70,000 MWD/MTU for BWR zircaloy clad fuel. These enrichments are based on References [5.2.6] and [5.2.7]. Table 8 of reference [5.2.6] presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, for burnups up to 50,000 MWD/MTU, are approximately the average enrichments from Table 8 of reference [5.2.6] for the burnup range that is 5,000 MWD/MTU less than the ranges listed in

Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] and some additional data from nuclear plants was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the shielding analysis are adequate to bound the fuel authorized by the limits in Section 2.1.9 for loading in the HI-STORM system. Since the enrichment does affect the source term evaluation, it is recommended that the site-specific dose evaluation consider the enrichment for the fuel being stored.

The neutron source calculated for the design basis fuel assemblies for the MPC-24, MPC-32, and MPC-68 and the design basis 6x6 fuel are listed in Tables 5.2.15 through 5.2.18 in neutrons/s for varying burnup and cooling times. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly.  $^{244}\text{Cm}$  accounts for approximately 92-97% of the total number of neutrons produced. Alpha,n reactions in isotopes other than  $^{244}\text{Cm}$  account for approximately 0.3-2% of the neutrons produced while spontaneous fission in isotopes other than  $^{244}\text{Cm}$  account for approximately 2-8% of the neutrons produced within the  $\text{UO}_2$  fuel. In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cm-244 ranges from 0.81 to 0.95. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

### 5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.3 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.3 is actually

longer than the true active fuel length of 122 inches for the WE 15x15 and 83 inches for the LaCrosse 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this does not reflect the potential change in dose rates. As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 neutrons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches and 144 inches, respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 neutrons/s/inch and 1 neutron/s/inch for the stainless steel and zircaloy fuel, respectively. The result would be a higher neutron dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.8, 5.2.9, 5.2.19, and 5.2.20 list the gamma and neutron source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV, which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.4 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

#### 5.2.4 Non-fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM 100 System as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Section 2.1.9.

#### 5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding

BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable Co-60 source and decay heat from the BPRA and TPD were specified as: 141 curies Co-60 and 2.19 watts for each TPD and 895 curies Co-60 and 14.4 watts for each BPRA. Table 5.2.31 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted.

Section 5.4.6 discusses the increase in the cask dose rates due to the insertion of BPRAs or TPDs into fuel assemblies.

#### 5.2.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B<sub>4</sub>C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have

36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited. These devices are required to be stored in the locations as outlined in Section 2.1.9.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

#### Configuration 1: CRA and APSR

This configuration had the lower 15 inches of the CRA and APSR activated at full flux with two regions above the 15 inches activated at a reduced power level. This simulates a CRA or APSR which was operated at 10% insertion. The regions above the 15 inches reflect the upper portion of the fuel assembly.

#### Configuration 2: CRA and APSR

This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

#### Configuration 3: APSR

This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.



Tables 5.2.34 and 5.2.35 present the source terms, including decay heat, that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in Section 2.1.9. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Section 5.4.6 discusses the effect on dose rate of the insertion of APSRs into the inner four fuel assemblies in the MPC-24 and inner twelve fuel assemblies in the MPC-32, as well as the insertion of CRAs into the inner twelve fuel assemblies of the MPC-24 and MPC-32.

### 5.2.5 Choice of Design Basis Assembly

The analysis presented in this chapter was performed to bound the fuel assembly classes listed in Tables 2.1.1 and 2.1.2. In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly that produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the PWR MPCs (MPC-24 and MPC-32) and the BWR MPCs (MPC-68).

#### 5.2.5.1 PWR Design Basis Assembly

Table 2.1.1 lists the PWR fuel assembly classes that were evaluated to determine the design basis PWR fuel assembly. Within each class, the fuel assembly with the highest UO<sub>2</sub> mass was analyzed. Since the variations of fuel assemblies within a class are very minor (pellet diameter, clad thickness, etc.), it is conservative to choose the assembly with the highest UO<sub>2</sub> mass. For a given class of assemblies, the one with the highest UO<sub>2</sub> mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, the highest UO<sub>2</sub> mass will have produced the most energy and therefore the most fission products.

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table. The fuel assembly listed for each class is the assembly with the highest UO<sub>2</sub> mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Indian Point 1, Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time.

The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO<sub>2</sub> mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO<sub>2</sub> mass produces the highest radiation source term. The power/assembly values used in Table 5.2.25 were calculated by dividing 110% of the thermal power for commercial PWR reactors using that array class by the number of assemblies in the core. The higher thermal power, 110%, was used to account for potential power uprates. The power level used for the B&W15 is an additional 17% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

The Haddam Neck and San Onofre 1 classes are shorter stainless steel clad versions of the WE 15x15 and WE 14x14 classes, respectively. Since these assemblies have stainless steel clad, they were analyzed separately as discussed in Section 5.2.3. Based on the results in Table 5.2.27, which show that the WE 15x15 assembly class has a higher source term than the WE 14x14 assembly class, the Haddam Neck, WE 15x15, fuel assembly was analyzed as the bounding PWR stainless steel clad fuel assembly. The Indian Point 1 fuel assembly is a unique 14x14 design with a smaller mass of fuel and clad than the WE14x14. Therefore, it is also bounded by the WE 15x15 stainless steel fuel assembly.

As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 14x14A array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other PWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.

#### 5.2.5.2 BWR Design Basis Assembly

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO<sub>2</sub> mass was analyzed. Since the variations of fuel assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO<sub>2</sub> mass. For a given array type of assemblies, the one with the highest UO<sub>2</sub> mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment,

it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table. The fuel assembly listed for each array type is the assembly that has the highest  $\text{UO}_2$  mass. All fuel assemblies in Table 5.2.26 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.28. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.2. This fuel assembly also has the highest  $\text{UO}_2$  mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest  $\text{UO}_2$  mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in Section 2.1.9. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type. The power/assembly values used in Table 5.2.26 were calculated by dividing 120% of the thermal power for commercial BWR reactors by the number of assemblies in the core. The higher thermal power, 120%, was used to account for potential power uprates. The power level used for the 7x7 is an additional 4% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling time for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in Section 2.1.9. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher  $\text{UO}_2$  mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies, which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher  $\text{UO}_2$  mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis 6x6 fuel assembly can be intact or damaged, the analysis presented in Section 5.4.2 for the damaged 6x6 fuel assembly also demonstrates the acceptability of storing intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 9x9G array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other BWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.

### 5.2.5.3 Decay Heat Loads and Allowable Burnup and Cooling Times

Section 2.1.6 describes the calculation of the MPC maximum decay heat limits per assembly. These limits, which differ for uniform and regionalized loading, are presented in Section 2.1.9. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits. Since the decay heat of an assembly will vary slightly with enrichment for a fixed burnup and cooling time, an equation is used to represent burnup as a function of decay heat and enrichment. This equation is of the form:

$$B_u = A * q + B * q^2 + C * q^3 + D * E_{235}^2 + E * E_{235} * q + F * E_{235} * q^2 + G$$

where:

$B_u$  = Burnup in MWD/MTU

$q$  = assembly decay heat (kW)

$E_{235}$  = wt.%  $^{235}\text{U}$

The coefficients for this equation were developed by fitting ORIGEN-S calculated data for a specific cooling time using GNU PLOT [5.2.16]. ORIGEN-S calculations were performed for enrichments ranging from 0.7 to 5.0 wt.%  $^{235}\text{U}$  and burnups from 10,000 to 65,000 MWD/MTU for BWRs and 10,000 to 70,000 MWD/MTU for PWRs. The burnups were increased in 2,500 MWD/MTU increments. Using the ORIGEN-S data, the coefficients A through G were determined and then the constant, G, was adjusted so that all data points were bounded (i.e. calculated burnup less than or equal to ORIGEN-S value) by the fit. The coefficients were calculated using ORIGEN-S data for cooling times from 2 to 40 years. As a result, Section 2.1.9 provides different equation coefficients for each cooling time from 2 to 40 years. Additional discussion on the determination of the equation coefficients is provided in Appendix 5.F. Since the decay heat increases as the enrichment decreases, the allowable burnup will decrease as the enrichment decreases. Therefore, the enrichment used to calculate the allowable burnups becomes a minimum enrichment value and assemblies with an enrichment higher than the value used in the equation are acceptable for storage assuming they also meet the corresponding burnup

and decay heat requirements. Even though the lower limit of 0.7 wt.%  $^{235}\text{U}$  was used in developing the coefficients, these equations are valid for the few assemblies that might exist with enrichments below 0.7 wt.%  $^{235}\text{U}$ . This is because the curve fit is very well behaved in the enrichment range from 0.7 to 5.0 wt.%  $^{235}\text{U}$  and, therefore, it is expected that the curve fit will remain accurate for enrichments below 0.7 wt.%  $^{235}\text{U}$ .

Different array classes or combinations of classes were analyzed separately to determine the allowable burnup as a function of cooling time for the specified allowable decay heat limits. Calculating allowable burnups for individual array classes is appropriate because even two assemblies with the same MTU may have a different allowable burnup for the same allowable cooling time and permissible decay heat. The heavy metal mass specified in Table 5.2.25 and 5.2.26 and Section 2.1.9 for the various array classes is the value that was used in the determination of the coefficients as a function of cooling time and is the maximum for the respective assembly class. Equation coefficients for each array class listed in Tables 5.2.25 and 5.2.26 were developed. In the end, the equation for the 17x17B and 17x17C array classes resulted in almost identical burnups. Therefore, in Section 2.1.9 these array classes were combined and the coefficients for the 17x17C array class were used since these coefficients produce slightly lower allowable burnups.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. To estimate this uncertainty, an approach similar to the one in Reference [5.2.14] was used. The potential error in the ORIGEN-S decay heat calculations was estimated to be in the range of 3.5 to 5.5% for cooling times 2 to 40 years. The difference is due to the change in isotopes important to decay heat as a function of cooling time. In order to be conservative in the derivation of the coefficients for the burnup equation, a uniform 5% decay heat penalty was applied for both the PWR and BWR array classes.

As a demonstration that the decay heat values used to determine the allowable burnups are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

As mentioned above, the fuel assembly burnup and cooling times in Section 2.1.9 were calculated using the decay heat limits which are also stipulated in Section 2.1.9. The burnup and cooling times for the non-fuel hardware, in Section 2.1.9, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup, decay heat, and enrichment equations were derived without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in Section 2.1.9 regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel

assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.

#### 5.2.5.4 Burnup, Enrichment and Cooling time values for Site Specific Dose Analyses

As discussed earlier in this Chapter, site-specific dose evaluations are required to show compliance with the regulatory requirements, and those need to consider the types, burnups, enrichments and cooling times of the fuel to be stored. Since it is impractical to evaluate every fuel assembly individually, a bounding approach is typically used where assemblies are grouped and bounding characteristics are selected and evaluated for each group. Recommendations and guidance for those selections are as follows:

For the fuel assembly type, the one approach would be to use the design basis assembly type, since this has been shown to bound all other assembly types (see Subsections 5.2.5.1 and 5.2.5.2). However, if this approach is considered too conservative, it is also acceptable to utilize a site-specific fuel assembly type. In this case, that fuel assembly type needs to be considered in both the radiation transport analyses and the source term evaluations.

For burnups, enrichments and cooling times, selecting an appropriate burnup and enrichment combination or combinations (for given lower bound cooling times) could be difficult, since the more conservative values are a higher burnup but a lower enrichment (see Subsection 5.2.2). One approach would be to have a single group, i.e. select bounding values for all those parameters: upper bound burnup, lower bound enrichment, and lower bound cooling time. However, this could be excessively conservative, since combinations of high burnup and low enrichment are typically not found in spent fuel. A more practical approach would be to establish several groups, each with an upper bound burnup and lower bound enrichment. A separate evaluation may be required for each group in that case. However, with the help of the information shown in Table 5.2.24 it may be possible to avoid multiple analyses and demonstrate compliance with a single set of burnup, enrichment and cooling time. To support this approach, the results of source term calculation for all burnup and enrichment combinations listed in the table were compared, for the entire neutron and photon energy spectrum used in the dose analyses. The comparison shows, that in all cases, the source terms for the higher burnup, with the correspondingly listed enrichment, bound those with any lower burnup in the same burnup column, again with the correspondingly listed enrichment. The burnup column refers to either the column of the upper bound or lower bound burnups of the burnup ranges. For example, BWR fuel with 45,000 MWD/MTU and 3.0 wt% enrichment bounds BWR fuel with 40,000 MWD/MTU and 2.9 wt% enrichment, or PWR fuel with 50,000 MWD/MTU and 3.9 wt% bounds PWR fuel with 45,000 MWD/MTU and 3.6 wt% enrichment. Also, the maximum burnup for PWR and BWR, together with the correspondingly listed enrichment, bounds all other burnup and enrichment combination for PWR and BWR fuel, respectively. Using this comparison, it may be possible to bound the entire inventory with a single burnup and enrichment value (in combination with a lower bound cooling time). Additional considerations for this approach are as follows:

- The comparison was performed for design basis fuel. It is expected that the same conclusions would be valid for other fuel types, but that has not been demonstrated.
- Small deviations from the combinations listed in Table 5.2.24 for a limited number of assemblies would have an insignificant effect on doses (see Section 5.2.2) and may therefore be accepted without further evaluation.
- Assemblies with an enrichment below that listed in Table 5.2.24 for a given burnup may still be bounded by the assemblies with a higher burnup listed in that table, since the comparison shows margin in most cases. Therefore, a site-specific expansion of the table using source term comparisons may still allow a single bounding burnup and enrichment combination to be used in the dose analysis.

### 5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STORM 100 System.

### 5.2.7 Fuel Assembly Neutron Sources

Neutron source assemblies (NSAs) are used in reactors for startup. There are different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, polonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

#### 5.2.7.1 PWR Neutron Source Assemblies

During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. Reference [5.2.5] provides the masses of steel and inconel for the NSAs. Using these masses it was determined that the total activation of a primary or secondary source is bound by the total activation of a BPRA (see Table 5.2.31). Therefore, storage of NSAs is acceptable and a detailed dose rate analysis using the gamma source from activated NSAs is not performed. The burnup and cooling time limits for TPDs, as listed in Section 2.1.9, are being applied to NSAs since they cover a larger range of burnups.

Antimony-beryllium sources are used as secondary (regenerative) neutron sources in reactor cores. The Sb-Be source produces neutrons from a gamma-n reaction in the beryllium, where the gamma originates from the decay of neutron-activated antimony. The very short half-life of  $^{124}\text{Sb}$ , 60.2 days, however results in a complete decay of the initial amount generated in the reactor within a few years after removal from the reactor. The production of neutrons by the Sb-Be source through regeneration in the MPC is orders of magnitude lower than the design-basis fuel assemblies. Therefore Sb-Be sources do not contribute to the total neutron source in the MPC.

Primary neutron sources (californium, americium-beryllium, plutonium-beryllium and polonium-beryllium) are usually placed in the reactor with a source-strength on the order of  $5\text{E}+08$  n/s. This source strength is similar to, but not greater than, the maximum design-basis fuel assembly source strength listed in Tables 5.2.15 and 5.2.16.

By the time NSAs are stored in the MPC, the primary neutron sources will have been decaying for many years since they were first inserted into the reactor (typically greater than 10 years). For the  $^{252}\text{Cf}$  source, with a half-life of 2.64 years, this means a significant reduction in the source intensity; while the  $^{210}\text{Po}$ -Be source, with a half-life of 138 days, is virtually eliminated. The  $^{238}\text{Pu}$ -Be and  $^{241}\text{Am}$ -Be sources, however, have a significantly longer half-life, 87.4 years and 433 years, respectively. As a result, their source intensity does not decrease significantly before storage in the MPC. Since the  $^{238}\text{Pu}$ -Be and  $^{241}\text{Am}$ -Be sources may have a source intensity similar to a design-basis fuel assembly when they are stored in the MPC, only a single NSA is permitted for storage in the MPC. Since storage of a single NSA would not significantly increase the total neutron source in an MPC, storage of NSAs is acceptable and detailed dose rate analysis of the neutron source from NSAs is not performed.

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Section 2.1.9.

#### 5.2.7.2 BWR Neutron Source Assemblies

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STORM 100 System.



### 5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and Section 2.1.9 requires that these channels be stored in the inner sixteen locations.

The activation of a single stainless steel channel was calculated by simulating the irradiation of the channels with ORIGEN-S using the flux calculated from the LaCrosse fuel assembly. The mass of the steel channel in the active fuel zone (83 inches) was used in the analysis. For burnups beyond 22,500 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 22,500 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 22,500 MWD/MTU.

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming the plant operated continually from 11/69 until 4/87, approximately 17.5 years or 6388 days, the accumulated burnup for the channels would be 186,000 MWD/MTU (6388 days times 29.17 MW/MTU from Table 5.2.3). Therefore, the cobalt activity calculated for a single stainless steel channel irradiated for 180,000 MWD/MTU was calculated to be 667 curies of Co-60 for 13 years cooling. This is equivalent to a source of  $4.94\text{E}+13$  photons/sec in the energy range of 1.0-1.5 MeV.

In order to demonstrate that sixteen stainless steel channels are acceptable for storage in an MPC-68, a comparison of source terms is performed. Table 5.2.8 indicates that the source term for the LaCrosse design basis fuel assembly in the 1.0-1.5 MeV range is  $6.34\text{E}+13$  photons/sec for 10 years cooling, assuming a 144 inch active fuel length. This is equivalent to  $4.31\text{E}+15$  photons/sec/cask. At 13 years cooling, the fuel source term in that energy range decreases to  $4.31\text{E}+13$  photons/sec which is equivalent to  $2.93\text{E}+15$  photons/sec/cask. If the source term from the stainless steel channels is scaled to 144 inches and added to the 13 year fuel source term the result is  $4.30\text{E}+15$  photons/sec/cask ( $2.93\text{E}+15$  photons/sec/cask +  $4.94\text{E}+13$  photons/sec/channel x 144 inch/83 inch x 16 channels/cask). This number is equivalent to the 10 year  $4.31\text{E}+15$  photons/sec/cask source calculated from Table 5.2.8 and used in the shielding analysis in this chapter. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.

Table 5.2.1

## DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	<b>PWR</b>	<b>BWR</b>
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	3.6	3.2
Specific power (MW/MTU)	40	30
Weight of UO <sub>2</sub> (kg) <sup>††</sup>	562.029	225.177
Weight of U (kg) <sup>††</sup>	495.485	198.516

## Notes:

1. The B&W 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, CE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun.
2. The GE 7x7 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

†† Derived from parameters in this table.

Table 5.2.1 (continued)

## DESCRIPTION OF DESIGN BASIS FUEL

	<b>PWR</b>	<b>BWR</b>
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2

## DESCRIPTION OF DESIGN BASIS GE 6x6 ZIRCALOY CLAD FUEL

	<b>BWR</b>
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	129.5
Weight of U (kg) <sup>†</sup>	114.2

## Notes:

1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

<sup>†</sup> Derived from parameters in this table.

Table 5.2.3

## DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

	<b>PWR</b>	<b>BWR</b>
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	3.5	3.5
Burnup (MWD/MTU) <sup>†</sup>	40,000 (MPC-24 and 32)	22,500 (MPC-68)
Cooling Time (years) <sup>†</sup>	8 (MPC-24), 9 (MPC-32)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

## Notes:

1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: Indian Point 1, Haddam Neck, and San Onofre 1.
2. The LaCrosse 10x10 is the design basis assembly for the following fuel assembly class listed in Table 2.1.2: LaCrosse.

<sup>†</sup> Burnup and cooling time combinations are equivalent to or conservatively bound the limits in Section 2.1.9.

Table 5.2.4

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	45,000 MWD/MTU 3 Year Cooling		69,000 MWD/MTU 5 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	3.05E+15	5.30E+15	3.26E+15	5.67E+15
0.7	1.0	1.37E+15	1.62E+15	1.23E+15	1.44E+15
1.0	1.5	2.96E+14	2.37E+14	2.69E+14	2.15E+14
1.5	2.0	2.91E+13	1.66E+13	1.41E+13	8.08E+12
2.0	2.5	3.79E+13	1.68E+13	7.56E+12	3.36E+12
2.5	3.0	1.14E+12	4.13E+11	3.56E+11	1.29E+11
Total		4.78E+15	7.18E+15	4.78E+15	7.34E+15

Table 5.2.5

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	60,000 MWD/MTU 3 Year Cooling		75,000 MWD/MTU 5 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	4.11E+15	7.14E+15	3.55E+15	6.17E+15
0.7	1.0	1.98E+15	2.33E+15	1.36E+15	1.60E+15
1.0	1.5	4.04E+14	3.23E+14	2.94E+14	2.35E+14
1.5	2.0	3.41E+13	1.95E+13	1.50E+13	8.59E+12
2.0	2.5	3.95E+13	1.76E+13	7.63E+12	3.39E+12
2.5	3.0	1.29E+12	4.70E+11	3.72E+11	1.35E+11
Total		6.57E+15	9.84E+15	5.23E+15	8.02E+15

Table 5.2.6

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>50,000 MWD/MTU 3 Year Cooling</b>	
		(MeV/s)	(Photons/s)
0.45	0.7	1.28E+15	2.23E+15
0.7	1.0	5.76E+14	6.77E+14
1.0	1.5	1.18E+14	9.47E+13
1.5	2.0	1.04E+13	5.92E+12
2.0	2.5	1.20E+13	5.33E+12
2.5	3.0	4.04E+11	1.47E+11
Total		2.00E+15	3.01E+15



Table 5.2.7

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

Lower Energy	Upper Energy	30,000 MWD/MTU 18-Year Cooling	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
4.5e-01	7.0e-01	1.53e+14	2.65e+14
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+09	5.99e+08
2.5	3.0	7.30e+07	2.66e+07
Totals		1.61e+14	2.73e+14

Table 5.2.8

**CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL**

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>22,500 MWD/MTU 10-Year Cooling</b>	
		<b>(MeV/s)</b>	<b>(Photons/s)</b>
(MeV)	(MeV)		
4.5e-01	7.0e-01	2.72e+14	4.74+14
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		3.72e+14	5.61e+14

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 83-inch active fuel length.

Table 5.2.9

**CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL**

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>40,000 MWD/MTU 8-Year Cooling</b>		<b>40,000 MWD/MTU 9-Year Cooling</b>	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.37e+15	2.38e+15	1.28E+15	2.22E+15
7.0e-01	1.0	2.47e+14	2.91e+14	1.86E+14	2.19E+14
1.0	1.5	4.59e+14	3.67e+14	4.02E+14	3.21E+14
1.5	2.0	3.99e+12	2.28e+12	3.46E+12	1.98E+12
2.0	2.5	5.85e+11	2.60e+11	2.69E+11	1.20E+11
2.5	3.0	3.44e+10	1.25e+10	1.77E+10	6.44E+09
Totals		2.08e+15	3.04e+15	1.87E+15	2.76E+15

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 122-inch active fuel length.

Table 5.2.10

SCALING FACTORS USED IN CALCULATING THE  $^{60}\text{Co}$  SOURCE

<b>Region</b>	<b>PWR</b>	<b>BWR</b>
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.11

CALCULATED MPC-32  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
ZIRCALOY CLAD FUEL  
AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b>45,000 MWD/MTU and 3-Year Cooling (curies)</b>	<b>69,000 MWD/MTU and 5-Year Cooling (curies)</b>
Lower End Fitting	217.58	208.12
Gas Plenum Springs	16.60	15.88
Gas Plenum Spacer	9.52	9.11
Expansion Springs	N/A	N/A
Incore Grid Spacers	563.50	539.00
Upper End Fitting	106.72	102.08
Handle	N/A	N/A

Table 5.2.12

CALCULATED MPC-24  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
ZIRCALOY CLAD FUEL  
AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b>60,000 MWD/MTU and 3-Year Cooling (curies)</b>	<b>75,000 MWD/MTU and - 5 Year Cooling (curies)</b>
Lower End Fitting	249.74	219.47
Gas Plenum Springs	19.05	16.74
Gas Plenum Spacer	10.93	9.61
Expansion Springs	N/A	N/A
Incore Grid Spacers	646.80	568.40
Upper End Fitting	122.50	107.65
Handle	N/A	N/A

Table 5.2.13

CALCULATED MPC-68  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
ZIRCALOY CLAD FUEL  
AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b>50,000 MWD/MTU and 3-Year Cooling (curies)</b>
Lower End Fitting	90.55
Gas Plenum Springs	27.67
Gas Plenum Spacer	N/A
Expansion Springs	5.03
Grid Spacer Springs	41.50
Upper End Fitting	25.15
Handle	3.14

Table 5.2.14

THIS TABLE INTENTIONALLY DELETED



Table 5.2.15

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>45,000 MWD/MTU 3-Year Cooling (Neutrons/s)</b>	<b>69,000 MWD/MTU 5-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	1.77E+07	5.31E+07
4.0e-01	9.0e-01	9.03E+07	2.71E+08
9.0e-01	1.4	8.27E+07	2.48E+08
1.4	1.85	6.09E+07	1.82E+08
1.85	3.0	1.08E+08	3.21E+08
3.0	6.43	9.77E+07	2.92E+08
6.43	20.0	8.66E+06	2.60E+07
Totals		4.65E+08	1.39E+09

Table 5.2.16

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>60,000 MWD/MTU 3-Year Cooling (Neutrons/s)</b>	<b>75,000 MWD/MTU 5-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	3.76E+07	6.82E+07
4.0e-01	9.0e-01	1.92E+08	3.48E+08
9.0e-01	1.4	1.76E+08	3.18E+08
1.4	1.85	1.29E+08	2.34E+08
1.85	3.0	2.28E+08	4.11E+08
3.0	6.43	2.08E+08	3.75E+08
6.43	20.0	1.84E+07	3.34E+07
Totals		9.89E+08	1.79E+09

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>50,000 MWD/MTU 3-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	9.79E+06
4.0e-01	9.0e-01	5.00E+07
9.0e-01	1.4	4.57E+07
1.4	1.85	3.37E+07
1.85	3.0	5.93E+07
3.0	6.43	5.40E+07
6.43	20.0	4.79E+06
Totals		2.57E+08

Table 5.2.18

**CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL**

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>30,000 MWD/MTU 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Total		2.20e+7

Table 5.2.19

**CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL**

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>22,500 MWD/MTU 10-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 83-inch active fuel length.

Table 5.2.20

**CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL**

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>40,000 MWD/MTU 8-Year Cooling (Neutrons/s)</b>	<b>40,000 MWD/MTU 9-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	1.04e+7	1.01E+07
4.0e-01	9.0e-01	5.33e+7	5.14E+07
9.0e-01	1.4	4.89e+7	4.71E+07
1.4	1.85	3.61e+7	3.48E+07
1.85	3.0	6.41e+7	6.18E+07
3.0	6.43	5.79e+7	5.58E+07
6.43	20.0	5.11e+6	4.92E+06
Totals		2.76e+8	2.66E+08

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 122-inch active fuel length.

Table 5.2.21

## DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	<b>BWR</b>
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO <sub>2</sub> and PuUO <sub>2</sub>
No. of UO <sub>2</sub> Rods	27
No. of PuUO <sub>2</sub> rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U) <sup>†</sup>	2.24 (UO <sub>2</sub> rods) 0.711 (PuUO <sub>2</sub> rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO <sub>2</sub> ,PuUO <sub>2</sub> (kg) <sup>††</sup>	123.3
Weight of U,Pu (kg) <sup>††</sup>	108.7

<sup>†</sup> See Table 5.3.3 for detailed composition of PuUO<sub>2</sub> rods.

<sup>††</sup> Derived from parameters in this table.

Table 5.2.22

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>30,000 MWD/MTU 18-Year Cooling</b>	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
4.5e-01	7.0e-01	1.45e+14	2.52e+14
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		1.53e+14	2.60e+14



Table 5.2.23

**CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL**

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>30,000 MWD/MTU 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.24

## INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

<b>Burnup Range (MWD/MTU)</b>	<b>Initial Enrichment (wt.% <sup>235</sup>U)</b>
<b>BWR Fuel</b>	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	3.6
55,000-60,000	4.0
60,000-65,000	4.4
65,000-70,000	4.8
<b>PWR Fuel</b>	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	3.9
55,000-60,000	4.2
60,000-65,000	4.5
65,000-70,000	4.8
70,000-75,000	5.0

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. This note does not apply to the maximum burnups of 70,000 and 75,000 MWD/MTU.

Table 5.2.25 (page 1 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	WE 14×14	WE 14x14	WE 15×15	WE 17×17	WE 17x17
Fuel assembly array class	14x14B	14x14A	15x15AB C	17x17B	17x17A
Active fuel length (in.)	144	144	144	144	144
No. of fuel rods	179	179	204	264	264
Rod pitch (in.)	0.556	0.556	0.563	0.496	0.496
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.422	0.4	0.422	0.374	0.36
Cladding thickness (in.)	0.0243	0.0243	0.0245	0.0225	0.0225
Pellet diameter (in.)	0.3659	0.3444	0.3671	0.3232	0.3088
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (% of theoretical)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.4	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	15.0	15.0	18.6	20.4	20.4
Specific power (MW/MTU)	36.409	41.097	39.356	43.031	47.137
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	467.319	414.014	536.086	537.752	490.901
Weight of U (kg) <sup>†</sup>	411.988	364.994	472.613	474.082	432.778
No. of Guide Tubes	17	17	21	25	25
Guide Tube O.D. (in.)	0.539	0.539	0.546	0.474	0.474
Guide Tube Thickness (in.)	0.0170	0.0170	0.0170	0.0160	0.0160

† Derived from parameters in this table.

Table 5.2.25 (page 2 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Fuel assembly array class	14x14C	16x16A	15x15DEF HI	17x17C
Active fuel length (in.)	144	150	144	144
No. of fuel rods	176	236	208	264
Rod pitch (in.)	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	0.3805	0.3255	0.3742	0.3252
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (95% of theoretical)	10.522 (96%)	10.522 (96%)	10.412 (95%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Power/assembly (MW)	13.7	17.5	19.819	20.4
Specific power (MW/MTU)	31.275	39.083	40	42.503
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	496.887	507.9	562.029	544.428
Weight of U (kg) <sup>†</sup>	438.055	447.764	495.485	479.968
No. of Guide Tubes	5	5	17	25
Guide Tube O.D. (in.)	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0400	0.0400	0.0160	0.0175

<sup>†</sup> Derived from parameters in this table.

Table 5.2.26 (page 1 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	7x7	8x8	8x8	9x9	9x9
Fuel assembly array class	7x7B	8x8B	8x8CDE	9x9A	9x9B
Active fuel length (in.)	144	144	150	144	150
No. of fuel rods	49	64	62	74	72
Rod pitch (in.)	0.738	0.642	0.64	0.566	0.572
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.484	0.493	0.44	0.433
Cladding thickness (in.)	0.0355	0.02725	0.034	0.028	0.026
Pellet diameter (in.)	0.488	0.4195	0.416	0.376	0.374
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (% of theoretical)	10.412 (95%)	10.412 (95%)	10.412 (95%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	5.96	5.75	5.75	5.75	5.75
Specific power (MW/MTU)	30	30	30.24	31.97	31.88
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	225.177	217.336	215.673	204.006	204.569
Weight of U (kg) <sup>†</sup>	198.516	191.603	190.137	179.852	180.348
No. of Water Rods	0	0	2	2	1
Water Rod O.D. (in.)	n/a	n/a	0.493	0.98	1.516
Water Rod Thickness (in.)	n/a	n/a	0.034	0.03	0.0285

† Derived from parameters in this table.

Table 5.2.26 (page 2 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	9x9	9x9	9x9	10x10	10x10
Fuel assembly array class	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Active fuel length (in.)	150	144	150	144	150
No. of fuel rods	80	76	72	92	96
Rod pitch (in.)	0.572	0.572	0.572	0.510	0.488
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.423	0.443	0.424	0.404	0.378
Cladding thickness (in.)	0.0295	0.0285	0.03	0.0260	0.0243
Pellet diameter (in.)	0.3565	0.3745	0.3565	0.345	0.3224
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (% of theoretical)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	5.75	5.75	5.75	5.75	5.75
Specific power (MW/MTU)	31.58	31.38	35.09	30.54	32.18
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	206.525	207.851	185.873	213.531	202.687
Weight of U (kg) <sup>†</sup>	182.073	183.242	163.865	188.249	178.689
No. of Water Rods	1	5	1	2	1
Water Rod O.D. (in.)	0.512	0.546	1.668	0.980	Note 1
Water Rod Thickness (in.)	0.02	0.0120	0.032	0.0300	Note 1

Note 1: 10x10C has a diamond shaped water rod with 4 additional segments dividing the fuel rods into four quadrants.

† Derived from parameters in this table.

Table 5.2.27

**COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD PWR FUEL**  
 3.4 wt.% <sup>235</sup>U - 40,000 MWD/MTU - 5 years cooling

Assembly	WE 14x14	WE 14x14	WE 15x15	WE 17x17	WE 17x17	CE 14x14	CE 16x16	B&W 15x15	B&W 17x17
Array class	14x14A	14x14B	15x15 ABC	17x17A	17x17B	14x14C	16x16A	15x15 DEFHI	17x17C
Neutrons/sec	1.76E+8 1.78E+8	2.32E+8 2.35E+8	2.70E+8 2.73E+8	2.18E+8	2.68E+8	2.32E+8	2.38E+8	2.94E+8	2.68E+8
Photons/sec (0.45-3.0 MeV)	2.88E+15 2.93E+15	3.28E+15 3.32E+15	3.80E+15 3.86E+15	3.49E+15	3.85E+15	3.37E+15	3.57E+15	4.01E+15	3.89E+15
Thermal power (watts)	809.5 820.7	923.5 933.7	1073 1086	985.6	1090	946.6	1005	1137	1098

## Note:

The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.28

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD BWR FUEL  
3.0 wt.% <sup>235</sup>U - 40,000 MWD/MTU - 5 years cooling

Assembly	7x7	8x8	8x8	9x9	9x9	9x9	9x9	9x9	10x10	10x10
Array Class	7x7B	8x8B	8x8CDE	9x9A	9x9B	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Neutrons/sec	1.33E+8	1.22E+8	1.22E+8	1.13E+8	1.06E+8	1.09E+8	1.24E+8	9.15E+7	1.24E+8	1.07E+8
Photons/sec (0.45-3.0 MeV)	1.55E+15	1.49E+15	1.48E+15	1.41E+15	1.40E+15	1.42E+15	1.45E+15	1.28E+15	1.48E+15	1.40E+15
Thermal power (watts)	435.5	417.3	414.2	394.2	389.8	395	405.8	356.9	413.5	389.2



Table 5.2.29

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL  
AND VALUES REPORTED IN THE  
DOE CHARACTERISTICS DATABASE<sup>†</sup> FOR  
30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Source Term Calculations (watts/assembly)
<b>PWR Fuel</b>		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	802.7
CE 16x16	653.7	734.3
CE 14x14	601.3	694.9
WE 17x17	742.5	795.4
WE 15x15	762.2	796.2
WE 14x14	649.6	682.9
<b>BWR Fuel</b>		
7x7	310.9	315.7
8x8	296.6	302.8
9x9	275.0	286.8

## Notes:

1. The decay heat from the source term calculations is the maximum value calculated for that fuel assembly class.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.
4. The enrichments used for the column labeled “Decay Heat from Source Term Calculations” were consistent with Table 5.2.24.

<sup>†</sup> Reference [5.2.7].

Table 5.2.30

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY  
AND THIMBLE PLUG DEVICE

<b>Region</b>	<b>BPRA</b>	<b>TPD</b>
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.31

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD  
ASSEMBLIES AND THIMBLE PLUG DEVICES

<b>Region</b>	<b>BPRA</b>	<b>TPD</b>
Upper End Fitting (curies Co-60)	32.7	71.1
Gas Plenum Spacer (curies Co-60)	5.0	25.5
Gas Plenum Springs (curies Co-60)	8.9	44.4
In-core (curies Co-60)	848.4	N/A

Table 5.2.32

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY  
CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Inconel)	Mass of absorber (kg AgInCd)
Start (in)	Finish (in)	Length (in)			
<b>Configuration 1 - 10% Inserted</b>					
0.0	15.0	15.0	1.0	1.32	7.27
15.0	18.8125	3.8125	0.2	0.34	1.85
18.8125	28.25	9.4375	0.1	0.83	4.57
<b>Configuration 2 - Fully Removed</b>					
0.0	3.8125	3.8125	0.2	0.34	1.85
3.8125	13.25	9.4375	0.1	0.83	4.57

Table 5.2.33

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD  
CONFIGURATION S FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Steel)	Mass of absorber (kg Inconel)
Start (in)	Finish (in)	Length (in)			
<b>Configuration 1 - 10% Inserted</b>					
0.0	15.0	15.0	1.0	1.26	5.93
15.0	18.8125	3.8125	0.2	0.32	1.51
18.8125	28.25	9.4375	0.1	0.79	3.73
<b>Configuration 2 - Fully Removed</b>					
0.0	3.8125	3.8125	0.2	0.32	1.51
3.8125	13.25	9.4375	0.1	0.79	3.73
<b>Configuration 3 - Fully Inserted</b>					
0.0	63.0	63.0	1.0	5.29	24.89
63.0	66.8125	3.8125	0.2	0.32	1.51
66.8125	76.25	9.4375	0.1	0.79	3.73

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD  
ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Photons/sec from AgInCd			Curies Co-60 from Inconel
Start (in)	Finish (in)	Length (in)	0.3-0.45 MeV	0.45-0.7 MeV	0.7-1.0 MeV	
<b>Configuration 1 - 10% Inserted - 80.8 watts decay heat</b>						
0.0	15.0	15.0	1.91e+14	1.78e+14	1.42e+14	1111.38
15.0	18.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
18.8125	28.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92
<b>Configuration 2 - Fully Removed - 8.25 watts decay heat</b>						
0.0	3.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
3.8125	13.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER  
SHAPING ROD CONFIGURATIONS

<b>Axial Dimensions Relative to Bottom of Active Fuel</b>			<b>Curies of Co-60</b>
<b>Start (in)</b>	<b>Finish (in)</b>	<b>Length (in)</b>	
<b>Configuration 1 - 10% Inserted - 46.2 watts decay heat</b>			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78
<b>Configuration 2 - Fully Removed - 4.72 watts decay heat</b>			
0.0	3.8125	3.8125	136.36
3.8125	13.25	9.4375	168.78
<b>Configuration 3 - Fully Inserted - 178.9 watts decay heat</b>			
0.0	63.0	63.0	11266.80
63.0	66.8125	3.8125	136.36
66.8125	76.25	9.4375	168.78

Table 5.2.36

DESCRIPTION OF FUEL ASSEMBLY USED TO ANNALYZE  
THORIA RODS IN THE THORIA ROD CANISTER

	<b>BWR</b>
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO <sub>2</sub> fuel rods	55
No. of UO <sub>2</sub> /ThO <sub>2</sub> fuel rods	9
Rod pitch (in.)	0.523
Cladding material	zircaloy
Rod diameter (in.)	0.412
Cladding thickness (in.)	0.025
Pellet diameter (in.)	0.358
Pellet material	98.2% ThO <sub>2</sub> and 1.8% UO <sub>2</sub> for UO <sub>2</sub> /ThO <sub>2</sub> rods or 98.5% ThO <sub>2</sub> and 1.5% UO <sub>2</sub> for UO <sub>2</sub> /ThO <sub>2</sub> rods
Pellet density (gm/cc)	10.412
Enrichment (w/o <sup>235</sup> U)	93.5 in UO <sub>2</sub> for UO <sub>2</sub> /ThO <sub>2</sub> rods and 1.8 for UO <sub>2</sub> rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of ThO <sub>2</sub> and UO <sub>2</sub> (kg) <sup>†</sup>	121.46
Weight of U (kg) <sup>†</sup>	92.29
Weight of Th (kg) <sup>†</sup>	14.74

<sup>†</sup> Derived from parameters in this table.



Table 5.2.37

CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD  
CANISTER CONTAINING EIGHTEEN THORIA RODS

98.2% ThO<sub>2</sub> and 1.8% UO<sub>2</sub> for UO<sub>2</sub>/ThO<sub>2</sub> rods

Lower Energy	Upper Energy	16,000 MWD/MTIHM 18-Year Cooling	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
4.5e-01	7.0e-01	3.07e+13	5.34e+13
7.0e-01	1.0	5.79e+11	6.81e+11
1.0	1.5	3.79e+11	3.03e+11
1.5	2.0	4.25e+10	2.43e+10
2.0	2.5	4.16e+8	1.85e+8
2.5	3.0	2.31e+11	8.39e+10
Totals		3.20e+13	5.45e+13

98.5% ThO<sub>2</sub> and 1.5% UO<sub>2</sub> for UO<sub>2</sub>/ThO<sub>2</sub> rods

Lower Energy	Upper Energy	16,000 MWD/MTIHM 18-Year Cooling	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
4.5e-01	7.0e-01	2.88e+13	5.02e+13
7.0e-01	1.0	5.38e+11	6.33e+11
1.0	1.5	3.48e+11	2.79e+11
1.5	2.0	4.04e+10	2.31e+10
2.0	2.5	3.92e+08	1.74e+08
2.5	3.0	2.39e+11	8.67e+10
Totals		3.00e+13	5.12e+13

Table 5.2.38

**CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD  
CANISTER CONTAINING EIGHTEEN THORIA RODS**

98.2% ThO<sub>2</sub> and 1.8% UO<sub>2</sub> for UO<sub>2</sub>/ThO<sub>2</sub> rods

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	5.65e+2
4.0e-01	9.0e-01	3.19e+3
9.0e-01	1.4	6.79e+3
1.4	1.85	1.05e+4
1.85	3.0	3.68e+4
3.0	6.43	1.41e+4
6.43	20.0	1.60e+2
Totals		7.21e+4

98.5% ThO<sub>2</sub> and 1.5% UO<sub>2</sub> for UO<sub>2</sub>/ThO<sub>2</sub> rods

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	5.99e+2
4.0e-01	9.0e-01	3.39e+3
9.0e-01	1.4	7.21e+3
1.4	1.85	1.11e+4
1.85	3.0	3.91e+4
3.0	6.43	1.50e+4
6.43	20.0	1.69e+2
Totals		7.66e+4

### 5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM 100 System, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accidents that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC and the 30 day 100% blockage of air inlets for the HI-STORM overpack. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. The MCNP model of the accident condition HI-STORM assumes a bounding volume of the neutron shield (concrete) loses all hydrogen and partial oxygen, and has corresponding lower density. This bounding volume correlates to all concrete in the cask body and lid that is at a temperature of at least 350°F.

#### 5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM 100 System, including the HI-TRAC transfer casks. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 through 5.3.6 show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. Figures 5.3.1 through 5.3.3 were created with the MCNP two-dimensional plotter and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figure 5.3.7 shows a cross sectional view of the 100-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. The dimensions for the HI-STORM 100S and HI-STORM 100S Version B overpacks are also shown on Figure 5.3.10. This figure notes two different dimensions for the inner and outer shells. These values apply only to the HI-STORM 100 and 100S. In these overpacks, the inner and outer shells can be manufactured from 1.25 and 0.75 inch thick steel, respectively, or both shells can be manufactured from 1 inch thick steel. The HI-STORM 100 and 100S were modeled as 1.25 and 0.75 inch thick shells.

Figures 5.3.11, 5.3.18, and 5.3.22 are axial representations of the HI-STORM 100, HI-STORM 100S, and HI-STORM 100S Version B overpacks, respectively, with the various as-modeled dimensions indicated.

Only the HI-STORM 100S Version B is analyzed in this chapter. This is reasonable because the HI-STORM 100S Version B overpack is shorter than the other overpacks, and the MPC is positioned closer to the inlet vent which results in higher dose rates at the inlet vent compared to the other overpacks. In addition, the HI-STORM 100S Version B has slightly higher offsite dose than the other overpacks.

Figures 5.3.12, 5.3.13, and 5.3.23 show axial cross-sectional views of the 100-, 125-ton, and 100D HI-TRAC transfer casks, respectively, with the as-modeled dimensions and materials specified. Figures 5.3.14, 5.3.15, and 5.3.20 show fully labeled radial cross-sectional views of the HI-TRAC 100, 125, and 125D transfer casks, respectively. Figure 5.3.14 also provides the information for the HI-TRAC 100D. Finally, Figures 5.3.16 and 5.3.17 show fully labeled diagrams of the transfer lids for the HI-TRAC 100 and 125 transfer casks. Since lead plate may be used instead of poured lead in the pool and transfer lids, there exists the possibility of a gap between the lead plate and the surrounding steel walls. This gap was accounted for in the analysis as depicted on Figures 5.3.16 and 5.3.17. The gap was not modeled in the pool lid since the gap will only exist on the outer edges of the pool lid and the highest dose rate is in the center. (All results presented in this chapter were calculated with the gap with the exception of the results presented in Figures 5.1.6, 5.1.7, and 5.1.11 which did not include the gap.) The HI-TRAC 100D and 125D do not utilize the transfer lid, rather they utilize the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid in this chapter are applicable to both the HI-TRAC 125 and 125D and the HI-TRAC 100 and 100D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 100 and 125. Consistent with the analysis of the transfer lid in which only the portion of the lid directly below the MPC was modeled, the structure of the mating device which surrounds the pool lid was not modeled.

Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. The HI-TRAC 125D has steel ribs in the lower water jacket while the HI-TRAC 125 does not. These additional ribs in the lower water jacket reduce the dose rate in the vicinity of the pool lid for the HI-TRAC 125D compared to the HI-TRAC 125. Since the dose rates at the midplane of the HI-TRAC 125D are higher than the HI-TRAC 125, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.

To reduce the gamma dose around the inlet and outlet vents, stainless steel cross plates, designated gamma shield cross plates<sup>†</sup> (see Figures 5.3.11 and 5.3.18), have been installed inside

<sup>†</sup> This design embodiment, formally referred to as “Duct Photon Attenuator,” has been disclosed as an invention by Holtec International for consideration by the US Patent Office for issuance of a patent under U.S. law.

all vents in all overpacks. The steel in these plates effectively attenuates the fuel and  $^{60}\text{Co}$  gammas that dominated the dose at these locations prior to their installation. Figure 5.3.19 shows three designs for the gamma shield cross plates to be used in the inlet and outlet vents. The designs in the top portion of the figure are mandatory for use in the HI-STORM 100 and 100S overpacks during normal storage operations and were assumed to be in place in the shielding analysis. The designs in the middle portion of the figure may be used instead of the mandatory designs in the HI-STORM 100S overpack to further reduce the radiation dose rates at the vents. These optional gamma shield cross plates could further reduce the dose rate at the vent openings by as much as a factor of two. The designs in the bottom portion of the figure are mandatory for use in the HI-STORM 100S Version B overpack during normal storage operations and were assumed to be in place in the shielding analysis.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 15 times the pitch and 7 times the pitch, respectively. Homogenization resulted in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The basket material in the top and bottom 0.9 inches where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9-inch area at the top and bottom of the MPC basket.
2. The upper and lower fuel spacers are not modeled, as the fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The fuel spacer length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis, but the fuel spacer materials are not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
3. For the MPC-32, MPC-24, and MPC-68, the MPC basket supports (or basket shims) are not modeled. This is conservative since it removes steel that would provide a small increase in shielding. The optional aluminum heat conduction elements are also conservatively not modeled.
4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch thick steel. The Boral and

sheathing are modeled explicitly. This is conservative since it removes steel that would provide a small amount of additional shielding.

5. In the modeling of the BWR fuel assemblies, the zircaloy flow channels were not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.
6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches compared to 6.25 inches or 7.5 inches.
7. The MPC-68 is designed for two lid thicknesses: 9.5 inches and 10 inches. Conservatively, all calculations reported in this chapter were performed with the 9.5 inch thick lid.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STORM 100 and HI-TRAC. Therefore, the models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

#### MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in both the HI-TRAC and the HI-STORM outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-TRAC or the HI-STORM.
2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Since the new MPC-24 configuration and the configuration of the MPC-24E are almost identical, the analysis of the earlier MPC-24 configuration is valid for the MPC-24E as well. Figure 5.3.21 shows the superseded and current configuration for the MPC-24 for comparison.
3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches to 0.0235 inches. However, the model still uses 0.06 inches. This discrepancy is compensated for by the use of 9/32 inch cell walls and 5 inch boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration in the 100-ton HI-TRAC for comparison to the superseded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface for a burnup of

60,000 MWD/MTU and a cooling time of 3 years. These results demonstrate that using the superceded MPC-24 design is conservative.

4. The MPC lid can be made either entirely from stainless steel, or from a stainless steel / carbon steel combination. In MCNP, the lid is modeled entirely as stainless steel. The difference in material would have a negligible effect on dose rates.

#### HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the HI-TRAC 125 was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces effective shielding thickness. The HI-TRAC 125D does not use pocket trunnions.
2. The lifting blocks in the top lid of the 125-ton HI-TRACs were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.
3. The door side plates that are in the middle of the transfer lid of the HI-TRAC 125 are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.
4. The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRACs was modeled as 4 inches larger than it is due to a design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.

#### HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.
3. In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.
4. The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.

5. The pedestal baseplate, which is steel with holes for pouring concrete, in the HI-STORM 100 and 100S overpacks was modeled as concrete rather than steel. This is acceptable because this piece of steel is positioned at the bottom of the pedestal below 5 inches of steel and a minimum of 11.5 inches of concrete and therefore will have no impact on the dose rates at the bottom vent.
6. Minor penetrations in the body of the overpack (e.g. holes for grounding straps) are not modeled as these are small localized effects which will not affect the off-site dose rates.
7. Deleted
8. The drawings in Section 1.5 indicate that the HI-STORM 100S has a variable height. This is achieved by adjusting the height of the body of the overpack. The pedestal height is not adjusted. Conservatively, all calculations in this chapter used the shorter height for the HI-STORM 100S.
9. In February 2002, the top plate on the HI-STORM 100 overpack was modified to be two pieces in a shear ring arrangement. The total thickness of the top plate was not changed. However, there is approximately a 0.5 inch gap between the two pieces of the top plate. This gap was not modeled in MCNP since it will result in a small increase in the dose rate on the overpack lid in an area where the dose rate is greatly reduced compared to other locations on the lid.
10. The MPC base support in the HI-STORM 100S Version B was conservatively modeled as a 1 inch thick plate resting on a two inch tall ring as shown in Figure 5.3.22. The design of the overpack utilizes a solid three inch plate.
11. The gussets in the inside lower corners of the HI-STORM 100S Version B overpack were not modeled. Concrete was modeled instead.

#### 5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, basket, inlet vents, and outlet vents are shown in these figures.



### 5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

The MCNP model of the HI-TRAC transfer cask describes the lifting trunnions, pocket trunnions, and the opening in the HI-TRAC top lid. The fins through the HI-TRAC water jacket are also modeled. Streaming considerations through these trunnions and fins are discussed in Section 5.4.1.

The design of the HI-STORM 100 System, as described in the drawings in Chapter 1, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the HI-TRAC transfer cask.
- The pocket trunnions of the HI-TRAC are modeled as solid blocks of steel. No credit is taken for any part of the pocket trunnion that extends beyond the water jacket.
- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.
- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. The holes in the vent and drain port cover plates are filled with a set screw and plug weld. The steel lost in the MPC lid at the port location is replaced with a block of steel located directly below the port opening and attached to the underside of the lid. This design feature is shown on the drawings in Chapter 1. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid plate.

### 5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM 100 System and HI-TRAC shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

The concrete density shown in Table 5.3.2 is the minimum concrete density analyzed in this chapter. The HI-STORM 100 overpacks are designed in such a way that the concrete density in the body of the overpack can be increased to approximately 3.2 gm/cc (200 lb/cu-ft). It is generally recommended to use concrete with a density of 155 lb/cu-ft or more. Increasing the density beyond this value would result in a significant reduction in the dose rates. This may be

beneficial based on on-site and off-site ALARA considerations. Lower densities may be necessary to address weight restrictions at certain sites.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorbing materials: Boral or Metamic. Both materials are made of aluminum and B<sub>4</sub>C powder. The Boral contains an aluminum and B<sub>4</sub>C powder mixture sandwiched between two aluminum plates while the Metamic is a single plate. The minimum <sup>10</sup>B areal density is the same for Boral and Metamic while the thicknesses are essentially the same. Therefore, the mass of Aluminum and B<sub>4</sub>C are essentially equivalent and there is no distinction between the two materials from a shielding perspective. As a result, Table 5.3.2 identifies the composition for Boral and no explicit calculations were performed with Metamic.

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Section 4.4 indicates that there are localized areas in the concrete in the lid of the overpack which approach 390°F. An increase in temperature from 300°F to 390°F results in an approximate 0.666% overall density reduction due to the loss of chemically unbound water. This density reduction results in a reduction in the mass fraction of hydrogen from 0.6% to 0.529% in the area affected by the temperature excursion. This is a localized effect with the maximum loss occurring at the bottom center of the lid where the temperature is the hottest and reduced loss occurring as the temperature decreases to 300°F.

Based on these considerations, the presence of localized temperatures up to 390°F in the lid concrete has a negligible effect on the shielding effectiveness of the HI-STORM 100 overpack lid.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there are two accident conditions that have any significant impact on the shielding configuration. One of these accidents is the loss of the neutron shield (water) in the HI-TRAC as

a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void. The other of these accidents is the 30 day 100% blockage of air inlets. The change is that a bounding volume of the neutron shield (concrete) is conservatively assumed to lose all hydrogen and partial oxygen, and has a lower corresponding density.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES<sup>†</sup>

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
<b>PWR</b>					
Lower End Fitting	0.0	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
<b>BWR</b>					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

<sup>†</sup> All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

## COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Uranium Oxide	10.412	<sup>235</sup> U	2.9971(BWR) 3.2615(PWR)
		<sup>238</sup> U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral <sup>†</sup>	2.644	<sup>10</sup> B	4.4226 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM)4.367 (MPC-24 in HI-TRAC)
		<sup>11</sup> B	20.1474 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

<sup>†</sup> All B-10 loadings in the Boral compositions are conservatively lower than the values defined in the Bill of Materials.

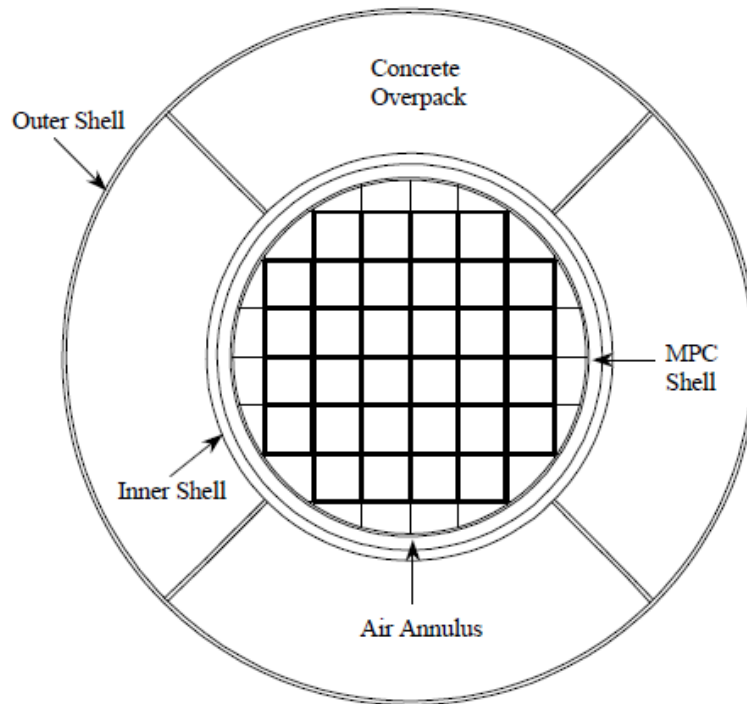
Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		<sup>10</sup> B	0.14087
		<sup>11</sup> B	0.64174
BWR Fuel Region Mixture	4.29251	<sup>235</sup> U	2.4966
		<sup>238</sup> U	70.9315
		O	9.8709
		Zr	16.4046
		N	8.35E-05
		Cr	0.0167
		Fe	0.0209
		Sn	0.2505
PWR Fuel Region Mixture	3.869939	<sup>235</sup> U	2.7652
		<sup>238</sup> U	71.9715
		O	10.0469
		Zr	14.9015
		Cr	0.0198
		Fe	0.0365
		Sn	0.2587

<b>Component</b>	<b>Density (g/cm<sup>3</sup>)</b>	<b>Elements</b>	<b>Mass Fraction (%)</b>
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.4862	SS304	100
Gas Plenum Springs (BWR)	0.2653	SS304	100
Expansion Springs (BWR)	0.6775	SS304	100
Upper End Fitting (BWR)	1.3692	SS304	100
Handle (BWR)	0.2572	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.9140 (water jacket)	H	11.2
	0.9619 (inside MPC)	O	88.8

<b>Component</b>	<b>Density (g/cm<sup>3</sup>)</b>	<b>Elements</b>	<b>Mass Fraction (%)</b>
Concrete	2.24	H	0.6
		O	50.0
		Si	31.5
		Al	4.8
		Na	1.7
		Ca	8.3
		Fe	1.2
		K	1.9

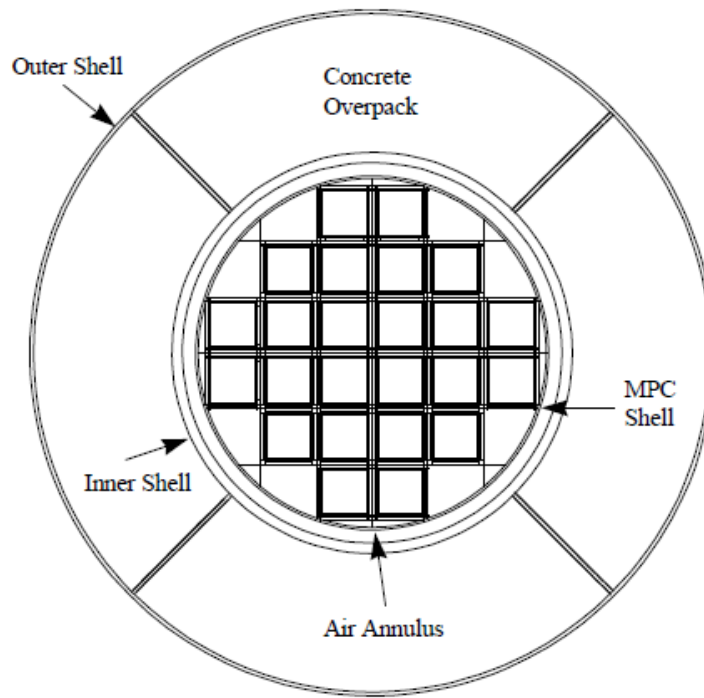


<b>Component</b>	<b>Density (g/cm<sup>3</sup>)</b>	<b>Elements</b>	<b>Mass Fraction (%)</b>
Mixed Oxide Pellets	10.412	<sup>238</sup> U	85.498
		<sup>235</sup> U	0.612
		<sup>238</sup> Pu	0.421
		<sup>239</sup> Pu	1.455
		<sup>240</sup> Pu	0.034
		<sup>241</sup> Pu	0.123
		<sup>242</sup> Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	<sup>238</sup> U	86.175
		<sup>235</sup> U	1.975
		O	11.85



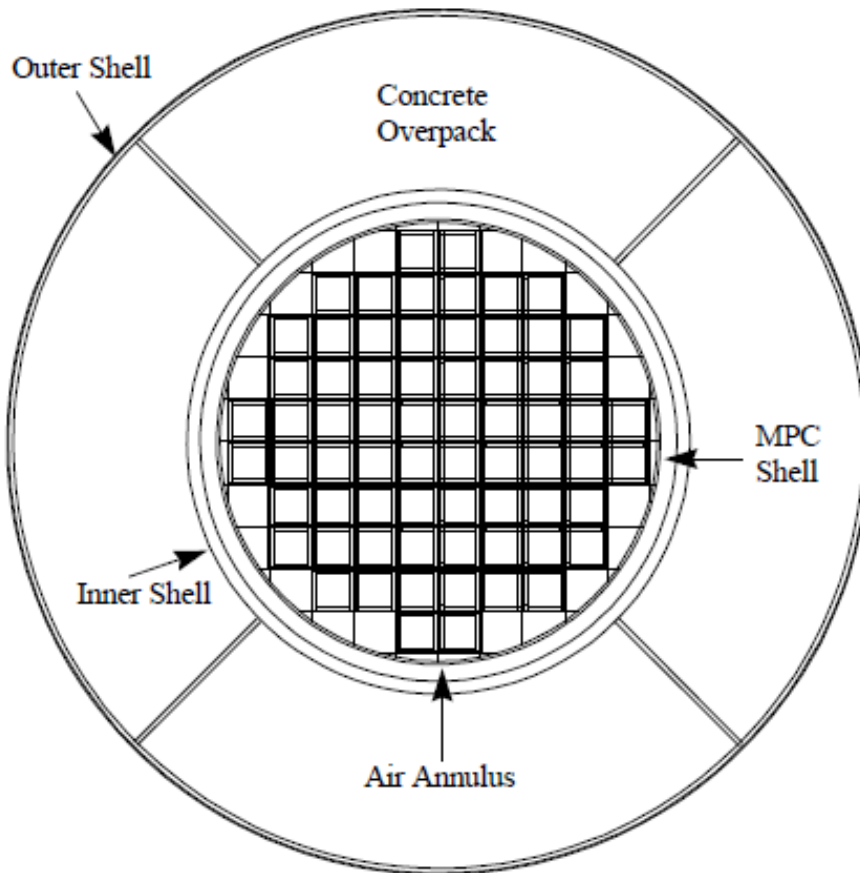
**FIGURE 5.3.1: HI-STORM 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>**

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.



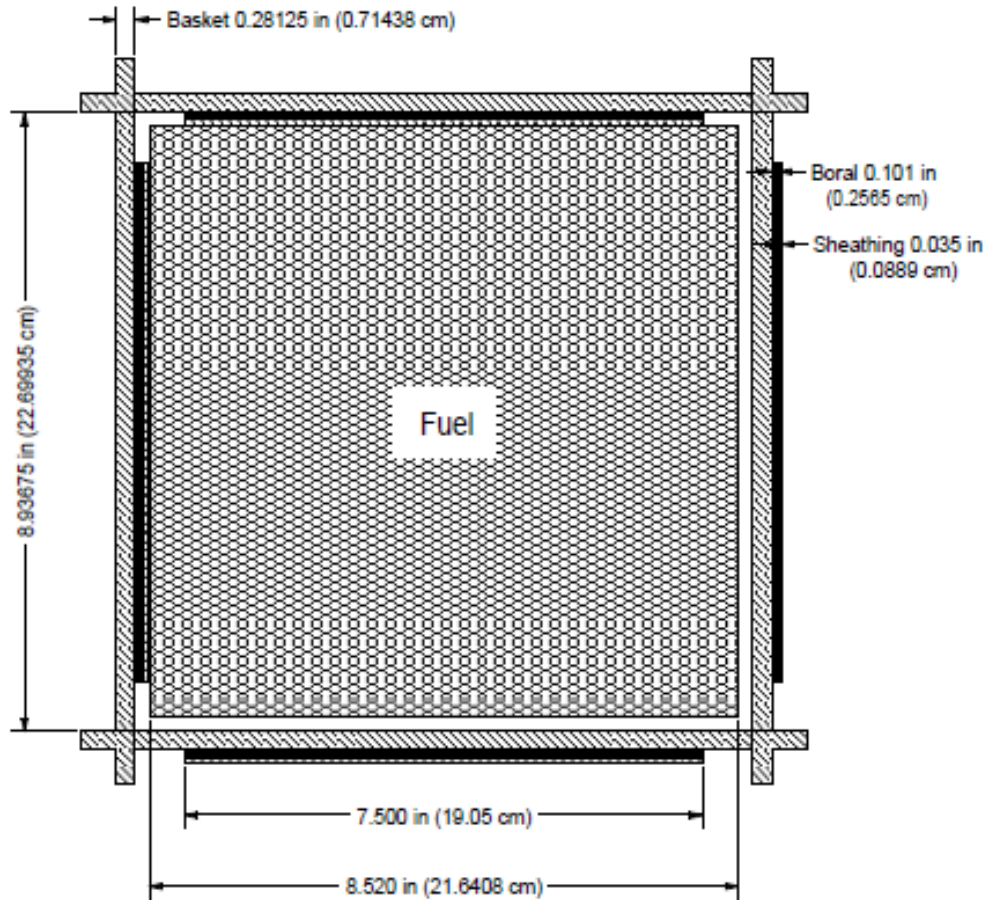
**FIGURE 5.3.2: HI-STORM 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>**

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

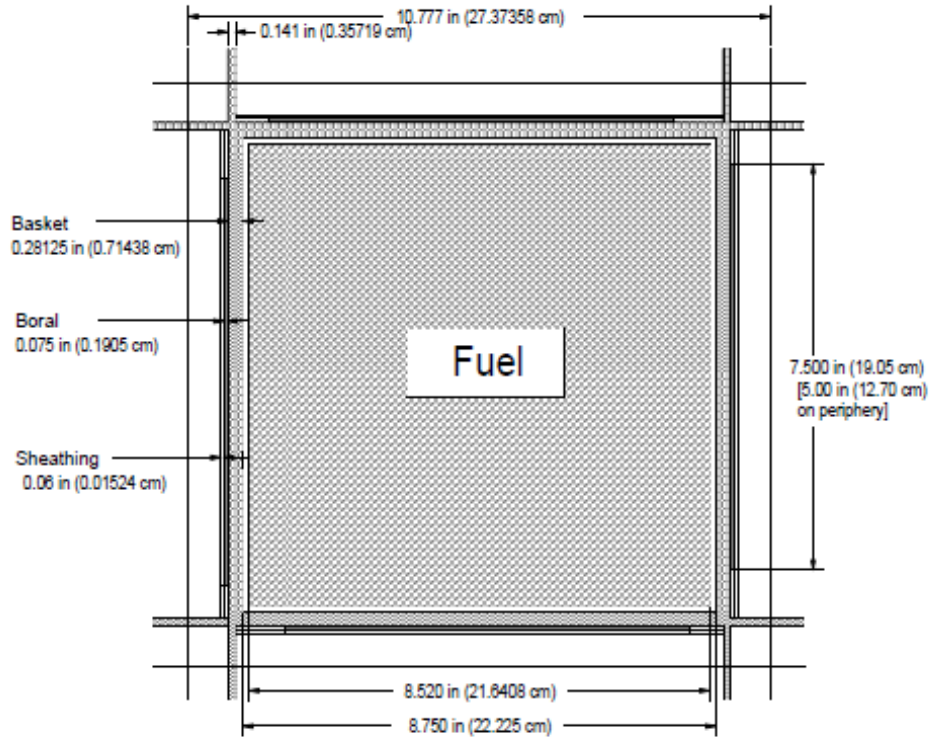


**FIGURE 5.3.3: HI-STORM 100 OVERPACK WITH MPC-68 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>**

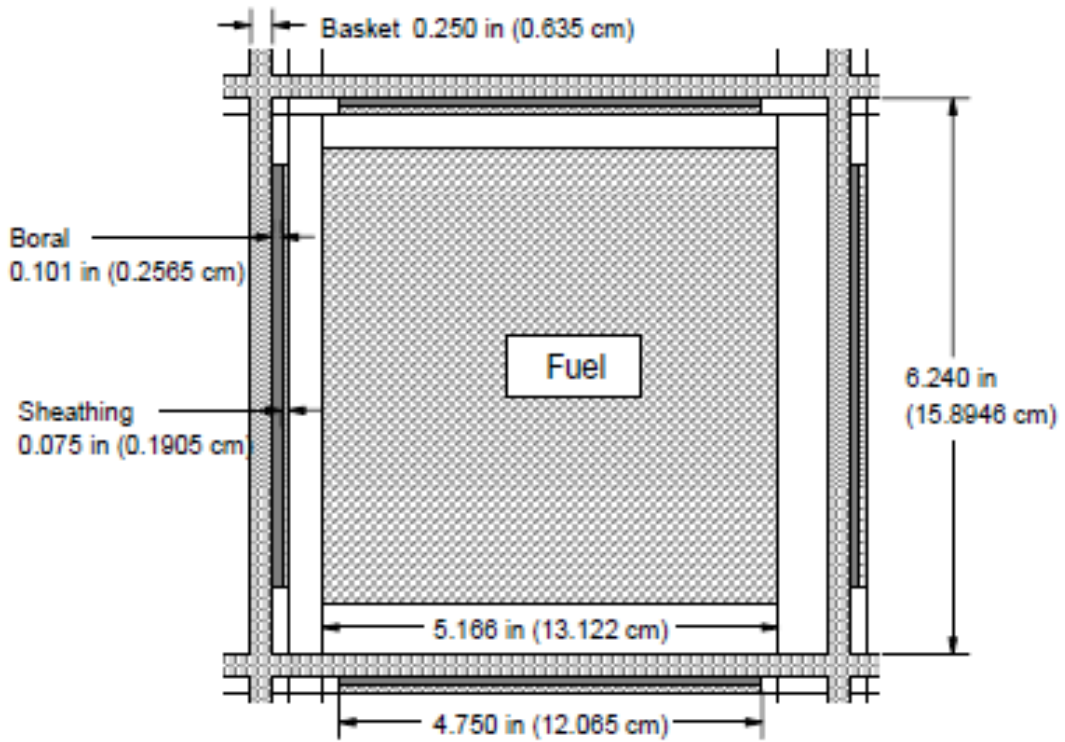
<sup>†</sup> This figure is drawn to scale using the MCNP plotter.



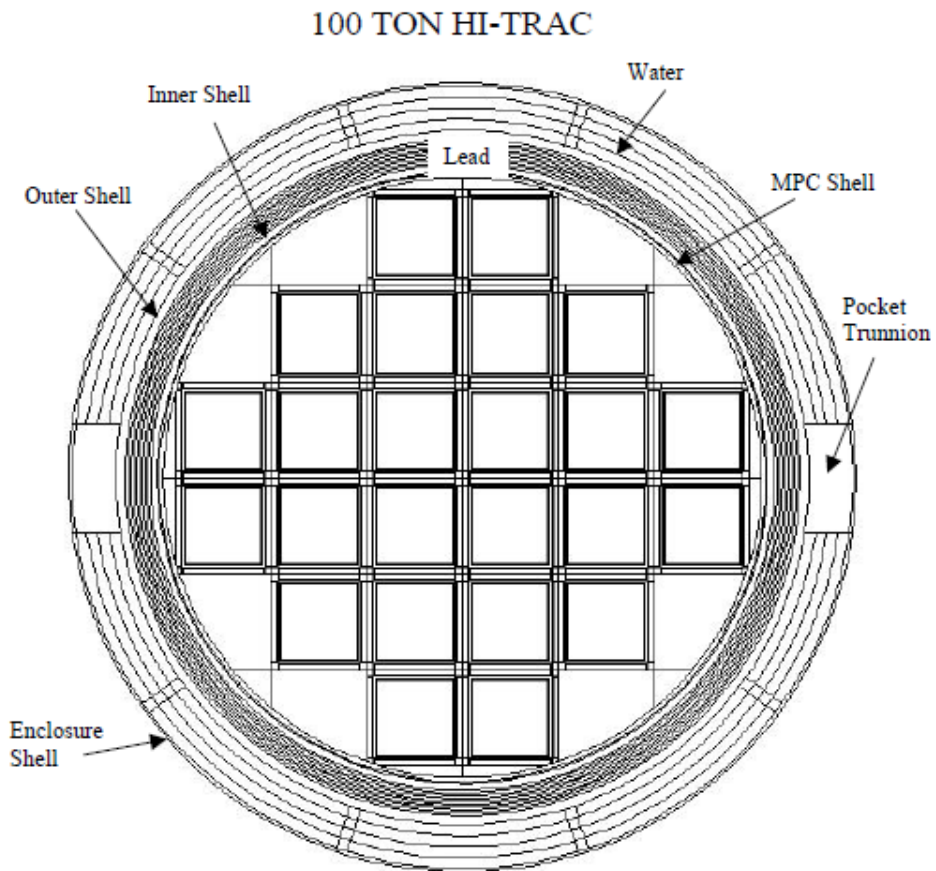
**FIGURE 5.3.4: CROSS SECTIONAL VIEW OF AN MPC-32 BASKET CELL AS MODELED IN MCNP**



**FIGURE 5.3.5: CROSS SECTIONAL VIEW OF AN MPC-24 BASKET CELL AS MODELED IN MCNP**



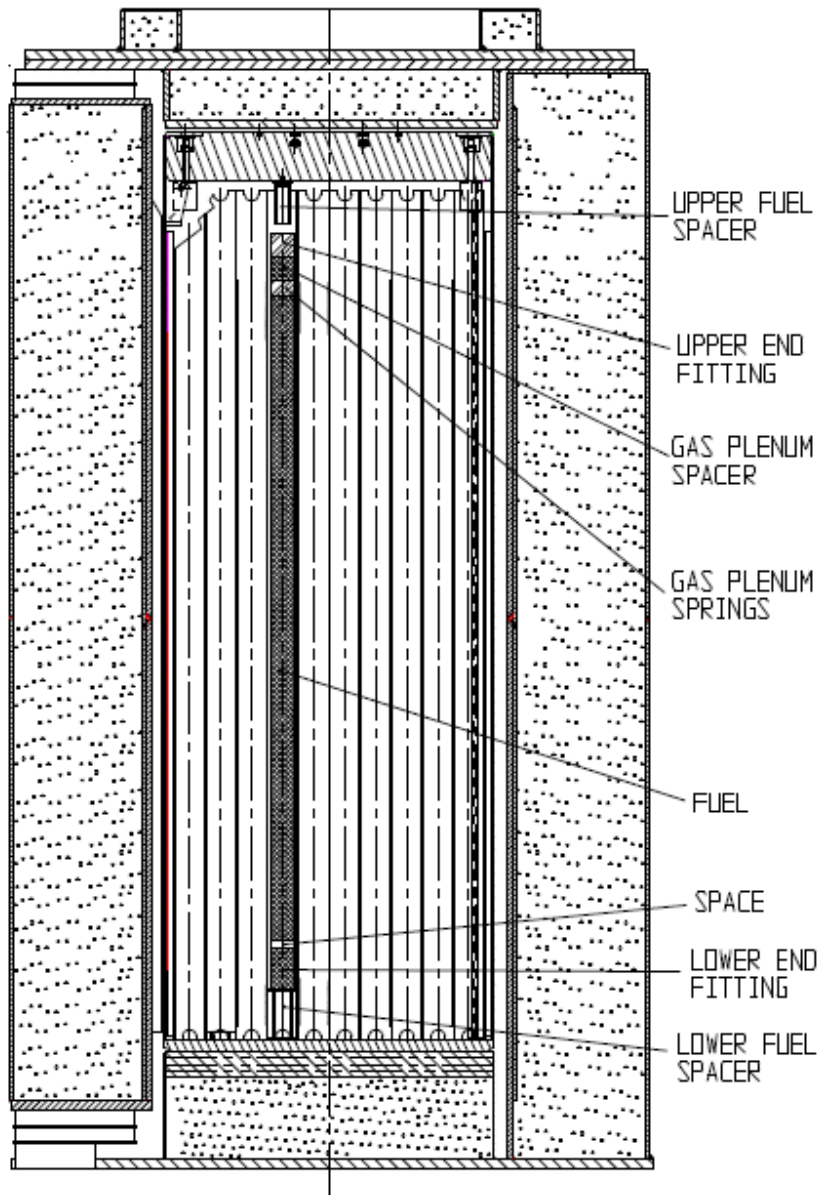
**FIGURE 5.3.6: CROSS SECTIONAL VIEW OF AN MPC-68 BASKET CELL AS MODELED IN MCNP**



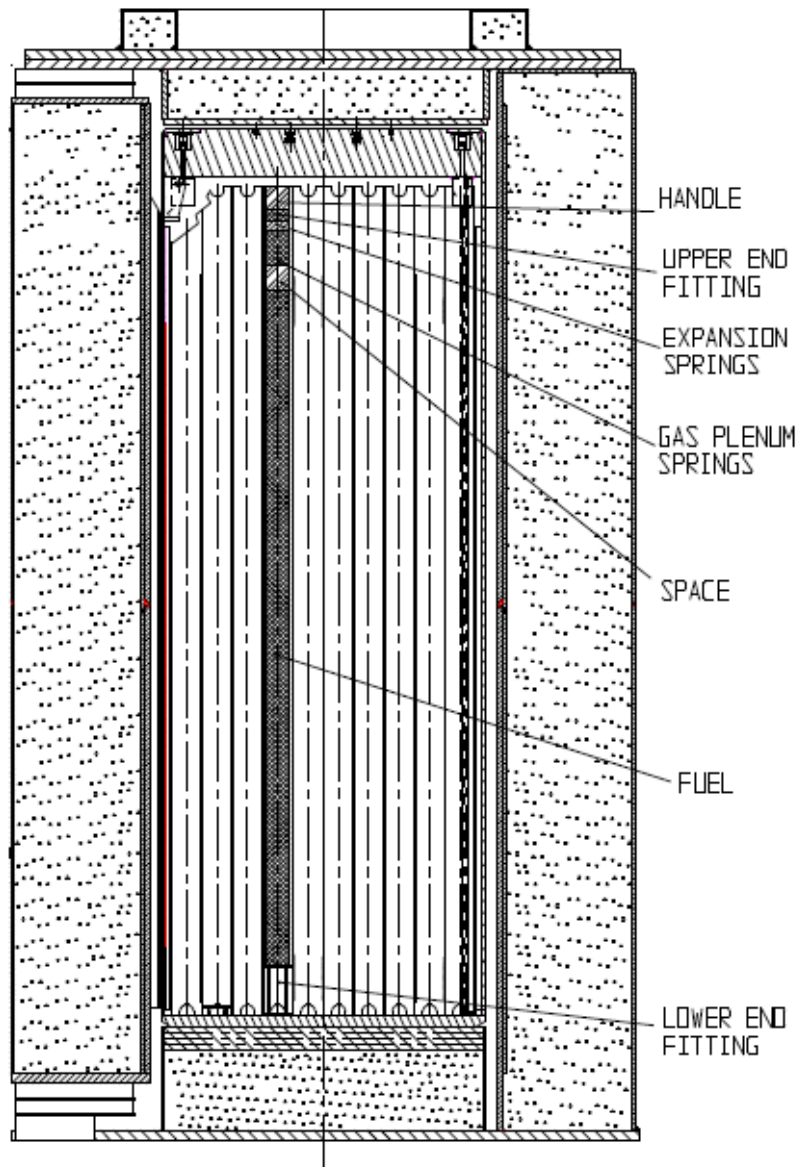
**FIGURE 5.3.7: HI-TRAC OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELED IN MCNP<sup>†</sup>**

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

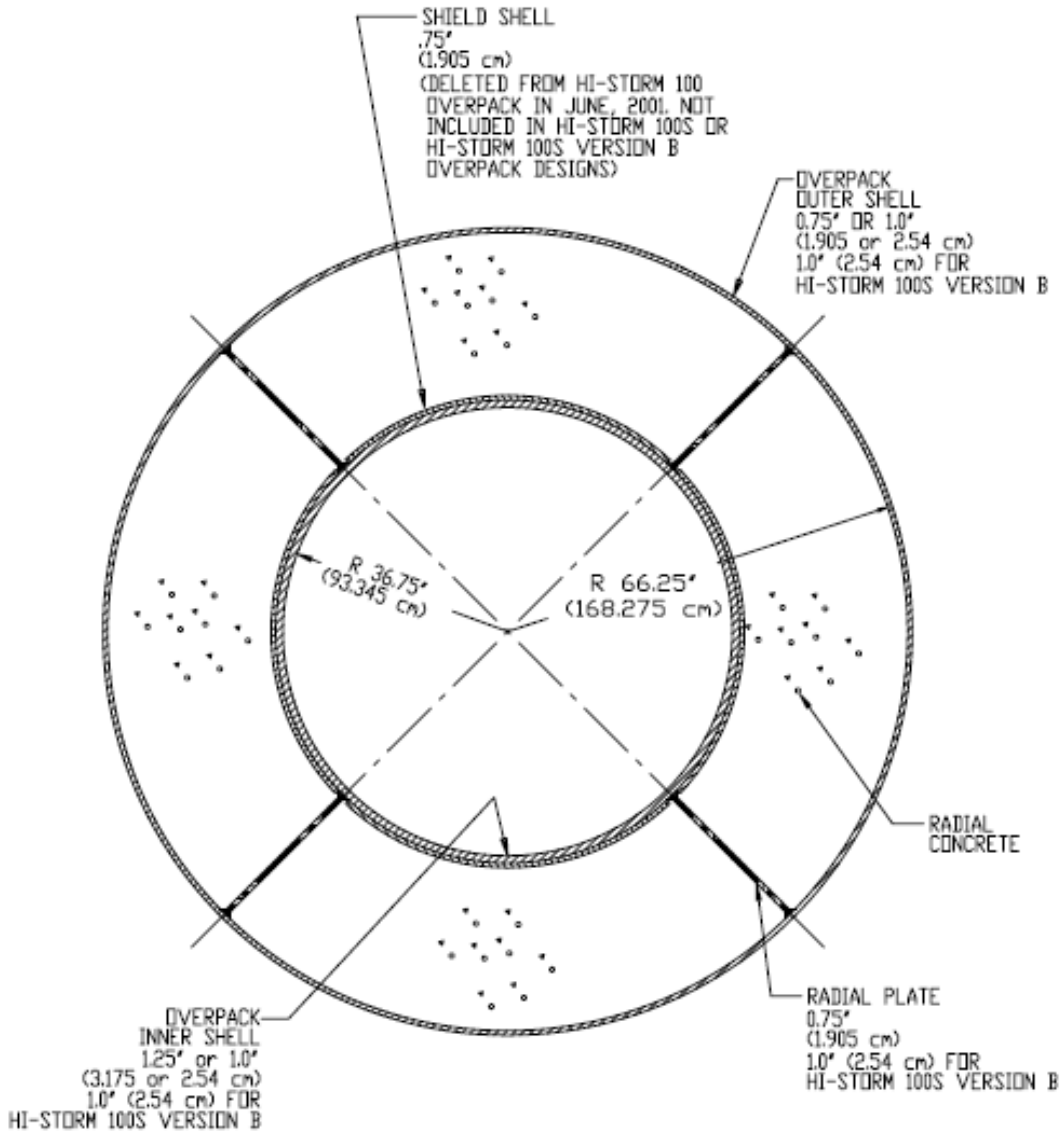




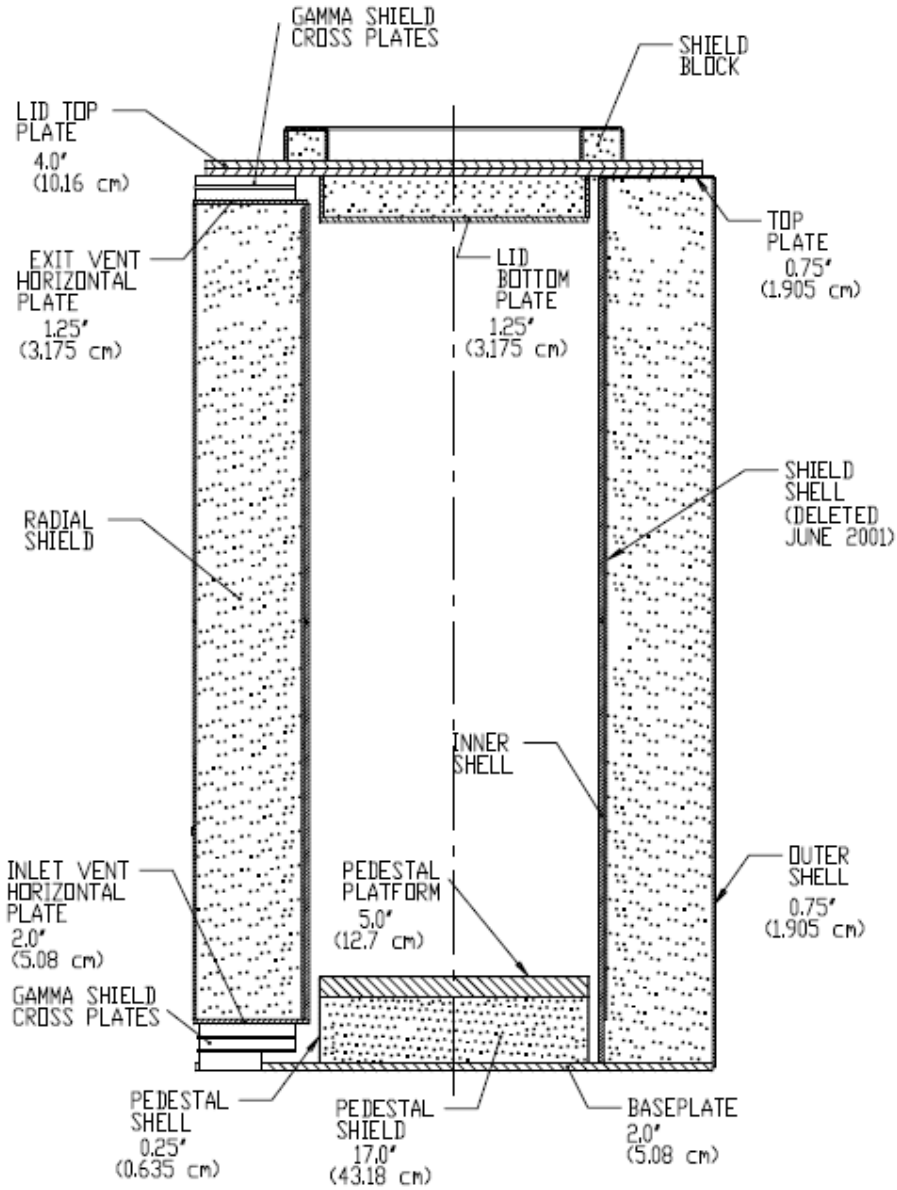
**FIGURE 5.3.8: AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM OVERPACK**



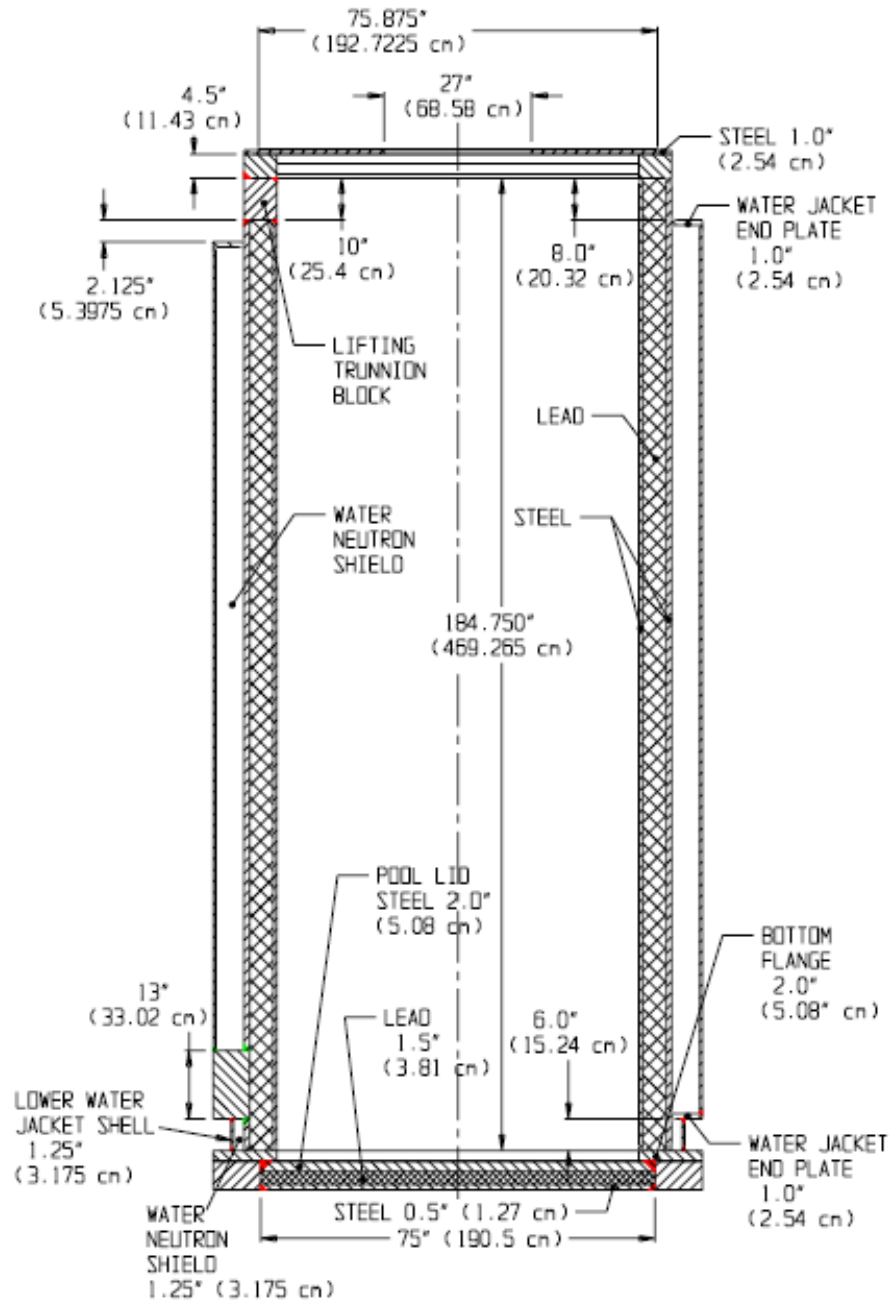
**FIGURE 5.3.9: AXIAL LOCATION OF BWR DESIGN FUEL IN THE HI-STORM OVERPACK**



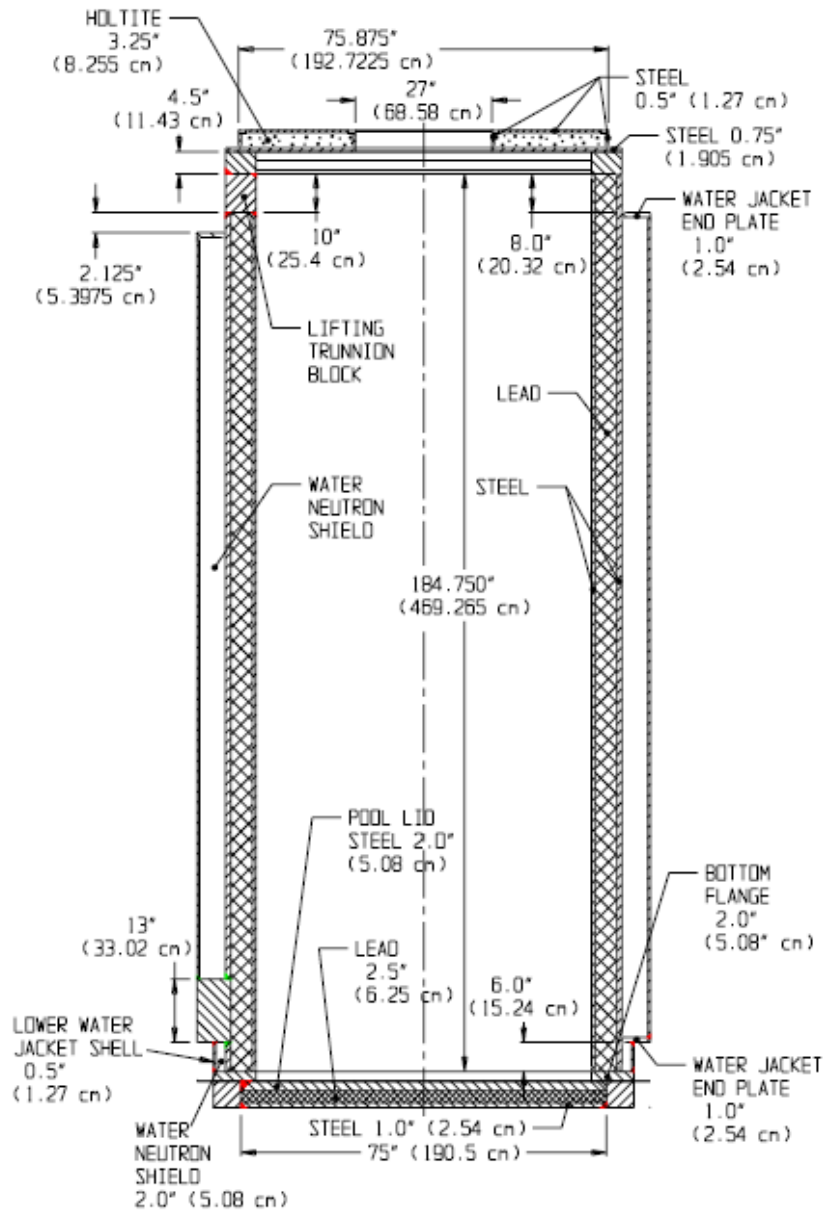
**FIGURE 5.3.10: CROSS SECTION OF HI-STORM OVERPACK**



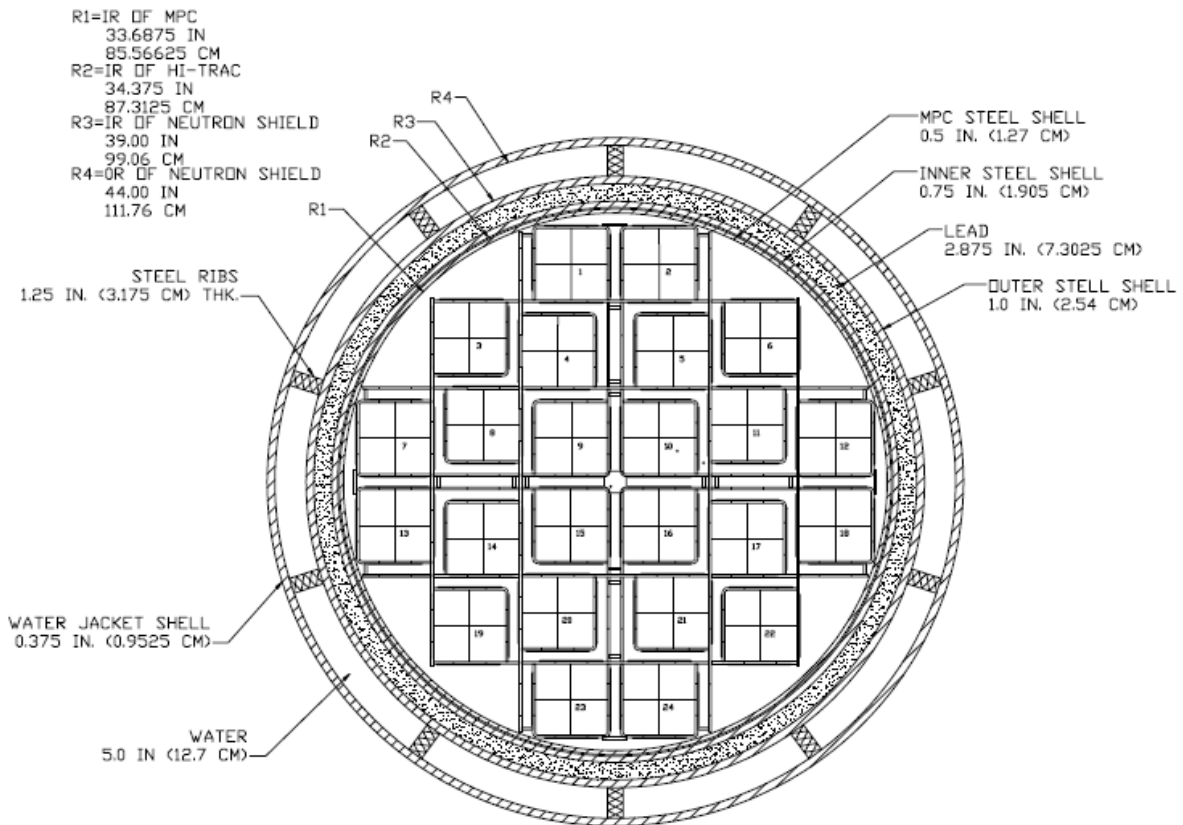
**FIGURE 5.3.11: HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW**



**FIGURE 5.3.12: 100-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW (AS MODELED)**

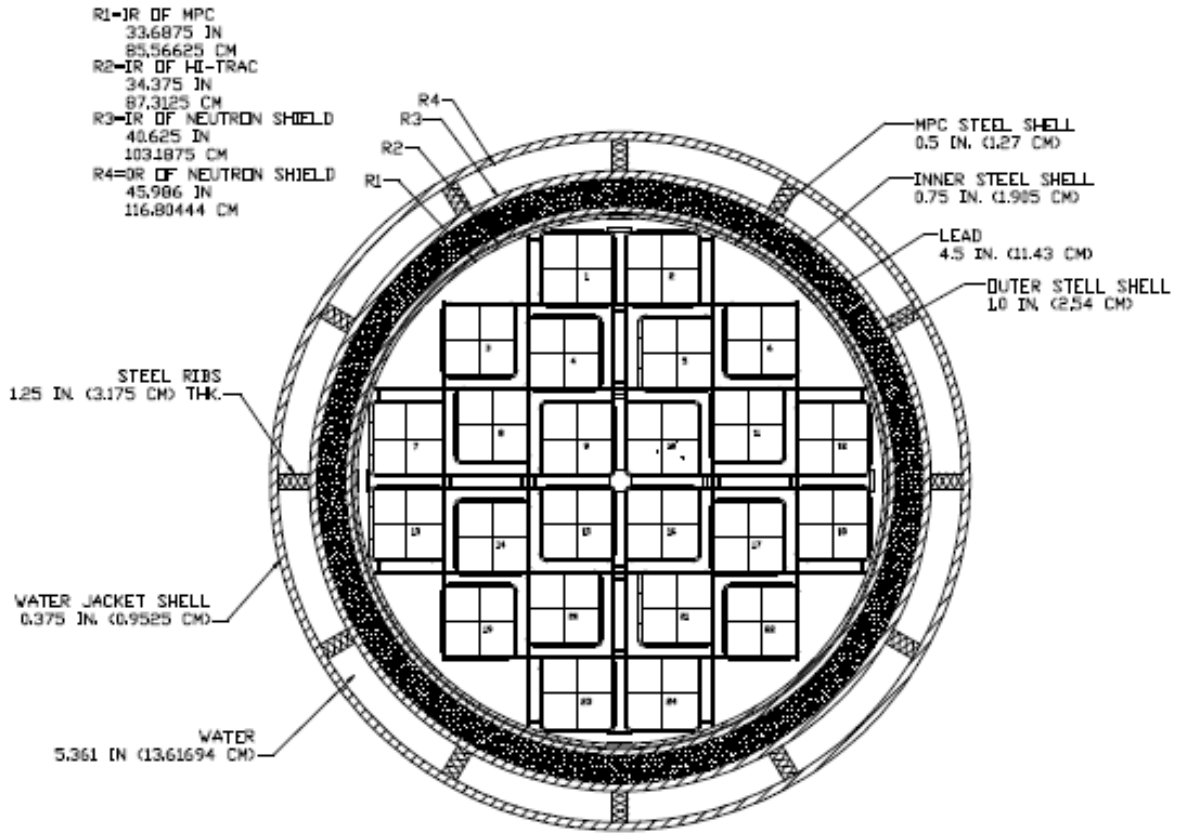


**FIGURE 5.3.13: 125-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW (AS MODELED)**



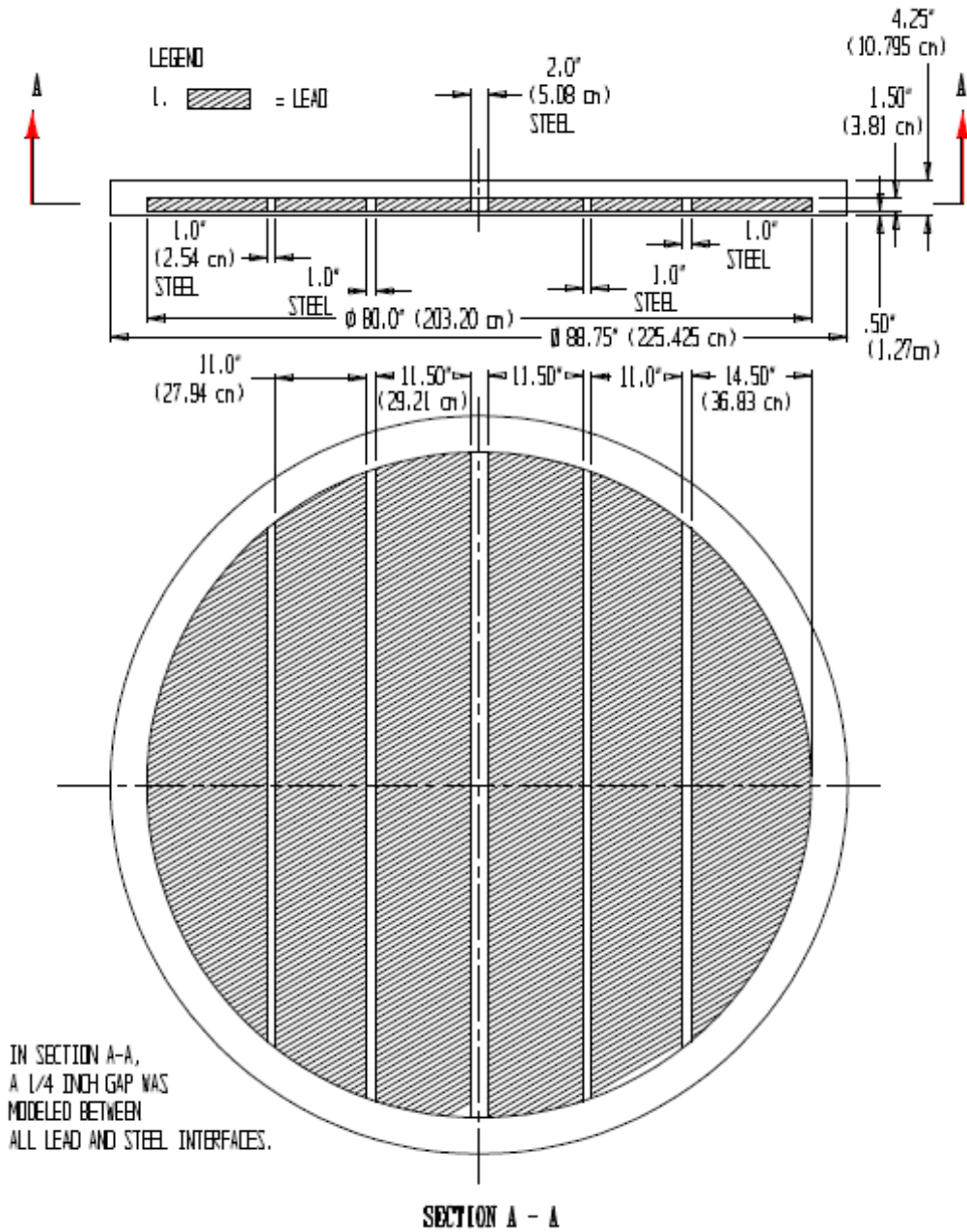
Note: The HI-TRAC 100 has 10 steel ribs as shown. The HI-TRAC 100D has 8 steel ribs evenly spaced with thickness as shown.

**FIGURE 5.3.14: HI-TRAC 100 AND 100D TRANSFER CASK CROSS SECTIONAL VIEW (AS MODELED)**



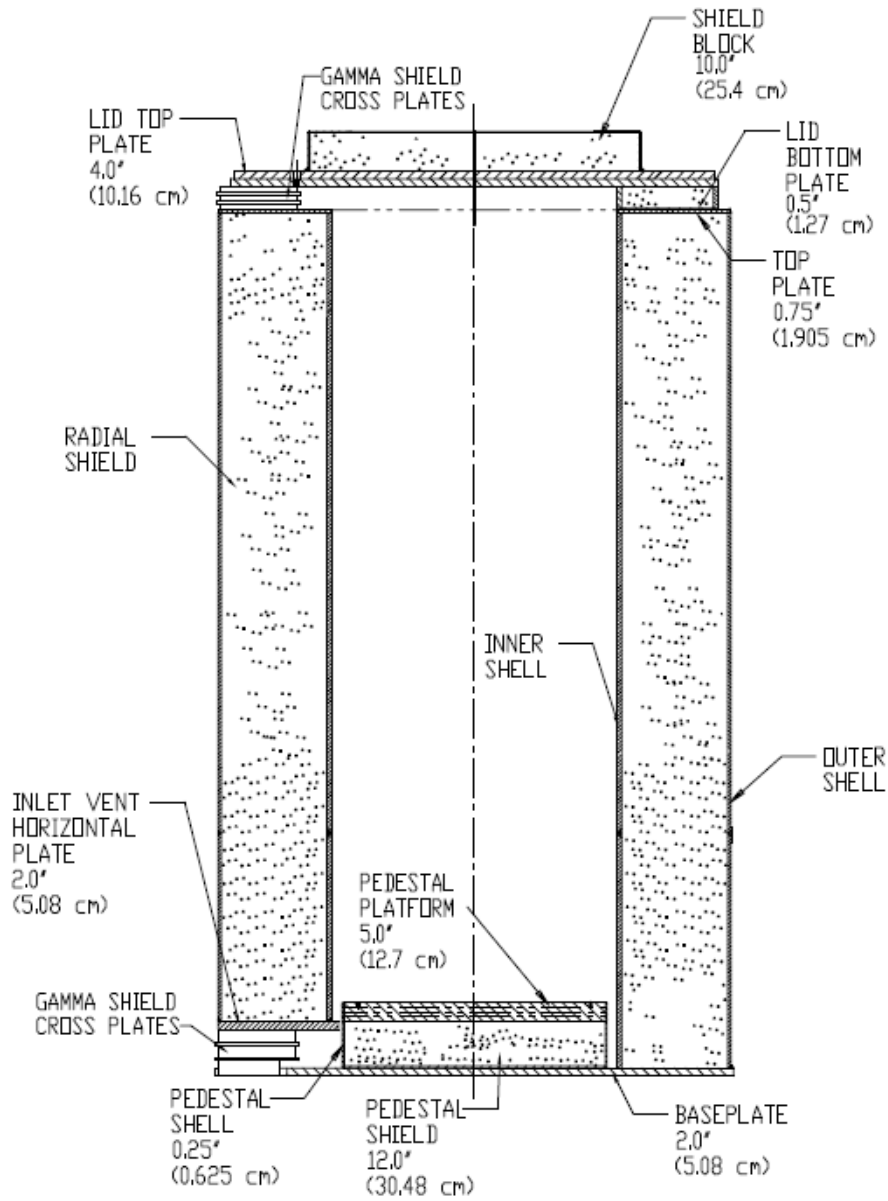
**FIGURE 5.3.15: HI-TRAC 125 TRANSFER CASK CROSS SECTIONAL VIEW (AS MODELED)**



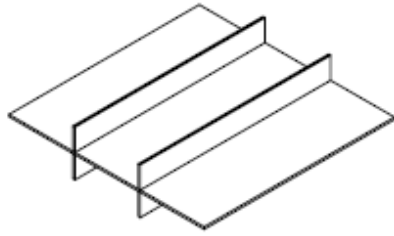


**FIGURE 5.3.16: 100-TON HI-TRAC TRANSFER LID (AS MODELED)**

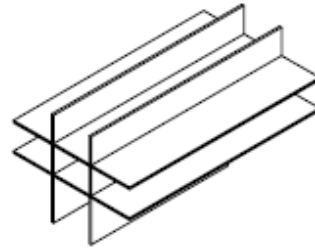




**FIGURE 5.3.18: HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW**



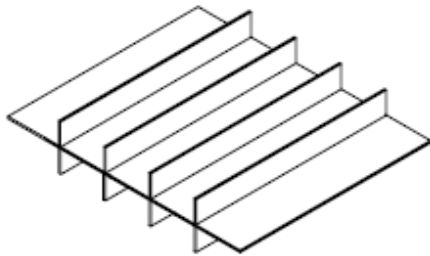
OUTLET VENT



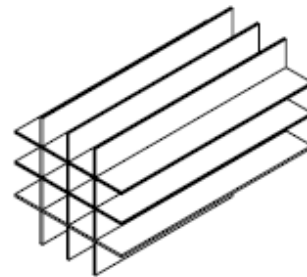
INLET VENT

MANDATORY GAMMA SHIELD CROSS PLATES FOR HI-STORM 100  
AND HI-STORM 100S

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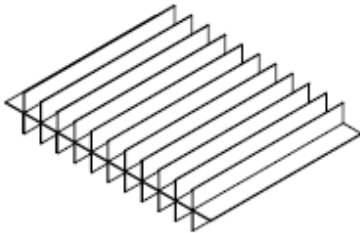
OUTLET VENT



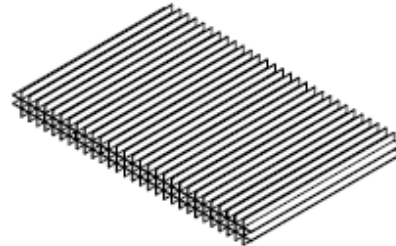
INLET VENT

OPTIONAL GAMMA SHIELD CROSS PLATES FOR HI-STORM 100S  
THAT MAY BE USED INSTEAD OF THE MANDATORY DEVICES.

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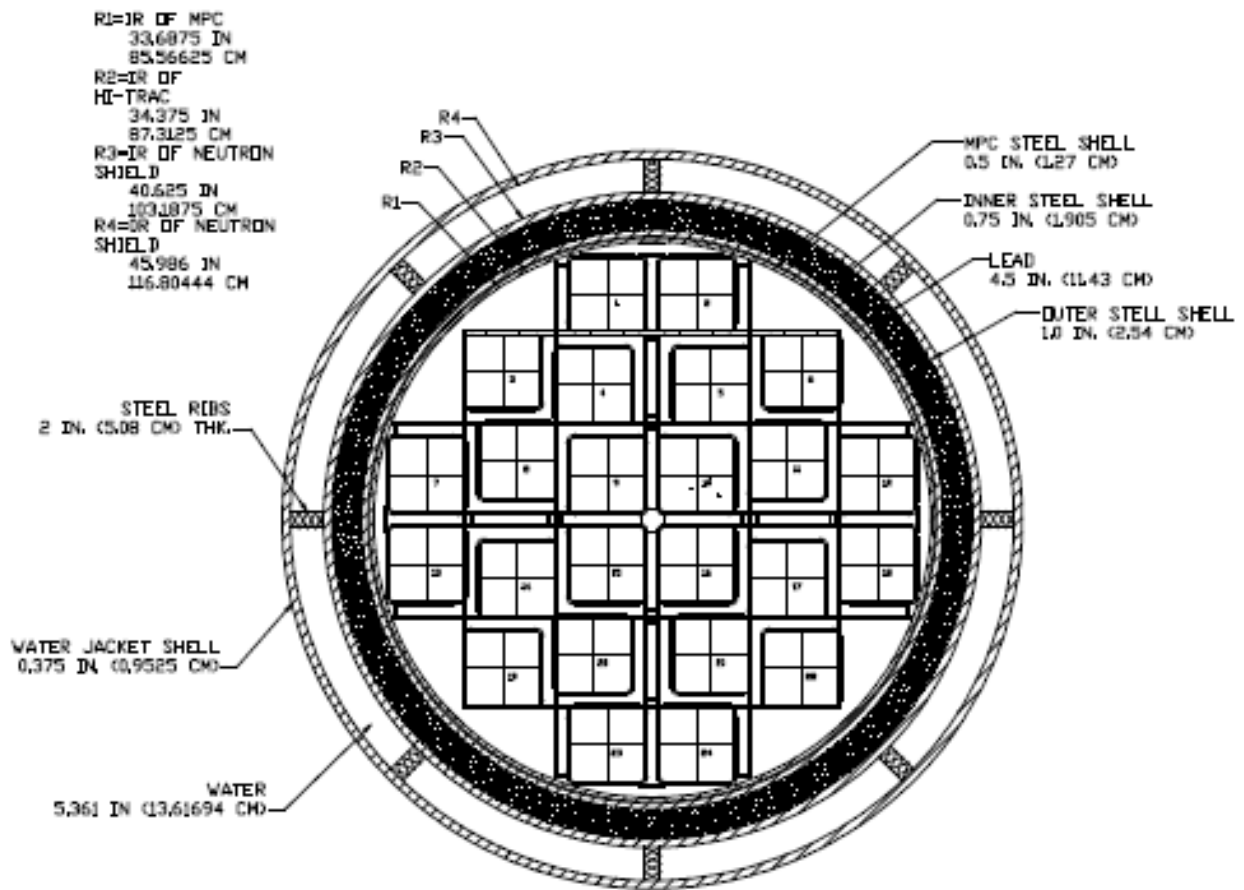
OUTLET VENT



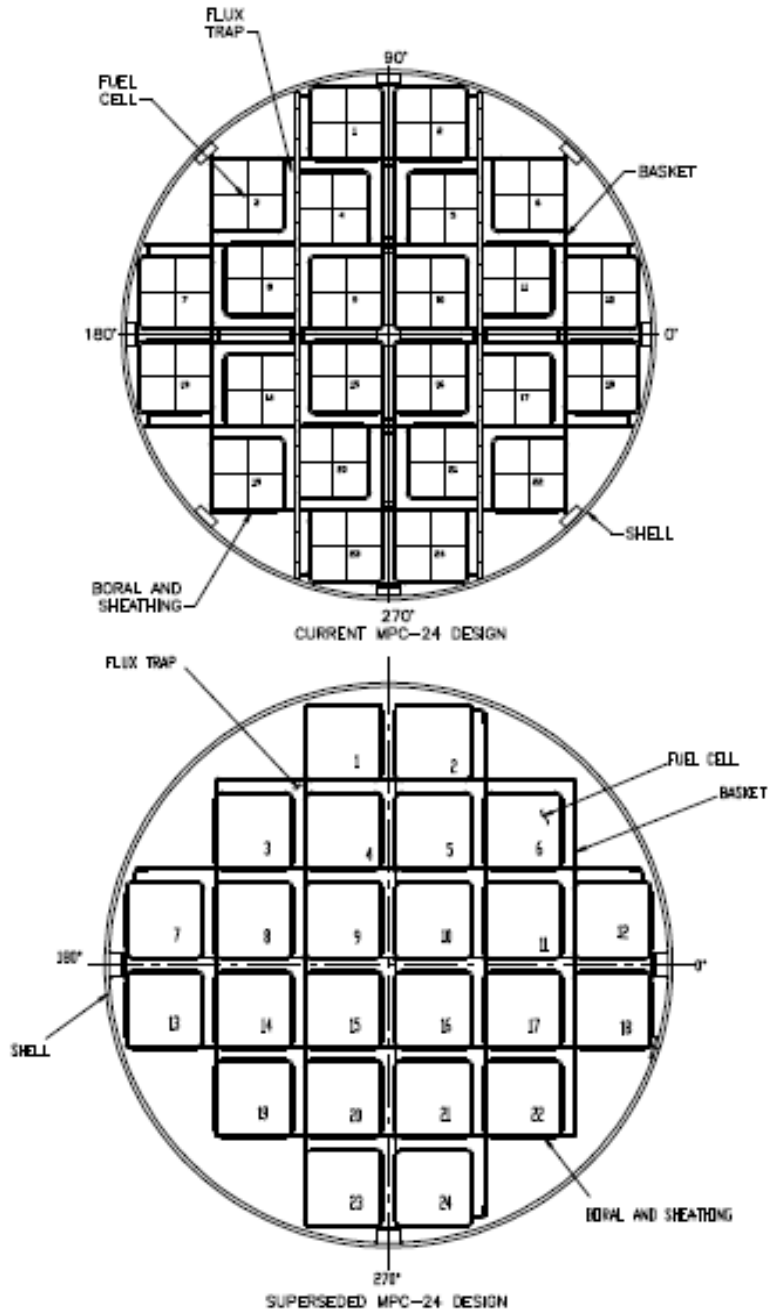
INLET VENT

MANDATORY GAMMA SHIELD CROSS PLATES FOR HI-STORM 100S VERSION B

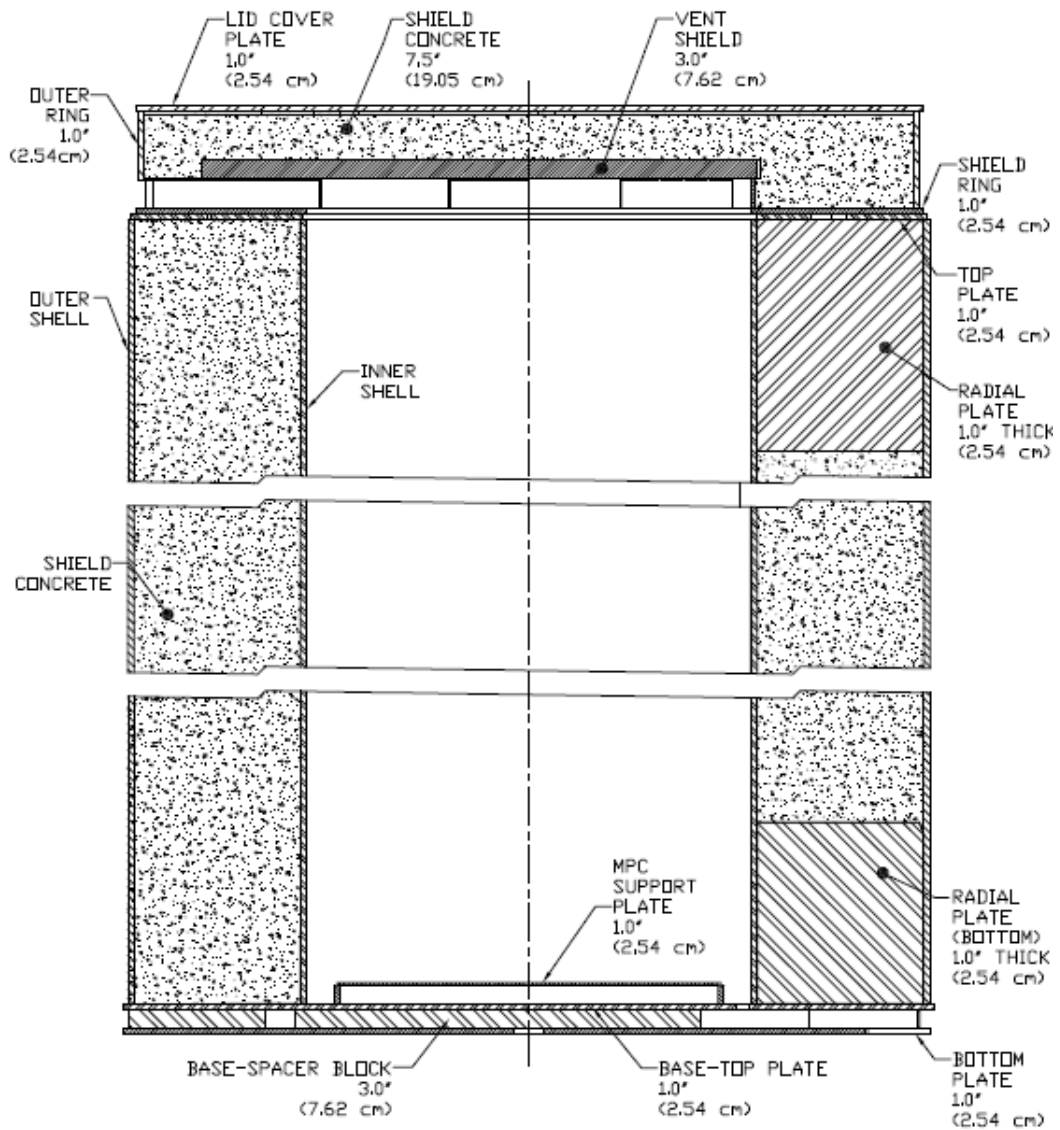
**FIGURE 5.3.19: GAMMA SHIELD CROSS PLATE CONFIGURATION OF  
HI-STORM 100, HI-STORM 100S, AND HI-STORM 100S VERSION B**



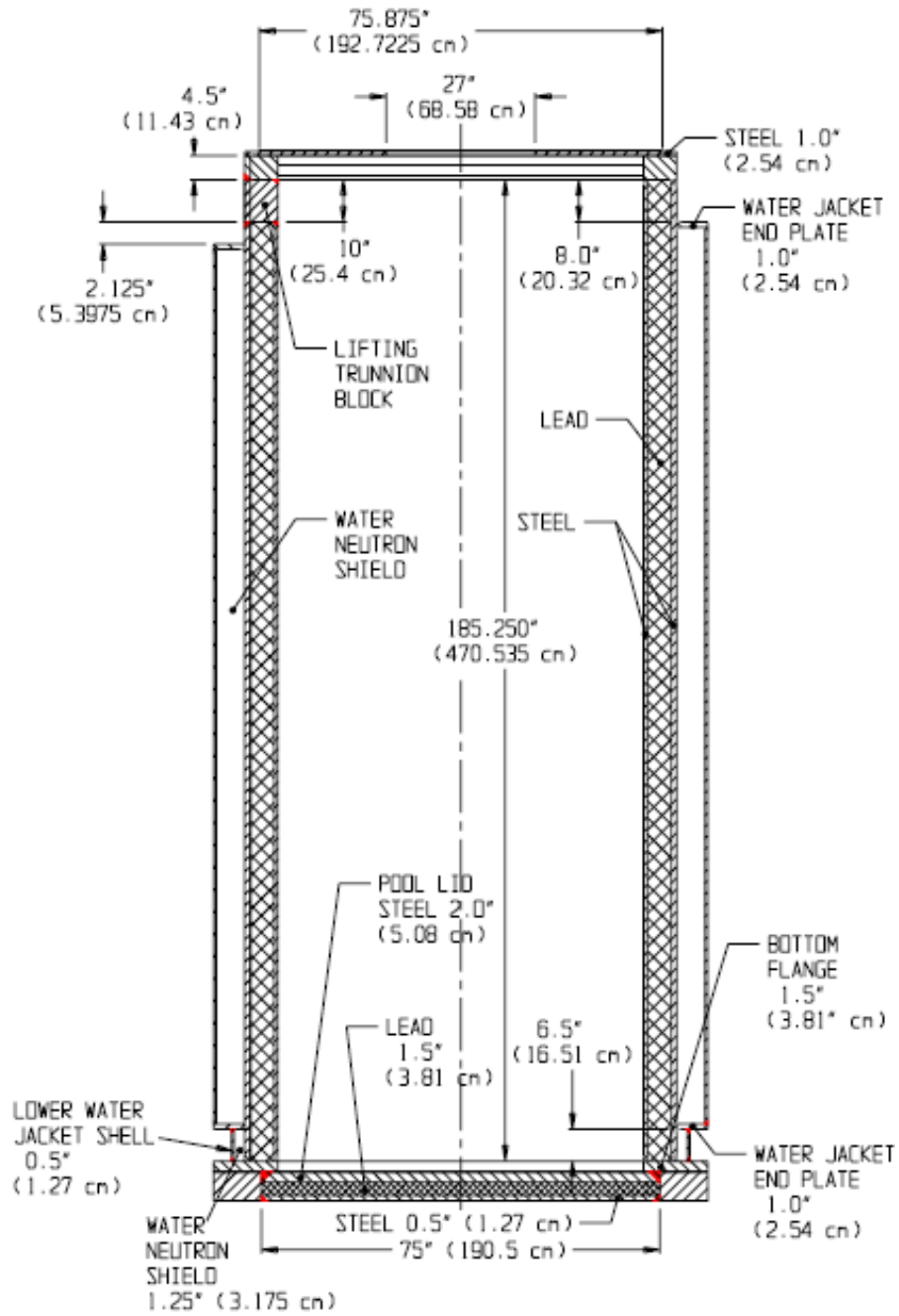
**FIGURE 5.3.20: HI-TRAC 125D TRANSFER CASK CROSS SECTIONAL VIEW (AS MODELED)**



**FIGURE 5.3.21: CROSS SECTIONAL VIEWS OF THE CURRENT MPC-24 DESIGN AND THE SUPERSEDED MPC-24 WHICH IS USED IN THE MCNP MODELS**



**FIGURE 5.3.22: HI-STORM 100S VERSION B OVERPACK CROSS SECTIONAL ELEVATION VIEW**



**FIGURE 5.3.23: HI-TRAC 100D TRANSFER CASK WITH POOL LID CORSS SECTIONAL ELEVATION VIEW (AS MODELED)**



## 5.4 SHIELDING EVALUATION

The MCNP-4A code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and  $^{60}\text{Co}$ ). The axial distribution of the fuel source term is described in Table 2.1.11 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The  $^{60}\text{Co}$  source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.11 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ( $1.105^{4.2}/1.105$ ) and 76.8% ( $1.195^{4.2}/1.195$ ) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel, respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.C. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

The dose rates at the various locations were calculated with MCNP using a two step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in the fuel and each axial location in the end fittings. The second and last step was to multiply the dose rate per starting particle for each group or starting location by

the source strength (i.e. particles/sec) in that group or location and sum the resulting dose rates for all groups in each dose location. The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location.

The HI-STORM shielding analysis was performed for conservative burnup and cooling time combinations which bound the uniform and regionalized loading specifications for zircaloy clad fuel specified in Section 2.1.9. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

Tables 5.1.11 through 5.1.13 provide the maximum dose rates adjacent to the HI-STORM overpack during normal conditions for each of the MPCs. Tables 5.1.14 through 5.1.16 provide the maximum dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Sections 5.1.1 and 5.1.2.

Tables 5.1.7 and 5.1.8 provide dose rates for the 100-ton and 125-ton HI-TRAC transfer casks, respectively, with the MPC-24 loaded with design basis fuel in the normal condition, in which the MPC is dry and the HI-TRAC water jacket is filled with water. Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with the water jacket filled with water (condition in which welding operations are performed). Dose locations 4 and 5, which are on the top and bottom of the HI-TRAC were not calculated at the one-meter distance for these configurations. For the conditions involving a fully flooded MPC, the internal water level was 10 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.7 indicates that the dose rates in the upper and lower portions of the HI-TRAC are reduced by about 50% with the water in the MPC. The dose at the center of the HI-TRAC is reduced by approximately 50% when there is also water in the water jacket and is essentially unchanged when there is no water in the water jacket as compared to the normal condition results shown in Table 5.1.7.

The burnup and cooling time combination of 60,000 MWD/MTU and 3 years was selected for the 100-ton MPC-24 HI-TRAC analysis because this combination of burnup and cooling time results in the highest dose rates, and therefore, bounds all other requested combinations in the 100-ton HI-TRAC. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results in Table 5.1.7 clearly indicate that gammas are the dominant portion of the total dose rate. Therefore, as the burnup and cooling time increase, the reduction in the gamma dose rate due to the increased cooling time results in a net decrease in the total dose rate.

In contrast, the dose rates surrounding the HI-TRAC 125 and 125D transfer casks have significantly higher neutron component. Therefore, the dose rates at 75,000 MWD/MTU burnup

and 5 year cooling are higher than the dose rates at 60,000 MWD/MTU burnup and 3 year cooling. The dose rates for the 125-ton HI-TRACs with the MPC-24 at 75,000 MWD/MTU and 5 year cooling are listed in Table 5.1.8 of Section 5.1.

Table 5.4.9 provides dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-68 at a burnup and cooling time combination of 50,000 MWD/MTU and 3 years. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-68 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-68 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Table 5.4.11 provides dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-32 at burnup and cooling time combination of 45,000 MWD/MTU and 3 years. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top of the 100-ton HI-TRAC are somewhat higher in the MPC-32 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-32 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Table 5.4.19 provides dose rates adjacent to and one meter away from the radial surface of the HI-TRAC 100D with the MPC-32 at a burnup of 35,000 MWD/MTU and a cooling time of 3 years. Results are presented only for dose locations 1 through 3 since the differences between the HI-TRAC 100 and the 100D will only affect the dose rates on the side of the transfer cask. A comparison of these results to those provided in Table 5.4.11 indicates that the dose rates at 1 meter from the transfer cask are very similar to the dose rates for the 100-ton HI-TRAC.

As mentioned in Section 5.0, all MPCs offer a regionalized loading pattern as described in Section 2.1.9. This loading pattern authorizes fuel of higher decay heat than uniform loading (i.e. higher burnups and shorter cooling times) to be stored in either the center region, region 1, of the MPC or the outer region, region 2. From a shielding perspective, placing the older fuel on the outside provides shielding for the inner fuel in the radial direction. Note that for the MPC-24 there are some inner region assemblies that have a corner area with a direct line of sight to the MPC shell and therefore the outer region will not provide quite as much additional shielding over the whole surface area of the cask as it does for the MPC-32 and MPC-68. Based on analysis for the MPC-32 and MPC-68 using the same burnup and cooling times in region 1 and 2 the following percentages were calculated for dose location 2 on the 100-ton HI-TRAC.

- Approximately 21% and 27% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. Region 1 contains 12 (38% of total) and 32 (47% of total) assemblies in the MPC-32 and MPC-68, respectively. For the MPC-24, where region 1 contains 12 (50% of total) assemblies, the contribution would be similar.
- Approximately 1% and 2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. For the MPC-24, where some corners of the region 1 assemblies are not completely shielded by the outer assemblies, the contribution of the photon dose from region 1 would be slightly larger than for the MPC-32 and MPC-68, but the gamma dose rates would still be dominated by the outer assemblies.

These results clearly indicate that the outer fuel assemblies shield almost the entire gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. Note that for the MPC-24 there may be localized higher dose rates using regionalized loading, since the inner region is not totally surrounded by the outer region. However, the dose rates would always be bounded by the values presented in this chapter. In the axial direction, regionalized loading with higher burnup fuel on the inside results in higher dose rates in the center portion of the cask since the region 2 assemblies are not shielding the region 1 assemblies for axial dose locations.

Note that the regionalized loading scheme also allows placing higher burned or shorter cooled assemblies on the periphery of the basket. In this case, dose rate would be closer to the bounding values presented here. This configuration should only be used if it is not feasible to place such assemblies in the center of the cask.

Burnup and cooling time combinations which bound both regionalized loading and uniform loading patterns were analyzed. Therefore, dose rates for specific regionalized loading patterns are not presented in this chapter. Section 5.4.9 provides a brief additional discussion on regionalized loading dose rates.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

#### 5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations

The HI-STORM 100 overpack and the HI-TRAC utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose. In addition to the fins, the pocket trunnions in the HI-TRAC 100 and 125 are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Analysis of the pocket trunnions in the HI-TRAC 100 and 125 and the steel fins in the HI-TRAC 100, 125, and 125D indicate that neutron streaming is noticeable at the surface of the transfer cask. The neutron dose rate on the surface of the pocket trunnion is approximately 5 times higher than the circumferential average dose rate at that location. The gamma dose rate is approximately 10 times lower than the circumferential average dose rate at that location. The streaming at the rib location is the largest in the HI-TRAC 125D because the ribs are thicker than in the HI-TRAC 100 or 125. The neutron dose rate on the surface of the rib in the 125D is approximately 3 times higher than the circumferential average dose rate at that location. The gamma dose rate on the surface of the rib in the 125D is approximately 3 times lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins and the pocket trunnions compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask only surface average dose rates are reported in this chapter.

Below each lifting trunnion, there is a localized area where the water jacket has been reduced in height by 4.125 inches to accommodate the lift yoke (see Figures 5.3.12 and 5.3.13). This area experiences a significantly higher than average dose rate on contact of the HI-TRAC. The peak dose in this location is 2.9 Rem/hr for the MPC-32, 2.0 Rem/hr for the MPC-68 and 2.4 Rem/hr for the MPC-24 in the 100-ton HI-TRAC and 1.7 Rem/hr for the MPC-24 in the HI-TRAC 125D. At a distance of 1 to 2 feet from the edge of the HI-TRAC the localized effect is greatly reduced. This dose rate is acceptable because during lifting operations the lift yoke will be in place, which, due to the additional lift yoke steel (~3 inches), will greatly reduce the dose rate. However, more importantly, people will be prohibited from being in the vicinity of the lifting trunnions during lifting operations as a standard rigging practice. In addition the lift yoke is remote in its attachment and detachment, further minimizing personnel exposure. Immediately following the detachment of the lift yoke, in preparation for closure operations, temporary shielding may be

placed in this area. Any temporary shielding (e.g., lead bricks, water tanks, lead blankets, steel plates, etc.) is sufficient to attenuate the localized hot spot. The operating procedure in Chapter 8 discusses the placement of temporary shielding in this area. For the 100-ton HI-TRAC, the optional temporary shield ring will replace the water that was lost from the axial reduction in the water jacket thereby eliminating the localized hot spot. When the HI-TRAC is in the horizontal position, during transport operations, it will (at a minimum) be positioned a few feet off the ground by the transport vehicle and therefore this location below the lifting trunnions will be positioned above people which will minimize the effect on personnel exposure. In addition, good operating practice will dictate that personnel remain at least a few feet away from the transport vehicle. During vertical transport of a loaded HI-TRAC, the localized hot spot will be even further from the operating personnel. Based on these considerations, the conclusion is that this localized hot spot does not significantly impact the personnel exposure.

#### 5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

##### 5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed that the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Chapter 1, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Dividing the total fuel gamma source for a 6x6 fuel assembly in Table 5.2.7 by the 80 inch rubble height provides a gamma source per inch of  $3.41\text{E}+12$  photon/s. Dividing the total neutron source for a 6x6 fuel assembly in Table 5.2.18 by 80 inches provides a neutron source per inch of  $2.75\text{E}+05$  neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of  $1.08\text{E}+13$  photon/s and  $9.17\text{E}+05$  neutron/s, respectively, for a burnup and cooling time of 40,000 MWD/MTU and 5 years. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling by the active fuel length of 144 inches. Therefore, damaged Dresden 1 and Humboldt Bay fuel assemblies are bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates from Dresden 1 or Humboldt Bay fuel assemblies are provided as they are bounded by the intact fuel analysis.

#### 5.4.2.2 Generic PWR and BWR Damaged Fuel

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact. As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was performed to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Only the 100-ton HI-TRAC was analyzed because it can be concluded that if the dose rate change is not significant for the 100-ton HI-TRAC then the change will not be significant for the 125-ton HI-TRACs or the HI-STORM overpacks.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density which is higher than the fuel density for an intact fuel assembly. If the damaged fuel assembly was to fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. This scenario was analyzed with MCNP-4A in a conservative bounding fashion to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 (4 peripheral locations in the MPC-24E or MPC-24EF) and the MPC-68 (16 peripheral locations) with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

Tables 5.4.13 and 5.4.14 provide the results for the MPC-24 and MPC-68, respectively. Only the radial dose rates are provided since the axial dose rates will not be significantly affected because the damaged fuel assemblies are located on the periphery of the baskets. A comparison of these results to the results in Tables 5.1.7 and 5.4.9 indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase by less than 27% while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the bottom and top is due to the assumed flat power distribution. The dose rates shown in Tables 5.4.13 and 5.4.14 were averaged over the circumference of the cask. Since almost all of the peripheral cells in the MPC-68 are filled with DFCs, an azimuthal variation would not be expected for the MPC-68. However, since

there are only 4 DFCs in the MPC-24E, an azimuthal variation in dose due to the damaged fuel/fuel debris might be expected. Therefore, the dose rates were evaluated in four smaller regions, one outside each DFC, that encompass about 44% of the circumference. There was no significant change in the dose rate as a result of the localized dose calculation. These results indicate that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter.

The MPC-32 was not explicitly analyzed for damaged fuel or fuel debris in this chapter. However, based on the analysis described above for the MPC-24 and the MPC-68, it can be concluded that the shielding performance of the MPC-32 will not be significantly affected by the storage of damaged fuel.

#### 5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single HI-STORM overpack loaded with an MPC-24 and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters were evaluated with MCNP. In the model, the casks were placed on an infinite slab of dirt to account for earth-shine effects. The atmosphere was represented by dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 700 meters. This is more than sufficient to properly account for skyshine effects. The models included either 500 or 1050 meters of air around the cask. Based on the behavior of the dose rate as a function of distance, 50 meters of air, beyond the detector locations, is sufficient to account for back-scattering. Therefore, the HI-STORM MCNP off-site dose models account for back scattering by including more than 50 meters of air beyond the detector locations for all cited dose rates. Since gamma back-scattering has an effect on the off-site dose, it is recommended that the site-specific evaluation under 10CFR72.212 include at least 50 to 100 meters of air, beyond the detector locations, in the calculational models.

The MCNP calculations of the off-site dose used a two-stage process. In the first stage a binary surface source file (MCNP terminology) containing particle track information was written for particles crossing the outer radial and top surfaces of the HI-STORM overpack. In the second stage of the calculation, this surface source file was used with the particle tracks originating on the outer edge of the overpack and the dose rate was calculated at the desired location (hundreds



of meters away from the overpack). The results from this two-stage process are statistically the same as the results from a single calculation. However, the advantage of the two-stage process is that each stage can be optimized independently.

The annual dose, assuming 100% occupancy (8760 hours), at 350 meters from a single HI-STORM 100S Version B cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed. This table indicates that the dose due to neutrons is 2.8 % of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques may not properly account for the neutron transmissions and could lead to low estimates of the site boundary dose.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM 100S Version B overpack was calculated at the distance desired. Dose value = A.
2. The annual dose from the radiation leaving the top of the HI-STORM 100S Version B overpack was calculated at the distance desired. Dose value = B.
3. The annual dose from the radiation leaving the side of a HI-STORM 100S Version B overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7 for the bounding burnup and cooling time of 60,000 MWD/MTU and 3-year cooling. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM 100S Version B overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

$$\text{Dose} = ZA + 2ZB + ZC$$

As an example, the dose from a 2x3 array at 450 meters is presented.

1. The annual dose from the side of a single cask: Dose A = 6.81
2. The annual dose from the top of a single cask: Dose B = 1.78E-2
3. The annual dose from the side of a cask positioned behind another cask:  
Dose C = 1.36

Using the formula shown above (Z=3), the total dose at 450 meters from a 2x3 array of HI-STORM overpacks is 24.62 mrem/year, assuming a 8760 hour occupancy.

An important point to notice here is that the dose from the side of the back row of casks is approximately 16 % of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis. Therefore, a site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

#### 5.4.4 Stainless Steel Clad Fuel Evaluation

Table 5.4.8 presents the dose rates at the center of the HI-STORM 100 overpack, adjacent and at one meter distance, from the stainless steel clad fuel. These dose rates, when compared to Tables 5.1.1 through 5.1.6, are similar to the dose rates from the design basis zircaloy clad fuel, indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STORM 100 overpack. Since the true active fuel length is shorter than 144 inches and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

#### 5.4.5 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the source terms for the BWR design basis zircaloy clad fuel assembly (GE 7x7) which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.22 by the 110 inch active fuel height provides a gamma source per inch of  $2.36\text{E}+12$  photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.23 by 110 inches provides a neutron source strength per inch of  $3.06\text{E}+5$  neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of  $1.08\text{E}+13$  photons/s and  $9.17\text{E}+5$  neutrons/s for 40,000 MWD/MTU and 5 year cooling. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling by the active fuel length of 144 inches. This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Section 5.4.2.1, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches. The resulting gamma and neutron source strengths are  $3.25\text{E}+12$  photons/s and  $4.21\text{E}+5$  neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

#### 5.4.6 Non-Fuel Hardware

As discussed in Section 5.2.4, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM 100 System. Since each of these devices occupy the same location within an assembly (i.e., the guide tubes), only one of these devices will be present in a given assembly. ITTRs, which are installed after core discharge and do not contain radioactive material, may also be stored in the assembly. BPRAs, TPDs, and ITTRs are authorized for unrestricted storage in an MPC. The CRAs are restricted to the center twelve locations in an MPC while the APSRs are restricted to the center four locations in the MPC-24, MPC-24E, MPC-24EF and the center twelve locations in the MPC-32. The calculation of the source term and a description of the bounding fuel devices was provided in Section 5.2.4. The dose rate due to BPRAs and TPDs being stored in a fuel assembly was explicitly calculated. Table 5.4.15 provides the dose rates at various locations on the surface and one meter from the 100-ton HI-TRAC due to the BPRAs and TPDs for the MPC-24 and MPC-32. These results were added to the totals in the other table to provide the total dose rate with BPRAs. As indicated in Table 5.4.15, the dose rates on the side of the cask containing BPRAs are much larger than the dose rates on the side of the cask containing TPDs. Calculations have been performed to show that the side cask dose rates are the dominant contributor to the site boundary dose rates. Therefore, using cask containing BPRAs in the analyses that demonstrate compliance with the site boundary dose limits is bounding for the TPDs

The analyses in this chapter that consider presence of BPRAs assume that a full-length rod with burnable poison is present in all principal locations. In reality, many BPRAs contain full-length poison rods in some locations, and thimble rodlets in others. The burnup and cooling time combinations listed in Table 2.1.25 for BPRAs and TPDs were selected to ensure the Co-60 activity of those devices is below the value of 895 Ci (BPRAs) and 141 Ci (TPDs). These activities are used in the dose evaluations presented in this chapter. Apart from the total activity, the axial distribution of the material in those devices is important for the dose rates. This axial distribution is shown in Table 5.2.30 (masses) and 5.2.31 (activities). It can be observed from Table 5.2.31, while TPDs have a lower overall activity, their activity in the upper end fitting and gas plenum region of the assembly is higher compared to that of the BPRAs. These activities were used to calculate the dose rates in Table 5.4.15. The results in this table show that the maximum dose effect for BPRAs is at the side of the cask, while the maximum dose effect of TPDs is near and on the top of the cask. It is to be noted that BPRAs with several thimble plugs may result higher dose rate near and on the top of the cask than that reported in Table 5.4.15. However, the potential local increase in dose near and on the top of the cask due to the presence of several thimble plug rodlets instead of full length BPRAs rods would be more than compensated by the reduction of the dose from the side of the cask at larger distances. Therefore, using BPRAs with all burnable poison rods in the analyses that demonstrate compliance with the site boundary dose limits would be bounding, and hence the burnup and cooling time combinations for BPRAs in Table 2.1.25 are conservative.

As discussed in Section 5.2.4, two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs. The dose rate due to CRAs and APSRs was explicitly calculated for dose locations around the 100-ton HI-TRAC. Tables 5.4.16 and 5.4.17 provide the results for the different configurations of CRAs and APSRs, respectively, in the MPC-24 and MPC-32. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (except for the surface dose rate at the bottom, where the value for the CRA is comparable to or higher than the value from the BPRAs, depending on the CRA configuration) and the dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied. Therefore, even though the dose rates calculated (using a very conservative source term evaluation) are very high, they do not pose a risk from an operations perspective because they are localized in nature. Section 5.1.1 provides additional discussion on the acceptability of the relatively high localized doses on the bottom of the HI-TRACs.

It has to be noted that when twelve CRAs or APSRs are placed in a basket, not all of them are shielded equally well: the components placed in the center four locations have a better shielding than the eight components placed around them. The dose contribution per component will

therefore be lower for components placed in the center four locations. It is therefore recommended from an ALARA perspective to minimize the number of CRAs and APSRs per basket if possible, and place the components close to the center of the basket.

#### 5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately  $1\text{E}+8$  neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as  $5.8\text{E}-6$  neutrons per gamma ( $1\text{E}+8/865/3.7\text{E}+10/0.54$ ) with energy greater than 1.666 MeV or  $1.16\text{E}+5$  neutrons/curie ( $1\text{E}+8/865$ ) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the

beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be  $1.04\text{E}+8$  gammas/sec which would produce a neutron source of 603.2 neutrons/sec ( $1.04\text{E}+8 * 5.8\text{E}-6$ ). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies. This activity level would produce a neutron source of  $4.63\text{E}+6$  neutrons/sec ( $39.9 * 1.16\text{E}+5$ ) or  $6.0\text{E}+4$  neutrons/sec/inch ( $4.63\text{E}+6/77.25$ ). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.18, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, storage of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

#### 5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.37 and 5.2.7 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.18, bounds the thoria rod neutron spectra, Table 5.2.38, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the 100-ton HI-TRAC and the HI-STORM overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher for the 100-ton HI-TRAC and only 20% lower for the HI-STORM overpack than the dose rate from an MPC full of thoria rod

canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

Per Supplement 5.III, the effect of the design differences between the MPC-68 and MPC-68M on the dose rates is small and all results and conclusions from the MPC-68 are applicable to the MPC-68M. Therefore, the thoria rod canister is also acceptable for storage in the MPC-68M.

#### 5.4.9 Regionalized Loading Dose Rate Evaluation

Section 2.1.9 describes the regionalized loading scheme available in the HI-STORM 100 system. Depending on the choice of  $X$  (the ratio of inner region assembly heat load to outer region assembly heat load), higher heat load fuel (higher burnup and shorter cooling time) may be placed in either region 1 or region 2. If  $X$  is greater than 1, the higher heat load fuel is placed in region 1 and shielded by lower heat load fuel in region 2. This configuration produces the lowest dose rates since the older colder fuel is being used as shielding for the younger hotter fuel. If  $X$  is less than 1, then the younger hotter fuel is placed on the periphery of the basket and the older colder fuel is placed on the interior of the basket. This configuration will result in higher radial dose rates than for configurations with  $X$  greater than or equal to 1. In order to perform a bounding shielding analysis, the burnup and cooling time combinations listed in Section 5.1 were chosen to bound all values of  $X$ . All fuel assemblies in an MPC were assumed to have the same burnup and cooling time in the shielding analysis. This approach results in dose rates calculated in this chapter that bound all allowable regionalized and uniform loading burnup and cooling time combinations.

#### 5.4.10 Fuel Assemblies with Stainless Steel Replacement Rods Dose Rate Evaluation

A dose rate evaluation for the HI-STORM 100S Version B containing the MPC-32 and the MPC-68 is performed to determine the impact of storing fuel assemblies with irradiated stainless steel replacement rods. The stainless steel rods are irradiated in the same neutron flux and for the same time period as the design basis PWR and BWR UO<sub>2</sub> fuel rods. The dose rates at several locations, adjacent to and at 1 meter, from the HI-STORM containing the MPC-32 are presented in Table 5.1.11 and Table 5.1.14, respectively. The dose rates for the HI-STORM containing the MPC-68 are presented in Tables 5.1.13 and Table 5.1.16. The dose rates at the same locations are calculated assuming all 32 design basis PWR assemblies contain 4 irradiated stainless steel replacement rods and all 68 design basis BWR assemblies contain 2 irradiated stainless steel replacement rods. The dose rates with the 4 irradiated stainless steel replacement rods in the design basis PWR assembly are approximately 10% higher at the sides and top of the HI-STORM containing the MPC-32. The dose rates with the 2 irradiated stainless steel replacement rods in the design basis BWR assembly are approximately 33% higher at the sides and top of the HI-STORM containing the MPC-68. Therefore, fuel assemblies containing irradiated stainless

steel replacement rods are acceptable for storage and, if present in a fuel assembly, need to be considered in the site specific dose calculations.



Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])

<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])

<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)

**FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])**

<b>Neutron Energy (MeV)</b>	<b>Quality Factor</b>	<b>(rem/hr)<sup>†</sup>/(n/cm<sup>2</sup>-s)</b>
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC  
CONDITION WITH AN EMPTY NEUTRON SHIELD  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
60,000 MWD/MTU AND 3-YEAR COOLING

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	44.75	370.31	55.57	470.63	473.46
2	2649.74	0.79	810.25	3460.78	3664.65
3	9.98	484.21	11.35	505.54	637.13
4	39.13	367.92	2.10	409.15	511.41
5 (pool lid)	126.99	2071.23	5.62	2203.84 <sup>†††</sup>	2214.03
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	351.97	74.73	115.53	542.22	568.65
2	1171.07	7.08	265.57	1443.73	1533.97
3	140.14	117.78	47.64	305.55	349.87

Note: MPC internal water level is 10 inches below the MPC lid.

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.3

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC  
CONDITION WITH A FULL NEUTRON SHIELD  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
60,000 MWD/MTU AND 3-YEAR COOLING

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	34.56	318.37	6.08	359.01	361.11
2	1565.21	0.50	53.06	1618.77	1734.33
3	6.38	483.03	1.07	490.48	621.41
4	39.09	367.91	1.93	408.92	511.18
5 (pool lid)	126.63	2071.47	5.41	2203.51 <sup>†††</sup>	2213.68
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	202.31	49.47	7.09	258.86	273.65
2	681.82	3.93	19.97	705.72	756.28
3	80.54	81.26	2.41	164.21	193.09

Note: MPC internal water level is 10 inches below the MPC lid.

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.4

[INTENTIONALLY DELETED]

Table 5.4.5

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Table 5.4.6

ANNUAL DOSE AT 350 METERS FROM A SINGLE  
HI-STORM 100S VERSION B OVERPACK WITH AN MPC-24 WITH DESIGN BASIS  
ZIRCALOY CLAD FUEL<sup>†</sup>

Dose Component	60,000 MWD/MTU 3-Year Cooling (mrem/yr)
Fuel gammas <sup>††</sup>	17.54
<sup>60</sup> Co Gammas	1.18
Neutrons	0.54
Total	19.26

<sup>†</sup> 8760 hour annual occupancy is assumed.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM  
VARIOUS HI-STORM 100S VERSION B ISFSI CONFIGURATIONS  
60,000 MWD/MTU AND 3-YEAR COOLING ZIRCALOY CLAD FUEL<sup>†</sup>

<b>Distance</b>	<b>A Side of Overpack (mrem/yr)</b>	<b>B Top of Overpack (mrem/yr)</b>	<b>C Side of Shielded Overpack (mrem/yr)</b>
100 meters	880.47	2.79	176.09
150 meters	299.26	1.04	59.85
200 meters	129.22	0.48	25.84
250 meters	64.50	0.22	12.90
300 meters	34.11	0.12	6.82
350 meters	19.20	6.06E-02	3.84
400 meters	11.26	3.25E-02	2.25
450 meters	6.81	1.78E-02	1.36
500 meters	4.22	1.07E-02	0.84
550 meters	2.71	6.94E-03	0.54
600 meters	1.74	4.13E-03	0.35

<sup>†</sup> 8760 hour annual occupancy is assumed.



Table 5.4.8

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR  
DESIGN BASIS STAINLESS STEEL CLAD FUEL  
WITHOUT BPRAs

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
<b>MPC-24 (40,000 MWD/MTU AND 8-YEAR COOLING)</b>				
2 (Adjacent)	36.97	0.02	1.11	38.10
2 (One Meter)	18.76	0.17	0.50	19.43
<b>MPC-32 (40,000 MWD/MTU AND 9-YEAR COOLING)</b>				
2 (Adjacent)	37.58	0.00	1.49	39.08
2 (One Meter)	18.74	0.25	0.58	19.57
<b>MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)</b>				
2 (Adjacent)	17.79	0.01	0.10	17.90
2 (One Meter)	8.98	0.13	0.04	9.15

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.9

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
50,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	118.31	25.51	1260.91	340.66	1745.39
2	2638.99	125.79	0.80	227.40	2992.98
3	7.38	2.57	833.34	147.88	991.17
3 (temp)	4.33	4.53	408.34	2.70	419.89
4	25.26	0.90	494.35	195.30	715.82
4 (outer)	7.49	0.70	132.37	124.46	265.02
5 (pool lid)	351.88	31.12	5646.34	2040.15	8069.50
5 (transfer lid)	509.02	1.43	8507.35	1288.83	10306.63
5 (t-outer)	187.09	0.61	750.08	479.11	1416.90
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	359.24	15.46	117.76	60.21	552.67
2	1134.59	35.59	8.69	78.41	1257.27
3	88.49	6.19	178.43	16.85	289.97
3 (temp)	88.38	6.52	142.94	8.14	245.97
4	7.94	0.52	151.01	47.61	207.07
5 (transfer lid)	260.67	0.62	3774.45	344.08	4379.82
5 (t-outer)	32.63	1.12	318.78	97.39	449.92

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5 (t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.10

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Table 5.4.11

**DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL  
45,000 MWD/MTU AND 3-YEAR COOLING**

<b>Dose Point Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><math>^{60}\text{Co}</math> Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	123.09	17.99	1134.82	253.68	1529.57	1539.91
2	3059.28	77.64	1.63	145.58	3284.12	3576.87
3	42.51	3.43	703.72	203.47	953.12	1178.63
4	102.68	1.55	527.34	253.44	885.02	1064.21
4 (outer)	28.17	0.91	132.30	174.27	335.65	380.68
5 (pool)	745.11	26.66	6237.41	1665.29	8674.47	8757.27
5 (transfer)	1366.50	1.04	9401.97	966.70	11736.20	11837.01
5 (t-outer)	246.52	0.48	787.00	363.83	1397.82	1418.02
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	407.43	10.23	168.91	42.45	629.02	667.20
2	1350.21	24.30	12.79	54.21	1441.51	1571.36
3	176.77	5.64	144.81	19.97	347.18	414.23
4	31.95	0.40	157.25	63.15	252.75	305.95
5 (transfer)	567.40	0.23	3723.34	260.88	4551.86	4598.19
5 (t-outer)	58.97	0.96	330.25	76.48	466.67	471.29

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5 (t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.12

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Table 5.4.13

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS  
WITH FOUR DAMAGED FUEL CONTAINERS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
60,000 MWD/MTU AND 3-YEAR COOLING  
WITHOUT BPRAs

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	162.53	40.07	958.00	607.90	1768.50
2	3154.37	145.11	1.14	264.13	3564.75
3	47.06	9.00	528.22	583.57	1167.85
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	449.03	20.58	142.41	92.37	704.39
2	1404.60	44.91	11.27	99.48	1560.26
3	202.22	11.75	118.30	54.30	386.56

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.14

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS  
WITH SIXTEEN DAMAGED FUEL CONTAINERS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL  
50,000 MWD/MTU AND 3-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	276.49	36.57	1260.91	629.42	2203.39
2	2488.31	126.39	0.80	213.81	2829.31
3	9.10	5.48	833.34	324.77	1172.69
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	417.32	19.09	117.76	90.45	644.62
2	1091.56	38.82	8.69	85.73	1224.79
3	121.49	8.49	178.43	33.62	342.02

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.15

## DOSE RATES DUE TO BPRAs AND TPDs FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS

Dose Point Location	MPC-24		MPC-32	
	BPRAs (mrem/hr)	TPDs (mrem/hr)	BPRAs (mrem/hr)	TPDs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>				
1	8.54	0.00	10.34	0.03
2	247.24	0.08	292.75	0.11
3	148.53	354.62	225.50	530.55
3 (temp)	68.15	158.51	93.63	217.14
4	124.50	300.92	179.19	440.31
4 (outer)	31.33	76.48	45.02	110.91
5 (pool lid)	64.77	0.00	82.79	0.03
5 (transfer lid)	96.84	0.00	100.81	0.00
5 (t-outer)	21.47	0.00	20.20	0.00
<b>ONE METER FROM THE 100-TON HI-TRAC</b>				
1	32.48	0.51	38.18	0.68
2	109.47	3.38	129.85	4.60
3	49.43	109.78	67.05	155.41
3 (temp)	44.57	98.73	59.32	138.04
4	38.57	94.10	53.20	133.08
5 (transfer lid)	39.13	0.00	46.33	0.00
5 (t-outer)	3.90	0.00	4.63	0.00

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5 (t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.



Table 5.4.16

**DOSE RATES DUE TO CRAs FROM THE 100-TON HI-TRAC  
FOR NORMAL CONDITIONS**

<b>Dose Point Location</b>	<b>MPC-24</b>		<b>MPC-32</b>	
	<b>Config. 1 (mrem/hr)</b>	<b>Config. 2 (mrem/hr)</b>	<b>Config. 1 (mrem/hr)</b>	<b>Config. 2 (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>				
1	49.65	8.99	25.25	5.12
2	3.15	0.08	0.12	0.01
3	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5 (pool lid)	1061.33	213.34	1504.83	300.86
5 (transfer lid)	1846.04	349.09	2785.92	521.49
5 (t-outer)	36.32	5.82	15.21	2.76
<b>ONE METER FROM THE 100-TON HI-TRAC</b>				
1	28.42	3.59	5.11	0.97
2	13.64	1.19	0.67	0.09
3	0.20	0.02	0.02	0.00
4	0.00	0.00	0.00	0.00
5 (transfer lid)	545.93	100.60	638.62	119.05
5 (t-outer)	29.81	5.51	30.89	5.88

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5 (t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.17

**DOSE RATES DUE TO APSRs FROM THE 100-TON HI-TRAC  
FOR NORMAL CONDITIONS**

Dose Point Location	MPC-24			MPC-32		
	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	12.42	2.35	12.25	57.80	11.73	56.80
2	0.21	0.01	9.12	0.29	0.02	17.70
3	0.00	0.00	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00	0.00	0.09
5 (pool lid)	1996.57	371.98	1941.51	3243.57	645.11	3698.77
5 (transfer)	3021.08	572.85	2994.54	5522.01	1035.17	5664.10
5 (t-outer)	3.41	0.54	3.57	33.25	5.97	31.46
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	2.73	0.46	3.49	11.69	2.22	11.36
2	0.61	0.07	3.31	1.55	0.20	6.53
3	0.02	0.00	0.04	0.05	0.01	0.08
4	0.00	0.00	0.00	0.00	0.00	0.09
5 (transfer)	458.06	84.81	444.44	1288.04	240.05	1211.44
5 (t-outer)	17.11	3.19	17.36	64.18	12.16	62.83

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5 (t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.18

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR  
DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

<b>Assembly</b>	<b>Active fuel length (inch)</b>	<b>Neutrons per sec per inch</b>	<b>Neutrons per sec per inch with Sb-Be source</b>	<b>Reference for neutrons per sec per inch</b>
7x7 design basis	144	9.17E+5	N/A	40 GWD/MTU and 5 year cooling
6x6 design basis	110	2.0E+5	2.6E+5	Table 5.2.18
6x6 design basis MOX	110	3.06E+5	3.66E+5	Table 5.2.23

Table 5.4.19

DOSE RATES FROM THE HI-TRAC 100D FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL  
35,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	156.55	10.12	1558.18	77.72	1802.56	1818.78
2	2575.94	34.87	1.88	63.16	2675.84	2972.05
3	36.72	1.51	596.76	89.80	724.78	950.60
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	349.37	4.72	192.55	16.34	562.97	602.58
2	1120.77	10.83	12.08	22.98	1166.66	1297.14
3	146.00	2.46	124.89	8.66	282.02	350.18

## Notes:

- Refer to Figure 5.1.4 for dose locations.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

## 5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this FSAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

This chapter has evaluated these shielding SSCs important to safety and has assessed the impact on health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM 100 System.

It has been shown that the design of the shielding system of the HI-STORM 100 System is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel.

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**APPENDIX 5.A**  
**SAMPLE INPUT FILE FOR SAS2H**

```

=SAS2H    PARM='halt05,skipshipdata'
bw 15x15 PWR assembly
' fuel temp 923
44groupndf5    LATTICECELL
UO2 1 0.95 923 92234 0.03204 92235 3.6 92236 0.01656
    92238 96.3514 END
,
' Zirc 4 composition
ARBM-ZIRC4 6.55 4 1 0 0 50000 1.7 26000 0.24 24000 0.13 40000 97.93
    2 1.0 595 END
,
' water with 652.5 ppm boron
H2O    3 DEN=0.7135 1 579 END
ARBM-BORMOD 0.7135 1 1 0 0 5000 100 3 652.5E-6 579 END
,
co-59  3 0 1-20 579 end
kr-83  1 0 1-20 923 end
kr-84  1 0 1-20 923 end
kr-85  1 0 1-20 923 end
kr-86  1 0 1-20 923 end
sr-90  1 0 1-20 923 end
y-89   1 0 1-20 923 end
zr-94  1 0 1-20 923 end
zr-95  1 0 1-20 923 end
mo-94  1 0 1-20 923 end
mo-95  1 0 1-20 923 end
nb-94  1 0 1-20 923 end
nb-95  1 0 1-20 923 end
tc-99  1 0 1-20 923 end
ru-106 1 0 1-20 923 end
rh-103 1 0 1-20 923 end
rh-105 1 0 1-20 923 end
sb-124 1 0 1-20 923 end
sn-126 1 0 1-20 923 end
xe-131 1 0 1-20 923 end
xe-132 1 0 1-20 923 end
xe-134 1 0 1-20 923 end
,
xe-135 1 0 1-09 923 end
,
xe-136 1 0 1-20 923 end
cs-133 1 0 1-20 923 end
cs-134 1 0 1-20 923 end
cs-135 1 0 1-20 923 end
cs-137 1 0 1-20 923 end
ba-136 1 0 1-20 923 end
la-139 1 0 1-20 923 end
ce-144 1 0 1-20 923 end
pr-143 1 0 1-20 923 end
nd-143 1 0 1-20 923 end
nd-144 1 0 1-20 923 end
nd-145 1 0 1-20 923 end
nd-146 1 0 1-20 923 end
nd-147 1 0 1-20 923 end
nd-148 1 0 1-20 923 end
nd-150 1 0 1-20 923 end
pm-147 1 0 1-20 923 end
pm-148 1 0 1-20 923 end
pm-149 1 0 1-20 923 end

```

```

sm-147  1 0 1-20 923 end
sm-148  1 0 1-20 923 end
sm-149  1 0 1-20 923 end
sm-150  1 0 1-20 923 end
sm-151  1 0 1-20 923 end
sm-152  1 0 1-20 923 end
eu-151  1 0 1-20 923 end
eu-153  1 0 1-20 923 end
eu-154  1 0 1-20 923 end
eu-155  1 0 1-20 923 end
gd-154  1 0 1-20 923 end
gd-155  1 0 1-20 923 end
gd-157  1 0 1-20 923 end
gd-158  1 0 1-20 923 end
gd-160  1 0 1-20 923 end
'
END COMP
'
' -----
'
'   FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH  1.44272 0.950468 1 3 1.08712 2 0.97028 0 END
'
' -----
'
'   MTU in this model is 0.495485 based on fuel dimensions provided
'
'   1 power cycle will be used and a library will be generated every
'   2500 MWD/MTU power level is 40 MW/MTU
'   therefore 62.5 days per 2500 MWD/MTU
'   Below
'   BURN=62.5*NLIB/CYC
'   POWER=MTU*40
'
'   Number of libraries is 20 which is 50,000 MWD/MTU burnup (20*2500)
'
'   ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=208 FUELNGTH=365.76 NCYCLES=1 NLIB/CYC=20
PRINTLEVEL=1
LIGHTEL=5  INPLEVEL=1  NUMHOLES=17
NUMINStr= 0  ORTUBE= 0.6731  SRTUBE=0.63246  END
POWER=19.81938  BURN=1250.0  END
'
O 66.54421
FE 0.24240868
ZR 98.78151  CR 0.1311304  SN 1.714782
'
END
=SAS2H  PARM='restarts, halt10, skipshipdata'
bw 15x15 PWR assembly
END
=SAS2H  PARM='restarts, halt15, skipshipdata'
bw 15x15 PWR assembly
END
=SAS2H  PARM='restarts, halt20, skipshipdata'
bw 15x15 PWR assembly
END

```

**APPENDIX 5.B**  
**SAMPLE INPUT FILE FOR ORIGEN-S**

```

#ORIGENS
0$$ A4 33 A8 26 A11 71 E
1$$ 1 T
bw 15x15 FUEL -- FT33F001 -
'
' SUBCASE 1 LIBRARY POSITION 1
'
' lib pos grms photon group
3$$ 33 A3 1 0 A16 2 E T
35$$ 0 T
56$$ 5 5 A6 3 A10 0 A13 9 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
FUEL 3.6
BW 15x15 0.495485 MTU
58** 19.81938 19.81938 19.81938 19.81938 19.81938
60** 1.0000 3.0000 15.0000 30.0000 62.5
66$$ A1 2 A5 2 A9 2 E
73$$ 922350 922340 922360 922380 80000 500000
260000 240000 400000
74** 17837.45 158.7533 82.05225 477406.4 66544.21 1714.782
242.0868 131.1304 98781.51
75$$ 2 2 2 2 4 4 4 4 4 T
'
' SUBCASE 2 LIBRARY POSITION 2
'
3$$ 33 A3 2 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 5 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 3 LIBRARY POSITION 3
'
3$$ 33 A3 3 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 4 LIBRARY POSITION 4
'
3$$ 33 A3 4 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'

```

```

' SUBCASE 5 LIBRARY POSITION 5
,
3$$ 33 A3 5 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
,
' SUBCASE 6 LIBRARY POSITION 6
,
3$$ 33 A3 6 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
,
' SUBCASE 7 LIBRARY POSITION 7
,
3$$ 33 A3 7 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
,
' SUBCASE 8 LIBRARY POSITION 8
,
3$$ 33 A3 8 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
,
' SUBCASE 9 LIBRARY POSITION 9
,
3$$ 33 A3 9 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T

```

```

'
' SUBCASE 10 LIBRARY POSITION 10
'
3$$ 33 A3 10 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 11 LIBRARY POSITION 11
'
3$$ 33 A3 11 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 12 LIBRARY POSITION 12
'
3$$ 33 A3 12 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 13 LIBRARY POSITION 13
'
3$$ 33 A3 13 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 14 LIBRARY POSITION 14
'
3$$ 33 A3 14 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5

```

```

66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 15 LIBRARY POSITION 15
'
3$$ 33 A3 15 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 16 LIBRARY POSITION 16
'
3$$ 33 A3 16 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 17 LIBRARY POSITION 17
'
3$$ 33 A3 17 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 18 LIBRARY POSITION 18
'
3$$ 33 A3 18 A4 7 0 A16 2 A33 18 E T
35$$ 0 T
56$$ 3 3 A6 1 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE - decay
'
54$$ A8 1 E
56$$ 0 9 A6 1 A10 3 A14 3 A15 1 A19 1 E
57** 0.0 0 1.E-5 E T
fuel enrichment above
60** 0.5 0.75 1.0 4.0 8.0 12.0 24.0 48.0 96.0
61** F0.1
65$$

```



```

'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z T
,
' SUBCASE - decay
,
54$$$ A8 1 E
56$$$ 0 9 A6 1 A10 9 A14 4 A15 1 A19 1 E
57** 4.0 0 1.E-5 E T
  fuel enrichment above
60** 10.0 20.0 30.0 60.0 90.0 120.0 180.0 240.0 365.0
61** F0.1
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z T
,
' SUBCASE - decay
,
54$$$ A8 0 E
56$$$ 0 9 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 1.0 0 1.E-5 E T
  fuel enrichment above
60** 1.5 3.0 4.0 5.0 6.0 7.0 8.0 9.0 10.0
61** F1.0e-5
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
81$$$ 2 0 26 1 E
82$$$ 0 2 2 2 2 2 2 2 2
83** 1.1E+7 8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6 7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4 2.0E+4 1.0E+4
84** 20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
,
,
,
,
,
,
,
,
' SUBCASE - decay
,
54$$$ A8 0 E
56$$$ 0 10 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 10.0 0 1.E-5 E T
  fuel enrichment above
60** 11.0 12.0 13.0 14.0 15.0 16.0 17.0 18.0 19.0 20.0
61** F1.0e-5
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z

```



**APPENDIX 5.C**  
**SAMPLE INPUT FILE FOR MCNP**

message: outp=hs24c11o srctp=hs24c11s runtpe=hs24c11r  
mctal=hs24c11m wssa=hs24c11w rssa=pt001w

hs24c11

```

c
c   origin is 6 inches below mpc
c
c   only cells that contain material are split axially
c   importance splitting is not done in cells with 0 material
c
c   axial segmentation is at the following boundaries
c   615, 620, 420, 430, 445, 455, 675, 651, 652 ,653
c   654, 655, 656, 657, 680
c
c   universe 1
c
301  0          (-40:41:-42:43)  -400 u=1
302  0   37 -38      -12          400 -410 u=1
303  0   37 -38    15           400 -410 u=1
304  0   35 -36      -20         400 -410 u=1
305  0   35 -36        23         400 -410 u=1
306  0   37 -38      -12          435 -460 u=1
307  0   37 -38    15           435 -460 u=1
308  0   35 -36      -20         435 -460 u=1
309  0   35 -36        23         435 -460 u=1
310  0   37 -38      -10          410 -435 u=1
311  0   37 -38    17           410 -435 u=1
312  0   35 -36      -18         410 -435 u=1
313  0   35 -36        25         410 -435 u=1
314  5  -7.92    10 -11 26 -27 410 -420 u=-1 $ left
315  6  -2.644   11 -12 26 -27 410 -420 u=-1 $ left
316  6  -2.644   15 -16 26 -27 410 -420 u=-1 $ right
317  5  -7.92    16 -17 26 -27 410 -420 u=-1 $ right
318  5  -7.92    28 -29 18 -19 410 -420 u=-1 $ bot
319  6  -2.644   28 -29 19 -20 410 -420 u=-1 $ bot
320  6  -2.644   28 -29 23 -24 410 -420 u=-1 $ top
321  5  -7.92    28 -29 24 -25 410 -420 u=-1 $ top
322  5  -7.92    10 -11 26 -27 420 -430 u=-1 $ left
323  6  -2.644   11 -12 26 -27 420 -430 u=-1 $ left
324  6  -2.644   15 -16 26 -27 420 -430 u=-1 $ right
325  5  -7.92    16 -17 26 -27 420 -430 u=-1 $ right
326  5  -7.92    28 -29 18 -19 420 -430 u=-1 $ bot
327  6  -2.644   28 -29 19 -20 420 -430 u=-1 $ bot
328  6  -2.644   28 -29 23 -24 420 -430 u=-1 $ top
329  5  -7.92    28 -29 24 -25 420 -430 u=-1 $ top
330  5  -7.92    10 -11 26 -27 430 -435 u=-1 $ left
331  6  -2.644   11 -12 26 -27 430 -435 u=-1 $ left
332  6  -2.644   15 -16 26 -27 430 -435 u=-1 $ right
333  5  -7.92    16 -17 26 -27 430 -435 u=-1 $ right
334  5  -7.92    28 -29 18 -19 430 -435 u=-1 $ bot
335  6  -2.644   28 -29 19 -20 430 -435 u=-1 $ bot
336  6  -2.644   28 -29 23 -24 430 -435 u=-1 $ top
337  5  -7.92    28 -29 24 -25 430 -435 u=-1 $ top
338  0          10 -12 27 -38 410 -435 u=-1 $ left
339  0          10 -12 37 -26 410 -435 u=-1 $ left
340  0          15 -17 27 -38 410 -435 u=-1 $ right
341  0          15 -17 37 -26 410 -435 u=-1 $ right
342  0          35 -28 18 -20 410 -435 u=-1 $ bot
343  0          29 -36 18 -20 410 -435 u=-1 $ bot
344  0          35 -28 23 -25 410 -435 u=-1 $ top
345  0          29 -36 23 -25 410 -435 u=-1 $ top
346  5  -7.92    12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-1
347  5  -7.92    12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-1
348  5  -7.92    12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-1
349  5  -7.92    12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-1

```

Attachment 6 to Holtec Letter 5014874

```

350 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-1
351 9 -1.17e-3 460 u=1
c fuel element
352 0 40 -41 42 -43 -415 u=1
353 5 -1.0783 40 -41 42 -43 415 -420 u=-1 $ lower nozzle
354 0 40 -41 42 -43 420 -425 u=-1 $ space
355 2 -3.8699 40 -41 42 -43 425 -430 u=-1 $ active fuel
356 5 -0.1591 40 -41 42 -43 430 -440 u=-1 $ space
357 5 -0.1591 40 -41 42 -43 440 -445 u=-1 $ plenum spacer
358 5 -1.5410 40 -41 42 -43 445 -455 u=-1 $ top nozzle
359 0 13 -14 21 -22 455 -460 u=-1
c
360 5 -7.92 38 -23 -12 400 -420 u=1
361 5 -7.92 20 -37 -12 400 -420 u=1
362 5 -7.92 12 -35 23 400 -420 u=1
363 5 -7.92 12 -35 -20 400 -420 u=1
364 5 -7.92 36 -15 23 400 -420 u=1
365 5 -7.92 36 -15 -20 400 -420 u=1
366 5 -7.92 38 -23 15 400 -420 u=1
367 5 -7.92 20 -37 15 400 -420 u=1
368 5 -7.92 38 -23 -12 420 -430 u=1
369 5 -7.92 20 -37 -12 420 -430 u=1
370 5 -7.92 12 -35 23 420 -430 u=1
371 5 -7.92 12 -35 -20 420 -430 u=1
372 5 -7.92 36 -15 23 420 -430 u=1
373 5 -7.92 36 -15 -20 420 -430 u=1
374 5 -7.92 38 -23 15 420 -430 u=1
375 5 -7.92 20 -37 15 420 -430 u=1
376 5 -7.92 38 -23 -12 430 -445 u=1
377 5 -7.92 20 -37 -12 430 -445 u=1
378 5 -7.92 12 -35 23 430 -445 u=1
379 5 -7.92 12 -35 -20 430 -445 u=1
380 5 -7.92 36 -15 23 430 -445 u=1
381 5 -7.92 36 -15 -20 430 -445 u=1
382 5 -7.92 38 -23 15 430 -445 u=1
383 5 -7.92 20 -37 15 430 -445 u=1
384 5 -7.92 38 -23 -12 445 -460 u=1
385 5 -7.92 20 -37 -12 445 -460 u=1
386 5 -7.92 12 -35 23 445 -460 u=1
387 5 -7.92 12 -35 -20 445 -460 u=1
388 5 -7.92 36 -15 23 445 -460 u=1
389 5 -7.92 36 -15 -20 445 -460 u=1
390 5 -7.92 38 -23 15 445 -460 u=1
391 5 -7.92 20 -37 15 445 -460 u=1
392 0 23 -12 400 -460 u=1
393 0 23 15 400 -460 u=1
394 0 15 -20 400 -460 u=1
395 0 -12 -20 400 -460 u=1
c
c universe 2
c
401 0 (-40:41:-42:43) -400 u=2
402 0 37 -38 -12 400 -410 u=2
403 0 37 -38 15 400 -410 u=2
404 0 35 -36 -20 400 -410 u=2
405 0 35 -36 23 400 -410 u=2
406 0 37 -38 -12 435 -460 u=2
407 0 37 -38 15 435 -460 u=2
408 0 35 -36 -20 435 -460 u=2
409 0 35 -36 23 435 -460 u=2
410 0 37 -38 -10 410 -435 u=2
411 0 37 -38 17 410 -435 u=2
412 0 35 -36 -18 410 -435 u=2
413 0 35 -36 25 410 -435 u=2
414 5 -7.92 10 -11 26 -27 410 -420 u=-2 $ left

```

Attachment 6 to Holtec Letter 5014874

```

415 6 -2.644 11 -12 26 -27 410 -420 u=-2 $ left
416 6 -2.644 15 -16 30 -31 410 -420 u=-2 $ right
417 5 -7.92 16 -17 30 -31 410 -420 u=-2 $ right
418 5 -7.92 28 -29 18 -19 410 -420 u=-2 $ bot
419 6 -2.644 28 -29 19 -20 410 -420 u=-2 $ bot
420 6 -2.644 32 -33 23 -24 410 -420 u=-2 $ top
421 5 -7.92 32 -33 24 -25 410 -420 u=-2 $ top
422 5 -7.92 10 -11 26 -27 420 -430 u=-2 $ left
423 6 -2.644 11 -12 26 -27 420 -430 u=-2 $ left
424 6 -2.644 15 -16 30 -31 420 -430 u=-2 $ right
425 5 -7.92 16 -17 30 -31 420 -430 u=-2 $ right
426 5 -7.92 28 -29 18 -19 420 -430 u=-2 $ bot
427 6 -2.644 28 -29 19 -20 420 -430 u=-2 $ bot
428 6 -2.644 32 -33 23 -24 420 -430 u=-2 $ top
429 5 -7.92 32 -33 24 -25 420 -430 u=-2 $ top
430 5 -7.92 10 -11 26 -27 430 -435 u=-2 $ left
431 6 -2.644 11 -12 26 -27 430 -435 u=-2 $ left
432 6 -2.644 15 -16 30 -31 430 -435 u=-2 $ right
433 5 -7.92 16 -17 30 -31 430 -435 u=-2 $ right
434 5 -7.92 28 -29 18 -19 430 -435 u=-2 $ bot
435 6 -2.644 28 -29 19 -20 430 -435 u=-2 $ bot
436 6 -2.644 32 -33 23 -24 430 -435 u=-2 $ top
437 5 -7.92 32 -33 24 -25 430 -435 u=-2 $ top
438 0 10 -12 27 -38 410 -435 u=-2 $ left
439 0 10 -12 37 -26 410 -435 u=-2 $ left
440 0 15 -17 31 -38 410 -435 u=-2 $ right
441 0 15 -17 37 -30 410 -435 u=-2 $ right
442 0 35 -28 18 -20 410 -435 u=-2 $ bot
443 0 29 -36 18 -20 410 -435 u=-2 $ bot
444 0 35 -32 23 -25 410 -435 u=-2 $ top
445 0 33 -36 23 -25 410 -435 u=-2 $ top
446 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-2
447 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-2
448 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-2
449 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-2
450 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-2
451 9 -1.17e-3 460 u=2
c
fuel element
452 0 40 -41 42 -43 -415 u=2
453 5 -1.0783 40 -41 42 -43 415 -420 u=-2 $ lower nozzle
454 0 40 -41 42 -43 420 -425 u=-2 $ space
455 2 -3.8699 40 -41 42 -43 425 -430 u=-2 $ active fuel
456 5 -0.1591 40 -41 42 -43 430 -440 u=-2 $ space
457 5 -0.1591 40 -41 42 -43 440 -445 u=-2 $ plenum spacer
458 5 -1.5410 40 -41 42 -43 445 -455 u=-2 $ top nozzle
459 0 13 -14 21 -22 455 -460 u=-2
c
460 5 -7.92 38 -23 -12 400 -420 u=2
461 5 -7.92 20 -37 -12 400 -420 u=2
462 0 12 -35 23 400 -420 u=2
463 5 -7.92 12 -35 -20 400 -420 u=2
464 0 36 -15 23 400 -420 u=2
465 5 -7.92 36 -15 -20 400 -420 u=2
466 0 38 -23 15 400 -420 u=2
467 0 20 -37 15 400 -420 u=2
468 5 -7.92 38 -23 -12 420 -430 u=2
469 5 -7.92 20 -37 -12 420 -430 u=2
470 0 12 -35 23 420 -430 u=2
471 5 -7.92 12 -35 -20 420 -430 u=2
472 0 36 -15 23 420 -430 u=2
473 5 -7.92 36 -15 -20 420 -430 u=2
474 0 38 -23 15 420 -430 u=2
475 0 20 -37 15 420 -430 u=2
476 5 -7.92 38 -23 -12 430 -445 u=2
477 5 -7.92 20 -37 -12 430 -445 u=2

```

Attachment 6 to Holtec Letter 5014874

478	0	12 -35 23	430 -445	u=2
479	5 -7.92	12 -35 -20	430 -445	u=2
480	0	36 -15 23	430 -445	u=2
481	5 -7.92	36 -15 -20	430 -445	u=2
482	0	38 -23 15	430 -445	u=2
483	0	20 -37 15	430 -445	u=2
484	5 -7.92	38 -23 -12	445 -460	u=2
485	5 -7.92	20 -37 -12	445 -460	u=2
486	0	12 -35 23	445 -460	u=2
487	5 -7.92	12 -35 -20	445 -460	u=2
488	0	36 -15 23	445 -460	u=2
489	5 -7.92	36 -15 -20	445 -460	u=2
490	0	38 -23 15	445 -460	u=2
491	0	20 -37 15	445 -460	u=2
492	0	23 -12	400 -460	u=2
493	0	23 15	400 -460	u=2
494	0	15 -20	400 -460	u=2
495	0	-12 -20	400 -460	u=2
c				
c	universe 3			
c				
501	0	(-40:41:-42:43)	-400	u=3
502	0 37 -38	-12	400 -410	u=3
503	0 37 -38	15	400 -410	u=3
504	0 35 -36		-20 400 -410	u=3
505	0 35 -36		23 400 -410	u=3
506	0 37 -38	-12	435 -460	u=3
507	0 37 -38	15	435 -460	u=3
508	0 35 -36		-20 435 -460	u=3
509	0 35 -36		23 435 -460	u=3
510	0 37 -38	-10	410 -435	u=3
511	0 37 -38	17	410 -435	u=3
512	0 35 -36		-18 410 -435	u=3
513	0 35 -36		25 410 -435	u=3
514	5 -7.92	10 -11 30 -31	410 -420	u=-3 \$ left
515	6 -2.644	11 -12 30 -31	410 -420	u=-3 \$ left
516	6 -2.644	15 -16 26 -27	410 -420	u=-3 \$ right
517	5 -7.92	16 -17 26 -27	410 -420	u=-3 \$ right
518	5 -7.92	28 -29 18 -19	410 -420	u=-3 \$ bot
519	6 -2.644	28 -29 19 -20	410 -420	u=-3 \$ bot
520	6 -2.644	32 -33 23 -24	410 -420	u=-3 \$ top
521	5 -7.92	32 -33 24 -25	410 -420	u=-3 \$ top
522	5 -7.92	10 -11 30 -31	420 -430	u=-3 \$ left
523	6 -2.644	11 -12 30 -31	420 -430	u=-3 \$ left
524	6 -2.644	15 -16 26 -27	420 -430	u=-3 \$ right
525	5 -7.92	16 -17 26 -27	420 -430	u=-3 \$ right
526	5 -7.92	28 -29 18 -19	420 -430	u=-3 \$ bot
527	6 -2.644	28 -29 19 -20	420 -430	u=-3 \$ bot
528	6 -2.644	32 -33 23 -24	420 -430	u=-3 \$ top
529	5 -7.92	32 -33 24 -25	420 -430	u=-3 \$ top
530	5 -7.92	10 -11 30 -31	430 -435	u=-3 \$ left
531	6 -2.644	11 -12 30 -31	430 -435	u=-3 \$ left
532	6 -2.644	15 -16 26 -27	430 -435	u=-3 \$ right
533	5 -7.92	16 -17 26 -27	430 -435	u=-3 \$ right
534	5 -7.92	28 -29 18 -19	430 -435	u=-3 \$ bot
535	6 -2.644	28 -29 19 -20	430 -435	u=-3 \$ bot
536	6 -2.644	32 -33 23 -24	430 -435	u=-3 \$ top
537	5 -7.92	32 -33 24 -25	430 -435	u=-3 \$ top
538	0	10 -12 31 -38	410 -435	u=-3 \$ left
539	0	10 -12 37 -30	410 -435	u=-3 \$ left
540	0	15 -17 27 -38	410 -435	u=-3 \$ right
541	0	15 -17 37 -26	410 -435	u=-3 \$ right
542	0	35 -28 18 -20	410 -435	u=-3 \$ bot
543	0	29 -36 18 -20	410 -435	u=-3 \$ bot
544	0	35 -32 23 -25	410 -435	u=-3 \$ top

## Attachment 6 to Holtec Letter 5014874

```

545 0 33 -36 23 -25 410 -435 u=-3 $ top
546 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-3
547 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-3
548 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-3
549 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-3
550 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-3
551 9 -1.17e-3 460 u=3
c fuel element
552 0 40 -41 42 -43 -415 u=3
553 5 -1.0783 40 -41 42 -43 415 -420 u=-3 $ lower nozzle
554 0 40 -41 42 -43 420 -425 u=-3 $ space
555 2 -3.8699 40 -41 42 -43 425 -430 u=-3 $ active fuel
556 5 -0.1591 40 -41 42 -43 430 -440 u=-3 $ space
557 5 -0.1591 40 -41 42 -43 440 -445 u=-3 $ plenum spacer
558 5 -1.5410 40 -41 42 -43 445 -455 u=-3 $ top nozzle
559 0 13 -14 21 -22 455 -460 u=-3
c
560 0 38 -23 -12 400 -420 u=3
561 0 20 -37 -12 400 -420 u=3
562 0 12 -35 23 400 -420 u=3
563 5 -7.92 12 -35 -20 400 -420 u=3
564 0 36 -15 23 400 -420 u=3
565 5 -7.92 36 -15 -20 400 -420 u=3
566 5 -7.92 38 -23 15 400 -420 u=3
567 5 -7.92 20 -37 15 400 -420 u=3
568 5 -7.92 38 -23 -12 420 -430 u=3
569 5 -7.92 20 -37 -12 420 -430 u=3
570 0 12 -35 23 420 -430 u=3
571 0 12 -35 -20 420 -430 u=3
572 5 -7.92 36 -15 23 420 -430 u=3
573 0 36 -15 -20 420 -430 u=3
574 5 -7.92 38 -23 15 420 -430 u=3
575 5 -7.92 20 -37 15 420 -430 u=3
576 5 -7.92 38 -23 -12 430 -445 u=3
577 5 -7.92 20 -37 -12 430 -445 u=3
578 0 12 -35 23 430 -445 u=3
579 0 12 -35 -20 430 -445 u=3
580 5 -7.92 36 -15 23 430 -445 u=3
581 0 36 -15 -20 430 -445 u=3
582 5 -7.92 38 -23 15 430 -445 u=3
583 5 -7.92 20 -37 15 430 -445 u=3
584 5 -7.92 38 -23 -12 445 -460 u=3
585 5 -7.92 20 -37 -12 445 -460 u=3
586 0 12 -35 23 445 -460 u=3
587 0 12 -35 -20 445 -460 u=3
588 5 -7.92 36 -15 23 445 -460 u=3
589 0 36 -15 -20 445 -460 u=3
590 5 -7.92 38 -23 15 445 -460 u=3
591 5 -7.92 20 -37 15 445 -460 u=3
592 0 23 -12 400 -460 u=3
593 0 23 15 400 -460 u=3
594 0 15 -20 400 -460 u=3
595 0 -12 -20 400 -460 u=3
c
c universe 4
c
601 0 (-40:41:-42:43) -400 u=4
602 0 37 -38 -12 400 -410 u=4
603 0 37 -38 15 400 -410 u=4
604 0 35 -36 -20 400 -410 u=4
605 0 35 -36 23 400 -410 u=4
606 0 37 -38 -12 435 -460 u=4
607 0 37 -38 15 435 -460 u=4
608 0 35 -36 -20 435 -460 u=4
609 0 35 -36 23 435 -460 u=4

```



## Attachment 6 to Holtec Letter 5014874

610	0	37 -38	-10	410	-435	u=4	
611	0	37 -38	17	410	-435	u=4	
612	0	35 -36		-18	410	-435	u=4
613	0	35 -36	25	410	-435	u=4	
614	5	-7.92	10 -11 30	-31	410	-420	u=-4 \$ left
615	6	-2.644	11 -12 30	-31	410	-420	u=-4 \$ left
616	6	-2.644	15 -16 26	-27	410	-420	u=-4 \$ right
617	5	-7.92	16 -17 26	-27	410	-420	u=-4 \$ right
618	5	-7.92	32 -33 18	-19	410	-420	u=-4 \$ bot
619	6	-2.644	32 -33 19	-20	410	-420	u=-4 \$ bot
620	6	-2.644	28 -29 23	-24	410	-420	u=-4 \$ top
621	5	-7.92	28 -29 24	-25	410	-420	u=-4 \$ top
622	5	-7.92	10 -11 30	-31	420	-430	u=-4 \$ left
623	6	-2.644	11 -12 30	-31	420	-430	u=-4 \$ left
624	6	-2.644	15 -16 26	-27	420	-430	u=-4 \$ right
625	5	-7.92	16 -17 26	-27	420	-430	u=-4 \$ right
626	5	-7.92	32 -33 18	-19	420	-430	u=-4 \$ bot
627	6	-2.644	32 -33 19	-20	420	-430	u=-4 \$ bot
628	6	-2.644	28 -29 23	-24	420	-430	u=-4 \$ top
629	5	-7.92	28 -29 24	-25	420	-430	u=-4 \$ top
630	5	-7.92	10 -11 30	-31	430	-435	u=-4 \$ left
631	6	-2.644	11 -12 30	-31	430	-435	u=-4 \$ left
632	6	-2.644	15 -16 26	-27	430	-435	u=-4 \$ right
633	5	-7.92	16 -17 26	-27	430	-435	u=-4 \$ right
634	5	-7.92	32 -33 18	-19	430	-435	u=-4 \$ bot
635	6	-2.644	32 -33 19	-20	430	-435	u=-4 \$ bot
636	6	-2.644	28 -29 23	-24	430	-435	u=-4 \$ top
637	5	-7.92	28 -29 24	-25	430	-435	u=-4 \$ top
638	0		10 -12 31	-38	410	-435	u=-4 \$ left
639	0		10 -12 37	-30	410	-435	u=-4 \$ left
640	0		15 -17 27	-38	410	-435	u=-4 \$ right
641	0		15 -17 37	-26	410	-435	u=-4 \$ right
642	0		35 -32 18	-20	410	-435	u=-4 \$ bot
643	0		33 -36 18	-20	410	-435	u=-4 \$ bot
644	0		35 -28 23	-25	410	-435	u=-4 \$ top
645	0		29 -36 23	-25	410	-435	u=-4 \$ top
646	5	-7.92	12 -15 20	-23	(-13:-21:22:14)	400 -420	u=-4
647	5	-7.92	12 -15 20	-23	(-13:-21:22:14)	420 -430	u=-4
648	5	-7.92	12 -15 20	-23	(-13:-21:22:14)	430 -445	u=-4
649	5	-7.92	12 -15 20	-23	(-13:-21:22:14)	445 -460	u=-4
650	0		13 -14 21	-22	(-40:41:-42:43)	400 -455	u=-4
651	9	-1.17e-3			460	u=4	
c		fuel element					
652	0		40 -41 42	-43	-415	u=4	
653	5	-1.0783	40 -41 42	-43	415 -420	u=-4	\$ lower nozzle
654	0		40 -41 42	-43	420 -425	u=-4	\$ space
655	2	-3.8699	40 -41 42	-43	425 -430	u=-4	\$ active fuel
656	5	-0.1591	40 -41 42	-43	430 -440	u=-4	\$ space
657	5	-0.1591	40 -41 42	-43	440 -445	u=-4	\$ plenum spacer
658	5	-1.5410	40 -41 42	-43	445 -455	u=-4	\$ top nozzle
659	0		13 -14 21	-22	455 -460	u=-4	
c							
660	0		38 -23 -12	400 -420	u=4		
661	0		20 -37 -12	400 -420	u=4		
662	5	-7.92	12 -35 23	400 -420	u=4		
663	0		12 -35 -20	400 -420	u=4		
664	5	-7.92	36 -15 23	400 -420	u=4		
665	0		36 -15 -20	400 -420	u=4		
666	5	-7.92	38 -23 15	400 -420	u=4		
667	5	-7.92	20 -37 15	400 -420	u=4		
668	0		38 -23 -12	420 -430	u=4		
669	0		20 -37 -12	420 -430	u=4		
670	5	-7.92	12 -35 23	420 -430	u=4		
671	0		12 -35 -20	420 -430	u=4		
672	5	-7.92	36 -15 23	420 -430	u=4		

Attachment 6 to Holtec Letter 5014874

673	0	36	-15	-20	420	-430	u=4	
674	5	-7.92	38	-23	15	420	-430	u=4
675	5	-7.92	20	-37	15	420	-430	u=4
676	0		38	-23	-12	430	-445	u=4
677	0		20	-37	-12	430	-445	u=4
678	5	-7.92	12	-35	23	430	-445	u=4
679	0		12	-35	-20	430	-445	u=4
680	5	-7.92	36	-15	23	430	-445	u=4
681	0		36	-15	-20	430	-445	u=4
682	5	-7.92	38	-23	15	430	-445	u=4
683	5	-7.92	20	-37	15	430	-445	u=4
684	0		38	-23	-12	445	-460	u=4
685	0		20	-37	-12	445	-460	u=4
686	5	-7.92	12	-35	23	445	-460	u=4
687	0		12	-35	-20	445	-460	u=4
688	5	-7.92	36	-15	23	445	-460	u=4
689	0		36	-15	-20	445	-460	u=4
690	5	-7.92	38	-23	15	445	-460	u=4
691	5	-7.92	20	-37	15	445	-460	u=4
692	0		23	-12		400	-460	u=4
693	0		23	15		400	-460	u=4
694	0		15	-20		400	-460	u=4
695	0		-12	-20		400	-460	u=4
c								
c	universe 5							
c								
701	0		(-40:41:-42:43)			-400	u=5	
702	0	37	-38	-12		400	-410	u=5
703	0	37	-38	15		400	-410	u=5
704	0	35	-36		-20	400	-410	u=5
705	0	35	-36		23	400	-410	u=5
706	0	37	-38	-12		435	-460	u=5
707	0	37	-38	15		435	-460	u=5
708	0	35	-36		-20	435	-460	u=5
709	0	35	-36		23	435	-460	u=5
710	0	37	-38	-10		410	-435	u=5
711	0	37	-38	17		410	-435	u=5
712	0	35	-36		-18	410	-435	u=5
713	0	35	-36		25	410	-435	u=5
714	5	-7.92	10	-11	26	-27	410	-420 u=-5 \$ left
715	6	-2.644	11	-12	26	-27	410	-420 u=-5 \$ left
716	6	-2.644	15	-16	30	-31	410	-420 u=-5 \$ right
717	5	-7.92	16	-17	30	-31	410	-420 u=-5 \$ right
718	5	-7.92	32	-33	18	-19	410	-420 u=-5 \$ bot
719	6	-2.644	32	-33	19	-20	410	-420 u=-5 \$ bot
720	6	-2.644	28	-29	23	-24	410	-420 u=-5 \$ top
721	5	-7.92	28	-29	24	-25	410	-420 u=-5 \$ top
722	5	-7.92	10	-11	26	-27	420	-430 u=-5 \$ left
723	6	-2.644	11	-12	26	-27	420	-430 u=-5 \$ left
724	6	-2.644	15	-16	30	-31	420	-430 u=-5 \$ right
725	5	-7.92	16	-17	30	-31	420	-430 u=-5 \$ right
726	5	-7.92	32	-33	18	-19	420	-430 u=-5 \$ bot
727	6	-2.644	32	-33	19	-20	420	-430 u=-5 \$ bot
728	6	-2.644	28	-29	23	-24	420	-430 u=-5 \$ top
729	5	-7.92	28	-29	24	-25	420	-430 u=-5 \$ top
730	5	-7.92	10	-11	26	-27	430	-435 u=-5 \$ left
731	6	-2.644	11	-12	26	-27	430	-435 u=-5 \$ left
732	6	-2.644	15	-16	30	-31	430	-435 u=-5 \$ right
733	5	-7.92	16	-17	30	-31	430	-435 u=-5 \$ right
734	5	-7.92	32	-33	18	-19	430	-435 u=-5 \$ bot
735	6	-2.644	32	-33	19	-20	430	-435 u=-5 \$ bot
736	6	-2.644	28	-29	23	-24	430	-435 u=-5 \$ top
737	5	-7.92	28	-29	24	-25	430	-435 u=-5 \$ top
738	0		10	-12	27	-38	410	-435 u=-5 \$ left
739	0		10	-12	37	-26	410	-435 u=-5 \$ left

Attachment 6 to Holtec Letter 5014874

```

740 0 15 -17 31 -38 410 -435 u=-5 $ right
741 0 15 -17 37 -30 410 -435 u=-5 $ right
742 0 35 -32 18 -20 410 -435 u=-5 $ bot
743 0 33 -36 18 -20 410 -435 u=-5 $ bot
744 0 35 -28 23 -25 410 -435 u=-5 $ top
745 0 29 -36 23 -25 410 -435 u=-5 $ top
746 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-5
747 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-5
748 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-5
749 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-5
750 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-5
751 9 -1.17e-3 460 u=5
c fuel element
752 0 40 -41 42 -43 -415 u=5
753 5 -1.0783 40 -41 42 -43 415 -420 u=-5 $ lower nozzle
754 0 40 -41 42 -43 420 -425 u=-5 $ space
755 2 -3.8699 40 -41 42 -43 425 -430 u=-5 $ active fuel
756 5 -0.1591 40 -41 42 -43 430 -440 u=-5 $ space
757 5 -0.1591 40 -41 42 -43 440 -445 u=-5 $ plenum spacer
758 5 -1.5410 40 -41 42 -43 445 -455 u=-5 $ top nozzle
759 0 13 -14 21 -22 455 -460 u=-5
c
760 5 -7.92 38 -23 -12 400 -420 u=5
761 5 -7.92 20 -37 -12 400 -420 u=5
762 5 -7.92 12 -35 23 400 -420 u=5
763 0 12 -35 -20 400 -420 u=5
764 5 -7.92 36 -15 23 400 -420 u=5
765 0 36 -15 -20 400 -420 u=5
766 0 38 -23 15 400 -420 u=5
767 0 20 -37 15 400 -420 u=5
768 5 -7.92 38 -23 -12 420 -430 u=5
769 5 -7.92 20 -37 -12 420 -430 u=5
770 5 -7.92 12 -35 23 420 -430 u=5
771 0 12 -35 -20 420 -430 u=5
772 5 -7.92 36 -15 23 420 -430 u=5
773 0 36 -15 -20 420 -430 u=5
774 0 38 -23 15 420 -430 u=5
775 0 20 -37 15 420 -430 u=5
776 5 -7.92 38 -23 -12 430 -445 u=5
777 5 -7.92 20 -37 -12 430 -445 u=5
778 5 -7.92 12 -35 23 430 -445 u=5
779 0 12 -35 -20 430 -445 u=5
780 5 -7.92 36 -15 23 430 -445 u=5
781 0 36 -15 -20 430 -445 u=5
782 0 38 -23 15 430 -445 u=5
783 0 20 -37 15 430 -445 u=5
784 5 -7.92 38 -23 -12 445 -460 u=5
785 5 -7.92 20 -37 -12 445 -460 u=5
786 5 -7.92 12 -35 23 445 -460 u=5
787 0 12 -35 -20 445 -460 u=5
788 5 -7.92 36 -15 23 445 -460 u=5
789 0 36 -15 -20 445 -460 u=5
790 0 38 -23 15 445 -460 u=5
791 0 20 -37 15 445 -460 u=5
792 0 23 -12 400 -460 u=5
793 0 23 15 400 -460 u=5
794 0 15 -20 400 -460 u=5
795 0 -12 -20 400 -460 u=5
c
c egg crate
c
c storage locations
c
c 201 0 -301 -112 101 620 -675
202 0 -301 112 -113 101 620 -675

```

```

203  0 -301 113 -114 101      620 -675
c    204   0 -301 114      101      620 -675
c
205  0 -301      -111 102      620 -675
206  0 -301 111 -112 102 -101 620 -675
101  0 -301 112 -113 102 -101 620 -675
fill=3 (-13.68679 68.43395 0.0)
102  0 -301 113 -114 102 -101 620 -675
fill=2 ( 13.68679 68.43395 0.0)
207  0 -301 114 -115 102 -101 620 -675
208  0 -301 115      102      620 -675
c
c    209   0 -301      -110 103      620 -675
210  0 -301 110 -111 103 -102 620 -675
103  0      111 -112 103 -102 620 -675
fill=3 (-41.06037 41.06037 0.0)
104  0      112 -113 103 -102 620 -675
fill=1 (-13.68679 41.06037 0.0)
105  0      113 -114 103 -102 620 -675
fill=1 ( 13.68679 41.06037 0.0)
106  0      114 -115 103 -102 620 -675
fill=2 ( 41.06037 41.06037 0.0)
211  0 -301 115 -116 103 -102 620 -675
c    212   0 -301 116      103      620 -675
c
213  0 -301      -110 104 -103 620 -675
107  0 -301 110 -111 104 -103 620 -675
fill=3 (-68.43395 13.68679 0.0)
108  0      111 -112 104 -103 620 -675
fill=1 (-41.06037 13.68679 0.0)
109  0      112 -113 104 -103 620 -675
fill=1 (-13.68679 13.68679 0.0)
110  0      113 -114 104 -103 620 -675
fill=1 ( 13.68679 13.68679 0.0)
111  0      114 -115 104 -103 620 -675
fill=1 ( 41.06037 13.68679 0.0)
112  0 -301 115 -116 104 -103 620 -675
fill=2 ( 68.43395 13.68679 0.0)
214  0 -301 116      104 -103 620 -675
c
215  0 -301      -110 105 -104 620 -675
113  0 -301 110 -111 105 -104 620 -675
fill=4 (-68.43395 -13.68679 0.0)
114  0      111 -112 105 -104 620 -675
fill=1 (-41.06037 -13.68679 0.0)
115  0      112 -113 105 -104 620 -675
fill=1 (-13.68679 -13.68679 0.0)
116  0      113 -114 105 -104 620 -675
fill=1 ( 13.68679 -13.68679 0.0)
117  0      114 -115 105 -104 620 -675
fill=1 ( 41.06037 -13.68679 0.0)
118  0 -301 115 -116 105 -104 620 -675
fill=5 ( 68.43395 -13.68679 0.0)
216  0 -301 116      105 -104 620 -675
c
c    217   0 -301      -110      -105 620 -675
218  0 -301 110 -111 106 -105 620 -675
119  0      111 -112 106 -105 620 -675
fill=4 (-41.06037 -41.06037 0.0)
120  0      112 -113 106 -105 620 -675
fill=1 (-13.68679 -41.06037 0.0)
121  0      113 -114 106 -105 620 -675
fill=1 ( 13.68679 -41.06037 0.0)
122  0      114 -115 106 -105 620 -675
fill=5 ( 41.06037 -41.06037 0.0)

```

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```

219 0 -301 115 -116 106 -105 620 -675
c 220 0 -301 116 -105 620 -675
c
221 0 -301 -111 -106 620 -675
222 0 -301 111 -112 107 -106 620 -675
123 0 -301 112 -113 107 -106 620 -675
fill=4 (-13.68679 -68.43395 0.0)
124 0 -301 113 -114 107 -106 620 -675
fill=5 ( 13.68679 -68.43395 0.0)
223 0 -301 114 -115 107 -106 620 -675
224 0 -301 115 -106 620 -675
c
c 225 0 -301 -112 -107 620 -675
226 0 -301 112 -113 -107 620 -675
227 0 -301 113 -114 -107 620 -675
c 228 0 -301 114 -107 620 -675
c
1001 5 -7.92 301 -302 610 -615 $ MPC shell
1003 5 -7.92 301 -302 615 -616 $ MPC shell
1005 5 -7.92 301 -302 616 -620 $ MPC shell
1007 5 -7.92 301 -302 620 -420 $ MPC shell
1009 5 -7.92 301 -302 420 -430 $ MPC shell
1011 5 -7.92 301 -302 430 -445 $ MPC shell
1013 5 -7.92 301 -302 445 -460 $ MPC shell
1014 5 -7.92 301 -302 460 -675 $ MPC shell
1015 5 -7.92 301 -302 675 -651 $ MPC shell
1017 5 -7.92 301 -302 651 -652 $ MPC shell
1019 5 -7.92 301 -302 652 -653 $ MPC shell
1021 5 -7.92 301 -302 653 -654 $ MPC shell
1023 5 -7.92 301 -302 654 -655 $ MPC shell
1025 5 -7.92 301 -302 655 -656 $ MPC shell
1027 5 -7.92 301 -302 656 -657 $ MPC shell
1028 5 -7.92 301 -302 657 -658 $ MPC shell
1029 5 -7.92 301 -302 658 -659 $ MPC shell
1031 5 -7.92 301 -302 659 -680 $ MPC shell
c
1051 5 -7.92 -301 610 -615 $ MPC baseplate
1052 5 -7.92 -301 615 -616 $ MPC baseplate
1053 5 -7.92 -301 616 -620 $ MPC baseplate
1060 5 -7.92 -301 675 -651 $ MPC lid
1061 5 -7.92 -301 651 -652 $ MPC lid
1062 5 -7.92 -301 652 -653 $ MPC lid
1063 5 -7.92 -301 653 -654 $ MPC lid
1064 5 -7.92 -301 654 -655 $ MPC lid
1065 5 -7.92 -301 655 -656 $ MPC lid
1066 5 -7.92 -301 656 -657 $ MPC lid
1067 5 -7.92 -301 657 -658 $ MPC lid
1068 5 -7.92 -301 658 -659 $ MPC lid
1069 5 -7.92 -301 659 -680 $ MPC lid
c
c overpack universes
c
c pedestal
c
2001 8 -7.82 -302 801 -610
2002 8 -7.82 -302 802 -801
2003 8 -7.82 -302 803 -802
2004 8 -7.82 -302 804 -803
2005 8 -7.82 -302 805 -804
c
2006 7 -2.35 -306 806 -805
2007 7 -2.35 -306 807 -806
2008 7 -2.35 -306 808 -807
2009 7 -2.35 -306 809 -808
2010 7 -2.35 -306 810 -809

```

Attachment 6 to Holtec Letter 5014874

2011	7	-2.35		-306	811	-810
2012	7	-2.35		-306	812	-811
2013	7	-2.35		-306	813	-812
2014	7	-2.35		-306	814	-813
c						
2016	8	-7.82	306	-302	806	-805
2017	8	-7.82	306	-302	807	-806
2028	8	-7.82	306	-302	808	-807
2019	8	-7.82	306	-302	809	-808
2020	8	-7.82	306	-302	810	-809
2021	8	-7.82	306	-302	811	-810
2022	8	-7.82	306	-302	812	-811
2023	8	-7.82	306	-302	813	-812
2024	8	-7.82	306	-302	814	-813
c						
overpack baseplate						
2031	8	-7.82		-302	815	-814
2032	8	-7.82		-302	816	-815
2033	7	-2.35		-302	817	-816
c						
gap between overpack and lid						
3001	9	-1.17e-3		-302	680	-901
c						
c						
lid						
c						
3002	8	-7.82		-307	901	-902
3003	8	-7.82		-307	902	-903
c						
3004	7	-2.35		-305	903	-904
3005	7	-2.35		-305	904	-905
3006	7	-2.35		-305	905	-906
3007	7	-2.35		-305	906	-907
3008	7	-2.35		-305	907	-908
3009	7	-2.35		-305	908	-909
c						
3010	8	-7.82	305	-307	903	-904
3011	8	-7.82	305	-307	904	-905
3012	8	-7.82	305	-307	905	-906
3013	8	-7.82	305	-307	906	-907
3014	8	-7.82	305	-307	907	-908
3015	8	-7.82	305	-307	908	-909
c						
3021	8	-7.82		-307	909	-910
3022	8	-7.82		-307	910	-911
3023	8	-7.82		-307	911	-912
3024	8	-7.82		-307	912	-913
c						
3030	0			-303	913	-914
3031	0			-303	914	-915
3032	0			-303	915	-916
3033	0			-303	916	-917
3034	0			-303	917	-918
c						
3035	8	-7.82	303	-304	913	-914
3036	8	-7.82	303	-304	914	-915
3037	8	-7.82	303	-304	915	-916
3038	8	-7.82	303	-304	916	-917
3039	0		303	-304	917	-918
c						
3040	7	-2.35	304	-307	913	-914
3041	7	-2.35	304	-307	914	-915
3042	7	-2.35	304	-307	915	-916
3043	7	-2.35	304	-307	916	-917
3044	0		304	-307	917	-918
c						
c						
c						
steel,concrete and air in gap between mpc and overpack						

c

4000	7	-2.35	302	-700	817	-816
4001	8	-7.82	302	-700	816	-815
4002	8	-7.82	302	-700	815	-814
4003	9	-1.17e-3	302	-700	814	-813
4004	9	-1.17e-3	302	-700	813	-812
4005	9	-1.17e-3	302	-700	812	-811
4006	9	-1.17e-3	302	-700	811	-810
4007	9	-1.17e-3	302	-700	810	-809
4008	9	-1.17e-3	302	-700	809	-808
4009	9	-1.17e-3	302	-700	808	-807
4010	9	-1.17e-3	302	-700	807	-806
4011	9	-1.17e-3	302	-700	806	-805
4012	9	-1.17e-3	302	-700	805	-804
4013	9	-1.17e-3	302	-700	804	-803
4014	9	-1.17e-3	302	-700	803	-802
4015	9	-1.17e-3	302	-700	802	-801
4016	9	-1.17e-3	302	-700	801	-610
4017	9	-1.17e-3	302	-700	610	-615
4018	9	-1.17e-3	302	-700	615	-620
4019	9	-1.17e-3	302	-700	620	-420
4020	9	-1.17e-3	302	-700	420	-430
4021	9	-1.17e-3	302	-700	430	-445
4022	9	-1.17e-3	302	-700	445	-460
4023	9	-1.17e-3	302	-700	460	-675
4024	9	-1.17e-3	302	-700	675	-651
4025	9	-1.17e-3	302	-700	651	-652
4026	9	-1.17e-3	302	-700	652	-653
4027	9	-1.17e-3	302	-700	653	-654
4028	9	-1.17e-3	302	-700	654	-655
4029	9	-1.17e-3	302	-700	655	-656
4030	9	-1.17e-3	302	-700	656	-657
4031	9	-1.17e-3	302	-700	657	-658
4032	9	-1.17e-3	302	-700	658	-659
4033	9	-1.17e-3	302	-700	659	-680
4034	9	-1.17e-3	302	-700	680	-901
4035	9	-1.17e-3	307	-700	901	-902
4036	9	-1.17e-3	307	-700	902	-903
4037	9	-1.17e-3	307	-700	903	-904
4038	9	-1.17e-3	307	-700	904	-905
4039	9	-1.17e-3	307	-700	905	-906
4040	9	-1.17e-3	307	-700	906	-907
4041	9	-1.17e-3	307	-700	907	-908
4042	9	-1.17e-3	307	-700	908	-909
4043	8	-7.82	307	-700	909	-910
4044	8	-7.82	307	-700	910	-911
4045	8	-7.82	307	-700	911	-912
4046	8	-7.82	307	-700	912	-913
4047	7	-2.35	307	-700	913	-914
4048	7	-2.35	307	-700	914	-915
4049	7	-2.35	307	-700	915	-916
4050	7	-2.35	307	-700	916	-917
4051	0		307	-700	917	-918

c

c importance splitting regions in overpack

c

5000	7	-2.35	700	-701	817	-816
5001	0		700	-701	816	-815 fill=10
5002	0		700	-701	815	-814 fill=10
5003	0		700	-701	814	-813 fill=11
5004	0		700	-701	813	-812 fill=11
5005	0		700	-701	812	-811 fill=11
5006	0		700	-701	811	-810 fill=11
5007	0		700	-701	810	-809 fill=11
5008	0		700	-701	809	-808 fill=11

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5009	0	700	-701	808	-807	fill=11
5010	8	-7.82	700	-701	807	-806
5011	8	-7.82	700	-701	806	-805
5012	8	-7.82	700	-701	805	-804
5013	8	-7.82	700	-701	804	-803
5014	8	-7.82	700	-701	803	-802
5015	8	-7.82	700	-701	802	-801
5016	8	-7.82	700	-701	801	-610
5017	8	-7.82	700	-701	610	-615
5018	8	-7.82	700	-701	615	-620
5019	8	-7.82	700	-701	620	-420
5020	8	-7.82	700	-701	420	-430
5021	8	-7.82	700	-701	430	-445
5022	8	-7.82	700	-701	445	-460
5023	8	-7.82	700	-701	460	-675
5024	8	-7.82	700	-701	675	-651
5025	8	-7.82	700	-701	651	-652
5026	8	-7.82	700	-701	652	-653
5027	8	-7.82	700	-701	653	-654
5028	8	-7.82	700	-701	654	-655
5029	8	-7.82	700	-701	655	-656
5030	8	-7.82	700	-701	656	-657
5031	8	-7.82	700	-701	657	-658
5032	8	-7.82	700	-701	658	-659
5033	8	-7.82	700	-701	659	-680
5034	8	-7.82	700	-701	680	-901
5035	8	-7.82	700	-701	901	-902
5036	8	-7.82	700	-701	902	-903
5037	8	-7.82	700	-701	903	-904
5038	0	700	-701	904	-905	fill=13
5039	0	700	-701	905	-906	fill=13
5040	0	700	-701	906	-907	fill=13
5041	0	700	-701	907	-908	fill=13
5042	0	700	-701	908	-909	fill=13
5043	8	-7.82	700	-701	909	-910
5044	8	-7.82	700	-701	910	-911
5045	8	-7.82	700	-701	911	-912
5046	8	-7.82	700	-701	912	-913
5047	7	-2.35	700	-701	913	-914
5048	7	-2.35	700	-701	914	-915
5049	7	-2.35	700	-701	915	-916
5050	7	-2.35	700	-701	916	-917
5051	0	700	-701	917	-918	
c						
5100	7	-2.35	701	-702	817	-816
5101	0	701	-702	816	-815	fill=10
5102	0	701	-702	815	-814	fill=10
5103	0	701	-702	814	-813	fill=11
5104	0	701	-702	813	-812	fill=11
5105	0	701	-702	812	-811	fill=11
5106	0	701	-702	811	-810	fill=11
5107	0	701	-702	810	-809	fill=11
5108	0	701	-702	809	-808	fill=11
5109	0	701	-702	808	-807	fill=11
5110	8	-7.82	701	-311	807	-806
5111	8	-7.82	701	-311	806	-805
5112	8	-7.82	701	-311	805	-804
5113	8	-7.82	701	-311	804	-803
5114	8	-7.82	701	-311	803	-802
5115	8	-7.82	701	-311	802	-801
5116	8	-7.82	701	-311	801	-610
5117	8	-7.82	701	-311	610	-615
5118	8	-7.82	701	-311	615	-620
5119	8	-7.82	701	-311	620	-420
5120	8	-7.82	701	-311	420	-430



## Attachment 6 to Holtec Letter 5014874

5121	8	-7.82	701	-311	430	-445
5122	8	-7.82	701	-311	445	-460
5123	8	-7.82	701	-311	460	-675
5124	8	-7.82	701	-311	675	-651
5125	8	-7.82	701	-311	651	-652
5126	8	-7.82	701	-311	652	-653
5127	8	-7.82	701	-311	653	-654
5128	8	-7.82	701	-311	654	-655
5129	8	-7.82	701	-311	655	-656
5130	8	-7.82	701	-311	656	-657
5131	8	-7.82	701	-311	657	-658
5132	8	-7.82	701	-311	658	-659
5133	8	-7.82	701	-311	659	-680
5134	8	-7.82	701	-311	680	-901
5135	8	-7.82	701	-311	901	-902
5136	8	-7.82	701	-311	902	-903
5137	8	-7.82	701	-311	903	-904
5138	0		701	-702	904	-905 fill=13
5139	0		701	-702	905	-906 fill=13
5140	0		701	-702	906	-907 fill=13
5141	0		701	-702	907	-908 fill=13
5142	0		701	-702	908	-909 fill=13
5143	8	-7.82	701	-702	909	-910
5144	8	-7.82	701	-702	910	-911
5145	8	-7.82	701	-702	911	-912
5146	8	-7.82	701	-702	912	-913
5147	7	-2.35	701	-702	913	-914
5148	7	-2.35	701	-702	914	-915
5149	7	-2.35	701	-702	915	-916
5150	7	-2.35	701	-702	916	-917
5151	0		701	-702	917	-918
c						
5200	7	-2.35	702	-703	817	-816
5201	0		702	-703	816	-815 fill=10
5202	0		702	-703	815	-814 fill=10
5203	0		702	-703	814	-813 fill=11
5204	0		702	-703	813	-812 fill=11
5205	0		702	-703	812	-811 fill=11
5206	0		702	-703	811	-810 fill=11
5207	0		702	-703	810	-809 fill=11
5208	0		702	-703	809	-808 fill=11
5209	0		702	-703	808	-807 fill=11
5210	0		311	-703	807	-806 fill=112
5211	0		311	-703	806	-805 fill=112
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5213	0		311	-703	804	-803 fill=112
5214	0		311	-703	803	-802 fill=112
5215	0		311	-703	802	-801 fill=112
5216	0		311	-703	801	-610 fill=112
5217	0		311	-703	610	-615 fill=112
5218	0		311	-703	615	-620 fill=112
5219	0		311	-703	620	-420 fill=112
5220	0		311	-703	420	-430 fill=112
5221	0		311	-703	430	-445 fill=112
5222	0		311	-703	445	-460 fill=112
5223	0		311	-703	460	-675 fill=112
5224	0		311	-703	675	-651 fill=112
5225	0		311	-703	651	-652 fill=112
5226	0		311	-703	652	-653 fill=112
5227	0		311	-703	653	-654 fill=112
5228	0		311	-703	654	-655 fill=112
5229	0		311	-703	655	-656 fill=112
5230	0		311	-703	656	-657 fill=112
5231	0		311	-703	657	-658 fill=112
5232	0		311	-703	658	-659 fill=112

## Attachment 6 to Holtec Letter 5014874

5233	0	311	-703	659	-680	fill=112
5234	0	311	-703	680	-901	fill=112
5235	0	311	-703	901	-902	fill=112
5236	0	311	-703	902	-903	fill=112
5237	0	311	-703	903	-904	fill=112
5238	0	702	-703	904	-905	fill=13
5239	0	702	-703	905	-906	fill=13
5240	0	702	-703	906	-907	fill=13
5241	0	702	-703	907	-908	fill=13
5242	0	702	-703	908	-909	fill=13
5243	8	-7.82	702	-703	909	-910
5244	8	-7.82	702	-703	910	-911
5245	8	-7.82	702	-703	911	-912
5246	8	-7.82	702	-703	912	-913
5247	7	-2.35	702	-703	913	-914
5248	7	-2.35	702	-703	914	-915
5249	7	-2.35	702	-703	915	-916
5250	7	-2.35	702	-703	916	-917
5251	0	702	-703	917	-918	
c						
5300	7	-2.35	703	-705	817	-816
5301	0	703	-705	816	-815	fill=10
5302	0	703	-705	815	-814	fill=10
5303	0	703	-705	814	-813	fill=11
5304	0	703	-705	813	-812	fill=11
5305	0	703	-705	812	-811	fill=11
5306	0	703	-705	811	-810	fill=11
5307	0	703	-705	810	-809	fill=11
5308	0	703	-705	809	-808	fill=11
5309	0	703	-705	808	-807	fill=11
5310	0	703	-705	807	-806	fill=112
5311	0	703	-705	806	-805	fill=112
5312	0	703	-705	805	-804	fill=112
5313	0	703	-705	804	-803	fill=112
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5318	0	703	-705	615	-620	fill=112
5319	0	703	-705	620	-420	fill=112
5320	0	703	-705	420	-430	fill=112
5321	0	703	-705	430	-445	fill=112
5322	0	703	-705	445	-460	fill=112
5323	0	703	-705	460	-675	fill=112
5324	0	703	-705	675	-651	fill=112
5325	0	703	-705	651	-652	fill=112
5326	0	703	-705	652	-653	fill=112
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5328	0	703	-705	654	-655	fill=112
5329	0	703	-705	655	-656	fill=112
5330	0	703	-705	656	-657	fill=112
5331	0	703	-705	657	-658	fill=112
5332	0	703	-705	658	-659	fill=112
5333	0	703	-705	659	-680	fill=112
5334	0	703	-705	680	-901	fill=112
5335	0	703	-705	901	-902	fill=112
5336	0	703	-705	902	-903	fill=112
5337	0	703	-705	903	-904	fill=112
5338	0	703	-705	904	-905	fill=13
5339	0	703	-705	905	-906	fill=13
5340	0	703	-705	906	-907	fill=13
5341	0	703	-705	907	-908	fill=13
5342	0	703	-705	908	-909	fill=13
5343	8	-7.82	703	-705	909	-910
5344	8	-7.82	703	-705	910	-911

## Attachment 6 to Holtec Letter 5014874

5345	8	-7.82	703	-705	911	-912	
5346	8	-7.82	703	-705	912	-913	
5347	0		703	-705	913	-914	fill=14
5348	0		703	-705	914	-915	fill=14
5349	0		703	-705	915	-916	fill=14
5350	0		703	-705	916	-917	fill=14
5351	0		703	-705	917	-918	
c							
c	5401	0	704	-705	816	-815	fill=10
c	5402	0	704	-705	815	-814	fill=10
c	5403	0	704	-705	814	-813	fill=11
c	5404	0	704	-705	813	-812	fill=11
c	5405	0	704	-705	812	-811	fill=11
c	5406	0	704	-705	811	-810	fill=11
c	5407	0	704	-705	810	-809	fill=11
c	5408	0	704	-705	809	-808	fill=11
c	5409	0	704	-705	808	-807	fill=11
c	5410	0	704	-705	807	-806	fill=112
c	5411	0	704	-705	806	-805	fill=112
c	5412	0	704	-705	805	-804	fill=112
c	5413	0	704	-705	804	-803	fill=112
c	5414	0	704	-705	803	-802	fill=112
c	5415	0	704	-705	802	-801	fill=112
c	5416	0	704	-705	801	-610	fill=112
c	5417	0	704	-705	610	-615	fill=112
c	5418	0	704	-705	615	-620	fill=112
c	5419	0	704	-705	620	-420	fill=112
c	5420	0	704	-705	420	-430	fill=112
c	5421	0	704	-705	430	-445	fill=112
c	5422	0	704	-705	445	-460	fill=112
c	5423	0	704	-705	460	-675	fill=112
c	5424	0	704	-705	675	-651	fill=112
c	5425	0	704	-705	651	-652	fill=112
c	5426	0	704	-705	652	-653	fill=112
c	5427	0	704	-705	653	-654	fill=112
c	5428	0	704	-705	654	-655	fill=112
c	5429	0	704	-705	655	-656	fill=112
c	5430	0	704	-705	656	-657	fill=112
c	5431	0	704	-705	657	-658	fill=112
c	5432	0	704	-705	658	-659	fill=112
c	5433	0	704	-705	659	-680	fill=112
c	5434	0	704	-705	680	-901	fill=112
c	5435	0	704	-705	901	-902	fill=112
c	5436	0	704	-705	902	-903	fill=112
c	5437	0	704	-705	903	-904	fill=112
c	5438	0	704	-705	904	-905	fill=13
c	5439	0	704	-705	905	-906	fill=13
c	5440	0	704	-705	906	-907	fill=13
c	5441	0	704	-705	907	-908	fill=13
c	5442	0	704	-705	908	-909	fill=13
c	5443	8	-7.82	704	-705	909	-910
c	5444	8	-7.82	704	-705	910	-911
c	5445	8	-7.82	704	-705	911	-912
c	5446	8	-7.82	704	-705	912	-913
c	5447	0	704	-705	913	-914	fill=14
c	5448	0	704	-705	914	-915	fill=14
c	5449	0	704	-705	915	-916	fill=14
c	5450	0	704	-705	916	-917	fill=14
c	5451	0	704	-705	917	-918	
c							
5500	7	-2.35	705	-707	817	-816	
5501	0		705	-707	816	-815	fill=10
5502	0		705	-707	815	-814	fill=10
5503	0		705	-707	814	-813	fill=11
5504	0		705	-707	813	-812	fill=11

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5505	0	705	-707	812	-811	fill=11
5506	0	705	-707	811	-810	fill=11
5507	0	705	-707	810	-809	fill=11
5508	0	705	-707	809	-808	fill=11
5509	0	705	-707	808	-807	fill=11
5510	0	705	-707	807	-806	fill=112
5511	0	705	-707	806	-805	fill=112
5512	0	705	-707	805	-804	fill=112
5513	0	705	-707	804	-803	fill=112
5514	0	705	-707	803	-802	fill=112
5515	0	705	-707	802	-801	fill=112
5516	0	705	-707	801	-610	fill=112
5517	0	705	-707	610	-615	fill=112
5518	0	705	-707	615	-620	fill=112
5519	0	705	-707	620	-420	fill=112
5520	0	705	-707	420	-430	fill=112
5521	0	705	-707	430	-445	fill=112
5522	0	705	-707	445	-460	fill=112
5523	0	705	-707	460	-675	fill=112
5524	0	705	-707	675	-651	fill=112
5525	0	705	-707	651	-652	fill=112
5526	0	705	-707	652	-653	fill=112
5527	0	705	-707	653	-654	fill=112
5528	0	705	-707	654	-655	fill=112
5529	0	705	-707	655	-656	fill=112
5530	0	705	-707	656	-657	fill=112
5531	0	705	-707	657	-658	fill=112
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5535	0	705	-707	901	-902	fill=112
5536	0	705	-707	902	-903	fill=112
5537	0	705	-707	903	-904	fill=112
5538	0	705	-707	904	-905	fill=113
5539	0	705	-707	905	-906	fill=113
5540	0	705	-707	906	-907	fill=113
5541	0	705	-707	907	-908	fill=113
5542	0	705	-707	908	-909	fill=113
5543	8	-7.82	705	-707	909	-910
5544	8	-7.82	705	-707	910	-911
5545	8	-7.82	705	-707	911	-912
5546	8	-7.82	705	-707	912	-913
5547	0	705	-716	913	-914	
5548	0	705	-716	914	-915	
5549	0	705	-716	915	-916	
5550	0	705	-716	916	-917	
5551	0	705	-716	917	-918	
c						
c	5601	0	706	-707	816	-815 fill=10
c	5602	0	706	-707	815	-814 fill=10
c	5603	0	706	-707	814	-813 fill=11
c	5604	0	706	-707	813	-812 fill=11
c	5605	0	706	-707	812	-811 fill=11
c	5606	0	706	-707	811	-810 fill=11
c	5607	0	706	-707	810	-809 fill=11
c	5608	0	706	-707	809	-808 fill=11
c	5609	0	706	-707	808	-807 fill=11
c	5610	0	706	-707	807	-806 fill=112
c	5611	0	706	-707	806	-805 fill=112
c	5612	0	706	-707	805	-804 fill=112
c	5613	0	706	-707	804	-803 fill=112
c	5614	0	706	-707	803	-802 fill=112
c	5615	0	706	-707	802	-801 fill=112
c	5616	0	706	-707	801	-610 fill=112
c	5617	0	706	-707	610	-615 fill=112

## Attachment 6 to Holtec Letter 5014874

c	5618	0	706	-707	615	-620	fill=112
c	5619	0	706	-707	620	-420	fill=112
c	5620	0	706	-707	420	-430	fill=112
c	5621	0	706	-707	430	-445	fill=112
c	5622	0	706	-707	445	-460	fill=112
c	5623	0	706	-707	460	-675	fill=112
c	5624	0	706	-707	675	-651	fill=112
c	5625	0	706	-707	651	-652	fill=112
c	5626	0	706	-707	652	-653	fill=112
c	5627	0	706	-707	653	-654	fill=112
c	5628	0	706	-707	654	-655	fill=112
c	5629	0	706	-707	655	-656	fill=112
c	5630	0	706	-707	656	-657	fill=112
c	5631	0	706	-707	657	-658	fill=112
c	5632	0	706	-707	658	-659	fill=112
c	5633	0	706	-707	659	-680	fill=112
c	5634	0	706	-707	680	-901	fill=112
c	5635	0	706	-707	901	-902	fill=112
c	5636	0	706	-707	902	-903	fill=112
c	5637	0	706	-707	903	-904	fill=112
c	5638	0	706	-707	904	-905	fill=13
c	5639	0	706	-707	905	-906	fill=13
c	5640	0	706	-707	906	-907	fill=13
c	5641	0	706	-707	907	-908	fill=13
c	5642	0	706	-707	908	-909	fill=13
c	5643	8	-7.82	706	-707	909	-910
c	5644	8	-7.82	706	-707	910	-911
c	5645	8	-7.82	706	-707	911	-912
c	5646	8	-7.82	706	-707	912	-913
c	5647	0	706	-707	913	-914	
c	5648	0	706	-707	914	-915	
c	5649	0	706	-707	915	-916	
c	5650	0	706	-707	916	-917	
c	5651	0	706	-707	917	-918	
c							
5700	7	-2.35	707	-709	817	-816	
5701	0		707	-709	816	-815	fill=10
5702	0		707	-709	815	-814	fill=10
5703	0		707	-709	814	-813	fill=11
5704	0		707	-709	813	-812	fill=11
5705	0		707	-709	812	-811	fill=11
5706	0		707	-709	811	-810	fill=11
5707	0		707	-709	810	-809	fill=11
5708	0		707	-709	809	-808	fill=11
5709	0		707	-709	808	-807	fill=11
5710	0		707	-709	807	-806	fill=112
5711	0		707	-709	806	-805	fill=112
5712	0		707	-709	805	-804	fill=112
5713	0		707	-709	804	-803	fill=112
5714	0		707	-709	803	-802	fill=112
5715	0		707	-709	802	-801	fill=112
5716	0		707	-709	801	-610	fill=112
5717	0		707	-709	610	-615	fill=112
5718	0		707	-709	615	-620	fill=112
5719	0		707	-709	620	-420	fill=112
5720	0		707	-709	420	-430	fill=112
5721	0		707	-709	430	-445	fill=112
5722	0		707	-709	445	-460	fill=112
5723	0		707	-709	460	-675	fill=112
5724	0		707	-709	675	-651	fill=112
5725	0		707	-709	651	-652	fill=112
5726	0		707	-709	652	-653	fill=112
5727	0		707	-709	653	-654	fill=112
5728	0		707	-709	654	-655	fill=112
5729	0		707	-709	655	-656	fill=112

## Attachment 6 to Holtec Letter 5014874

5730	0	707	-709	656	-657	fill=112
5731	0	707	-709	657	-658	fill=112
5732	0	707	-709	658	-659	fill=112
5733	0	707	-709	659	-680	fill=112
5734	0	707	-709	680	-901	fill=112
5735	0	707	-709	901	-902	fill=112
5736	0	707	-709	902	-903	fill=112
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5738	0	707	-709	904	-905	fill=113
5739	0	707	-709	905	-906	fill=113
5740	0	707	-709	906	-907	fill=113
5741	0	707	-709	907	-908	fill=113
5742	0	707	-709	908	-909	fill=113
5743	8	-7.82	707	-709	909	-910
5744	8	-7.82	707	-709	910	-911
5745	8	-7.82	707	-709	911	-912
5746	8	-7.82	707	-709	912	-913
c	5747	0	707	-709	913	-914
c	5748	0	707	-709	914	-915
c	5749	0	707	-709	915	-916
c	5750	0	707	-709	916	-917
c	5751	0	707	-709	917	-918
c						
c	5801	0	708	-709	816	-815 fill=10
c	5802	0	708	-709	815	-814 fill=10
c	5803	0	708	-709	814	-813 fill=11
c	5804	0	708	-709	813	-812 fill=11
c	5805	0	708	-709	812	-811 fill=11
c	5806	0	708	-709	811	-810 fill=11
c	5807	0	708	-709	810	-809 fill=11
c	5808	0	708	-709	809	-808 fill=11
c	5809	0	708	-709	808	-807 fill=11
c	5810	0	708	-709	807	-806 fill=112
c	5811	0	708	-709	806	-805 fill=112
c	5812	0	708	-709	805	-804 fill=112
c	5813	0	708	-709	804	-803 fill=112
c	5814	0	708	-709	803	-802 fill=112
c	5815	0	708	-709	802	-801 fill=112
c	5816	0	708	-709	801	-610 fill=112
c	5817	0	708	-709	610	-615 fill=112
c	5818	0	708	-709	615	-620 fill=112
c	5819	0	708	-709	620	-420 fill=112
c	5820	0	708	-709	420	-430 fill=112
c	5821	0	708	-709	430	-445 fill=112
c	5822	0	708	-709	445	-460 fill=112
c	5823	0	708	-709	460	-675 fill=112
c	5824	0	708	-709	675	-651 fill=112
c	5825	0	708	-709	651	-652 fill=112
c	5826	0	708	-709	652	-653 fill=112
c	5827	0	708	-709	653	-654 fill=112
c	5828	0	708	-709	654	-655 fill=112
c	5829	0	708	-709	655	-656 fill=112
c	5830	0	708	-709	656	-657 fill=112
c	5831	0	708	-709	657	-658 fill=112
c	5832	0	708	-709	658	-659 fill=112
c	5833	0	708	-709	659	-680 fill=112
c	5834	0	708	-709	680	-901 fill=112
c	5835	0	708	-709	901	-902 fill=112
c	5836	0	708	-709	902	-903 fill=112
c	5837	0	708	-709	903	-904 fill=112
c	5838	0	708	-709	904	-905 fill=113
c	5839	0	708	-709	905	-906 fill=113
c	5840	0	708	-709	906	-907 fill=113
c	5841	0	708	-709	907	-908 fill=113
c	5842	0	708	-709	908	-909 fill=113

## Attachment 6 to Holtec Letter 5014874

c	5843	8	-7.82	708	-709	909	-910
c	5844	8	-7.82	708	-709	910	-911
c	5845	8	-7.82	708	-709	911	-912
c	5846	8	-7.82	708	-709	912	-913
c	5847	0		708	-709	913	-914
c	5848	0		708	-709	914	-915
c	5849	0		708	-709	915	-916
c	5850	0		708	-709	916	-917
c	5851	0		708	-709	917	-918
c							
5900	7	-2.35		709	-711	817	-816
5901	0			709	-711	816	-815
5902	0			709	-711	815	-814
5903	0			709	-711	814	-813
5904	0			709	-711	813	-812
5905	0			709	-711	812	-811
5906	0			709	-711	811	-810
5907	0			709	-711	810	-809
5908	0			709	-711	809	-808
5909	0			709	-711	808	-807
5910	0			709	-711	807	-806
5911	0			709	-711	806	-805
5912	0			709	-711	805	-804
5913	0			709	-711	804	-803
5914	0			709	-711	803	-802
5915	0			709	-711	802	-801
5916	0			709	-711	801	-610
5917	0			709	-711	610	-615
5918	0			709	-711	615	-620
5919	0			709	-711	620	-420
5920	0			709	-711	420	-430
5921	0			709	-711	430	-445
5922	0			709	-711	445	-460
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5934	0			709	-711	680	-901
5935	0			709	-711	901	-902
5936	0			709	-711	902	-903
5937	0			709	-711	903	-904
5938	0			709	-711	904	-905
5939	0			709	-711	905	-906
5940	0			709	-711	906	-907
5941	0			709	-711	907	-908
5942	0			709	-711	908	-909
5943	8	-7.82		709	-711	909	-910
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5945	8	-7.82		709	-711	911	-912
5946	8	-7.82		709	-711	912	-913
c	5947	0		709	-711	913	-914
c	5948	0		709	-711	914	-915
c	5949	0		709	-711	915	-916
c	5950	0		709	-711	916	-917
c	5951	0		709	-711	917	-918
c							
c	6001	0		710	-711	816	-815
c	6002	0		710	-711	815	-814

## Attachment 6 to Holtec Letter 5014874

c	6003	0	710	-711	814	-813	fill=11
c	6004	0	710	-711	813	-812	fill=11
c	6005	0	710	-711	812	-811	fill=11
c	6006	0	710	-711	811	-810	fill=11
c	6007	0	710	-711	810	-809	fill=11
c	6008	0	710	-711	809	-808	fill=11
c	6009	0	710	-711	808	-807	fill=11
c	6010	0	710	-711	807	-806	fill=112
c	6011	0	710	-711	806	-805	fill=112
c	6012	0	710	-711	805	-804	fill=112
c	6013	0	710	-711	804	-803	fill=112
c	6014	0	710	-711	803	-802	fill=112
c	6015	0	710	-711	802	-801	fill=112
c	6016	0	710	-711	801	-610	fill=112
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c	6018	0	710	-711	615	-620	fill=112
c	6019	0	710	-711	620	-420	fill=112
c	6020	0	710	-711	420	-430	fill=112
c	6021	0	710	-711	430	-445	fill=112
c	6022	0	710	-711	445	-460	fill=112
c	6023	0	710	-711	460	-675	fill=112
c	6024	0	710	-711	675	-651	fill=112
c	6025	0	710	-711	651	-652	fill=112
c	6026	0	710	-711	652	-653	fill=112
c	6027	0	710	-711	653	-654	fill=112
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c	6031	0	710	-711	657	-658	fill=112
c	6032	0	710	-711	658	-659	fill=112
c	6033	0	710	-711	659	-680	fill=112
c	6034	0	710	-711	680	-901	fill=112
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c	6036	0	710	-711	902	-903	fill=112
c	6037	0	710	-711	903	-904	fill=112
c	6038	0	710	-711	904	-905	fill=13
c	6039	0	710	-711	905	-906	fill=13
c	6040	0	710	-711	906	-907	fill=13
c	6041	0	710	-711	907	-908	fill=13
c	6042	0	710	-711	908	-909	fill=13
c	6043	8	-7.82	710	-711	909	-910
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c	6045	8	-7.82	710	-711	911	-912
c	6046	8	-7.82	710	-711	912	-913
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c	6048	0	710	-711	914	-915	
c	6049	0	710	-711	915	-916	
c	6050	0	710	-711	916	-917	
c	6051	0	710	-711	917	-918	
c							
6100	7	-2.35	711	-713	817	-816	
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6102	0	711	-713	815	-814	fill=10	
6103	0	711	-713	814	-813	fill=11	
6104	0	711	-713	813	-812	fill=11	
6105	0	711	-713	812	-811	fill=11	
6106	0	711	-713	811	-810	fill=11	
6107	0	711	-713	810	-809	fill=11	
6108	0	711	-713	809	-808	fill=11	
6109	0	711	-713	808	-807	fill=11	
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6111	0	711	-713	806	-805	fill=112	
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6113	0	711	-713	804	-803	fill=112	
6114	0	711	-713	803	-802	fill=112	



## Attachment 6 to Holtec Letter 5014874

6115	0	711	-713	802	-801	fill=112
6116	0	711	-713	801	-610	fill=112
6117	0	711	-713	610	-615	fill=112
6118	0	711	-713	615	-620	fill=112
6119	0	711	-713	620	-420	fill=112
6120	0	711	-713	420	-430	fill=112
6121	0	711	-713	430	-445	fill=112
6122	0	711	-713	445	-460	fill=112
6123	0	711	-713	460	-675	fill=112
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6128	0	711	-713	654	-655	fill=112
6129	0	711	-713	655	-656	fill=112
6130	0	711	-713	656	-657	fill=112
6131	0	711	-713	657	-658	fill=112
6132	0	711	-713	658	-659	fill=112
6133	0	711	-713	659	-680	fill=112
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6135	0	711	-713	901	-902	fill=112
6136	0	711	-713	902	-903	fill=112
6137	0	711	-713	903	-904	fill=112
6138	0	711	-713	904	-905	fill=113
6139	0	711	-713	905	-906	fill=113
6140	0	711	-713	906	-907	fill=113
6141	0	711	-713	907	-908	fill=113
6142	0	711	-713	908	-909	fill=113
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6146	8	-7.82	711	-713	912	-913
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c	6148	0	711	-713	914	-915
c	6149	0	711	-713	915	-916
c	6150	0	711	-713	916	-917
c	6151	0	711	-713	917	-918
c						
c	6201	0	712	-713	816	-815 fill=10
c	6202	0	712	-713	815	-814 fill=10
c	6203	0	712	-713	814	-813 fill=11
c	6204	0	712	-713	813	-812 fill=11
c	6205	0	712	-713	812	-811 fill=11
c	6206	0	712	-713	811	-810 fill=11
c	6207	0	712	-713	810	-809 fill=11
c	6208	0	712	-713	809	-808 fill=11
c	6209	0	712	-713	808	-807 fill=11
c	6210	0	712	-713	807	-806 fill=112
c	6211	0	712	-713	806	-805 fill=112
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c	6213	0	712	-713	804	-803 fill=112
c	6214	0	712	-713	803	-802 fill=112
c	6215	0	712	-713	802	-801 fill=112
c	6216	0	712	-713	801	-610 fill=112
c	6217	0	712	-713	610	-615 fill=112
c	6218	0	712	-713	615	-620 fill=112
c	6219	0	712	-713	620	-420 fill=112
c	6220	0	712	-713	420	-430 fill=112
c	6221	0	712	-713	430	-445 fill=112
c	6222	0	712	-713	445	-460 fill=112
c	6223	0	712	-713	460	-675 fill=112
c	6224	0	712	-713	675	-651 fill=112
c	6225	0	712	-713	651	-652 fill=112
c	6226	0	712	-713	652	-653 fill=112
c	6227	0	712	-713	653	-654 fill=112

## Attachment 6 to Holtec Letter 5014874

c	6228	0	712	-713	654	-655	fill=112
c	6229	0	712	-713	655	-656	fill=112
c	6230	0	712	-713	656	-657	fill=112
c	6231	0	712	-713	657	-658	fill=112
c	6232	0	712	-713	658	-659	fill=112
c	6233	0	712	-713	659	-680	fill=112
c	6234	0	712	-713	680	-901	fill=112
c	6235	0	712	-713	901	-902	fill=112
c	6236	0	712	-713	902	-903	fill=112
c	6237	0	712	-713	903	-904	fill=112
c	6238	0	712	-713	904	-905	fill=13
c	6239	0	712	-713	905	-906	fill=13
c	6240	0	712	-713	906	-907	fill=13
c	6241	0	712	-713	907	-908	fill=13
c	6242	0	712	-713	908	-909	fill=13
c	6243	8	-7.82	712	-713	909	-910
c	6244	8	-7.82	712	-713	910	-911
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c	6246	8	-7.82	712	-713	912	-913
c	6247	0	712	-713	913	-914	
c	6248	0	712	-713	914	-915	
c	6249	0	712	-713	915	-916	
c	6250	0	712	-713	916	-917	
c	6251	0	712	-713	917	-918	
c							
6300	7	-2.35	713	-714	817	-816	
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6302	0		713	-714	815	-814	fill=10
6303	0		713	-714	814	-813	fill=11
6304	0		713	-714	813	-812	fill=11
6305	0		713	-714	812	-811	fill=11
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6307	0		713	-714	810	-809	fill=11
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6311	0		713	-714	806	-805	fill=112
6312	0		713	-714	805	-804	fill=112
6313	0		713	-714	804	-803	fill=112
6314	0		713	-714	803	-802	fill=112
6315	0		713	-714	802	-801	fill=112
6316	0		713	-714	801	-610	fill=112
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6318	0		713	-714	615	-620	fill=112
6319	0		713	-714	620	-420	fill=112
6320	0		713	-714	420	-430	fill=112
6321	0		713	-714	430	-445	fill=112
6322	0		713	-714	445	-460	fill=112
6323	0		713	-714	460	-675	fill=112
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6330	0		713	-714	656	-657	fill=112
6331	0		713	-714	657	-658	fill=112
6332	0		713	-714	658	-659	fill=112
6333	0		713	-714	659	-680	fill=112
6334	0		713	-714	680	-901	fill=112
6335	0		713	-714	901	-902	fill=112
6336	0		713	-714	902	-903	fill=112
6337	0		713	-714	903	-904	fill=112
6338	0		713	-714	904	-905	fill=13
6339	0		713	-714	905	-906	fill=13

## Attachment 6 to Holtec Letter 5014874

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6342	0	713	-714	908	-909	fill=13
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6344	8	-7.82	713	-714	910	-911
6345	8	-7.82	713	-714	911	-912
6346	8	-7.82	713	-714	912	-913
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c	6348	0	713	-714	914	-915
c	6349	0	713	-714	915	-916
c	6350	0	713	-714	916	-917
c	6351	0	713	-714	917	-918
c						
6400	7	-2.35	714	-715	817	-816
6401	0	714	-715	816	-815	fill=10
6402	0	714	-715	815	-814	fill=10
6403	0	714	-715	814	-813	fill=11
6404	0	714	-715	813	-812	fill=11
6405	0	714	-715	812	-811	fill=11
6406	0	714	-715	811	-810	fill=11
6407	0	714	-715	810	-809	fill=11
6408	0	714	-715	809	-808	fill=11
6409	0	714	-715	808	-807	fill=11
6410	0	714	-715	807	-806	fill=112
6411	0	714	-715	806	-805	fill=112
6412	0	714	-715	805	-804	fill=112
6413	0	714	-715	804	-803	fill=112
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6415	0	714	-715	802	-801	fill=112
6416	0	714	-715	801	-610	fill=112
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6418	0	714	-715	615	-620	fill=112
6419	0	714	-715	620	-420	fill=112
6420	0	714	-715	420	-430	fill=112
6421	0	714	-715	430	-445	fill=112
6422	0	714	-715	445	-460	fill=112
6423	0	714	-715	460	-675	fill=112
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6425	0	714	-715	651	-652	fill=112
6426	0	714	-715	652	-653	fill=112
6427	0	714	-715	653	-654	fill=112
6428	0	714	-715	654	-655	fill=112
6429	0	714	-715	655	-656	fill=112
6430	0	714	-715	656	-657	fill=112
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6433	0	714	-715	659	-680	fill=112
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6435	0	714	-715	901	-902	fill=112
6436	0	714	-715	902	-903	fill=112
6437	0	714	-715	903	-904	fill=112
6438	0	714	-715	904	-905	fill=13
6439	0	714	-715	905	-906	fill=13
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6443	0	714	-715	909	-910	fill=15
6444	0	714	-715	910	-911	fill=15
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c	6448	0	714	-715	914	-915
c	6449	0	714	-715	915	-916
c	6450	0	714	-715	916	-917
c	6451	0	714	-715	917	-918

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c
6500 7 -2.35 715 -716 817 -816
6501 0 715 -716 816 -815 fill=10
6502 0 715 -716 815 -814 fill=10
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6504 0 715 -716 813 -812 fill=11
6505 0 715 -716 812 -811 fill=11
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6508 0 715 -716 809 -808 fill=11
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6532 0 715 -716 658 -659 fill=12
6533 0 715 -716 659 -680 fill=12
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6539 0 715 -716 905 -906 fill=13
6540 0 715 -716 906 -907 fill=13
6541 0 715 -716 907 -908 fill=13
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6544 0 715 -716 910 -911
6545 0 715 -716 911 -912
6546 0 715 -716 912 -913
c 6547 0 715 -716 913 -914
c 6548 0 715 -716 914 -915
c 6549 0 715 -716 915 -916
c 6550 0 715 -716 916 -917
c 6551 0 715 -716 917 -918
c
6601 0 716 -717 816 -815
6602 0 716 -717 815 -814
6603 0 716 -717 814 -813
6604 0 716 -717 813 -812
6605 0 716 -717 812 -811
6606 0 716 -717 811 -810
6607 0 716 -717 810 -809
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6609 0 716 -717 808 -807
6610 0 716 -717 807 -806
6611 0 716 -717 806 -805

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6612	0	716	-717	805	-804	
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6615	0	716	-717	802	-801	
6616	0	716	-717	801	-610	
6617	0	716	-717	610	-615	
6618	0	716	-717	615	-620	
6619	0	716	-717	620	-420	
6620	0	716	-717	420	-430	
6621	0	716	-717	430	-445	
6622	0	716	-717	445	-460	
6623	0	716	-717	460	-675	
6624	0	716	-717	675	-651	
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6646	0	716	-717	912	-913	
6647	0	716	-717	913	-914	
6648	0	716	-717	914	-915	
6649	0	716	-717	915	-916	
6650	0	716	-717	916	-917	
6651	0	716	-717	917	-918	
c						
c overpack universes						
c overpack baseplate						
10001	8	-7.82	371	-372	373	-374 u=10
10002	8	-7.82	374	-362		u=10
10003	8	-7.82	374	363		u=10
10004	9	-1.17e-3	374	362	-363	u=10
10005	8	-7.82	-371	367	-374	u=10
10006	8	-7.82	-371	373	-366	u=10
10007	9	-1.17e-3	-371	366	-367	u=10
10008	8	-7.82	-373	-362		u=10
10009	8	-7.82	-373	363		u=10
10010	9	-1.17e-3	-373	362	-363	u=10
10011	8	-7.82	372	373	-366	u=10
10012	8	-7.82	372	367	-374	u=10
10013	9	-1.17e-3	372	366	-367	u=10
c						
c walls and top of bottom duct						
10101	8	-7.82	361	-362	-932	u=11
10102	8	-7.82	363	-364	-932	u=11
10103	8	-7.82	362	-363	931	-932 u=11
c						
10104	8	-7.82	365	-366	-932	u=11
10105	8	-7.82	367	-368	-932	u=11
10106	8	-7.82	366	-367	931	-932 u=11

```

c      inner and outer shell between bottom ducts
10107 8 -7.82      368 364 -932 315      u=11
10108 8 -7.82      368 -361 -932 315      u=11
10109 8 -7.82      -365 364 -932 315      u=11
10110 8 -7.82      -365 -361 -932 315      u=11
c
10111 8 -7.82      368 364 -932 -310      u=11
10112 8 -7.82      368 -361 -932 -310      u=11
10113 8 -7.82      -365 364 -932 -310      u=11
10114 8 -7.82      -365 -361 -932 -310      u=11
c      concrete and radial plates between bottom ducts
10121 8 -7.82      310 -315 391 -392      -932 u=11
10122 8 -7.82      310 -315 393 -394      -932 u=11
c
10131 7 -2.35      310 -315 394 -365      -932 u=11
10132 7 -2.35      310 -315 368 -391      -932 u=11
10133 7 -2.35      310 -315 392 364      -932 u=11
10134 7 -2.35      310 -315 -361 394      -932 u=11
10135 7 -2.35      310 -315 368 -393      -932 u=11
10136 7 -2.35      310 -315 392 -365      -932 u=11
10137 7 -2.35      310 -315 -391 -361      -932 u=11
10138 7 -2.35      310 -315 364 -393      -932 u=11
c      air and grid spacers in bottom ducts
10141 9 -1.17e-3    362 -363 -931 263 -264      u=11
10142 9 -1.17e-3    366 -367 -931 261 -262      u=11
c
10143 9 -1.17e-3    362 -201 -931 (-263:264) u=11
10144 9 -1.17e-3    206 -363 -931 (-263:264) u=11
10145 9 -1.17e-3    201 -202 -221 (-263:264) u=11
10146 5 -7.92        202 -203 -221 (-273:274) u=11
11146 9 -1.17e-3    202 -203 -221 273 -274 (-263:264) u=11
10147 9 -1.17e-3    203 -204 -221 (-263:264) u=11
10148 5 -7.92        204 -205 -221 (-273:274) u=11
11148 9 -1.17e-3    204 -205 -221 273 -274 (-263:264) u=11
10149 9 -1.17e-3    205 -206 -221 (-263:264) u=11
10150 5 -7.92        201 -202 221 -222 (-263:264) u=11
10151 5 -7.92        202 -203 221 -222 (-263:264) u=11
10152 5 -7.92        203 -204 221 -222 (-263:264) u=11
10153 5 -7.92        204 -205 221 -222 (-263:264) u=11
10154 5 -7.92        205 -206 221 -222 (-263:264) u=11
10155 9 -1.17e-3    201 -202 222 -223 (-263:264) u=11
10156 5 -7.92        202 -203 222 -223 (-263:264) u=11
10157 9 -1.17e-3    203 -204 222 -223 (-263:264) u=11
10158 5 -7.92        204 -205 222 -223 (-263:264) u=11
10159 9 -1.17e-3    205 -206 222 -223 (-263:264) u=11
10160 5 -7.92        201 -202 223 -224 (-263:264) u=11
10161 5 -7.92        202 -203 223 -224 (-263:264) u=11
10162 5 -7.92        203 -204 223 -224 (-263:264) u=11
10163 5 -7.92        204 -205 223 -224 (-263:264) u=11
10164 5 -7.92        205 -206 223 -224 (-263:264) u=11
10165 9 -1.17e-3    201 -202 224 -225 (-263:264) u=11
10166 5 -7.92        202 -203 224 -225 (-263:264) u=11
10167 9 -1.17e-3    203 -204 224 -225 (-263:264) u=11
10168 5 -7.92        204 -205 224 -225 (-263:264) u=11
10169 9 -1.17e-3    205 -206 224 -225 (-263:264) u=11
10170 9 -1.17e-3    201 -206 225 -931 (-263:264) u=11
c
10243 9 -1.17e-3    366 -211 -931 (-261:262) u=11
10244 9 -1.17e-3    216 -367 -931 (-261:262) u=11
10245 9 -1.17e-3    211 -212 -221 (-261:262) u=11
10246 5 -7.92        212 -213 -221 (-261:262) u=11
10247 9 -1.17e-3    213 -214 -221 (-261:262) u=11
10248 5 -7.92        214 -215 -221 (-261:262) u=11
10249 9 -1.17e-3    215 -216 -221 (-261:262) u=11
10250 5 -7.92        211 -212 221 -222 (-261:262) u=11

```

Attachment 6 to Holtec Letter 5014874

10251 5 -7.92 212 -213 221 -222 (-261:262) u=11  
 10252 5 -7.92 213 -214 221 -222 (-261:262) u=11  
 10253 5 -7.92 214 -215 221 -222 (-261:262) u=11  
 10254 5 -7.92 215 -216 221 -222 (-261:262) u=11  
 10255 9 -1.17e-3 211 -212 222 -223 (-261:262) u=11  
 10256 5 -7.92 212 -213 222 -223 (-261:262) u=11  
 10257 9 -1.17e-3 213 -214 222 -223 (-261:262) u=11  
 10258 5 -7.92 214 -215 222 -223 (-261:262) u=11  
 10259 9 -1.17e-3 215 -216 222 -223 (-261:262) u=11  
 10260 5 -7.92 211 -212 223 -224 (-261:262) u=11  
 10261 5 -7.92 212 -213 223 -224 (-261:262) u=11  
 10262 5 -7.92 213 -214 223 -224 (-261:262) u=11  
 10263 5 -7.92 214 -215 223 -224 (-261:262) u=11  
 10264 5 -7.92 215 -216 223 -224 (-261:262) u=11  
 10265 9 -1.17e-3 211 -212 224 -225 (-261:262) u=11  
 10266 5 -7.92 212 -213 224 -225 (-261:262) u=11  
 10267 9 -1.17e-3 213 -214 224 -225 (-261:262) u=11  
 10268 5 -7.92 214 -215 224 -225 (-261:262) u=11  
 10269 9 -1.17e-3 215 -216 224 -225 (-261:262) u=11  
 10270 9 -1.17e-3 211 -216 225 -931 (-261:262) u=11

c

c inner, outer shells and concrete between top and bot ducts

c

10301 8 -7.82 932 -311 u=11  
 10302 8 -7.82 932 315 u=11  
 10303 8 -7.82 932 311 -315 391 -392 u=11  
 10304 8 -7.82 932 311 -315 393 -394 u=11  
 10305 7 -2.35 932 311 -315 394 -391 u=11  
 10306 7 -2.35 932 311 -315 392 394 u=11  
 10307 7 -2.35 932 311 -315 392 -393 u=11  
 10308 7 -2.35 932 311 -315 -391 -393 u=11

c

11302 8 -7.82 315 u=12  
 11303 8 -7.82 -315 391 -392 u=12  
 11304 8 -7.82 -315 393 -394 u=12  
 11305 7 -2.35 -315 394 -391 u=12  
 11306 7 -2.35 -315 392 394 u=12  
 11307 7 -2.35 -315 392 -393 u=12  
 11308 7 -2.35 -315 -391 -393 u=12

c

13303 8 -7.82 391 -392 u=112  
 13304 8 -7.82 393 -394 u=112  
 13305 7 -2.35 394 -391 u=112  
 13306 7 -2.35 392 394 u=112  
 13307 7 -2.35 392 -393 u=112  
 13308 7 -2.35 -391 -393 u=112

c

12301 8 -7.82 -933 -311 u=13  
 12302 8 -7.82 -933 315 u=13  
 12303 8 -7.82 -933 311 -315 391 -392 u=13  
 12304 8 -7.82 -933 311 -315 393 -394 u=13  
 12305 7 -2.35 -933 311 -315 394 -391 u=13  
 12306 7 -2.35 -933 311 -315 392 394 u=13  
 12307 7 -2.35 -933 311 -315 392 -393 u=13  
 12308 7 -2.35 -933 311 -315 -391 -393 u=13

c

top duct bottom plates  
 10309 8 -7.82 933 -934 351 -354 u=13  
 10310 8 -7.82 933 -934 355 -358 u=13

c

top duct walls  
 10311 8 -7.82 934 351 -352 u=13  
 10312 8 -7.82 934 353 -354 u=13  
 10313 8 -7.82 934 355 -356 u=13  
 10314 8 -7.82 934 357 -358 u=13

c

inner and outer shell between top ducts  
 10407 8 -7.82 358 354 933 315 u=13

## Attachment 6 to Holtec Letter 5014874

10408 8 -7.82 358 -351 933 315 u=13  
 10409 8 -7.82 -355 354 933 315 u=13  
 10410 8 -7.82 -355 -351 933 315 u=13  
 c  
 10411 8 -7.82 358 354 933 -310 u=13  
 10412 8 -7.82 358 -351 933 -310 u=13  
 10413 8 -7.82 -355 354 933 -310 u=13  
 10414 8 -7.82 -355 -351 933 -310 u=13  
 c concrete and radial plates next to top ducts  
 10421 8 -7.82 310 -315 391 -392 933 -935 u=13  
 10422 8 -7.82 310 -315 393 -394 933 -935 u=13  
 c  
 10431 7 -2.35 310 -315 394 -355 933 -935 u=13  
 10432 7 -2.35 310 -315 358 -391 933 -935 u=13  
 10433 7 -2.35 310 -315 392 354 933 -935 u=13  
 10434 7 -2.35 310 -315 -351 394 933 -935 u=13  
 10435 7 -2.35 310 -315 358 -393 933 -935 u=13  
 10436 7 -2.35 310 -315 392 -355 933 -935 u=13  
 10437 7 -2.35 310 -315 -391 -351 933 -935 u=13  
 10438 7 -2.35 310 -315 354 -393 933 -935 u=13  
 c  
 c air and grid spacers in top ducts  
 10441 9 -1.17e-3 352 -353 934 263 -264 u=13  
 10442 9 -1.17e-3 356 -357 934 261 -262 u=13  
 c  
 10443 9 -1.17e-3 352 -231 934 (-263:264) u=13  
 10444 9 -1.17e-3 236 -353 934 (-263:264) u=13  
 c  
 10445 9 -1.17e-3 231 -232 934 -251 (-263:264) u=13  
 10446 5 -7.92 232 -233 934 -251 (-263:264) u=13  
 10447 9 -1.17e-3 233 -234 934 -251 (-263:264) u=13  
 10448 5 -7.92 234 -235 934 -251 (-263:264) u=13  
 10449 9 -1.17e-3 235 -236 934 -251 (-263:264) u=13  
 10450 5 -7.92 231 -232 251 -252 (-263:264) u=13  
 10451 5 -7.92 232 -233 251 -252 (-263:264) u=13  
 10452 5 -7.92 233 -234 251 -252 (-263:264) u=13  
 10453 5 -7.92 234 -235 251 -252 (-263:264) u=13  
 10454 5 -7.92 235 -236 251 -252 (-263:264) u=13  
 10455 9 -1.17e-3 231 -232 252 -253 (-263:264) u=13  
 10456 5 -7.92 232 -233 252 -253 (-263:264) u=13  
 10457 9 -1.17e-3 233 -234 252 -253 (-263:264) u=13  
 10458 5 -7.92 234 -235 252 -253 (-263:264) u=13  
 10459 9 -1.17e-3 235 -236 252 -253 (-263:264) u=13  
 10470 9 -1.17e-3 231 -236 253 (-263:264) u=13  
 c  
 10543 9 -1.17e-3 356 -241 934 (-261:262) u=13  
 10544 9 -1.17e-3 246 -357 934 (-261:262) u=13  
 c  
 10545 9 -1.17e-3 241 -242 934 -251 (-261:262) u=13  
 10546 5 -7.92 242 -243 934 -251 (-261:262) u=13  
 10547 9 -1.17e-3 243 -244 934 -251 (-261:262) u=13  
 10548 5 -7.92 244 -245 934 -251 (-261:262) u=13  
 10549 9 -1.17e-3 245 -246 934 -251 (-261:262) u=13  
 10550 5 -7.92 241 -242 251 -252 (-261:262) u=13  
 10551 5 -7.92 242 -243 251 -252 (-261:262) u=13  
 10552 5 -7.92 243 -244 251 -252 (-261:262) u=13  
 10553 5 -7.92 244 -245 251 -252 (-261:262) u=13  
 10554 5 -7.92 245 -246 251 -252 (-261:262) u=13  
 10555 9 -1.17e-3 241 -242 252 -253 (-261:262) u=13  
 10556 5 -7.92 242 -243 252 -253 (-261:262) u=13  
 10557 9 -1.17e-3 243 -244 252 -253 (-261:262) u=13  
 10558 5 -7.92 244 -245 252 -253 (-261:262) u=13  
 10559 9 -1.17e-3 245 -246 252 -253 (-261:262) u=13  
 10570 9 -1.17e-3 241 -246 253 (-261:262) u=13  
 c top plate



Attachment 6 to Holtec Letter 5014874

```

10641  8 -7.82      358 354 935 310 -315  u=13
10642  8 -7.82      358 -351 935 310 -315  u=13
10643  8 -7.82     -355 354 935 310 -315  u=13
10644  8 -7.82     -355 -351 935 310 -315  u=13
c
10701  8 -7.82     -314 u=15
10702  0          314 u=15
c
10711  7 -2.35      -312      u=14
10712  8 -7.82      312 -313 u=14
10713  0          313      u=14
c
99999  0 -817:918:717:(716 -816)
c
c      BLANK LINE

c      BLANK LINE
c
c      MPC surfaces\ / \ / \ / \ / \ /
c
10      px          -12.169775
11      px          -12.017375
12      px          -11.826875
13      px          -11.1125
14      px          11.1125
15      px          11.826875
16      px          12.017375
17      px          12.169775
18      py          -12.169775
19      py          -12.017375
20      py          -11.826875
21      py          -11.1125
22      py          11.1125
23      py          11.826875
24      py          12.017375
25      py          12.169775
c
26      py          -9.525
27      py           9.525
28      px          -9.525
29      px           9.525
30      py          -6.35
31      py           6.35
32      px          -6.35
33      px           6.35
c
35      px          -11.46969
36      px           11.46969
37      py          -11.46969
38      py           11.46969
c
40      px          -10.8204
41      px           10.8204
42      py          -10.8204
43      py           10.8204
c
101     py           82.12074
102     py           54.74716
103     py           27.37358
104     py            0.0
105     py          -27.37358
106     py          -54.74716
107     py          -82.12074
c
116     px           82.12074

```



```

c    bottom duct planes
361 px    -20.955  $ start of item 13
362 px    -19.05  $ end of item 13
363 px     19.05  $ start of item 13
364 px     20.955  $ end of item 13
c
365 py    -20.955  $ start of item 13
366 py    -19.05  $ end of item 13
367 py     19.05  $ start of item 13
368 py     20.955  $ end of item 13
c    cutouts in item 1
371 px    -123.19
372 px     123.19
373 py    -123.19
374 py     123.19
c    item 14
391 1 py    -0.9525  $ steel plate in concrete at 45/225 degrees
392 1 py     0.9525  $ steel plate in concrete at 45/225 degrees
393 1 px    -0.9525  $ steel plate in concrete at 135/315 degrees
394 1 px     0.9525  $ steel plate in concrete at 135/315 degrees
c
c    bottom shielding cross plates
c
201 px    -18.57375
202 px     -6.35
203 px    -5.715
204 px     5.715
205 px     6.35
206 px    18.57375
c
211 py    -18.57375
212 py     -6.35
213 py    -5.715
214 py     5.715
215 py     6.35
216 py    18.57375
c
221 pz    -32.8168
222 pz    -32.1818
223 pz    -24.3586
224 pz    -23.7236
225 pz    -15.9004
c
c    top shielding cross plates
c
231 px    -31.27375
232 px    -10.795
233 px    -10.16
234 px     10.16
235 px     10.795
236 px    31.27375
c
241 py    -31.27375
242 py    -10.795
243 py    -10.16
244 py     10.16
245 py     10.795
246 py    31.27375
c
251 pz     523.24
252 pz     523.875
253 pz     530.86
c    end of cross plates in openings
261 px    -107.315
262 px     107.315

```

263 py -107.315  
 264 py 107.315  
 c end of part of bottom cross plates  
 271 px -124.46  
 272 px 124.46  
 273 py -124.46  
 274 py 124.46  
 c  
 c radial planes in overpack  
 700 cz 93.345 \$ ID of overpack  
 701 cz 95.885  
 702 cz 98.5 \$ slightly diff from 311  
 703 cz 103.505  
 704 cz 108.585  
 705 cz 113.665  
 706 cz 118.745  
 707 cz 123.825  
 708 cz 128.905  
 709 cz 133.985  
 710 cz 139.065  
 711 cz 144.145  
 712 cz 149.225  
 713 cz 154.305  
 714 cz 159.385  
 715 cz 164.465  
 716 cz 168.275  
 717 cz 169.275  
 c  
 c planes in pedestal  
 c  
 801 pz 12.7  
 802 pz 10.16  
 803 pz 7.62  
 804 pz 5.08  
 805 pz 2.54 \$ bottom of item 24  
 806 pz -2.54  
 807 pz -7.62  
 808 pz -12.7  
 809 pz -17.78  
 810 pz -22.86  
 811 pz -27.94  
 812 pz -33.02  
 813 pz -38.1  
 814 pz -40.64 \$ start of item 1  
 815 pz -43.18 \$  
 816 pz -45.72 \$ ground  
 817 pz -76.20  
 c  
 c planes in lid  
 c  
 901 pz 501.65 \$ start of item 6  
 902 pz 502.285 \$ 0.25 inch segment from start  
 903 pz 504.825 \$ end of item 6  
 904 pz 509.905  
 905 pz 513.715  
 906 pz 516.3 \$ end of item 8 plus a little  
 907 pz 521.335  
 908 pz 526.415  
 909 pz 531.495 \$ end of concrete start of item 10  
 910 pz 534.035  
 911 pz 536.575  
 912 pz 539.115  
 913 pz 541.655 \$ end of item 10  
 914 pz 546.735  
 915 pz 551.815

## Attachment 6 to Holtec Letter 5014874

916	pz	556.895	
917	pz	561.975	
918	pz	562.975	
c			
c	planes in overpack		
c			
931	pz	-15.24	\$ bottom of item 11
932	pz	-10.16	\$ top of item 11
933	pz	513.08	\$ bottom of item 8 and top of item 28
934	pz	516.255	\$ top of item 8
935	pz	529.59	\$ start of item 9
c			
c	for tallying		
c			
501	pz	-45.72	
502	pz	-30.48	
503	pz	-15.24	
504	pz	0.00	
505	pz	15.24	
506	pz	30.48	
507	pz	45.72	
508	pz	60.96	
509	pz	76.20	
510	pz	91.44	
511	pz	106.68	
512	pz	121.92	
513	pz	137.16	
514	pz	152.40	
515	pz	167.64	
516	pz	182.88	
517	pz	198.12	
518	pz	213.36	
519	pz	228.60	
520	pz	243.84	
521	pz	259.08	
522	pz	274.32	
523	pz	289.56	
524	pz	304.80	
525	pz	320.04	
526	pz	335.28	
527	pz	350.52	
528	pz	365.76	
529	pz	381.00	
530	pz	396.24	
531	pz	411.48	
532	pz	426.72	
533	pz	441.96	
534	pz	457.20	
535	pz	472.44	
536	pz	487.68	
537	pz	502.92	
538	pz	518.16	
539	pz	533.40	
c			
550	cz	15.24	
551	cz	30.48	
552	cz	45.72	
553	cz	60.96	
554	cz	76.20	
555	cz	91.44	
556	cz	106.68	
557	cz	121.92	
558	cz	137.16	
559	cz	152.40	
560	cz	167.64	

```

c
c   BLANK LINE

c   BLANK LINE
c
*tr1   0 0 0 45   315   90 135   45   90 90 90 0
c
c       PHOTON MATERIALS
c
c   fuel 3.4 w/o U235   10.412 gm/cc
m1      92235.01p   -0.029971
        92238.01p   -0.851529
        8016.01p    -0.1185
c   homogenized fuel density 3.8699 gm/cc
m2      92235.01p   -0.027652
        92238.01p   -0.719715
        8016.01p    -0.100469
        40000.01p   -0.149015
        50000.01p   -0.002587
        26000.01p   -0.000365
        24000.01p   -0.000198
c   zirconium 6.55 gm/cc
m3      40000.01p   1.          $ Zr Clad
c   stainless steel 7.92 gm/cc
m5      24000.01p   -0.19
        25055.01p   -0.02
        26000.01p   -0.695
        28000.01p   -0.095
c   boral 2.644 gm/cc
m6      5010.01p   -0.044226
        5011.01p   -0.201474
        13027.01p   -0.6861
        6000.01p    -0.0682
c   Concrete (NBS Ordinary) @ 2.35 g/cc (Ref: LA-12827-M)
m7      14000.01p   -0.315
        13027.01p   -0.048
        8016.01p    -0.500
        1001.01p    -0.006
        11023.01p   -0.017
        20000.01p   -0.083
        26000.01p   -0.012
        19000.01p   -0.019
c   carbon steel 7.82 gm/cc
m8      6000.01p   -0.005 26000.01p -0.995
c   air density 1.17e-3 gm/cc
m9      7014.01p 0.78 8016.01p 0.22
c
c       NEUTRON MATERIALS
c
c   fuel 3.4 w/o U235   10.412 gm/cc
c   m1      92235.50c   -0.029971
        92238.50c   -0.851529
        8016.50c    -0.1185
c   c   homogenized fuel density 3.8699 gm/cc
c   m2      92235.50c   -0.027652
        92238.50c   -0.719715
        8016.50c    -0.100469
        40000.35c   -0.149015
        50000.35c   -0.002587
        26000.55c   -0.000365
        24000.50c   -0.000198
c   c   helium 1e-4 gm/cc
c   m3      2004.50c 1.0
c   c   stainless steel 7.92 gm/cc
c   m5      24000.50c   -0.19

```

```

c          25055.50c  -0.02
c          26000.55c  -0.695
c          28000.50c  -0.095
c  c      borat  2.644 gm/cc
c  m6      5010.50c  -0.044226
c          5011.56c  -0.201474
c          13027.50c  -0.6861
c          6000.50c  -0.0682
c  c      Concrete (NBS Ordinary) @ 2.35 g/cc (Ref: LA-12827-M)
c  m7      14000.50c  -0.315
c          13027.50c  -0.048
c          8016.50c  -0.500
c          1001.50c  -0.006
c          11023.50c  -0.017
c          20000.50c  -0.083
c          26000.55c  -0.012
c          19000.50c  -0.019
c  mt7     lwtr.01t
c  c      carbon steel 7.82 gm/cc
c  m8      6000.50c -0.005 26000.55c -0.995
c  c      air density 1.17e-3 gm/cc
c  m9      7014.50c 0.78 8016.50c 0.22
c
phys:n    20 0.0
phys:p    100 0
c  imp:n   1 228r 0
c  imp:p   1 228r 0
nps      13500000
prdmp    j  -60  1  2
c  print   10 110 160 161 20 170
print
mode p
ssw      716 917
c
sdef par=2  erg=d1  axs=0 0 1  x=d4  y=fx  d5  z=d3
c
c  energy dist for gammas in the fuel
c
c  si1  h  0.7 1.0 1.5 2.0 2.5 3.0
c  sp1   0  0.43 0.27 0.22 0.04 0.04
c
c  energy dist for neutrons in the fuel
c
c  si1  h  0.1 0.4 0.9 1.4 1.85 3.0 6.43 20.0
c  sp1   0  0.03787 0.1935 0.1773 0.1310 0.2320 0.2098 0.01853
c
c  energy dist for Co60 gammas
c
c  si1  1 1.3325 1.1732
c  sp1  0.5 0.5
c
c  axial dist for phot in fuel
c
c  si3  h  53.2765 68.5165 83.7565 114.2365 175.1965 236.1565
c          297.1165 358.0765 388.5565 403.7965 419.0365
c  sp3  0 0.022854 0.035321 0.08975 0.184167 0.183 0.179833
c          0.175017 0.080033 0.030575 0.019458
c  sb3  0 1 1 1 1 1 1 1 1 1
c
c  axial dist for Co60 - a zero prob is in the fuel
c
c  si3  h  32.004 50.7365 419.0365 428.72025 439.83275 452.6915
c  sp3  0 0.44 0.0 0.05 0.05 0.46
c  sb3  0 0.50 0.0 0.05 0.10 0.35
c

```

```

si4  s          13 14
           12 13 14 15
        11 12 13 14 15 16
        11 12 13 14 15 16
           12 13 14 15
           13 14

sp4  1 23r
c
ds5  s          26 26
           25 25 25 25
        24 24 24 24 24 24
        23 23 23 23 23 23
           22 22 22 22
           21 21

c
si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639
si14  2.86639  24.50719
si15  30.23997  51.88077
si16  57.61355  79.25435

c
si21 -79.25435 -57.61355
si22 -51.88077 -30.23997
si23 -24.50719 -2.86639
si24  2.86639  24.50719
si25  30.23997  51.88077
si26  57.61355  79.25435

c
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1

c
#  imp:p
314  2
315  2
316  2
317  2
318  2
319  2
320  2
321  2
346  2
353  2
360  2
361  2
362  2
363  2
364  2
365  2
366  2
367  2
414  2
415  2
416  2
417  2

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Attachment 6 to Holtec Letter 5014874

418	2
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420	2
421	2
446	2
453	2
460	2
461	2
462	2
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465	2
466	2
467	2
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567	2
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617	2
618	2
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620	2
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720	2
721	2
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753	2
760	2
761	2
762	2
763	2
764	2

Attachment 6 to Holtec Letter 5014874

765	2
766	2
767	2
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Attachment 6 to Holtec Letter 5014874

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729	1
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755	1
768	1
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770	1
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774	1
775	1
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Attachment 6 to Holtec Letter 5014874

483	2
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778	2
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781	2
782	2
783	2
349	4
358	4
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386	4
387	4
388	4

Attachment 6 to Holtec Letter 5014874

389	4
390	4
391	4
449	4
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485	4
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785	4
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789	4
790	4
791	4
351	8
451	8
551	8
651	8
751	8
1051	16
1052	8
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1060	24
1061	24
1062	72
1063	72
1064	216
1065	216
1066	648
1067	648
1068	1944
1069	1944
1001	16
1003	8
1005	8
1007	4

Attachment 6 to Holtec Letter 5014874

1009	2
1011	4
1013	8
1014	16
1015	48
1017	24
1019	72
1021	72
1023	216
1025	216
1027	648
1028	648
1029	1944
1031	1944
c	
2001	32
2002	64
2003	128
2004	256
2005	512
2006	1024
2007	1024
2008	2048
2009	2048
2010	4096
2011	4096
2012	8192
2013	8192
2014	16384
c	
2016	1024
2017	1024
2028	2048
2019	2048
2020	4096
2021	4096
2022	8192
2023	8192
2024	16384
c	
2031	16384
2032	16384
2033	16384
c	
3001	3888
3002	3888
3003	3888
c	
3004	7776
3005	7776
3006	7776
3007	7776
3008	15552
3009	15552
c	
3010	7776
3011	7776
3012	7776
3013	7776
3014	15552
3015	15552
c	
3021	46656
3022	46656
3023	139968

Attachment 6 to Holtec Letter 5014874

3024	139968
c	
3030	1
3031	1
3032	1
3033	1
3034	1
c	
3035	279936
3036	279936
3037	559872
3038	559872
3039	1
c	
3040	279936
3041	279936
3042	559872
3043	559872
3044	1
c	
4000	32768
4001	32768
4002	32768
4003	32768
4004	16384
4005	16384
4006	8192
4007	8192
4008	4096
4009	4096
4010	2048
4011	2048
4012	1024
4013	512
4014	256
4015	128
4016	64
4017	32
4018	16
4019	8
4020	4
4021	8
4022	16
4023	32
4024	96
4025	48
4026	144
4027	144
4028	432
4029	432
4030	1296
4031	1296
4032	3888
4033	3888
4034	7776
4035	7776
4036	7776
4037	15552
4038	15552
4039	15552
4040	15552
4041	31104
4042	31104
4043	93312
4044	93312

Attachment 6 to Holtec Letter 5014874

4045	279936
4046	279936
4047	559872
4048	559872
4049	1119744
4050	1119744
4051	1

c

5000	32768
5001	32768
5002	32768
5003	32768
5004	16384
5005	16384
5006	8192
5007	8192
5008	4096
5009	4096
5010	2048
5011	2048
5012	1024
5013	512
5014	256
5015	128
5016	64
5017	32
5018	16
5019	8
5020	4
5021	8
5022	16
5023	32
5024	96
5025	48
5026	144
5027	144
5028	432
5029	432
5030	1296
5031	1296
5032	3888
5033	3888
5034	7776
5035	7776
5036	7776
5037	15552
5038	15552
5039	15552
5040	15552
5041	31104
5042	31104
5043	93312
5044	93312
5045	279936
5046	279936
5047	559872
5048	559872
5049	1119744
5050	1119744
5051	1

c

5100	65536
5101	65536
5102	65536
5103	65536



Attachment 6 to Holtec Letter 5014874

5104	32768
5105	32768
5106	16384
5107	16384
5108	8192
5109	8192
5110	4096
5111	4096
5112	2048
5113	1024
5114	512
5115	256
5116	128
5117	64
5118	32
5119	16
5120	8
5121	16
5122	32
5123	64
5124	192
5125	96
5126	288
5127	288
5128	864
5129	864
5130	2592
5131	2592
5132	7776
5133	7776
5134	15552
5135	15552
5136	15552
5137	31104
5138	31104
5139	31104
5140	31104
5141	62208
5142	62208
5143	186624
5144	186624
5145	559872
5146	559872
5147	1119744
5148	1119744
5149	2239488
5150	2239488
5151	1
c	
5200	131072
5201	131072
5202	131072
5203	131072
5204	65536
5205	65536
5206	32768
5207	32768
5208	16384
5209	16384
5210	8192
5211	8192
5212	4096
5213	2048
5214	1024
5215	512

Attachment 6 to Holtec Letter 5014874

5216	256
5217	128
5218	64
5219	32
5220	16
5221	32
5222	64
5223	128
5224	384
5225	192
5226	576
5227	576
5228	1728
5229	1728
5230	5184
5231	5184
5232	15552
5233	15552
5234	31104
5235	31104
5236	31104
5237	62208
5238	62208
5239	62208
5240	62208
5241	124416
5242	124416
5243	373248
5244	373248
5245	1119744
5246	1119744
5247	2239488
5248	2239488
5249	4478976
5250	4478976
5251	1
c	
5300	262144
5301	262144
5302	262144
5303	262144
5304	131072
5305	131072
5306	65536
5307	65536
5308	32768
5309	32768
5310	16384
5311	16384
5312	8192
5313	4096
5314	2048
5315	1024
5316	512
5317	256
5318	128
5319	64
5320	32
5321	64
5322	128
5323	256
5324	768
5325	384
5326	1152
5327	1152

Attachment 6 to Holtec Letter 5014874

5328	3456
5329	3456
5330	10368
5331	10368
5332	31104
5333	31104
5334	62208
5335	62208
5336	62208
5337	124416
5338	124416
5339	124416
5340	124416
5341	248832
5342	248832
5343	746496
5344	746496
5345	2239488
5346	2239488
5347	4478976
5348	4478976
5349	8957952
5350	8957952
5351	1
c	
5500	524288
5501	524288
5502	524288
5503	524288
5504	262144
5505	262144
5506	131072
5507	131072
5508	65536
5509	65536
5510	32768
5511	32768
5512	16384
5513	8192
5514	4096
5515	2048
5516	1024
5517	512
5518	256
5519	128
5520	64
5521	128
5522	256
5523	512
5524	1536
5525	768
5526	2304
5527	2304
5528	6912
5529	6912
5530	20736
5531	20736
5532	62208
5533	62208
5534	124416
5535	124416
5536	124416
5537	248832
5538	248832
5539	248832

Attachment 6 to Holtec Letter 5014874

5540	248832
5541	497664
5542	497664
5543	1492992
5544	1492992
5545	4478976
5546	4478976
5547	1
5548	1
5549	1
5550	1
5551	1

c

5700	1048576
5701	1048576
5702	1048576
5703	1048576
5704	524288
5705	524288
5706	262144
5707	262144
5708	131072
5709	131072
5710	65536
5711	65536
5712	32768
5713	16384
5714	8192
5715	4096
5716	2048
5717	1024
5718	512
5719	256
5720	128
5721	256
5722	512
5723	1024
5724	3072
5725	1536
5726	4608
5727	4608
5728	13824
5729	13824
5730	41472
5731	41472
5732	124416
5733	124416
5734	248832
5735	248832
5736	248832
5737	497664
5738	497664
5739	497664
5740	497664
5741	995328
5742	995328
5743	2985984
5744	2985984
5745	8957952
5746	8957952

c

5900	2097152
5901	2097152
5902	2097152
5903	2097152

Attachment 6 to Holtec Letter 5014874

5904	1048576
5905	1048576
5906	524288
5907	524288
5908	262144
5909	262144
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5911	131072
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5913	32768
5914	16384
5915	8192
5916	4096
5917	2048
5918	1024
5919	512
5920	256
5921	512
5922	1024
5923	2048
5924	6144
5925	3072
5926	9216
5927	9216
5928	27648
5929	27648
5930	82944
5931	82944
5932	248832
5933	248832
5934	497664
5935	497664
5936	497664
5937	995328
5938	995328
5939	995328
5940	995328
5941	1990656
5942	1990656
5943	5971968
5944	5971968
5945	17915904
5946	17915904
c	
6100	4194304
6101	4194304
6102	4194304
6103	4194304
6104	2097152
6105	2097152
6106	1048576
6107	1048576
6108	524288
6109	524288
6110	262144
6111	262144
6112	131072
6113	65536
6114	32768
6115	16384
6116	8192
6117	4096
6118	2048
6119	1024
6120	512

Attachment 6 to Holtec Letter 5014874

6121	1024
6122	2048
6123	4096
6124	12288
6125	6144
6126	18432
6127	18432
6128	55296
6129	55296
6130	165888
6131	165888
6132	497664
6133	497664
6134	995328
6135	995328
6136	995328
6137	1990656
6138	1990656
6139	1990656
6140	1990656
6141	3981312
6142	3981312
6143	11943936
6144	11943936
6145	35831808
6146	35831808
c	
6300	8388608
6301	8388608
6302	8388608
6303	8388608
6304	4194304
6305	4194304
6306	2097152
6307	2097152
6308	1048576
6309	1048576
6310	524288
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6312	262144
6313	131072
6314	65536
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6322	4096
6323	8192
6324	24576
6325	12288
6326	36864
6327	36864
6328	110592
6329	110592
6330	331776
6331	331776
6332	995328
6333	995328
6334	1990656
6335	1990656
6336	1990656
6337	3981312

Attachment 6 to Holtec Letter 5014874

6338	3981312
6339	3981312
6340	3981312
6341	7962624
6342	7962624
6343	23887872
6344	23887872
6345	71663616
6346	71663616
c	
6400	8388608
6401	8388608
6402	8388608
6403	8388608
6404	4194304
6405	4194304
6406	2097152
6407	2097152
6408	1048576
6409	1048576
6410	524288
6411	524288
6412	262144
6413	131072
6414	65536
6415	32768
6416	16384
6417	8192
6418	4096
6419	2048
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6422	4096
6423	8192
6424	24576
6425	12288
6426	36864
6427	36864
6428	110592
6429	110592
6430	331776
6431	331776
6432	995328
6433	995328
6434	1990656
6435	1990656
6436	1990656
6437	3981312
6438	3981312
6439	3981312
6440	3981312
6441	7962624
6442	7962624
6443	23887872
6444	23887872
6445	71663616
6446	71663616
c	
6500	16777216
6501	16777216
6502	16777216
6503	16777216
6504	8388608
6505	8388608
6506	4194304

Attachment 6 to Holtec Letter 5014874

6507	4194304
6508	2097152
6509	2097152
6510	1048576
6511	1048576
6512	524288
6513	262144
6514	131072
6515	65536
6516	32768
6517	16384
6518	8192
6519	4096
6520	2048
6521	4096
6522	8192
6523	16384
6524	49152
6525	24576
6526	73728
6527	73728
6528	221184
6529	221184
6530	663552
6531	663552
6532	1990656
6533	1990656
6534	3981312
6535	3981312
6536	3981312
6537	7962624
6538	7962624
6539	7962624
6540	7962624
6541	15925248
6542	15925248
6543	47775744
6544	47775744
6545	143327232
6546	143327232
c	
6601	1
6602	1
6603	1
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6613	1
6614	1
6615	1
6616	1
6617	1
6618	1
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6620	1
6621	1
6622	1
6623	1
6624	1



Attachment 6 to Holtec Letter 5014874

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6629	1
6630	1
6631	1
6632	1
6633	1
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6635	1
6636	1
6637	1
6638	1
6639	1
6640	1
6641	1
6642	1
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6644	1
6645	1
6646	1
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c      9.26e-5 1.32e-4 1.25e-4 1.56e-4 1.47e-4 1.47e-4 2.08e-4 2.27e-4
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c      photon dose factors
c
c      0.01 0.03 0.05 0.07 0.1 0.15 0.2 0.25 0.3 0.35 0.4 0.45
c      0.5 0.55 0.6 0.65 0.7 0.8 1.0 1.4 1.8 2.2 2.6 2.8 3.25
c      3.75 4.25 4.75 5.0 5.25 5.75 6.25 6.75 7.5 9.0 11.0
c      13.0 15.0
c      3.96e-06 5.82e-07 2.90e-07 2.58e-07 2.83e-07 3.79e-07 5.01e-07
c      6.31e-07 7.59e-07 8.78e-07 9.85e-07 1.08e-06 1.17e-06 1.27e-06
c      1.36e-06 1.44e-06 1.52e-06 1.68e-06 1.98e-06 2.51e-06 2.99e-06
c      3.42e-06 3.82e-06 4.01e-06 4.41e-06 4.83e-06 5.23e-06 5.60e-06
c      5.80e-06 6.01e-06 6.37e-06 6.74e-06 7.11e-06 7.66e-06 8.77e-06
c      1.03e-05 1.18e-05 1.33e-05
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c      PHOTON TALLIES
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c      ft102  scx 3
c      de102  0.01 0.03 0.05 0.07 0.1 0.15 0.2 0.25 0.3 0.35 0.4 0.45
c      0.5 0.55 0.6 0.65 0.7 0.8 1.0 1.4 1.8 2.2 2.6 2.8 3.25
c      3.75 4.25 4.75 5.0 5.25 5.75 6.25 6.75 7.5 9.0 11.0
c      13.0 15.0
c      df102  3.96e-06 5.82e-07 2.90e-07 2.58e-07 2.83e-07 3.79e-07 5.01e-07
c      6.31e-07 7.59e-07 8.78e-07 9.85e-07 1.08e-06 1.17e-06 1.27e-06
c      1.36e-06 1.44e-06 1.52e-06 1.68e-06 1.98e-06 2.51e-06 2.99e-06
c      3.42e-06 3.82e-06 4.01e-06 4.41e-06 4.83e-06 5.23e-06 5.60e-06
c      5.80e-06 6.01e-06 6.37e-06 6.74e-06 7.11e-06 7.66e-06 8.77e-06
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**APPENDIX 5.D**

**DOSE RATE COMPARISON FOR DIFFERENT COBALT IMPURITY LEVELS**



The dose rate adjacent to and one meter from the 100-ton HI-TRAC and the HI-STORM overpack are presented on Tables 5.D.1 through 5.D.4 for the MPC-24 with different burnup and cooling times and different assumed Cobalt-59 impurity levels for inconel. The HI-TRAC results were calculated for an earlier design which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below. The following burnup and cooling time combinations are presented.

#### 100-ton HI-TRAC

- 35,000 MWD/MTU and 5 year cooling  
1000 ppm (1.0 gm/kg) Cobalt-59 impurity in inconel
- 45,000 MWD/MTU and 9 year cooling  
4700 ppm (4.7 gm/kg) Cobalt-59 impurity in inconel

#### HI-STORM

- 45,000 MWD/MTU and 5 year cooling  
1000 ppm (1.0 gm/kg) Cobalt-59 impurity in inconel
- 45,000 MWD/MTU and 9 year cooling  
4700 ppm (4.7 gm/kg) Cobalt-59 impurity in inconel

On Tables 5.D.1 through 5.D.4, the contribution to the dose rate from activation in incore grid spacers is explicitly shown.

These results demonstrate that the dose rates at the longer cooling time are essentially equivalent to (within 11%) or bounded by the dose rates at the shorter cooling times even though a very conservative Cobalt-59 impurity level of 4700 ppm was assumed for the longer cooling times.

Table 5.2.1 shows the masses of inconel and steel that are used in the modeling of the PWR fuel assembly. When 4700 ppm was used for the impurity level in the inconel, an effective Cobalt-59 impurity level was used for the regions containing both steel and inconel. The following table summarizes the impurity levels that were used.

Region	Regional Co-59 impurity when 1000 ppm in inconel assumed	Regional Co-59 impurity when 4700 ppm in inconel assumed
Lower end fitting	1000 ppm	1340 ppm
Incore grid spacers	1000 ppm	4700 ppm
Gas plenum springs	1000 ppm	3417 ppm
Gas plenum spacer	1000 ppm	3417 ppm
Upper end fitting	1000 ppm	1000 ppm

Table 5.D.1

DOSE RATES ADJACENT TO 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point † Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in inconel</b>						
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>						
1	12.20	11.89	12.65	507.16	171.98	715.87
2	345.15	332.90	52.81	0.75	88.26	819.87
3	4.72	5.19	1.98	369.24	214.70	595.83
4	7.13	8.11	0.97	197.76	180.55	394.52
5 (pool lid)	48.63	54.68	17.52	2557.68	1194.17	3872.68
5(transfer lid)	137.52	155.17	0.97	3811.45	674.42	4779.53
<b>1000 ppm Co-59 in inconel</b>						
<b>35,000 MWD/MTU AND 5-YEAR COOLING</b>						
1	3.73	27.05	6.51	543.06	88.57	668.92
2	105.37	696.56	27.19	0.80	45.46	875.38
3	1.44	11.44	1.02	473.51	110.57	597.98
4	2.18	17.87	0.50	241.22	92.97	354.74
5 (pool lid)	28.09	186.49	8.27	2751.19	554.51	3528.55
5(transfer lid)	41.98	293.57	0.50	4081.28	347.33	4764.66

† Refer to Figure 5.1.4.

Table 5.D.2

DOSE RATES AT 1 METER FROM 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point † Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in inconel</b>						
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>						
1	44.32	43.03	7.14	73.99	27.98	196.46
2	148.49	143.94	16.40	6.18	32.38	347.39
3	18.76	18.25	3.88	72.75	13.83	127.47
4	2.19	2.80	0.17	61.70	44.87	111.73
5(transfer lid)	55.57	64.25	0.18	1556.99	188.18	1865.16
<b>1000 ppm Co-59 in inconel</b>						
<b>35,000 MWD/MTU AND 5-YEAR COOLING</b>						
1	13.53	91.20	3.68	79.16	14.41	201.97
2	45.33	302.99	8.44	6.13	16.68	379.57
3	5.73	38.74	2.00	71.95	7.12	125.54
4	0.67	6.21	0.09	74.47	23.11	104.55
5(transfer lid)	16.96	128.14	0.09	1667.22	96.91	1909.32

† Refer to Figure 5.1.4.

Table 5.D.3

DOSE RATES ADJACENT TO HI-STORM OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point † Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas †† (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in inconel</b>					
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
1	0.54	2.95	3.86	2.37	9.72
2	7.73	10.40	0.03	1.48	19.63
3	0.36	1.97	2.59	1.18	6.11
4	0.10	0.53	0.33	3.10	4.06
<b>1000 ppm Co-59 in inconel</b>					
<b>45,000 MWD/MTU AND 5-YEAR COOLING</b>					
1	0.20	5.68	4.87	2.76	13.51
2	2.73	28.93	0.03	1.88	33.58
3	3.87	0.13	3.21	1.38	8.59
4	0.04	0.91	0.36	3.60	4.91

† Refer to Figures 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.D.4

DOSE RATES ONE METER FROM HI-STORM OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point † Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas †† (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in inconel</b>					
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
1	0.77	2.01	2.30	0.46	5.54
2	3.90	5.14	0.39	0.64	10.08
3	0.44	1.09	1.72	0.18	3.42
4	0.05	0.23	0.14	0.94	1.37
<b>1000 ppm Co-59 in inconel</b>					
<b>45,000 MWD/MTU AND 5-YEAR COOLING</b>					
1	0.28	4.49	2.90	0.54	8.21
2	1.41	14.98	0.25	0.78	17.42
3	0.16	2.57	2.09	0.21	5.03
4	0.02	0.42	0.16	1.10	1.70

† Refer to Figures 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

**APPENDIX 5.E**

**DELETED**

## **APPENDIX 5.F**

### **Additional Information on the Burnup Versus Decay Heat and Enrichment Equation**

The equation in Section 5.2.5.3 was determined to be the best equation capable of reproducing the burnup versus enrichment and decay heat data calculated with ORIGEN-S. As an example, Figure 5.F.1 graphically presents ORIGEN-S burnup versus decay heat data for various enrichments for the 9x9C/D fuel assembly array/classes with 20 years cooling time. In these calculations, the decay heat data from ORIGEN-S are conservatively adjusted for uncertainty of 5%, as discussed in Section 5.2. This data could also be represented graphically as a surface on a three dimensional plot. However, the 2D plot is easier to visualize. Additional enrichments were used in the ORIGEN-S calculations and have been omitted for clarity.

Figures 5.F.2 through 5.F.4 show ORIGEN-S burnup versus decay heat data for specific enrichments. In addition to the ORIGEN-S data, these figures present the results of the best fit curve using the least squares method and Equation 5.2. This curve is referred here to as the “original fit.” Since in the original fit, some burnups are higher than the burnups in the ORIGEN-S file, the original fit is adjusted by modifying the constant coefficient “G” such that all burnups are conservatively bounded by ORIGEN-S burnups. This curve is referred to as the adjusted curve fit. Table 5.F.1 below shows the equation coefficients used for both curve fits. As these figures indicate, the original curve fit faithfully reproduces the ORIGEN-S data.

Figure 5.F.5 provides a different representation of the curve fit versus ORIGEN-S comparison. This figure was generated by taking the ORIGEN-S enrichment and decay heat data from Figure 5.F.1 for a constant burnup of 30,000 MWD/MTU and calculating the burnup using the fitted equation with coefficients from Table 5.F.1. The resulting burnup versus enrichment is plotted. Table 5.F.2 presents the ORIGEN-S and curve fit data in tabular form used to generate Figure 5.F.5. Since the ORIGEN-S calculations were performed for a specific burnup of 30,000 MWD/MTU, the ORIGEN-S data is represented as a straight line. Figures 5.F.6 and 5.F.7 provide the same representation for burnups of 45,000 and 65,000 MWD/MTU. These results also indicate that the non-adjusted curve fit provides a very good representation of the ORIGEN-S data. It is also clear that the adjusted curve fit always bounds the ORIGEN-S data by predicting a lower burnup which results in a more restrictive and conservative limit for the user.



Table 5.F.1

COEFFICIENTS FOR EQUATION IN SECTION 5.2.5.3 FOR THE 9X9C/D FUEL  
ASSEMBLY ARRAY/CLASSES WITH A COOLING TIME OF 20 YEARS

<b>Coefficient</b>	<b>Original Curve Fit</b>	<b>Adjusted Curve Fit</b>
A	238022	238022
B	-324930	-324930
C	227066	227066
D	-158.319	-158.32
E	7284.25	7284.25
F	5103.45	5103.45
G	-985.96	-1464.16

Table 5.F.2

**ORIGEN-S AND CURVE FIT DATA FOR THE 9X9C/D FUEL ASSEMBLY  
ARRAY/CLASSES  
WITH A COOLING TIME OF 20 YEARS**

<b>Specified Enrichment</b>	<b>ORIGEN-S calculated decay heat per assembly (kw)<sup>1</sup></b>	<b>ORIGEN-S calculated burnup (MWD/MTU)</b>	<b>Burnup calculated with original curve fit (MWD/MTU)<sup>2</sup></b>	<b>Burnup calculated with adjusted curve fit (MWD/MTU)</b>
0.7	1.55E-01	30000	29792	29314
1	1.53E-01	30000	29763	29284
1.35	1.52E-01	30000	29767	29289
1.7	1.50E-01	30000	29829	29351
2	1.50E-01	30000	29962	29484
2.3	1.49E-01	30000	30009	29531
2.6	1.49E-01	30000	30080	29602
2.9	1.49E-01	30000	30198	29720
3.2	1.50E-01	30000	30327	29848
3.4	1.50E-01	30000	30358	29880
3.6	1.49E-01	30000	30200	29721
3.9	1.48E-01	30000	30162	29684
4.2	1.48E-01	30000	30115	29636
4.5	1.48E-01	30000	30059	29581
4.8	1.48E-01	30000	29996	29518
5	1.49E-01	30000	30008	29530

<sup>1</sup> Decay heat from ORIGEN-S is increased by uncertainty as discussed in Section 5.2

<sup>2</sup> Burnups are rounded down to whole integers.

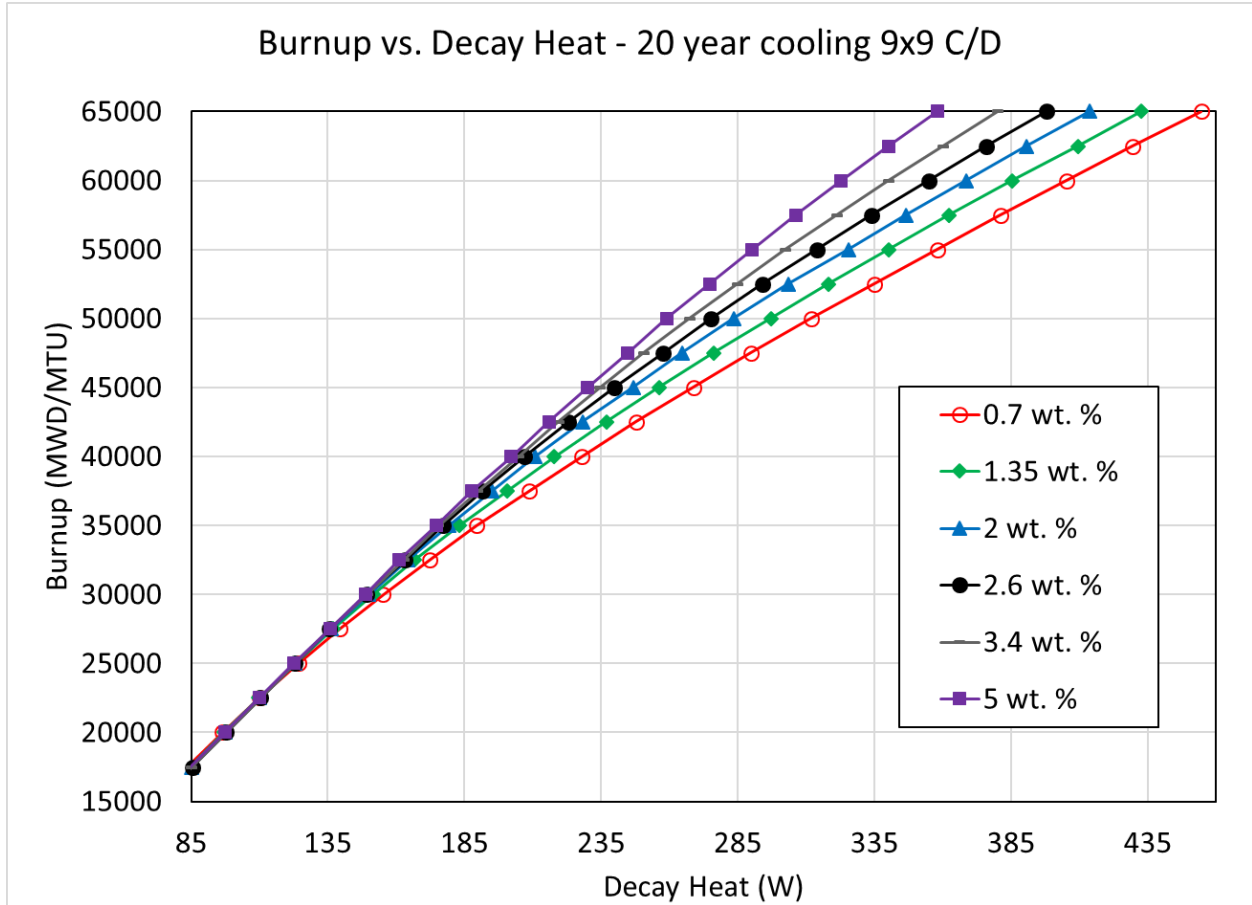


FIGURE 5.F.1; ORIGEN-S CALCULATED BURNUP VERSUS DECAY HEAT FOR VARIOUS ENRICHMENTS

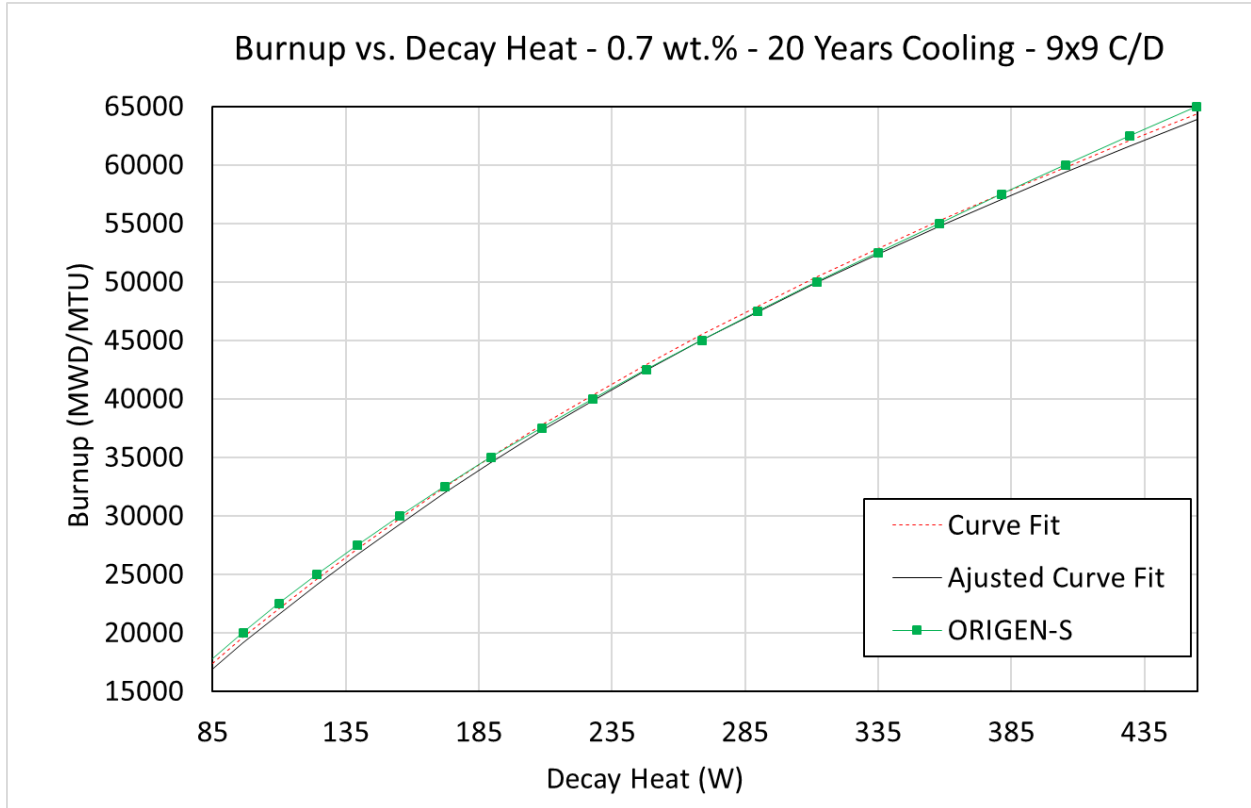


FIGURE 5.F.2; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 0.7 WT.% <sup>235</sup>U.

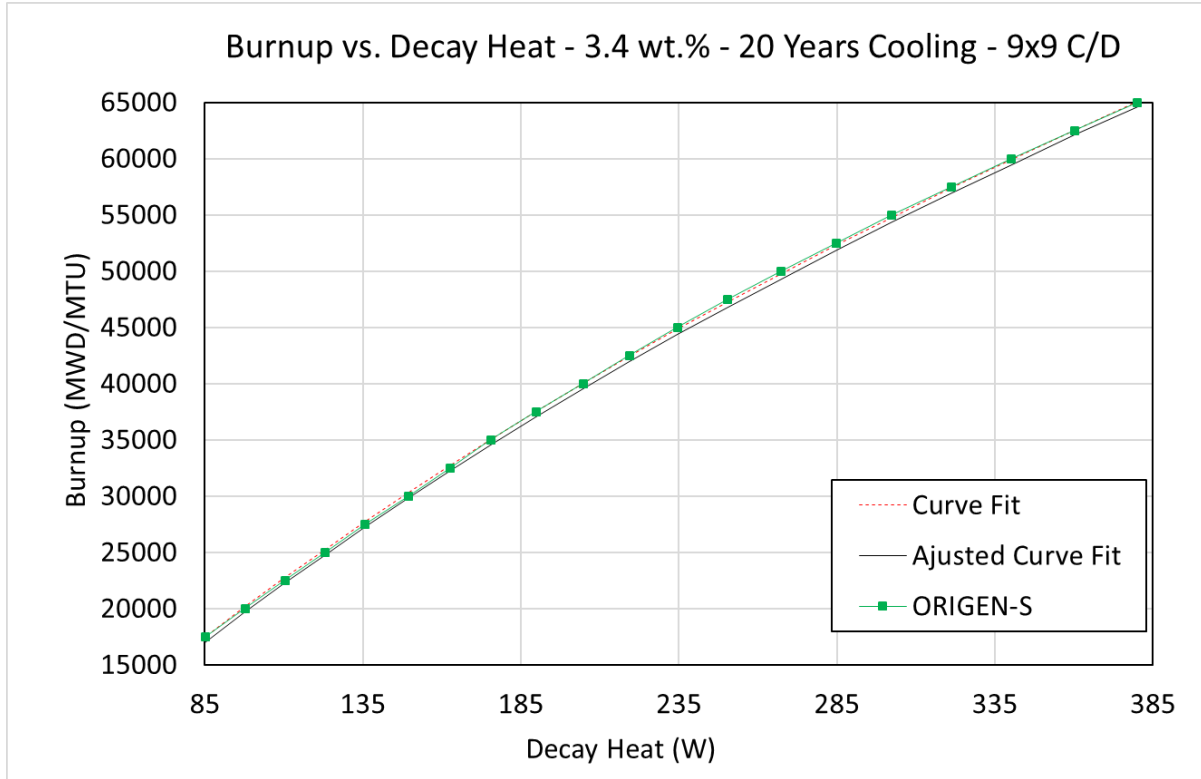


FIGURE 5.F.3; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 3.4 WT.% <sup>235</sup>U.

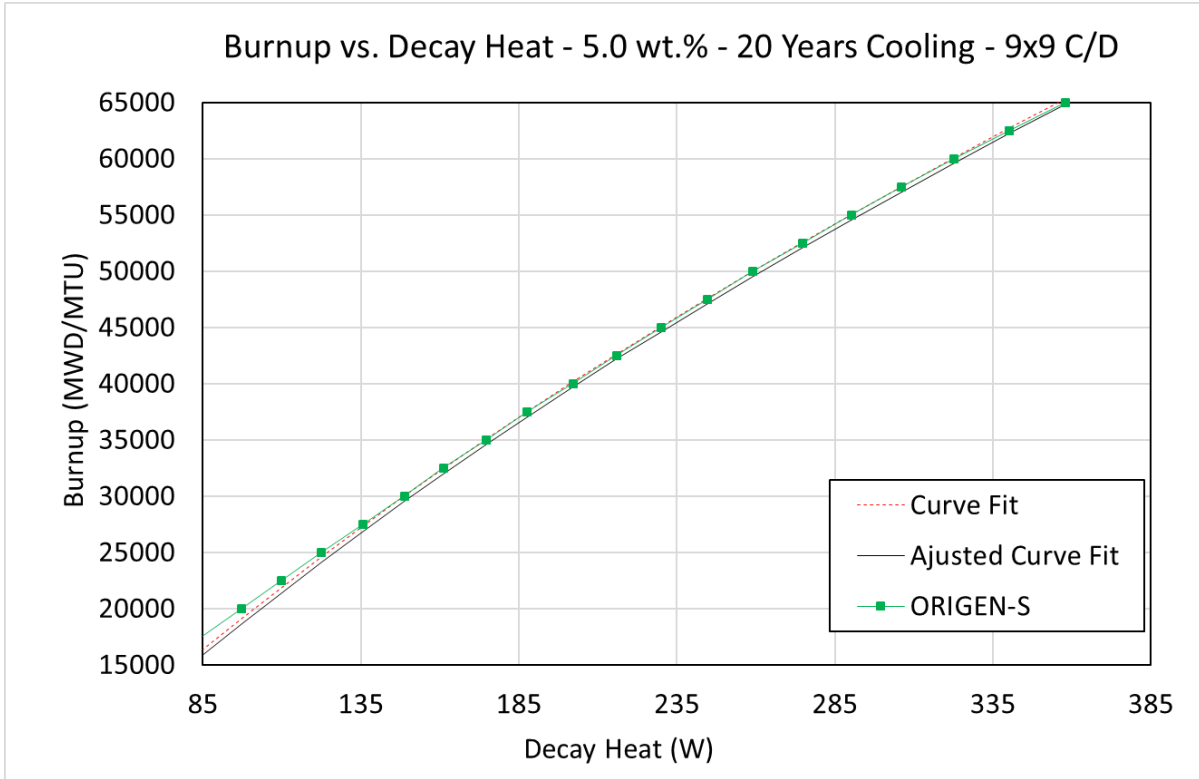


FIGURE 5.F.4; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 5.0 WT.% <sup>235</sup>U.

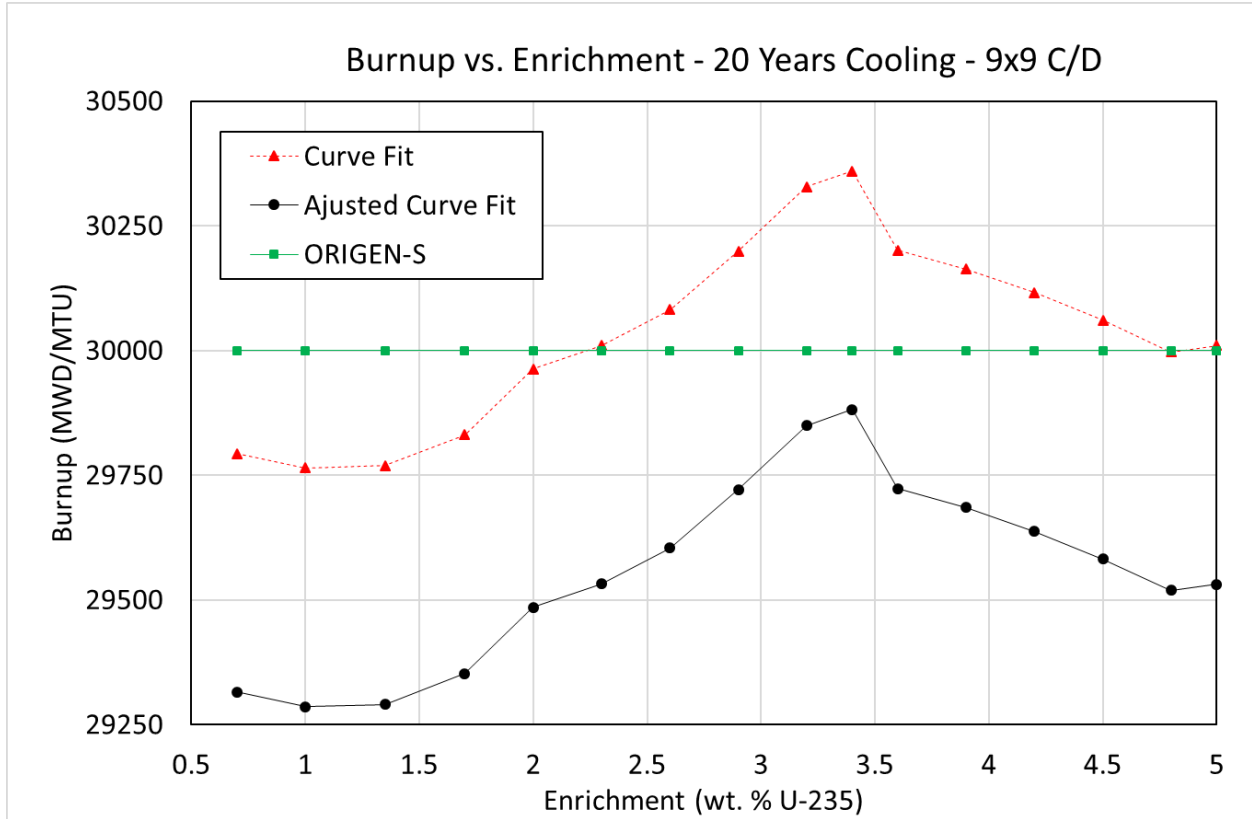


FIGURE 5.F.5; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 30,000 MWD/MTU.

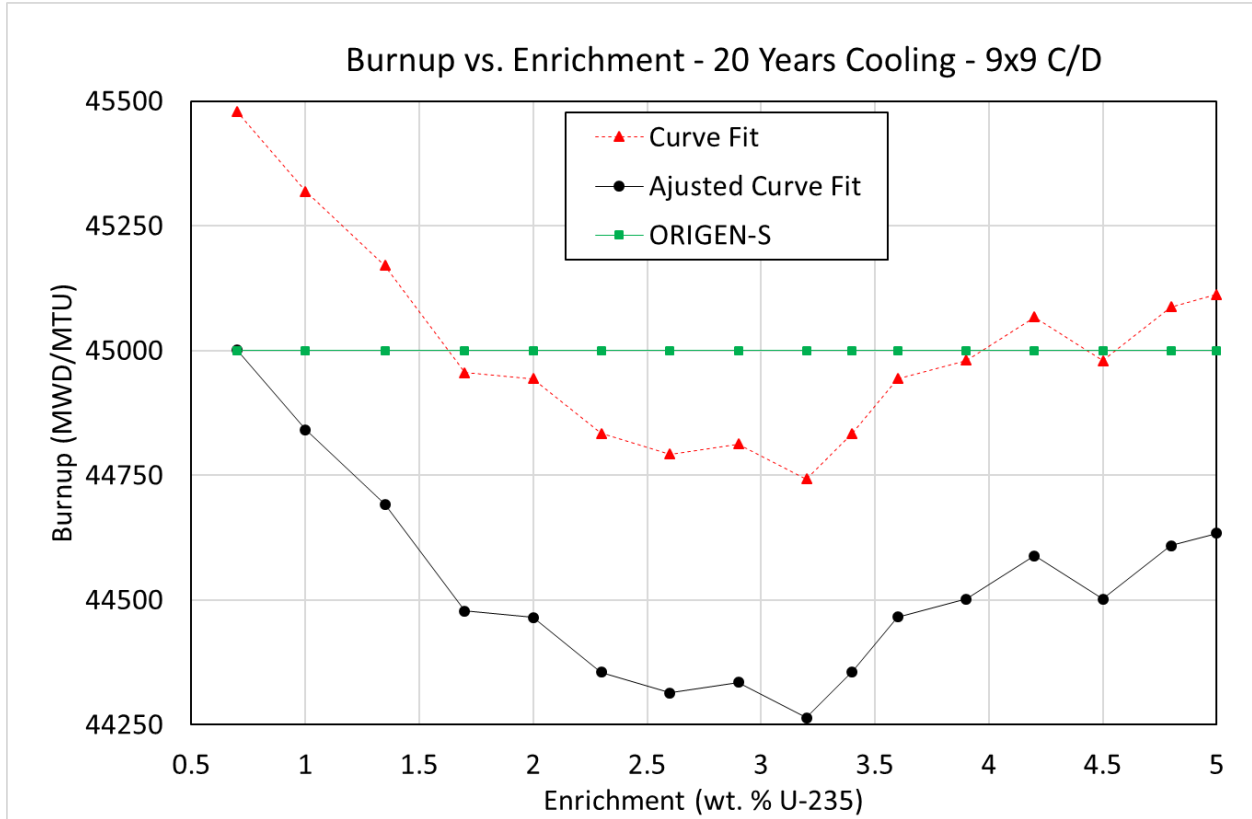


FIGURE 5.F.6; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 45,000 MWD/MTU.



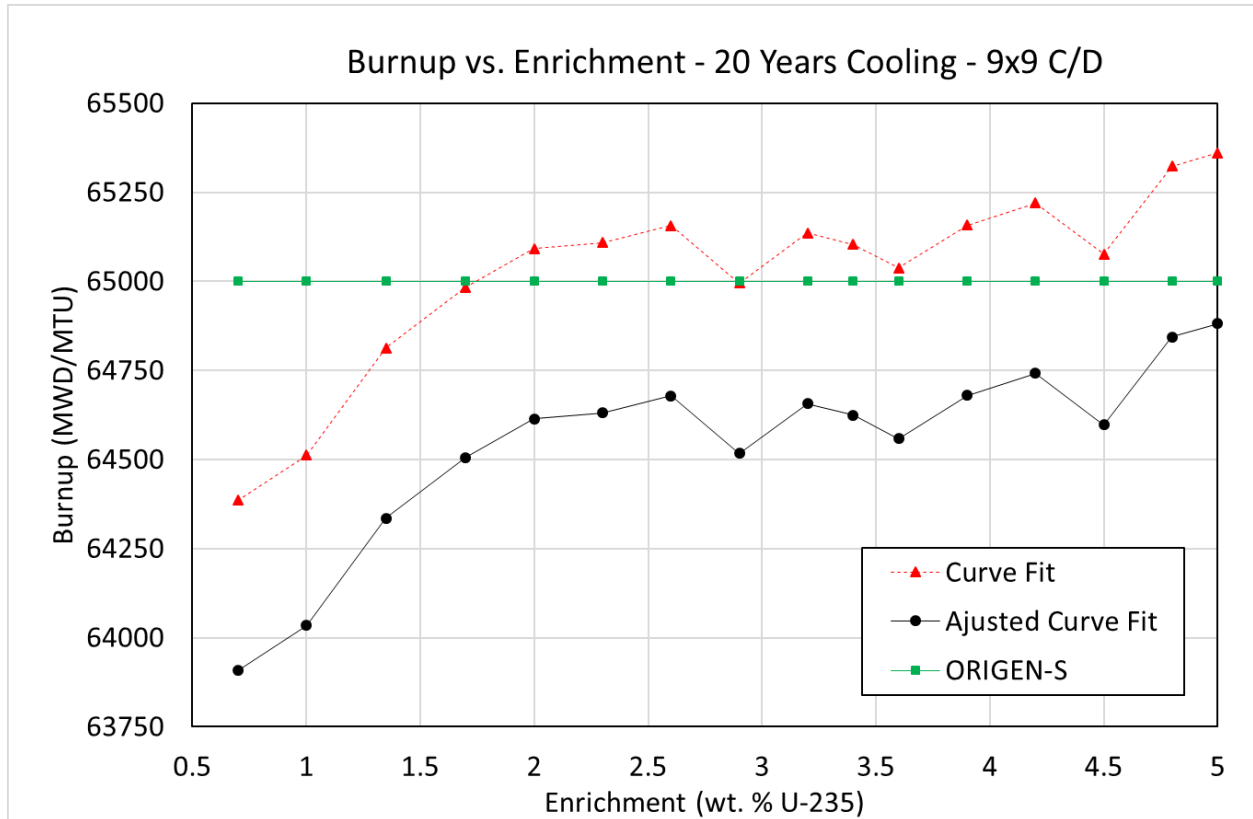


FIGURE 5.F.7; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 65,000 MWD/MTU.

## SUPPLEMENT 5.I

### SHIELDING EVALUATION OF THE HI-STORM 100U SYSTEM

#### 5.I.0 INTRODUCTION

This supplement is focused on providing a shielding evaluation of the HI-STORM 100U system pursuant to the guidelines in NUREG-1536. The evaluation presented herein supplements those evaluations of the HI-STORM overpacks contained in the main body of Chapter 5 of this FSAR, and information in the main body of Chapter 5 that remains applicable to the HI-STORM 100U is not repeated in this supplement. To aid the reader, the sections in this supplement are numbered in the same fashion as the corresponding sections in the main body of this chapter, i.e., Sections 5.I.1 through 5.I.6 correspond to Sections 5.1 through 5.6. Tables and figures in this supplement are labeled sequentially.

#### 5.I.1 DISCUSSION AND RESULTS

The HI-STORM 100U system differs from the HI-STORM system evaluated in the main body of this chapter only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs and HI-TRAC transfer casks are identical between the systems. All calculations, results and conclusions regarding the HI-TRAC transfer cask presented in the main body of Chapter 5 are therefore directly applicable to the HI-STORM 100U system, and no further calculations for the HI-TRAC transfer cask are presented in this supplement.

The shielding design of the HI-STORM 100U VVM is similar to the overpack designs evaluated in the main body of this chapter, with gamma shielding provided by the concrete and the steel of the module, and neutron shielding provided by the module concrete. However, the VVM is mostly located below the surface of the surrounding soil. This results in additional shielding, and a significant reduction in the directly accessible surface for the VVM compared to the other overpacks. Dose rates from a HI-STORM 100U VVM at the site boundary are therefore significantly lower than, and bounded by, dose rates from the above ground HI-STORM systems evaluated in the main body of this chapter.

Shielding analyses were performed for the HI-STORM 100U with an MPC-32 loaded with intact design basis zircaloy clad fuel assemblies. As discussed in Section 5.1, three burnup and cooling time combinations are analyzed for the MPC-32, namely 45,000 MWD/MTU and 3 years, 60,000 MWD/MTU and 4 years, and 69,000 MWD/MTU and 5 years cooling time. These burnup and cooling time combinations bound all assemblies permitted to be loaded in any of the uniform or regionalized loading configurations in the MPC-32. All calculations for the HI-STORM 100 are performed for all three combinations, and the results corresponding to the highest total dose rate at each dose location are reported. Dose rates at some locations are more dominated by the contribution from the neutron source. In this case, the highest burnup will result in the highest dose rates. At other locations, dose rates are more dominated by the contribution from the photon source terms. In this case, the shortest cooling time will result in the highest dose rates. In the

result table, the burnup and cooling time combination that leads to the highest dose rate is indicated for each dose location. It should further be noted that while the number and location of assemblies with such burnup and cooling time combinations would be restricted for any actually loaded cask, it was assumed in the shielding evaluations presented here that all assemblies have the same characteristics. To indicate the level of conservatism in this approach, note that the burnup and cooling time combinations evaluated here correspond to a heat load of about 2.2 kW per assembly, or about 70 kW for the entire MPC-32. This is an important fact in respect to the choice to perform analyses for the MPC-32, and not for the other MPCs. While calculated dose rates for the other baskets could potentially be slightly higher than those for the MPC-32, based on the differences in the MPC design, fuel type, and burnup and cooling times, it is not expected that actual dose rates for any MPC would exceed those calculated here for the MPC-32. This justifies restricting the analyses here to the MPC-32. Table 5.I.1 presents the results for those burnup and cooling time combinations out of those listed above that resulted in the maximum dose rates at each location. Figure 5.I.1 identifies the locations of the dose points referenced in the table. Dose Points #1 and #2 are the locations of the inlet and outlet vents, respectively. The dose values reported adjacent to these dose points were averaged over the vent opening while the dose values reported at 1 meter from these dose locations were taken at the mid-plane of the vent. Dose Point #3, which is positioned approximately over the air flow annulus, is the location of the highest dose rate on the lid in the final storage configuration. Dose Point #4 is averaged over the vertical air flow passage shown in Figure 5.I.1. Dose Point #4, adjacent to the overpack, is not accessible in the final storage configuration as depicted in Figure 5.I.1. Dose Point #5 is located over a tube that would be required for the ICCPS test station. Dose Point #6 is located over an empty VVM located adjacent to a loaded VVM. Except for conditions during construction discussed further below, calculations were only performed for normal conditions, since Subsections 5.1.1 and 5.1.2 concluded that, aside from the 30 day 100% blockage of air inlets accident condition, off-normal and accident conditions for the HI-STORM overpack are identical or equivalent to normal conditions for the purpose of the shielding evaluation.

The tube for the ICCPS test station is modeled as a cylindrical hole that extends from the VIP down to the base plate of the MPC. The tube is modeled with a diameter of 4 inches, located about 5.5 feet from the center of the VVM. If the actual tube has characteristics that could result in higher dose rates, i.e. is larger or closer to the VVM than modeled here, the actual tube characteristics should be considered in the site specific dose calculations. Depending on the results of those calculations, additional measures, such as added shielding at the top of the tube, may be required.

A comparison between the dose rates in Table 5.I.1 and dose rates presented in Tables 5.1.11 and 5.1.14 of the main body of this chapter show that the maximum dose rate for the HI-STORM 100U module with an MPC-32 is well below the maximum dose rate for the HI-STORM 100S Version B with an MPC-32. Furthermore, the area associated with the maximum dose rate for the HI-STORM 100U module is the inlet vent. This area is much smaller than the area associated with the maximum dose rate for the overpack analyzed in the main part of this chapter, which is the outer radial surface of the overpack. It can therefore be concluded that the HI-STORM 100U is bounded by the HI-STORM 100 systems analyzed in the main body of this chapter, since the

HI-STORM 100U has a smaller directly accessible surface, lower maximum dose rates and smaller areas associated with these maximum dose rates. Nevertheless, calculations were performed to determine the dose rate from the HI-STORM 100U at a distance of 100 meters from the inlet vent. These results, which are presented in Table 5.I.2, indicate that the HI-STORM 100U easily meets the requirements of 10CFR72.104 at 100 meters. Comparing these results to the results in Table 5.4.7 demonstrates that the off-site dose from the HI-STORM 100U is a very small fraction of the off-site dose from an above ground overpack.

ISFSIs with HI-STORM 100U VVMs might be built in one stage or in several stages. If the ISFSI is built in several stages, then excavation work will be necessary in the vicinity of the section of the ISFSI that is already in operation and contains loaded modules. To protect workers from radiation from the loaded modules, a radiation protection space (RPS) boundary is defined around the ISFSI in the drawing in Section 1.I.5. The RPS boundary is placed so that in a radial direction, a minimum of 10.5 ft of engineered fill remains between the construction site and the closest loaded module. For a loaded module on the periphery of the ISFSI, this places the boundary 14 ft from the center of the module (10.5 ft + radius of the VVM of 3.5 ft). For a loaded module not on the periphery of the ISFSI, this places the boundary 21 feet from the center of the module (10.5 ft + diameter of the outer empty VVM of 7 ft + radius of the loaded VVM of 3.5 ft). Calculations during excavations were performed for both normal and accident conditions. Under normal conditions, calculations were conservatively performed with the 6.5 ft remaining fill around a loaded VVM instead of the 10.5 ft required by the RPS. The calculations result in a maximum dose rate of only about 0.2 mrem/hr at the surface of the excavation. This dose rate is very low, specifically lower than the dose rates at 1 m from the inlet vents of the modules. The dose rates at a construction site might therefore be dominated by the direct dose rates from the inlet vents, and depending on the loading condition of the operating part of the ISFSI, temporary shielding might be used to reduce dose rates to the construction site. Note that the dose rate of 0.2 mrem/hr is calculated using soil with a density of 1.7 g/cm<sup>3</sup> for the 6.5 ft layer. This is conservative since the material in and around the VVM modules would be engineered fill which has a typical density of more than 1.7 g/cm<sup>3</sup>. The bounding accident condition is identified in Supplement 3.I, Section 3.I.4.8.2 to be the impact of a tornado missile with a diameter of 8 inches. This missile would penetrate the soil about 10 ft, and this case was used as the basis for selecting the size of the RPS. This is the bounding condition since smaller missiles have less energy, and larger missiles (automotive) have a much larger impact area thus resulting in a much smaller indentation of the soil. Under the bounding condition, the maximum dose over a period of 30 days at a distance of 100 m from the VVM would be 1.4 rem. This is less than the limit in 10CFR72, and also less than the dose for the bounding accident condition discussed in the main body of Chapter 5.

Finally, as observed in Section 1.I.2, the HI-STORM 100U VVM is deliberately engineered to permit the MPC cavity to be deepened such that the MPC is located deeper inside the module. The elevation of the MPC shown in the drawings in Section 1.5 and analyzed in this supplement is the highest permitted elevation. Of course, lowering the MPC in the VVM would further reduce the radiation dose rates below those computed herein.

### 5.1.2 SOURCE SPECIFICATION

The analyses in this supplement are performed for intact design basis zircaloy clad fuel assemblies as described in Section 5.2.

### 5.1.3 MODEL SPECIFICATIONS

The shielding analyses of the HI-STORM 100U module are performed with MCNP-4A, which is the same code used for the analyses presented in the main body of this chapter.

Section 1.1.5 provides the drawings that describe the HI-STORM 100U System. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figure 5.1.2 shows cross sectional views of the HI-STORM 100U module as it was modeled in MCNP for normal conditions. Note that the inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the dose calculations. Note again that the MPC is assumed to be positioned at its highest permissible elevation in relation to the inlet ducts (i.e., in the configuration shown in the drawings in Section 1.5) to maximize the calculated dose rates.

For the assumed accident condition during the construction phase, the model contains a VVM surrounded by 6.5 feet of soil (i.e. a thinner soil layer than the one required by the RPS of 10.5 ft), with a horizontal cylindrical hole (8 inch diameter) extending from the metal surface of the VVM to the outer surface of the soil. This condition is the result of the impact of a tornado missile with a diameter of 8 inches. Note that in the model the displaced soil and the missile itself are conservatively assumed to be lost. Also, the cylinder is conservatively assumed to be on the mid-height of the fuel. The dose location is on the axis of the cylindrical hole, 100 m from the VVM.

Since the HI-STORM 100U model uses principally the same MPC model as the calculations in the main body of this chapter, all figures, conservative modeling approximations, and modeling differences for the MPC shown in Section 5.3 are applicable to the calculations in this supplement. The differences between models and drawings for the module are listed and discussed here.

1. Minor penetrations in the body of the module (e.g. lift locations) are not modeled as these are small localized effects which will not affect the off-site dose rates.
2. The MPC supports and guides were conservatively neglected.
3. The closure lid cover plate was modeled as flat. This conservatively reduces the amount of concrete in the lid near the outlet vent.
4. The insulation installed on the divider shell was conservatively modeled as a void.

5. The cavities representing the ICCPS tube and the empty VVM are modeled as empty volumes surrounded by soil, i.e. any steel liner or other material in these areas, or any covers that would be located on top of those cavities, are conservatively neglected
6. The lid design contains a buttress rod at the center. This rod is not explicitly modeled, i.e. the steel of the rod is replaced with concrete in the model. This has a negligible effect on the dose rates.

Composition and densities of the various materials in Table 5.3.2 were used in the analyses, except for the soil composition. The soil composition and density is shown in Table 5.I.3, and represent typical soil conditions [5.I.1]. This is conservative, since the areas between and around the modules would contain engineered fill with a typical density higher than soil. Furthermore, the dose rates around the VVM are dominated by the streaming through the inlet and outlet vents, and not by direct radiation through the soil and concrete. This is evident by the fact that the dose rate calculated for the excavation site through 6.5 ft of radial soil (see Section 5.I.1) is much less than the dose rates around the inlet and outlet vents. To further substantiate this, a complete dose rate profile across the lid and the VIP was determined. For the VIP, two conditions were evaluated, the normal condition and a condition where the streaming from the inlets and outlets were artificially blocked. For this second condition, dose rates were also calculated at a distance of 100 m from the VVM. This would indicate what portion of the dose rate results from direct radiation through the concrete and soil of the VIP as opposed to radiation from the streaming from the air inlet and outlet. The dose locations for the profile are shown in Figure 5.I.3, and are labeled alphabetically (A through X). The calculated dose rates are listed in Table 5.I.4. For the VIP, dose rates are shown with the open inlets and outlets, and with the artificially closed inlets and outlets. The following conclusions can be drawn from the results:

- The profile did not reveal any locations with dose rates higher than those shown in Figure 5.I.1 and Table 5.I.1.
- On the VIP, the dose rates are fairly low compared to other areas.
- Comparison of the results on the VIP surface with open and artificially closed inlets and outlets show that closer to the VVM, the dose rate on the surface is dominated by the direct radiation through the concrete and soil, while at larger distances, the relative contribution of the direct radiation through the soil and concrete reduces. At a distance of 100 m, the dose rate from the VIP surface contributes about one third of the total dose rate.

Note that the dose location 1 m from the inlet vents (see Figure 5.I.1) is approximately above dose locations W and X shown in Figure 5.I.3. The 1 m dose will therefore include contributions from the inlet vent as well as the VIP surface, and does therefore represent both the contribution from streaming through the inlet vent and from direct radiation through the soil and concrete of the VIP.

In summary, soil properties are conservative and its effect is included in the dose rate at 1 m from the inlet vent. The use of typical soil properties is therefore appropriate. Nevertheless, site specific analyses to demonstrate compliance with regulatory requirements should use appropriate site specific soil properties if these are substantially different from the properties used in this Supplement.

#### 5.I.4 SHIELDING EVALUATION

Table 5.I.1 provides dose rates adjacent to and at 1 meter distance from the HI-STORM 100U module during normal conditions for the MPC-32. The table also includes dose rates at the top of the ICCPS tube and at the top of an empty cavity next to a loaded VVM. These results demonstrate that the dose rates around the HI-STORM 100U are exceptionally low for the very conservative burnup and cooling time combination analyzed. These results also show that the higher dose rate at the inlet vent is reduced by more than an order of magnitude at a distance of 1 meter from the vent.

Table 5.I.2 provides the annual dose at 100 meters from a HI-STORM 100U module for the MPC-32 including the contribution from BPRAs. These results clearly demonstrate that the off-site contribution from a HI-STORM 100U is a small fraction of the off-site dose from the above ground HI-STORM overpacks.

The bounding accident condition is the impact of a tornado missile during the expansion of an array of 100U modules. The dose calculated for this condition at a distance of 100 m is 1.4 rem, which is below the limit specified in 10CFR72, and also below the dose for the bounding accident condition determined in the main section of Chapter 5.

#### 5.I.5 REGULATORY COMPLIANCE

In summary it can be concluded that dose rates from the HI-STORM 100U module are bounded by the dose rates for the overpacks analyzed in the main body of the report. The shielding system of the HI-STORM 100U System is therefore in compliance with 10CFR72 and satisfies the applicable design and acceptance criteria including 10CFR20. Thus, the shielding evaluation presented in this supplement provides reasonable assurance that the HI-STORM 100U System will allow safe storage of spent fuel.

#### 5.I.6 REFERENCES

[5.I.1] ANSI/ANS-6.6.1-1987, "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants"

Table 5.I.1

DOSE RATES ADJACENT TO AND 1 METER FROM THE HI-STORM 100U MODULE FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING BURNUP AND COOLING TIME

<b>Dose Point<sup>†</sup> Location</b>	<b>Burnup and Cooling Time (MWD/MTU / Years)</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
<b>Surface</b>						
1	69,000 / 5	41.70	8.24	15.08	65.02	69.43
2	69,000 / 5	3.03	1.07	4.84	8.94	9.39
3	69,000 / 5	16.83	3.59	5.08	25.50	27.91
4	69,000 / 5	11.62	10.23	25.69	47.54	51.73
5	45,000 / 3	3.38	2.83	7.60E-02	6.29	7.49
6 <sup>‡</sup>	45,000 / 3	0.40	6.6E-02	1.85E-02	0.48	0.52
<b>One Meter</b>						
1	69,000 / 5	3.44	0.70	1.41	5.56	5.92
2	69,000 / 5	0.93	0.44	0.92	2.30	2.49
3	69,000 / 5	5.10	1.07	1.50	7.67	8.49
4	69,000 / 5	2.97	0.68	1.96	5.61	6.03

<sup>†</sup> Refer to Figure 5.I.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>‡</sup> Calculated for an empty VVM surrounded by four loaded VVMs.



Table 5.I.2

ANNUAL DOSE AT 100 METERS FROM A SINGLE  
HI-STORM 100U OVERPACK WITH AN MPC-32 WITH DESIGN BASIS  
ZIRCALOY CLAD FUEL<sup>†</sup>

Dose Component	69,000 MWD/MTU 5-Year Cooling (mrem/yr)
Fuel gammas <sup>††</sup>	4.01
<sup>60</sup> Co Gammas	1.22
Neutrons	3.27
Total	8.50

<sup>†</sup> 8760 hour annual occupancy is assumed.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

Table 5.I.3

## COMPOSITION OF THE MATERIALS IN THE HI-STORM 100U SYSTEM

<b>Component</b>	<b>Density (g/cm<sup>3</sup>)</b>	<b>Elements</b>	<b>Mass Fraction (%)</b>
Soil	1.7	H	0.962
		O	54.361
		Al	12.859
		Si	31.818

Table 5.I.1

## DOSE RATES ADJACENT TO THE HI-STORM 100U MODULE FOR NORMAL CONDITIONS AT DOSE LOCATIONS SHOWN IN FIGURE 5.I.3.

Note: All Values Are For 69,000 MWD/MTU And 5 Years Cooling,  
Except For Dose Location I Which Is For 60,000 MWD/MTU And 4 Years Cooling

Dose Location <sup>††</sup>	Dose Rate (mrem/hr unless noted)		Dose Location	Dose Rate (mrem/hr unless noted)	
	Inlet/Outlet Open	Inlet/Outlet Artificially Closed		Inlet/Outlet Open	Inlet/Outlet Artificially Closed
A	33.1	N/C <sup>†</sup>	N	2.0	N/C
B	16.7	N/C	O	3.5	N/C
C	15.7	N/C	P	9.3	N/C
D	9.3	N/C	Q	6.8	6.5
E	6.5	N/C	R	4.1	3.5
F	7.0	N/C	S	3.0	2.0
G	4.8	N/C	T	2.5	2.0
H	6.6	N/C	U	2.2	1.5
I	33.2	N/C	V	2.0	1.5
J	27.8	N/C	W	1.7	1.0
K	3.7	N/C	X	1.6	1.0
L	1.1	N/C	Y	8.5 mrem/year	2.8 mrem/year
M	1.3	N/C			

<sup>††</sup> See Figure 5.I.3

<sup>†</sup> N/C = Not Calculated

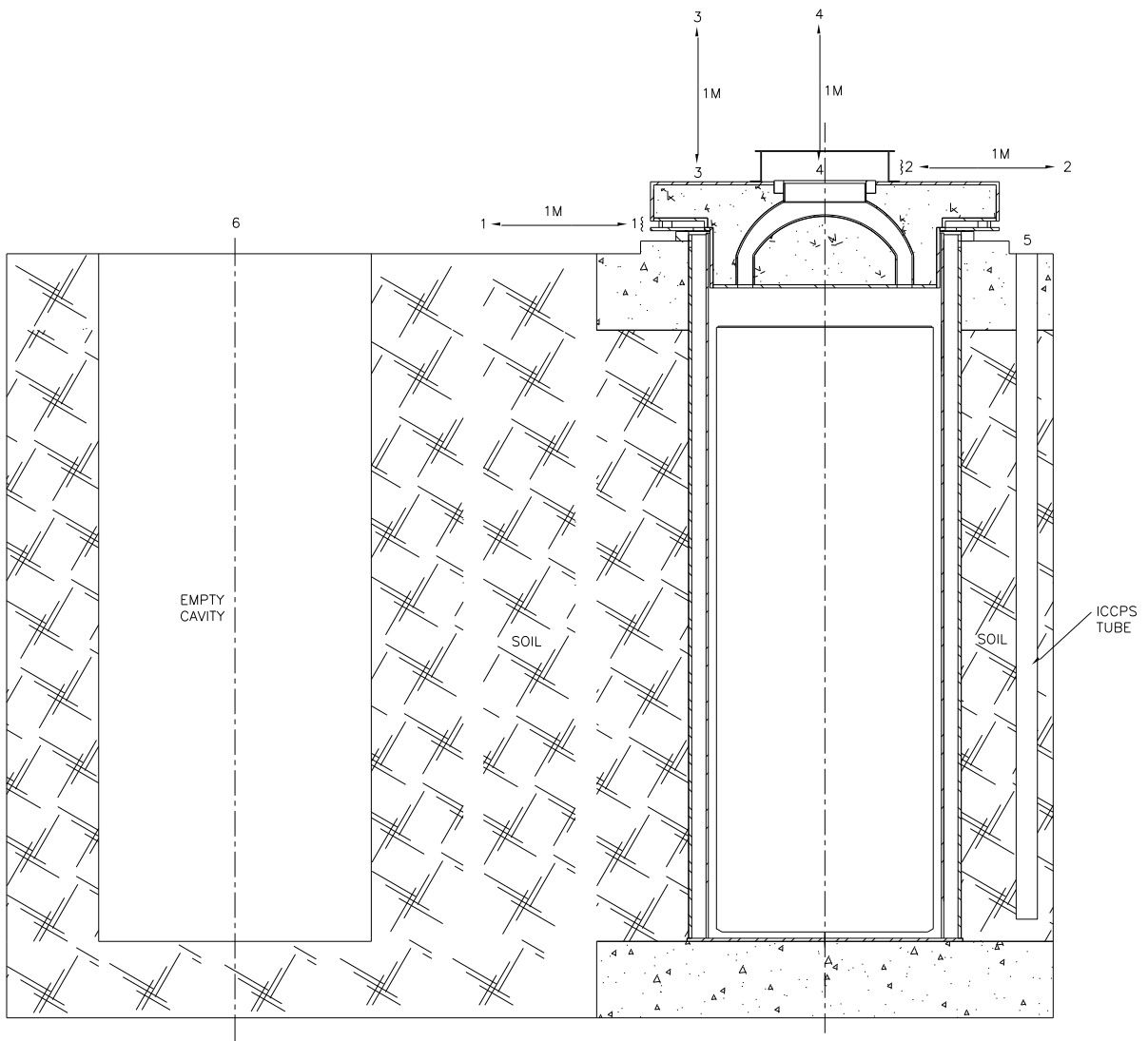


FIGURE 5.1.1; HI-STORM 100U MODULE CROSS SECTIONAL ELEVATION VIEW WITH DOSE POINT LOCATION

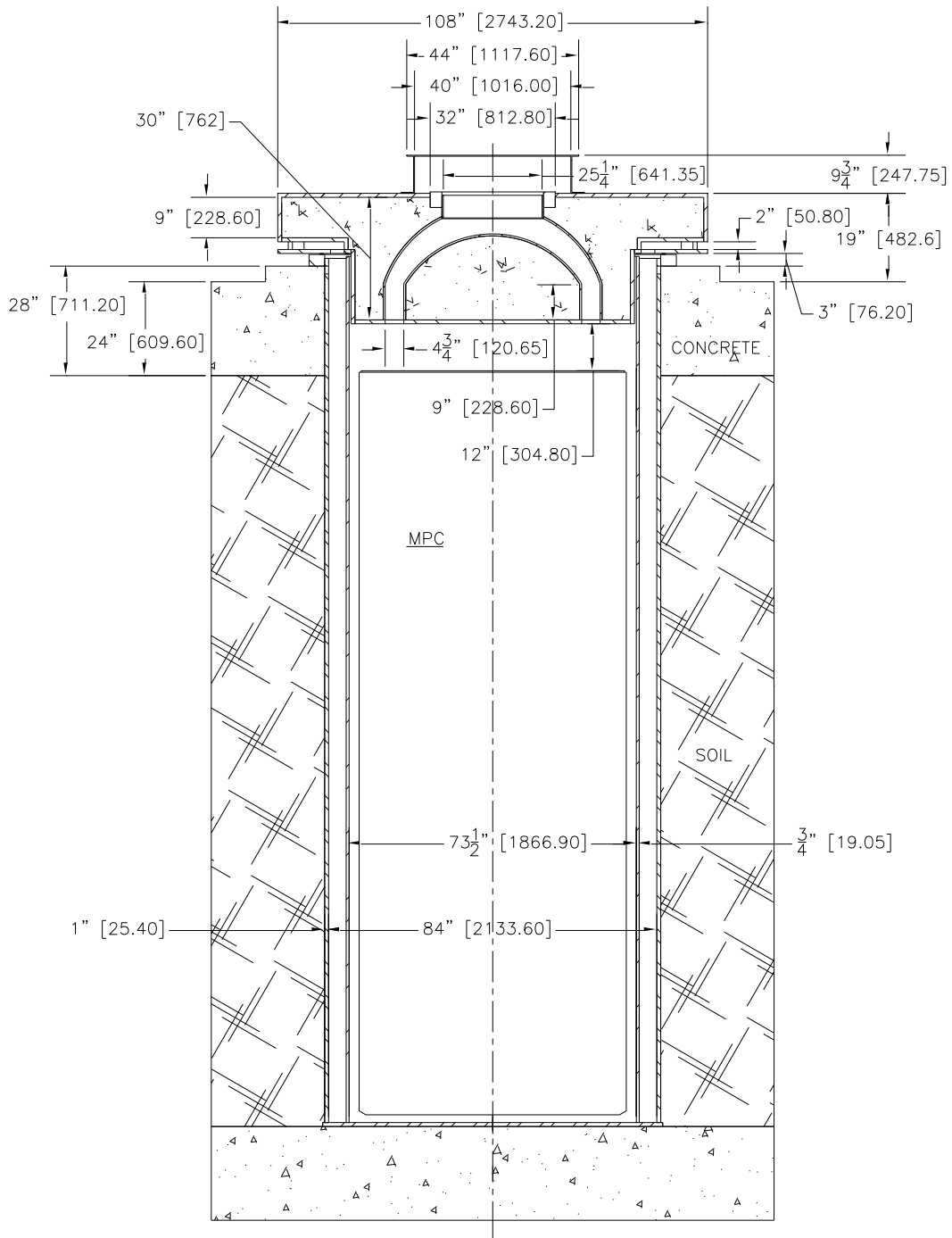


FIGURE 5.1.2; HI-STORM 100U MODULE CROSS SECTIONAL ELEVATION VIEW. VALUES IN BRACKETS ARE IN MILLIMETERS.

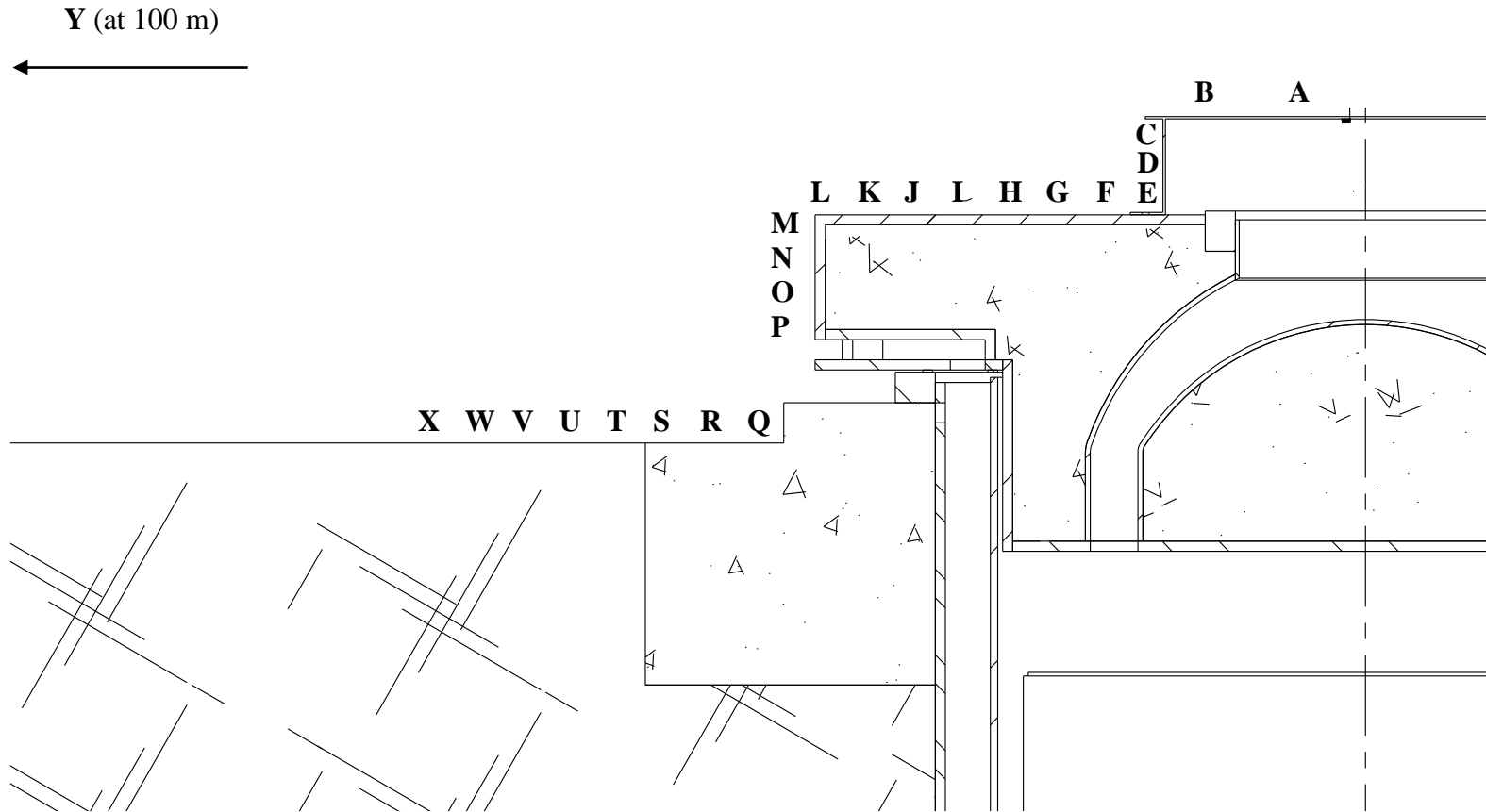


FIGURE 5.I.3; HI-STORM 100U MODULE CROSS SECTIONAL ELEVATION VIEW WITH DOSE POINT LOCATIONS

## SUPPLEMENT 5.III

### EVALUATION OF THE MPC-68M BASKET, AND THE 10x10F AND 10x10G ASSEMBLY CLASSES

#### 5.III.0 DISCUSSION

The MPC-68M is a variation of the 68 cell BWR canister MPC-68 evaluated in the main part of this chapter, but with a basket design consisting of aluminum oxide and finely ground boron carbide dispersed in a metal matrix of pure aluminum. The boron carbide content is 10% (minimum) by weight. This results in a B-10 areal density that is slightly above that in the MPC-68. The relevant differences between the baskets are listed below, and then discussed in respect to its effect on the photon and neutron dose rates.

Differences between the MPC-68M compared to the MPC-68, in respect to the characteristics important for the dose calculations, are as follows:

- The MPC-68M has a slightly higher B-10 content
- The MPC-68M is lighter, since it consists of aluminum and boron carbide, but no steel
- In the enclosure shell, the MPC-68M is surrounded by aluminum basket shims

To evaluate the effect of these differences, studies in the main part of Chapter 5 regarding dose contributions from a regionalized loading scheme are utilized. These studies, described in Section 5.4, show that the inner region on an MPC-68 (32 assemblies = 47 % of the content) contributes about 27% of the neutron dose rate, but only about 2 % of the photon dose rate. This means that the self-shielding of the fuel and basket for neutron radiation is low, while for photon radiation it is very high. The low neutron self-shielding means that the neutron doses are not significantly affected by the reduced basket weight, since the majority of the neutron shielding function is provided by the overpack around the MPC. Also, for MPCs filled with water, there is a further reduction in neutron dose due to the increased absorption of thermal neutrons from the increased B-10 loading. The high self-shielding for photons means that only the outer basket panels are effective for gamma shielding. For the MPC-68M, the shielding in this area is enhanced due to the presence of the basket shims, and therefore comparable to the absorption in the steel basket walls. In summary, the effect of the design differences between MPC-68 and MPC-68M on dose rates is small.

Additionally, two BWR array classes designated 10x10F and 10x10G have been added as approved contents in the MPC-68M only. From a radiological perspective, the additional array classes are bounded by the design basis GE 7x7 source term calculations, since those design basis assemblies have higher initial uranium masses. In terms of grouping assemblies for the polynomial factors presented in Section 2.1.9, the new array classes are added to groups which represent assemblies of a higher mass. This is conservative since a heavier assembly results in a

higher decay heat, which reduces the allowable assembly burnup. In summary, no new analyses are necessary to qualify those additional array classes.

Therefore, the main body of this chapter remains fully applicable for the HI-STORM 100 System using an MPC-68M and the new assembly classes.

As discussed in Subsection 2.0.1, each MPC basket, except MPC-68F, allows for two loading strategies, namely the uniform fuel loading and the regionalized loading with two regions. An additional 3-region loading pattern, shown in Figure 2.III.1, is evaluated to determine acceptability as approved contents in the MPC-68M only. This evaluation performs a dose rate comparison between a uniform loading pattern (results shown in Table 5.4.9) and a 3-region pattern (Table 5.III.1) in which one region contains 2-year cooled spent nuclear fuel.

It should be noted that the design basis GE 7x7 source term calculations, discussed in Section 5.2 of the main part of this chapter, are performed using the SCALE 4.3 system [5.1.2, 5.1.3]. The evaluation in this Supplement 5.III is performed with an updated version of SCALE (SCALE 5.1) which is consistent with other approved Holtec applications [5.III.3]. (SCALE 4.3 is no longer maintained by Oak Ridge National Laboratory, and does not work on contemporary computer operating systems.) To ensure that the dose rate comparison is not affected by the SCALE code version, a comparison between results generated using SCALE 4.3 and SCALE 5.1 was performed. There were no significant differences in the neutron and fuel gamma source terms between the two SCALE versions. The Cobalt-60 photon source calculated with SCALE 5.1 were substantially higher than the Cobalt-60 source calculated using SCALE 4.3 which is encompassed in the first two dose rate results columns of Table 5.III.2. The remaining comparisons shown in Table 5.III.2 are performed using the updated version of SCALE (SCALE 5.1), which compared the uniform loading pattern (50,000 MWD/MTU, 3-year cooling time) against the 3-region loading pattern presented in Table 5.III.1 for both the MPC-68<sup>1</sup> and MPC-68M.

Table 5.III.2 shows the dose rates for the 3-region loading pattern (Figure 2.III.1 and Table 5.III.1) are bounded by the uniform loading pattern (50,000 MWD/MTU and 3-year cooling). For this reason, the 3-region loading pattern shown in Figure 2.III.1 is added as approved contents of the MPC-68M. The minimum cooling time criteria for the MPC-68M is updated to 2.0 years (see Table 2.0.1). The minimum cooling time criteria for the MPC-68 remains at 3.0 years (see Table 2.0.1).

<sup>1</sup> The 3-region loading pattern shown in Figure 2.III.1 and Table 5.III.1 is approved for the MPC-68M only. All results related to the 3-region loading pattern (in Figure 2.III.1 and Table 5.III.1) in the MPC-68 are for the purposes of comparison only.



### 5.III.1 REFERENCES

- [5.III.1] I.C. Gauld and O.W. Hermann, "SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module," ORNL/TM-2005/39, Revision 5.1, , Oak Ridge National Laboratory, November 2006.
- [5.III.2] I.C. Gauld, O.W. Hermann, and R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," ORNL/TM-2005/39, Version 5.1, Oak Ridge National Laboratory, November 2006.
- [5.III.3] Holtec International Report HI-2114830, Final Safety Analysis Report on the HI-STORM FW System, USNRC Docket 72-1032, latest revision.
- [5.III.4] Holtec International Report HI-2146423 Revision 1, HI-STAR 190 Source Terms and Loading Patterns.

Table 5.III.1

## 3-REGION LOADING PATTERN FOR MPC-68M

<b>Region</b>	<b>Assemblies Per Region</b>	<b>Heat Load Limit<sup>1</sup> (kW)</b>	<b>Burnup (MWD/MTU)</b>	<b>Enrichment (wt%)</b>	<b>Cooling Time (years)</b>	<b>Calculated Heat Load<sup>2</sup> (kW)</b>
1	24	0.4	50000	3.6	7	0.436
2	16	1.2			2	1.336
3	28	0.5			5	0.550

<sup>1</sup> As shown in Figure 2.III.1.

<sup>2</sup> Decay heat is calculated in Reference [5.III.4]; this shielding analysis reference case conservatively exceeds the heat load limits set in Figure 2.III.1.

Table 5.III.2

DOSE RATES COMPARISON BETWEEN UNIFORM<sup>6</sup> AND 3-REGION<sup>7</sup> LOADING PATTERN FOR 100-TON HI-TRAC

<b>Dose Point Location</b>	<b>Totals Uniform MPC-68 From Table 5.4.9 (SCALE 4.3)</b> (mrem/hr)	<b>Totals Uniform MPC-68 (SCALE 5.1)</b> (mrem/hr)	<b>Totals 3-Region MPC-68 (SCALE 5.1)</b> (mrem/hr)	<b>Totals Uniform MPC-68M (SCALE 5.1)</b> (mrem/hr)	<b>Totals 3-Region MPC-68M (SCALE 5.1)</b> (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	1745.39	1978.3	1631.8	2068.0	1706.5
2	2992.98	3025.3	1803.6	2742.7	1672.9
3	991.17	1145.0	950.0	1526.4	1240.3
4	715.82	767.4	570.2	866.7	625.6
5 (pool lid)	8069.5	9414.2	6815.3	9848.6	6979.4
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	552.67	575.2	381.7	541.2	369.0
2	1257.27	1278.1	750.5	1169.9	702.7
3	289.97	323.3	242.9	375.1	284.7

<sup>6</sup> Design Basis GE 7x7 spent fuel with a burnup of 50,000 MWD/MTU, cooling time of 3 years, and initial enrichment of 3.6 wt% U-235.

<sup>7</sup> As shown in Figure 2.III.1 and Table 5.III.1.

## CHAPTER 6<sup>†</sup>: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STORM 100 System is consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor ( $k_{\text{eff}}$ ), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.
2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.
3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both.
4. Criticality safety of the cask system should not rely on use of the following credits:
  - a. burnup of the fuel
  - b. fuel-related burnable neutron absorbers
  - c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test<sup>††</sup>.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STORM 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. Analyses for the HI-STAR 100 System, which are applicable to the HI-STORM 100 System, have been previously submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

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

<sup>††</sup> For greater credit allowance, fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.

## 6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor ( $k_{\text{eff}}$ ) of the HI-STORM 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STORM 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM 100 System depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24 and MPC-24E);
2. The incorporation of permanent fixed neutron-absorbing panels in the fuel basket structure;
3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and
4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24 and MPC-24E, and for loading/unloading fuel in the MPC-32.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the Boral fixed neutron absorber
- more than 90 percent of the B-10 content for the Metamic fixed neutron absorber, with comprehensive fabrication tests as described in Section 9.1.5.3.2.

The following four interchangeable basket designs are available for use in the HI-STORM 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 24-cell basket (MPC-24E) for intact and damaged PWR fuel assemblies. This is a variation of the MPC-24, with an optimized cell arrangement, increased  $^{10}\text{B}$  content in the fixed neutron absorber and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC) containing damaged PWR fuel assemblies or PWR fuel debris. The MPC-24E is designed for fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 32-cell basket (MPC-32), designed for intact, damaged PWR fuel assemblies, and PWR fuel debris of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading. And
- a 68-cell basket (MPC-68), designed for intact and damaged BWR fuel assemblies, and BWR fuel debris with a specified maximum planar-average enrichment. Additionally, a variation in the MPC-68, designated MPC-68F, is designed for specific intact and damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.

Two interchangeable neutron absorber materials are used in these baskets, Boral and Metamic. For Boral, 75 percent of the minimum B-10 content is credited in the criticality analysis, while for Metamic, 90 percent of the minimum B-10 content is credited, based on the neutron absorber tests specified in Section 9.1.5.3. However, the B-10 content in Metamic is chosen to be lower than the B-10 content in Boral, and is chosen so that the absolute B-10 content credited in the criticality analysis is the same for the two materials. This makes the two materials identical from a criticality perspective. This is confirmed by comparing results for a selected number of cases that were

performed with both materials (see Section 6.4.11). Calculations in this chapter are therefore only performed for the Boral neutron absorber, with results directly applicable to Metamic.

The HI-STORM 100 System includes the HI-TRAC transfer cask and the HI-STORM storage cask. The HI-TRAC transfer cask is required for loading and unloading fuel into the MPC and for transfer of the MPC into the HI-STORM storage cask. HI-TRAC uses a lead shield for gamma radiation and a water-filled jacket for neutron shielding. The HI-STORM storage cask uses concrete as a shield for both gamma and neutron radiation. Both the HI-TRAC transfer cask and the HI-STORM storage cask, as well as the HI-STAR System<sup>†</sup>, accommodate the interchangeable MPC designs. The three cask designs (HI-STAR, HI-STORM, and HI-TRAC) differ only in the overpack reflector materials (steel for HI-STAR, concrete for HI-STORM, and lead for HI-TRAC), which do not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 system and vice versa. Therefore, the majority of criticality calculations to support both the HI-STAR and the HI-STORM System have been performed for only one of the two systems, namely the HI-STAR System. Only a selected number of analyses has been performed for both systems to demonstrate that this approach is valid. Therefore, unless specifically noted otherwise, all analyses documented throughout this chapter have been performed for the HI-STAR System. For the cases where analyses were performed for both the HI-STORM and HI-STAR System, this is clearly indicated.

The HI-STORM 100 System for storage (concrete overpack) is dry (no moderator), and thus, the reactivity is very low ( $k_{\text{eff}} < 0.52$ ). However, the HI-STORM 100 System for cask transfer (HI-TRAC, lead overpack) is flooded for loading and unloading operations, and thus, represents the limiting case in terms of reactivity.

The MPC-24EF, MPC-32F and MPC-68FF contain the same basket as the MPC-24E, MPC-32 and MPC-68, respectively. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF, the MPC-32 and MPC-32F, and the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-24E, MPC-32 and MPC-68 are valid for the MPC-24EF, MPC-32F and MPC-68FF, respectively, and no separate analyses for the MPC-24EF, MPC-32F and MPC-68FF are necessary. Therefore, throughout this chapter and unless otherwise noted, ‘MPC-68’ refers to ‘MPC-68 and/or MPC-68FF’, ‘MPC-24E’ or ‘MPC-24E/EF’ refers to ‘MPC-24E and/or MPC-24EF’, and ‘MPC-32’ or ‘MPC-32/32F’ refers to ‘MPC-32 and/or MPC-32F’.

Confirmation of the criticality safety of the HI-STORM 100 System was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package [6.4.1]. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors

<sup>†</sup> Analyses for the HI-STAR System have previously been submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, and MCNP4a [6.1.4] were used. The CASMO-3 and MCNP4a calculations identify those tolerances that cause a positive reactivity effect, enabling the subsequent Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24 and MPC-24E) or cell spacing (MPC-32, MPC-68, and MPC-68F), (3) the  $^{10}\text{B}$  loading of the neutron absorber panels, and (4) the soluble boron concentration in the water. The critical experiment benchmarking is presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.
- 10CFR72.124, Criteria For Nuclear Criticality Safety.
- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.
- Consistent with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.
- Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber.



- The fuel stack density is conservatively assumed to be at least 96% of theoretical (10.522 g/cm<sup>3</sup>) for all criticality analyses. Fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels may be slightly greater than 96% of theoretical, the actual stack density will be less.
- No credit is taken for the <sup>234</sup>U and <sup>236</sup>U in the fuel.
- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.
- When credit is taken for soluble boron, a <sup>10</sup>B content of 18.0 wt% in boron is assumed.
- Neutron absorption in minor structural members and optional heat conduction elements is neglected, i.e., spacer grids, basket supports, basket shims, and optional aluminum heat conduction elements are replaced by water.
- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.
- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.
- Planar-averaged enrichments are assumed for BWR fuel. (Consistent with NUREG-1536, analysis is presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)
- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.
- For evaluation of the bias, all benchmark calculations that result in a  $k_{\text{eff}}$  greater than 1.0 are conservatively truncated to 1.0000, consistent with NUREG-1536.
- The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.
- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.
- Regarding the position of assemblies in the basket, configurations with centered and eccentric positioning of assemblies in the fuel storage locations are considered. For further discussions see Section 6.3.3.

- For intact fuel assemblies, as defined in the glossary, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods. The number of dummy rods used to replace missing fuel rods is not limited.

Results of the design basis criticality safety calculations for single internally flooded HI-TRAC transfer casks with full water reflection on all sides (limiting cases for the HI-STORM 100 System), and for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System), loaded with intact fuel assemblies are listed in Tables 6.1.1 through 6.1.8, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. Comparing corresponding results for the HI-TRAC and HI-STAR demonstrates that the overpack material does not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 System and vice versa. In addition, a few results for single internally dry (no moderator) HI-STORM storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are listed to confirm the low reactivity of the HI-STORM 100 System in storage.

For each of the MPC designs, minimum soluble boron concentration (if applicable) and fuel assembly classes<sup>††</sup>, Tables 6.1.1 through 6.1.8 list the bounding maximum  $k_{\text{eff}}$  value, and the associated maximum allowable enrichment. The maximum allowed enrichments and the minimum soluble boron concentrations are also listed in Section 2.1.9. The candidate fuel assemblies, that are bounded by those listed in Tables 6.1.1 through 6.1.8, are given in Section 6.2.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) loaded with damaged fuel assemblies or a combination of intact and damaged fuel assemblies are listed in Tables 6.1.9 through 6.1.12. The results include the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs qualified for damaged fuel and/or fuel debris (MPC-24E, MPC-68, MPC-68F, MPC-32), Tables 6.1.9 through 6.1.12 indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum  $k_{\text{eff}}$  value, the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration. For the permissible location of DFCs see Subsection 6.4.4.2. The maximum allowed enrichments are also listed in Section 2.1.9.

A table listing the maximum  $k_{\text{eff}}$  (including bias, uncertainties, and calculational statistics), calculated  $k_{\text{eff}}$ , standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum  $k_{\text{eff}}$  values for the HI-STORM 100 System are below the limiting design criteria ( $k_{\text{eff}} < 0.95$ ) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is

<sup>††</sup> For each array size (e.g., 6x6, 7x7, 14x14, etc.), the fuel assemblies have been subdivided into a number of assembly classes, where an assembly class is defined in terms of the (1) number of fuel rods; (2) pitch; (3) number and location of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Section 6.2.

most limiting, are presented in Section 6.4. The capability of the HI-STORM 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

Table 6.1.1  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24  
(no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	4.6	0.3080	0.9283	0.9296
14x14B	4.6	---	0.9237	0.9228
14x14C	4.6	---	0.9274	0.9287
14x14D	4.0	---	0.8531	0.8507
14x14E	5.0 <sup>‡</sup>	---	0.7627	0.7627
15x15A	4.1	---	0.9205	0.9204
15x15B	4.1	---	0.9387	0.9388
15x15C	4.1	---	0.9362	0.9361
15x15D	4.1	---	0.9354	0.9367
15x15E	4.1	---	0.9392	0.9368
15x15F	4.1	0.3648	0.9393 <sup>††</sup>	0.9395 <sup>†††</sup>
15x15G	4.0	---	0.8878	0.8876
15x15H	3.8	---	0.9333	0.9337
16x16A	4.6	0.3588	0.9322	0.9327
16x16B	4.6	---	---	0.9370
16x16C	4.6	---	---	0.9090
17x17A	4.0	0.3243	0.9378	0.9368
17x17B	4.0	---	0.9318	0.9324
17x17C	4.0	---	0.9319	0.9336

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

<sup>†</sup> The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible  $k_{effective}$ , including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

<sup>‡</sup> (Footnote deleted)

<sup>††</sup> KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9383.

<sup>†††</sup> KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9378.

Table 6.1.2  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24  
WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8884
14x14B	5.0	---	---	0.8900
14x14C	5.0	---	---	0.8950
14x14D	5.0	---	---	0.8518
14x14E	5.0 <sup>‡</sup>	---	---	0.7132
15x15A	5.0	---	---	0.9119
15x15B	5.0	---	---	0.9284
15x15C	5.0	---	---	0.9236
15x15D	5.0	---	---	0.9261
15x15E	5.0	---	---	0.9265
15x15F	5.0	0.4013	0.9301	0.9314
15x15G	5.0	---	---	0.8939
15x15H	5.0	---	0.9345	0.9366
16x16A	5.0	---	---	0.8993
16x16B	5.0	---	---	0.9023
16x16C	5.0	---	---	0.8808
17x17A	5.0	---	---	0.9264
17x17B	5.0	---	---	0.9284
17x17C	5.0	---	0.9296	0.9294

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible  $k$ -effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡ (Footnote deleted)

Table 6.1.3  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E  
(no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.9380
14x14B	5.0	---	---	0.9312
14x14C	5.0	---	---	0.9356
14x14D	5.0	---	---	0.8875
14x14E	5.0 <sup>‡</sup>	---	---	0.7651
15x15A	4.5	---	---	0.9336
15x15B	4.5	---	---	0.9465
15x15C	4.5	---	---	0.9462
15x15D	4.5	---	---	0.9440
15x15E	4.5	---	---	0.9455
15x15F	4.5	0.3699	0.9465	0.9468
15x15G	4.5	---	---	0.9054
15x15H	4.2	---	---	0.9423
16x16A	5.0	---	---	0.9394
16x16B	5.0	---	---	0.9436
16x16C	5.0	---	---	0.9207
17x17A	4.4	---	0.9467	0.9447
17x17B	4.4	---	---	0.9421
17x17C	4.4	---	---	0.9433

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡ (Footnote deleted)

Table 6.1.4  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E  
WITH 300 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8963
14x14B	5.0	---	---	0.8974
14x14C	5.0	---	---	0.9031
14x14D	5.0	---	---	0.8588
14x14E	5.0 <sup>‡</sup>	---	---	0.7249
15x15A	5.0	---	---	0.9161
15x15B	5.0	---	---	0.9321
15x15C	5.0	---	---	0.9271
15x15D	5.0	---	---	0.9290
15x15E	5.0	---	---	0.9309
15x15F	5.0	0.3897	0.9333	0.9332
15x15G	5.0	---	---	0.8972
15x15H	5.0	---	0.9399	0.9399
16x16A	5.0	---	---	0.9060
16x16B	5.0	---	---	0.9080
16x16C	5.0	---	---	0.8875
17x17A	5.0	---	0.9320	0.9332
17x17B	5.0	---	---	0.9316
17x17C	5.0	---	---	0.9312

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡ (Footnote deleted)

Table 6.1.5  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32  
FOR 4.1% ENRICHMENT

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Concentration (ppm)*	Maximum <sup>†</sup> $k_{eff}$		
			HI-STORM	HI-TRAC	HI-STAR
14x14A	4.1	1300	---	---	0.9041
14x14B	4.1	1300	---	---	0.9257
14x14C	4.1	1300	---	---	0.9423
14x14D	4.1	1300	---	---	0.8970
14x14E	4.1	1300	---	---	0.7340
15x15A	4.1	1800	---	---	0.9206
15x15B	4.1	1800	---	---	0.9397
15x15C	4.1	1800	---	---	0.9266
15x15D	4.1	1900	---	---	0.9384
15x15E	4.1	1900	---	---	0.9365
15x15F	4.1	1900	0.4691	0.9403	0.9411
15x15G	4.1	1800	---	---	0.9147
15x15H	4.1	1900	---	---	0.9276
15x15I	4.1	1800	---	---	0.9340
16x16A	4.1	1400	---	---	0.9375
16x16B	4.1	1400	---	---	0.9354
16x16C	4.1	1400	---	---	0.9178
17x17A	4.1	1600	---	---	0.9421
17x17B	4.1	1900	---	---	0.9309
17x17C	4.1	1900	---	0.9365	0.9355

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

\* For maximum allowable enrichments between 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible  $k$ -effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.



Table 6.1.6  
Attachment 6 to Docket Letter 5044874  
BOUNDED MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32  
FOR 5.0% ENRICHMENT

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Concentration (ppm)*	Maximum <sup>†</sup> $k_{eff}$		
			HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	1900	---	---	0.9000
14x14B	5.0	1900	---	---	0.9214
14x14C	5.0	1900	---	---	0.9480
14x14D	5.0	1900	---	---	0.9050
14x14E	5.0	1900	---	---	0.7415
15x15A	5.0	2500	---	---	0.9230
15x15B	5.0	2500	---	---	0.9429
15x15C	5.0	2500	---	---	0.9307
15x15D	5.0	2600	---	---	0.9466
15x15E	5.0	2600	---	---	0.9434
15x15F	5.0	2600	0.5142	0.9470	0.9483
15x15G	5.0	2500	---	---	0.9251
15x15H	5.0	2600	---	---	0.9333
15x15I	5.0	2500	---	---	0.9402
16x16A	5.0	2000	---	---	0.9429
16x16B	5.0	2000	---	---	0.9378
16x16C	5.0	2000	---	---	0.9208
17x17A	5.0	2200	---	---	0.9475
17x17B	5.0	2600	---	---	0.9371
17x17C	5.0	2600	---	0.9436	0.9437

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

\* For maximum allowable enrichments between 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible  $k_{eff}$ , including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.7  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7 <sup>††</sup>	---	0.7886	0.7888 <sup>†††</sup>
6x6B <sup>‡</sup>	2.7 <sup>††</sup>	---	0.7833	0.7824 <sup>†††</sup>
6x6C	2.7 <sup>††</sup>	0.2759	0.8024	0.8021 <sup>†††</sup>
7x7A	2.7 <sup>††</sup>	---	0.7963	0.7974 <sup>†††</sup>
7x7B	4.2	0.4061	0.9385	0.9386
8x8A	2.7 <sup>††</sup>	---	0.7690	0.7697 <sup>†††</sup>
8x8B	4.2	0.3934	0.9427	0.9416
8x8C	4.2	0.3714	0.9429	0.9425
8x8D	4.2	---	0.9408	0.9403
8x8E	4.2	---	0.9309	0.9312
8x8F	4.0	---	0.9396	0.9411

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† This calculation was performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum  $k_{eff}$  value is conservative.

††† This calculation was performed for a  $^{10}\text{B}$  loading of 0.0067 g/cm<sup>2</sup>, which is 75% of a minimum  $^{10}\text{B}$  loading of 0.0089 g/cm<sup>2</sup>. The minimum  $^{10}\text{B}$  loading in the MPC-68 is at least 0.0310 g/cm<sup>2</sup>. Therefore, the listed maximum  $k_{eff}$  value is conservative.

‡ Assemblies in this class contain both MOX and UO<sub>2</sub> pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the specification of authorized contents in Section 2.1.9.

Table 6.1.7 (continued)  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
9x9A	4.2	0.3365	0.9434	0.9417
9x9B	4.2	---	0.9417	0.9436
9x9C	4.2	---	0.9377	0.9395
9x9D	4.2	---	0.9387	0.9394
9x9E	4.0		0.9402	0.9401
9x9F	4.0	---	0.9402	0.9401
9x9G	4.2	---	0.9307	0.9309
10x10A	4.2	0.3379	0.9448 <sup>‡‡</sup>	0.9457*
10x10B	4.2	---	0.9443	0.9436
10x10C	4.2	---	0.9430	0.9433
10x10D	4.0	---	0.9383	0.9376
10x10E	4.0	---	0.9157	0.9185

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡‡ KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9451.

\* KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9453.

Table 6.1.8  
Attachment 6 to Holtec Letter 5014874

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7 <sup>††</sup>	---	0.7886	0.7888
6x6B <sup>†††</sup>	2.7	---	0.7833	0.7824
6x6C	2.7	0.2759	0.8024	0.8021
7x7A	2.7	---	0.7963	0.7974
8x8A	2.7	---	0.7690	0.7697

Notes:

The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

These calculations were performed for a  $^{10}\text{B}$  loading of 0.0067 g/cm<sup>2</sup>, which is 75% of a minimum  $^{10}\text{B}$  loading of 0.0089 g/cm<sup>2</sup>. The minimum  $^{10}\text{B}$  loading in the MPC-68F is 0.010 g/cm<sup>2</sup>. Therefore, the listed maximum  $k_{eff}$  values are conservative.

<sup>†</sup> The term "maximum  $k_{eff}$ " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

<sup>††</sup> These calculations were performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum  $k_{eff}$  values are conservative.

<sup>†††</sup> Assemblies in this class contain both MOX and UO<sub>2</sub> pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in the specification of authorized contents in Section 2.1.9.

Table 6.1.9  
Attachment 6 to Holtec Letter 5014874  
BOUNDED MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-24E  
WITH UP TO 4 DFCs

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )		Minimum Soluble Boron Concentration (ppm)	Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris		HI-TRAC	HI-STAR
All PWR Classes	4.0	4.0	0	0.9486	0.9480
All PWR Classes <sup>‡</sup>	5.0	5.0	600	0.9177	0.9185

Table 6.1.10  
BOUNDED MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68 AND MPC-68F  
WITH UP TO 68 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )		Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
6x6A, 6x6B, 6x6C, 7x7A, 8x8A	2.7	2.7	0.8024	0.8021

Table 6.1.11  
BOUNDED MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68  
WITH UP TO 16 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )		Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
All BWR Classes	3.7	4.0	0.9328	0.9328

<sup>‡</sup> (Footnote deleted)

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-32  
WITH UP TO 8 DFCs

Fuel Assembly Class of Intact Fuel	Maximum Allowable Enrichment for Intact Fuel and Damaged Fuel/Fuel Debris (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Content (ppm) <sup>†</sup>	Maximum $k_{eff}$	
			HI-TRAC	HI-STAR
14x14A, B, C, D, E	4.1	1500	---	0.9336
	5.0	2300	---	0.9269
15x15A, B, C, G, I	4.1	1900	0.9349	0.9350
	5.0	2700	---	0.9365
15x15D, E, F, H	4.1	2100	---	0.9340
	5.0	2900	0.9382	0.9397
16x16A, B, C	4.1	1500	---	0.9348
	5.0	2300	---	0.9299
17x17A	4.1	1800	---	0.9311
	5.0	2600	---	0.9298
17x17B, C	4.1	2100	---	0.9294
	5.0	2900	---	0.9367

<sup>†</sup> For maximum allowable enrichments between 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

## 6.2 SPENT FUEL LOADING

Specifications for the BWR and PWR fuel assemblies that were analyzed are given in Tables 6.2.1 and 6.2.2, respectively. For the BWR fuel characteristics, the number and dimensions for the water rods are the actual number and dimensions. For the PWR fuel characteristics, the actual number and dimensions of the control rod guide tubes and thimbles are used. Table 6.2.1 lists 72 unique BWR assemblies while Table 6.2.2 lists 47 unique PWR assemblies, all of which were explicitly analyzed for this evaluation. Examination of Tables 6.2.1 and 6.2.2 reveals that there are a large number of minor variations in fuel assembly dimensions.

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in defining the authorized contents could limit the applicability of the HI-STORM 100 System. To resolve this limitation, bounding criticality analyses are presented in this section for a number of defined fuel assembly classes for both fuel types (PWR and BWR). The results of the bounding criticality analyses justify using bounding fuel dimensions for defining the authorized contents.

### 6.2.1 Definition of Assembly Classes

For each array size (e.g., 6x6, 7x7, 15x15, etc.), the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; (3) number and locations of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Tables 6.2.1 and 6.2.2, respectively. It should be noted that these assembly classes are unique to this evaluation and are not known to be consistent with any class designations in the open literature.

For each assembly class, calculations have been performed for all of the dimensional variations for which data is available (i.e., all data in Tables 6.2.1 and 6.2.2). These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

Therefore, for each assembly class, a bounding assembly was defined based on the above characteristics and a calculation for the bounding assembly was performed to demonstrate compliance with the regulatory requirement of  $k_{\text{eff}} < 0.95$ . In some assembly classes this bounding assembly corresponds directly to one of the actual (real) assemblies; while in most assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than

one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, the authorized contents in Section 2.1.9 are defined in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R). Additionally, parametric studies were performed for a PWR assembly (the 15x15F assembly class) in the MPC-24 and MPC-32 with soluble boron in the water flooding the MPC. The results of these studies are shown in Table 6.2.3 through 6.2.5, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod/guide tube thickness, (5) artificially replacing the Zircaloy water rod tubes/guide tubes with water, (6) maximizing the channel thickness (for BWR Assemblies), and (7) increasing the active length. These results, and the many that follow, justify the approach for using bounding dimensions for defining the authorized contents. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. As these studies were performed with and without soluble boron, they also demonstrate that the bounding dimensions are valid independent of the soluble boron concentration.

As mentioned, the bounding approach used in these analyses often results in a maximum  $k_{\text{eff}}$  value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

## 6.2.2 Intact PWR Fuel Assemblies

### 6.2.2.1 Intact PWR Fuel Assemblies in the MPC-24 without Soluble Boron

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9395). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.20) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.6 through 6.2.22 for the fully flooded condition without soluble boron in the water.

Tables 6.2.6 through 6.2.22 show the maximum  $k_{\text{eff}}$  values for the assembly classes that are acceptable for storage in the MPC-24. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All



calculations for the MPC-24 were performed for a  $^{10}\text{B}$  loading of  $0.020 \text{ g/cm}^2$ , which is 75% of the minimum loading of  $0.0267 \text{ g/cm}^2$  for Boral, or 90% of the minimum loading of  $0.0223 \text{ g/cm}^2$  for Metamic. The maximum allowable enrichment in the MPC-24 varies from 3.8 to 5.0 wt%  $^{235}\text{U}$ , depending on the assembly class, and is defined in Tables 6.2.6 through 6.2.22. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.6 through 6.2.22 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.6). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.9), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

The results of the analyses for the MPC-24, which were performed for all assemblies in each class (see Tables 6.2.6 through 6.2.22), further confirm the validity of the bounding dimensions established in Section 6.2.1. Thus, for all following calculations, namely analyses of the MPC-24E, MPC-32, and MPC-24 with soluble boron present in the water, only the bounding assembly in each class is analyzed.

#### 6.2.2.2 Intact PWR Fuel Assemblies in the MPC-24 with Soluble Boron

Additionally, the HI-STAR 100 system is designed to allow credit for the soluble boron typically present in the water of PWR spent fuel pools. For a minimum soluble boron concentration of 400ppm, the maximum allowable fuel enrichment is 5.0 wt%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.2 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9366). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

#### 6.2.2.3 Intact PWR Assemblies in the MPC-24E with and without Soluble Boron

The MPC-24E is a variation of the MPC-24, which provides for storage of higher enriched fuel than the MPC-24 through optimization of the storage cell layout. The MPC-24E also allows for the loading of up to 4 PWR Damaged Fuel Containers (DFC) with damaged PWR fuel and PWR fuel debris. The requirements for damaged fuel and fuel debris in the MPC-24E is discussed in Section 6.2.4.3.

Without credit for soluble boron, the maximum allowable fuel enrichment varies between 4.2 and 5.0 wt%  $^{235}\text{U}$ , depending on the assembly classes as identified in Tables 6.2.6 through 6.2.22. The maximum allowable enrichment for each assembly class is listed in Table 6.1.3, together with the maximum  $k_{\text{eff}}$  for the bounding assembly in the assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15F assembly class at 4.5% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  shown in Table 6.1.3). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

For a minimum soluble boron concentration of 300ppm, the maximum allowable fuel enrichment is 5.0 wt%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.4 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  shown in Table 6.1.4). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

#### 6.2.2.4 Intact PWR Assemblies in the MPC-32

When loading any PWR fuel assembly in the MPC-32, a minimum soluble boron concentration is required.

For a maximum allowable fuel enrichment of 4.1 wt%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1300ppm and 1900ppm is required, depending on the assembly class. Table 6.1.5 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 17x17A assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  shown in Table 6.1.5). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

For a maximum allowable fuel enrichment of 5.0 wt%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1900ppm and 2600ppm is required, depending on the assembly class. Table 6.1.6 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15F assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  shown in Table 6.1.6). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

It is desirable to limit the soluble boron concentration to a level appropriate for the maximum enrichment in a basket, since this prevents adding soluble boron unnecessarily to the spent fuel pool during loading and unloading operations. This approach requires a minimum soluble boron level as a function of the maximum allowable enrichment, which can be directly derived by linear interpolation from the calculations at 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$  shown in Tables 6.1.5 and 6.1.6. Since the maximum  $k_{\text{eff}}$  is a near linear function of both enrichment and soluble boron

concentration, linear interpolation is both appropriate and sufficient. Further, studies have shown that this approach results in maximum  $k_{\text{eff}}$  values for enrichments between 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$  that are lower than those maximum  $k_{\text{eff}}$  values calculated at 4.1 wt% and 5.0 wt%  $^{235}\text{U}$  in Tables 6.1.5 and 6.1.6.

### 6.2.3 Intact BWR Fuel Assemblies in the MPC-68

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 are summarized in Tables 6.2.23 through 6.2.40 for the fully flooded condition. In all cases, the gadolinia ( $\text{Gd}_2\text{O}_3$ ) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.23 through 6.2.40 show the maximum  $k_{\text{eff}}$  values for assembly classes that are acceptable for storage in the MPC-68. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 were performed with a  $^{10}\text{B}$  loading of  $0.0279 \text{ g/cm}^2$ , which is 75% of the minimum loading of  $0.0372 \text{ g/cm}^2$  for Boral, or 90% of the minimum loading of  $0.031 \text{ g/cm}^2$  for Metamic. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a  $^{10}\text{B}$  loading of  $0.0067 \text{ g/cm}^2$ . The maximum allowable enrichment in the MPC-68 varies from 2.7 to 4.2 wt%  $^{235}\text{U}$ , depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.7 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68.

Tables 6.2.23 through 6.2.40 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.27), the authorized contents dimensions are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain assemblies with thicker channels (e.g., 120 mils), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.23). All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum  $k_{\text{eff}}$  values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the specification of the authorized contents has no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum  $k_{\text{eff}}$  value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils thick; and (3) the active fuel length of the full length rods is 155 inches. Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

#### 6.2.4 BWR and PWR Damaged Fuel Assemblies and Fuel Debris

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in the glossary. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs). Five different DFC types with different cross sections are considered; three types for BWR fuel and two for PWR fuel. DFCs containing BWR damaged fuel assemblies or fuel debris may be stored in the MPC-68 or MPC-68F. DFCs containing PWR damaged fuel or fuel debris may be stored in the MPC-24E or MPC-32. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4.

##### 6.2.4.1 Damaged BWR Fuel Assemblies and BWR Fuel Debris in Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

Tables 6.2.41 through 6.2.45 show the maximum  $k_{\text{eff}}$  values for the five assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a  $^{10}\text{B}$  loading of  $0.0067 \text{ g/cm}^2$ , which is 75% of a minimum loading,  $0.0089 \text{ g/cm}^2$ . However, because the practical manufacturing lower limit for minimum

$^{10}\text{B}$  loading is  $0.01 \text{ g/cm}^2$ , the minimum  $^{10}\text{B}$  loading of  $0.01 \text{ g/cm}^2$  is specified on the drawing in Section 1.5, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt%  $^{235}\text{U}$ , while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt%  $^{235}\text{U}$  in the specification of the authorized contents. Therefore, the maximum  $k_{\text{eff}}$  values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.41 through 6.2.45 for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt%  $^{235}\text{U}$  enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual  $^{10}\text{B}$  loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a  $^{10}\text{B}$  loading of  $0.0089 \text{ g/cm}^2$ , which conservatively bounds the analysis of damaged BWR fuel assemblies in an MPC-68 with a minimum  $^{10}\text{B}$  loading of  $0.0372 \text{ g/cm}^2$ , damaged BWR fuel assemblies or fuel debris may be stored in the MPC-68 or MPC-68F.

Tables 6.2.41 through 6.2.45 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.43), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

#### 6.2.4.2 Damaged BWR Fuel Assemblies and Fuel Debris in the MPC-68

Damaged BWR fuel assemblies and fuel debris from all BWR classes may be loaded into the MPC-68 by restricting the locations of the DFCs to 16 specific cells on the periphery of the fuel basket. The MPC-68 may be loaded with up to 16 DFCs containing damaged fuel assemblies, with up to 8 DFCs containing fuel debris.

For all assembly classes, the enrichment of the damaged fuel or fuel debris is limited to a maximum of 4.0 wt%  $^{235}\text{U}$ , while the enrichment of the intact assemblies stored together with the damaged fuel is limited to a maximum of 3.7 wt%  $^{235}\text{U}$ . The maximum  $k_{\text{eff}}$  is 0.9328. The criticality evaluation of the damaged fuel assemblies and fuel debris in the MPC-68 is presented in Section 6.4.4.2.

### 6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris

In addition to storing intact PWR fuel assemblies, the HI-STORM 100 System is designed to store damaged PWR fuel assemblies and fuel debris (MPC-24E and MPC-32). Damaged fuel assemblies and fuel debris are defined in the glossary. Damaged PWR fuel assemblies and fuel debris are required to be loaded into PWR Damaged Fuel Containers (DFCs).

#### 6.2.4.3.1 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-24E

Up to four DFCs may be stored in the MPC-24E. When loaded with damaged fuel and/or fuel debris, the maximum enrichment for intact and damaged fuel is 4.0 wt%  $^{235}\text{U}$  for all assembly classes listed in Table 6.2.6 through 6.2.22 without credit for soluble boron. The maximum  $k_{\text{eff}}$  for these classes is 0.9486. For a minimum soluble boron concentration of 600ppm, the maximum enrichment for intact and damaged fuel is 5.0 wt%  $^{235}\text{U}$  for all assembly classes listed in Table 6.2.6 through 6.2.22. The criticality evaluation of the damaged fuel is presented in Subsection 6.4.4.2.

#### 6.2.4.3.2 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-32

Up to eight DFCs may be stored in the MPC-32. For a maximum allowable fuel enrichment of 4.1 wt%  $^{235}\text{U}$  for intact fuel, damaged fuel and fuel debris for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1500ppm and 2100ppm is required, depending on the assembly class of the intact assembly. For a maximum allowable fuel enrichment of 5.0 wt%  $^{235}\text{U}$  for intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration between 2300ppm and 2900ppm is required, depending on the assembly class of the intact assembly. Table 6.1.12 shows the maximum  $k_{\text{eff}}$  by assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.

As discussed in Section 6.2.2.4, it is desirable to limit the soluble boron concentration to a level appropriate for the maximum enrichment in a basket. The discussion presented in Section 6.2.2.4 is also applicable for the MPC-32 with damaged fuel or fuel debris. Further, studies with damaged fuel have shown that this approach also results in maximum  $k_{\text{eff}}$  values that are lower than those  $k_{\text{eff}}$  values calculated for 4.1 wt% and 5.0 wt%  $^{235}\text{U}$  in Table 6.1.12.

### 6.2.5 Thoria Rod Canister

Additionally, the HI-STORM 100 System is designed to store a Thoria Rod Canister in the MPC-68 or MPC-68F. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC is very low compared to the approved fuel assemblies (The  $^{235}\text{U}$  content of these rods correspond to  $\text{UO}_2$  rods with an initial enrichment of approximately up to 1.7 wt%  $^{235}\text{U}$ ). It is therefore permissible to the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. Specifications of the canister and the Thoria Rods that are used in the criticality evaluation are given in Table 6.2.46. The criticality

evaluations are presented in Subsection 6.4.6.

Table 6.2.1 (page 1 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694	36	0.5645	0.0350	0.4940	110.0	0	n/a	n/a	0.060	4.290
6x6A02	Zr	0.694	36	0.5645	0.0360	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A03	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A04	Zr	0.694	36	0.5550	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A05	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A06	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6A07	Zr	0.700	36	0.5555	0.03525	0.4780	110.0	0	n/a	n/a	0.060	4.290
6x6A08	Zr	0.710	36	0.5625	0.0260	0.4980	110.0	0	n/a	n/a	0.060	4.290
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B02	Zr	0.694	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B03	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B04	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6B05	Zr	0.710	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6C Assembly Class												
6x6C01	Zr	0.740	36	0.5630	0.0320	0.4880	77.5	0	n/a	n/a	0.060	4.542
7x7A Assembly Class												
7x7A01	Zr	0.631	49	0.4860	0.0328	0.4110	80	0	n/a	n/a	0.060	4.542



Table 6.2.1 (page 2 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738	49	0.5630	0.0320	0.4870	150	0	n/a	n/a	0.080	5.278
7x7B02	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.102	5.291
7x7B03	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.080	5.278
7x7B04	Zr	0.738	49	0.5700	0.0355	0.4880	150	0	n/a	n/a	0.080	5.278
7x7B05	Zr	0.738	49	0.5630	0.0340	0.4775	150	0	n/a	n/a	0.080	5.278
7x7B06	Zr	0.738	49	0.5700	0.0355	0.4910	150	0	n/a	n/a	0.080	5.278
8x8A Assembly Class												
8x8A01	Zr	0.523	64	0.4120	0.0250	0.3580	110	0	n/a	n/a	0.100	4.290
8x8A02	Zr	0.523	63	0.4120	0.0250	0.3580	120	0	n/a	n/a	0.100	4.290

Table 6.2.1 (page 3 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B02	Zr	0.636	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B03	Zr	0.640	63	0.4930	0.0340	0.4160	150	1	0.493	0.425	0.100	5.278
8x8B04	Zr	0.642	64	0.5015	0.0360	0.4195	150	0	n/a	n/a	0.100	5.278
8x8C Assembly Class												
8x8C01	Zr	0.641	62	0.4840	0.0350	0.4050	150	2	0.484	0.414	0.100	5.278
8x8C02	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.000	no channel
8x8C03	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.080	5.278
8x8C04	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C05	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C06	Zr	0.640	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.100	5.278
8x8C07	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C08	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.493	0.425	0.100	5.278
8x8C09	Zr	0.640	62	0.4930	0.0340	0.4160	150	2	0.493	0.425	0.100	5.278
8x8C10	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C11	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.215
8x8C12	Zr	0.636	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.120	5.215

Table 6.2.1 (page 4 of 7)

**BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS**  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640	60	0.4830	0.0320	0.4110	150	2 large/ 2 small	0.591/ 0.483	0.531/ 0.433	0.100	5.278
8x8D02	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.591	0.531	0.100	5.278
8x8D03	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.483	0.433	0.100	5.278
8x8D04	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.100	5.278
8x8D05	Zr	0.640	60	0.4830	0.0320	0.4100	150	1	1.34	1.26	0.100	5.278
8x8D06	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.120	5.278
8x8D07	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.080	5.278
8x8D08	Zr	0.640	61	0.4830	0.0300	0.4140	150	3	0.591	0.531	0.080	5.278
8x8E Assembly Class												
8x8E01	Zr	0.640	59	0.4930	0.0340	0.4160	150	5	0.493	0.425	0.100	5.278
8x8F Assembly Class												
8x8F01	Zr	0.609	64	0.4576	0.0290	0.3913	150	4 <sup>†</sup>	0.291 <sup>†</sup>	0.228 <sup>†</sup>	0.055	5.390
9x9A Assembly Class												
9x9A01	Zr	0.566	74	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A02	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A03	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.100	5.278
9x9A04	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.120	5.278

<sup>†</sup> Four rectangular water cross segments dividing the assembly into four quadrants

Table 6.2.1 (page 5 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9B Assembly Class												
9x9B01	Zr	0.569	72	0.4330	0.0262	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B02	Zr	0.569	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B03	Zr	0.572	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9C Assembly Class												
9x9C01	Zr	0.572	80	0.4230	0.0295	0.3565	150	1	0.512	0.472	0.100	5.278
9x9D Assembly Class												
9x9D01	Zr	0.572	79	0.4240	0.0300	0.3565	150	2	0.424	0.364	0.100	5.278
9x9E Assembly Class <sup>†</sup>												
9x9E01	Zr	0.572	76	0.4170	0.0265	0.3530	150	5	0.546	0.522	0.120	5.215
9x9E02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215

<sup>†</sup> The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in Section 2.1.9.

Table 6.2.1 (page 6 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9F Assembly Class*												
9x9F01	Zr	0.572	76	0.4430	0.0285	0.3745	150	5	0.546	0.522	0.120	5.215
9x9F02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215
9x9G Assembly Class												
9x9G01	Zr	0.572	72	0.4240	0.0300	0.3565	150	1	1.668	1.604	0.120	5.278
10x10A Assembly Class												
10x10A01	Zr	0.510	92	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A02	Zr	0.510	78	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A03	Zr	0.510	92/78	0.4040	0.0260	0.3450	155/90	2	0.980	0.920	0.100	5.278
10x10B Assembly Class												
10x10B01	Zr	0.510	91	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B02	Zr	0.510	83	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B03	Zr	0.510	91/83	0.3957	0.0239	0.3413	155/90	1	1.378	1.321	0.100	5.278

\* The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in Section 2.1.9.

Table 6.2.1 (page 7 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10C Assembly Class												
10x10C01	Zr	0.488	96	0.3780	0.0243	0.3224	150	5	1.227	1.165	0.055	5.347
10x10D Assembly Class												
10x10D01	SS	0.565	100	0.3960	0.0200	0.3500	83	0	n/a	n/a	0.08	5.663
10x10E Assembly Class												
10x10E01	SS	0.557	96	0.3940	0.0220	0.3430	83	4	0.3940	0.3500	0.08	5.663

Table 6.2.2 (page 1 of 5)  
 PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.527	0.493	0.0170
14x14A02	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.528	0.490	0.0190
14x14A03	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.526	0.492	0.0170
14x14B Assembly Class											
14x14B01	Zr	0.556	179	0.422	0.0243	0.3659	150	17	0.539	0.505	0.0170
14x14B02	Zr	0.556	179	0.417	0.0295	0.3505	150	17	0.541	0.507	0.0170
14x14B03	Zr	0.556	179	0.424	0.0300	0.3565	150	17	0.541	0.507	0.0170
14x14B04	Zr	0.556	179	0.426	0.0310	0.3565	150	17	0.541	0.507	0.0170
14x14C Assembly Class											
14x14C01	Zr	0.580	176	0.440	0.0280	0.3765	150	5	1.115	1.035	0.0400
14x14C02	Zr	0.580	176	0.440	0.0280	0.3770	150	5	1.115	1.035	0.0400
14x14C03	Zr	0.580	176	0.440	0.0260	0.3805	150	5	1.111	1.035	0.0380
14x14D Assembly Class											
14x14D01	SS	0.556	180	0.422	0.0165	0.3835	144	16	0.543	0.514	0.0145

Table 6.2.2 (page 2 of 5)  
 PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14E Assembly Class											
14x14E01 <sup>†</sup>	SS	0.453 and 0.441	162 3 8	0.3415 0.3415 0.3415	0.0120 0.0285 0.0200	0.313 0.280 0.297	102	0	n/a	n/a	n/a
14x14E02 <sup>†</sup>	SS	0.453 and 0.441	173	0.3415	0.0120	0.313	102	0	n/a	n/a	n/a
14x14E03 <sup>†</sup>	SS	0.453 and 0.441	173	0.3415	0.0285	0.0280	102	0	n/a	n/a	n/a
15x15A Assembly Class											
15x15A01	Zr	0.550	204	0.418	0.0260	0.3580	150	21	0.533	0.500	0.0165

<sup>†</sup> This is the fuel assembly used at Indian Point 1 (IP-1). This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. It has a different pitch in different sections of the assembly, and different fuel rod dimensions in some rods.



Table 6.2.2 (page 3 of 5)  
 PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.533	0.499	0.0170
15x15B02	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.546	0.512	0.0170
15x15B03	Zr	0.563	204	0.422	0.0243	0.3660	150	21	0.533	0.499	0.0170
15x15B04	Zr	0.563	204	0.422	0.0243	0.3659	150	21	0.545	0.515	0.0150
15x15B05	Zr	0.563	204	0.422	0.0242	0.3659	150	21	0.545	0.515	0.0150
15x15B06	Zr	0.563	204	0.420	0.0240	0.3671	150	21	0.544	0.514	0.0150
15x15C Assembly Class											
15x15C01	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.493	0.0255
15x15C02	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.511	0.0165
15x15C03	Zr	0.563	204	0.424	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15C04	Zr	0.563	204	0.417	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15D Assembly Class											
15x15D01	Zr	0.568	208	0.430	0.0265	0.3690	150	17	0.530	0.498	0.0160
15x15D02	Zr	0.568	208	0.430	0.0265	0.3686	150	17	0.530	0.498	0.0160
15x15D03	Zr	0.568	208	0.430	0.0265	0.3700	150	17	0.530	0.499	0.0155
15x15D04	Zr	0.568	208	0.430	0.0250	0.3735	150	17	0.530	0.500	0.0150
15x15E Assembly Class											
15x15E01	Zr	0.568	208	0.428	0.0245	0.3707	150	17	0.528	0.500	0.0140
15x15F Assembly Class											
15x15F01	Zr	0.568	208	0.428	0.0230	0.3742	150	17	0.528	0.500	0.0140

Table 6.2.2 (page 4 of 5)  
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563	204	0.422	0.0165	0.3825	144	21	0.543	0.514	0.0145
15x15H Assembly Class											
15x15H01	Zr	0.568	208	0.414	0.0220	0.3622	150	17	0.528	0.500	0.0140
15x15I Assembly Class											
15x15I01 <sup>†</sup>	Zr	0.550	216	0.413	0.0230	0.3600	150	9	n/a	n/a	0.0140
16x16A Assembly Class											
16x16A01	Zr	0.506	236	0.382	0.0250	0.3255	150	5	0.980	0.900	0.0400
16x16A02	Zr	0.506	236	0.382	0.0250	0.3250	150	5	0.980	0.900	0.0400
16x16A03	Zr	0.506	236	0.382	0.0235	0.3255	150	5	0.970	0.900	0.0350
16x16B Assembly Class											
16x16B01	Zr	0.506	236	0.374	0.0225	0.3225	150	5	0.980	0.900	0.0400
16x16C Assembly Class											
16x16C01	Zr	0.485	235	0.374	0.0225	0.3225	150	21	0.467	0.435	0.0157
17x17A Assembly Class											
17x17A01	Zr	0.496	264	0.360	0.0225	0.3088	150	25	0.474	0.442	0.0160
17x17A02	Zr	0.496	264	0.360	0.0250	0.3030	150	25	0.480	0.448	0.0160

<sup>†</sup> This assembly is a 15x15 assembly with 1 instrument tube and 8 guide bars (solid Zr) with outer diameter of 0.413 inches.

Table 6.2.2 (page 5 of 5)  
 PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
17x17B Assembly Class											
17x17B01	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.482	0.450	0.0160
17x17B02	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.474	0.442	0.0160
17x17B03	Zr	0.496	264	0.376	0.0240	0.3215	150	25	0.480	0.448	0.0160
17x17B04	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.427	0.399	0.0140
17x17B05	Zr	0.496	264	0.374	0.0240	0.3195	150	25	0.482	0.450	0.0160
17x17B06	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.480	0.452	0.0140
17x17C Assembly Class											
17x17C01	Zr	0.502	264	0.379	0.0240	0.3232	150	25	0.472	0.432	0.0200
17x17C02	Zr	0.502	264	0.377	0.0220	0.3252	150	25	0.472	0.432	0.0200

Table 6.2.3  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-68  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	water rod thickness	channel thickness
8x8C04 (GE8x8R)	reference	0.9307	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
increase pellet OD (+0.001)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.411	0.030	0.100
decrease pellet OD (-0.001)	-0.0008	0.9299	0.0009	0.483	0.419	0.032	0.409	0.030	0.100
increase clad ID (+0.004)	+0.0027	0.9334	0.0007	0.483	0.423	0.030	0.410	0.030	0.100
decrease clad ID (-0.004)	-0.0034	0.9273	0.0007	0.483	0.415	0.034	0.410	0.030	0.100
increase clad OD (+0.004)	-0.0041	0.9266	0.0008	0.487	0.419	0.034	0.410	0.030	0.100
decrease clad OD (-0.004)	+0.0023	0.9330	0.0007	0.479	0.419	0.030	0.410	0.030	0.100
increase water rod thickness (+0.015)	-0.0019	0.9288	0.0008	0.483	0.419	0.032	0.410	0.045	0.100
decrease water rod thickness (-0.015)	+0.0001	0.9308	0.0008	0.483	0.419	0.032	0.410	0.015	0.100
remove water rods (i.e., replace the water rod tubes with water)	+0.0021	0.9328	0.0008	0.483	0.419	0.032	0.410	0.000	0.100
remove channel	-0.0039	0.9268	0.0009	0.483	0.419	0.032	0.410	0.030	0.000
increase channel thickness (+0.020)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.410	0.030	0.120
reduced active length (120 Inches)	-0.0007	0.9300	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
reduced active length (90 Inches)	-0.0043	0.9264	0.0007	0.483	0.419	0.032	0.410	0.030	0.100

Table 6.2.4  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC 24 with 400ppm soluble boron concentration  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9271	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	-0.0008	0.9263	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	-0.0002	0.9269	0.0005	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0040	0.9311	0.0005	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0033	0.9238	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0042	0.9229	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0035	0.9306	0.0005	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0008	0.9263	0.0005	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	+0.0006	0.9277	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	+0.0028	0.9299	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	-0.0318	0.8953	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2.5  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-32 with 2600ppm soluble boron concentration  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9389	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	+0.0019	0.9408	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	0.0000	0.9389	0.0004	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0015	0.9404	0.0004	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0015	0.9374	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0002	0.9387	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0007	0.9397	0.0004	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0003	0.9387	0.0004	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	-0.0005	0.9384	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	-0.0005	0.9385	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	+0.0039	0.9428	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2. 6  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 14X14A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14A (4.6% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14A01	0.9295	0.9252	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
14x14A02	0.9286	0.9242	0.0008	0.400	0.3514	0.0243	0.3444	150	0.019
14x14A03	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
Dimensions Listed for Authorized Contents				0.400 (min.)	0.3514 (max.)		0.3444 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (14x14A03)	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017

Table 6.2.7  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 14X14B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14B (4.6% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14B01	0.9159	0.9117	0.0007	0.422	0.3734	0.0243	0.3659	150	0.017
14x14B02	0.9169	0.9126	0.0008	0.417	0.3580	0.0295	0.3505	150	0.017
14x14B03	0.9110	0.9065	0.0009	0.424	0.3640	0.0300	0.3565	150	0.017
14x14B04	0.9084	0.9039	0.0009	0.426	0.3640	0.0310	0.3565	150	0.017
Dimensions Listed for Authorized Contents				0.417 (min.)	0.3734 (max.)		0.3659 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (B14x14B01)	0.9228	0.9185	0.0008	0.417	0.3734	0.0218	0.3659	150	0.017



Table 6.2.8  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14C (4.6% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 176 fuel rods, 5 guide tubes, pitch=0.580, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9258	0.9215	0.0008	0.440	0.3840	0.0280	0.3765	150	0.040
14x14C02	0.9265	0.9222	0.0008	0.440	0.3840	0.0280	0.3770	150	0.040
14x14C03	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038
Dimensions Listed for Authorized Contents				0.440 (min.)	0.3880 (max.)		0.3805 (max.)	150 (max.)	0.038 (min.)
bounding dimensions (14x14C03)	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038

Table 6.2.9  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 14X14D ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14D (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 180 fuel rods, 16 guide tubes, pitch=0.556, SS clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14D01	0.8507	0.8464	0.0008	0.422	0.3890	0.0165	0.3835	144	0.0145
Dimensions Listed for Authorized Contents				0.422 (min.)	0.3890 (max.)		0.3835 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.10  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14E ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14E (5.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 173 fuel rods, 0 guide tubes, pitch=0.453 and 0.441, SS clad <sup>†</sup>									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length <sup>††</sup>	guide tube thickness
14x14E01	0.7598	0.7555	0.0008	0.3415	0.3175 0.2845 0.3015	0.0120 0.0285 0.0200	0.3130 0.2800 0.2970	102	0.0000
14x14E02	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000
14x14E03	0.6952	0.6909	0.0008	0.3415	0.2845	0.0285	0.2800	102	0.0000
Dimensions Listed for Authorized Contents				0.3415 (min.)	0.3175 (max.)		0.3130 (max.)	102 (max.)	0.0000 (min.)
Bounding dimensions (14x14E02)	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000

<sup>†</sup> This is the IP-1 fuel assembly at Indian Point. This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. Fuel rod dimensions are bounding for each of the three types of rods found in the IP-1 fuel assembly.

<sup>††</sup> Calculations were conservatively performed for a fuel length of 150 inches.

Table 6.2.11  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15A (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 204 fuel rods, 21 guide tubes, pitch=0.550, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15A01	0.9204	0.9159	0.0009	0.418	0.3660	0.0260	0.3580	150	0.0165
Dimensions Listed for Authorized Contents				0.418 (min.)	0.3660 (max.)		0.3580 (max.)	150 (max.)	0.0165 (min.)

Table 6.2.12  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15B (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15B01	0.9369	0.9326	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B02	0.9338	0.9295	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B03	0.9362	0.9318	0.0008	0.422	0.3734	0.0243	0.3660	150	0.017
15x15B04	0.9370	0.9327	0.0008	0.422	0.3734	0.0243	0.3659	150	0.015
15x15B05	0.9356	0.9313	0.0008	0.422	0.3736	0.0242	0.3659	150	0.015
15x15B06	0.9366	0.9324	0.0007	0.420	0.3720	0.0240	0.3671	150	0.015
Dimensions Listed for Authorized Contents				0.420 (min.)	0.3736 (max.)		0.3671 (max.)	150 (max.)	0.015 (min.)
bounding dimensions (B15x15B01)	0.9388	0.9343	0.0009	0.420	0.3736	0.0232	0.3671	150	0.015

Table 6.2.13  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15C (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15C01	0.9255	0.9213	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0255
15x15C02	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0165
15x15C03	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3565	150	0.0165
15x15C04	0.9311	0.9268	0.0008	0.417	0.3570	0.0300	0.3565	150	0.0165
Dimensions Listed for Authorized Contents				0.417 (min.)	0.3640 (max.)		0.3570 (max.)	150 (max.)	0.0165 (min.)
bounding dimensions (B15x15C01)	0.9361	0.9316	0.0009	0.417	0.3640	0.0265	0.3570	150	0.0165

Table 6.2.14  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15D ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15D (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15D01	0.9341	0.9298	0.0008	0.430	0.3770	0.0265	0.3690	150	0.0160
15x15D02	0.9367	0.9324	0.0008	0.430	0.3770	0.0265	0.3686	150	0.0160
15x15D03	0.9354	0.9311	0.0008	0.430	0.3770	0.0265	0.3700	150	0.0155
15x15D04	0.9339	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150
Dimensions Listed for Authorized Contents				0.430 (min.)	0.3800 (max.)		0.3735 (max.)	150 (max.)	0.0150 (min.)
bounding dimensions (15x15D04)	0.9339 <sup>†</sup>	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150

<sup>†</sup> The  $k_{\text{eff}}$  value listed for the 15x15D02 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9367 (15x15D02) value is listed in Table 6.1.1 as the maximum.

Table 6.2.15  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15E ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15E (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15E01	0.9368	0.9325	0.0008	0.428	0.3790	0.0245	0.3707	150	0.0140
Dimensions Listed for Authorized Contents				0.428 (min.)	0.3790 (max.)		0.3707 (max.)	150 (max.)	0.0140 (min.)



Table 6.2.16  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15F ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15F (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15F01	0.9395 <sup>†</sup>	0.9350	0.0009	0.428	0.3820	0.0230	0.3742	150	0.0140
Dimensions Listed for Authorized Contents				0.428 (min.)	0.3820 (max.)		0.3742 (max.)	150 (max.)	0.0140 (min.)

<sup>†</sup> KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9383.

Table 6.2.17  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15G ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15G (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 204 fuel rods, 21 guide tubes, pitch=0.563, SS clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15G01	0.8876	0.8833	0.0008	0.422	0.3890	0.0165	0.3825	144	0.0145
Dimensions Listed for Authorized Contents				0.422 (min.)	0.3890 (max.)		0.3825 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.18  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15H (3.8% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9337	0.9292	0.0009	0.414	0.3700	0.0220	0.3622	150	0.0140
Dimensions Listed for Authorized Contents				0.414 (min.)	0.3700 (max.)		0.3622 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.19 (Sheet 1 of 3)  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

16x16A (4.6% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400
16x16A02	0.9263	0.9221	0.0007	0.382	0.3320	0.0250	0.3250	150	0.0400
16x16A03	0.9327	0.9282	0.0009	0.382	0.3350	0.0235	0.3255	150	0.0350
Dimensions Listed for Authorized Contents				0.382 (min.)	0.3350 (max.)		0.3255 (max.)	150 (max.)	0.0350 (min.)
bounding dimensions (16x16A03)	0.9327	0.9282	0.0009	0.382	0.3350	0.0235	0.3255	150	0.0350

Table 6.2.19 (Sheet 2 of 3)  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 16X16B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

16x16B (4.6% Enrichment, fixed neutron absorber $^{10}B$ minimum loading of 0.02 g/cm <sup>2</sup> ) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16B01	0.9370	0.9327	0.0008	0.374	0.329	0.0225	0.3225	150	0.04
Dimensions Listed for Authorized Contents				0.374 (min.)	0.329 (max.)		0.3225 (max.)	150 (max.)	0.04 (min.)

Table 6.2.19 (Sheet 3 of 3)  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 16X16C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

16x16C (4.1% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 235 fuel rods, 21 guide tubes, pitch=0.485, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16C	0.9090	0.9046	0.0008	0.374	0.329	0.0225	0.3225	150	0.0157
Dimensions Listed for Authorized Contents				0.374 (min.)	0.329 (max.)		0.3225 (max.)	150 (max.)	0.0157 (min.)

Table 6.2.20  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17A (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17A01	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016
17x17A02	0.9329	0.9286	0.0008	0.360	0.3100	0.0250	0.3030	150	0.016
Dimensions Listed for Authorized Contents				0.360 (min.)	0.3150 (max.)		0.3088 (max.)	150 (max.)	0.016 (min.)
bounding dimensions (17x17A01)	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016

Table 6.2.21  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17B (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9288	0.9243	0.0009	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B02	0.9290	0.9247	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B03	0.9243	0.9199	0.0008	0.376	0.3280	0.0240	0.3215	150	0.016
17x17B04	0.9324	0.9279	0.0009	0.372	0.3310	0.0205	0.3232	150	0.014
17x17B05	0.9266	0.9222	0.0008	0.374	0.3260	0.0240	0.3195	150	0.016
17x17B06	0.9311	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014
Dimensions Listed for Authorized Contents				0.372 (min.)	0.3310 (max.)		0.3232 (max.)	150 (max.)	0.014 (min.)
bounding dimensions (17x17B06)	0.9311 <sup>†</sup>	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014

<sup>†</sup> The  $k_{\text{eff}}$  value listed for the 17x17B04 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9324 (17x17B04) value is listed in Table 6.1.1 as the maximum.



Table 6.2.22  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17C (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 264 fuel rods, 25 guide tubes, pitch=0.502, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9293	0.9250	0.0008	0.379	0.3310	0.0240	0.3232	150	0.020
17x17C02	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020
Dimensions Listed for Authorized Contents				0.377 (min.)	0.3330 (max.)		0.3252 (max.)	150 (max.)	0.020 (min.)
bounding dimensions (17x17C02)	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020

Table 6.2.23  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

7x7B (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 49 fuel rods, 0 water rods, pitch=0.738, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630	0.4990	0.0320	0.4870	150	n/a	0.080
7x7B02	0.9301	0.9260	0.0007	0.5630	0.4890	0.0370	0.4770	150	n/a	0.102
7x7B03	0.9313	0.9271	0.0008	0.5630	0.4890	0.0370	0.4770	150	n/a	0.080
7x7B04	0.9311	0.9270	0.0007	0.5700	0.4990	0.0355	0.4880	150	n/a	0.080
7x7B05	0.9350	0.9306	0.0008	0.5630	0.4950	0.0340	0.4775	150	n/a	0.080
7x7B06	0.9298	0.9260	0.0006	0.5700	0.4990	0.0355	0.4910	150	n/a	0.080
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4910 (max.)	150 (max.)	n/a	0.120 (max.)
bounding dimensions (B7x7B01)	0.9375	0.9332	0.0008	0.5630	0.4990	0.0320	0.4910	150	n/a	0.102
bounding dimensions with 120 mil channel (B7x7B02)	0.9386	0.9344	0.0007	0.5630	0.4990	0.0320	0.4910	150	n/a	0.120

Table 6.2.24  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

8x8B (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 63 or 64 fuel rods <sup>†</sup> , 1 or 0 water rods <sup>†</sup> , pitch <sup>†</sup> = 0.636-0.642, Zr clad												
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	Fuel rods	Pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	63	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B02	0.9227	0.9185	0.0007	63	0.636	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B03	0.9299	0.9257	0.0008	63	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8B04	0.9236	0.9194	0.0008	64	0.642	0.5015	0.4295	0.0360	0.4195	150	n/a	0.100
Dimensions Listed for Authorized Contents				63 or 64	0.636-0.642	0.4840 (min.)	0.4295 (max.)		0.4195 (max.)	150 (max.)	0.034	0.120 (max.)
bounding (pitch=0.636) (B8x8B01)	0.9346	0.9301	0.0009	63	0.636	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.640) (B8x8B02)	0.9385	0.9343	0.0008	63	0.640	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.642) (B8x8B03)	0.9416	0.9375	0.0007	63	0.642	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120

<sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.25  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

8x8C (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 62 fuel rods, 2 water rods, pitch <sup>†</sup> = 0.636-0.641, Zr clad											
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8C02	0.9313	0.9268	0.0009	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.000
8x8C03	0.9329	0.9286	0.0008	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.800
8x8C04	0.9348 <sup>††</sup>	0.9307	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.100
8x8C05	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.120
8x8C06	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8C07	0.9314	0.9273	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.100
8x8C08	0.9339	0.9298	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.034	0.100
8x8C09	0.9301	0.9260	0.0007	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8C10	0.9317	0.9275	0.0008	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C11	0.9328	0.9287	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C12	0.9285	0.9242	0.0008	0.636	0.4830	0.4190	0.0320	0.4110	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.636-0.641	0.4830 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
Bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120

Table 6.2.26

- <sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch.  
<sup>††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9343.

MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68  
(all dimensions are in inches)

8x8D (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 60-61 fuel rods, 1-4 water rods <sup>†</sup> , pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830	0.4190	0.0320	0.4110	150	0.03/0.025	0.100
8x8D02	0.9325	0.9284	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8D03	0.9351	0.9309	0.0008	0.4830	0.4190	0.0320	0.4110	150	0.025	0.100
8x8D04	0.9338	0.9296	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.100
8x8D05	0.9339	0.9294	0.0009	0.4830	0.4190	0.0320	0.4100	150	0.040	0.100
8x8D06	0.9365	0.9324	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.120
8x8D07	0.9341	0.9297	0.0009	0.4830	0.4190	0.0320	0.4110	150	0.040	0.080
8x8D08	0.9376	0.9332	0.0009	0.4830	0.4230	0.0300	0.4140	150	0.030	0.080
Dimensions Listed for Authorized Contents				0.4830 (min.)	0.4230 (max.)		0.4140 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B8x8D01)	0.9403	0.9363	0.0007	0.4830	0.4230	0.0300	0.4140	150	0.000	0.120

<sup>†</sup> Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.

Table 6.2.27  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

8x8E (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
59 fuel rods, 5 water rods, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
Dimensions Listed for Authorized Contents				0.4930 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.034 (min.)	0.100 (max.)

Table 6.2.28  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

8x8F (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9411	0.9366	0.0009	0.4576	0.3996	0.0290	0.3913	150	0.0315	0.055
Dimensions Listed for Authorized Contents				0.4576 (min.)	0.3996 (max.)		0.3913 (max.)	150 (max.)	0.0315 (min.)	0.055 (max.)

Table 6.2.29  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

9x9A (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
74/66 fuel rods <sup>†</sup> , 2 water rods, pitch=0.566, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400	0.3840	0.0280	0.3760	150/90	0.030	0.100
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400	0.3840	0.0280	0.3760	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.4400 (min.)	0.3840 (max.)		0.3760 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.000	0.120

<sup>†</sup> This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.



Table 6.2.30  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

9x9B (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 to 0.572 <sup>†</sup> , Zr clad											
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9380	0.9336	0.0008	0.569	0.4330	0.3807	0.0262	0.3737	150	0.0285	0.100
9x9B02	0.9373	0.9329	0.0009	0.569	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
9x9B03	0.9417	0.9374	0.0008	0.572	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
Dimensions Listed for Authorized Contents				0.572	0.4330 (min.)	0.3810 (max.)		0.3740 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B9x9B01)	0.9436	0.9394	0.0008	0.572	0.4330	0.3810	0.0260	0.3740 <sup>††</sup>	150	0.000	0.120

<sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch.

<sup>††</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.31  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

9x9C (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
80 fuel rods, 1 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230	0.3640	0.0295	0.3565	150	0.020	0.100
Dimensions Listed for Authorized Contents				0.4230 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.020 (min.)	0.100 (max.)

Table 6.2.32  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

9x9D (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 79 fuel rods, 2 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.9394	0.9350	0.0009	0.4240	0.3640	0.0300	0.3565	150	0.0300	0.100
Dimensions Listed for Authorized Contents				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0300 (min.)	0.100 (max.)

Table 6.2.33  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

9x9E (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.9334	0.9293	0.0007	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
9x9E02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents <sup>†</sup>				0.4170 (min.)	0.3640 (max.)		0.3530 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9E02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

<sup>†</sup> This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.34  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

9x9F (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.9307	0.9265	0.0007	0.4430	0.3860	0.0285	0.3745	150	0.0120	0.120
9x9F02	0.9401	0.9359	0.0008	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
				0.4430	0.3860	0.0285	0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents <sup>†</sup>				0.4430 (min.)	0.3860 (max.)		0.3745 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9F02)	0.9401	0.9359	0.0008	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
				0.4430	0.3860	0.0285	0.3745			

<sup>†</sup> This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.35  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9G ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

9x9G (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9G01	0.9309	0.9265	0.0008	0.4240	0.3640	0.0300	0.3565	150	0.0320	0.120
Dimensions Listed for Authorized Contents				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0320 (min.)	0.120 (max.)

Table 6.2.36  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10A (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
92/78 fuel rods <sup>†</sup> , 2 water rods, pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040	0.3520	0.0260	0.3450	155/90	0.030	0.100
Dimensions Listed for Authorized Contents				0.4040 (min.)	0.3520 (max.)		0.3455 (max.)	150 <sup>††</sup> (max.)	0.030 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457 <sup>†††</sup>	0.9414	0.0008	0.4040	0.3520	0.0260	0.3455 <sup>‡</sup>	155	0.030	0.120

<sup>†</sup> This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

<sup>††</sup> Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for the authorized contents limits the active fuel length to 150 inches. This is due to the fact that the fixed neutron absorber panels are 156 inches in length.

<sup>†††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9453.

<sup>‡</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.37  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68  
 (all dimensions are in inches)

10x10B (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 91/83 fuel rods <sup>†</sup> , 1 water rods (square, replacing 9 fuel rods), pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957	0.3480	0.0239	0.3413	155/90	0.0285	0.100
Dimensions Listed for Authorized Contents				0.3957 (min.)	0.3480 (max.)		0.3420 (max.)	150 <sup>††</sup> (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957	0.3480	0.0239	0.3420 <sup>†††</sup>	155	0.000	0.120

<sup>†</sup> This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

<sup>††</sup> Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for the authorized contents limits the active fuel length to 150 inches. This is due to the fact that the fixed neutron absorber panels are 156 inches in length.

<sup>†††</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.



Table 6.2.38  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10C (4.2% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.9433	0.9392	0.0007	0.3780	0.3294	0.0243	0.3224	150	0.031	0.055
Dimensions Listed for Authorized Contents				0.3780 (min.)	0.3294 (max.)		0.3224 (max.)	150 (max.)	0.031 (min.)	0.055 (max.)

Table 6.2.39  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10D (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 100 fuel rods, 0 water rods, pitch=0.565, SS clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960	0.3560	0.0200	0.350	83	n/a	0.080
Dimensions Listed for Authorized Contents				0.3960 (min.)	0.3560 (max.)		0.350 (max.)	83 (max.)	n/a	0.080 (max.)

Table 6.2.40  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10E (4.0% Enrichment, fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
96 fuel rods, 4 water rods, pitch=0.557, SS clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940	0.3500	0.0220	0.3430	83	0.022	0.080
Dimensions Listed for Authorized Contents				0.3940 (min.)	0.3500 (max.)		0.3430 (max.)	83 (max.)	0.022 (min.)	0.080 (max.)

Table 6.2.41  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F and MPC-68  
 (all dimensions are in inches)

6x6A (3.0% Enrichment <sup>†</sup> , fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> ) 35 or 36 fuel rods <sup>††</sup> , 1 or 0 water rods <sup>††</sup> , pitch <sup>††</sup> =0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4940	110	n/a	0.060
6x6A02	0.7517	0.7476	0.0007	0.694	36	0.5645	0.4925	0.0360	0.4820	110	n/a	0.060
6x6A03	0.7545	0.7501	0.0008	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6A04	0.7537	0.7494	0.0008	0.694	36	0.5550	0.4850	0.0350	0.4820	110	n/a	0.060
6x6A05	0.7555	0.7512	0.0008	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6A06	0.7618	0.7576	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6A07	0.7588	0.7550	0.0005	0.700	36	0.5555	0.4850	0.03525	0.4780	110	n/a	0.060
6x6A08	0.7808	0.7766	0.0007	0.710	36	0.5625	0.5105	0.0260	0.4980	110	n/a	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5550 (min.)	0.5105 (max.)	0.02225	0.4980 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6A01)	0.7727	0.7685	0.0007	0.694	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A02)	0.7782	0.7738	0.0008	0.700	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A03)	0.7888	0.7846	0.0007	0.710	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060

Table 6.2.42

- <sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.
- <sup>††</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

MAXIMUM  $K_{EFF}$  VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F and MPC-68  
(all dimensions are in inches)

6x6B (3.0% Enrichment <sup>†</sup> , fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> ) 35 or 36 fuel rods <sup>††</sup> (up to 9 MOX rods), 1 or 0 water rods <sup>††</sup> , pitch <sup>††</sup> =0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	0.7604	0.7563	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6B02	0.7618	0.7577	0.0007	0.694	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B03	0.7619	0.7578	0.0007	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B04	0.7686	0.7644	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6B05	0.7824	0.7785	0.0006	0.710	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5625 (min.)	0.4945 (max.)		0.4820 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6B01)	0.7822 <sup>†††</sup>	0.7783	0.0006	0.710	35	0.5625	0.4945	0.0340	0.4820	120	0.0	0.060

Note:

1. These assemblies contain up to 9 MOX pins. The composition of the MOX fuel pins is given in Table 6.3.4.

<sup>†</sup> The <sup>235</sup>U enrichment of the MOX and UO<sub>2</sub> pins is assumed to be 0.711% and 3.0%, respectively.

<sup>††</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

<sup>†††</sup> The  $k_{eff}$  value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1 $\sigma$ ). Therefore, the 0.7824 value is listed in Tables 6.1.7 and 6.1.8 as the maximum.

Table 6.2.43  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F and MPC-68

(all dimensions are in inches)

6x6C (3.0% Enrichment <sup>†</sup> , fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )										
36 fuel rods, 0 water rods, pitch=0.740, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630	0.4990	0.0320	0.4880	77.5	n/a	0.060
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4880 (max.)	77.5 (max.)	n/a	0.060 (max.)

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

Table 6.2.44  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F and MPC-68

(all dimensions are in inches)

7x7A (3.0% Enrichment <sup>†</sup> , fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )										
49 fuel rods, 0 water rods, pitch=0.631, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.7974	0.7932	0.0008	0.4860	0.4204	0.0328	0.4110	80	n/a	0.060
Dimensions Listed for Authorized Contents				0.4860 (min.)	0.4204 (max.)		0.4110 (max.)	80 (max.)	n/a	0.060 (max.)

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

Table 6.2.45  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F and MPC-68

(all dimensions are in inches)

8x8A (3.0% Enrichment <sup>†</sup> , fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )											
63 or 64 fuel rods <sup>††</sup> , 0 water rods, pitch=0.523, Zr clad											
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	64	0.4120	0.3620	0.0250	0.3580	110	n/a	0.100
8x8A02	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100
Dimensions Listed for Authorized Contents				63	0.4120 (min.)	0.3620 (max.)		0.3580 (max.)	120 (max.)	n/a	0.100 (max.)
bounding dimensions (8x8A02)	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

<sup>††</sup> This assembly class was analyzed and qualified for a variation in the number of fuel rods.



Table 6.2.46

## SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	4.81''
Canister Wall Thickness	0.11''
Separator Assembly Plates Thickness	0.11''
Cladding OD	0.412''
Cladding ID	0.362''
Pellet OD	0.358''
Active Length	110.5''
Fuel Composition	1.8% UO <sub>2</sub> and 98.2% ThO <sub>2</sub> or 1.5% UO <sub>2</sub> and 98.5% ThO <sub>2</sub>
Initial Enrichment	93.5 wt% <sup>235</sup> U for 1.8% of the fuel
Maximum $k_{eff}^{\dagger}$	0.1813
Calculated $k_{eff}$	0.1779
Standard Deviation	0.0004

<sup>†</sup> The maximum calculated  $k_{eff}$  of 0.1813 assumes an average ThO<sub>2</sub> content of 98.2 wt%. It is also based on a UO<sub>2</sub> content of 1.8 wt%. Reducing the UO<sub>2</sub> content from 1.8 wt% to the average value of 1.5 wt% would result in a reduction of the already low reactivity, due to the reduction in the fissile material. Therefore the values listed in the table are bounding.

## 6.3 MODEL SPECIFICATION

### 6.3.1 Description of Calculational Model

Figures 6.3.1, 6.3.1.a, 6.3.2 and 6.3.3 show representative horizontal cross sections of the four types of cells used in the calculations, and Figures 6.3.4 through 6.3.6 illustrate the basket configurations used. Four different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket
- an optimized 24 PWR assembly basket (24E)
- a 32 PWR assembly basket
- a 68 BWR assembly basket.

For all four basket designs, the same techniques and the same level of detail are used in the calculational models.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7. Although the fixed neutron absorber panels are 156 inches in length, which is much longer than the active fuel length (maximum of 150 inches), they are assumed equal to or less than the active fuel length in the calculations. As shown on the Drawings in Section 1.5, 16 of the 24 periphery fixed neutron absorber panels on the MPC-24 and MPC-24E have reduced width (i.e., 6.25 inches wide as opposed to 7.5 inches). However, the calculational models for these baskets conservatively assume all of the periphery fixed neutron absorber panels are 6.25 inches in width. Note that Figures 6.3.1 through 6.3.3 show Boral as the fixed neutron absorber. The effect of using Metamic as fixed neutron absorber is discussed in Subsection 6.4.11.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design conditions important to criticality safety (see Subsection 6.4.2.5), and thus from a criticality standpoint, the normal, off-normal, and accident conditions are identical and do not require individual models.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and fixed neutron absorber panels on the stainless steel walls of the storage cells. Under the conditions of storage, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ( $k_{\text{eff}} < 0.52$ ). For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel rod pellet-to-clad gaps. Appendix 6.D provides sample input files for two of the MPC basket designs (MPC-68 and

MPC-24) in the HI-STORM 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included. Furthermore, the water above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

As indicated in Figures 6.3.1 through 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO-3 and MCNP4a were used to determine the direction of the manufacturing tolerances, which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity.

CASMO-3 was used for one of each of the two principal basket designs, i.e. for the flux trap design MPC-24 and for the non-fluxtrap design MPC-68. The effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. Generally, the conclusions in Table 6.3.1 are directly applicable to the MPC-24E and the MPC-32. Exceptions are the conclusions for the water temperature and void percentage, which are not directly applicable to the MPC-32 due to the presence of high soluble boron concentrations in this canister. This condition is addressed in Section 6.4.2.1 where the optimum moderation is determined for the MPC-32.

Additionally, MCNP4a calculations are performed to evaluate the tolerances of the various basket dimensions of the MPC-68, MPC-24 and MPC-32 in further detail. The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a. Calculated  $k_{\text{eff}}$  results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR assemblies. Each of the basket dimensions are evaluated for their minimum, nominal and maximum values from the Drawings of section 1.5. For PWR MPC designs, the reactivity effect of tolerances with soluble boron present in the water is additionally determined. Due to the close similarity between the MPC-24 and MPC-24E, the basket dimensions are only evaluated for the MPC-24, and the same dimensional assumptions are applied to both MPC designs.

Based on the MCNP4a and CASMO-3 calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined. Because the reactivity effect (positive or negative) of the

manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

As demonstrated in this section, design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, the fixed neutron absorbing panels and the soluble boron concentration in the water during loading/unloading operations. As shown in Chapter 11, none of these parameters are affected during any of the design basis off-normal or accident conditions involving handling, packaging, transfer or storage.

The MPC-32 criticality model uses a sheathing thickness of 0.075 inches, whereas the actual MPC-32 design uses a sheathing thickness of 0.035 inches. For the minimum cell pitch of 9.158 inches, the thicker sheathing results in a slightly smaller cell ID of 8.69 inches (minimum), compared to 8.73 inches (minimum) for the thinner sheathing. To demonstrate that the dimensions used in the criticality model are acceptable and conservative, calculations were performed for both sheathing thicknesses and the results are compared in Table 6.3.5. To bound various soluble boron levels, two comparisons were performed. The first comparison uses the bounding case for the MPC-32 (see Table 6.1.6), which is for assembly class 15x15F at 5 wt%  $^{235}\text{U}$  and a soluble boron level of 2600 ppm. To bound lower soluble boron levels, the second comparison uses the same assembly class (15x15F), 0 ppm soluble boron (i.e. pure water), and an arbitrary enrichment of 1.7 wt%  $^{235}\text{U}$ . In both comparisons, the results of the 0.075 inch sheathing are slightly higher, i.e. more conservative, than the results for 0.035 inch sheathing, although the differences are within the statistical uncertainties. Using a sheathing thickness of 0.075 inches in the criticality models of the MPC-32 is therefore acceptable, and potentially more conservative, than using the actual value of 0.035 inches. This validates the choice of the dimensional assumptions for the MPC-32 shown in Table 6.3.3, which are used for all further MPC-32 criticality calculations, unless otherwise noted.

### 6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM 100 System are listed in Table 6.3.4.

The HI-STORM 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of fixed neutron absorber are provided in Section 1.2.1.3.1.

The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Section 9.1.5.3, to validate the  $^{10}\text{B}$  (poison) concentration in the fixed neutron absorber. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an MCNP4a calculation of the number of neutrons absorbed in the  $^{10}\text{B}$  was performed. The calculation conservatively assumed a constant neutron

source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of  $^{10}\text{B}$  atoms destroyed is only  $2.6\text{E}-09$  in 50 years. Thus, the reduction in  $^{10}\text{B}$  concentration in the fixed neutron absorber by neutron absorption is negligible. In addition, the results presented in Subsection 3.4.4.3.1.8 demonstrate that the sheathing, which affixes the fixed neutron absorber panel, remains in place during all credible accident conditions, and thus, the fixed neutron absorber panel remains permanently fixed. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

### 6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 1 of this FSAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel positioning was judged to be not significant. Starting in Revision 2 of this FSAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for all new or changed conditions. The calculations in this subsection serve to determine for which of these conditions the eccentric positioning of assemblies in the fuel storage locations results in a higher maximum  $k_{\text{eff}}$  value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum  $k_{\text{eff}}$  value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4. All other calculations throughout this chapter, such as studies to determine bounding fuel dimensions, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- Cell Center Configuration: All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;
- Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and
- Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such

configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be  $(1/5)^{24}$  or approximately  $10^{-17}$  for the MPC-24,  $(1/5)^{32}$  or approximately  $10^{-23}$  for the MPC-32, and  $(1/5)^{68}$  or approximately  $10^{-48}$  for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.6, results are presented for all conditions that were introduced in Revision 2 of this FSAR, namely results for the MPC-24E with intact and damaged fuel at 5 wt%  $^{235}\text{U}$ , for the MPC-32 with soluble boron levels lower than 2600 ppm for 5 wt%  $^{235}\text{U}$  and lower than 1900 ppm for 4.1 wt%  $^{235}\text{U}$ , and for the MPC-32 with intact and damaged fuel. The table shows the maximum  $k_{\text{eff}}$  value for centered and the two eccentric configurations for each condition, and the difference in  $k_{\text{eff}}$  between the centered and eccentric positioning. The results and conclusions are summarized as follows:

- In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.
- For the MPC-24E, moving the assemblies and DFCs towards the center of the basket also results in a minor reduction. The cell centered configuration is therefore bounding for this condition and is used in the design basis calculations reported in Section 6.1 and Section 6.4.
- For the MPC-32 cases listed in Table 6.3.6, the maximum reactivity is shown for the basket center configuration. However, for some of the cases with intact and damaged fuel in the MPC-32, the cell centered configuration results in a higher maximum reactivity. Therefore, both the cell centered and basket centered configuration are analyzed for the MPC-32 design basis calculation, and the higher results are listed in the tables in Section 6.1. and 6.4. This applies to the cases with intact and damaged fuel, and to cases with intact fuel only and soluble boron levels lower than 2600 ppm for 5 wt%  $^{235}\text{U}$  and lower than 1900 ppm for 4.1 wt%  $^{235}\text{U}$ .

Table 6.3.1

## CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter <sup>†</sup>	$\Delta k$ for Maximum Tolerance		Action/Modeling Assumption
	MPC-24 <sup>‡</sup>	MPC-68	
Reduce Fixed Neutron Absorber Width to Minimum	N/A <sup>†††</sup> min.= nom.= 7.5" and 6.25"	N/A <sup>†††</sup> min. = nom. = 4.75"	Assume minimum fixed neutron absorber width
Increase UO <sub>2</sub> Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO <sub>2</sub> density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min.= 8.86" nom. = 8.92"	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" nom. = 8.92"	-0.0030 max. = 6.113" nom. = 6.053"	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" nom. = 1.15"	N/A	Assume minimum water gap in the MPC-24

<sup>†</sup> Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

<sup>‡</sup> Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

<sup>†††</sup> The fixed neutron absorber width for the MPC-68 is 4.75" +0.125", -0" , the fixed neutron absorber widths for the MPC-24 are 7.5" +0.125", -0" and 6.25" +0.125" -0" (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

## CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	$\Delta k$ Maximum Tolerance		Action/Modeling Assumption
	MPC-24 <sup>‡</sup>	MPC-68	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0030	-0.0039	
70°C	-0.0089	-0.0136	
100°C	-0.0162	-0.0193	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0251	-0.0241	
100°C	-0.0412	-0.0432	
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

<sup>‡</sup> Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].



Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES<sup>†</sup>

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated $k_{eff}$
MPC-24 <sup>††</sup> (17x17A01 @ 4.0% Enrichment)						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9325±0.0008 <sup>†††</sup>
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9300±0.0008
nominal	(10.906")	nom. - 0.04"	(8.88")	nom. + 0.05"	(0.3625")	0.9305±0.0007
MPC-68 (8x8C04 @ 4.2% Enrichment)						
minimum	(6.43")	minimum	(5.993")	nominal	(1/4")	0.9307±0.0007
nominal	(6.49")	nominal	(6.053")	nominal	(1/4")	0.9274±0.0007
maximum	(6.55")	maximum	(6.113")	nominal	(1/4")	0.9272±0.0008
nom. + 0.05"	(6.54")	nominal	(6.053")	nom. + 0.05"	(0.30")	0.9267±0.0007

Notes:

1. Values in parentheses are the actual value used.

<sup>†</sup> Tolerance for pitch and box I.D. are  $\pm 0.06"$ .  
Tolerance for box wall thickness is  $+0.05"$ ,  $-0.00"$ .

<sup>††</sup> All calculations for the MPC-24 assume minimum water gap thickness (1.09").

<sup>†††</sup> Numbers are  $1\sigma$  statistical uncertainties.

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES<sup>†</sup>

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated $k_{\text{eff}}$
MPC-24 (17x17A @ 5.0% Enrichment) 400ppm soluble boron						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9236±0.0007 <sup>††</sup>
maximum	(10.966")	maximum	(8.98")	nominal	(5/16")	0.9176±0.0008
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9227±0.0010
minimum	(10.846")	minimum	(8.86")	nominal	(5/16")	0.9159±0.0008
nominal	(10.906")	nominal-0.04"	(8.88")	nom.+0.05"	(0.3625")	0.9232±0.0009
nominal	(10.906")	nominal	(8.92")	nominal	(5/16")	0.9158±0.0007
MPC-32 (17x17A @ 5.0% Enrichment) 2200 ppm soluble boron <sup>†††</sup>						
minimum	(9.158")	minimum	(8.69")	nominal	(9/32")	0.9399±0.0007
nominal	(9.218")	nominal	(8.75")	nominal	(9/32")	0.9370±0.0007
maximum	(9.278")	maximum	(8.81")	nominal	(9/32")	0.9313±0.0008
nominal+0.05"	(9.268")	nominal	(8.75")	nominal+0.05"	(0.331")	0.9356±0.0007
minimum+0.05"	(9.208")	minimum	(8.69")	nominal+0.05"	(0.331")	0.9395±0.0008
maximum	(9.278")	Maximum-0.05"	(8.76")	nominal+0.05"	(0.331")	0.9330±0.0008

Notes:

1. Values in parentheses are the actual value used.

<sup>†</sup> Tolerance for pitch and box I.D. are ± 0.06".  
Tolerance for box wall thickness is +0.05", -0.00".

<sup>††</sup> Numbers are 1 $\sigma$  statistical uncertainties.

<sup>†††</sup> for 0.075" sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.

Table 6.3.3

## BASKET DIMENSIONAL ASSUMPTIONS

<b>Basket Type</b>	<b>Pitch</b>	<b>Box I.D.</b>	<b>Box Wall Thickness</b>	<b>Water-Gap Flux Trap</b>
MPC-24	nominal (10.906")	maximum (8.98")	nominal (5/16")	minimum (1.09")
MPC-24E	nominal (10.847")	maximum (8.81", 9.11" for DFC Positions)	nominal (5/16")	minimum (1.076", 0.776" for DFC Positions)
MPC-32	Minimum (9.158")	Minimum <sup>†</sup> (8.69")	Nominal (9/32")	N/A
MPC-68	minimum (6.43")	Minimum (5.993")	nominal (1/4")	N/A

<sup>†</sup> for 0.075" sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.

Table 6.3.4

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>MPC-24, MPC-24E and MPC-32</b>		
<b>UO<sub>2</sub> 5.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.696E-02	1.185E-01
92235	1.188E-03	4.408E-02
92238	2.229E-02	8.374E-01
<b>UO<sub>2</sub> 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
<b>BORAL (0.02 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
<b>BORAL (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24E and MPC-32)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01

Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>METAMIC (0.02 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.648 (MPC-24)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	6.314E-03	3.965E-02
5011	2.542E-02	1.755E-01
6012	7.932E-02	5.975E-02
13027	4.286E-02	7.251E-01
<b>METAMIC (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.646 (MPC-24E and MPC-32)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01

Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>BORATED WATER, 300 PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wt. Fraction</b>
5010	3.248E-06	5.400E-05
5011	1.346E-05	2.460E-04
1001	6.684E-02	1.1186E-01
8016	3.342E-02	8.8784E-01
<b>BORATED WATER, 400PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	4.330E-06	7.200E-05
5011	1.794E-05	3.280E-04
1001	6.683E-02	1.1185E-01
8016	3.341E-02	8.8775E-01
<b>BORATED WATER, 1900PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	2.057E-05	3.420E-04
5011	8.522E-05	1.558E-03
1001	6.673E-02	1.1169E-01
8016	3.336E-02	8.8641E-01
<b>BORATED WATER, 2600PPM, DENSITY (g/cc)=0.93</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	2.618e-05	4.680E-04
5011	1.085e-04	2.132E-03
1001	6.201e-02	1.1161E-01
8016	3.101e-02	8.8579E-01

Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>MPC-68</b>		
<b>UO<sub>2</sub> 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
<b>UO<sub>2</sub> 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
<b>MOX FUEL<sup>†</sup>, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

<sup>†</sup> The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>BORAL (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
<b>METAMIC (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.646</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01
<b>FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.798E-02	1.212E-01
92235	4.001E-04	1.484E-02
92238	2.742E-05	1.030E-03
90232	2.357E-02	8.630E-01
<b>COMMON MATERIALS</b>		
<b>ZR CLAD, DENSITY (g/cc) = 6.550</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
40000	4.323E-02	1.000E+00
<b>MODERATOR (H<sub>2</sub>O), DENSITY (g/cc) = 1.000</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01



Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>STAINLESS STEEL, DENSITY (g/cc) = 7.840</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
<b>ALUMINUM, DENSITY (g/cc) = 2.700</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
13027	6.026E-02	1.000E+00

Table 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>CONCRETE, DENSITY (g/cc) = 2.35</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
1001	8.806E-03	6.000E-03
8016	4.623E-02	5.000E-01
11000	1.094E-03	1.700E-02
13027	2.629E-04	4.800E-03
14000	1.659E-02	3.150E-01
19000	7.184E-04	1.900E-02
20000	3.063E-03	8.300E-02
26000	3.176E-04	1.200E-02
<b>LEAD, DENSITY (g/cc) = 11.34</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
82000	3.296E-02	1.0
<b>HOLTITE-A, DENSITY (g/cc) = 1.61</b>		
1001	5.695E-02	5.920E-02
5010	1.365E-04	1.410E-03
5011	5.654E-04	6.420E-03
6012	2.233E-02	2.766E-01
7014	1.370E-03	1.980E-02
8016	2.568E-02	4.237E-01
13027	7.648E-03	2.129E-01

Table 6.3.5

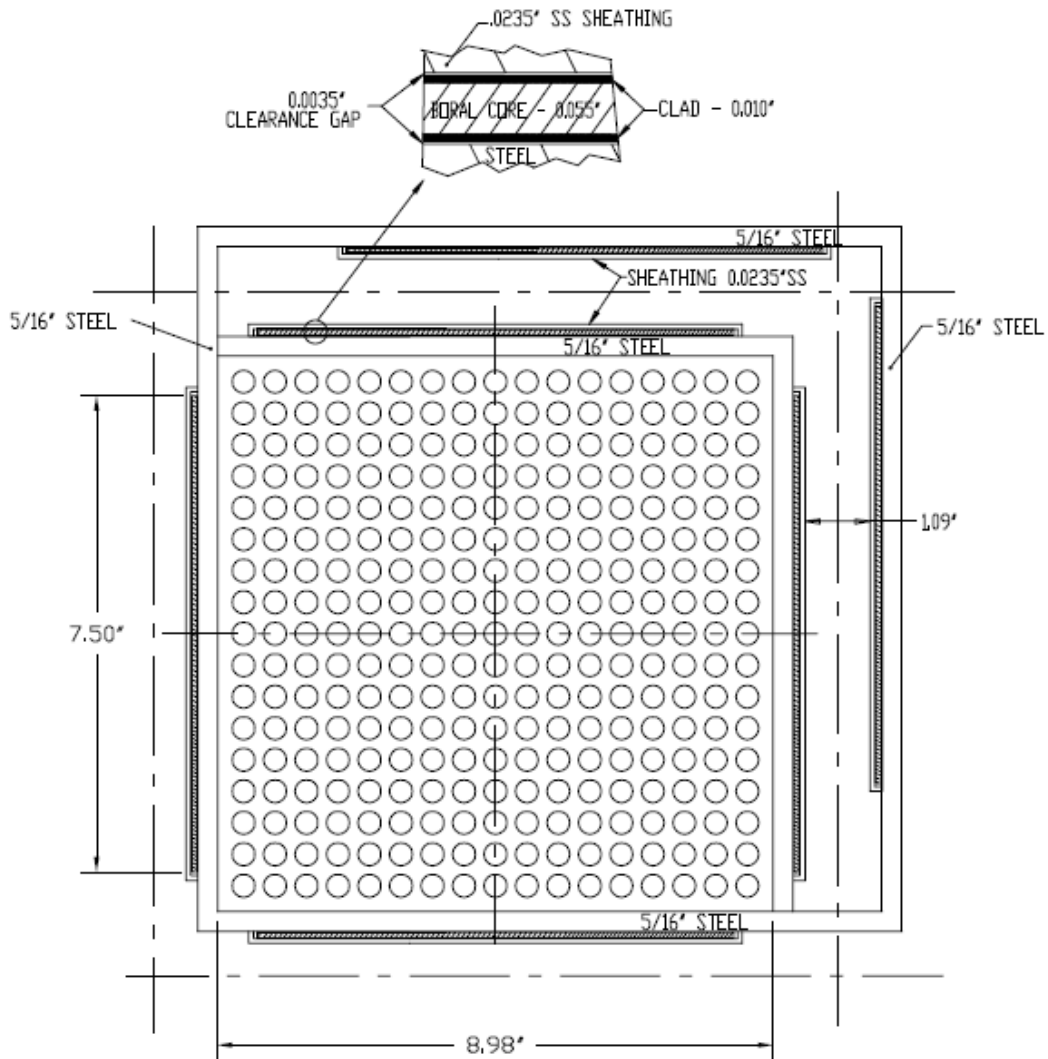
## REACTIVITY EFFECT OF SHEATHING THICKNESS FOR THE MPC-32

Assembly Class	Enrichment (wt% <sup>235</sup> U)	Soluble Boron (ppm)	Maximum k <sub>eff</sub>		Difference in Maximum k <sub>eff</sub>
			Sheathing 0.075” Min. Cell ID 8.69”	Sheathing 0.035” Min. Cell ID 8.73”	
15x15F	5.0	2600	0.9483	0.9476	-0.0008
15x15F	1.7	0	0.8914	0.8909	-0.0005

Table 6.3.6

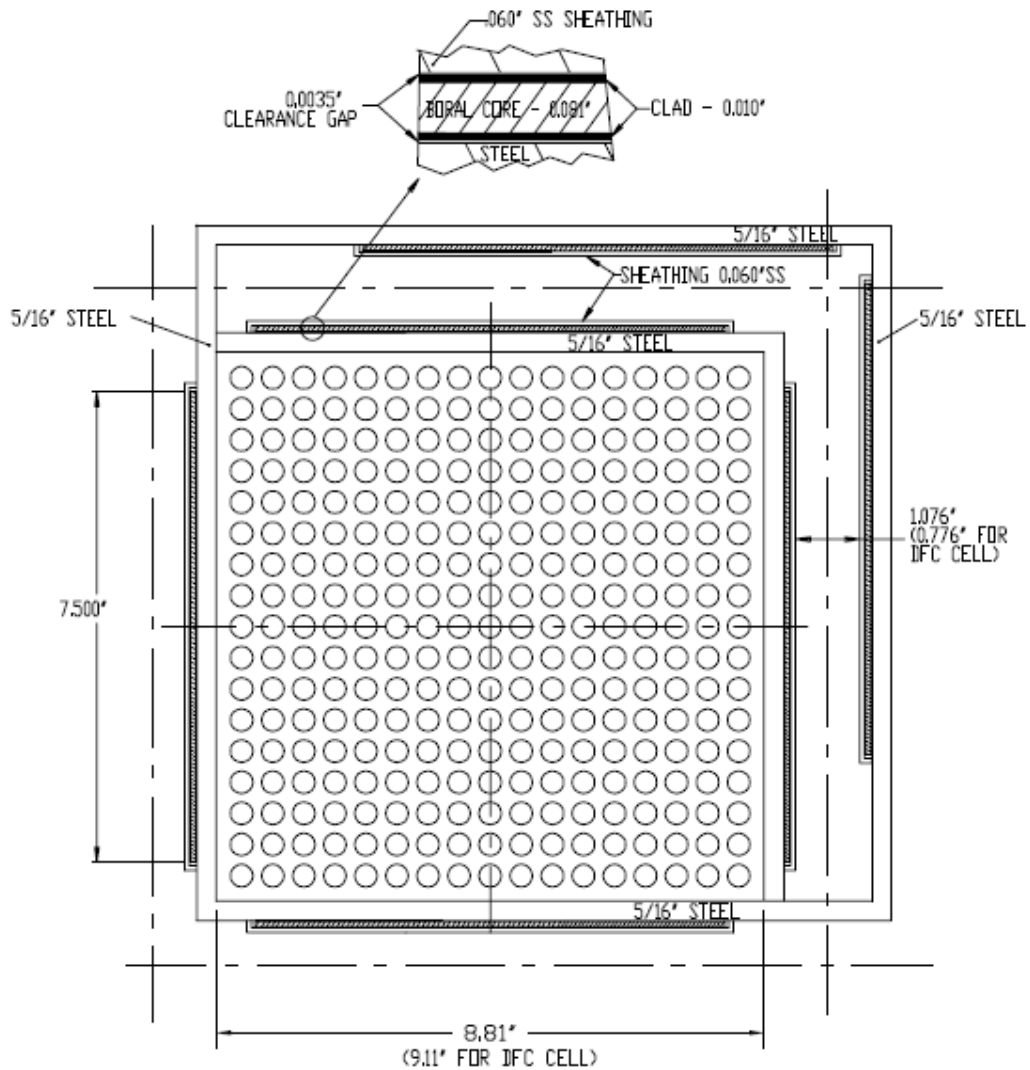
**REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT  
(FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS**

<b>CASE</b>	<b>Contents centered (Reference)</b>	<b>Content moved towards center of basket</b>		<b>Content moved towards basket periphery</b>	
	<b>Maximum <math>k_{eff}</math></b>	<b>Maximum <math>k_{eff}</math></b>	<b><math>k_{eff}</math> Difference to Reference</b>	<b>Maximum <math>k_{eff}</math></b>	<b><math>k_{eff}</math> Difference to Reference</b>
MPC-24E, Intact Fuel and Damaged Fuel/Fuel Debris, 5% Enrichment, 600ppm Soluble Boron	0.9185	0.9178	-0.0007	0.9132	-0.0053
MPC-32, Intact Fuel, Assembly Class 16x16A, 4.1% Enrichment, 1400ppm Soluble Boron	0.9332	0.9367	0.0035	0.8992	-0.0340
MPC-32, Intact Fuel, Assembly Class 15x15B, 5.0% Enrichment, 2400ppm Soluble Boron	0.9473	0.9493	0.0020	0.9306	-0.0167
MPC-32, Intact Fuel and Damaged Fuel/Fuel Debris, Assembly Class 15x15F (Intact), 5% Enrichment, 2900ppm Soluble Boron	0.9378	0.9397	0.0019	0.9277	-0.0101



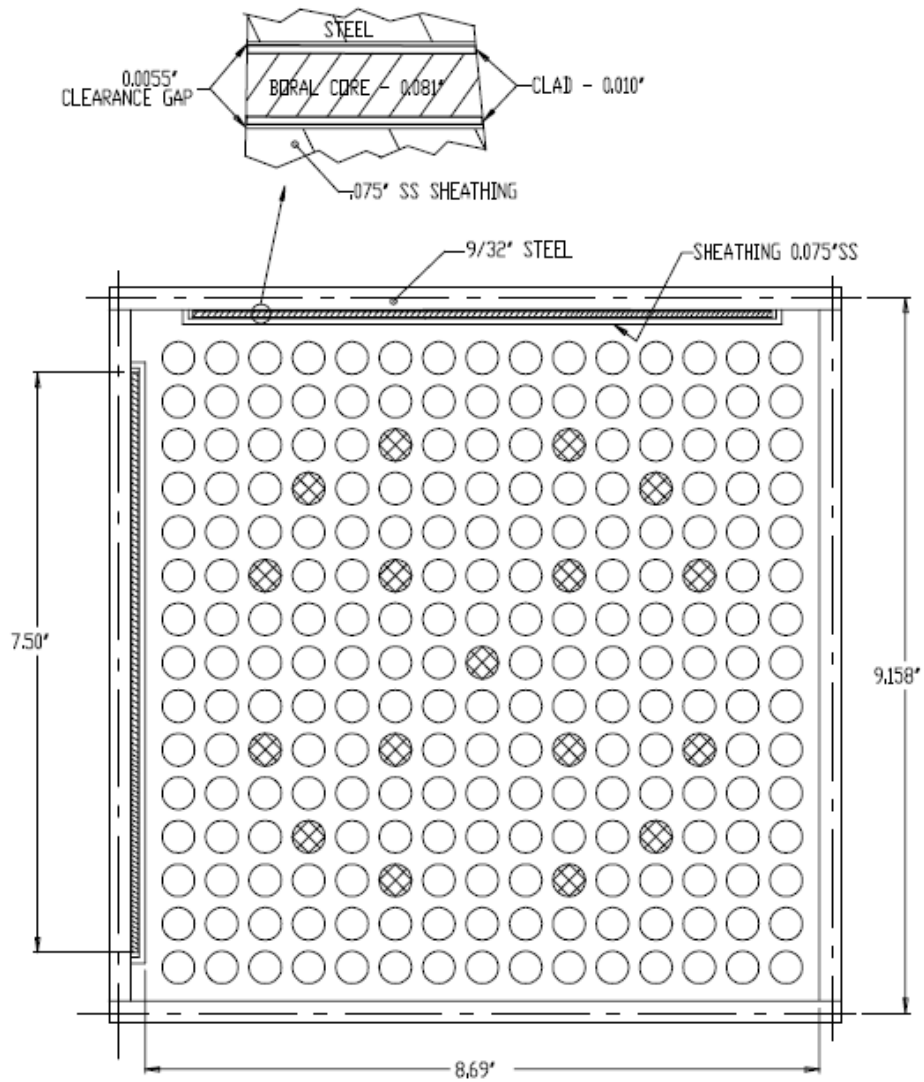
**FIGURE 6.3.1: TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-24 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



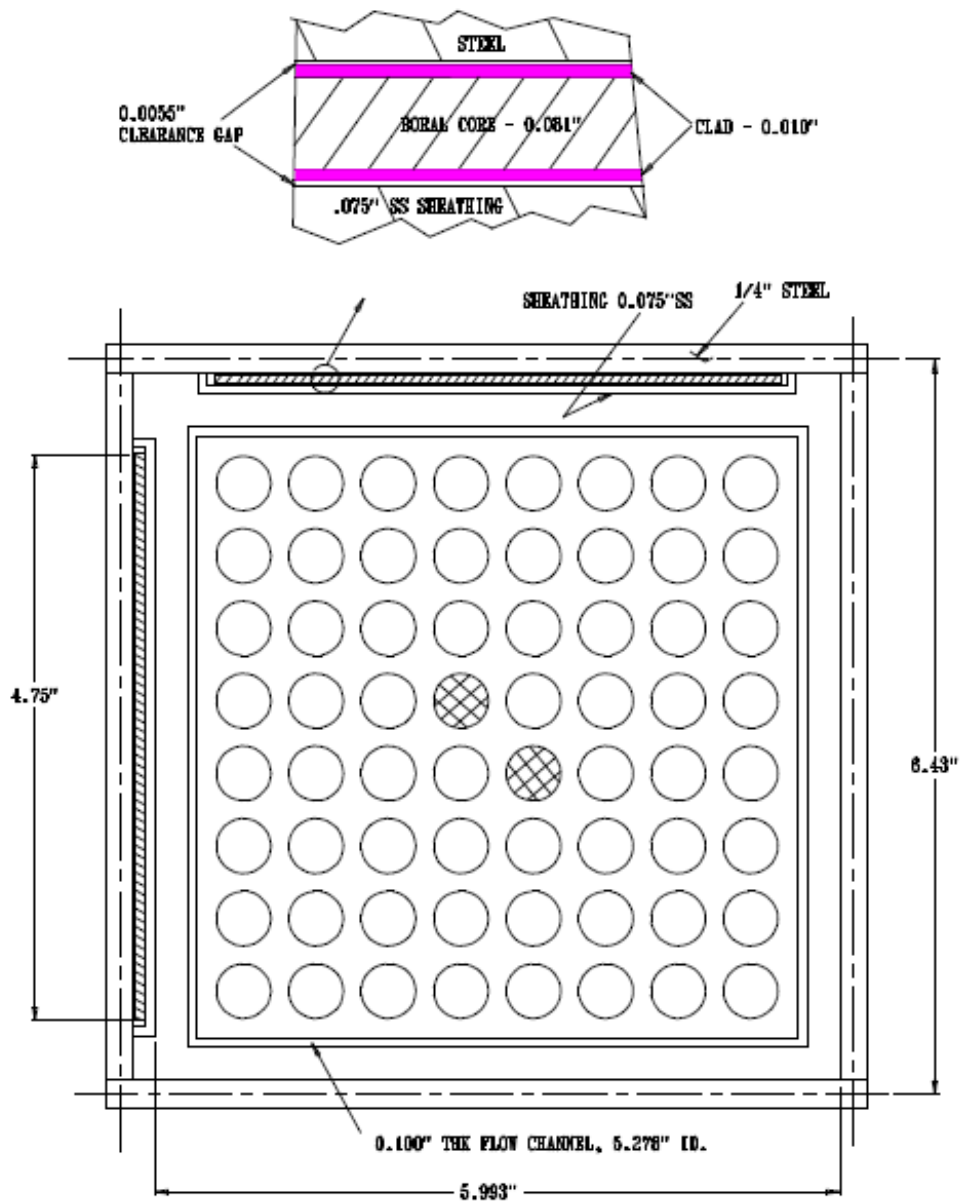
**FIGURE 6.3.1A: TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-24E BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



**FIGURE 6.3.2: TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-32 BASKET (SEE CHAPTER 1 FOR THE TRUE BASKET DIMENSIONS)**

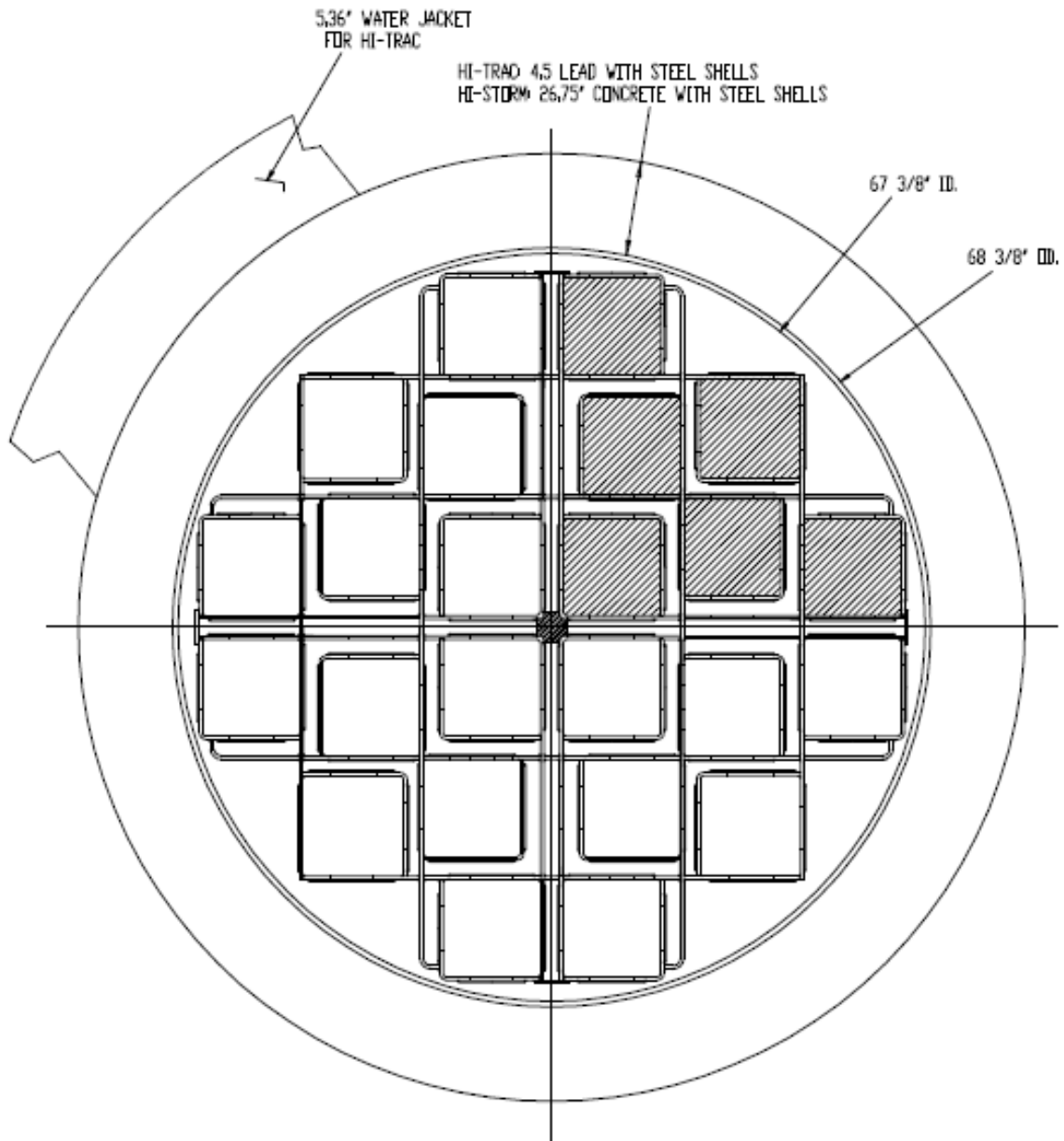
**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



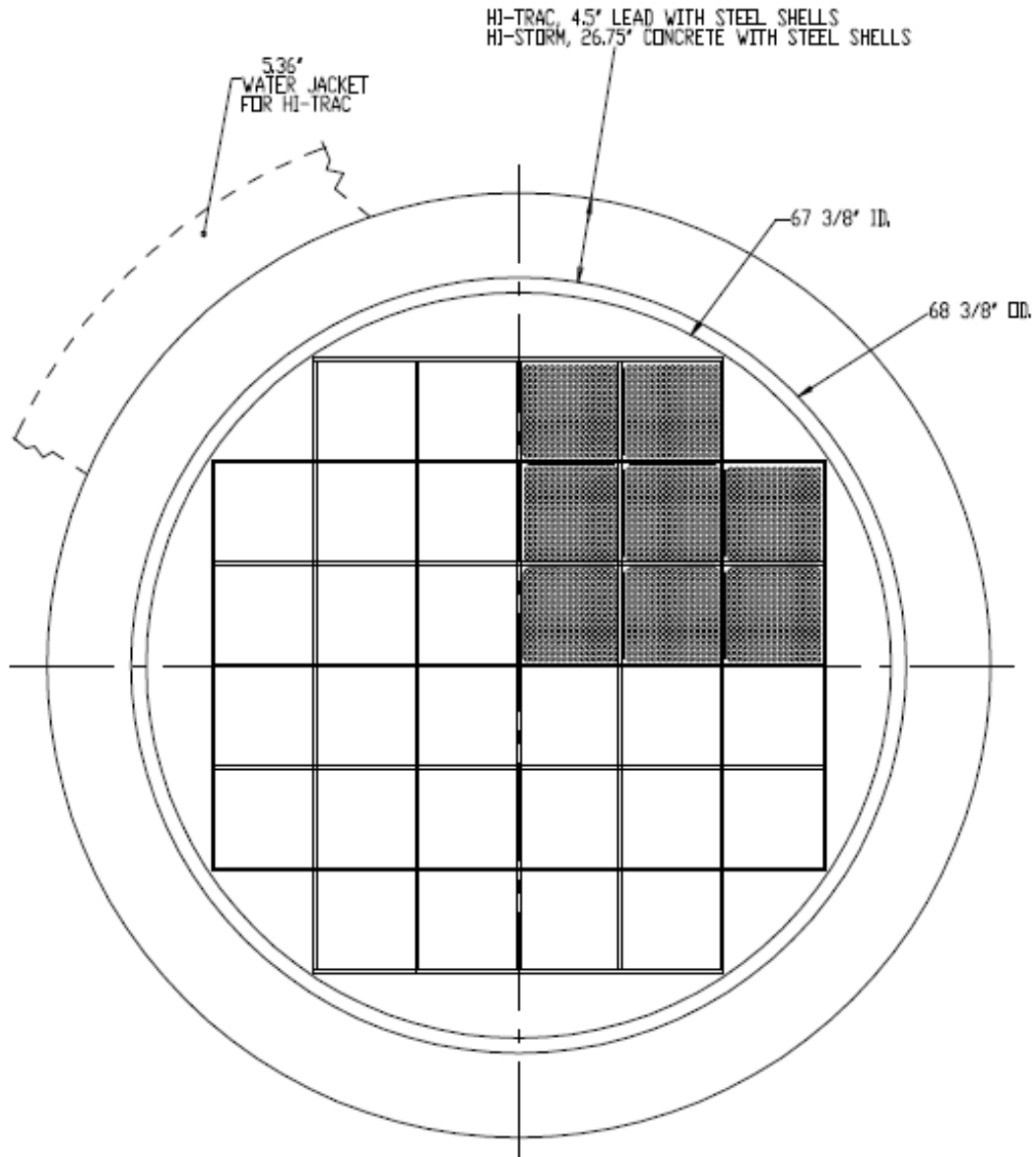
**FIGURE 6.3.3: TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**

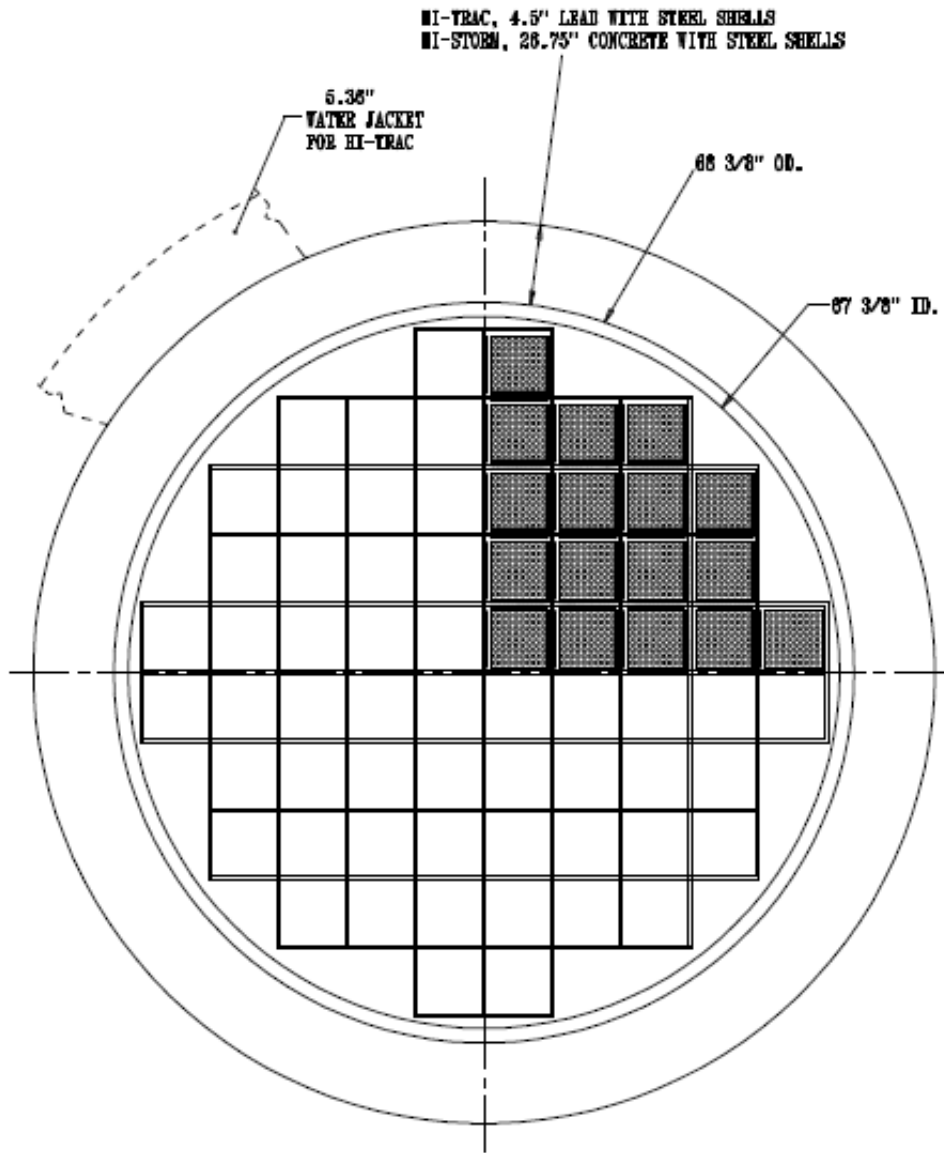




**FIGURE 6.3.4: CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-24 AND THE MPC-24E (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

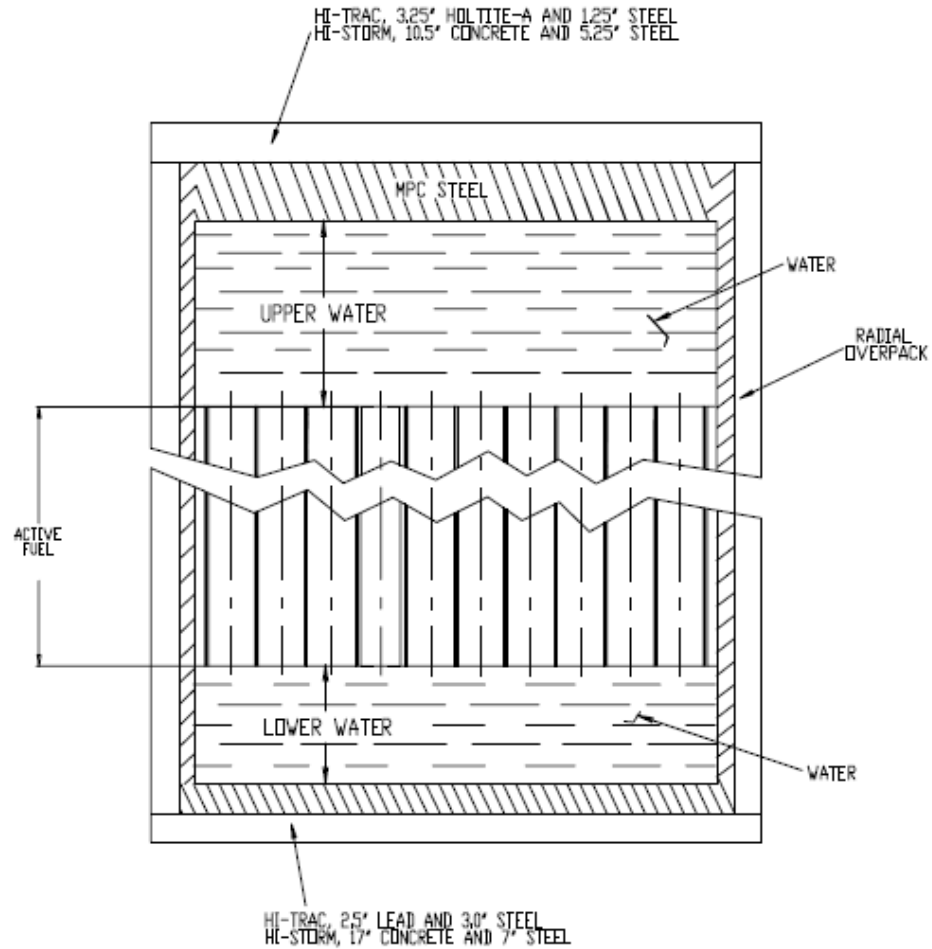


**FIGURE 6.3.5: CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-32 (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**



**FIGURE 6.3.6: CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-68 (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

	ACTIVE FUEL LENGTH	LOWER WATER THICKNESS	UPPER WATER THICKNESS
MPC-68	SEE TABLE 6.2.1	7.30 IN.	8.46 IN.
MPC-24, -24E & -32	SEE TABLE 6.2.2	4.0 IN.	6.0 IN.



**FIGURE 6.3.7: SKETCH OF THE CALCULATIONAL MODEL IN THE AXIAL DIRECTION**

## 6.4 CRITICALITY CALCULATIONS

### 6.4.1 Calculational or Experimental Method

#### 6.4.1.1 Basic Criticality Safety Calculations

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, a minimum of 5,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket in the HI-STORM 100 System.

CASMO-3 [6.1.9] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO-3 has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO-3 calculations.

## 6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Tables 6.1.1 through 6.1.8. These calculations were based on the assumption that the HI-STORM 100 System (HI-TRAC transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

### 6.4.2.1 Internal and External Moderation

As required by NUREG-1536, calculations in this section demonstrate that the HI-STORM 100 System remains subcritical for all credible conditions of moderation.

#### 6.4.2.1.1 Unborated Water

With a neutron absorber present (i.e., the fixed neutron absorber sheets or the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask were made to confirm that the phenomenon does not occur with low density water inside or outside the casks.

Calculations for the MPC designs with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case 1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Figure 6.4.10 plots calculated  $k_{\text{eff}}$  values ( $\pm 2\sigma$ ) as a function of internal moderator density for both MPC designs with 100% external moderator density (i.e., full water reflection). Results listed in Table 6.4.1 support the following conclusions:

- For each type of MPC, the calculated  $k_{\text{eff}}$  for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and
- For each type of MPC, reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STORM 100 System.

For each of the MPC designs, the maximum  $k_{\text{eff}}$  values are shown to be less than or statistically

equal to that of a single internally flooded unreflected cask and are below the regulatory limit of 0.95.

#### 6.4.2.1.2 Borated Water

With the presence of a soluble neutron absorber in the water, the discussion in the previous section is not always applicable. Calculations were made to determine the optimum moderator density for the MPC designs that require a minimum soluble boron concentration.

Calculations for the MPC designs with various internal moderator densities are shown in Table 6.4.6. As shown in the previous section, the external moderator density has a negligible effect on the reactivity, and is therefore not varied. Water containing soluble boron has a slightly higher density than pure water. Therefore, water densities up to 1.005 g/cm<sup>3</sup> were analyzed for the higher soluble boron concentrations. Additionally, for the higher soluble boron concentrations, analyses have been performed with empty (voided) guide tubes. This variation is discussed in detail in Section 6.4.8. Results listed in the Table 6.4.6 support the following conclusions:

- For all cases with a soluble boron concentration of up to 1900ppm, and for a soluble boron concentration of 2600ppm assuming voided guide tubes, the conclusion of the Section 6.4.2.1.1 applies, i.e. the maximum reactivity corresponds to 100% moderator density.
- For 2600ppm soluble boron concentration with filled guide tubes, the results presented in Table 6.4.6 indicate that there is a maximum of the reactivity somewhere between 0.90 g/cm<sup>3</sup> and 1.00 g/cm<sup>3</sup> moderator density. However, a distinct maximum can not be identified, as the reactivities in this range are very close. For the purpose of the calculations with 2600ppm soluble boron concentration, a moderator density of 0.93 g/cm<sup>3</sup> was chosen, which corresponds to the highest calculated reactivity listed in Table 6.4.6.

The calculations documented in this chapter also use soluble boron concentrations other than 1900 ppm and 2600 ppm in the MPC-32. For the MPC-32 loaded with intact fuel only, soluble boron concentrations between 1300 ppm and 2600 ppm are used. For the MPC-32 loaded with intact fuel, damaged fuel and fuel debris, soluble boron concentrations between 1500 ppm and 2900 ppm are used. In order to determine the optimum moderation condition for each assembly class at the corresponding soluble boron level, evaluations are performed with filled and voided guide tubes, and for water densities of 1.0 g/cm<sup>3</sup> and 0.93 g/cm<sup>3</sup> for each class and enrichment level. Results for the MPC-32 loaded with intact fuel only are listed in Table 6.4.10 for an initial enrichment of 5.0 wt% <sup>235</sup>U and in Table 6.4.11 for an initial enrichment of 4.1 wt% <sup>235</sup>U. Corresponding results for the MPC-32 loaded with intact fuel, damaged fuel and fuel debris are listed in Table 6.4.14. The highest value listed in these tables for each assembly class is listed as the bounding value in Section 6.1.

#### 6.4.2.2 Partial Flooding

As required by NUREG-1536, calculations in this section address partial flooding in the HI-STORM 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

#### 6.4.2.3 Clad Gap Flooding

As required by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum  $k_{\text{eff}}$  values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

#### 6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e. different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 11). The fuel cladding satisfies the “acceptance criteria to limit spent fuel reconfiguration in storage casks” (ISG-11, Rev. 3), since temperatures remain below their design limits (as demonstrated in Chapter 4). For damaged fuel assemblies and fuel debris, the assemblies or debris are loaded into stainless steel Damaged Fuel Containers fitted with mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. This condition would be the result of the water tension across the mesh screens. The maximum



water level inside the DFCs for this condition is calculated from the dimensions of the mesh screen and the surface tension of water. The wetted perimeter of the screen openings is no more than 50 ft per square inch of screen. With a surface tension of water of 0.005 lbf/ft, this results in a maximum pressure across the screen of 0.25 psi, corresponding to a maximum water height in the DFC of 7 inches. For added conservatism, a value of 12 inches is used. Assuming this condition, calculations are performed for all three possible DFC configurations:

- MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)
- MPC-68 with 16 DFCs (All BWR Assembly Classes)
- MPC-24E with 4 DFCs (All PWR Assembly Classes)
- MPC-32 with 8 DFCs (All PWR Assembly Classes)

For each configuration, the case resulting in the highest maximum  $k_{\text{eff}}$  for the fully flooded condition (see Section 6.4.4) is re-analyzed assuming the preferential flooding condition. For these analyses, the lower 12 inches of the active fuel in the DFCs and the water region below the active fuel (see Figure 6.3.7) are filled with full density water (1.0 g/cc). The remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). Table 6.4.4 lists the maximum  $k_{\text{eff}}$  for the four configurations in comparison with the maximum  $k_{\text{eff}}$  for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum  $k_{\text{eff}}$  than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC confinement boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Section 6.4.4.

#### 6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 11 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC transfer cask and minor damage to the concrete radiation shield for the HI-STORM storage cask, which have no adverse effect on the design parameters important to criticality safety.

As reported in Chapter 3, Table 3.4.4, the minimum factor of safety for either MPC as a result of the hypothetical cask drop or tip-over accident is 1.1 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change will be insignificant compared to the characteristic dimension of the flux trap.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM 100 System is in full compliance with the requirement of 10CRF72.124, which states that “before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.”

### 6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) are presented in section 6.2 and summarized in Section 6.1. To demonstrate the applicability of the HI-STAR analyses, results of the design basis criticality safety calculations for the HI-STAR cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor ( $k_{\text{eff}}$ ), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-3) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{\text{eff}}^{\text{max}} = k_c + K_c \sigma_c + \text{Bias} + \sigma_B$$

where:

- ⇒  $k_c$  is the calculated  $k_{\text{eff}}$  under the worst combination of tolerances;
- ⇒  $K_c$  is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final  $k_{\text{eff}}$  value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle  $k_{\text{eff}}$  values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;
- ⇒  $\sigma_c$  is the standard deviation of the calculated  $k_{\text{eff}}$ , as determined by the computer code (MCNP4a or KENO5a);
- ⇒  $\text{Bias}$  is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and
- ⇒  $\sigma_B$  is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

#### 6.4.4 Damaged Fuel and Fuel Debris

Damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs). Five (5) different DFC types with different cross sections are analyzed. Three (3) of these DFCs are designed for BWR fuel assemblies, two (2) are designed for PWR fuel assemblies. Two of the DFCs for BWR fuel are specifically designed for fuel assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. These assemblies have a smaller cross section, a shorter active length and a low initial enrichment of 2.7 wt%  $^{235}\text{U}$ , and therefore a low reactivity. The analysis for these assembly classes is presented in the following Section 6.4.4.1. The remaining three DFCs are generic DFCs designed for all BWR and PWR assembly classes. The criticality analysis for these generic DFCs is presented in Section 6.4.4.2.

##### 6.4.4.1 MPC-68 or MPC-68F loaded with Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

This section only addresses criticality calculations and results for assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A, loaded into the MPC-68 or MPC-68F. Up to 68 DFCs with these assembly classes are permissible to be loaded into the MPC. Two different DFC types with slightly different cross-sections are analyzed. Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 has a higher specified  $^{10}\text{B}$  loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68. Although the maximum planar-average enrichment of the damaged fuel is limited to 2.7%  $^{235}\text{U}$  as specified in Section 2.1.9, analyses have been made for three possible scenarios, conservatively assuming fuel<sup>††</sup> of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.2 through 6.4.8.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This

<sup>††</sup> 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

- configuration is illustrated in Figure 6.4.9.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.5, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in  $^{238}\text{U}$  neutron capture (higher effective resonance integral for  $^{238}\text{U}$  absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.5.

The analyses performed and summarized in Table 6.4.5 provide the relative magnitude of the effects on the reactivity. This information coupled with the maximum  $k_{\text{eff}}$  values listed in Table 6.1.3 and the conservatism in the analyses, demonstrates that the maximum  $k_{\text{eff}}$  of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of  $k_{\text{eff}} < 0.95$ .

#### 6.4.4.2 Generic BWR and PWR Damaged Fuel and Fuel Debris

The MPC-24E, MPC-32, and MPC-68 are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into generic DFCs. The number of generic DFCs is limited to 16 for the MPC-68, to 4 for the MPC-24E, and to 8 for the MPC-32. The permissible locations of the DFCs are shown in Figure 6.4.11 for the MPC-68, in Figure 6.4.12 for the MPC-24E and in Figure 6.4.16 for the MPC-32.

Damaged fuel assemblies are assemblies with known or suspected cladding defects greater than pinholes or hairlines, or with missing rods, but excluding fuel assemblies with gross defects (for a full definition see glossary). Therefore, apart from possible missing fuel rods, damaged fuel assemblies have the same geometric configuration as intact fuel assemblies and consequently the same reactivity. Missing fuel rods can result in a slight increase of reactivity. After a drop accident, however, it can not be assumed that the initial geometric integrity is still maintained. For a drop on either the top or bottom of the cask, the damaged fuel assemblies could collapse. This would result in a configuration with a reduced length, but increased amount of fuel per unit length. For a side drop, fuel rods could be compacted to one side of the DFC. In either case, a significant relocation of fuel within the DFC is possible, which creates a greater amount of fuel in some areas of the DFC, whereas the amount of fuel in other areas is reduced. Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

In the cases of fuel debris or relocated damaged fuel, there is the potential that fuel could be present in axial sections of the DFCs that are outside the basket height covered with the fixed neutron absorber. However, in these sections, the DFCs are not surrounded by any intact fuel, only by basket cell walls, non-fuel hardware, and water and for the MPC-68 by a maximum of one other DFC. Studies have shown that this condition does not result in any significant effect on reactivity, compared to a condition where the damaged fuel and fuel debris is restricted to the axial section of the basket covered by the fixed neutron absorber. All calculations for generic BWR and PWR damaged fuel and fuel debris are therefore performed assuming that fuel is present only in the axial sections covered by the fixed neutron absorber, and the results are directly applicable to any situation where damaged fuel and fuel debris is located outside these sections in the DFCs.

To address all the situations listed above and identify the configuration or configurations leading to the highest reactivity, it is impractical to analyze a large number of different geometrical configurations for each of the fuel classes. Instead, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following sections.

All calculations for generic damaged fuel and fuel debris are performed using a full cask model with the maximum permissible number of Damaged Fuel Containers. For the MPC-68, the model therefore contains 52 intact assemblies, and 16 DFCs in the locations shown in Figure 6.4.11. For the MPC-24E, the model consists of 20 intact assemblies, and 4 DFCs in the locations shown in Figure 6.4.12. For the MPC-32, the model consists of 24 intact assemblies, and 8 DFCs in the locations shown in Figure 6.4.16. The bounding assumptions regarding the intact assemblies and the modeling of the damaged fuel and fuel debris in the DFCs are discussed in the following sections.

Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term ‘damaged fuel’ as used throughout this chapter designates both damaged fuel and fuel debris.

#### 6.4.4.2.1 Bounding Intact Assemblies

Intact BWR assemblies stored together with DFCs are limited to a maximum planar average enrichment of 3.7 wt%  $^{235}\text{U}$ , regardless of the fuel class. The results presented in Table 6.1.7 are for different enrichments for each class, ranging between 2.7 and 4.2 wt%  $^{235}\text{U}$ , making it difficult to identify the bounding assembly. Therefore, additional calculations were performed for the bounding assembly in each assembly class with a planar average enrichment of 3.7 wt%. The results are summarized in Table 6.4.7 and demonstrate that the assembly classes 9x9E and 9x9F have the highest reactivity. These two classes share the same bounding assembly (see footnotes for Tables 6.2.33 and 6.2.34 for further details). This bounding assembly is used as the intact BWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-24E are limited to a maximum enrichment of 4.0 wt%  $^{235}\text{U}$  without credit for soluble boron and to a maximum enrichment of 5.0 wt% with credit for soluble boron, regardless of the fuel class. The results presented in Table 6.1.3 are for different enrichments for each class, ranging between 4.2 and 5.0 wt%  $^{235}\text{U}$ , making it difficult to directly identify the bounding assembly. However, Table 6.1.4 shows results for an enrichment of 5.0 wt% for all fuel classes, with a soluble boron concentration of 300 ppm. The assembly class 15x15H has the highest reactivity. This is consistent with the results in Table 6.1.3, where the assembly class 15x15H is among the classes with the highest reactivity, but has the lowest initial enrichment. Therefore, in the MPC-24E, the 15x15H assembly is used as the intact PWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-32 are limited to a maximum enrichment of 5.0 wt%, regardless of the fuel class. Table 6.1.5 and Table 6.1.6 show results for enrichments of 4.1 wt% and 5.0 wt%, respectively, for all fuel classes. Since different minimum soluble boron concentrations are used for different groups of assembly classes, the assembly class with the highest reactivity in each group is used as the intact assembly for the calculations with DFCs in the MPC-32. These assembly classes are

- 14x14C for all 14x14 assembly classes;
- 15x15B for assembly classes 15x15A, B, C, G and I;
- 15x15F for assembly classes 15x15D, E, F and H;
- 16x16A for assembly classes 16x16A, B and C;
- 17x17A assembly class and
- 17x17C for assembly classes 17x17B and C.

#### 6.4.4.2.2 Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e. all cladding and other structural material in the DFC is replaced by water.
- For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.
- The active length of these rods is chosen to be the maximum active fuel length of all fuel assemblies listed in Section 6.2, which is 155 inch for BWR fuel and 150 inch for PWR fuel.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 9 (3x3) and 189 (17x17) for BWR fuel, and between 64 (8x8) and 729 (27x27) for PWR fuel.
- Analyses are performed for the minimum, maximum and typical pellet diameter of PWR and BWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

This is also a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 155 inch (BWR) or 150 inch (PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

To demonstrate the level of conservatism, additional analyses are performed with the DFC containing various realistic assembly configurations such as intact assemblies, assemblies with missing fuel rods and collapsed assemblies, i.e. assemblies with increased number of rods and decreased rod pitch.

As discussed in Section 6.4.4.2, all calculations are performed for full cask models, containing the maximum permissible number of DFCs together with intact assemblies.

As an example of the damaged fuel model used in the analyses, Figure 6.4.17 shows the basket cell of an MPC-32 with a DFC containing a 17x17 array of bare fuel rods.

Graphical presentations of the calculated maximum  $k_{\text{eff}}$  for typical cases as a function of the fuel mass per unit length of the DFC are shown in Figures 6.4.13 (BWR) and 6.4.14 (PWR, MPC-24E with pure water). The results for the bare fuel rods show a distinct peak in the maximum  $k_{\text{eff}}$  at about 2 kg UO<sub>2</sub>/inch for BWR fuel, and at about 3.5 kgUO<sub>2</sub>/inch for PWR fuel.

The realistic assembly configurations are typically about 0.01 (delta-k) or more below the peak results for the bare fuel rods, demonstrating the conservatism of this approach to model damaged fuel and fuel debris configurations such as severely damaged assemblies and bundles of fuel rods.

For fuel debris configurations consisting of bare fuel pellets only, the fuel mass per unit length would be beyond the value corresponding to the peak reactivity. For example, for DFCs filled with a mixture of 60 vol% fuel and 40 vol% water the fuel mass per unit length is 3.36 kgUO<sub>2</sub>/inch for

the BWR DFC and 7.92 kgUO<sub>2</sub>/inch for the PWR DFC. The corresponding reactivities are significantly below the peak reactivity. The difference is about 0.005 (delta-k) for BWR fuel and 0.01 (delta-k) or more for PWR fuel. Furthermore, the filling height of the DFC would be less than 70 inches in these examples due to the limitation of the fuel mass per basket position, whereas the calculation is conservatively performed for a height of 155 inch (BWR) or 150 inch (PWR). These results demonstrate that even for the fuel debris configuration of bare fuel pellets, the model using bare fuel rods is a conservative approach.

#### 6.4.4.2.3 Distributed Enrichment in BWR Fuel

BWR fuel usually has an enrichment distribution in each planar cross section, and is characterized by the maximum planar average enrichment. For intact fuel it has been shown that using the average enrichment for each fuel rod in a cross section is conservative, i.e. the reactivity is higher than calculated for the actual enrichment distribution (See Appendix 6.B). For damaged fuel assemblies, additional configurations are analyzed to demonstrate that the distributed enrichment does not have a significant impact on the reactivity of the damaged assembly under accident conditions. Specifically, the following two scenarios were analyzed:

- As a result of an accident, fuel rods with lower enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the top part, but at the same time the amount of fuel in that area is reduced compared to the intact assembly.
- As a result of an accident, fuel rods with higher enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the bottom part, and at the same time the amount of fuel in that area is increased compared to the intact assembly, leading to a reduction of the water content.

In both scenarios, a compensation of effects on reactivity is possible, as the increase of reactivity due to the increased planar average enrichment might be offset by the possible reduction of reactivity due to the change in the fuel to water ratio. A selected number of calculations have been performed for these scenarios and the results show that there is only a minor change in reactivity. These calculations are shown in Figure 6.4.13 in the group of the explicit assemblies. Consequently, it is appropriate to qualify damaged BWR fuel assemblies and fuel debris based on the maximum planar average enrichment. For assemblies with missing fuel rods, this maximum planar average enrichment has to be determined based on the enrichment and number of rods still present in the assembly when loaded into the DFC.

#### 6.4.4.2.4 Results for the MPC-68

The MPC-68 allows the storage of up to sixteen DFCs in the shaded cells on the periphery of the basket shown in Figure 6.4.11. Additionally, up to 8 of these cells may contain DFCs with fuel



debris. The various configurations outlined in Sections 6.4.4.2.2 and 6.4.4.2.3 are analyzed with an enrichment of the intact fuel of 3.7%  $^{235}\text{U}$  and an enrichment of damaged fuel or fuel debris of 4.0%  $^{235}\text{U}$ . For the intact assembly, the bounding assembly of the 9x9E and 9x9F fuel classes was chosen. This assembly has the highest reactivity of all BWR assembly classes for the initial enrichment of 3.7 wt%  $^{235}\text{U}$ , as demonstrated in Table 6.4.7. The results for the various configurations are summarized in Figure 6.4.13 and in Table 6.4.8. Figure 6.4.13 shows the maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, for various actual and hypothetical damaged fuel or fuel debris configurations as a function of the fuel mass per unit length of the DFC. Table 6.4.8 lists the highest maximum  $k_{\text{eff}}$  for the various configurations. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.

#### 6.4.4.2.5 Results for the MPC-24E

The MPC-24E allows the storage of up to four DFCs with damaged fuel or fuel debris in the four outer fuel baskets cells shaded in Figure 6.4.12. These locations are designed with a larger box ID to accommodate the DFCs. For an enrichment of 4.0 wt%  $^{235}\text{U}$  for the intact fuel, damaged fuel and fuel debris, and assuming no soluble boron, the results for the various configurations outlined in Section 6.4.4.2.2 are summarized in Figure 6.4.14 and in Table 6.4.9. Figure 6.4.14 shows the maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, for various actual and hypothetical damaged fuel and fuel debris configurations as a function of the fuel mass per unit length of the DFC. For the intact assemblies, the 15x15H assembly class was chosen. This assembly class has the highest reactivity of all PWR assembly classes for a given initial enrichment. This is demonstrated in Table 6.1.4. Table 6.4.9 lists the highest maximum  $k_{\text{eff}}$  for the various configurations. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.

For an enrichment of 5.0 wt%  $^{235}\text{U}$  for the intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration of 600 ppm is required. For this condition, calculations are performed for various hypothetical fuel debris configurations (i.e. bare fuel rods) as a function of the fuel mass per unit length of the DFC. Additionally, calculations are performed with reduced water densities in the DFC. The various conditions of damaged fuel, such as assemblies with missing rods or collapsed assemblies, were not analyzed, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods. Again, the 15x15H assembly class was chosen as the intact assembly since this assembly class has the highest reactivity of all PWR assembly classes as demonstrated in Table 6.1.4. The results are summarized in Table 6.4.12. Similar to the calculations with pure water (see Figure 6.4.14), the results for borated water show a distinct peak of the maximum  $k_{\text{eff}}$  as a function of the fuel mass per unit length. Therefore, for each condition, the table lists only the highest maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the reactivity decreases with decreasing water density. This demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. All maximum  $k_{\text{eff}}$  values are below

the 0.95 regulatory limit.

#### 6.4.4.2.6 Results for the MPC-32

The MPC-32 allows the storage of up to eight DFCs with damaged fuel or fuel debris in the outer fuel basket cells shaded in Figure 6.4.16. For the MPC-32, additional cases are analyzed due to the high soluble boron level required for this basket:

- The assembly classes of the intact assemblies are grouped, and minimum required soluble boron levels are determined separately for each group. The analyses are performed for the bounding assembly class in each group. The bounding assembly classes are listed in Section 6.4.4.2.1.
- Evaluations of conditions with voided and filled guide tubes and various water densities in the MPC and DFC are performed to identify the most reactive condition.

In general, all calculations performed for the MPC-32 show the same principal behavior as for the MPC-24 (see Figure 6.4.14), i.e. the reactivity as a function of the fuel mass per unit length for the bare fuel rod array shows a distinct peak. Therefore, for each condition analyzed, only the highest maximum  $k_{\text{eff}}$ , i.e. the calculated peak reactivity, is listed in the tables. Evaluations of different diameters of the bare fuel pellets and the reduced water density in the DFC have been performed for a representative case using the 15x15F assembly class as the intact assembly, with voided guide tubes, a water density of 1.0 g/cc in the DFC and MPC, 2900 ppm soluble boron, and an enrichment of 5.0 wt%  $^{235}\text{U}$  for the intact and damaged fuel and fuel debris. For this case, results are summarized in Table 6.4.13. For each condition, the table lists the highest maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the fuel pellet diameter in the DFC has an insignificant effect on reactivity, and that reactivity decreases with decreasing water density. The latter demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. Therefore, a typical fuel pellet diameter and a water density of 1.0 in the DFCs are used for all further analyses. Two enrichment levels are analyzed, 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , consistent with the analyses for intact fuel only. In any calculation, the same enrichment is used for the intact fuel and the damaged fuel and fuel debris. For both enrichment levels, analyses are performed with voided and filled guide tubes, each with water densities of 0.93 and 1.0 g/cm<sup>3</sup> in the MPC. In all cases, the water density inside the DFCs is assumed to be 1.0 g/cm<sup>3</sup>, since this is the most reactive condition as shown in Table 6.4.13. Results are summarized in Table 6.4.14. For each group of assembly classes, the table shows the soluble boron level and the highest maximum  $k_{\text{eff}}$  for the various moderation conditions of the intact assembly. The highest maximum  $k_{\text{eff}}$  is the highest value of any of the hypothetical fuel debris configurations, i.e. various arrays of bare fuel rods. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. Conditions of damaged fuel such as assemblies with missing rods or collapsed assemblies were not analyzed in the MPC-32, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the

hypothetical model for fuel debris based on regular arrays of bare fuel rods.

#### 6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods. The number of dummy rods that is used to replace missing and/or damaged fuel rods is not limited.

#### 6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.15. The  $k_{\text{eff}}$  value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in  $^{235}\text{U}$  (equivalent to  $\text{UO}_2$  fuel with an enrichment of approximately up to 1.7 wt%  $^{235}\text{U}$ ), the large spacing between the rods (the pitch is approximately 1", the cladding OD is 0.412") and the absorption in the separator assembly. Together with the maximum  $k_{\text{eff}}$  values listed in Tables 6.1.7 and 6.1.8 this result demonstrates, that the  $k_{\text{eff}}$  for a Thoria Rod Canister loaded into the MPC-68 or the MPC-68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of  $k_{\text{eff}} < 0.95$ . The evaluation presented here was performed with fresh fuel, but for a condition where all basket locations were assumed to be filled with a Thoria Rod canister, while only a single canister is permitted. The conversion of Th-232 to U-233 during depletion may result in a slight increase in reactivity for this hypothetical case of a basket entirely filled with Thoria Rod Canisters, however, the real condition of a single canister loaded together with spent fuel would still be bounded by the design basis case with fuel assemblies only.

#### 6.4.7 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

#### 6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware such as Thimble Plugs (TPs), Burnable Poison Rod Assemblies (BPRAs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and similar devices are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged. This conclusion is supported by the calculation listed in Table 6.2.4, which shows a significant reduction in reactivity as a result of voided guide tubes, i.e. the removal of the water from the guide tubes.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes are filled or assumed empty. For the MPC-24 with 400 ppm and the MPC-32 with 1900 ppm, voiding the guide tubes results in a reduction of reactivity. All calculations for the MPC-24 and MPC-24E are therefore performed with water in the guide tubes. For the MPC-32 with 2600 ppm, the reactivity for voided guide tubes slightly exceeds the reactivity for filled guide tubes. However, this effect is not consistent across all assembly classes. Table 6.4.10, Table 6.4.11 and Table 6.4.14 show results with filled and voided guide tubes for all assembly classes in the MPC-32 at 4.1 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ . Some classes show an increase, other classes show a decrease as a result of voiding the guide tubes. Therefore, for the results presented in the Section 6.1, Table 6.1.5, Table 6.1.6 and Table 6.1.12, the maximum value for each class is chosen for each enrichment level.

An Instrument Tube Tie Rod (ITTR) is inserted into the instrument tube and is permitted for storage with all PWR assemblies. Studies for representative PWR assemblies, including the assembly with the lowest margin (15x15F in the MPC-32), with voided instrument tubes confirm that this condition is equivalent to or bounded by the condition with flooded instrument tubes. An ITTR in the assembly instrument tube is therefore acceptable in all PWR assembly types, and all results and conclusions for PWR fuel assemblies without an ITTR are directly applicable to PWR assemblies with an ITTR.

In summary, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

#### 6.4.9 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a  $k_{eff}$  less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

#### 6.4.10 Applicability of HI-STAR Analyses to HI-STORM 100 System

Calculations previously supplied to the NRC in applications for the HI-STAR 100 System (Docket Numbers 71-9261 and 72-1008) are directly applicable to the HI-STORM storage and HI-TRAC transfer casks. The MPC designs are identical. The cask systems differ only in the overpack shield material. The limiting condition for the HI-STORM 100 System is the fully flooded HI-TRAC transfer cask. As demonstrated by the comparative calculations presented in Tables 6.1.1 through 6.1.8, the shield material in the overpack (steel and lead for HI-TRAC, steel for HI-STAR) has a negligible impact on the eigenvalue of the cask systems. As a result, this analysis for the 125-ton HI-TRAC transfer cask is applicable to the 100-ton HI-TRAC transfer cask. In all cases, for the reference fuel assemblies, the maximum  $k_{eff}$  values are in good agreement and are conservatively less than the limiting  $k_{eff}$  value (0.95).

#### 6.4.11 Fixed Neutron Absorber Material

The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorber materials: Boral or Metamic. Both materials are made of aluminum and  $B_4C$  powder. Boral has an inner core consisting of  $B_4C$  and aluminum between two outer layers consisting of aluminum only. This configuration is explicitly modeled in the criticality evaluation and shown in Figures 6.3.1 through 6.3.3 for each basket. Metamic is a single layer material with a slightly higher overall thickness and the same credited  $^{10}B$  loading (in  $g/cm^2$ ) for each basket. The majority of the criticality evaluations documented in this chapter are performed using Boral as the fixed neutron absorber. For a selected number of bounding cases, analyses are also performed using Metamic instead of Boral. (Note that the Metamic cases use the same absorber thickness as the corresponding Boral case, instead of the slightly increased thickness for Metamic. This is acceptable since analyses of slight thickness increases for a fixed  $^{10}B$  loading (in  $g/cm^2$ ) indicate

that such increases have a negligible effect on reactivity.) The results for these cases are listed in Table 6.4.15, together with the corresponding result using Boral and the difference between the two materials for each case. Individual cases show small differences for the two materials. However, the differences are mostly below two times the standard deviation (the standard deviation is about 0.0008 for all cases in Table 6.4.15), indicating that the results are statistically equivalent. Furthermore, the average difference is well below one standard deviation, and all cases are below the regulatory limit of 0.95. In some cases listed in Table 6.4.15, the reactivity difference between Metamic and Boral might be larger than expected for two equivalent materials. Also, for four out of the five cases with MPC-24 type baskets, Metamic shows the higher reactivity, which could potentially indicate a trend rather than a statistical variation. Therefore, in order to confirm that the materials are equivalent, a second set of calculations was performed for Metamic, which was statistically independent from the set shown in Table 6.4.15. This was achieved by selecting a different starting value for the random number generator in the Monte Carlo calculations. The second set also shows some individual variations of the differences, and a low average difference. However, there is no apparent trend regarding the MPC-24 type baskets compared to the MPC-32 and MPC-68, and the maximum positive reactivity difference for Metamic in an MPC-24 type basket is only 0.0005. Overall, the calculations demonstrate that the two fixed neutron absorber materials are identical from a criticality perspective. All results obtained for Boral are therefore directly applicable to Metamic and no further evaluations using Metamic are required.

#### 6.4.12 Annular Fuel Pellets

Typically, PWR fuel assemblies are designed with solid fuel pellets throughout the entire active fuel length. However, some PWR assemblies contain annular fuel pellets in the top and bottom 6 to 8 inches of the active fuel length. This changes the fuel to water ratio in these areas, which could have an effect on reactivity. However, the top and bottom of the active length are areas with high neutron leakage, and changes in these areas typically have no significant effect on reactivity. Studies with up to 12 inches of annular pellets at the top and bottom, with various pellet IDs confirm this, i.e., shown no significant reactivity effects, even if the annular region of the pellet is flooded with pure water. All calculations for PWR fuel assemblies are therefore performed with solid fuel pellets along the entire length of the active fuel region, and the results are directly applicable to those PWR assemblies with annular fuel pellets.

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS<sup>†</sup>

Case Number	Water Density		MCNP4a Maximum $k_{eff}$ <sup>††</sup>	
	Internal	External	MPC-24 (17x17A01 @ 4.0%)	MPC-68 (8x8C04 @ 4.2%)
1	100%	single cask	0.9368	0.9348
2	100%	100%	0.9354	0.9339
3	100%	70%	0.9362	0.9339
4	100%	50%	0.9352	0.9347
5	100%	20%	0.9372	0.9338
6	100%	10%	0.9380	0.9336
7	100%	5%	0.9351	0.9333
8	100%	0%	0.9342	0.9338
9	70%	0%	0.8337	0.8488
10	50%	0%	0.7426	0.7631
11	20%	0%	0.5606	0.5797
12	10%	0%	0.4834	0.5139
13	5%	0%	0.4432	0.4763
14	10%	100%	0.4793	0.4946

<sup>†</sup> For an infinite square array of casks with 60cm spacing between cask surfaces.

<sup>††</sup> Maximum  $k_{eff}$  includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

## REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

<b>MPC-24 (17x17A01 @ 4.0% ENRICHMENT) (no soluble boron)</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
<b>MPC-68 (8x8C04 @ 4.2% ENRICHMENT)</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348
<b>MPC-32 (15x15F @ 5.0 % ENRICHMENT) 2600ppm Soluble Boron</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.8927	31.25	0.9213
50	0.9215	50	0.9388
75	0.9350	68.75	0.9401
100	0.9445	100	0.9445

Notes:

1. All values are maximum  $k_{\text{eff}}$  which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.



Table 6.4.3

## REACTIVITY EFFECT OF FLOODING THE PELLETT-TO-CLAD GAP

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded with unborated water	0.9368	0.9348

## Notes:

1. All values are maximum  $k_{\text{eff}}$  which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.4

## REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Preferential Flooding	Fully Flooded
MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)	0.6560	0.7857
MPC-68 with 16 DFCs (All BWR Assembly Classes)	0.6646	0.9328
MPC-24E with 4 DFCs (All PWR Assembly Classes)	0.7895	0.9480
MPC-32 with 8 DFCs (All PWR Assembly Classes)	0.7213	0.9378

## Notes:

1. All values are maximum  $k_{\text{eff}}$  which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.5

MAXIMUM  $k_{\text{eff}}$  VALUES<sup>†</sup> IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Maximum <sup>††</sup> $k_{\text{eff}}$	
	DFC Dimensions: ID 4.93” THK. 0.12”	DFC Dimensions: ID 4.81” THK. 0.11”
<u>6x6 Fuel Assembly</u>		
6x6 Intact Fuel	0.7086	0.7016
w/32 Rods Standing	0.7183	0.7117
w/28 Rods Standing	0.7315	0.7241
w/24 Rods Standing	0.7086	0.7010
w/18 Rods Standing	0.6524	0.6453
Collapsed to 8x8 array	0.7845	0.7857
Dispersed Powder	0.7628	0.7440
<u>7x7 Fuel Assembly</u>		
7x7 Intact Fuel	0.7463	0.7393
w/41 Rods Standing	0.7529	0.7481
w/36 Rods Standing	0.7487	0.7444
w/25 Rods Standing	0.6718	0.6644

<sup>†</sup> These calculations were performed with a planar-average enrichment of 3.0% and a <sup>10</sup>B loading of 0.0067 g/cm<sup>2</sup>, which is 75% of a minimum <sup>10</sup>B loading of 0.0089 g/cm<sup>2</sup>. The minimum <sup>10</sup>B loading in the MPC-68F is 0.010 g/cm<sup>2</sup>. Therefore, the listed maximum  $k_{\text{eff}}$  values are conservative

<sup>††</sup> Maximum  $k_{\text{eff}}$  includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.6

MAXIMUM  $k_{eff}$  VALUES WITH REDUCED BORATED WATER DENSITIES

Internal Water Density <sup>†</sup> in g/cm <sup>3</sup>	Maximum $k_{eff}$				
	MPC-24 (400ppm) @ 5.0 %	MPC-32 (1900ppm) @ 4.1 %		MPC-32 (2600ppm) @ 5.0 %	
Guide Tubes	filled	filled	void	filled	void
1.005	NC <sup>††</sup>	0.9403	0.9395	NC	0.9481
1.00	0.9314	0.9411	0.9400	0.9445	0.9483
0.99	NC	0.9393	0.9396	0.9438	0.9462
0.98	0.9245	0.9403	0.9376	0.9447	0.9465
0.97	NC	0.9397	0.9391	0.9453	0.9476
0.96	NC	NC	NC	0.9446	0.9466
0.95	0.9186	0.9380	0.9384	0.9451	0.9468
0.94	NC	NC	NC	0.9445	0.9467
0.93	0.9130	0.9392	0.9352	0.9465	0.9460
0.92	NC	NC	NC	0.9458	0.9450
0.91	NC	NC	NC	0.9447	0.9452
0.90	0.9061	0.9384	NC	0.9449	0.9454
0.80	0.8774	0.9322	NC	0.9431	0.9390
0.70	0.8457	0.9190	NC	0.9339	0.9259
0.60	0.8095	0.8990	NC	0.9194	0.9058
0.40	0.7225	0.8280	NC	0.8575	0.8410
0.20	0.6131	0.7002	NC	0.7421	0.7271
0.10	0.5486	0.6178	NC	0.6662	0.6584

<sup>†</sup> External moderator is modeled at 0%. This is consistent with the results demonstrated in Table 6.4.1.

<sup>††</sup> NC: Not Calculated

Table 6.4.7

MAXIMUM  $k_{\text{eff}}$  VALUES FOR INTACT BWR FUEL ASSEMBLIES WITH A MAXIMUM PLANAR AVERAGE ENRICHMENT OF 3.7 wt%  $^{235}\text{U}$

Fuel Assembly Class	Maximum $k_{\text{eff}}$
6x6A	0.8287
6x6C	0.8436
7x7A	0.8399
7x7B	0.9109
8x8A	0.8102
8x8B	0.9131
8x8C	0.9115
8x8D	0.9125
8x8E	0.9049
8x8F	0.9233
9x9A	0.9111
9x9B	0.9134
9x9C	0.9103
9x9D	0.9096
9x9E	0.9237
9x9F	0.9237
9x9G	0.9005
10x10A	0.9158
10x10B	0.9156
10x10C	0.9152
10x10D	0.9182
10x10E	0.8970

Table 6.4.8

MAXIMUM  $k_{eff}$  VALUES IN THE GENERIC BWR DAMAGED FUEL CONTAINER FOR A  
 MAXIMUM INITIAL ENRICHMENT OF 4.0 wt%  $^{235}\text{U}$  FOR DAMAGED FUEL AND 3.7  
 wt%  $^{235}\text{U}$  FOR INTACT FUEL

<b>Model Configuration inside the DFC</b>	<b>Maximum <math>k_{eff}</math></b>
Intact Assemblies (4 assemblies analyzed)	0.9241
Assemblies with missing rods (7 configurations analyzed)	0.9240
Assemblies with distributed enrichment (4 configurations analyzed)	0.9245
Collapsed Assemblies (6 configurations analyzed)	0.9258
Regular Arrays of Bare Fuel Rods (31 configurations analyzed)	0.9328

Table 6.4.9

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-24E WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt%  $^{235}\text{U}$  AND NO SOLUBLE BORON.

<b>Model Configuration inside the DFC</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
Intact Assemblies (2 assemblies analyzed)	0.9340
Assemblies with missing rods (4 configurations analyzed)	0.9350
Collapsed Assemblies (6 configurations analyzed)	0.9360
Regular Arrays of Bare Fuel Rods (36 configurations analyzed)	0.9480

Table 6.4.10

MAXIMUM  $k_{eff}$  VALUES WITH FILLED AND VOIDED GUIDE TUBES  
FOR THE MPC-32 AT 5.0 wt% ENRICHMENT

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 5.0 %			
		Guide Tubes Filled,		Guide Tubes Voided,	
		1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>	1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>
14x14A	1900	0.8984	0.9000	0.8953	0.8943
14x14B	1900	0.9210	0.9214	0.9164	0.9118
14x14C	1900	0.9371	0.9376	0.9480	0.9421
14x14D	1900	0.9050	0.9027	0.8947	0.8904
14x14E	1900	0.7415	0.7301	n/a	n/a
15x15A	2500	0.9210	0.9223	0.9230	0.9210
15x15B	2500	0.9402	0.9420	0.9429	0.9421
15x15C	2500	0.9258	0.9292	0.9307	0.9293
15x15D	2600	0.9426	0.9419	0.9466	0.9440
15x15E	2600	0.9394	0.9415	0.9434	0.9442
15x15F	2600	0.9445	0.9465	0.9483	0.9460
15x15G	2500	0.9228	0.9244	0.9251	0.9243
15X15H	2600	0.9271	0.9301	0.9317	0.9333
15x15I <sup>‡</sup>	2500	0.9402	0.9363	-	-
16X16A	2000	0.9377	0.9375	0.9429	0.9389
16X16B	2000	0.9326	0.9338	0.9378	0.9358
16X16C	2000	0.9208	0.9193	0.9091	0.9055
17x17A	2200	0.9472	0.9468	0.9475	0.9459
17x17B	2600	0.9345	0.9358	0.9371	0.9356
17X17C	2600	0.9417	0.9431	0.9437	0.9430

<sup>‡</sup> This array/class has solid guide rods that cannot be filled or voided.



Table 6.4.11

MAXIMUM  $k_{eff}$  VALUES WITH FILLED AND VOIDED GUIDE TUBES  
FOR THE MPC-32 AT 4.1 wt% ENRICHMENT

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 4.1 %			
		Guide Tubes Filled		Guide Tubes Voided	
		1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>	1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
14x14E	1300	0.7340	0.7204	n/a	n/a
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
15x15I <sup>§</sup>	1800	0.9340	0.9316	-	-
16X16A	1400	0.9367	0.9347	0.9375	0.9308
16X16B	1400	0.9336	0.9319	0.9354	0.9283
16X16C	1400	0.9178	0.9130	0.9039	0.8965
17x17A	1600	0.9421	0.9413	0.9396	0.9357
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

<sup>§</sup> This array/class has solid guide rods that cannot be filled or voided.

Table 6.4.12

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-24E WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt%  $^{235}\text{U}$  AND 600 PPM SOLUBLE BORON.

<b>Water Density inside the DFC</b>	<b>Bare Fuel Pellet Diameter</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
1.00	minimum	0.9185
1.00	typical	0.9181
1.00	maximum	0.9171
0.95	typical	0.9145
0.90	typical	0.9125
0.60	typical	0.9063
0.10	typical	0.9025
0.02	typical	0.9025

Table 6.4.13

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-32 WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt%  $^{235}\text{U}$ , 2900 PPM SOLUBLE BORON AND THE 15x15F ASSEMBLY CLASS AS INTACT ASSEMBLY.

<b>Water Density inside the DFC</b>	<b>Bare Fuel Pellet Diameter</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
1.00	minimum	0.9374
1.00	typical	0.9372
1.00	maximum	0.9373
0.95	typical	0.9369
0.90	typical	0.9365
0.60	typical	0.9308
0.10	typical	0.9295
0.02	typical	0.9283

Table 6.4.14

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR THE MPC-32  
WITH UP TO 8 DFCs UNDER VARIOUS MODERATION CONDITIONS.

Fuel Assembly Class of Intact Fuel	Initial Enrichment (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Content (ppm)	Maximum $k_{\text{eff}}$			
			Filled Guide Tubes		Voided Guide Tubes	
			1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>	1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>
14x14A through 14x14E	4.1	1500	0.9277	0.9283	0.9336	0.9298
	5.0	2300	0.9139	0.9180	0.9269	0.9262
15x15A, B, C, G, I	4.1	1900	0.9345	0.9350	0.9350	0.9326
	5.0	2700	0.9307	0.9346	0.9347	0.9365
15x15D, E, F, H	4.1	2100	0.9322	0.9336	0.9340	0.9329
	5.0	2900	0.9342	0.9375	0.9385	0.9397
16x16A, B, C	4.1	1500	0.9330	0.9332	0.9348	0.9333
	5.0	2300	0.9212	0.9246	0.9283	0.9299
17x17A	4.1	1800	0.9298	0.9310	0.9311	0.9283
	5.0	2600	0.9228	0.9274	0.9273	0.9298
17x17B, C	4.1	2100	0.9284	0.9290	0.9294	0.9285
	5.0	2900	0.9308	0.9338	0.9355	0.9367

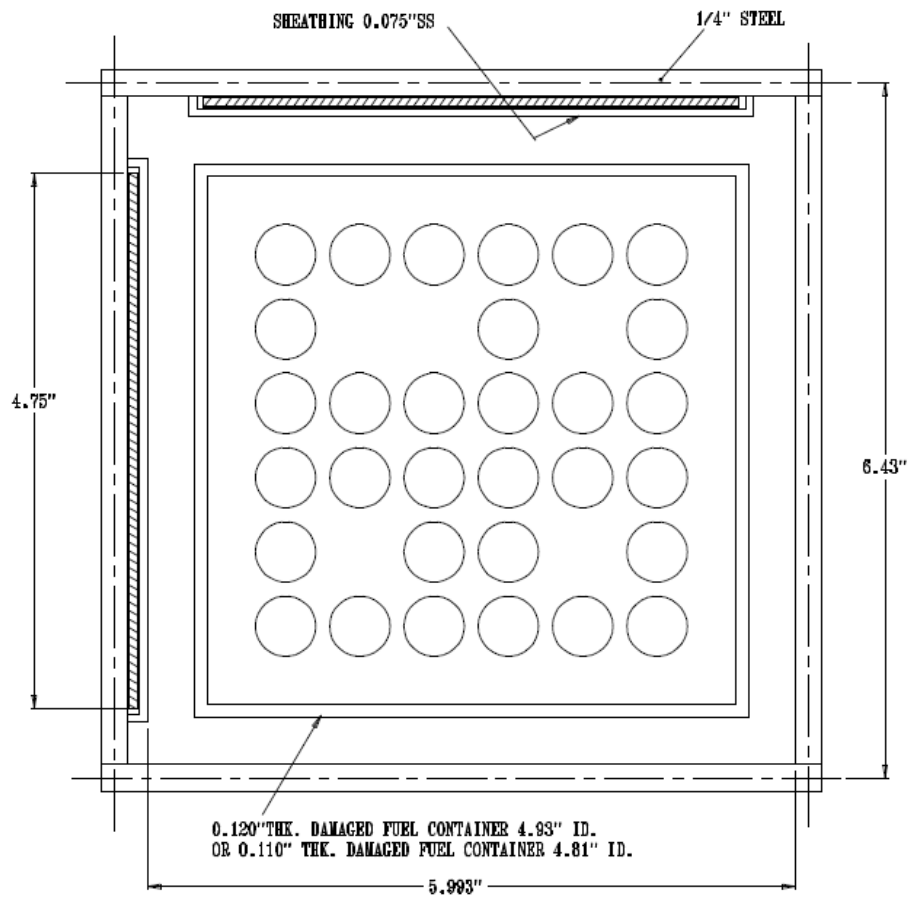
Table 6.4.15

COMPARISON OF MAXIMUM  $k_{eff}$  VALUES FOR DIFFERENT FIXED NEUTRON  
ABSORBER MATERIALS

Case	Maximum $k_{eff}$		Reactivity Difference
	BORAL	METAMIC	
MPC-68, Intact Assemblies	0.9457	0.9452	-0.0005
MPC-68, with 16 DFCs	0.9328	0.9315	-0.0013
MPC-68F with 68 DFCs	0.8021	0.8019	-0.0002
MPC-24, 0ppm	0.9478	0.9491	+0.0013
MPC-24, 400ppm	0.9447	0.9457	+0.0010
MPC-24E, Intact Assemblies, 0ppm	0.9468	0.9494	+0.0026
MPC-24E, Intact Assemblies, 300ppm	0.9399	0.9410	+0.0011
MPC-24E, with 4 DFCs, 0ppm	0.9480	0.9471	-0.0009
MPC-32, Intact Assemblies, 1900ppm	0.9411	0.9397	-0.0014
MPC-32, Intact Assemblies, 2600ppm	0.9483	0.9471	-0.0012
<b>Average Difference</b>			<b>+0.0001</b>

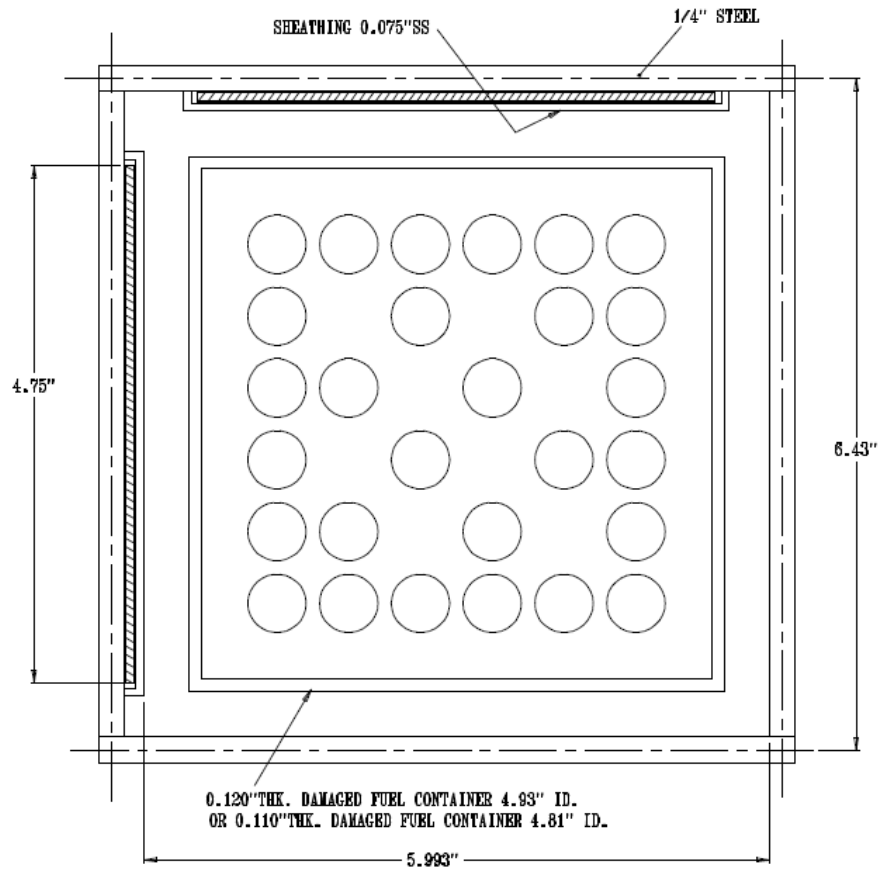
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FIGURE 6.4.1



**FIGURE 6.4.2: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 4 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

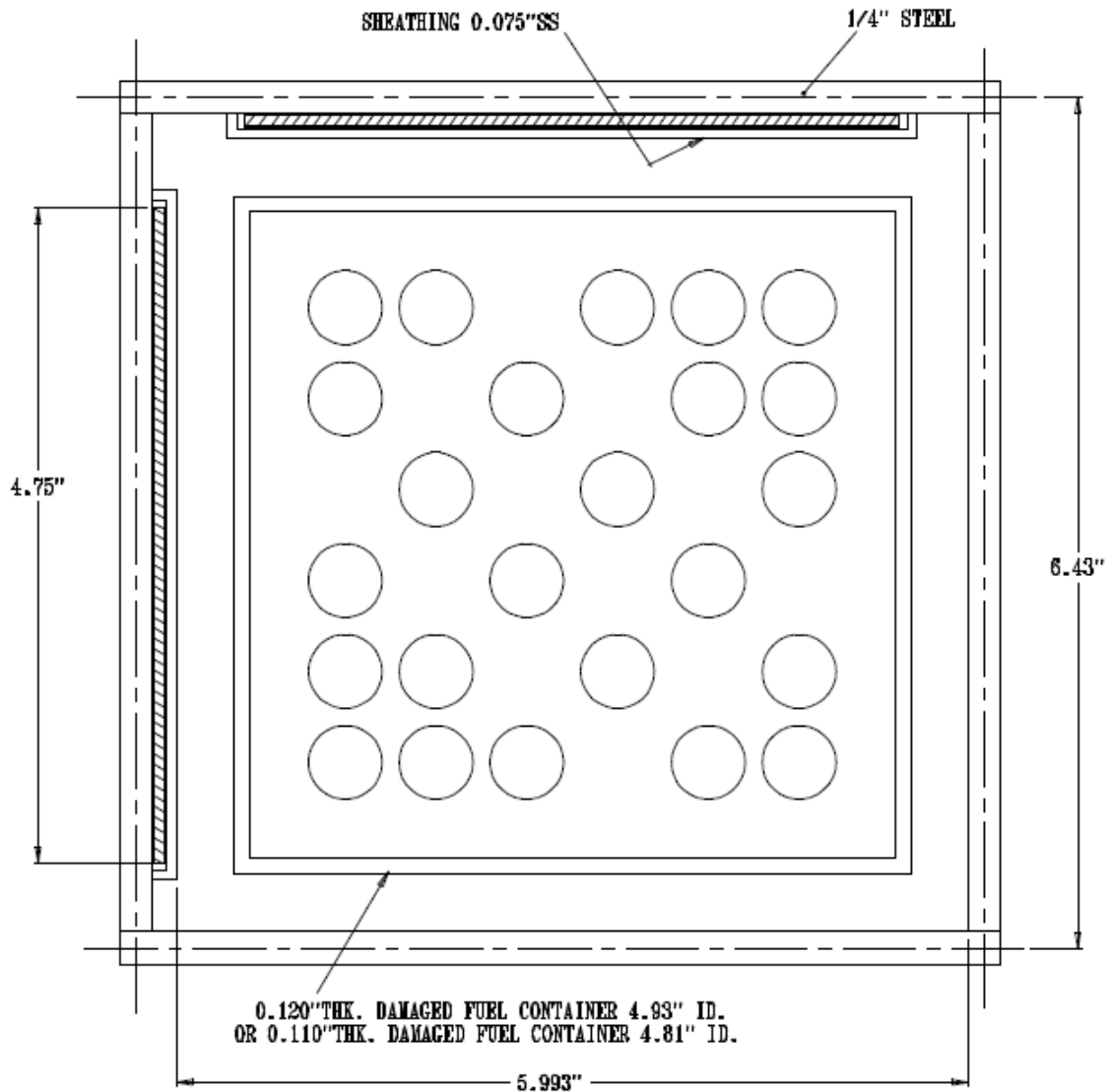
**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



**FIGURE 6.4.3: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

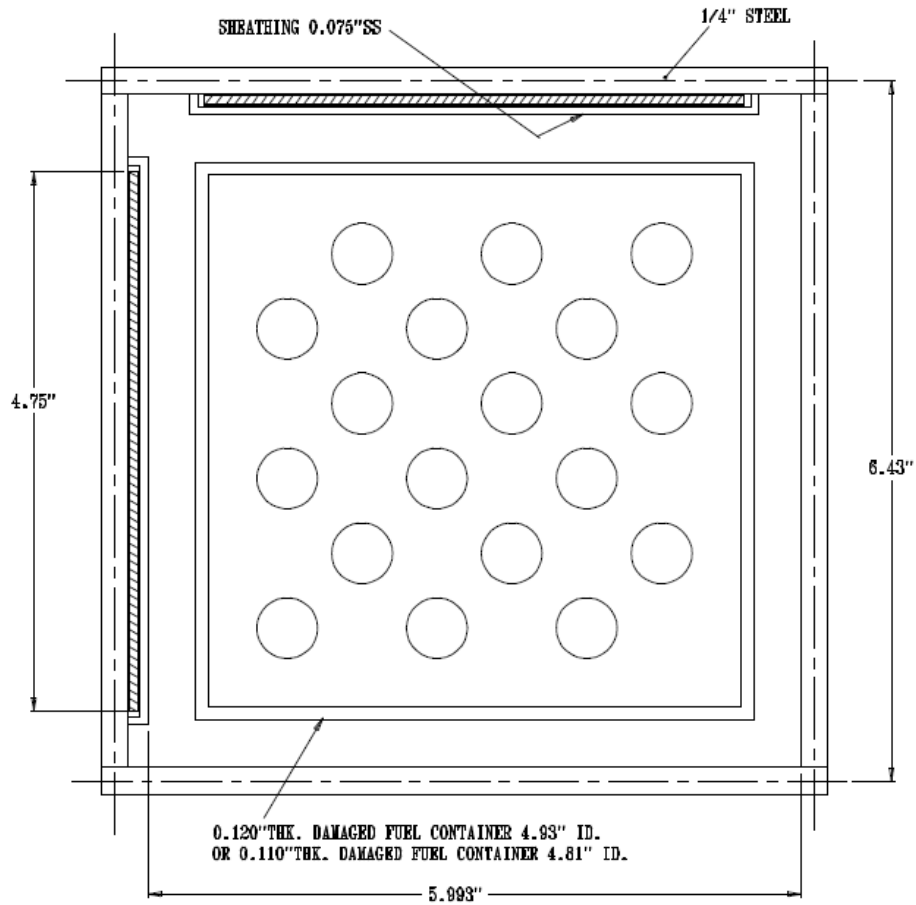
**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**





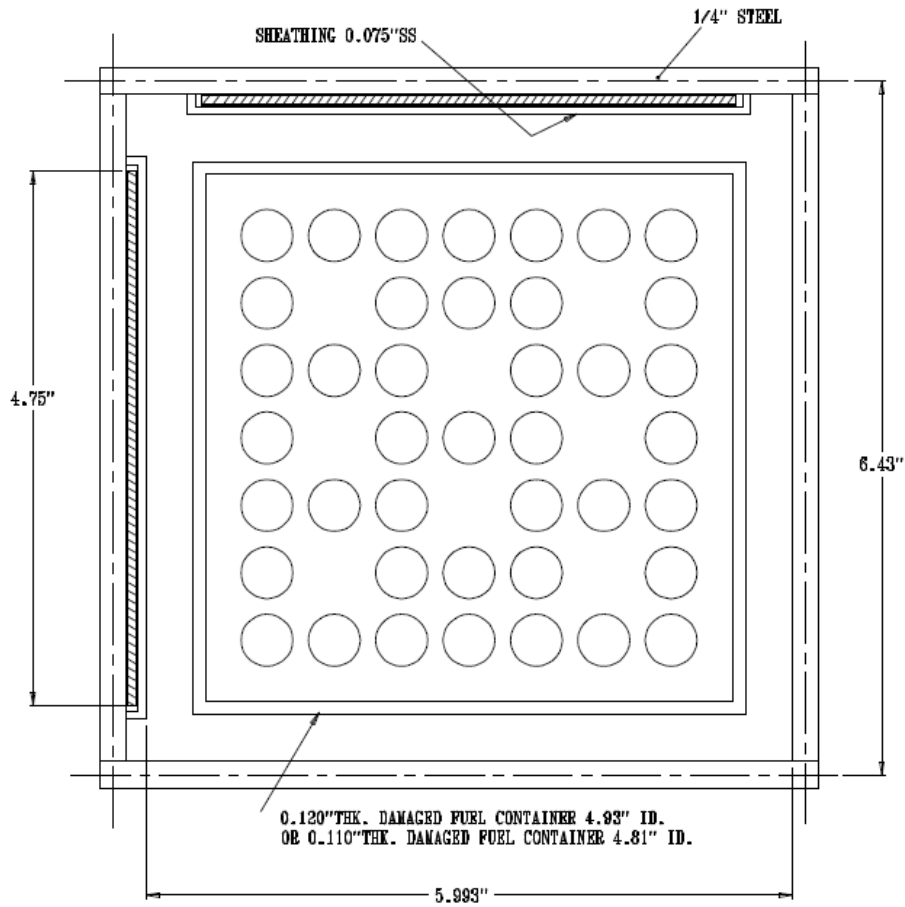
**FIGURE 6.4.4: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 12 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



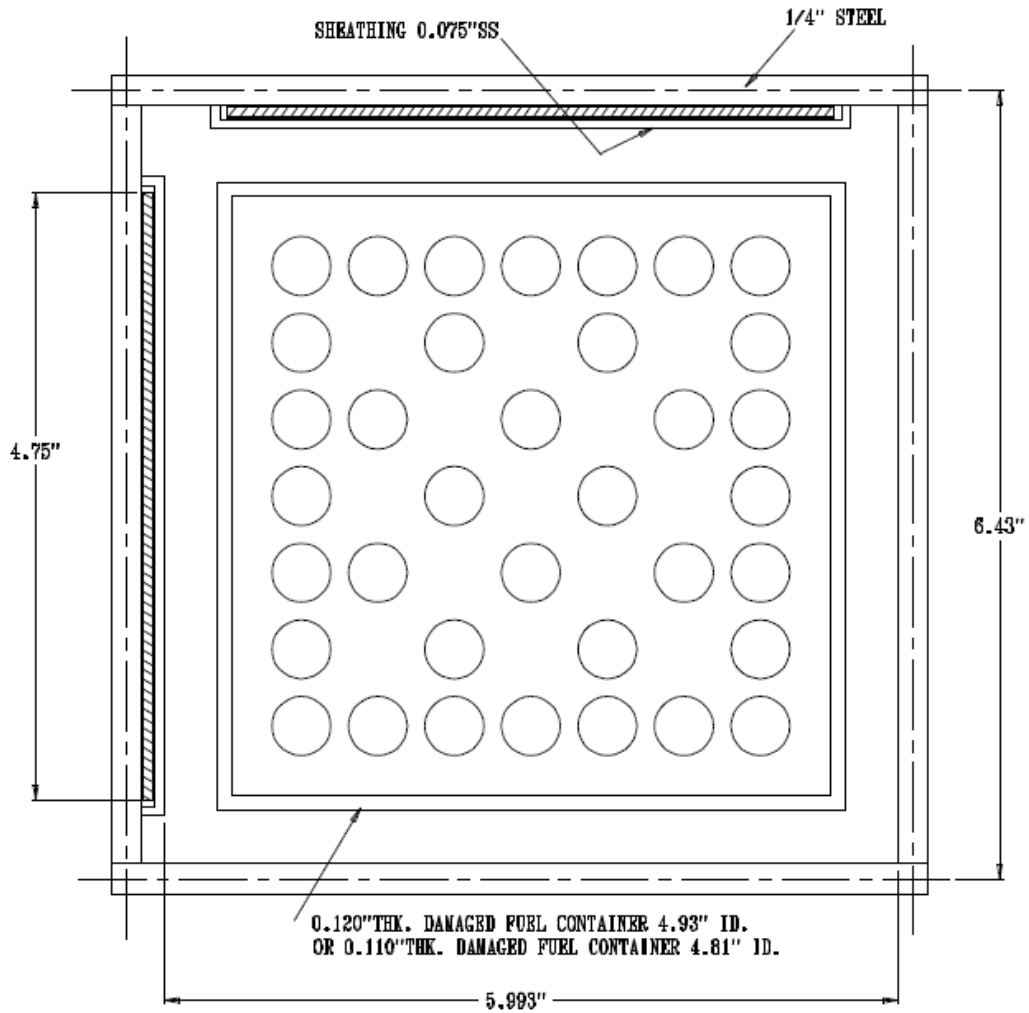
**FIGURE 6.4.5: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 18 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**

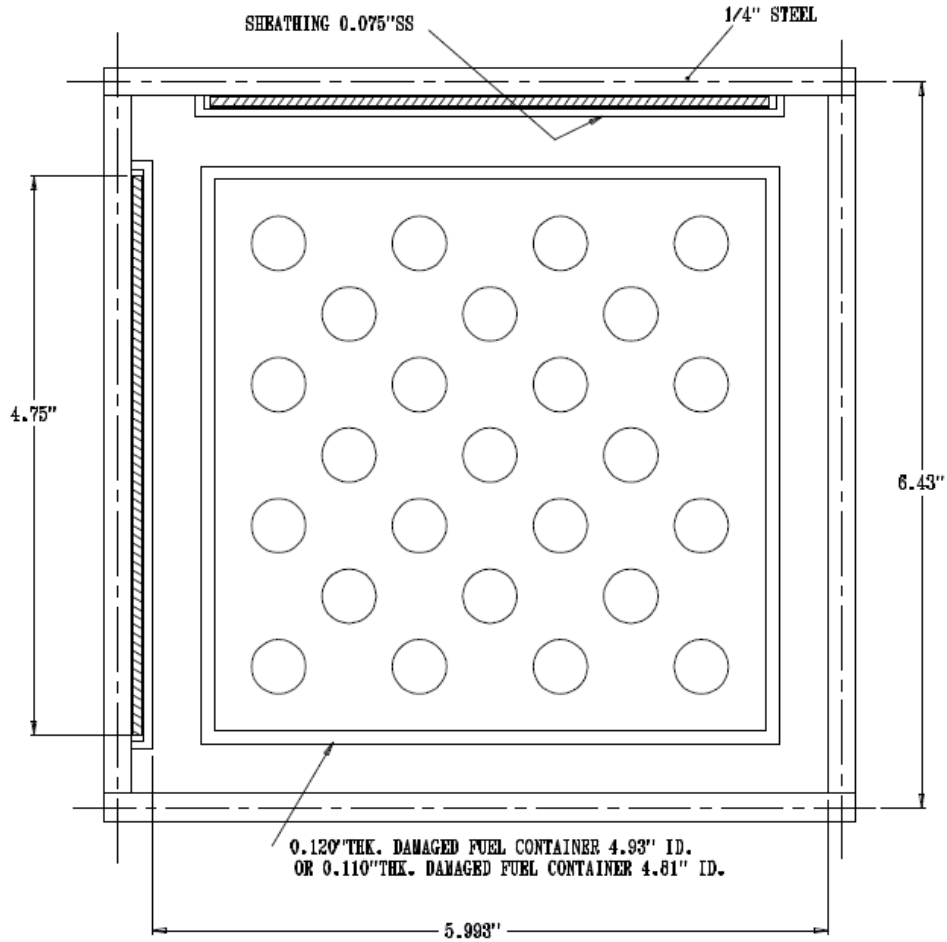


**FIGURE 6.4.6: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

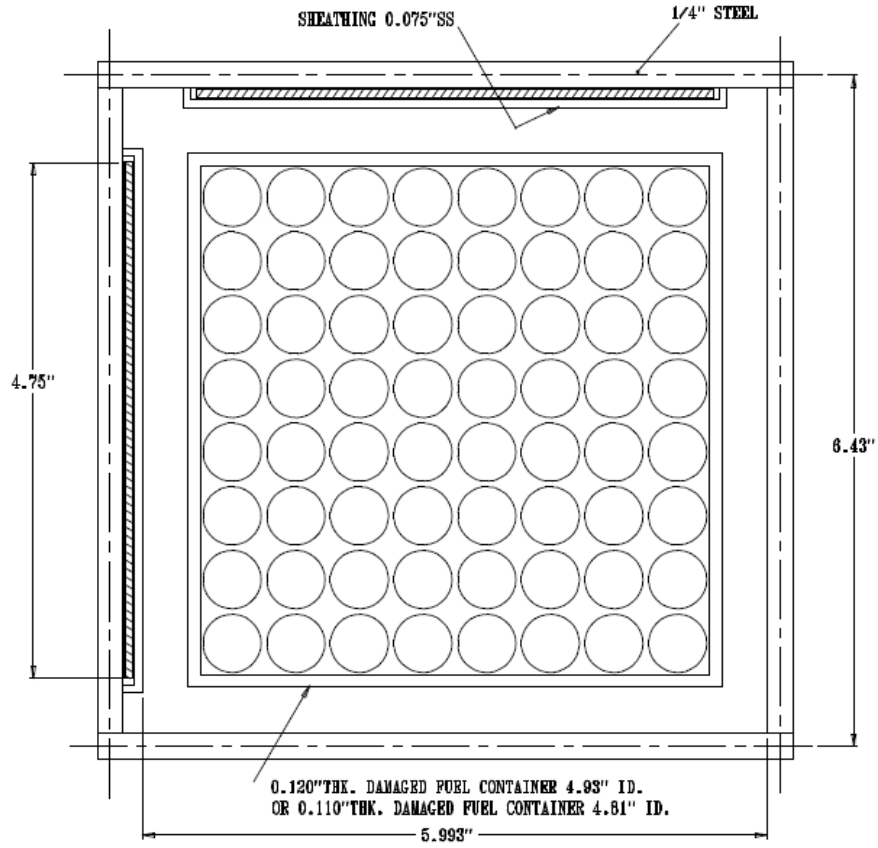
**NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.**



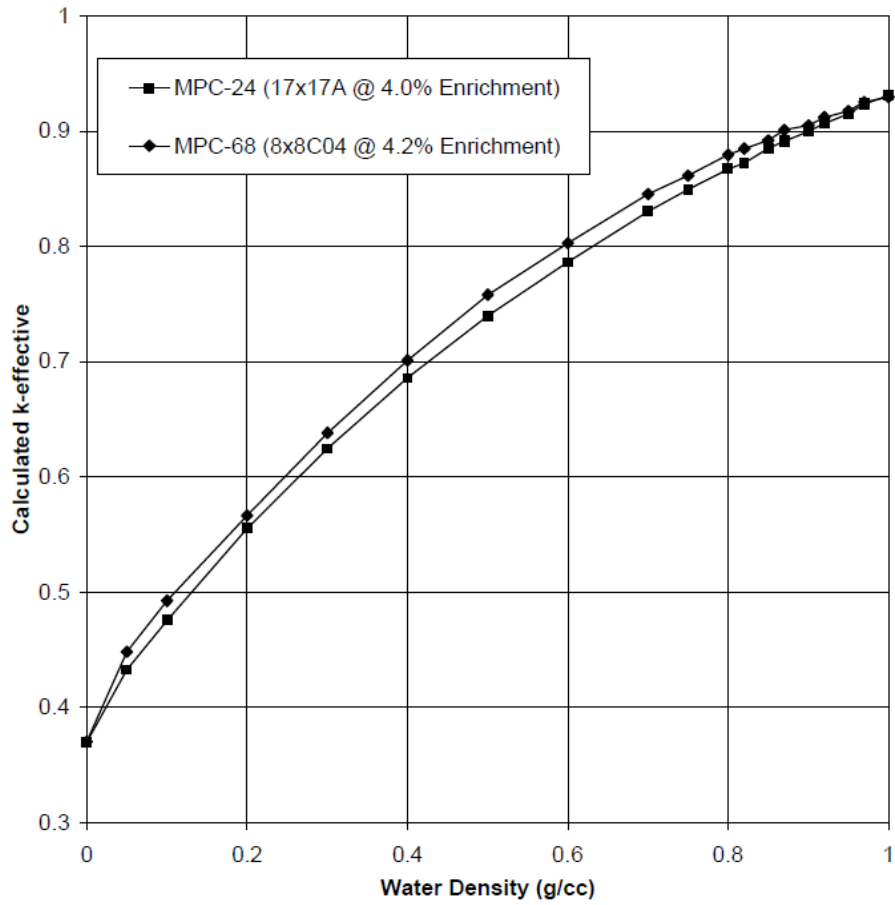
**FIGURE 6.4.7: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH 7X7 ARRAY WITH 13 MISSING RODS IN THE MPC-68 BASKET  
(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**



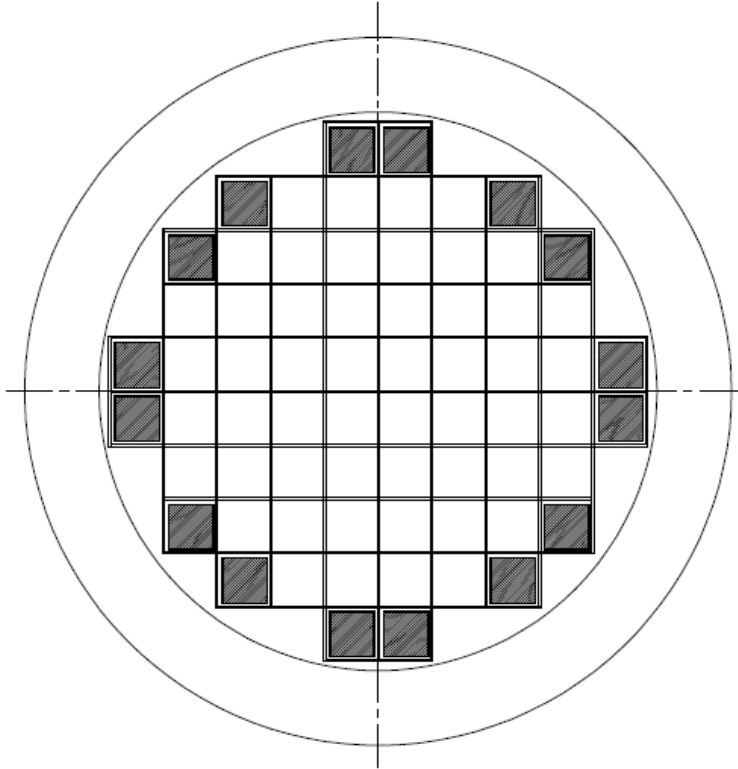
**FIGURE 6.4.8: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH 7X7 ARRAY WITH 24 MISSING RODS IN THE MPC-68 BASKET  
(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**



**FIGURE 6.4.9: FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH DAMAGED FUEL COLLAPSED INTO 8X8 ARRAY IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

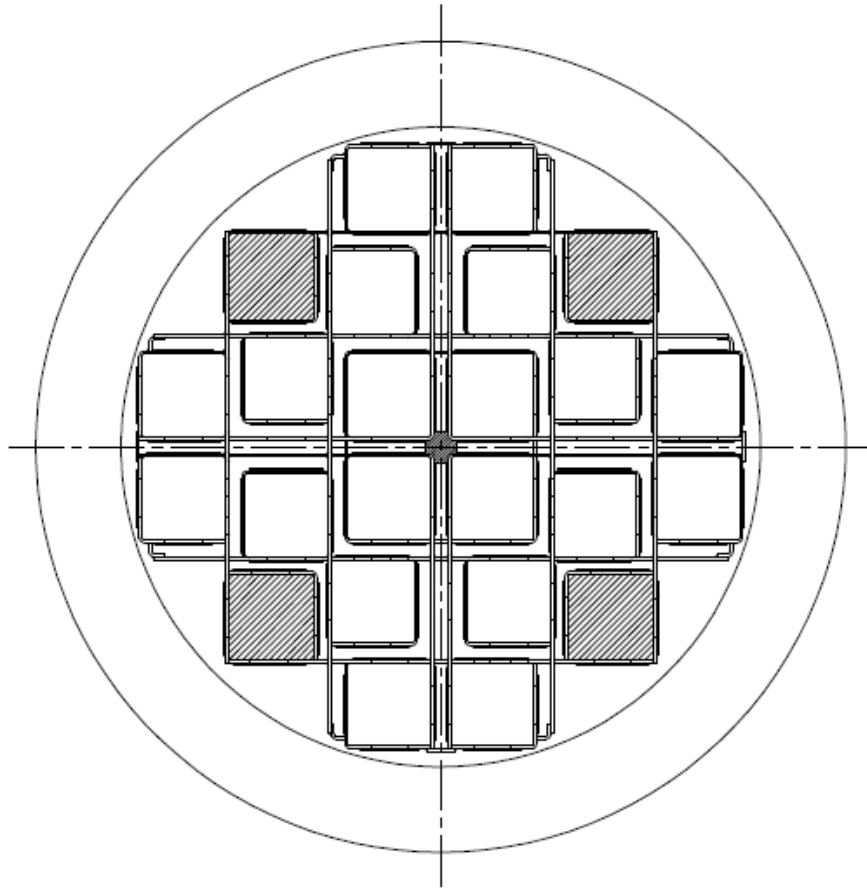


**FIGURE 6.4.10: CALCULATED K-EFFECTIVE AS A FUNCTION OF INTERNAL MODERATOR DENSITY**

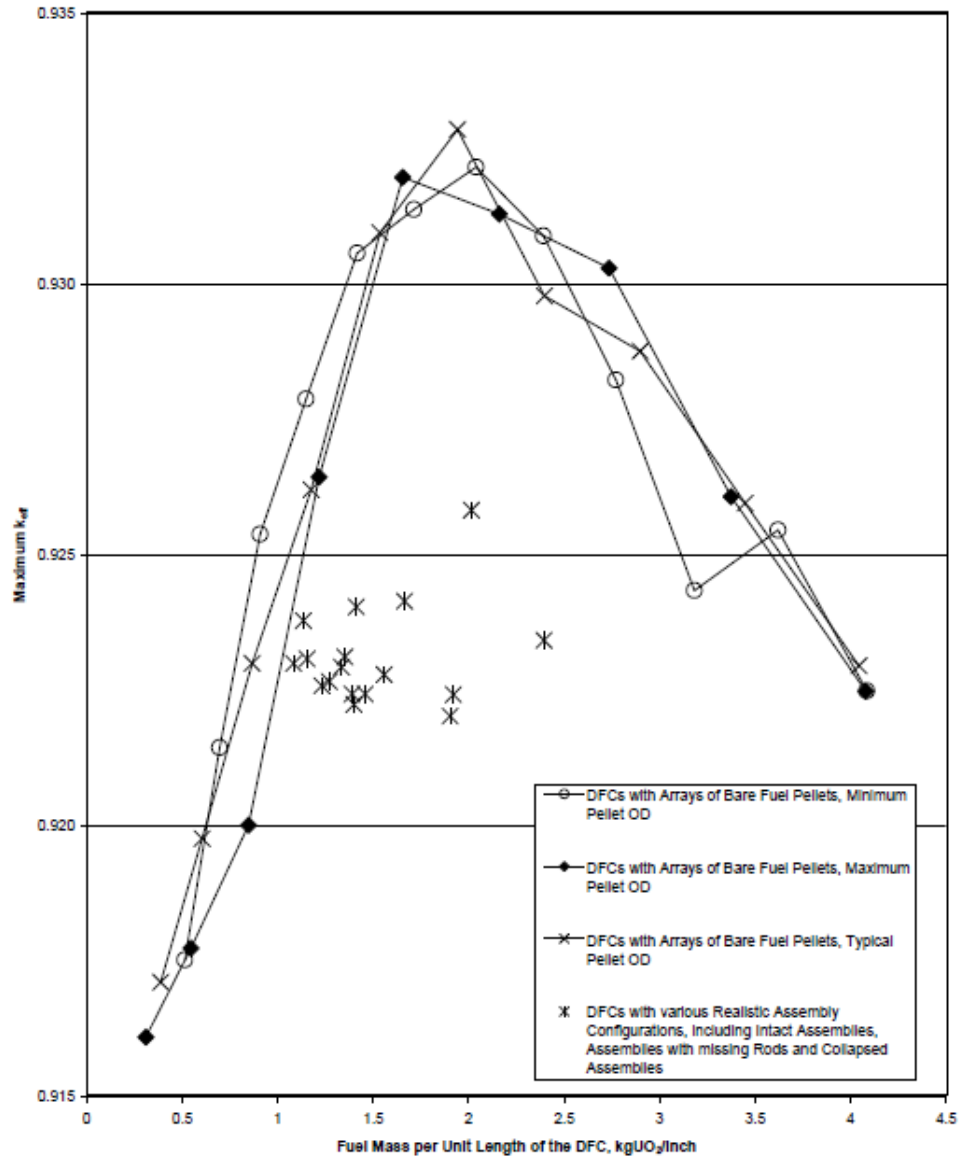


**FIGURE 6.4.11: LOCATIONS OF THE DAMAGED FUEL CONTAINER IN THE MPC-68 AND MPC-68FF**

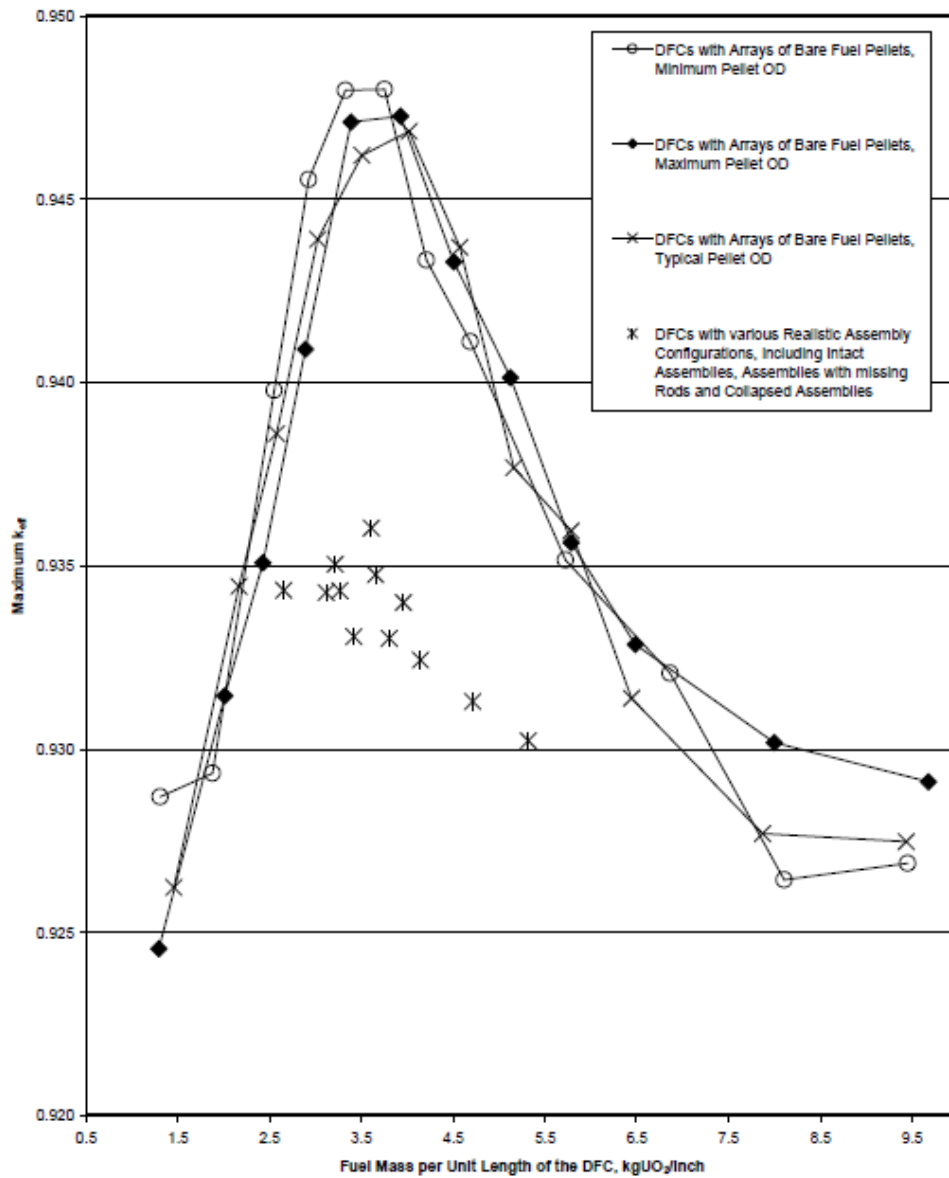




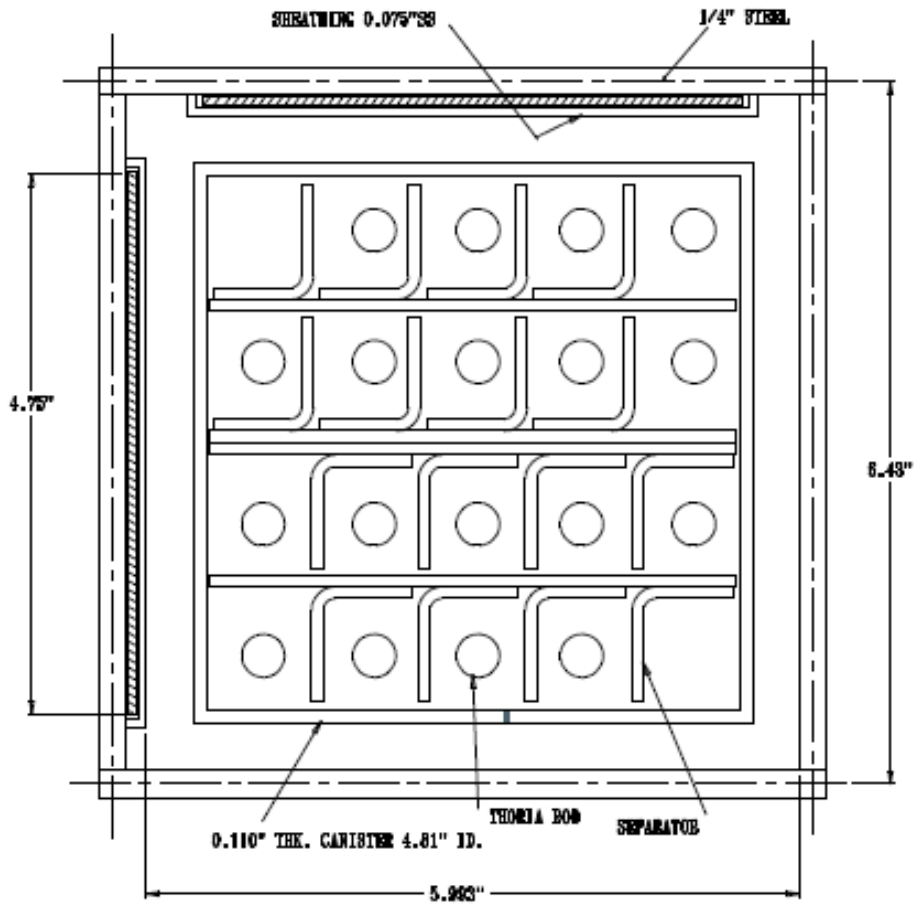
**FIGURE 6.4.12: LOCATIONS OF THE DAMAGED FUEL CONTAINERS IN THE MPC-24E**



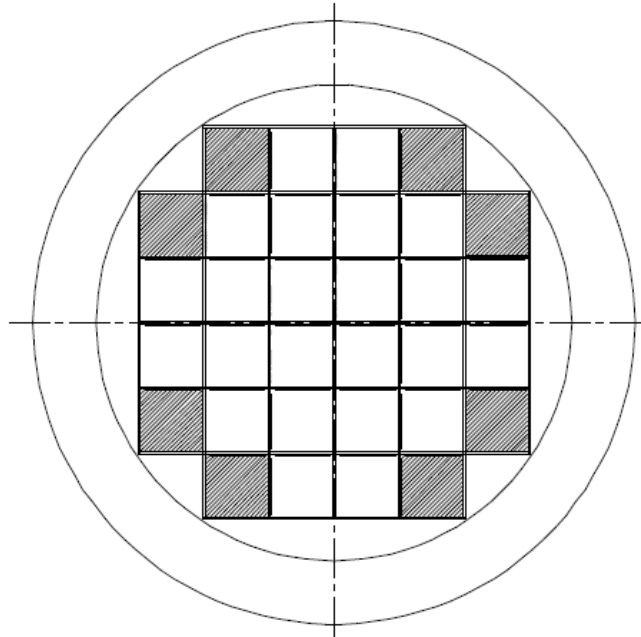
**FIGURE 6.4.13: MAXIMUM  $K_{eff}$  FOR THE MPC-68 WITH GENERIC BWR DAMAGED FUEL CONTAINER, INITIAL ENRICHMENT OF 4.0 wt% FOR DAMAGED AND 3.7 wt% FOR INTACT FUEL**



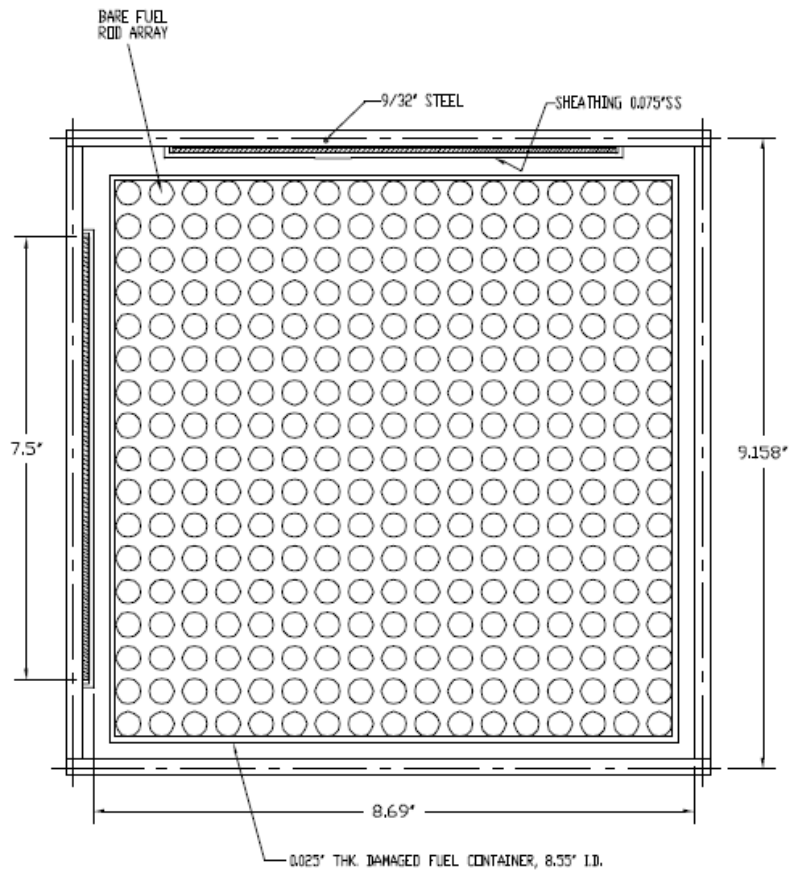
**FIGURE 6.4.14: MAXIMUM  $K_{eff}$  FOR THE MPC-24E WITH GENERIC PWR DAMAGED FUEL CONTAINER, INITIAL ENRICHMENT OF 4.0 wt% FOR DAMAGED AND INTACT FUEL**



**FIGURE 6.4.15: THORIA ROD CANISTER (PLANAR CROSS-SECTION) WITH 18 THORIA RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**



**FIGURE 6.4.16: LOCATIONS OF THE DAMAGED FUEL CONTAINERS IN THE MPC-32**



**FIGURE 6.4.17: DAMAGED FUEL/FUEL DEBRIS CALCULATION MODEL (PLANAR CROSS-SECTION) WITH BARE FUEL RODS IN THE MPC-32 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)**

## 6.5 CRITICALITY BENCHMARK EXPERIMENTS

Benchmark calculations have been made on selected critical experiments, chosen, insofar as possible, to bound the range of variables in the cask designs. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24) or cell spacing (MPC-68), and (3) the  $^{10}\text{B}$  loading of the neutron absorber panels. Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included. No significant trends were evident in the benchmark calculations or the derived bias. Detailed benchmark calculations are presented in Appendix 6.A.

The benchmark calculations were performed with the same computer codes and cross-section data, described in Section 6.4, that were used to calculate the  $k_{\text{eff}}$  values for the cask. Further, all calculations were performed on the same computer hardware, specifically, personal computers using the pentium processor.

## 6.6 REGULATORY COMPLIANCE

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STORM 100 System is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety, as well as the limiting fuel characteristics, are described in sufficient detail in this chapter to enable an evaluation of their effectiveness.

The HI-STORM 100 System is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons (Boral). An appraisal of the fixed neutron poisons has shown that they will remain effective for a storage period greater than 20 years, and there is no credible way to lose it, therefore there is no need to provide a positive means to verify their continued efficacy as required by 10CFR72.124(b).

The criticality evaluation has demonstrated that the cask will enable the storage of spent fuel for a minimum of 20 years with an adequate margin of safety. Further, the evaluation has demonstrated that the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, the HI-STORM 100 System is in full compliance with the double contingency requirements of 10CFR72.124. Therefore, it is concluded that the criticality design features for the HI-STORM 100 System are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The criticality evaluation provides reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel.



6.7

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## APPENDIX 6.A: BENCHMARK CALCULATIONS

### 6.A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the cask designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [6.A.1] is a continuous energy Monte Carlo code and KENO5a [6.A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [6.A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors<sup>†</sup> (trends) that have been reported (e.g., [6.A.3 through 6.A.5]) for calculations with collapsed cross section sets.

In cask designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the <sup>10</sup>B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included in the analyses.

Table 6.A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission". In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the energy of the average lethargy causing fission determined (post-processing).

Figures 6.A.1 and 6.A.2 show the calculated  $k_{\text{eff}}$  for the benchmark critical experiments as a function of the "energy of the average lethargy causing fission" for MCNP4a and KENO5a, respectively (UO<sub>2</sub> fuel only). The scatter in the data (even for comparatively minor variation in

<sup>†</sup> Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

critical parameters) represents experimental error<sup>†</sup> in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 6.A.1 and 6.A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a  $k_{\text{eff}}$  of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a		
	Total	Truncated
MCNP4a	0.0009 ± 0.0011	0.0021 ± 0.0006
KENO5a	0.0030 ± 0.0012	0.0036 ± 0.0009

The values of bias shown in this table include both the bias derived directly from the calculated  $k_{\text{eff}}$  values in Table 6.A.1, and a more conservative value derived by arbitrarily truncating to 1.000 any calculated value that exceeds 1.000. The bias and standard error of the bias were calculated by the following equations<sup>††</sup>, with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [6.A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i=1}^n k_i \quad (6.A.1)$$

<sup>†</sup> A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

<sup>††</sup> These equations may be found in any standard text on statistics, for example, reference [6.A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^n k_i^2 - (\sum_{i=1}^n k_i)^2 / n}{n(n-1)} \quad (6.A.2)$$

$$\text{Bias} = (1 - \bar{k}) \pm K\sigma_{\bar{k}} \quad (6.A.3)$$

where  $k_i$  are the calculated reactivities for  $n$  critical experiments;  $\sigma_{\bar{k}}$  is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); and  $K$  is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [6.A.18]).

Formula 6.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 6.A-2. The first portion of the equation,  $(1 - \bar{k})$ , is the actual bias which is added to the MCNP4a and KENO5a results. The second term,  $K\sigma_{\bar{k}}$ , which corresponds to  $\sigma_B$  in Section 6.4.3, is the uncertainty or standard error associated with the bias. The  $K$  values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual  $K$  values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The larger of the calculational biases (truncated bias) was used to evaluate the maximum  $k_{\text{eff}}$  values for the cask designs.

#### 6.A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46% to 5.74% and therefore span the enrichment range for the MPC designs. Figures 6.A.3 and 6.A.4 show the calculated  $k_{\text{eff}}$  values (Table 6.A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, the MPC-68 configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 6.A.2 and Figure 6.A.5, confirm no significant difference in the calculated values of  $k_{\text{eff}}$  for the two independent codes as evidenced by the 45° slope of the

curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

### 6.A.3 Effect of $^{10}\text{B}$ Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the cask designs. Of these critical experiments, those performed by B&W are the most representative of the cask designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 6.A.3 lists the subset of experiments using thin neutron absorbers (from Table 6.A.1) and shows the reactivity worth ( $\Delta k$ ) of the absorber.<sup>†</sup>

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 6.A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with  $^{10}\text{B}$  concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 6.A.6 and Table 6.A.4 for the MPC-68 cask<sup>††</sup> geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

<sup>†</sup> The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental ( $\Delta k$ ) change in reactivity due to the absorber.

<sup>††</sup> The MPC-68 geometry was chosen for this comparison since it contains the greater number of Boral panels and would therefore be expected to be the most sensitive to trends (errors) in calculations.

## 6.A.4 Miscellaneous and Minor Parameters

### 6.A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.<sup>†</sup> Analysis of these critical experiments are listed in Table 6.A.5 (subset of data in Table 6.A.1). There appears to be a small tendency toward overprediction of  $k_{\text{eff}}$  at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the cask calculations may be slightly more conservative than otherwise.

### 6.A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the cask designs, the fuel pellet diameters range from 0.303 to 0.3835 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.498 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of the fuel in the MPC designs. Based on the data in Table 6.A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments or the cask designs.

### 6.A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 6.A.6. Analyses of the very high boron concentration experiments (>1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the MPC-32 with various soluble boron concentration could be slightly conservative for the high soluble boron concentration.

## 6.A.5 MOX Fuel

<sup>†</sup>Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec cask design. A lead reflector is also not directly pertinent, but might be used in future designs.

The number of critical experiments with PuO<sub>2</sub> bearing fuel (MOX) is more limited than for UO<sub>2</sub> fuel. However, a number of MOX critical experiments have been analyzed and the results are shown in Table 6.A.7. Results of these analyses are generally above a  $k_{\text{eff}}$  of 1.00, indicating that when Pu is present, MCNP4a and KENO5a overpredict the reactivity.

This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggested that a small trend may exist with KENO5a. It is also possible that the overprediction in  $k_{\text{eff}}$  in both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated  $k_{\text{eff}}$  over a wide range of the spectral index (energy of the average lethargy causing fission).

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**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

Reference	Identification	Enrich.	Calculated $k_{eff}$		EALF (eV)		
			MCNP4a	KENO5a	MCNP4a	KENO5a	
1	B&W-1484 (6.A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898 ± 0.0006	0.1759	0.1753
2	B&W-1484 (6.A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (6.A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (6.A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (6.A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (6.A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (6.A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (6.A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (6.A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (6.A.7)	Core XV <sup>††</sup>	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (6.A.7)	Core XVI <sup>††</sup>	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (6.A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

Reference	Identification	Enrich.	Calculated $k_{eff}$		EALF (eV)		
			MCNP4a	KENO5a	MCNP4a	KENO5a	
13	B&W-1484 (6.A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708
14	B&W-1484 (6.A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (6.A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (6.A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (6.A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (6.A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (6.A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (6.A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (6.A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (6.A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (6.A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (6.A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

Reference	Identification	Enrich.	Calculated $k_{eff}$		EALF (eV)		
			MCNP4a	KENO5a	MCNP4a	KENO5a	
25	French (6.A.10)	Water Moderator 10 cm gap	4.75	$0.9979 \pm 0.0010$	NC	0.1736	NC
26	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	2.35	NC	$1.0004 \pm 0.0006$	NC	0.1018
27	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	2.35	$0.9980 \pm 0.0009$	$0.9992 \pm 0.0006$	0.1000	0.0909
28	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	2.35	$0.9968 \pm 0.0009$	$0.9964 \pm 0.0006$	0.0981	0.0975
29	PNL-3602 (6.A.11)	Steel Reflector, 3.912 cm separation	2.35	$0.9974 \pm 0.0010$	$0.9980 \pm 0.0006$	0.0976	0.0970
30	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	2.35	$0.9962 \pm 0.0008$	$0.9939 \pm 0.0006$	0.0973	0.0968
31	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	4.306	NC	$1.0003 \pm 0.0007$	NC	0.3282
32	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	4.306	$0.9997 \pm 0.0010$	$1.0012 \pm 0.0007$	0.3016	0.3039
33	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	4.306	$0.9994 \pm 0.0012$	$0.9974 \pm 0.0007$	0.2911	0.2927
34	PNL-3602 (6.A.11)	Steel Reflector, 5.405 cm separation	4.306	$0.9969 \pm 0.0011$	$0.9951 \pm 0.0007$	0.2828	0.2860
35	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	4.306	$0.9910 \pm 0.0020$	$0.9947 \pm 0.0007$	0.2851	0.2864
36	PNL-3602 (6.A.11)	Steel Reflector, with Boral Sheets	4.306	$0.9941 \pm 0.0011$	$0.9970 \pm 0.0007$	0.3135	0.3150

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

Reference	Identification	Enrich.	<u>Calculated <math>k_{eff}</math></u>		<u>EALF (eV)</u>		
			MCNP4a	KENO5a	MCNP4a	KENO5a	
37	PNL-3626 (6.A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3626 (6.A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3626 (6.A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930
40	PNL-3626 (6.A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (6.A.13)	Experiment 004/032 – no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (6.A.13)	Experiment 030 – Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (6.A.13)	Experiment 013 – Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (6.A.13)	Experiment 014 – Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (6.A.13)	Exp. 009 1.05% Boron Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (6.A.13)	Exp. 009 1.62% Boron Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (6.A.13)	Exp. 031 – Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (6.A.14)	Experiment 214R – with flux traps	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

Reference	Identification	Enrich.	Calculated $k_{eff}$		EALF (eV)		
			MCNP4a	KENO5a	MCNP4a	KENO5a	
49	PNL-7167 (6.A.14)	Experiment 214V3 –with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (6.A.15)	Case 173 – 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (6.A.15)	Case 177 – 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868
53	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1339	0.1165
56	WCAP-3385 (6.A.17)	Saxton Case 52 PuO <sub>2</sub> 0.52” pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (6.A.17)	Saxton Case 52 U 0.52” pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (6.A.17)	Saxton Case 56 PuO <sub>2</sub> 0.56” pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (6.A.17)	Saxton Case 56 borated PuO <sub>2</sub>	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (6.A.17)	Saxton Case 56 U 0.56” pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	<u>Calculated <math>k_{eff}</math></u>		<u>EALF (eV)</u>	
				MCNP4a	KENO5a	MCNP4a	KENO5a
61	WCAP-3385 (6.A.17)	Saxton Case 79 PuO <sub>2</sub> 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (6.A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission

†† The experimental results appear to be statistical outliers ( $>3\sigma$ ) suggesting the possibility of unusually large experimental error. Although they could be justifiably excluded, for conservatism, they were retained in determining the calculational basis.



Table 6.A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES<sup>†</sup>  
FOR VARIOUS ENRICHMENTS (UO<sub>2</sub>)

Enrichment	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	KENO5a
3.0	0.8465 $\pm$ 0.0011	0.8478 $\pm$ 0.0004
3.5	0.8820 $\pm$ 0.0011	0.8841 $\pm$ 0.0004
3.75	0.9019 $\pm$ 0.0011	0.8987 $\pm$ 0.0004
4.0	0.9132 $\pm$ 0.0010	0.9140 $\pm$ 0.0004
4.2	0.9276 $\pm$ 0.0011	0.9237 $\pm$ 0.0004
4.5	0.9400 $\pm$ 0.0011	0.9388 $\pm$ 0.0004

<sup>†</sup> Based on the MPC-68 with the GE 8x8R

Table 6.A.3

MCNP4a CALCULATED REACTIVITIES FOR  
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS (UO<sub>2</sub>)

<b>Ref.</b>	<b>Experiment</b>		<b><math>\Delta k</math> Worth of Absorber</b>	<b>MCNP4a Calculated <math>k_{eff}</math></b>	<b>EALF<sup>†</sup> (eV)</b>
6.A.13	PNL-2615	Boral Sheet	0.0139	$0.9994 \pm 0.0012$	0.1165
6.A.7	BAW-1484	Core XX	0.0165	$1.0008 \pm 0.0011$	0.1724
6.A.13	PNL-2615	1.62% Boron-steel	0.0165	$0.9996 \pm 0.0012$	0.1161
6.A.7	BAW-1484	Core XIX	0.0202	$0.9961 \pm 0.0012$	0.2103
6.A.7	BAW-1484	Core XXI	0.0243	$0.9994 \pm 0.0010$	0.1544
6.A.7	BAW-1484	Core XVII	0.0519	$0.9962 \pm 0.0012$	0.2083
6.A.11	PNL-3602	Boral Sheet	0.0708	$0.9941 \pm 0.0011$	0.3135
6.A.7	BAW-1484	Core XV	0.0786	$0.9910 \pm 0.0011$	0.2092
6.A.7	BAW-1484	Core XVI	0.0845	$0.9935 \pm 0.0010$	0.1757
6.A.7	BAW-1484	Core XIV	0.1575	$0.9953 \pm 0.0011$	0.2022
6.A.7	BAW-1484	Core XIII	0.1738	$1.0020 \pm 0.0011$	0.1988
6.A.14	PNL-7167	Expt 214R flux trap	0.1931	$0.9991 \pm 0.0011$	0.3722

<sup>†</sup> EALF is the energy of the average lethargy causing fission

Table 6.A.4  
 COMPARISON OF MCNP4a AND KENO5a  
 CALCULATED REACTIVITIES<sup>†</sup> FOR VARIOUS BORON LOADINGS (UO<sub>2</sub>)

<sup>10</sup> B, g/cm <sup>2</sup>	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	KENO5a
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

<sup>†</sup> based on 4.5% enrichment GE 8x8R in the MPC-68 cask.

Table 6.A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH  
THICK LEAD AND STEEL REFLECTORS<sup>†</sup> (UO<sub>2</sub>)

Ref.	Case	Enrichment, wt%	Separation, cm	MCNP4a $k_{eff}$	KENO5a $k_{eff}$
6.A.11	Steel Reflector	2.35	1.321	0.9980 ± 0.0009	0.9992 ± 0.0006
		2.35	2.616	0.9968 ± 0.0009	0.9964 ± 0.0006
		2.35	3.912	0.9974 ± 0.0010	0.9980 ± 0.0006
		2.35	∞	0.9962 ± 0.0008	0.9939 ± 0.0006
6.A.11	Steel Reflector	4.306	1.321	0.9997 ± 0.0010	1.0012 ± 0.0007
		4.306	2.616	0.9994 ± 0.0012	0.9974 ± 0.0007
		4.306	3.405	0.9969 ± 0.0011	0.9951 ± 0.0007
		4.306	∞	0.9910 ± 0.0020	0.9947 ± 0.0007
6.A.11	Lead Reflector	4.306	0.55	1.0025 ± 0.0011	0.9997 ± 0.0007
		4.306	1.956	1.0000 ± 0.0012	0.9985 ± 0.0007
		4.306	5.405	0.9971 ± 0.0012	0.9946 ± 0.0007

<sup>†</sup> Arranged in order of increasing reflector fuel spacing.

Table 6.A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE  
BORON CONCENTRATIONS (UO<sub>2</sub>)

Reference	Experiment	Boron Concentration ppm	Calculated $k_{eff}$	
			MCNP4a	KENO5a
6.A.15	PNL-4267	0	$0.9974 \pm 0.0012$	-
6.A.8	BAW-1645-4	886	$0.9970 \pm 0.0010$	$0.9924 \pm 0.0006$
6.A.9	BAW-1810	1337	$1.0023 \pm 0.0010$	-
6.A.9	BAW-1810	1899	$1.0060 \pm 0.0009$	-
6.A.15	PNL-4267	2550	$1.0057 \pm 0.0010$	-

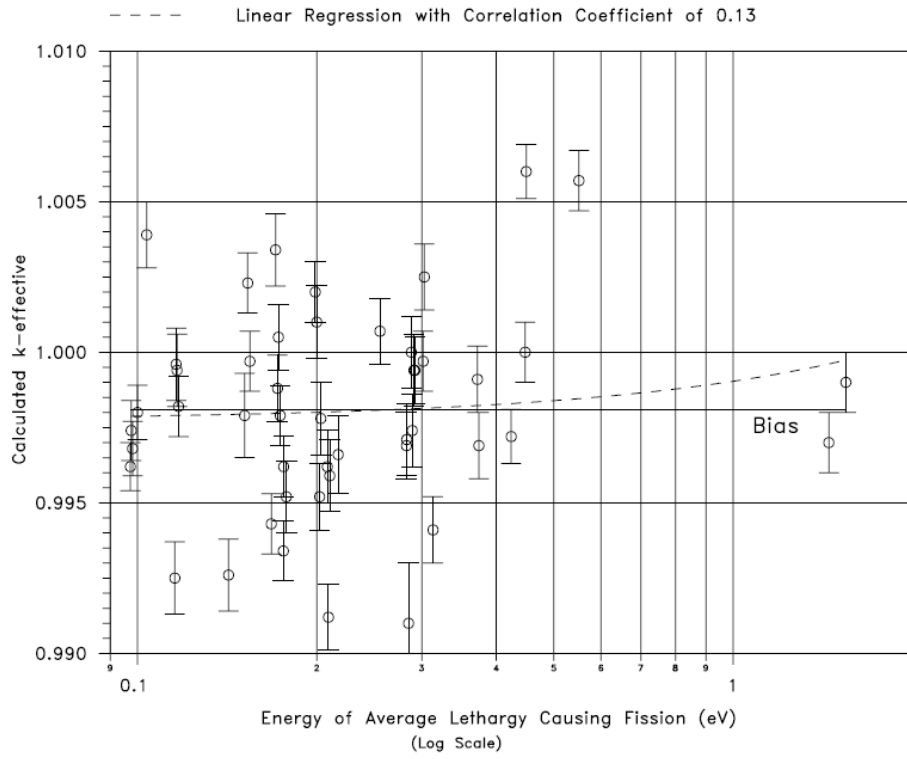
Table 6.A.7

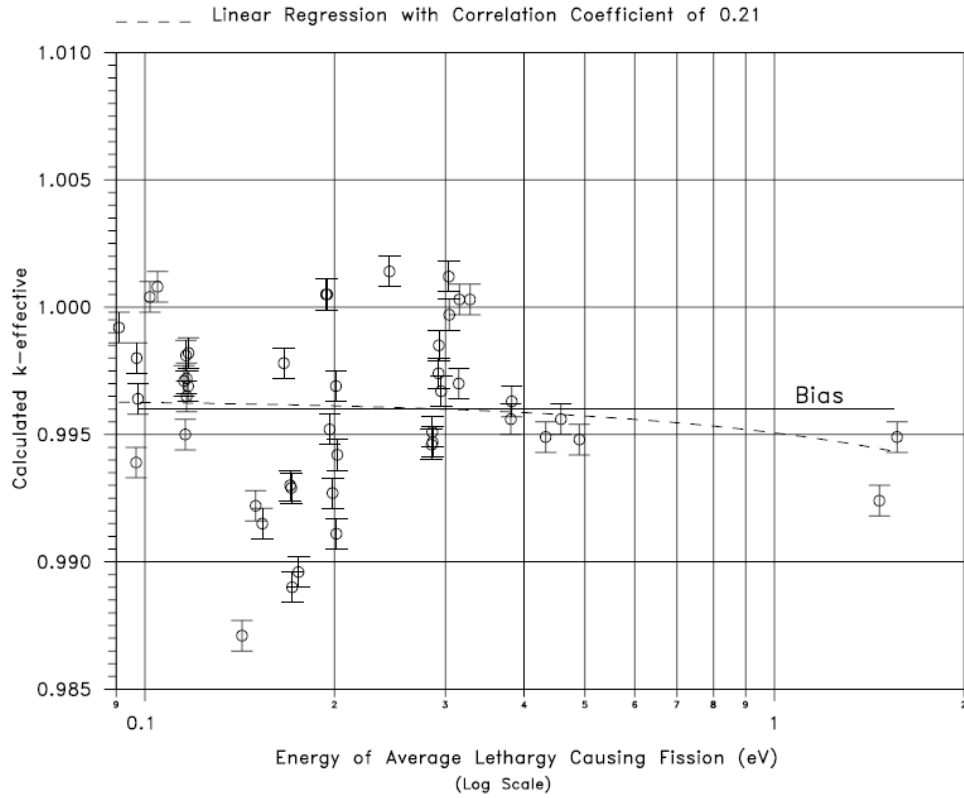
## CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case <sup>†</sup>	MCNP4a		KENO 5a	
		$k_{\text{eff}}$	EALF <sup>††</sup> (eV)	$k_{\text{eff}}$	EALF <sup>††</sup> (eV)
PNL-5803 [6.A.16]	MOX Fuel – Exp No 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel – Exp No 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel – Exp No 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel – Exp No 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP- 3385- 54 [6.A.17]	Saxton @ 0.52” pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56” pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56” pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79” pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

<sup>†</sup> Arranged in order of increasing lattice spacing.

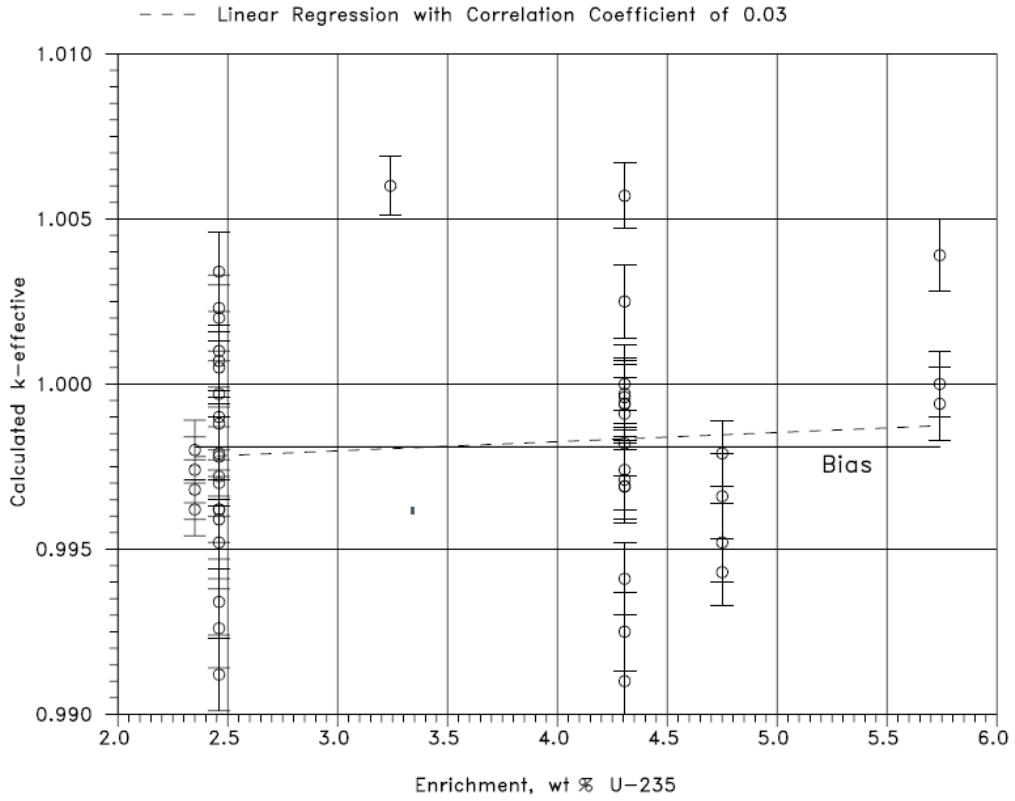
<sup>††</sup> EALF is the energy of the average lethargy causing fission.



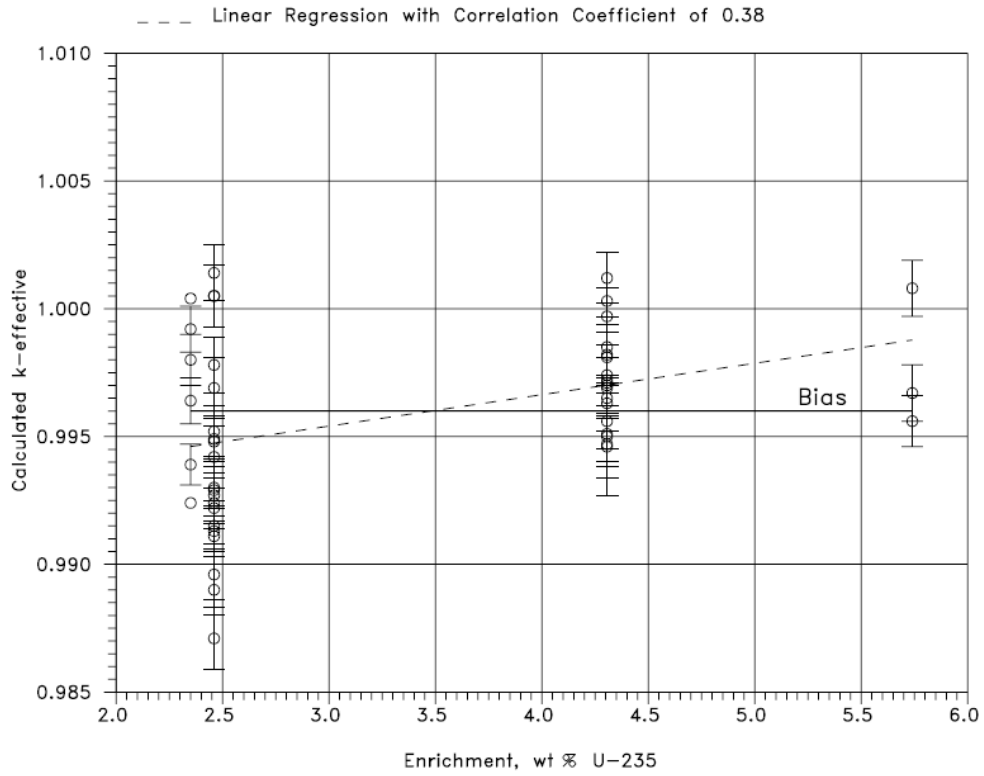


**FIGURE 6.A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX**

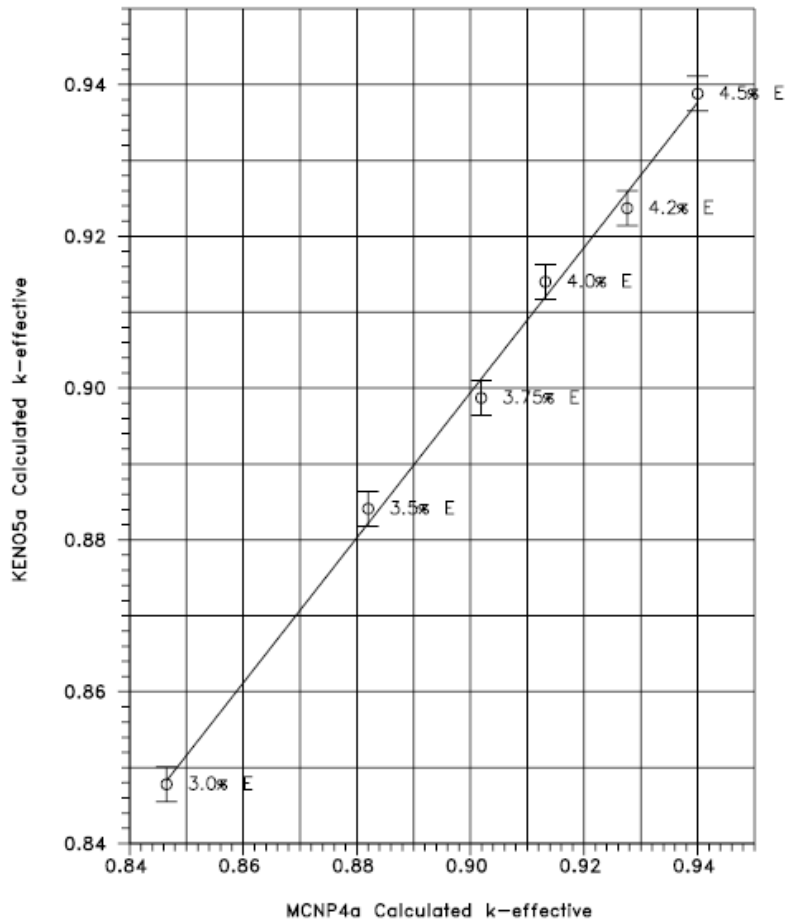




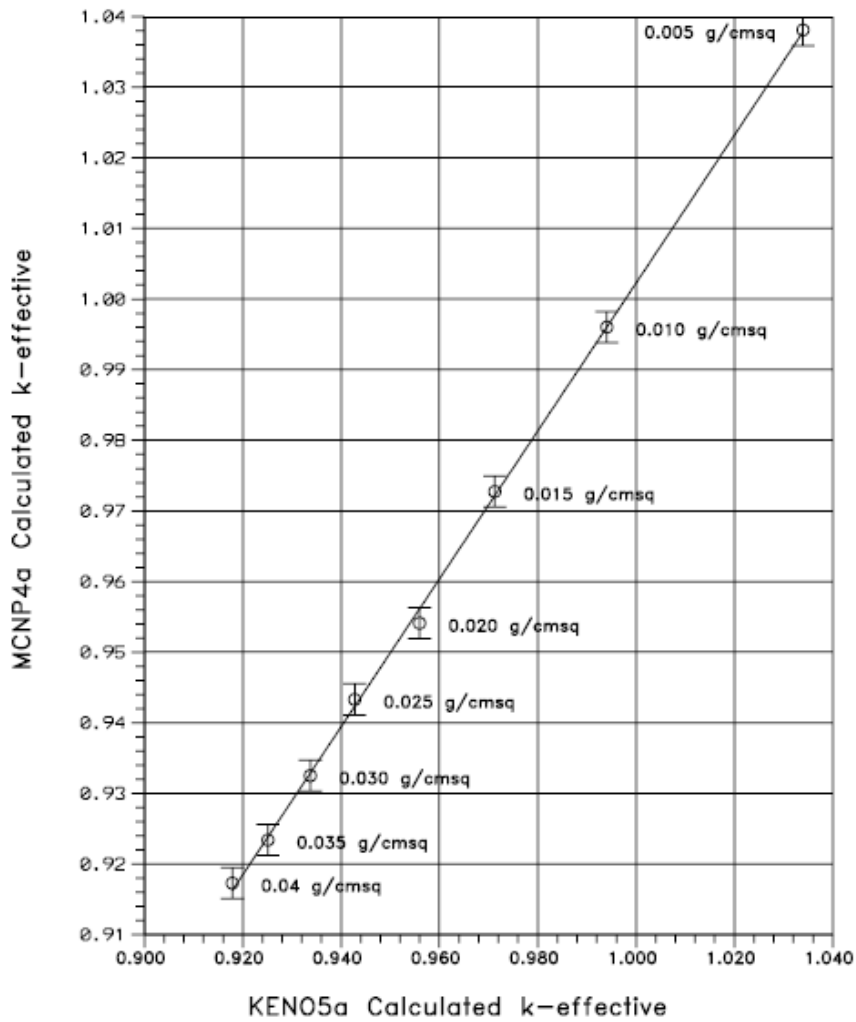
**FIGURE 6.A.3 MCNP4a CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS**



**FIGURE 6.A.4 KENO5a CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS**



**FIGURE 6.A.5 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS**



**FIGURE 6.A.6 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES**

## APPENDIX 6.B: DISTRIBUTED ENRICHMENTS IN BWR FUEL

Fuel assemblies used in BWRs utilize fuel rods of varying enrichments as a means of controlling power peaking during in-core operation. For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations have been performed to confirm that this statement remains valid in the geometry of the MPC-68. These calculations are based on fuel assembly designs currently in use and two hypothetical distributions, all intended to illustrate that calculations with uniform average enrichments are conservative.

The average enrichment is calculated as the linear average of the various fuel rod enrichments, i.e.,

$$\bar{E} = \frac{1}{n} \sum_{i=1}^n E_i,$$

where  $E_i$  is the enrichment in each of the  $n$  rods, and  $\bar{E}$  is the assembly average enrichment. This parameter conservatively characterizes the fuel assembly and is readily available for specific fuel assemblies in determining the acceptability of the assembly for placement in the MPC-68 cask.

The criticality calculations for average and distributed enrichment cases are compared in Table 6.B.1 to illustrate and confirm the conservatism inherent in using average enrichments. With two exceptions, the cases analyzed represent realistic designs currently in use and encompass fuel with different ratios of maximum pin enrichment to average assembly enrichment. The two exceptions are hypothetical cases intended to extend the models to higher enrichments and to demonstrate that using the average enrichment remains conservative.

Table 6.B.1 shows that, in all cases, the averaged enrichment yields conservative values of reactivity relative to distributed enrichments for both the actual fuel designs and the hypothetical higher enrichment cases. Thus, it is concluded that uniform average enrichments will always yield higher (more conservative) values for reactivity than the corresponding distributed enrichments.<sup>†</sup>

<sup>†</sup> This conclusion implicitly assumes the higher enrichment fuel rods are located internal to the assembly (as in BWR fuel), and the lower enriched rods are on the outside.

Table 6.B.1

**COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND  
DISTRIBUTED ENRICHMENTS**

<b>Case</b>	<b>Average %E</b>	<b>Peak Rod E%</b>	<b>Calculated <math>k_{eff}</math></b>	
			<b>Average E</b>	<b>Distributed E</b>
8x8C04	3.01	3.80	0.8549	0.8429
8x8C04	3.934	4.9	0.9128	0.9029
8x8D05	3.42	3.95	0.8790	0.8708
8x8D05	3.78	4.40	0.9030	0.8974
8x8D05	3.90	4.90	0.9062	0.9042
9x9B01	4.34	4.71	0.9347	0.9285
9x9D01	3.35	4.34	0.8793	0.8583
Hypothetical #1 (48 outer rods of 3.967%E, 14 inner rods of 5.0%)	4.20	5.00	0.9289	0.9151
Hypothetical #2 (48 outer rods of 4.354%E, 14 inner rods of 5.0%)	4.50	5.00	0.9422	0.9384

## APPENDIX 6.C: CALCULATIONAL SUMMARY

The following table lists the maximum  $k_{\text{eff}}$  (including bias, uncertainties, and calculational statistics), MCNP calculated  $k_{\text{eff}}$ , standard deviation, and energy of average lethargy causing fission (EALF) for each of the candidate fuel types and basket configurations.

Table 6.C.1  
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
AND BASKET CONFIGURATIONS

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{\text{eff}}</math></b>	<b>Calculated <math>k_{\text{eff}}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A01	HI-STAR	0.9295	0.9252	0.0008	0.2084
14x14A02	HI-STAR	0.9286	0.9242	0.0008	0.2096
14x14A03	HI-STORM	0.3080	0.3047	0.0003	3.37E+04
14x14A03	HI-TRAC	0.9283	0.9239	0.0008	0.2096
14x14A03	HI-STAR	0.9296	0.9253	0.0008	0.2093
14x14B01	HI-STAR	0.9159	0.9117	0.0007	0.2727
14x14B02	HI-STAR	0.9169	0.9126	0.0008	0.2345
14x14B03	HI-STAR	0.9110	0.9065	0.0009	0.2545
14x14B04	HI-STAR	0.9084	0.9039	0.0009	0.2563
B14x14B01	HI-TRAC	0.9237	0.9193	0.0008	0.2669
B14x14B01	HI-STAR	0.9228	0.9185	0.0008	0.2675
14x14C01	HI-TRAC	0.9273	0.9230	0.0008	0.2758
14x14C01	HI-STAR	0.9258	0.9215	0.0008	0.2729
14x14C02	HI-STAR	0.9265	0.9222	0.0008	0.2765
14x14C03	HI-TRAC	0.9274	0.9231	0.0008	0.2839
14x14C03	HI-STAR	0.9287	0.9242	0.0009	0.2825

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14D01	HI-TRAC	0.8531	0.8488	0.0008	0.3316
14x14D01	HI-STAR	0.8507	0.8464	0.0008	0.3308
14x14E01	HI-STAR	0.7598	0.7555	0.0008	0.3890
14x14E02	HI-TRAC	0.7627	0.7586	0.0007	0.3591
14x14E02	HI-STAR	0.7627	0.7586	0.0007	0.3607
14x14E03	HI-STAR	0.6952	0.6909	0.0008	0.2905
15x15A01	HI-TRAC	0.9205	0.9162	0.0008	0.2595
15x15A01	HI-STAR	0.9204	0.9159	0.0009	0.2608
15x15B01	HI-STAR	0.9369	0.9326	0.0008	0.2632
15C15B02	HI-STAR	0.9338	0.9295	0.0008	0.2640
15x15B03	HI-STAR	0.9362	0.9318	0.0008	0.2632
15x15B04	HI-STAR	0.9370	0.9327	0.0008	0.2612
15x15B05	HI-STAR	0.9356	0.9313	0.0008	0.2606
15x15B06	HI-STAR	0.9366	0.9324	0.0007	0.2638
B15x15B01	HI-TRAC	0.9387	0.9344	0.0008	0.2616
B15x15B01	HI-STAR	0.9388	0.9343	0.0009	0.2626
15x15C01	HI-STAR	0.9255	0.9213	0.0007	0.2493
15x15C02	HI-STAR	0.9297	0.9255	0.0007	0.2457
15x15C03	HI-STAR	0.9297	0.9255	0.0007	0.2440
15x15C04	HI-STAR	0.9311	0.9268	0.0008	0.2435
B15x15C01	HI-TRAC	0.9362	0.9319	0.0008	0.2374
B15x15C01	HI-STAR	0.9361	0.9316	0.0009	0.2385



<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
15x15D01	HI-STAR	0.9341	0.9298	0.0008	0.2822
15x15D02	HI-STAR	0.9367	0.9324	0.0008	0.2802
15x15D03	HI-STAR	0.9354	0.9311	0.0008	0.2844
15x15D04	HI-TRAC	0.9354	0.9309	0.0009	0.2963
15x15D04	HI-STAR	0.9339	0.9292	0.0010	0.2958
15x15E01	HI-TRAC	0.9392	0.9349	0.0008	0.2827
15x15E01	HI-STAR	0.9368	0.9325	0.0008	0.2826
15x15F01	HI-STORM	0.3648	0.3614	0.0003	3.03E+04
15x15F01	HI-TRAC	0.9393	0.9347	0.0009	0.2925
15x15F01	HI-STAR	0.9395	0.9350	0.0009	0.2903
15x15G01	HI-TRAC	0.8878	0.8836	0.0007	0.3347
15x15G01	HI-STAR	0.8876	0.8833	0.0008	0.3357
15x15H01	HI-TRAC	0.9333	0.9288	0.0009	0.2353
15x15H01	HI-STAR	0.9337	0.9292	0.0009	0.2349
16x16A01	HI-STAR	0.9287	0.9244	0.0008	0.2704
16x16A02	HI-STAR	0.9263	0.9221	0.0007	0.2702
16x16A03	HI-STORM	0.3588	0.3555	0.0003	2.11E+04
16x16A03	HI-TRAC	0.9322	0.9278	0.0008	0.2673
16x16A03	HI-STAR	0.9327	0.9282	0.0009	0.2661
16x16B01	HI-STAR	0.9370	0.9327	0.0008	0.2536
16x16C01	HI-STAR	0.9090	0.9046	0.0008	0.2890
17x17A01	HI-STORM	0.3243	0.3210	0.0003	3.23E+04

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
17x17A01	HI-TRAC	0.9378	0.9335	0.0008	0.2133
17x17A01	HI-STAR	0.9368	0.9325	0.0008	0.2131
17x17A02	HI-STAR	0.9329	0.9286	0.0008	0.2018
17x17B01	HI-STAR	0.9288	0.9243	0.0009	0.2607
17x17B02	HI-STAR	0.9290	0.9247	0.0008	0.2596
17x17B03	HI-STAR	0.9243	0.9199	0.0008	0.2625
17x17B04	HI-STAR	0.9324	0.9279	0.0009	0.2576
17x17B05	HI-STAR	0.9266	0.9222	0.0008	0.2539
17x17B06	HI-TRAC	0.9318	0.9275	0.0008	0.2570
17x17B06	HI-STAR	0.9311	0.9268	0.0008	0.2593
17x17C01	HI-STAR	0.9293	0.9250	0.0008	0.2595
17x17C02	HI-TRAC	0.9319	0.9274	0.0009	0.2610
17x17C02	HI-STAR	0.9336	0.9293	0.0008	0.2624

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
6x6A01	HI-STAR	0.7539	0.7498	0.0007	0.2754
6x6A02	HI-STAR	0.7517	0.7476	0.0007	0.2510
6x6A03	HI-STAR	0.7545	0.7501	0.0008	0.2494
6x6A04	HI-STAR	0.7537	0.7494	0.0008	0.2494
6x6A05	HI-STAR	0.7555	0.7512	0.0008	0.2470
6x6A06	HI-STAR	0.7618	0.7576	0.0008	0.2298
6x6A07	HI-STAR	0.7588	0.7550	0.0005	0.2360
6x6A08	HI-STAR	0.7808	0.7766	0.0007	0.2527
B6x6A01	HI-TRAC	0.7732	0.7691	0.0007	0.2458
B6x6A01	HI-STAR	0.7727	0.7685	0.0007	0.2460
B6x6A02	HI-TRAC	0.7785	0.7741	0.0008	0.2411
B6x6A02	HI-STAR	0.7782	0.7738	0.0008	0.2408
B6x6A03	HI-TRAC	0.7886	0.7846	0.0007	0.2311
B6x6A03	HI-STAR	0.7888	0.7846	0.0007	0.2310
6x6B01	HI-STAR	0.7604	0.7563	0.0007	0.2461
6x6B02	HI-STAR	0.7618	0.7577	0.0007	0.2450
6x6B03	HI-STAR	0.7619	0.7578	0.0007	0.2439
6x6B04	HI-STAR	0.7686	0.7644	0.0008	0.2286
6x6B05	HI-STAR	0.7824	0.7785	0.0006	0.2184
B6x6B01	HI-TRAC	0.7833	0.7794	0.0006	0.2181
B6x6B01	HI-STAR	0.7822	0.7783	0.0006	0.2190
6x6C01	HI-STORM	0.2759	0.2726	0.0003	1.59E+04

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
6x6C01	HI-TRAC	0.8024	0.7982	0.0008	0.2135
6x6C01	HI-STAR	0.8021	0.7980	0.0007	0.2139
7x7A01	HI-TRAC	0.7963	0.7922	0.0007	0.2016
7x7A01	HI-STAR	0.7974	0.7932	0.0008	0.2015
7x7B01	HI-STAR	0.9372	0.9330	0.0007	0.3658
7x7B02	HI-STAR	0.9301	0.9260	0.0007	0.3524
7x7B03	HI-STAR	0.9313	0.9271	0.0008	0.3438
7x7B04	HI-STAR	0.9311	0.9270	0.0007	0.3816
7x7B05	HI-STAR	0.9350	0.9306	0.0008	0.3382
7x7B06	HI-STAR	0.9298	0.9260	0.0006	0.3957
B7x7B01	HI-TRAC	0.9367	0.9324	0.0008	0.3899
B7x7B01	HI-STAR	0.9375	0.9332	0.0008	0.3887
B7x7B02	HI-STORM	0.4061	0.4027	0.0003	2.069E+04
B7x7B02	HI-TRAC	0.9385	0.9342	0.0008	0.3952
B7x7B02	HI-STAR	0.9386	0.9344	0.0007	0.3983
8x8A01	HI-TRAC	0.7662	0.7620	0.0008	0.2250
8x8A01	HI-STAR	0.7685	0.7644	0.0007	0.2227
8x8A02	HI-TRAC	0.7690	0.7650	0.0007	0.2163
8x8A02	HI-STAR	0.7697	0.7656	0.0007	0.2158
8x8B01	HI-STAR	0.9310	0.9265	0.0009	0.2935
8x8B02	HI-STAR	0.9227	0.9185	0.0007	0.2993
8x8B03	HI-STAR	0.9299	0.9257	0.0008	0.3319

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
8x8B04	HI-STAR	0.9236	0.9194	0.0008	0.3700
B8x8B01	HI-TRAC	0.9352	0.9310	0.0008	0.3393
B8x8B01	HI-STAR	0.9346	0.9301	0.0009	0.3389
B8x8B02	HI-TRAC	0.9401	0.9359	0.0007	0.3331
B8x8B02	HI-STAR	0.9385	0.9343	0.0008	0.3329
B8x8B03	HI-STORM	0.3934	0.3900	0.0004	1.815E+04
B8x8B03	HI-TRAC	0.9427	0.9385	0.0008	0.3278
B8x8B03	HI-STAR	0.9416	0.9375	0.0007	0.3293
8x8C01	HI-STAR	0.9315	0.9273	0.0007	0.2822
8x8C02	HI-STAR	0.9313	0.9268	0.0009	0.2716
8x8C03	HI-STAR	0.9329	0.9286	0.0008	0.2877
8x8C04	HI-STAR	0.9348	0.9307	0.0007	0.2915
8x8C05	HI-STAR	0.9353	0.9312	0.0007	0.2971
8x8C06	HI-STAR	0.9353	0.9312	0.0007	0.2944
8x8C07	HI-STAR	0.9314	0.9273	0.0007	0.2972
8x8C08	HI-STAR	0.9339	0.9298	0.0007	0.2915
8x8C09	HI-STAR	0.9301	0.9260	0.0007	0.3183
8x8C10	HI-STAR	0.9317	0.9275	0.0008	0.3018
8x8C11	HI-STAR	0.9328	0.9287	0.0007	0.3001
8x8C12	HI-STAR	0.9285	0.9242	0.0008	0.3062
B8x8C01	HI-TRAC	0.9348	0.9305	0.0008	0.3114
B8x8C01	HI-STAR	0.9357	0.9313	0.0009	0.3141

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
B8x8C02	HI-STORM	0.3714	0.3679	0.0004	2.30E+04
B8x8C02	HI-TRAC	0.9402	0.9360	0.0008	0.3072
B8x8C02	HI-STAR	0.9425	0.9384	0.0007	0.3081
B8x8C03	HI-TRAC	0.9429	0.9386	0.0008	0.3045
B8x8C03	HI-STAR	0.9418	0.9375	0.0008	0.3056
8x8D01	HI-STAR	0.9342	0.9302	0.0006	0.2733
8x8D02	HI-STAR	0.9325	0.9284	0.0007	0.2750
8x8D03	HI-STAR	0.9351	0.9309	0.0008	0.2731
8x8D04	HI-STAR	0.9338	0.9296	0.0007	0.2727
8x8D05	HI-STAR	0.9339	0.9294	0.0009	0.2700
8x8D06	HI-STAR	0.9365	0.9324	0.0007	0.2777
8x8D07	HI-STAR	0.9341	0.9297	0.0009	0.2694
8x8D08	HI-STAR	0.9376	0.9332	0.0009	0.2841
B8x8D01	HI-TRAC	0.9408	0.9368	0.0006	0.2773
B8x8D01	HI-STAR	0.9403	0.9363	0.0007	0.2778
8x8E01	HI-TRAC	0.9309	0.9266	0.0008	0.2834
8x8E01	HI-STAR	0.9312	0.9270	0.0008	0.2831
8x8F01	HI-TRAC	0.9396	0.9356	0.0006	0.2255
8x8F01	HI-STAR	0.9411	0.9366	0.0009	0.2264
9x9A01	HI-STAR	0.9353	0.9310	0.0008	0.2875
9x9A02	HI-STAR	0.9388	0.9345	0.0008	0.2228
9x9A03	HI-STAR	0.9351	0.9310	0.0007	0.2837

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
9x9A04	HI-STAR	0.9396	0.9355	0.0007	0.2262
B9x9A01	HI-STORM	0.3365	0.3331	0.0003	1.78E+04
B9x9A01	HI-TRAC	0.9434	0.9392	0.0007	0.2232
B9x9A01	HI-STAR	0.9417	0.9374	0.0008	0.2236
9x9B01	HI-STAR	0.9380	0.9336	0.0008	0.2576
9x9B02	HI-STAR	0.9373	0.9329	0.0009	0.2578
9x9B03	HI-STAR	0.9417	0.9374	0.0008	0.2545
B9x9B01	HI-TRAC	0.9417	0.9376	0.0007	0.2504
B9x9B01	HI-STAR	0.9436	0.9394	0.0008	0.2506
9x9C01	HI-TRAC	0.9377	0.9335	0.0008	0.2697
9x9C01	HI-STAR	0.9395	0.9352	0.0008	0.2698
9x9D01	HI-TRAC	0.9387	0.9343	0.0008	0.2635
9x9D01	HI-STAR	0.9394	0.9350	0.0009	0.2625
9x9E01	HI-STAR	0.9334	0.9293	0.0007	0.2227
9x9E02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9E02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9E02	HI-STAR	0.9401	0.9359	0.0008	0.2065
9x9F01	HI-STAR	0.9307	0.9265	0.0007	0.2899
9x9F02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9F02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9F02	HI-STAR	0.9401	0.9359	0.0008	0.2065
9x9G01	HI-TRAC	0.9307	0.9265	0.0007	0.2193

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
9x9G01	HI-STAR	0.9309	0.9265	0.0008	0.2191
10x10A01	HI-STAR	0.9377	0.9335	0.0008	0.3170
10x10A02	HI-STAR	0.9426	0.9386	0.0007	0.2159
10x10A03	HI-STAR	0.9396	0.9356	0.0007	0.3169
B10x10A01	HI-STORM	0.3379	0.3345	0.0003	1.74E+04
B10x10A01	HI-TRAC	0.9448	0.9405	0.0008	0.2214
B10x10A01	HI-STAR	0.9457	0.9414	0.0008	0.2212
10x10B01	HI-STAR	0.9384	0.9341	0.0008	0.2881
10x10B02	HI-STAR	0.9416	0.9373	0.0008	0.2333
10x10B03	HI-STAR	0.9375	0.9334	0.0007	0.2856
B10x10B01	HI-TRAC	0.9443	0.9401	0.0007	0.2380
B10x10B01	HI-STAR	0.9436	0.9395	0.0007	0.2366
10x10C01	HI-TRAC	0.9430	0.9387	0.0008	0.2424
10x10C01	HI-STAR	0.9433	0.9392	0.0007	0.2416
10x10D01	HI-TRAC	0.9383	0.9343	0.0007	0.3359
10x10D01	HI-STAR	0.9376	0.9333	0.0008	0.3355
10x10E01	HI-TRAC	0.9157	0.9116	0.0007	0.3301
10x10E01	HI-STAR	0.9185	0.9144	0.0007	0.2936



<b>MPC-24 400PPM SOLUBLE BORON</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.8884	0.8841	0.0008	0.2501
B14x14B01	HI-STAR	0.8900	0.8855	0.0009	0.3173
14x14C03	HI-STAR	0.8950	0.8907	0.0008	0.3410
14x14D01	HI-STAR	0.8518	0.8475	0.0008	0.4395
14x14E02	HI-STAR	0.7132	0.7090	0.0007	0.4377
15x15A01	HI-STAR	0.9119	0.9076	0.0008	0.3363
B15x15B01	HI-STAR	0.9284	0.9241	0.0008	0.3398
B15x15C01	HI-STAR	0.9236	0.9193	0.0008	0.3074
15x15D04	HI-STAR	0.9261	0.9218	0.0008	0.3841
15x15E01	HI-STAR	0.9265	0.9221	0.0008	0.3656
15x15F01	HI-STORM (DRY)	0.4013	0.3978	0.0004	28685
15x15F01	HI-TRAC	0.9301	0.9256	0.0009	0.3790
15x15F01	HI-STAR	0.9314	0.9271	0.0008	0.3791
15x15G01	HI-STAR	0.8939	0.8897	0.0007	0.4392
15x15H01	HI-TRAC	0.9345	0.9301	0.0008	0.3183
15x15H01	HI-STAR	0.9366	0.9320	0.0009	0.3175
16x16A03	HI-STAR	0.8993	0.8948	0.0009	0.3164
16x16B01	HI-STAR	0.9023	0.8978	0.0009	0.3011
16x16C01	HI-STAR	0.8808	0.8765	0.0008	0.3446
17x17A01	HI-STAR	0.9264	0.9221	0.0008	0.2801
17x17B06	HI-STAR	0.9284	0.9241	0.0008	0.3383
17x17C02	HI-TRAC	0.9296	0.9250	0.0009	0.3447

<b>MPC-24 400PPM SOLUBLE BORON</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
17x17C02	HI-STAR	0.9294	0.9249	0.0009	0.3433

<b>MPC-24E, UNBORATED WATER</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.9380	0.9337	0.0008	0.2277
B14x14B01	HI-STAR	0.9312	0.9269	0.0008	0.2927
14x14C01	HI-STAR	0.9356	0.9311	0.0009	0.3161
14x14D01	HI-STAR	0.8875	0.8830	0.0009	0.4026
14x14E02	HI-STAR	0.7651	0.7610	0.0007	0.3645
15x15A01	HI-STAR	0.9336	0.9292	0.0008	0.2879
B15x15B01	HI-STAR	0.9465	0.9421	0.0008	0.2924
B15x15C01	HI-STAR	0.9462	0.9419	0.0008	0.2631
15x15D04	HI-STAR	0.9440	0.9395	0.0009	0.3316
15x15E01	HI-STAR	0.9455	0.9411	0.0009	0.3178
15x15F01	HI-STORM (DRY)	0.3699	0.3665	0.0004	3.280e+04
15x15F01	HI-TRAC	0.9465	0.9421	0.0009	0.3297
15x15F01	HI-STAR	0.9468	0.9424	0.0008	0.3270
15x15G01	HI-STAR	0.9054	0.9012	0.0007	0.3781
15x15H01	HI-STAR	0.9423	0.9381	0.0008	0.2628
16x16A03	HI-STAR	0.9394	0.9351	0.0008	NC
16x16B01	HI-STAR	0.9436	0.9391	0.0009	0.2802
16x16C01	HI-STAR	0.9207	0.9162	0.0009	0.3201
17x17A01	HI-TRAC	0.9467	0.9425	0.0008	0.2372
17x17A01	HI-STAR	0.9447	0.9406	0.0007	0.2374
17x17B06	HI-STAR	0.9421	0.9377	0.0008	0.2888
17x17C02	HI-STAR	0.9433	0.9390	0.0008	0.2932

<b>MPC-24E, 300PPM BORATED WATER</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.8963	0.8921	0.0008	0.2231
B14x14B01	HI-STAR	0.8974	0.8931	0.0008	0.3214
14x14C01	HI-STAR	0.9031	0.8988	0.0008	0.3445
14x14D01	HI-STAR	0.8588	0.8546	0.0007	0.4407
14x14E02	HI-STAR	0.7249	0.7205	0.0008	0.4186
15x15A01	HI-STAR	0.9161	0.9118	0.0008	0.3408
B15x15B01	HI-STAR	0.9321	0.9278	0.0008	0.3447
B15x15C01	HI-STAR	0.9271	0.9227	0.0008	0.3121
15x15D04	HI-STAR	0.9290	0.9246	0.0009	0.3950
15x15E01	HI-STAR	0.9309	0.9265	0.0009	0.3754
15x15F01	HI-STORM (DRY)	0.3897	0.3863	0.0003	3.192E+04
15x15F01	HI-TRAC	0.9333	0.9290	0.0008	0.3900
15x15F01	HI-STAR	0.9332	0.9289	0.0008	0.3861
15x15G01	HI-STAR	0.8972	0.8930	0.0007	0.4473
15x15H01	HI-TRAC	0.9399	0.9356	0.0008	0.3235
15x15H01	HI-STAR	0.9399	0.9357	0.0008	0.3248
16x16A03	HI-STAR	0.9060	0.9017	0.0008	NC
16x16B01	HI-STAR	0.9080	0.9036	0.0008	0.3046
16x16C01	HI-STAR	0.8875	0.8830	0.0009	0.3488
17x17A01	HI-STAR	0.9332	0.9287	0.0009	0.2821
17x17B06	HI-STAR	0.9316	0.9273	0.0008	0.3455
17x17C02	HI-TRAC	0.9320	0.9277	0.0008	0.2819

<b>MPC-24E, 300PPM BORATED WATER</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
17X17C02	HI-STAR	0.9312	0.9270	0.0007	0.3530

<b>MPC-32, 4.1% Enrichment, Bounding Cases</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.9041	0.9001	0.0006	0.3185
B14x14B01	HI-STAR	0.9257	0.9216	0.0007	0.4049
14x14C01	HI-STAR	0.9423	0.9382	0.0007	0.4862
14x14D01	HI-STAR	0.8970	0.8931	0.0006	0.5474
14x14E02	HI-STAR	0.7340	0.7300	0.0006	0.6817
15x15A01	HI-STAR	0.9206	0.9167	0.0006	0.5072
B15x15B01	HI-STAR	0.9397	0.9358	0.0006	0.4566
B15x15C01	HI-STAR	0.9266	0.9227	0.0006	0.4167
15x15D04	HI-STAR	0.9384	0.9345	0.0006	0.5594
15x15E01	HI-STAR	0.9365	0.9326	0.0006	0.5403
15x15F01	HI-STORM (DRY)	0.4691	0.4658	0.0003	1.207E+04
15x15F01	HI-TRAC	0.9403	0.9364	0.0006	0.4938
15x15F01	HI-STAR	0.9411	0.9371	0.0006	0.4923
15x15G01	HI-STAR	0.9147	0.9108	0.0006	0.5880
15x15H01	HI-STAR	0.9276	0.9237	0.0006	0.4710
15x15I01	HI-STAR	0.9340	0.9301	0.0006	0.5488
16x16A03	HI-STAR	0.9375	0.9333	0.0007	0.4488
16x16B01	HI-STAR	0.9354	0.9315	0.0006	0.4253
16x16C01	HI-STAR	0.9178	0.9137	0.0007	0.4408
17x17A01	HI-STAR	0.9421	0.9381	0.0006	0.3534
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365

<b>MPC-32, 4.1% Enrichment, Bounding Cases</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
17x17C02	HI-TRAC	0.9365	0.9327	0.0006	0.4468
17x17C02	HI-STAR	0.9355	0.9317	0.0006	0.4469

<b>MPC-32, 5.0% Enrichment, Bounding Cases</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.9000	0.8959	0.0007	0.4651
B14x14B01	HI-STAR	0.9214	0.9175	0.0006	0.6009
14x14C01	HI-STAR	0.9480	0.9440	0.0006	0.6431
14x14D01	HI-STAR	0.9050	0.9009	0.0007	0.7276
14x14E02	HI-STAR	0.7415	0.7375	0.0006	0.9226
15x15A01	HI-STAR	0.9230	0.9189	0.0007	0.7143
B15x15B01	HI-STAR	0.9429	0.9390	0.0006	0.7234
B15x15C01	HI-STAR	0.9307	0.9268	0.0006	0.6439
15x15D04	HI-STAR	0.9466	0.9425	0.0007	0.7525
15x15E01	HI-STAR	0.9434	0.9394	0.0007	0.7215
15x15F01	HI-STORM (DRY)	0.5142	0.5108	0.0004	1.228E+04
15x15F01	HI-TRAC	0.9470	0.9431	0.0006	0.7456
15x15F01	HI-STAR	0.9483	0.9443	0.0007	0.7426
15x15G01	HI-STAR	0.9251	0.9212	0.0006	0.9303
15x15H01	HI-STAR	0.9333	0.9292	0.0007	0.7015
15x15I01	HI-STAR	0.9402	0.9359	0.0008	0.7318

<b>MPC-32, 5.0% Enrichment, Bounding Cases</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
16x16A03	HI-STAR	0.9429	0.9388	0.0007	0.5920
16x16B01	HI-STAR	0.9378	0.9339	0.0006	0.5632
16x16C01	HI-STAR	0.9208	0.9167	0.0007	0.5898
17x17A01	HI-STAR	0.9475	0.9435	0.0006	0.5285
17x17B06	HI-STAR	0.9371	0.9331	0.0006	0.6705
17x17C02	HI-TRAC	0.9436	0.9396	0.0006	0.6773
17x17C02	HI-STAR	0.9437	0.9399	0.0006	0.6780

Note: Maximum  $k_{eff}$  = Calculated  $k_{eff}$  +  $K_c \times \sigma_c$  + Bias +  $\sigma_B$

where:

$$K_c = 2.0$$

$\sigma_c$  = Std. Dev. (1-sigma)

Bias = 0.0021

$\sigma_B$  = 0.0006

See Subsection 6.4.3 for further explanation.



**APPENDIX 6.D: SAMPLE INPUT FILES****(Total number of pages in this appendix : 39)**

File Description	Starting Page
MCNP4a input file for MPC-24 in HI-TRAC	Appendix 6.D-2
MCNP4a input file for MPC-68 in HI-TRAC	Appendix 6.D-14
MCNP4a input file for MPC-24 in HI-STORM	Appendix 6.D-21
MCNP4a input file for MPC-68 in HI-STORM	Appendix 6.D-33

HI-TRAC Transfer Cask containing MPC24, 17x17 assembly @ 4.0 wt% Enrich.

```

c
c
c
c MPC-24/24E cell configuration
c
c HI-TRAC with active length 150 inch
c
c Cask Input Preprocessor
c cskinp 17a 17a mpc24n mpc24n hitrac trac150 4.0 4tf7a45 pure
c -----      cpp\17a.bat
c   added 17a.ce
c   added 17a.su
c   added 17a.sp
c -----      cpp\mpc24n.bat
c   added mpc24n.co
c   added mpc24n.ce
c   added mpc24n.su
c   added mpc24n.sp
c -----      cpp\hitrac.bat
c   added trac150.co
c   added hitrac.ce
c   added trac150.su
c   added trac150.sp
c end of comments
c
c start of cells
c
c 17x17a
c
c number of cells: 6
c cell numbers:      1 to 7
c univers numbers:   1 to 3
c surface numbers:   1 to 9
c
c number of cells: 1
1   1 -10.522   -1  u=2      $ fuel
2   4 -1.0      1  -2  u=2      $ gap
3   3 -6.55     2  -3  u=2      $ Zr Clad
4   4 -1.0      3  u=2      $ water in fuel region
5   4 -1.0  -4:5    u=3      $ water in guide tubes
6   3 -6.55     4  -5    u=3      $ guide tubes
7   4 -1.0  -6   +7   -8   +9    u=1  lat=1
    fill= -9:9   -9:9   0:0
    1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
    1 2 2 2 3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
    1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
    1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1

```

```

1 2 2 2 3 2 2 2 2 2 2 2 2 2 3 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1

```

c

c MPC-24

c

c number of cells: 102

c cell numbers : 400 to 699

c universe numbers : 4 to 9

c surface numbers : 400 to 699

c

c Right Side

c

```

408 0 -410 411 -412 413 u=4 fill=1 (1)
409 5 -7.84 410 -424 413 -426 u=4
410 4 -1.0 424 -428 448 -445 u=4
411 7 -2.7 428 -528 448 -445 u=4
412 6 -2.66 528 -532 448 -445 u=4
413 7 -2.7 532 -432 448 -445 u=4
414 4 -1.0 432 -436 448 -445 u=4
415 5 -7.84 436 -440 448 -445 u=4
416 4 -1.0 440 413 u=4
417 4 -1.0 424 -440 413 -447 u=4
418 4 -1.0 424 -440 446 u=4
419 5 -7.84 424 -440 447 -448 u=4
420 5 -7.84 424 -440 445 -446 u=4

```

c

c Left Side

c

```

421 5 -7.84 425 -411 413 u=4
422 4 -1.0 429 -425 448 -445 u=4
423 7 -2.7 529 -429 448 -445 u=4
424 6 -2.66 533 -529 448 -445 u=4
425 7 -2.7 433 -533 448 -445 u=4
426 4 -1.0 437 -433 448 -445 u=4
427 5 -7.84 441 -437 448 -445 u=4
428 4 -1.0 -441 413 u=4
429 4 -1.0 441 -425 413 -447 u=4
430 4 -1.0 441 -425 446 u=4
431 5 -7.84 441 -425 447 -448 u=4
432 5 -7.84 441 -425 445 -446 u=4

```

c

c Top

c

```

433 5 -7.84 411 -410 412 -426 u=4
434 4 -1.0 451 -452 426 -430 u=4
435 7 -2.7 451 -452 430 -530 u=4
436 6 -2.66 451 -452 530 -534 u=4
437 7 -2.7 451 -452 534 -434 u=4
438 4 -1.0 451 -452 434 -438 u=4
439 5 -7.84 451 -452 438 -442 u=4
440 4 -1.0 411 -424 442 u=4
441 4 -1.0 411 -450 426 -442 u=4
442 4 -1.0 453 -424 426 -442 u=4
443 5 -7.84 450 -451 426 -442 u=4
444 5 -7.84 452 -453 426 -442 u=4

```

```

c
c   Bottom
c
445  5 -7.84   427           -413   u=4
446  4 -1.0    451 -452   431 -427   u=4
447  7 -2.7    451 -452   531 -431   u=4
448  6 -2.66   451 -452   535 -531   u=4
449  7 -2.7    451 -452   435 -535   u=4
450  4 -1.0    451 -452   439 -435   u=4
451  5 -7.84   451 -452   443 -439   u=4
452  4 -1.0    411           -443   u=4
453  4 -1.0    411 -450   443 -427   u=4
454  4 -1.0    453           443 -427   u=4
455  5 -7.84   450 -451   443 -427   u=4
456  5 -7.84   452 -453   443 -427   u=4
457  5 -7.84   425 -411           -427   u=4
458  4 -1.0           -425           -427   u=4
c
c   TYPE B CELL - Short Boral on top and right
c
c   Right Side
c
459  0           -410   411 -412   413   u=5 fill=1 (1)
460  5 -7.84   410 -424   413 -426   u=5
470  4 -1.0    424 -428   548 -545   u=5
471  7 -2.7    428 -528   548 -545   u=5
472  6 -2.66   528 -532   548 -545   u=5
473  7 -2.7    532 -432   548 -545   u=5
474  4 -1.0    432 -436   548 -545   u=5
475  5 -7.84   436 -440   548 -545   u=5
476  4 -1.0    440           413   u=5
477  4 -1.0    424 -440   413 -547   u=5
478  4 -1.0    424 -440   546           u=5
479  5 -7.84   424 -440   547 -548   u=5
480  5 -7.84   424 -440   545 -546   u=5
c
c   Left Side
c
481  5 -7.84   425 -411   413           u=5
482  4 -1.0    429 -425   448 -445   u=5
483  7 -2.7    529 -429   448 -445   u=5
484  6 -2.66   533 -529   448 -445   u=5
485  7 -2.7    433 -533   448 -445   u=5
486  4 -1.0    437 -433   448 -445   u=5
487  5 -7.84   441 -437   448 -445   u=5
488  4 -1.0           -441   413   u=5
489  4 -1.0    441 -425   413 -447   u=5
490  4 -1.0    441 -425   446           u=5
491  5 -7.84   441 -425   447 -448   u=5
492  5 -7.84   441 -425   445 -446   u=5
c
c   Top
c
493  5 -7.84   411 -410   412 -426   u=5
494  4 -1.0    551 -552   426 -430   u=5
495  7 -2.7    551 -552   430 -530   u=5
496  6 -2.66   551 -552   530 -534   u=5
497  7 -2.7    551 -552   534 -434   u=5

```

Attachment 6 to Holtec Letter 5014874

498	4	-1.0	551	-552	434	-438	u=5
499	5	-7.84	551	-552	438	-442	u=5
500	4	-1.0	411	-424	442		u=5
501	4	-1.0	411	-550	426	-442	u=5
502	4	-1.0	553	-424	426	-442	u=5
503	5	-7.84	550	-551	426	-442	u=5
504	5	-7.84	552	-553	426	-442	u=5

c

c Bottom

c

505	5	-7.84	427			-413	u=5
506	4	-1.0	451	-452	431	-427	u=5
507	7	-2.7	451	-452	531	-431	u=5
508	6	-2.66	451	-452	535	-531	u=5
509	7	-2.7	451	-452	435	-535	u=5
510	4	-1.0	451	-452	439	-435	u=5
511	5	-7.84	451	-452	443	-439	u=5
512	4	-1.0	411			-443	u=5
513	4	-1.0	411	-450	443	-427	u=5
514	4	-1.0	453		443	-427	u=5
515	5	-7.84	450	-451	443	-427	u=5
516	5	-7.84	452	-453	443	-427	u=5
517	5	-7.84	425	-411		-427	u=5
518	4	-1.0		-425		-427	u=5

c

c

c

c TYPE D CELL - Short Boral on left and bottom, different cell ID

c

c number of cells: 51

c

c Right Side

c

1570	0		-1410	1411	-1412	1413	u=17 fill=1 (1)
1571	5	-7.84	1410	-1424	1413	-1426	u=17
1572	4	-1.0	1424	-1428	1448	-1445	u=17
1573	7	-2.7	1428	-1528	1448	-1445	u=17
1574	6	-2.66	1528	-1532	1448	-1445	u=17
1575	7	-2.7	1532	-1432	1448	-1445	u=17
1576	4	-1.0	1432	-1436	1448	-1445	u=17
1577	5	-7.84	1436	-1440	1448	-1445	u=17
1578	4	-1.0	1440		1413		u=17
1579	4	-1.0	1424	-1440	1413	-1447	u=17
1580	4	-1.0	1424	-1440	1446		u=17
1581	5	-7.84	1424	-1440	1447	-1448	u=17
1582	5	-7.84	1424	-1440	1445	-1446	u=17

c

c Left Side

c

1583	5	-7.84	1425	-1411	1413		u=17
1584	4	-1.0	1429	-1425	1548	-1545	u=17
1585	7	-2.7	1529	-1429	1548	-1545	u=17
1586	6	-2.66	1533	-1529	1548	-1545	u=17
1587	7	-2.7	1433	-1533	1548	-1545	u=17
1588	4	-1.0	1437	-1433	1548	-1545	u=17
1589	5	-7.84	1441	-1437	1548	-1545	u=17
1590	4	-1.0		-1441	1413		u=17
1591	4	-1.0	1441	-1425	1413	-1547	u=17

Attachment 6 to Holtec Letter 5014874

```

1592  4 -1.0    1441 -1425  1546          u=17
1593  5 -7.84   1441 -1425  1547 -1548    u=17
1594  5 -7.84   1441 -1425  1545 -1546    u=17
c
c      Top
c
1595  5 -7.84   1411 -1410   1412 -1426    u=17
1596  4 -1.0    1451 -1452   1426 -1430    u=17
1597  7 -2.7    1451 -1452   1430 -1530    u=17
1598  6 -2.66   1451 -1452   1530 -1534    u=17
1599  7 -2.7    1451 -1452   1534 -1434    u=17
1600  4 -1.0    1451 -1452   1434 -1438    u=17
1601  5 -7.84   1451 -1452   1438 -1442    u=17
1602  4 -1.0    1411 -1424   1442          u=17
1603  4 -1.0    1411 -1450   1426 -1442    u=17
1604  4 -1.0    1453 -1424   1426 -1442    u=17
1605  5 -7.84   1450 -1451   1426 -1442    u=17
1606  5 -7.84   1452 -1453   1426 -1442    u=17
c
c      Bottom
c
1607  5 -7.84   1427          -1413    u=17
1608  4 -1.0    1551 -1552   1431 -1427    u=17
1609  7 -2.7    1551 -1552   1531 -1431    u=17
1610  6 -2.66   1551 -1552   1535 -1531    u=17
1611  7 -2.7    1551 -1552   1435 -1535    u=17
1612  4 -1.0    1551 -1552   1439 -1435    u=17
1613  5 -7.84   1551 -1552   1443 -1439    u=17
1614  4 -1.0    1411          -1443    u=17
1615  4 -1.0    1411 -1550   1443 -1427    u=17
1616  4 -1.0    1553          1443 -1427    u=17
1617  5 -7.84   1550 -1551   1443 -1427    u=17
1618  5 -7.84   1552 -1553   1443 -1427    u=17
1619  5 -7.84   1425 -1411          -1427    u=17
1620  4 -1.0          -1425          -1427    u=17
c
c number of cells: 29
c
c empty cell no boron, no top
c
c
751   4 -1.0   -410  411  -412  413    u=14
752   5 -7.84   410  -424  413  -426    u=14
753   5 -7.84   425  -411  413          u=14
754   4 -1.0   411  -410  412  -426    u=14
755   5 -7.84   427          -413    u=14
756   5 -7.84   425  -411          -427    u=14
757   4 -1.0   411  426          u=14
758   4 -1.0   411  -427          u=14
759   4 -1.0  -425  413          u=14
760   4 -1.0   424  413  -426    u=14
761   4 -1.0  -425  -427          u=14
c
c
701   5 -7.84   701 -702  711 -713    u=9  $ steel post
702   5 -7.84   702 -703  711 -712    u=9  $ steel post
c
711   0          701 -705  711 -715 (702:713) (703:712)

```

Attachment 6 to Holtec Letter 5014874

```

fill=4 (13.8506 13.8506 0) u=9
712 0 704 (-706:-716) (705:715) -717 -710
fill=4 (17.9489 41.5518 0 0 1 0 -1 0 0 0 0 1) u=9
713 0 (705:715) -707 714 (-706:-716) 710
fill=4 (41.5518 17.9489 0 0 -1 0 1 0 0 0 0 1) u=9
714 0 701 -705 717 -719
fill=5 (13.8506 69.253 0) u=9
715 0 707 -709 711 -715
fill=5 (69.253 13.8506 0) u=9
716 0 706 -708 716 -718
fill=17 (45.6501 45.6501 0 -1 0 0 0 -1 0 0 0 1) u=9
717 0 705 -706 717 -719
fill=14 (41.5518 69.253 0) u=9
718 0 707 -709 715 -716
fill=14 (69.253 41.5518 0 0 1 0 1 0 0 0 0 1) u=9
719 0 701 -704 715 -717
fill=14 (-9.75233 41.5518 0 -1 0 0 0 1 0 0 0 1) u=9
720 0 705 -707 711 -714
fill=14 (41.5518 -9.75233 0 0 -1 0 1 0 0 0 0 1) u=9
721 4 -1.0 (706:719) (708:718) (709:716) u=9
c
c
c
731 4 -1.0 720 721 fill=9 (0 0 0) u=19
732 4 -1.0 -720 721 fill=9 (0 0 0
-1 0 0 0 1 0 0 0 1) u=19
733 4 -1.0 720 -721 fill=9 (0 0 0
1 0 0 0 -1 0 0 0 1) u=19
734 4 -1.0 -720 -721 fill=9 (0 0 0
-1 0 0 0 -1 0 0 0 1) u=19
c
673 0 -41 39 -40 fill=19
c
c number of cells: 20
374 4 -1.0 -41 300 -39 $ Water below Fuel (4 in.)
375 5 -7.84 -309 302 -300 $ MPC Steel below Fuel (2.5 in.)
376 5 -7.84 -315 320 -302 $ Transfer Cask Steel (2.0 in.)
377 30 -11.34 -315 321 -320 $ Transfer Cask Lead (2.5 in.)
378 5 -7.84 -315 322 -321 $ Transfer Cask Steel (1.0 in.)
c
379 4 -1.0 -41 40 -301 $ Water above Fuel (6 in.)
380 5 -7.84 -309 301 -303 $ MPC Steel above Fuel (9.5 - 0.06 in)
381 4 -1.0 -309 303 -330 $ Water (1.5 in.)
382 5 -7.84 -315 330 -331 $ Transfer Cask Steel (0.75 in.)
383 31 -1.61 -315 331 -332 $ Transfer Cask Neutron Shield (3.25 in.)
384 5 -7.84 -315 332 -333 $ Transfer Cask Steel (0.5 in.)
c
390 5 -7.84 41 -309 300 -301 $ Radial Steel - MPC shell
391 4 -1.00 309 -310 302 -330 $ Radial Water
392 5 -7.84 310 -311 302 -330 $ Radial Steel - inner shell of Trnsfr Cask
393 30 -11.34 311 -312 302 -330 $ Radial Lead - Transfer Cask lead
394 5 -7.84 312 -313 302 -330 $ Radial Steel - outer shell of Trnsfr Cask
395 4 -1.00 313 -314 302 -330 $ Radial Water - Water Jacket
396 5 -7.84 314 -315 302 -330 $ Radial Steel - outer shell of Water Jacket
c
300 4 -1.00 340 -341 -345 (315 :-322: 333) $ outer water reflector
301 0 345 :-340: 341 $ outside world
c end of cells

```

c --- empty line

c --- empty line

c start of surfaces

1	cz	0.3922	\$ fuel
2	cz	0.4001	\$ clad ID
3	cz	0.4572	\$ clad OD
4	cz	0.5613	\$ guide ID
5	cz	0.6020	\$ guide OD
6	px	0.6299	\$ pin pitch
7	px	-0.6299	
8	py	0.6299	
9	py	-0.6299	

c

c

c cell-id	8.98
c cell-pitch	10.906
c wall-thkns	5/16
c angle-thkns	5/16
c boral-gap	0.0035
c boral-gap-o	0.0035
c boral-thkns	0.075
c boral-clad	0.01
c sheathing	0.0235
c boral-wide	7.5
c boral-narrow	6.25

c

c gap size	1.09
c basket-od	67.335

c

410	px	11.40460	\$x 8.98/2	
411	px	-11.40460	\$x {410} *-1	
412	py	11.40460	\$x {410}	
413	py	-11.40460	\$x {411}	
416	px	13.85062	\$x (10.906 + 5/16 - 5/16) /2	
417	px	-13.85062	\$x -10.906 + {416}	
418	py	13.85062	\$x {416}	
419	py	-13.85062	\$x {417}	
424	px	12.19835	\$x {410} + 5/16	\$ angle
425	px	-12.19835	\$x {411} - 5/16	\$ box wall
426	py	12.19835	\$x {412} + 5/16	
427	py	-12.19835	\$x {413} - 5/16	
428	px	12.20724	\$x {424} + 0.0035	\$ wall to boral gap
429	px	-12.20724	\$x {425} - 0.0035	
430	py	12.20724	\$x {426} + 0.0035	
431	py	-12.20724	\$x {427} - 0.0035	
432	px	12.39774	\$x {428} + 0.075	\$ boral
433	px	-12.39774	\$x {429} - 0.075	
434	py	12.39774	\$x {430} + 0.075	
435	py	-12.39774	\$x {431} - 0.075	
436	px	12.40663	\$x {432} + 0.0035	\$ boral to sheathing gap
437	px	-12.40663	\$x {433} - 0.0035	
438	py	12.40663	\$x {434} + 0.0035	
439	py	-12.40663	\$x {435} - 0.0035	
440	px	12.46632	\$x {436} + 0.0235	\$ sheathing
441	px	-12.46632	\$x {437} - 0.0235	
442	py	12.46632	\$x {438} + 0.0235	
443	py	-12.46632	\$x {439} - 0.0235	



## Attachment 6 to Holtec Letter 5014874

```

445   py           9.52500 $x 7.5/2
446   py           9.58469 $x {445} + 0.0235   $ sheathing
447   py          -9.58469 $x {446} *-1
448   py          -9.52500 $x {445} *-1
450   px          -9.58469 $x {447}
451   px          -9.52500 $x {448}
452   px           9.52500 $x {445}
453   px           9.58469 $x {446}
528   px          12.23264 $x {428} + 0.01   $ Aluminum on the outside of boral
529   px         -12.23264 $x {429} - 0.01
530   py          12.23264 $x {430} + 0.01
531   py         -12.23264 $x {431} - 0.01
532   px          12.37234 $x {432} - 0.01
533   px         -12.37234 $x {433} + 0.01
534   py          12.37234 $x {434} - 0.01
535   py         -12.37234 $x {435} + 0.01
545   py           7.93750 $x 6.25/2
546   py           7.99719 $x {545} + 0.0235   $ sheathing
547   py          -7.99719 $x {546} *-1
548   py          -7.93750 $x {545} *-1
550   px          -7.99719 $x {547}
551   px          -7.93750 $x {548}
552   px           7.93750 $x {545}
553   px           7.99719 $x {546}
c
c cell-id-2 8.98
c gap-o      1.09
c
701   px          -5.0
702   px          1.90627 $x (10.906 - 8.98)/2 - 5/16 + 0.1
703   px          3.45694 $x 2.722/2
704   px          4.09829 $x 10.906 - 8.98 - 5/16
705   px          27.70124 $x 10.906
706   px          31.79953 $x 2 * 10.906 - (8.98+8.98)/2 - 5/16
707   px          55.40248 $x 2 * 10.906
708   px          59.50077 $x {707} + {704}
709   px          83.10372 $x 3 * 10.906
710   p           1 -1 0 0.1 $ diagonal x=y, offset by 0.1 to avoid intersecting corners
711   py          -4.99999 $x {701}
712   py          1.90627 $x {702}
713   py          3.45694 $x {703}
714   py          4.09829 $x {704}
715   py          27.70124 $x {705}
716   py          31.79953 $x {706}
717   py          55.40248 $x {707}
718   py          59.50077 $x {708}
719   py          83.10372 $x {709}
720   px           0.0
721   py           0.0
1410  px           11.40460 $x 8.98/2
1411  px          -11.40460 $x {1410} *-1
1412  py           11.40460 $x {1410}
1413  py          -11.40460 $x {1411}
1424  px           12.19835 $x {1410} + 5/16   $ angle
1425  px          -12.19835 $x {1411} - 5/16   $ box wall
1426  py           12.19835 $x {1412} + 5/16
1427  py          -12.19835 $x {1413} - 5/16
1428  px           12.20724 $x {1424} + 0.0035   $ wall to boral gap

```

## Attachment 6 to Holtec Letter 5014874

1429	px	-12.20724	\$x {1425}	- 0.0035	
1430	py	12.20724	\$x {1426}	+ 0.0035	
1431	py	-12.20724	\$x {1427}	- 0.0035	
1432	px	12.39774	\$x {1428}	+ 0.075	\$ boral
1433	px	-12.39774	\$x {1429}	- 0.075	
1434	py	12.39774	\$x {1430}	+ 0.075	
1435	py	-12.39774	\$x {1431}	- 0.075	
1436	px	12.40663	\$x {1432}	+ 0.0035	\$ boral to sheathing gap
1437	px	-12.40663	\$x {1433}	- 0.0035	
1438	py	12.40663	\$x {1434}	+ 0.0035	
1439	py	-12.40663	\$x {1435}	- 0.0035	
1440	px	12.46632	\$x {1436}	+ 0.0235	\$ sheathing
1441	px	-12.46632	\$x {1437}	- 0.0235	
1442	py	12.46632	\$x {1438}	+ 0.0235	
1443	py	-12.46632	\$x {1439}	- 0.0235	
1445	py	9.52500	\$x 7.5/2		
1446	py	9.58469	\$x {1445}	+ 0.0235	\$ sheathing
1447	py	-9.58469	\$x {1446}	*-1	
1448	py	-9.52500	\$x {1445}	*-1	
1450	px	-9.58469	\$x {1447}		
1451	px	-9.52500	\$x {1448}		
1452	px	9.52500	\$x {1445}		
1453	px	9.58469	\$x {1446}		
1528	px	12.23264	\$x {1428}	+ 0.01	\$ Aluminum on the outside of boral
1529	px	-12.23264	\$x {1429}	- 0.01	
1530	py	12.23264	\$x {1430}	+ 0.01	
1531	py	-12.23264	\$x {1431}	- 0.01	
1532	px	12.37234	\$x {1432}	- 0.01	
1533	px	-12.37234	\$x {1433}	+ 0.01	
1534	py	12.37234	\$x {1434}	- 0.01	
1535	py	-12.37234	\$x {1435}	+ 0.01	
1545	py	7.93750	\$x 6.25/2		
1546	py	7.99719	\$x {1545}	+ 0.0235	\$ sheathing
1547	py	-7.99719	\$x {1546}	*-1	
1548	py	-7.93750	\$x {1545}	*-1	
1550	px	-7.99719	\$x {1547}		
1551	px	-7.93750	\$x {1548}		
1552	px	7.93750	\$x {1545}		
1553	px	7.99719	\$x {1546}		
39	pz	0.0			\$ bottom of active fuel assembly
40	pz	381.0			\$ top of active fuel assembly
41	cz	85.57			\$ MPC
300	pz	-10.16			\$ lower water thkness = 4 in.
301	pz	396.24			\$ upper water thkness = 6 in.
302	pz	-16.51			\$ thkness of MPC baseplate = 2.5 in.
303	pz	420.22			\$ thkness of MPC lid = 9.5 -0.06 in.
309	cz	86.84			\$ I.D. = 68.375 in.
310	cz	87.31			\$ I.D. = 68.75 in.
311	cz	89.22			\$ I.D. = 70.25 in.
312	cz	100.65			\$ I.D. = 79.25 in.
313	cz	103.19			\$ I.D. = 81.25 in.
314	cz	116.80			\$ I.D. = 91.97 in.
315	cz	118.07			\$ I.D. = 92.972 in.
320	pz	-21.59			\$ thkness steel - 2.0 in.
321	pz	-27.94			\$ thkness lead - 2.5 in.
322	pz	-30.48			\$ thkness steel - 1.0 in.
330	pz	424.03			\$ thkness water - 1.5 in.
331	pz	425.93			\$ thkness steel - 0.75 in.

```

332 pz      434.19  $ thkness neutron shield - 3.25 in.
333 pz      435.46  $ thkness steel - 0.5 in.
c
*340 pz -60.48  $ lower boundary
*341 pz  465.46  $ upper boundary
*345 cz  148.07  $ outer radial boundary
c end of surfaces
c --- empty line

c --- empty line
tr1 0 0 0
kcode 10000 .94 20 120
dbcn 7j 1e7
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
sp3 0 1
c
si4 s      13 14
          12 13 14 15
          11 12 13 14 15 16
          11 12 13 14 15 16
          12 13 14 15
          13 14

sp4 1 23r
c
ds5 s      26 26
          25 25 25 25
          24 24 24 24 24 24
          23 23 23 23 23 23
          22 22 22 22
          21 21

c
si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639
si14  2.86639  24.50719
si15  30.23997  51.88077
si16  57.61355  79.25435
c
si21 -79.25435 -57.61355
si22 -51.88077 -30.23997
si23 -24.50719 -2.86639
si24  2.86639  24.50719
si25  30.23997  51.88077
si26  57.61355  79.25435
c
sp11 0 1
sp12 0 1
sp13 0 1
sp14 0 1
sp15 0 1
sp16 0 1
sp21 0 1
sp22 0 1
sp23 0 1
sp24 0 1

```

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```

sp25  0 1
sp26  0 1
c
m3      40000.56c  1.          $ Zr Clad
m4      1001.50c   0.6667        $ Water
        8016.50c   0.3333
m5      24000.50c  0.01761       $ Steel
        25055.50c  0.001761
        26000.55c  0.05977
        28000.50c  0.008239
m6      5010.50c  -0.054427     $ Boral Central Section @ 0.02 g/cmsq
        5011.50c  -0.241373
        13027.50c -0.6222
        6000.50c  -0.0821
m7      13027.50c  1.0
mt4      lwtr.01t
prdmpr  j  -120  j  2
fm4      1000  1  -6
f4:n     1
sd4      1000
e4      1.000E-11  1.000E-10  5.000E-10  7.500E-10  1.000E-09  1.200E-09
        1.500E-09  2.000E-09  2.500E-09  3.000E-09
        4.700E-09  5.000E-09  7.500E-09  1.000E-08  2.530E-08
        3.000E-08  4.000E-08  5.000E-08  6.000E-08  7.000E-08
        8.000E-08  9.000E-08  1.000E-07  1.250E-07  1.500E-07
        1.750E-07  2.000E-07  2.250E-07  2.500E-07  2.750E-07
        3.000E-07  3.250E-07  3.500E-07  3.750E-07  4.000E-07
        4.500E-07  5.000E-07  5.500E-07  6.000E-07  6.250E-07
        6.500E-07  7.000E-07  7.500E-07  8.000E-07  8.500E-07
        9.000E-07  9.250E-07  9.500E-07  9.750E-07  1.000E-06
        1.010E-06  1.020E-06  1.030E-06  1.040E-06  1.050E-06
        1.060E-06  1.070E-06  1.080E-06  1.090E-06  1.100E-06
        1.110E-06  1.120E-06  1.130E-06  1.140E-06  1.150E-06
        1.175E-06  1.200E-06  1.225E-06  1.250E-06  1.300E-06
        1.350E-06  1.400E-06  1.450E-06  1.500E-06  1.590E-06
        1.680E-06  1.770E-06  1.860E-06  1.940E-06  2.000E-06
        2.120E-06  2.210E-06  2.300E-06  2.380E-06  2.470E-06
        2.570E-06  2.670E-06  2.770E-06  2.870E-06  2.970E-06
        3.000E-06  3.050E-06  3.150E-06  3.500E-06  3.730E-06
        4.000E-06  4.750E-06  5.000E-06  5.400E-06  6.000E-06
        6.250E-06  6.500E-06  6.750E-06  7.000E-06  7.150E-06
        8.100E-06  9.100E-06  1.000E-05  1.150E-05  1.190E-05
        1.290E-05  1.375E-05  1.440E-05  1.510E-05  1.600E-05
        1.700E-05  1.850E-05  1.900E-05  2.000E-05  2.100E-05
        2.250E-05  2.500E-05  2.750E-05  3.000E-05  3.125E-05
        3.175E-05  3.325E-05  3.375E-05  3.460E-05  3.550E-05
        3.700E-05  3.800E-05  3.910E-05  3.960E-05  4.100E-05
        4.240E-05  4.400E-05  4.520E-05  4.700E-05  4.830E-05
        4.920E-05  5.060E-05  5.200E-05  5.340E-05  5.900E-05
        6.100E-05  6.500E-05  6.750E-05  7.200E-05  7.600E-05
        8.000E-05  8.200E-05  9.000E-05  1.000E-04  1.080E-04
        1.150E-04  1.190E-04  1.220E-04  1.860E-04  1.925E-04
        2.075E-04  2.100E-04  2.400E-04  2.850E-04  3.050E-04
        5.500E-04  6.700E-04  6.830E-04  9.500E-04  1.150E-03
        1.500E-03  1.550E-03  1.800E-03  2.200E-03  2.290E-03
        2.580E-03  3.000E-03  3.740E-03  3.900E-03  6.000E-03
        8.030E-03  9.500E-03  1.300E-02  1.700E-02  2.500E-02
        3.000E-02  4.500E-02  5.000E-02  5.200E-02  6.000E-02

```

Attachment 6 to Holtec Letter 5014874

	7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
	1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
	4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
	5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
	7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
	9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
	1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
	2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
	6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
	1.455E+01	1.568E+01	1.733E+01	2.000E+01	

si3 h 0 381.00

m30 82000.50c 1.0 \$ Lead

m31 6000.50c -27.660 \$ Neutron Shield Holtite-A (NS-4-FR)

1001.50c -5.920

13027.50c -21.285

7014.50c -1.98

8016.50c -42.372

5010.50c -0.141

5011.50c -0.642

imp:n 1 207r 0

c fuel enrichment 4.0 %

m1 92235.50c -0.03526

92238.50c -0.84624

8016.50c -0.11850

c end of file

c

HI-TRAC Transfer Cask containing MPC68, 08x08 assembly @ 4.2 wt% Enrich.  
 c reflected w/60cm of water, 0.0279 g/cmsq B-10 in Boral

```

c
c
1 1 -10.522 -1 u=2 $ fuel
2 4 -1.0 1 -2 u=2 $ gap
3 3 -6.55 2 -3 u=2 $ Zr Clad
4 4 -1.0 3 u=2 $ water in fuel region
5 4 -1.0 -4:5 u=3 $ water in guide tubes
6 4 -1.00 4 -5 u=3 $ guide tubes
7 4 -1.0 -6 +7 -8 +9 u=1 lat=1
  fill= -5:4 -5:4 0:0
1 1 1 1 1 1 1 1 1 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 3 2 2 2 2 1
1 2 2 2 2 3 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 1 1 1 1 1 1 1 1 1
  
```

c  
 C BOX TYPE R

```

c
8 0 -10 11 -12 13 u=4 fill=1 (0.8128 0.8128 0)
9 3 -6.55 60 -61 62 -63 #8 u=4 $ Zr flow channel
10 4 -1. 64 -65 66 -67 #8 #9 u=4 $ water
11 5 -7.84 20 -23 67 -14 u=4 $ 0.075" STEEL
12 4 -1. 20 -23 14 -15 u=4 $ WATER POCKET
13 7 -2.7 20 -23 15 -16 u=4 $ Al CLAD
14 6 -2.66 20 -23 16 -17 u=4 $ BORAL Absorber
15 7 -2.7 20 -23 17 -18 u=4 $ Al Clad
16 4 -1. 20 -23 18 -118 u=4 $ Water
17 5 -7.84 118:-129:65:-66 u=4 $ Steel
18 4 -1. 64 -21 67 -118 u=4 $ Water
19 4 -1. 24 -65 67 -118 u=4 $ water
20 5 -7.84 21 -20 67 -118 u=4 $ Steel
21 5 -7.84 23 -24 67 -118 u=4 $ Steel
22 4 -1. 129 -64 33 -118 u=4 $ Water
c
23 5 -7.84 25 -64 30 -31 u=4 $ Steel
24 4 -1. 26 -25 30 -31 u=4 $ Water
25 7 -2.7 27 -26 30 -31 u=4 $ Al clad
26 6 -2.66 28 -27 30 -31 u=4 $ Boral
27 7 -2.7 29 -28 30 -31 u=4 $ Al clad
28 4 -1. 129 -29 30 -31 u=4 $ water
29 5 -7.84 129 -64 32 -30 u=4 $ Steel ends
30 5 -7.84 129 -64 31 -33 u=4 $ Steel ends
31 4 -1. 129 -64 66 -32 u=4 $ Water
c
  
```

c Type A box - Boral only on left side

```

c
32 0 -10 11 -12 13 u=6 fill=1 (0.8128 0.8128 0)
33 3 -6.55 60 -61 62 -63 #8 u=6 $ Zr flow channel
34 4 -1. 64 -65 66 -118 #8 #9 u=6 $ water
35 5 -7.84 118:-129:65:-66 u=6 $ Steel
36 4 -1. 129 -64 67 -118 u=6 $ Water
  
```

```

c
37  5 -7.84   25  -64   30  -31           u=6   $ Steel
38  4  -1.    26  -25   30  -31           u=6   $ Water
39  7  -2.7   27  -26   30  -31           u=6   $ Al clad
40  6 -2.66   28  -27   30  -31           u=6   $ Boral
41  7  -2.7   29  -28   30  -31           u=6   $ Al clad
42  4  -1.    129 -29   30  -31           u=6   $ water
43  4  -1.    129 -64   33  -67           u=6   $ Water
44  5 -7.84   129 -64   32  -30           u=6   $ Steel ends
45  5 -7.84   129 -64   31  -33           u=6   $ Steel ends
46  4  -1.    129 -64   66  -32           u=6   $ Water
c
c   Type B box - Boral on Top only
c
47  0 -10   11  -12   13           u=7 fill=1 (0.8128 0.8128 0)
48  3 -6.55   60  -61   62  -63   #8   u=7   $ Zr flow channel
49  4  -1.    64  -65   66  -67   #8 #9 u=7   $ water
50  5 -7.84   20  -23   67  -14           u=7   $ 0.075" STEEL
51  4  -1.    20  -23   14  -15           u=7   $ WATER POCKET
52  7  -2.7   20  -23   15  -16           u=7   $ Al CLAD
53  6 -2.66   20  -23   16  -17           u=7   $ BORAL Absorber
54  7  -2.7   20  -23   17  -18           u=7   $ water
55  4  -1.    20  -23   18  -118          u=7   $ Water
56  5 -7.84   118:-129:65:-66          u=7   $ Steel
57  4  -1.    64  -21   67  -118          u=7   $ Water
58  4  -1.    24  -65   67  -118          u=7   $ water
59  5 -7.84   21  -20   67  -118          u=7   $ Steel
60  5 -7.84   23  -24   67  -118          u=7   $ Steel
61  4  -1.    129 -64   66  -118          u=7   $ Water
c
c   Type E box - No Boral Panels
c
62  0 -10   11  -12   13           u=8 fill=1 (0.8128 0.8128 0)
63  3 -6.55   60  -61   62  -63   #8   u=8   $ Zr flow channel
64  4  -1.    129 -65   66  -118   #8 #9 u=8   $ water
65  5 -7.84   118:-129:65:-66          u=8   $ Steel
c
c   Type F box - No Boral Panels or fuel
c
66  4  -1.    129 -65   66  -118          u=9   $ water
67  5 -7.84   118:-129:65:-66          u=9   $ Steel
c
68  4  -1.0   -34  35  -36  37  u=5 lat=1 fill=-7:6 -7:6 0:0
      5 5 5 5 5 5 5 5 5 5 5 5
      5 9 9 9 9 9 9 9 9 9 9 9 5
      5 9 9 9 9 9 7 4 9 9 9 9 5
      5 9 9 9 7 4 4 4 4 4 9 9 5
      5 9 9 7 4 4 4 4 4 4 4 9 5
      5 9 9 7 4 4 4 4 4 4 4 9 5
      5 9 7 4 4 4 4 4 4 4 4 9 5
      5 9 8 4 4 4 4 4 4 4 4 6 9 5
      5 9 9 7 4 4 4 4 4 4 4 9 9 5
      5 9 9 8 4 4 4 4 4 4 6 9 9 5
      5 9 9 9 8 4 4 4 6 6 9 9 9 5
      5 9 9 9 9 9 8 6 9 9 9 9 5
      5 9 9 9 9 9 9 9 9 9 9 9 5
      5 5 5 5 5 5 5 5 5 5 5 5
69  0           -41           50  -49 fill=5 (8.1661 8.1661 0)

```

Attachment 6 to Holtec Letter 5014874

c							
274	4	-1.0	-41	360	-50	\$ Water below Fuel (7.3 in.)	
275	5	-7.84	-42	362	-360	\$ MPC Steel below Fuel (2.5 in.)	
276	5	-7.84	-205	300	-362	\$ Transfer Cask Steel (2.0 in.)	
277	8	-11.34	-205	301	-300	\$ Transfer Cask Lead (2.5 in.)	
278	5	-7.84	-205	302	-301	\$ Transfer Cask Steel (1.0 in.)	
c							
279	4	-1.0	-41	49	-361	\$ Water above Fuel (8.46 in.)	
280	5	-7.84	-42	361	-363	\$ MPC Steel above Fuel (10.0 in)	
281	4	-1.0	-42	363	-400	\$ Water (1.5 in.)	
282	5	-7.84	-205	400	-401	\$ Transfer Cask Steel (0.75 in.)	
283	9	-1.61	-205	401	-402	\$ Transfer Cask Neutron Shield (3.25 in.)	
284	5	-7.84	-205	402	-403	\$ Transfer Cask Steel (0.5 in.)	
c							
290	5	-7.84	41	-42	360	-361	\$ Radial Steel - MPC shell
291	4	-1.00	42	-200	362	-400	\$ Radial Water
292	5	-7.84	200	-201	362	-400	\$ Radial Steel - inner shell of Trnsfr Cask
293	8	-11.34	201	-202	362	-400	\$ Radial Lead - Transfer Cask lead
294	5	-7.84	202	-203	362	-400	\$ Radial Steel - outer shell of Trnsfr Cask
295	4	-1.00	203	-204	362	-400	\$ Radial Water - Water Jacket
296	5	-7.84	204	-205	362	-400	\$ Radial Steel - outer shell of Water Jacket
c							
500	4	-1.00	500	-501	-505	(205 :-302: 403)	\$ outer water reflector
501	0		505	:-500:	501		\$ outside world
1	cz		0.5283				\$ Fuel OD
2	cz		0.5398				\$ Clad ID
3	cz		0.6134				\$ Clad OD
4	cz		0.6744				\$ Thimble ID
5	cz		0.7506				\$ Thimble OD
6	px		0.8128				\$ Pin Pitch
7	px		-0.8128				
8	py		0.8128				
9	py		-0.8128				
10	px		6.6231				\$ Channel ID
11	px		-6.6231				
12	py		6.6231				
13	py		-6.6231				
14	py		7.8016				
15	py		7.8155				
16	py		7.8410				
17	py		8.0467				
18	py		8.0721				
118	py		8.0861				
20	px		-6.0325				
21	px		-6.2230				
23	px		6.0325				
24	px		6.2230				
25	px		-7.8016				
26	px		-7.8155				
27	px		-7.8410				
28	px		-8.0467				
29	px		-8.0721				
129	px		-8.0861				
30	py		-6.0325				
31	py		6.0325				
32	py		-6.2230				
33	py		6.2230				



```

34  px      7.6111
35  px     -8.7211
36  py      8.7211
37  py     -7.6111
49  pz      381.          $ Top of Active Fuel
50  pz       0           $ Start of Active Fuel
60  px     -6.9279      $ Channel OD
61  px      6.9279
62  py     -6.9279
63  py      6.9279
64  px     -7.6111      $ Cell Box ID
65  px      7.6111
66  py     -7.6111
67  py      7.6111
360 pz     -18.54      $ lower water thkness = 7.30 in.
361 pz     402.49      $ upper water thkness = 8.46 in.
362 pz     -24.892     $ thkness of MPC baseplate = 2.5 in.
363 pz     427.89      $ thkness of MPC lid = 10. in.
41  cz     85.57       $ I.D. = 67.375 in.
42  cz     86.84       $ I.D. = 68.375 in.
200 cz     87.31       $ I.D. = 68.75 in.
201 cz     89.22       $ I.D. = 70.25 in.
202 cz    100.65       $ I.D. = 79.25 in.
203 cz    103.19       $ I.D. = 81.25 in.
204 cz    116.80       $ I.D. = 91.97 in.
205 cz    118.07       $ I.D. = 92.972 in.
300 pz    -29.97      $ thkness steel - 2.0 in.
301 pz    -36.32      $ thkness lead - 2.5 in.
302 pz    -38.86      $ thkness steel - 1.0 in.
400 pz    431.70      $ thkness water - 1.5 in.
401 pz    433.61      $ thkness steel - 0.75 in.
402 pz    441.87      $ thkness neutron shield - 3.25 in.
403 pz    443.14      $ thkness steel - 0.5 in.
c
*500 pz   -68.86      $ lower boundary
*501 pz   473.14      $ upper boundary
*505 cz   148.07      $ outer radial boundary

imp:n      1 87r 0
kcode     10000 0.94 20 120
c
sdef par=1  erg=d1  axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3 h 0 381.
sp3 0 1
c
c
si4 s          15 16
              13 14 15 16 17 18
              12 13 14 15 16 17 18 19
              12 13 14 15 16 17 18 19
11 12 13 14 15 16 17 18 19 20
11 12 13 14 15 16 17 18 19 20
              12 13 14 15 16 17 18 19
              12 13 14 15 16 17 18 19
              13 14 15 16 17 18

```

```

                15 16
sp4  1 67r
c
ds5  s                30 30
                29 29 29 29 29 29
                28 28 28 28 28 28 28 28
                27 27 27 27 27 27 27 27
                26 26 26 26 26 26 26 26 26
                25 25 25 25 25 25 25 25 25
                24 24 24 24 24 24 24 24
                23 23 23 23 23 23 23 23
                22 22 22 22 22 22
                21 21
c
si11 -80.6831 -67.6783
si12 -64.1985 -51.1937
si13 -47.7139 -34.7091
si14 -31.2293 -18.2245
si15 -14.7447 -1.7399
si16  1.7399  14.7447
si17  18.2245 31.2293
si18  34.7091 47.7139
si19  51.1937 64.1985
si20  67.6783 80.6831
c
si21 -80.6831 -67.6783
si22 -64.1985 -51.1937
si23 -47.7139 -34.7091
si24 -31.2293 -18.2245
si25 -14.7447 -1.7399
si26  1.7399  14.7447
si27  18.2245 31.2293
si28  34.7091 47.7139
si29  51.1937 64.1985
si30  67.6783 80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
ml          92235.50c   -0.03702   $ 4.20% E Fuel
            92238.50c   -0.84448

```

Attachment 6 to Holtec Letter 5014874

	8016.50c	-0.1185				
m3	40000.56c	1.	\$ Zr Clad			
m4	1001.50c	0.6667	\$ Water			
	8016.50c	0.3333				
m5	24000.50c	0.01761	\$ Steel			
	25055.50c	0.001761				
	26000.55c	0.05977				
	28000.50c	0.008239				
m6	5010.50c	8.0707E-03	\$ Boral			
	5011.50c	3.2553E-02				
	6000.50c	1.0146E-02				
	13027.50c	3.8054E-02				
m7	13027.50c	1.	\$ Al Clad			
m8	82000.50c	1.0	\$ Lead			
m9	6000.50c	-27.660	\$ Neutron Shield Holtite-A (NS-4-FR)			
	1001.50c	-5.920				
	13027.50c	-21.285				
	7014.50c	-1.98				
	8016.50c	-42.372				
	5010.50c	-0.141				
	5011.50c	-0.642				
mt4	lwtr.01t					
prdmp	j -120 j 2					
fm4	1000 1 -6					
f4:n	1					
sd4	1000					
e4	1.000E-11	1.000E-10	5.000E-10	7.500E-10	1.000E-09	1.200E-09
	1.500E-09	2.000E-09	2.500E-09	3.000E-09		
	4.700E-09	5.000E-09	7.500E-09	1.000E-08	2.530E-08	
	3.000E-08	4.000E-08	5.000E-08	6.000E-08	7.000E-08	
	8.000E-08	9.000E-08	1.000E-07	1.250E-07	1.500E-07	
	1.750E-07	2.000E-07	2.250E-07	2.500E-07	2.750E-07	
	3.000E-07	3.250E-07	3.500E-07	3.750E-07	4.000E-07	
	4.500E-07	5.000E-07	5.500E-07	6.000E-07	6.250E-07	
	6.500E-07	7.000E-07	7.500E-07	8.000E-07	8.500E-07	
	9.000E-07	9.250E-07	9.500E-07	9.750E-07	1.000E-06	
	1.010E-06	1.020E-06	1.030E-06	1.040E-06	1.050E-06	
	1.060E-06	1.070E-06	1.080E-06	1.090E-06	1.100E-06	
	1.110E-06	1.120E-06	1.130E-06	1.140E-06	1.150E-06	
	1.175E-06	1.200E-06	1.225E-06	1.250E-06	1.300E-06	
	1.350E-06	1.400E-06	1.450E-06	1.500E-06	1.590E-06	
	1.680E-06	1.770E-06	1.860E-06	1.940E-06	2.000E-06	
	2.120E-06	2.210E-06	2.300E-06	2.380E-06	2.470E-06	
	2.570E-06	2.670E-06	2.770E-06	2.870E-06	2.970E-06	
	3.000E-06	3.050E-06	3.150E-06	3.500E-06	3.730E-06	
	4.000E-06	4.750E-06	5.000E-06	5.400E-06	6.000E-06	
	6.250E-06	6.500E-06	6.750E-06	7.000E-06	7.150E-06	
	8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05	
	1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05	
	1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05	
	2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05	
	3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05	
	3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05	
	4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05	
	4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05	
	6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05	
	8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04	
	1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04	

Attachment 6 to Holtec Letter 5014874

2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04
5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03
1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03
2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03
8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02
3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02
7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	

```

HI-STORM Storage Cask containing MPC24, 17x17 assembly @ 4.0 wt% Enrich.
c
c
c
c MPC-24/24E cell configuration
c
c HI-STORM with active length 150 inch
c
c
c Cask Input Preprocessor
c cskinp 17a 17a mpc24n mpc24n historm historm 4.0 4sf7a45 empty
c ----- cpp\17a.bat
c   added 17a.ce
c   added 17a.su
c   added 17a.sp
c ----- cpp\mpc24n.bat
c   added mpc24n.co
c   added mpc24n.ce
c   added mpc24n.su
c   added mpc24n.sp
c ----- cpp\historm.bat
c   added historm.co
c   added historm.ce
c   added historm.su
c   added historm.sp
c end of comments
c
c start of cells
c
c 17x17a
c
c number of cells: 6
c cell numbers:      1 to 7
c univers numbers:  1 to 3
c surface numbers:  1 to 9
c
c number of cells: 1
1   1 -10.522      -1  u=2      $ fuel
2   4 -1.0         1  -2  u=2      $ gap
3   3 -6.55       2  -3  u=2      $ Zr Clad
4   2 -0.0002     3  u=2      $ water in fuel region
5   2 -0.0002    -4:5    u=3      $ water in guide tubes
6   3 -6.55      4 -5    u=3      $ guide tubes
7   2 -0.0002   -6  +7   -8   +9   u=1  lat=1
    fill= -9:9      -9:9    0:0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 1
1 2 2 2 3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 1

```

```

1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 3 2 2 2 2 2 2 2 2 2 3 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1

```

c

c MPC-24

c

c number of cells: 102

c cell numbers : 400 to 699

c universe numbers : 4 to 9

c surface numbers : 400 to 699

c

c Right Side

c

```

408 0 -410 411 -412 413 u=4 fill=1 (1)
409 5 -7.84 410 -424 413 -426 u=4
410 2 -0.0002 424 -428 448 -445 u=4
411 7 -2.7 428 -528 448 -445 u=4
412 6 -2.66 528 -532 448 -445 u=4
413 7 -2.7 532 -432 448 -445 u=4
414 2 -0.0002 432 -436 448 -445 u=4
415 5 -7.84 436 -440 448 -445 u=4
416 2 -0.0002 440 413 u=4
417 2 -0.0002 424 -440 413 -447 u=4
418 2 -0.0002 424 -440 446 u=4
419 5 -7.84 424 -440 447 -448 u=4
420 5 -7.84 424 -440 445 -446 u=4

```

c

c Left Side

c

```

421 5 -7.84 425 -411 413 u=4
422 2 -0.0002 429 -425 448 -445 u=4
423 7 -2.7 529 -429 448 -445 u=4
424 6 -2.66 533 -529 448 -445 u=4
425 7 -2.7 433 -533 448 -445 u=4
426 2 -0.0002 437 -433 448 -445 u=4
427 5 -7.84 441 -437 448 -445 u=4
428 2 -0.0002 -441 413 u=4
429 2 -0.0002 441 -425 413 -447 u=4
430 2 -0.0002 441 -425 446 u=4
431 5 -7.84 441 -425 447 -448 u=4
432 5 -7.84 441 -425 445 -446 u=4

```

c

c Top

c

```

433 5 -7.84 411 -410 412 -426 u=4
434 2 -0.0002 451 -452 426 -430 u=4
435 7 -2.7 451 -452 430 -530 u=4
436 6 -2.66 451 -452 530 -534 u=4
437 7 -2.7 451 -452 534 -434 u=4
438 2 -0.0002 451 -452 434 -438 u=4
439 5 -7.84 451 -452 438 -442 u=4
440 2 -0.0002 411 -424 442 u=4
441 2 -0.0002 411 -450 426 -442 u=4
442 2 -0.0002 453 -424 426 -442 u=4
443 5 -7.84 450 -451 426 -442 u=4

```

```

444 5 -7.84 452 -453 426 -442 u=4
c
c Bottom
c
445 5 -7.84 427 -413 u=4
446 2 -0.0002 451 -452 431 -427 u=4
447 7 -2.7 451 -452 531 -431 u=4
448 6 -2.66 451 -452 535 -531 u=4
449 7 -2.7 451 -452 435 -535 u=4
450 2 -0.0002 451 -452 439 -435 u=4
451 5 -7.84 451 -452 443 -439 u=4
452 2 -0.0002 411 -443 u=4
453 2 -0.0002 411 -450 443 -427 u=4
454 2 -0.0002 453 443 -427 u=4
455 5 -7.84 450 -451 443 -427 u=4
456 5 -7.84 452 -453 443 -427 u=4
457 5 -7.84 425 -411 -427 u=4
458 2 -0.0002 -425 -427 u=4
c
c TYPE B CELL - Short Boral on top and right
c
c Right Side
c
459 0 -410 411 -412 413 u=5 fill=1 (1)
460 5 -7.84 410 -424 413 -426 u=5
470 2 -0.0002 424 -428 548 -545 u=5
471 7 -2.7 428 -528 548 -545 u=5
472 6 -2.66 528 -532 548 -545 u=5
473 7 -2.7 532 -432 548 -545 u=5
474 2 -0.0002 432 -436 548 -545 u=5
475 5 -7.84 436 -440 548 -545 u=5
476 2 -0.0002 440 413 u=5
477 2 -0.0002 424 -440 413 -547 u=5
478 2 -0.0002 424 -440 546 u=5
479 5 -7.84 424 -440 547 -548 u=5
480 5 -7.84 424 -440 545 -546 u=5
c
c Left Side
c
481 5 -7.84 425 -411 413 u=5
482 2 -0.0002 429 -425 448 -445 u=5
483 7 -2.7 529 -429 448 -445 u=5
484 6 -2.66 533 -529 448 -445 u=5
485 7 -2.7 433 -533 448 -445 u=5
486 2 -0.0002 437 -433 448 -445 u=5
487 5 -7.84 441 -437 448 -445 u=5
488 2 -0.0002 -441 413 u=5
489 2 -0.0002 441 -425 413 -447 u=5
490 2 -0.0002 441 -425 446 u=5
491 5 -7.84 441 -425 447 -448 u=5
492 5 -7.84 441 -425 445 -446 u=5
c
c Top
c
493 5 -7.84 411 -410 412 -426 u=5
494 2 -0.0002 551 -552 426 -430 u=5
495 7 -2.7 551 -552 430 -530 u=5
496 6 -2.66 551 -552 530 -534 u=5

```

```

497  7 -2.7    551 -552  534 -434    u=5
498  2 -0.0002  551 -552  434 -438    u=5
499  5 -7.84    551 -552  438 -442    u=5
500  2 -0.0002  411 -424  442          u=5
501  2 -0.0002  411 -550  426 -442    u=5
502  2 -0.0002  553 -424  426 -442    u=5
503  5 -7.84    550 -551  426 -442    u=5
504  5 -7.84    552 -553  426 -442    u=5
c
c    Bottom
c
505  5 -7.84    427          -413    u=5
506  2 -0.0002  451 -452  431 -427    u=5
507  7 -2.7     451 -452  531 -431    u=5
508  6 -2.66    451 -452  535 -531    u=5
509  7 -2.7     451 -452  435 -535    u=5
510  2 -0.0002  451 -452  439 -435    u=5
511  5 -7.84    451 -452  443 -439    u=5
512  2 -0.0002  411          -443    u=5
513  2 -0.0002  411 -450  443 -427    u=5
514  2 -0.0002  453          443 -427    u=5
515  5 -7.84    450 -451  443 -427    u=5
516  5 -7.84    452 -453  443 -427    u=5
517  5 -7.84    425 -411          -427    u=5
518  2 -0.0002          -425          -427    u=5
c
c
c
c    TYPE D CELL - Short Boral on left and bottom, different cell ID
c
c number of cells: 51
c
c    Right Side
c
1570  0          -1410  1411 -1412  1413    u=17 fill=1 (1)
1571  5 -7.84    1410 -1424  1413 -1426    u=17
1572  2 -0.0002  1424 -1428  1448 -1445    u=17
1573  7 -2.7     1428 -1528  1448 -1445    u=17
1574  6 -2.66    1528 -1532  1448 -1445    u=17
1575  7 -2.7     1532 -1432  1448 -1445    u=17
1576  2 -0.0002  1432 -1436  1448 -1445    u=17
1577  5 -7.84    1436 -1440  1448 -1445    u=17
1578  2 -0.0002  1440          1413    u=17
1579  2 -0.0002  1424 -1440  1413 -1447    u=17
1580  2 -0.0002  1424 -1440  1446          u=17
1581  5 -7.84    1424 -1440  1447 -1448    u=17
1582  5 -7.84    1424 -1440  1445 -1446    u=17
c
c    Left Side
c
1583  5 -7.84    1425 -1411  1413          u=17
1584  2 -0.0002  1429 -1425  1548 -1545    u=17
1585  7 -2.7     1529 -1429  1548 -1545    u=17
1586  6 -2.66    1533 -1529  1548 -1545    u=17
1587  7 -2.7     1433 -1533  1548 -1545    u=17
1588  2 -0.0002  1437 -1433  1548 -1545    u=17
1589  5 -7.84    1441 -1437  1548 -1545    u=17
1590  2 -0.0002          -1441  1413          u=17

```



Attachment 6 to Holtec Letter 5014874

```

1591  2 -0.0002      1441  -1425   1413  -1547      u=17
1592  2 -0.0002      1441  -1425   1546                u=17
1593  5  -7.84       1441  -1425   1547  -1548      u=17
1594  5  -7.84       1441  -1425   1545  -1546      u=17
c
c      Top
c
1595  5  -7.84       1411  -1410   1412  -1426      u=17
1596  2 -0.0002      1451  -1452   1426  -1430      u=17
1597  7  -2.7         1451  -1452   1430  -1530      u=17
1598  6  -2.66        1451  -1452   1530  -1534      u=17
1599  7  -2.7         1451  -1452   1534  -1434      u=17
1600  2 -0.0002      1451  -1452   1434  -1438      u=17
1601  5  -7.84       1451  -1452   1438  -1442      u=17
1602  2 -0.0002      1411  -1424   1442                u=17
1603  2 -0.0002      1411  -1450   1426  -1442      u=17
1604  2 -0.0002      1453  -1424   1426  -1442      u=17
1605  5  -7.84       1450  -1451   1426  -1442      u=17
1606  5  -7.84       1452  -1453   1426  -1442      u=17
c
c      Bottom
c
1607  5  -7.84       1427                -1413      u=17
1608  2 -0.0002      1551  -1552   1431  -1427      u=17
1609  7  -2.7         1551  -1552   1531  -1431      u=17
1610  6  -2.66        1551  -1552   1535  -1531      u=17
1611  7  -2.7         1551  -1552   1435  -1535      u=17
1612  2 -0.0002      1551  -1552   1439  -1435      u=17
1613  5  -7.84       1551  -1552   1443  -1439      u=17
1614  2 -0.0002      1411                -1443      u=17
1615  2 -0.0002      1411  -1550   1443  -1427      u=17
1616  2 -0.0002      1553                1443  -1427      u=17
1617  5  -7.84       1550  -1551   1443  -1427      u=17
1618  5  -7.84       1552  -1553   1443  -1427      u=17
1619  5  -7.84       1425  -1411                -1427      u=17
1620  2 -0.0002                -1425                -1427      u=17
c
c      number of cells: 29
c
c      empty cell no boron, no top
c
c
751   2 -0.0002     -410   411   -412   413      u=14
752   5  -7.84       410  -424   413  -426      u=14
753   5  -7.84       425  -411   413                u=14
754   2 -0.0002       411  -410   412  -426      u=14
755   5  -7.84       427                -413      u=14
756   5  -7.84       425  -411                -427      u=14
757   2 -0.0002       411   426                u=14
758   2 -0.0002       411  -427                u=14
759   2 -0.0002      -425   413                u=14
760   2 -0.0002       424   413   -426                u=14
761   2 -0.0002      -425  -427                u=14
c
c
701   5  -7.84       701  -702  711  -713      u=9   $ steel post
702   5  -7.84       702  -703  711  -712      u=9   $ steel post
c

```

```

711 0          701 -705 711 -715 (702:713) (703:712)
      fill=4 (13.8506 13.8506 0) u=9
712 0          704 (-706:-716) (705:715) -717 -710
      fill=4 (17.9489 41.5518 0 0 1 0 -1 0 0 0 0 1) u=9
713 0          (705:715) -707 714 (-706:-716) 710
      fill=4 (41.5518 17.9489 0 0 -1 0 1 0 0 0 0 1) u=9
714 0          701 -705 717 -719
      fill=5 (13.8506 69.253 0) u=9
715 0          707 -709 711 -715
      fill=5 (69.253 13.8506 0) u=9
716 0          706 -708 716 -718
      fill=17 (45.6501 45.6501 0 -1 0 0 0 -1 0 0 0 1) u=9
717 0          705 -706 717 -719
      fill=14 (41.5518 69.253 0) u=9
718 0          707 -709 715 -716
      fill=14 (69.253 41.5518 0 0 1 0 1 0 0 0 0 1) u=9
719 0          701 -704 715 -717
      fill=14 (-9.75233 41.5518 0 -1 0 0 0 1 0 0 0 1) u=9
720 0          705 -707 711 -714
      fill=14 (41.5518 -9.75233 0 0 -1 0 1 0 0 0 0 1) u=9
721 2 -0.0002  (706:719) (708:718) (709:716) u=9
c
c
c
731 2 -0.0002  720 721 fill=9 (0 0 0) u=19
732 2 -0.0002  -720 721 fill=9 (0 0 0
      -1 0 0 0 1 0 0 0 1) u=19
733 2 -0.0002  720 -721 fill=9 (0 0 0
      1 0 0 0 -1 0 0 0 1) u=19
734 2 -0.0002  -720 -721 fill=9 (0 0 0
      -1 0 0 0 -1 0 0 0 1) u=19
c
673 0          -41          39 -40 fill=19
c
c number of cells: 19
374 2 -0.0002 -41          330 -39          $ Void below Fuel (4 in.)
375 5 -7.84 -309          332 -330          $ MPC Steel below Fuel (2.5 in.)
376 5 -7.84 -304          310 -332          $ Cask Steel (5.0 in.)
377 8 -2.35 -304          311 -310          $ Cask Concrete (17.0 in.)
378 5 -7.84 -304          312 -311          $ Cask Steel (2.0 in.)
c
379 2 -0.0002 -41          40 -331          $ Void above Fuel (6 in.)
380 5 -7.84 -309          331 -333          $ MPC Steel above Fuel (9.5 - 0.06 in)
381 4 -1.0 -309          333 -320          $ Water (1.0 in.)
382 5 -7.84 -304          320 -321          $ Cask Steel (1.25 in.)
383 8 -2.35 -304          321 -322          $ Cask Concrete (10.5 in.)
384 5 -7.84 -304          322 -323          $ Cask Steel (4.0 in.)
c
390 5 -7.84 41 -309 330 -331          $ Radial Steel - MPC shell
391 4 -1.00 309 -300 332 -320          $ Radial Water
392 5 -7.84 300 -301 332 -320          $ Radial Steel - overpack inner shell
394 5 -7.84 301 -302 332 -320          $ Radial Steel -
395 8 -2.35 302 -303 332 -320          $ Radial Shield - Concrete Overpack
396 5 -7.84 303 -304 332 -320          $ Radial Steel - overpack outer shell
c
300 4 -1.00 340 -341 -345          (304 :-312: 323) $ outer water reflector
301 0          345 :-340: 341          $ outside world
c end of cells

```

c --- empty line

c --- empty line

c start of surfaces

1	cz	0.3922	\$ fuel
2	cz	0.4001	\$ clad ID
3	cz	0.4572	\$ clad OD
4	cz	0.5613	\$ guide ID
5	cz	0.6020	\$ guide OD
6	px	0.6299	\$ pin pitch
7	px	-0.6299	
8	py	0.6299	
9	py	-0.6299	

c

c

c cell-id	8.98
c cell-pitch	10.906
c wall-thkns	5/16
c angle-thkns	5/16
c boral-gap	0.0035
c boral-gap-o	0.0035
c boral-thkns	0.075
c boral-clad	0.01
c sheathing	0.0235
c boral-wide	7.5
c boral-narrow	6.25

c

c gap size	1.09
c basket-od	67.335

c

410	px	11.40460	\$x 8.98/2	
411	px	-11.40460	\$x {410} *-1	
412	py	11.40460	\$x {410}	
413	py	-11.40460	\$x {411}	
416	px	13.85062	\$x (10.906 + 5/16 - 5/16) /2	
417	px	-13.85062	\$x -10.906 + {416}	
418	py	13.85062	\$x {416}	
419	py	-13.85062	\$x {417}	
424	px	12.19835	\$x {410} + 5/16	\$ angle
425	px	-12.19835	\$x {411} - 5/16	\$ box wall
426	py	12.19835	\$x {412} + 5/16	
427	py	-12.19835	\$x {413} - 5/16	
428	px	12.20724	\$x {424} + 0.0035	\$ wall to boral gap
429	px	-12.20724	\$x {425} - 0.0035	
430	py	12.20724	\$x {426} + 0.0035	
431	py	-12.20724	\$x {427} - 0.0035	
432	px	12.39774	\$x {428} + 0.075	\$ boral
433	px	-12.39774	\$x {429} - 0.075	
434	py	12.39774	\$x {430} + 0.075	
435	py	-12.39774	\$x {431} - 0.075	
436	px	12.40663	\$x {432} + 0.0035	\$ boral to sheathing gap
437	px	-12.40663	\$x {433} - 0.0035	
438	py	12.40663	\$x {434} + 0.0035	
439	py	-12.40663	\$x {435} - 0.0035	
440	px	12.46632	\$x {436} + 0.0235	\$ sheathing
441	px	-12.46632	\$x {437} - 0.0235	
442	py	12.46632	\$x {438} + 0.0235	
443	py	-12.46632	\$x {439} - 0.0235	

```

445 py 9.52500 $x 7.5/2
446 py 9.58469 $x {445} + 0.0235 $ sheathing
447 py -9.58469 $x {446} *-1
448 py -9.52500 $x {445} *-1
450 px -9.58469 $x {447}
451 px -9.52500 $x {448}
452 px 9.52500 $x {445}
453 px 9.58469 $x {446}
528 px 12.23264 $x {428} + 0.01 $ Aluminum on the outside of boral
529 px -12.23264 $x {429} - 0.01
530 py 12.23264 $x {430} + 0.01
531 py -12.23264 $x {431} - 0.01
532 px 12.37234 $x {432} - 0.01
533 px -12.37234 $x {433} + 0.01
534 py 12.37234 $x {434} - 0.01
535 py -12.37234 $x {435} + 0.01
545 py 7.93750 $x 6.25/2
546 py 7.99719 $x {545} + 0.0235 $ sheathing
547 py -7.99719 $x {546} *-1
548 py -7.93750 $x {545} *-1
550 px -7.99719 $x {547}
551 px -7.93750 $x {548}
552 px 7.93750 $x {545}
553 px 7.99719 $x {546}
c
c cell-id-2 8.98
c gap-o 1.09
c
701 px -5.0
702 px 1.90627 $x (10.906 - 8.98)/2 - 5/16 + 0.1
703 px 3.45694 $x 2.722/2
704 px 4.09829 $x 10.906 - 8.98 - 5/16
705 px 27.70124 $x 10.906
706 px 31.79953 $x 2 * 10.906 - (8.98+8.98)/2 - 5/16
707 px 55.40248 $x 2 * 10.906
708 px 59.50077 $x {707} + {704}
709 px 83.10372 $x 3 * 10.906
710 p 1 -1 0 0.1 $ diagonal x=y, offset by 0.1 to avoid intersecting corners
711 py -4.99999 $x {701}
712 py 1.90627 $x {702}
713 py 3.45694 $x {703}
714 py 4.09829 $x {704}
715 py 27.70124 $x {705}
716 py 31.79953 $x {706}
717 py 55.40248 $x {707}
718 py 59.50077 $x {708}
719 py 83.10372 $x {709}
720 px 0.0
721 py 0.0
1410 px 11.40460 $x 8.98/2
1411 px -11.40460 $x {1410} *-1
1412 py 11.40460 $x {1410}
1413 py -11.40460 $x {1411}
1424 px 12.19835 $x {1410} + 5/16 $ angle
1425 px -12.19835 $x {1411} - 5/16 $ box wall
1426 py 12.19835 $x {1412} + 5/16
1427 py -12.19835 $x {1413} - 5/16
1428 px 12.20724 $x {1424} + 0.0035 $ wall to boral gap

```

## Attachment 6 to Holtec Letter 5014874

1429	px	-12.20724	\$x {1425}	- 0.0035	
1430	py	12.20724	\$x {1426}	+ 0.0035	
1431	py	-12.20724	\$x {1427}	- 0.0035	
1432	px	12.39774	\$x {1428}	+ 0.075	\$ boral
1433	px	-12.39774	\$x {1429}	- 0.075	
1434	py	12.39774	\$x {1430}	+ 0.075	
1435	py	-12.39774	\$x {1431}	- 0.075	
1436	px	12.40663	\$x {1432}	+ 0.0035	\$ boral to sheathing gap
1437	px	-12.40663	\$x {1433}	- 0.0035	
1438	py	12.40663	\$x {1434}	+ 0.0035	
1439	py	-12.40663	\$x {1435}	- 0.0035	
1440	px	12.46632	\$x {1436}	+ 0.0235	\$ sheathing
1441	px	-12.46632	\$x {1437}	- 0.0235	
1442	py	12.46632	\$x {1438}	+ 0.0235	
1443	py	-12.46632	\$x {1439}	- 0.0235	
1445	py	9.52500	\$x 7.5/2		
1446	py	9.58469	\$x {1445}	+ 0.0235	\$ sheathing
1447	py	-9.58469	\$x {1446}	*-1	
1448	py	-9.52500	\$x {1445}	*-1	
1450	px	-9.58469	\$x {1447}		
1451	px	-9.52500	\$x {1448}		
1452	px	9.52500	\$x {1445}		
1453	px	9.58469	\$x {1446}		
1528	px	12.23264	\$x {1428}	+ 0.01	\$ Aluminum on the outside of boral
1529	px	-12.23264	\$x {1429}	- 0.01	
1530	py	12.23264	\$x {1430}	+ 0.01	
1531	py	-12.23264	\$x {1431}	- 0.01	
1532	px	12.37234	\$x {1432}	- 0.01	
1533	px	-12.37234	\$x {1433}	+ 0.01	
1534	py	12.37234	\$x {1434}	- 0.01	
1535	py	-12.37234	\$x {1435}	+ 0.01	
1545	py	7.93750	\$x 6.25/2		
1546	py	7.99719	\$x {1545}	+ 0.0235	\$ sheathing
1547	py	-7.99719	\$x {1546}	*-1	
1548	py	-7.93750	\$x {1545}	*-1	
1550	px	-7.99719	\$x {1547}		
1551	px	-7.93750	\$x {1548}		
1552	px	7.93750	\$x {1545}		
1553	px	7.99719	\$x {1546}		
39	pz	0.			
40	pz	381.0			\$ 150 inch active fuel length
330	pz	-10.16			\$ lower water thkness = 4 in.
331	pz	396.24			\$ upper water thkness = 6 in.
332	pz	-16.51			\$ thkness of MPC baseplate = 2.5 in.
333	pz	420.02			\$ thkness of MPC lid = 9.5 -0.06 in.
41	cz	85.57			\$ I.D. = 67.37 in
309	cz	86.84			\$ I.D. = 68.375 in.
300	cz	93.35			\$ I.D. = 73.50 in.
301	cz	96.52			\$ I.D. = 76.00 in.
302	cz	98.43			\$ I.D. = 77.50 in.
303	cz	166.37			\$ I.D. = 131.00 in.
304	cz	168.28			\$ I.D. = 132.50 in.
310	pz	-29.21			\$ thkness steel - 5.0 in.
311	pz	-72.39			\$ thkness concrete - 17.0 in.
312	pz	-77.47			\$ thkness steel - 2.0 in.
320	pz	422.76			\$ thkness water - 1.0 in.
321	pz	425.94			\$ thkness steel - 1.25 in.
322	pz	452.61			\$ thkness concrete - 10.5 in.

```

323 pz          462.765 $ thkness steel - 4.0 in.
c
*340 pz -107.47 $ lower boundary
*341 pz  492.765 $ upper boundary
*345 cz  198.28 $ outer radial boundary
c end of surfaces
c --- empty line

c --- empty line
tr1 0 0 0
kcode 10000 .94 20 120
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
sp3 0 1
c
si4 s          13 14
           12 13 14 15
           11 12 13 14 15 16
           11 12 13 14 15 16
           12 13 14 15
           13 14

sp4 1 23r
c
ds5 s          26 26
           25 25 25 25
           24 24 24 24 24 24
           23 23 23 23 23 23
           22 22 22 22
           21 21

c
si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639
si14  2.86639  24.50719
si15  30.23997  51.88077
si16  57.61355  79.25435
c
si21 -79.25435 -57.61355
si22 -51.88077 -30.23997
si23 -24.50719 -2.86639
si24  2.86639  24.50719
si25  30.23997  51.88077
si26  57.61355  79.25435
c
sp11 0 1
sp12 0 1
sp13 0 1
sp14 0 1
sp15 0 1
sp16 0 1
sp21 0 1
sp22 0 1
sp23 0 1
sp24 0 1
sp25 0 1
sp26 0 1

```

Attachment 6 to Holtec Letter 5014874

c						
m3	40000.56c	1.		\$ Zr Clad		
m4	1001.50c	0.6667		\$ Water		
	8016.50c	0.3333				
m5	24000.50c	0.01761		\$ Steel		
	25055.50c	0.001761				
	26000.55c	0.05977				
	28000.50c	0.008239				
m6	5010.50c	-0.054427		\$ Boral Central Section @ 0.02 g/cmsq		
	5011.50c	-0.241373				
	13027.50c	-0.6222				
	6000.50c	-0.0821				
m7	13027.50c	1.0				
mt4	lwtr.01t					
prdmp	j -120 j 2					
fm4	1000 1 -6					
f4:n	1					
sd4	1000					
e4	1.000E-11	1.000E-10	5.000E-10	7.500E-10	1.000E-09	1.200E-09
	1.500E-09	2.000E-09	2.500E-09	3.000E-09		
	4.700E-09	5.000E-09	7.500E-09	1.000E-08	2.530E-08	
	3.000E-08	4.000E-08	5.000E-08	6.000E-08	7.000E-08	
	8.000E-08	9.000E-08	1.000E-07	1.250E-07	1.500E-07	
	1.750E-07	2.000E-07	2.250E-07	2.500E-07	2.750E-07	
	3.000E-07	3.250E-07	3.500E-07	3.750E-07	4.000E-07	
	4.500E-07	5.000E-07	5.500E-07	6.000E-07	6.250E-07	
	6.500E-07	7.000E-07	7.500E-07	8.000E-07	8.500E-07	
	9.000E-07	9.250E-07	9.500E-07	9.750E-07	1.000E-06	
	1.010E-06	1.020E-06	1.030E-06	1.040E-06	1.050E-06	
	1.060E-06	1.070E-06	1.080E-06	1.090E-06	1.100E-06	
	1.110E-06	1.120E-06	1.130E-06	1.140E-06	1.150E-06	
	1.175E-06	1.200E-06	1.225E-06	1.250E-06	1.300E-06	
	1.350E-06	1.400E-06	1.450E-06	1.500E-06	1.590E-06	
	1.680E-06	1.770E-06	1.860E-06	1.940E-06	2.000E-06	
	2.120E-06	2.210E-06	2.300E-06	2.380E-06	2.470E-06	
	2.570E-06	2.670E-06	2.770E-06	2.870E-06	2.970E-06	
	3.000E-06	3.050E-06	3.150E-06	3.500E-06	3.730E-06	
	4.000E-06	4.750E-06	5.000E-06	5.400E-06	6.000E-06	
	6.250E-06	6.500E-06	6.750E-06	7.000E-06	7.150E-06	
	8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05	
	1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05	
	1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05	
	2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05	
	3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05	
	3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05	
	4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05	
	4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05	
	6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05	
	8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04	
	1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04	
	2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04	
	5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03	
	1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03	
	2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03	
	8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02	
	3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02	
	7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01	
	1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01	

4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	

si3 h 0 381.00

m2 8016.50c -1.0

m8 13027.50c -0.048 \$ Concrete

14000.50c -0.315

8016.50c -0.500

1001.50c -0.006

11023.50c -0.017

20000.50c -0.083

26000.55c -0.012

19000.50c -0.019

mt8 lwtr.01t

imp:n 1 206r 0

c fuel enrichment 4.0 %

m1 92235.50c -0.03526

92238.50c -0.84624

8016.50c -0.11850

c end of file

c



HI-STORM Storage Cask containing MPC68, 08x08 assembly @ 4.2 wt% Enrich.

c MPC68 reflected w/60cm of water, 0.0279 g/cmsq B-10 in Boral

c

c

```

1 1 -10.522      -1 u=2      $ fuel
2 2 -0.0002     1 -2 u=2      $ gap
3 2 -0.0002     2 -3 u=2      $ Zr Clad
4 2 -0.0002     3 u=2      $ water in fuel region
5 2 -0.0002    -4:5      u=3      $ water in guide tubes
6 2 -0.0002     4 -5      u=3      $ guide tubes
7 2 -0.0002    -6 +7      -8 +9      u=1 lat=1
  fill= -5:4      -5:4      0:0
1 1 1 1 1 1 1 1 1 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 3 2 2 2 2 1
1 2 2 2 2 3 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 1
1 1 1 1 1 1 1 1 1 1

```

c

C BOX TYPE R

c

```

8 0 -10 11 -12 13      u=4 fill=1 (0.8128 0.8128 0)
9 3 -6.55 60 -61 62 -63 #8 u=4 $ Zr flow channel
10 2 -0.0002 64 -65 66 -67 #8 #9 u=4 $ water
11 5 -7.84 20 -23 67 -14 u=4 $ 0.075" STEEL
12 2 -0.0002 20 -23 14 -15 u=4 $ WATER POCKET
13 7 -2.7 20 -23 15 -16 u=4 $ Al CLAD
14 6 -2.66 20 -23 16 -17 u=4 $ BORAL Absorber
15 7 -2.7 20 -23 17 -18 u=4 $ Al Clad
16 2 -0.0002 20 -23 18 -118 u=4 $ Water
17 5 -7.84 118:-129:65:-66 u=4 $ Steel
18 2 -0.0002 64 -21 67 -118 u=4 $ Water
19 2 -0.0002 24 -65 67 -118 u=4 $ water
20 5 -7.84 21 -20 67 -118 u=4 $ Steel
21 5 -7.84 23 -24 67 -118 u=4 $ Steel
22 2 -0.0002 129 -64 33 -118 u=4 $ Water
c
23 5 -7.84 25 -64 30 -31 u=4 $ Steel
24 2 -0.0002 26 -25 30 -31 u=4 $ Water
25 7 -2.7 27 -26 30 -31 u=4 $ Al clad
26 6 -2.66 28 -27 30 -31 u=4 $ Boral
27 7 -2.7 29 -28 30 -31 u=4 $ Al clad
28 2 -0.0002 129 -29 30 -31 u=4 $ water
29 5 -7.84 129 -64 32 -30 u=4 $ Steel ends
30 5 -7.84 129 -64 31 -33 u=4 $ Steel ends
31 2 -0.0002 129 -64 66 -32 u=4 $ Water
c

```

c

Type A box - Boral only on left side

c

```

32 0 -10 11 -12 13      u=6 fill=1 (0.8128 0.8128 0)
33 3 -6.55 60 -61 62 -63 #8 u=6 $ Zr flow channel
34 2 -0.0002 64 -65 66 -118 #8 #9 u=6 $ water
35 5 -7.84 118:-129:65:-66 u=6 $ Steel
36 2 -0.0002 129 -64 67 -118 u=6 $ Water

```

```

c
37 5 -7.84 25 -64 30 -31 u=6 $ Steel
38 2 -0.0002 26 -25 30 -31 u=6 $ Water
39 7 -2.7 27 -26 30 -31 u=6 $ Al clad
40 6 -2.66 28 -27 30 -31 u=6 $ Boral
41 7 -2.7 29 -28 30 -31 u=6 $ Al clad
42 2 -0.0002 129 -29 30 -31 u=6 $ water
43 2 -0.0002 129 -64 33 -67 u=6 $ Water
44 5 -7.84 129 -64 32 -30 u=6 $ Steel ends
45 5 -7.84 129 -64 31 -33 u=6 $ Steel ends
46 2 -0.0002 129 -64 66 -32 u=6 $ Water
c
c Type B box - Boral on Top only
c
47 0 -10 11 -12 13 u=7 fill=1 (0.8128 0.8128 0)
48 3 -6.55 60 -61 62 -63 #8 u=7 $ Zr flow channel
49 2 -0.0002 64 -65 66 -67 #8 #9 u=7 $ water
50 5 -7.84 20 -23 67 -14 u=7 $ 0.075" STEEL
51 2 -0.0002 20 -23 14 -15 u=7 $ WATER POCKET
52 7 -2.7 20 -23 15 -16 u=7 $ Al CLAD
53 6 -2.66 20 -23 16 -17 u=7 $ BORAL Absorber
54 7 -2.7 20 -23 17 -18 u=7 $ water
55 2 -0.0002 20 -23 18 -118 u=7 $ Water
56 5 -7.84 118:-129:65:-66 u=7 $ Steel
57 2 -0.0002 64 -21 67 -118 u=7 $ Water
58 2 -0.0002 24 -65 67 -118 u=7 $ water
59 5 -7.84 21 -20 67 -118 u=7 $ Steel
60 5 -7.84 23 -24 67 -118 u=7 $ Steel
61 2 -0.0002 129 -64 66 -118 u=7 $ Water
c
c Type E box - No Boral Panels
c
62 0 -10 11 -12 13 u=8 fill=1 (0.8128 0.8128 0)
63 3 -6.55 60 -61 62 -63 #8 u=8 $ Zr flow channel
64 2 -0.0002 129 -65 66 -118 #8 #9 u=8 $ water
65 5 -7.84 118:-129:65:-66 u=8 $ Steel
c
c Type F box - No Boral Panels or fuel
c
66 2 -0.0002 129 -65 66 -118 u=9 $ water
67 5 -7.84 118:-129:65:-66 u=9 $ Steel
c
68 2 -0.0002 -34 35 -36 37 u=5 lat=1 fill=-7:6 -7:6 0:0
5 5 5 5 5 5 5 5 5 5 5 5
5 9 9 9 9 9 9 9 9 9 9 9 5
5 9 9 9 9 9 7 4 9 9 9 9 5
5 9 9 9 7 4 4 4 4 4 9 9 5
5 9 9 7 4 4 4 4 4 4 4 9 5
5 9 9 7 4 4 4 4 4 4 4 9 5
5 9 7 4 4 4 4 4 4 4 4 9 5
5 9 8 4 4 4 4 4 4 4 4 6 9 5
5 9 9 7 4 4 4 4 4 4 4 9 5
5 9 9 8 4 4 4 4 4 4 4 6 9 5
5 9 9 9 8 4 4 4 6 6 9 9 5
5 9 9 9 9 9 8 6 9 9 9 9 5
5 9 9 9 9 9 9 9 9 9 9 9 5
5 5 5 5 5 5 5 5 5 5 5 5
69 0 -41 50 -49 fill=5 (8.1661 8.1661 0)

```

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c						
274	2	-0.0002	-41	360	-50	\$ space below Fuel (7.3 in.)
275	5	-7.84	-42	362	-360	\$ MPC Steel below Fuel (2.5 in.)
276	5	-7.84	-204	300	-362	\$ Cask Steel (5.0 in.)
277	8	-2.35	-204	301	-300	\$ Cask Concrete (17.0 in.)
278	5	-7.84	-204	302	-301	\$ Cask Steel (2.0 in.)
c						
279	2	-0.0002	-41	49	-361	\$ space above Fuel (8.46 in.)
280	5	-7.84	-42	361	-363	\$ MPC Steel above Fuel (10.0 in)
281	4	-1.00	-42	363	-400	\$ Water (1.0 in.)
282	5	-7.84	-204	400	-401	\$ Cask Steel (1.25 in.)
283	8	-2.35	-204	401	-402	\$ Cask Concrete (10.5 in.)
284	5	-7.84	-204	402	-403	\$ Cask Steel (4.0 in.)
c						
290	5	-7.84	41	-42	360	-361 \$ Radial Steel - MPC shell
291	4	-1.00	42	-200	362	-400 \$ Radial Water
292	5	-7.84	200	-201	362	-400 \$ Radial Steel - overpack inner shell
293	5	-7.84	201	-202	362	-400 \$ Radial Steel -
294	8	-2.35	202	-203	362	-400 \$ Radial Shield - Concrete Overpack
295	5	-7.84	203	-204	362	-400 \$ Radial Steel - overpack outer shell
c						
500	4	-1.00	500	-501	-505	(204 :-302: 403) \$ outer water reflector
501	0		505	:-500:	501	\$ outside world
1	cz		0.5283			\$ Fuel OD
2	cz		0.5398			\$ Clad ID
3	cz		0.6134			\$ Clad OD
4	cz		0.6744			\$ Thimble ID
5	cz		0.7506			\$ Thimble OD
6	px		0.8128			\$ Pin Pitch
7	px		-0.8128			
8	py		0.8128			
9	py		-0.8128			
10	px		6.6231			\$ Channel ID
11	px		-6.6231			
12	py		6.6231			
13	py		-6.6231			
14	py		7.8016			
15	py		7.8155			
16	py		7.8410			
17	py		8.0467			
18	py		8.0721			
118	py		8.0861			
20	px		-6.0325			
21	px		-6.2230			
23	px		6.0325			
24	px		6.2230			
25	px		-7.8016			
26	px		-7.8155			
27	px		-7.8410			
28	px		-8.0467			
29	px		-8.0721			
129	px		-8.0861			
30	py		-6.0325			
31	py		6.0325			
32	py		-6.2230			
33	py		6.2230			
34	px		7.6111			

```

35 px -8.7211
36 py 8.7211
37 py -7.6111
49 pz 381. $ Top of Active Fuel
50 pz 0 $ Start of Active Fuel
60 px -6.9279 $ Channel OD
61 px 6.9279
62 py -6.9279
63 py 6.9279
64 px -7.6111 $ Cell Box ID
65 px 7.6111
66 py -7.6111
67 py 7.6111
360 pz -18.54 $ lower thkness = 7.30 in.
361 pz 402.49 $ upper thkness = 8.46 in.
362 pz -24.892 $ thkness of MPC baseplate = 2.5 in.
363 pz 427.89 $ thkness of MPC lid = 10. in.
41 cz 85.57 $ I.D. = 67.375 in.
42 cz 86.84 $ I.D. = 68.375 in.
200 cz 93.35 $ I.D. = 73.50 in.
201 cz 96.52 $ I.D. = 76.00 in.
202 cz 98.43 $ I.D. = 77.50 in.
203 cz 166.37 $ I.D. = 131.00 in.
204 cz 168.28 $ I.D. = 132.50 in.
300 pz -37.59 $ thkness steel - 5.0 in.
301 pz -80.77 $ thkness concrete - 17.0 in.
302 pz -85.85 $ thkness steel - 2.0 in.
400 pz 430.43 $ thkness water - 1.0 in.
401 pz 433.605 $ thkness steel - 1.25 in.
402 pz 460.28 $ thkness concrete - 10.5 in.
403 pz 465.355 $ thkness steel - 4.0 in.
c
*500 pz -115.85 $ lower boundary
*501 pz 495.355 $ upper boundary
*505 cz 198.28 $ outer radial boundary

```

```

imp:n 1 86r 0
kcode 10000 0.94 20 120
c
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3 h 0 381.
sp3 0 1
c
c
si4 s 15 16
13 14 15 16 17 18
12 13 14 15 16 17 18 19
12 13 14 15 16 17 18 19
11 12 13 14 15 16 17 18 19 20
11 12 13 14 15 16 17 18 19 20
12 13 14 15 16 17 18 19
12 13 14 15 16 17 18 19
13 14 15 16 17 18
15 16
sp4 1 67r

```

```

c
ds5 s          30 30
              29 29 29 29 29 29
              28 28 28 28 28 28 28 28
              27 27 27 27 27 27 27 27
              26 26 26 26 26 26 26 26 26
              25 25 25 25 25 25 25 25 25
              24 24 24 24 24 24 24 24
              23 23 23 23 23 23 23 23
              22 22 22 22 22 22
              21 21

```

```

c
si11 -80.6831 -67.6783
si12 -64.1985 -51.1937
si13 -47.7139 -34.7091
si14 -31.2293 -18.2245
si15 -14.7447 -1.7399
si16  1.7399  14.7447
si17  18.2245 31.2293
si18  34.7091 47.7139
si19  51.1937 64.1985
si20  67.6783 80.6831

```

```

c
si21 -80.6831 -67.6783
si22 -64.1985 -51.1937
si23 -47.7139 -34.7091
si24 -31.2293 -18.2245
si25 -14.7447 -1.7399
si26  1.7399  14.7447
si27  18.2245 31.2293
si28  34.7091 47.7139
si29  51.1937 64.1985
si30  67.6783 80.6831

```

```

sp11 0 1
sp12 0 1
sp13 0 1
sp14 0 1
sp15 0 1
sp16 0 1
sp17 0 1
sp18 0 1
sp19 0 1
sp20 0 1
sp21 0 1
sp22 0 1
sp23 0 1
sp24 0 1
sp25 0 1
sp26 0 1
sp27 0 1
sp28 0 1
sp29 0 1
sp30 0 1

```

```

c
m1          92235.50c   -0.03702   $ 4.20% E Fuel
            92238.50c   -0.84448
            8016.50c    -0.1185
m2          8016.50c    1.           $ Void

```

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m3	40000.56c	1.		\$ Zr Clad		
m4	1001.50c	0.6667		\$ Water		
	8016.50c	0.3333				
m5	24000.50c	0.01761		\$ Steel		
	25055.50c	0.001761				
	26000.55c	0.05977				
	28000.50c	0.008239				
m6	5010.50c	8.0707E-03		\$ Boral		
	5011.50c	3.2553E-02				
	6000.50c	1.0146E-02				
	13027.50c	3.8054E-02				
m7	13027.50c	1.		\$ Al Clad		
m8	13027.50c	-0.0048		\$ Concrete		
	14000.50c	-0.315				
	8016.50c	-0.500				
	1001.50c	-0.006				
	11023.50c	-0.017				
	20000.50c	-0.083				
	26000.55c	-0.012				
	19000.50c	-0.019				
mt4	lwtr.01t					
mt8	lwtr.01t					
prdmp	j	-60	j	2		
fm4	1000	1	-6			
f4:n	1					
sd4	1000					
e4	1.000E-11	1.000E-10	5.000E-10	7.500E-10	1.000E-09	1.200E-09
	1.500E-09	2.000E-09	2.500E-09	3.000E-09		
	4.700E-09	5.000E-09	7.500E-09	1.000E-08	2.530E-08	
	3.000E-08	4.000E-08	5.000E-08	6.000E-08	7.000E-08	
	8.000E-08	9.000E-08	1.000E-07	1.250E-07	1.500E-07	
	1.750E-07	2.000E-07	2.250E-07	2.500E-07	2.750E-07	
	3.000E-07	3.250E-07	3.500E-07	3.750E-07	4.000E-07	
	4.500E-07	5.000E-07	5.500E-07	6.000E-07	6.250E-07	
	6.500E-07	7.000E-07	7.500E-07	8.000E-07	8.500E-07	
	9.000E-07	9.250E-07	9.500E-07	9.750E-07	1.000E-06	
	1.010E-06	1.020E-06	1.030E-06	1.040E-06	1.050E-06	
	1.060E-06	1.070E-06	1.080E-06	1.090E-06	1.100E-06	
	1.110E-06	1.120E-06	1.130E-06	1.140E-06	1.150E-06	
	1.175E-06	1.200E-06	1.225E-06	1.250E-06	1.300E-06	
	1.350E-06	1.400E-06	1.450E-06	1.500E-06	1.590E-06	
	1.680E-06	1.770E-06	1.860E-06	1.940E-06	2.000E-06	
	2.120E-06	2.210E-06	2.300E-06	2.380E-06	2.470E-06	
	2.570E-06	2.670E-06	2.770E-06	2.870E-06	2.970E-06	
	3.000E-06	3.050E-06	3.150E-06	3.500E-06	3.730E-06	
	4.000E-06	4.750E-06	5.000E-06	5.400E-06	6.000E-06	
	6.250E-06	6.500E-06	6.750E-06	7.000E-06	7.150E-06	
	8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05	
	1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05	
	1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05	
	2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05	
	3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05	
	3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05	
	4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05	
	4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05	
	6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05	
	8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04	
	1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04	

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2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04
5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03
1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03
2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03
8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02
3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02
7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	

## SUPPLEMENT 6.I

### CRITICALITY EVALUATION OF THE HI-STORM 100U SYSTEM

#### 6.I.0 INTRODUCTION

This supplement is solely focused on providing an evaluation of criticality safety of the HI-STORM 100U system. The evaluation presented herein supplements those evaluations of the HI-STORM 100 overpack contained in the main body of Chapter 6 of this FSAR.

#### 6.I.1 ACCEPTANCE CRITERIA

The acceptance criteria for criticality evaluations for the HI-STORM 100U system are identical to the criteria for the HI-STORM 100 system analyzed in the main body of Chapter 6.

#### 6.I.2 EVALUATION

The HI-STORM 100U system differs from the HI-STORM 100 system only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs and HI-TRAC transfer casks are identical between the two systems.

The limiting condition in the main body of Chapter 6 of this FSAR is the fully flooded MPC in a HI-TRAC. This limiting condition is unaffected by the different storage overpacks. All results and conclusions for the fully flooded condition from the main body of Chapter 6 are therefore directly applicable to the HI-STORM 100U system.

During storage conditions, the maximum  $k_{\text{eff}}$  is significantly below the limiting maximum  $k_{\text{eff}}$  since the MPC is internally dry (see Section 6.1). Under this condition, the configuration is very similar between the HI-STORM 100U VVM and the HI-STORM 100 overpack analyzed in the main body of Chapter 6, consisting of the internally dry MPC, the gap between the MPC and the overpack, a steel shell or shells and concrete (above-ground) or soil (below-ground). Results for the VVM would therefore be practically identical to the results listed for storage conditions in the main body of Chapter 6. Any small differences in results would not affect the principal conclusions, since the maximum  $k_{\text{eff}}$  under storage conditions is so low.

In summary, the limiting condition for the HI-STORM 100U is identical to the limiting condition for the HI-STORM 100, and other conditions are very similar between the two systems from a criticality perspective. All results and conclusions for the HI-STORM 100 system presented in the main body of Chapter 6 are therefore directly applicable to the HI-STORM 100U system, and no additional calculations to demonstrate criticality safety are required for the HI-STORM 100U system.



## SUPPLEMENT 6.III<sup>1</sup>: CRITICALITY EVALUATION OF THE MPC-68M

### 6.III.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.III.1.1], 10CFR72.124 [6.III.1.2], and NUREG-0800 Section 9.1.2 [6.III.1.3], the results in this supplement demonstrate that the effective multiplication factor ( $k_{\text{eff}}$ ) of the HI-STORM 100 System with the MPC-68M, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Criticality safety of the HI-STORM 100 System with the MPC-68M depends on the following principal design parameters:

- The inherent geometry of the fuel basket design of the MPC-68M;
- The incorporation of spatially distributed B-10 isotope in the Metamic-HT fuel basket structure. Based on the tests for the neutron absorber content in Metamic-HT (see Supplement 1.III and Supplement 9.III), and consistent with the approach taken for Metamic (see Section 9.1.5.3.2), 90% of the minimum B-10 ( $B_4C$ ) content is credited in the analysis. With a specified minimum  $B_4C$  content of 10 wt%, the concentration credited in the analysis is therefore 9 wt%.

The off-normal and accident conditions defined in Section 2.2 are applicable to the HI-STORM System using the MPC-68M. These accidents are considered in Supplement 11.III and have no adverse effect on the design parameters important to criticality safety, except for the non-mechanistic tip-over event, which could result in limited plastic deformation of the basket. However, a bounding basket deformation is already included in the criticality models for normal conditions, and thus, from the criticality safety standpoint, the off-normal and accident conditions are identical to those for normal conditions.

Results of the design basis criticality safety calculations for a single internally flooded HI-TRAC transfer cask with full water reflection on all sides (limiting cases for the HI-STORM 100 System), loaded with undamaged fuel assemblies (see definition in Supplement 1.III) are listed in Table 6.III.1.1 and 6.III.1.4, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.III.3), and including the calculational bias, uncertainties, and calculational statistics. Table 6.III.1.1 provides the information for undamaged fuel without known or suspected cladding defects larger than pinhole leaks or hairline cracks, while Table 6.III.1.4 provides information for low-enriched, channeled BWR undamaged fuel without known or suspected grossly breached fuel rods. In addition, a result for a single internally dry (no moderator) HI-STORM storage cask with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, is listed in Table 6.III.1.2 to confirm the low reactivity of the HI-STORM 100 System with an MPC-68M in storage. The maximum  $k_{\text{eff}}$  for an MPC-68M loaded with up to 16 DFCs is listed in Table 6.III.1.3.

<sup>1</sup> Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.III.1.4].

TABLE 6.III.1.1

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68M  
(HI-TRAC 100)

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% U-235)</b>	<b>Maximum <math>k_{eff}</math></b>
7x7B	4.8	0.9243
8x8B	4.8	0.9294
8x8C	4.8	0.9302
8x8D	4.8	0.9307
8x8E	4.8	0.9211
8x8F	4.5	0.9245
9x9A	4.8	0.9341
9x9B	4.8	0.9330
9x9C	4.8	0.9254
9x9D	4.8	0.9254
9x9E/F	4.5	0.9254
9x9G	4.8	0.9211
10x10A	4.8	0.9360
10x10B	4.8	0.9353
10x10C	4.8	0.9321
10x10F	4.7	0.9356
10x10G	4.75	0.9472

Note: The results presented in the table above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

TABLE 6.III.1.2

REPRESENTATIVE  $k_{eff}$  VALUES FOR MPC-68M IN THE HI-STORM 100 OVERPACK

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <math>^{235}\text{U}</math>)</b>	<b>Maximum <math>k_{eff}</math></b>
10x10A	4.8	0.3754

TABLE 6.III.1.3

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68M WITH UP TO 16 DFCs

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <math>^{235}\text{U}</math>)</b>	<b>Maximum <math>k_{eff}</math></b>
All BWR Classes except 8x8F, 9x9E/F, 10x10F and 10x10G	4.8	0.9408
8x8F and 9x9E/F	4.0	0.9028
10x10F and 10x10G	4.6	0.9453

TABLE 6.III.1.4

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68M WITH LOW ENRICHED, CHANNELED BWR FUEL

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <math>^{235}\text{U}</math>)</b>	<b>Maximum <math>k_{eff}</math></b>
All BWR Classes	3.3	0.9269

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Note: The results presented in Tables 6.III.1.2, 6.III.1.3, and 6.III.1.4 above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

### 6.III.2 SPENT FUEL LOADING

The BWR fuel assembly classes/arrays which are authorized for the MPC-68 are qualified for the MPC-68M, except for the 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D and 10x10E. Additionally, the MPC-68M is qualified for two new assembly classes, 10x10F and 10x10G. Information on those classes is provided in Supplement 2.III, Table 2.III.3. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for all other fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt%  $^{235}\text{U}$  are authorized in the MPC-68M.

BWR assemblies are specified in the Table 2.1.4 and Table 2.III.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in Appendix 6.B for the MPC-68 show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum  $k_{\text{eff}}$ . To confirm this for the higher enrichments analyzed here, additional calculations were performed for the assembly class 10x10A in the MPC-68M, and are presented in Table 6.III.2.1 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt%  $^{235}\text{U}$  is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt%  $^{235}\text{U}$  is close to the maximum rod enrichment of 5.0 wt%  $^{235}\text{U}$ , the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt%  $^{235}\text{U}$ , while the remainder of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results in Table 6.III.2.1 indicate that there may be a potential positive reactivity effect (+0.0021) due to distributed enrichments. Therefore, additional studies with distributed enrichments were performed and are presented in Table 6.III.2.2. These include all cases from Appendix B (for 8x8 and 9x9 assembly types), now evaluated in the MPC-68M, and additional cases for the 10x10G which has the highest reactivity of all assembly classes. The cases from Appendix B show no statistically significant increase, and in most cases a decrease in reactivity as a result of the distributed enrichment. Nevertheless, for conservatism an additional bias of 0.0021 is applied to the results for all 10x10 fuel assembly classes in Section 6.III.1, including the cases with damaged fuel and low enriched channeled fuel. Note that for the studies presented in the remainder of this supplement this bias is not included since those studies focus on reactivity differences rather than absolute values of  $k_{\text{eff}}$ .

TABLE 6.III.2.1

## REACTIVITY EFFECT OF DISTRIBUTED ENRICHMENTS IN BWR FUEL IN THE MPC-68M

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum $k_{eff}$	standard deviation
10x10A (Reference, full-length rods only)	Reference	0.9339	0.0004
distributed enrichment, Case 1	-0.0004	0.9335	0.0004
distributed enrichment, Case 2	+0.0021	0.9360	0.0004
distributed enrichment, Case 3	-0.0092	0.9247	0.0004
distributed enrichment, Case 4	-0.0118	0.9221	0.0004

TABLE 6.III.2.2

ADDITIONAL CALCULATIONS OF THE REACTIVITY EFFECT OF DISTRIBUTED  
ENRICHMENTS IN BWR FUEL IN THE MPC-68M

Assembly Class	Enrichment	Maximum $k_{eff}$	Description	Delta- $k_{eff}$
8x8C	4.8	0.8273	Average Enrichment	-0.0044
8x8C	4.8	0.8229	Distributed Enrichment	
8x8C	4.8	0.8876	Average Enrichment	-0.0040
8x8C	4.8	0.8836	Distributed Enrichment	
8x8D	4.8	0.8550	Average Enrichment	+0.0004
8x8D	4.8	0.8554	Distributed Enrichment	
8x8D	4.8	0.8774	Average Enrichment	-0.0017
8x8D	4.8	0.8757	Distributed Enrichment	
8x8D	4.8	0.8855	Average Enrichment	-0.0026
8x8D	4.8	0.8829	Distributed Enrichment	
9x9B	4.8	0.9103	Average Enrichment	-0.0023
9x9B	4.8	0.9080	Distributed Enrichment	
9x9D	4.8	0.8467	Average Enrichment	-0.0095
9x9D	4.8	0.8372	Distributed Enrichment	
8x8C	4.8	0.9023	Average Enrichment	-0.0025
8x8C	4.8	0.8998	Distributed Enrichment	
8x8C	4.8	0.9165	Average Enrichment	-0.0003
8x8C	4.8	0.9162	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0253
10x10G	4.75	0.9198	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0268
10x10G	4.75	0.9183	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0009
10x10G	4.75	0.9442	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0010
10x10G	4.75	0.9441	Distributed Enrichment	

### 6.III.3 MODEL SPECIFICATION

Calculational models for the MPC-68M are generally the same as those described in Section 6.3 except for the different basket as noted below.

Figures 6.III.3.1 and 6.III.3.2 show representative cross sections of the criticality models for the MPC-68M basket. Figure 6.III.3.1 shows a single cell of the basket, while Figure 6.III.3.2 shows the entire MPC-68M basket. All calculations are performed with eccentric fuel positioning, where the fuel is placed closest to the center of the basket in each basket cell. The wall thickness of the basket shims is modeled as 1 inch, while some of them are only ½ inch thick. This is conservative, since the model replaces water with aluminum, which reduces absorption and moderation outside of the basket.

To account for the potentially higher fuel density of higher enriched fuel, a conservative fuel stack density of 97.5% of the theoretical density (i.e.  $10.96 \text{ g/cm}^3 * 97.5\% = 10.686 \text{ g/cm}^3$ ) is used in all analyses for the MPC-68M.

The basket geometry can vary due to manufacturing tolerances and due to potential deflections of basket walls as the result of accident conditions. The basket tolerances are defined on the drawings in Section 1.5. The structural acceptance criterion for the basket during accident conditions is that the permanent deflection of the basket panels is limited to a fraction of 0.005 (0.5%) of the panel width (see Chapter 3). The analyses in Supplement 3.III demonstrate that permanent deformations of the basket walls during accident conditions are far below this limit. Nevertheless, it is conservatively assumed that 2 adjacent cell walls in each cell are deflected to the maximum extent possible over their entire length and width, i.e. that the cell ID is reduced by 0.5% of the cell width, or 0.03" for MPC-68M cells. Maximum  $k_{\text{eff}}$  results (including the bias, uncertainties, or calculational statistics), along with the selected dimensions, for a number of dimensional combinations are shown in Table 6.III.3.1 for various fuel types. The cell ID is evaluated for minimum (tolerance only), minimum with deformation, nominal and an increased value. The wall thickness is evaluated for nominal and minimum values.

Based on the calculations, the conservative dimensional assumptions listed in Table 6.III.3.2 were determined for the basket designs. Note that, as expected, the bounding basket condition correspond to the minimum wall thickness and minimum cell-id. Because the reactivity effect (positive or negative) of the manufacturing tolerances is not assembly dependent (see Table 6.III.3.1), these dimensional assumptions were employed for all criticality analyses.

The basket is manufactured from individual slotted panels. The panels are expected to be in direct contact with each other (see Drawing in Section 1.5). However, to show that small gaps between panels would have essentially no effect on criticality, calculations are performed with a postulated 0.06" gap between panels, repeated in the axial direction every 10" in all panels. The results are summarized in Tables 6.III.3.3 and show statistically equivalent results for calculations with and without the gap. This indicates that the effect of small gaps between the METAMIC panels is negligible. Therefore, all other calculations are performed without any gaps between panels.

The MPC-68M uses the same principal neutron poison material as the MPC-68 (i.e.  $^{10}\text{B}$  in the form of  $\text{B}_4\text{C}$ ). The evaluation provided in Section 6.3.2, which concludes that  $^{10}\text{B}$  depletion is negligible,

is therefore directly applicable to the MPC-68M, and no additional evaluations are required in that respect.

Composition of the Metamic-HT is listed in Table 6.III.3.4.



TABLE 6.III.3.1

## EVALUATION OF BASKET MANUFACTURING TOLERANCES FOR MPC-68M

<b>Box I.D.</b>	<b>Box Wall Thickness</b>	<b>Maximum keff</b>
10x10A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9307
increased (6.07")	minimum (0.38")	0.9288
minimum (5.99")	minimum (0.38")	0.9334
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9339</b>
7x7B, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9154
nominal (6.05")	minimum (0.38")	0.9196
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9243</b>
8x8D, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9230
nominal (6.05")	minimum (0.38")	0.9265
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9307</b>
9x9A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9301
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9341</b>
10x10G, 4.75% Enrichment		
nominal (6.05")	nominal (0.40")	0.9365
nominal (6.05")	minimum (0.38")	0.9409
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9451</b>
10x10A, 4.8% Enrichment, Damaged Fuel		
nominal (6.05")	nominal (0.40")	0.9316
nominal (6.05")	minimum (0.38")	0.9348
minimum, including deformation (5.96")	minimum (0.38")	<b>0.9387</b>

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

TABLE 6.III.3.2

MPC-68M BASKET DIMENSIONAL ASSUMPTIONS

<b>Box I.D.</b>	<b>Box Wall Thickness</b>
minimum, including deformation (5.96")	minimum (0.38")

TABLE 6.III.3.3

## REACTIVITY EFFECTS GAPS IN BASKET CELL PLATES

Gaps in Metamic-HT	MPC-68M (10x10A, 4.8% ENRICHMENT)	
	Max. $k_{eff}$	1 Sigma
None	0.9339	0.0004
0.06" every 10"	0.9346	0.0003

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

TABLE 6.III.3.4

## COMPOSITION OF THE METAMIC-HT

<b>METAMIC HT, 9% B<sub>4</sub>C, DENSITY 2.6 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
13027	0.91
6000	0.01956
5010	0.01289
5011	0.05755

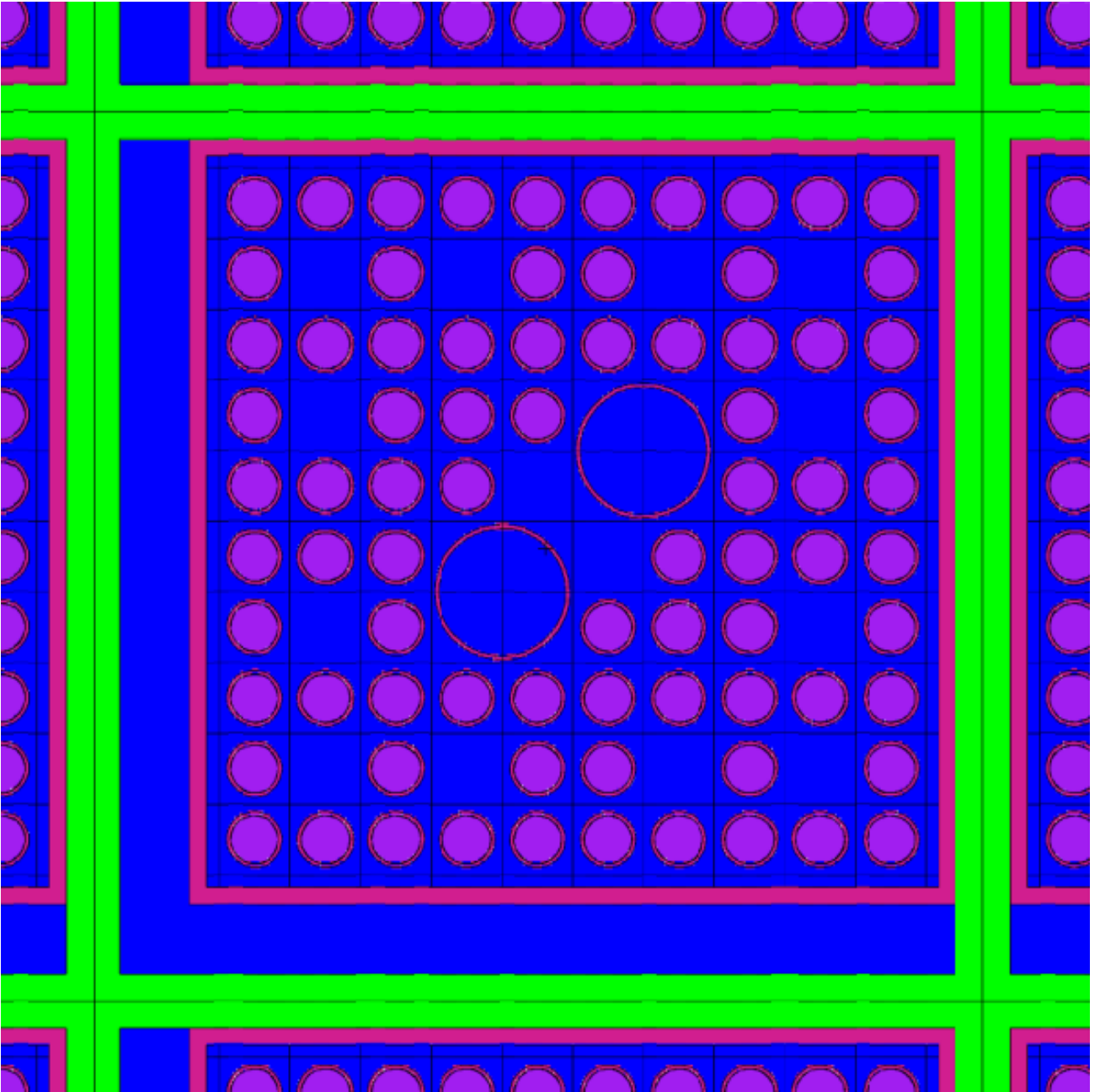


Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.III.3.2. For true dimensions see the drawing in Section 1.5. Note that the figure depicts an assembly with full and part length rods, showing the cross section where only full length rods are present.

Figure 6.III.3.1: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-68M

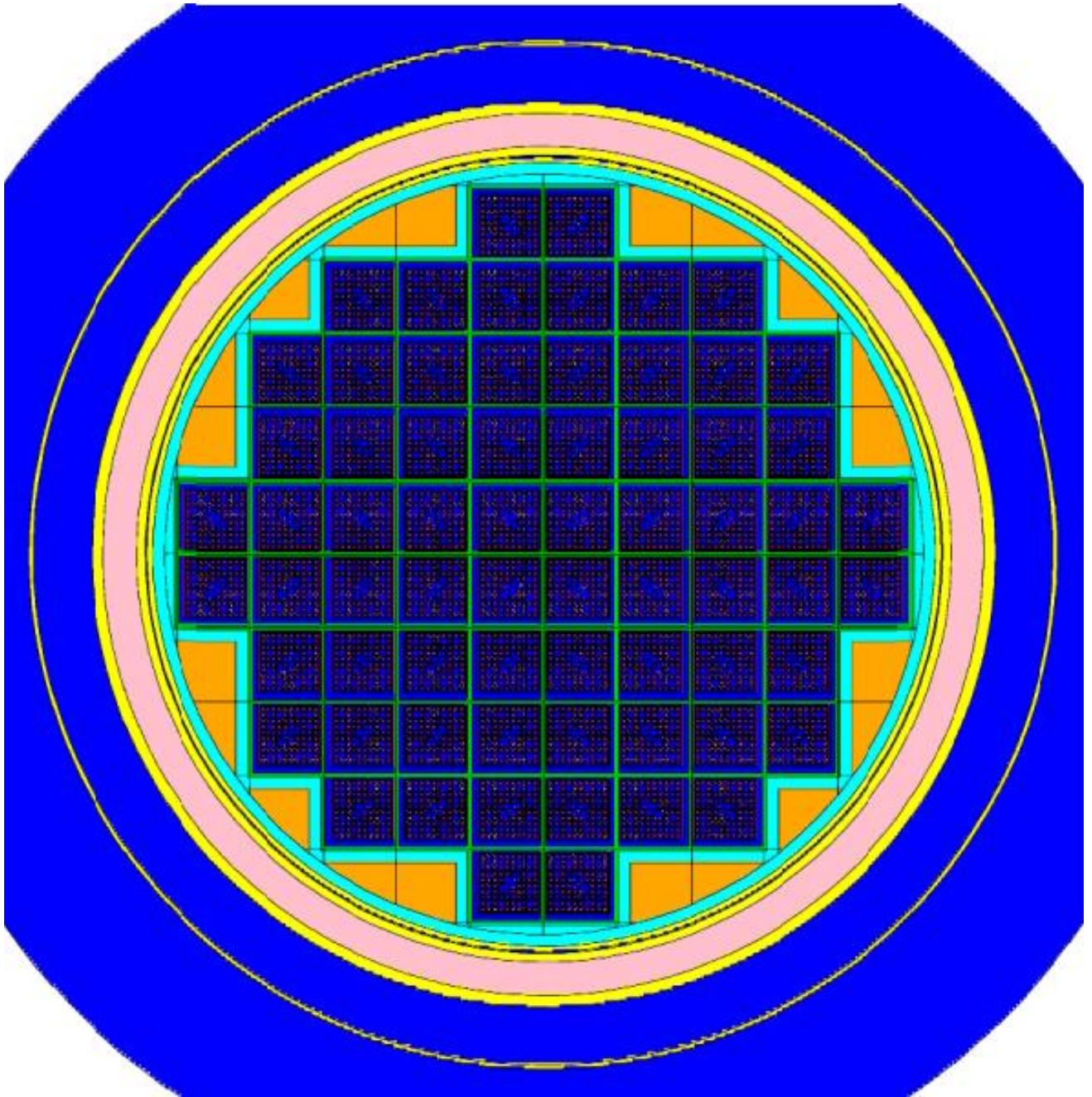


Figure generated directly from MCNP input file using the MCNP plot function. Radial dimensions of the HI-TRAC used in the analyses are unchanged from the analyses in the main part of this chapter. For true dimensions see the drawings in Section 1.5.

Figure 6.III.3.2: Calculational Model (planar cross-section) of the MPC-68M

### 6.III.4 CRITICALITY CALCULATIONS

The calculations in this supplement use the same computer codes and methodologies that are used in the main part of Chapter 6. Specifically, the conservative approach to model damaged fuel and fuel debris, using arrays of bare fuel rods, is the same (see discussion in Subsection 6.III.4.1 below).

The basket design of the MPC-68M is essentially identical to that of the MPC-68, in respect to the characteristics important to criticality safety. Specifically,

- The number and configuration of the cells for undamaged and damaged fuel/fuel debris are unchanged;
- The basket dimensions are essentially the same; and
- The same poison material (B<sub>4</sub>C) is used, but a larger <sup>10</sup>B content in the basket walls.

The content is also the same, except for the following

- Higher enrichments are qualified, consistent with the higher <sup>10</sup>B content in the basket walls; and
- Two additional fuel assembly types are analyzed, that are variations of existing types with slightly different dimension.

To verify that the bounding fuel parameter variations analyzed in the MPC-68 are also applicable to the MPC-68M, additional studies are performed and discussed in Subsection 6.III.4.2 below.

Due to the strong similarity in the basket design, the conclusions of the various studies presented in the main part of this Chapter on the MPC-68 are directly applicable to the MPC-68M. Nevertheless, to confirm this is also applicable to the MPC-68M, numerous studies with various moderation conditions that conclude that the fully flooded basket is the bounding case are re-analyzed and discussed in subsection 6.III.4.3. All analyzes are therefore performed under the following condition:

- Basket, and DFCs as applicable, are fully flooded with pure water at the maximum density; and
- Pellet-to-clad gaps of undamaged assemblies are assumed flooded (see also discussion in Subsection 6.III.4.2 below)
- All assemblies and DFCs are located eccentrically in the basket, closest to the center of the basket.

Results for all design basis calculations are listed in Subsection 6.III.1. All maximum  $k_{\text{eff}}$  values are below the regulatory limit of 0.95.

#### 6.III.4.1 Damaged Fuel and Fuel Debris

For damaged fuel and fuel debris in the MPC-68M the same conservative approach is used as in the main part of this chapter, see discussion in Section 6.4.4, specifically 6.4.4.2. Important aspects of this approach that ensure its conservatism are as follows:

- All damaged fuel and fuel debris must be in damaged fuel containers (DFCs), and located in specifically designated cells on the periphery of the basket as specified in Table 2.1.22.
- A conservative model is used that bounds both damaged fuel and fuel debris. In other words, damaged fuel is always conservatively modeled as fuel debris.
- The model consists of regular arrays of fuel rods without cladding. The rods pitch (array size) is varied to determine the optimum moderation condition.
- Undamaged and damaged fuel/fuel debris in the same basket have the same enrichment limit, which may be different from the enrichment limit for undamaged fuel only.
- The results for loading with undamaged fuel only in Table 6.III.1.1 utilize different enrichment limits for different assembly classes, to ensure that the maximum  $k_{\text{eff}}$  is always below 0.95. It is therefore not possible to establish a single bounding assembly class/enrichment combination to be used in all analyzes with damaged fuel/fuel debris. Therefore, and in order to optimize the enrichment for the loading of undamaged and damaged fuel/fuel debris for each assembly class, undamaged assemblies are grouped by enrichment limit, and the undamaged assembly with the highest maximum  $k_{\text{eff}}$  in each group is used for the calculations together with damaged fuel/fuel debris. These are:
  - Undamaged assemblies of 4.5 wt%: Assembly class 9x9E/F. For the calculations with undamaged and damaged fuel, an enrichment of 4.0 wt% is used.
  - Undamaged assembly of 4.7 and 4.75wt%: Assembly class 10x10G. For the calculations with undamaged and damaged fuel, an enrichment of 4.6 wt% is used.
  - Undamaged assembly of 4.8 wt%: Assembly class 10x10A. For the calculations with undamaged and damaged fuel, an enrichment of 4.8 wt% is used.
- Consistent with the results in the main part of this chapter for the MPC-68, array sizes of 9x9, 10x10 and 11x11 show the optimum moderation condition. This is confirmed for undamaged assembly classes 9x9E/F, 10x10A and 10x10G by evaluating all arrays from 3x3 to 17x17 rods.

#### 6.III.4.2 Fuel Parameters and Parameter Variations

In the main part of the FSAR, extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- maximum fuel rod pitch,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only)
- part length rods (if present) removed.

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely 1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:



- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.
- Fuel assemblies are predominantly characterized by the amount of fuel and the fuel-to-water (moderator) ratio. Increasing the amount of fuel, or the enrichment of the fuel, will increase the amount of fissile material, and therefore increase reactivity. Regarding the fuel-to-water ratio, it is important to note that commercial BWR assemblies are undermoderated, i.e. they do not contain enough water for a maximum possible reactivity.
- The neutron poison in the basket walls uses B-10, which is an absorber of thermal neutrons. This poison therefore also needs water (moderator) to be effective. This places a specific importance on the amount of water between the outer rows of the fuel assemblies and the basket cell walls. Note that this explains some of the differences in reactivity between the different assembly types in the same basket, even for the same enrichment, where assemblies with a smaller cross section, i.e. which have more water between the periphery of the assembly and the surrounding wall, generally have a lower reactivity.

Based on these characteristics, the following conclusions can be made:

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to-clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536, a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not by a positive increase in reactivity from the reduction in fuel (which would be counter-intuitive). Therefore, the maximum fuel pellet diameter is used for the fuel specification.

- Several assembly types contain part length rods (9x9A, 10x10A, B, F and G):
  - For 9x9A and 10x10A and B it was shown in the main part of the chapter (Tables 6.2.29, 36 and 37, respectively) that the condition with the part lengths rods completely removed is bounding (10x10F is covered in 10x10A). This condition is therefore used in the design basis calculations, so a specification of the lengths of the part length rods is not required (this includes assemblies with multiple length partial rods). Applicability to the MPC-68M is confirmed by showing that all assemblies are undermoderated, which means the increase of water from completely removing the part length rods increases reactivity. All calculations for the MPC-68M for assembly classes 9x9A and 10x10A and B are therefore performed with models where the part length rods are replaced by water.
  - For assembly class 10x10G, the part length rods are located near the periphery and the water rod of the assembly, i.e. not surrounded by other rods on all sides. For this fuel assembly class, calculations with various part length rod lengths were performed and show that using full length rods in place of part length rods is more conservative. These calculations are listed in Table 6.III.4.4. The case with all rods full length is therefore conservatively used in Table 6.III.1.1. This again removes the need to specify a length for the part lengths rods, and also covers assemblies with multiple length partial rods.

Since all assemblies have the same principal design, i.e. consist of bundles of clad fuel rods, most of them with embedded guide/instrument tubes or water rods or channels, the above conclusions apply to all of them, and the bounding dimensions are therefore also common to all fuel assemblies analyzed here. Nevertheless, to clearly demonstrate that the main assumption is true, i.e. that all assemblies are undermoderated, a study was performed for all assembly types where the pellet-to-clad gap is empty instead of being flooded (a conservative assumption for the design basis calculations) The results are listed in Table 6.III.4.2, in comparison with the results of the reference cases with the flooded gap. In all cases, the reactivity is reduced compared to the reference case. This verifies that all assembly types considered here are in fact undermoderated, and therefore validates the main assumption stated above. All assembly types are therefore behaving in a similar fashion, and the bounding dimensions are therefore applicable to all assembly types.

The discussion provided above regarding the principal characteristics of fuel poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR assembly type, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion above are listed below.

**Basket Manufacturing Tolerance:** The two aspects of the basket tolerance that are evaluated are the cell wall thickness and the cell ID. The reduced cell wall thickness results in a reduced amount of poison (since the material composition of the wall is fixed), and therefore in an increase in reactivity. The reduced cell ID reduces amount of water between the fuel the poison, and therefore the effectiveness of the poison material. Both effects are simply a function of the geometry, and are independent of the fuel type. However, calculations are performed for various fuel types and this study is presented in Table 6.III.3.1.

Panel Gaps: Similar to the basket manufacturing tolerance for the cell wall thickness, this tolerance has a small effect on the overall poison amount of the basket, which would affect the reactivity of the system independent of the fuel type. This study is presented in Table 6.III.4.5.

Eccentric positioning: When a fuel assembly is located in the center of a basket cell, it is surrounded by equal amounts of water on all sides, and hence the thermalization of the neutrons that occur between the assembly and the poison in the cell wall, and hence the effectiveness of the poison, is also equal on all sides. For an eccentric positioning, the effectiveness of the poison is now reduced on those sides where the assembly is located close to the cell walls, and increased on the opposite sides. This creates a compensatory situation for a single cell, where the net effect is not immediately clear. However, for the entire basket, and for the condition where all assemblies are located closest to the center of the basket, the four assemblies at the center of the basket are now located close to each other, separated by poison plates with a reduced effectiveness since they are not surrounded by water on any side. This now becomes the dominating condition in terms of reactivity increase. This effect is also applicable to all assembly types, since those assemblies are all located close to the center of the basket, i.e. the eccentric position with all assemblies moved towards the center will be bounding regardless of the assembly type. An additional study is presented in Table 6.III.4.6 to confirm this effect for the MPC-68M.

In addition, additional fuel assembly characteristics important to criticality control are the location of guide tubes, water rods, part length rods, and rods with differing dimensions (classes 9x9E/F only). These are identified in the assembly cross sections provided in Figure 6.III.4.1.

Various additional studies were performed to address conditions for specific assembly classes:

- Fuel assembly class 8x8B and 8x8D are specified with various fuel rod numbers and water rod locations (see Figure 6.III.4.1). Calculations are performed to show that the results listed in Table 6.III.1 are bounding. These calculations are listed in Table 6.III.4.3.
- Fuel assembly class 9x9E/F has two rod diameters, and two water rod patterns (see Figure 6.III.4.1). A study is performed and documented in Table 6.III.4.3 that shows that the result listed in Table 6.III.1 is bounding.

#### 6.III.4.3 Moderation Conditions

Additional studies were performed to verify that the fully flooded condition is bounding for the MPC-68M, as it is for the MPC-68 basket analyzed in the main part of this chapter:

- Internal and External Moderation: The studies presented in Table 6.III.4.2 show that all assemblies essentially behave identical in respect to water moderation, specifically, that all assemblies are undermoderated. The principal effect of changes to the internal and external moderation would therefore be independent of the fuel type. Calculations for the MPC-68M design with external moderators of various densities are shown in Table 6.III.4.7, all performed for the HI-TRAC and the MPC fully flooded. The results show that the maximum  $k_{\text{eff}}$  is essentially independent from the external water density. Nevertheless, all further evaluations are performed with full external water density. In a definitive study, Cano, et al. [6.4.2] have

demonstrated that the phenomenon of a peak in reactivity at low moderator densities (sometimes called "optimum" moderation) does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. All calculations are therefore performed with full water density inside the MPCs.

- **Partial Flooding:** The partial flooding of the basket, either in horizontal or vertical direction, reduces the amount of fuel that partakes effectively in the thermal fission process, while essentially maintaining the fuel-to-water ratio in the volume that is still flooded. This will therefore result in a reduction of the reactivity of the system (similar to that of the reduction of the active length), and due to the similarity of the fuel assemblies is not dependent on the specific fuel type. The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density ( $1.0 \text{ g/cm}^3$ ) water and the remainder of the cask is filled with steam consisting of ordinary water at a low partial density ( $0.002 \text{ g/cm}^3$  or less), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.III.4.8. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. Note that the studies for partial flooding are performed with the design basis model for the assembly class 10x10A that has the partial length rods removed for added conservatism, while the calculations in the main part of the chapter for the MPC-68 were performed for an assembly class that did not include partial length rods. This shows that the conclusion from partial flooding, i.e. that the fully flooded condition is bounding, applies equally to assemblies with and without partial lengths rods.
- **Pellet-to-clad Gap Flooding:** As demonstrated by the studies shown in Table 6.III.4.2, all assemblies are undermoderated. Flooding the pellet-to-clad gap will therefore improve the moderation and therefore increase reactivity for all assembly types.
- **Preferential Flooding:** The only preferential flooding situation that may be credible is the flooding of the bottom section of the DFCs while the rest of the MPC internal cavity is already drained. In this condition, the undamaged assemblies have a negligible effect on the system reactivity since they are not flooded with water. The dominating effect is from the damaged fuel model in the DFCs. However, the damaged fuel model is conservatively based on an optimum moderated array of bare fuel rods in water, and therefore representative of all fuel types and therefore the fully flooded condition is bounding of the preferential flooding condition.

#### 6.III.4.4 Low Enriched, Channeled BWR fuel

The calculations in this subsection show that low enriched, channeled BWR fuel with indeterminable cladding condition is acceptable for loading in all storage locations of the MPC-68M without placing those fuel assemblies into DFCs, hence classifying those assemblies as undamaged. The main characteristics that must be assured are:

- The channel is present and attached to the fuel assembly in the standard fashion; and
- The channel is essentially undamaged; and
- The maximum planar average enrichment of the assembly is less than or equal to  $3.3 \text{ wt}\%$   $^{235}\text{U}$

This analysis covers older assemblies, where the cladding integrity is uncertain, and where a verification of the cladding condition is prohibitive. An example of this type of fuel is the so-called CILC (Copper Induced Localized Corrosion) fuel, which has potential corrosion-induced damaged to the cladding but does not have grossly breached spent fuel rods.

The presence of the essentially undamaged and attached channel confines the fuel rods to a limited volume and the low enrichment, limits the reactivity of the fuel even under optimum moderation conditions. Due to the uncertain cladding condition, the analysis of this fuel follows essentially the same approach as that for the Damaged Fuel and Fuel Debris, i.e. bare fuel rod arrays of varying sizes are analyzed within the confines of the channel. This is an extremely conservative modeling approach for this condition, since reconfiguration is not expected and cladding would still be present.

Calculations are performed with these assemblies in all cells of the MPC-68M, without DFCs. The results of this conservative analysis are listed in Table 6.III.4.9 and show that the system remains below the regulatory limit.

In addition, calculations are performed for the MPC-68M with checkerboard configuration of normal undamaged fuel and low enriched, channeled BWR fuel fuel without DFCs. The results of this analysis are listed in Table 6.III.4.10 and show that the reactivity remains below the regulatory limit and bounded by the reference undamaged fuel assembly in all cells.

These results confirm that even with unknown cladding condition the maximum keff values are below the regulatory limit when fully flooded and loaded with any of the BWR candidate fuel assemblies, therefore if the cladding is not grossly breached and the fuel assembly is structurally sound it can be considered undamaged when loaded in an MPC-68M.

#### 6.III.4.5 Thoria Rod Canister

The criticality evaluation of thoria rod canister was performed for MPC-68 or MPC-68F and results presented in Section 6.4.6 show that it is permissible to load the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. While only a single canister is qualified for storage, the analysis assumes such a canister in every basket cell, and calculates a very low reactivity of less than 0.2 for this condition, based on a UO<sub>2</sub> content of 1.8 wt%. The conversion of Th-232 to U-233 during depletion may result in a slight increase in reactivity for the hypothetical case of a MPC-68 or MPC-68F entirely filled with Thoria Rod Canisters, however, the real condition of a single canister loaded together with spent fuel would still be bounded by the design basis case with fuel assemblies only. Since the MPC-68M has equal or better criticality performance than the MPC-68 due to the basket itself being made from the neutron absorber, Metamic-HT. Without any further evaluations it can therefore be concluded that, from a criticality perspective, the thoria rods with the actual composition can be safely stored in the HI-STORM 100 system in an MPC-68M canister.

TABLE 6.III.4.1

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M WITH UNDAMAGED AND  
DAMAGED FUEL/FUEL DEBRIS

Bare Rod Array inside the DFC	Maximum $k_{eff}$		
	Assembly Classes 8x8F and 9x9E/F (4.0 wt%)	Assembly Class 10x10F and 10x10G (4.6 wt%)	All other assembly classes (4.8 wt%)
3x3	0.8926	0.9310	0.9267
6x6	0.8942	0.9338	0.9295
8x8	0.8986	0.9395	0.9344
9x9	0.9028	0.9414	0.9371
10x10	0.9024	0.9432	0.9387
11x11	0.9024	0.9420	0.9381
12x12	0.9018	0.9412	0.9373
13x13	0.9007	0.9397	0.9353
14x14	0.8993	0.9385	0.9352
16x16	0.8985	0.9376	0.9335
17x17	0.8976	0.9366	0.9328

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.2

**MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-68M FOR VARIOUS  
FUEL TYPES WITH VOIDED PELLET TO CLAD GAP**

<b>Assembly Classes</b>	<b>Enrichment</b>	<b>Maximum <math>k_{\text{eff}}</math> (Voided Gap)</b>	<b>Reference <math>k_{\text{eff}}</math> (Flooded Gap)</b>	<b>Delta <math>k_{\text{eff}}</math></b>
7X7B	4.8	0.9185	0.9243	-0.0058
8x8B	4.8	0.9210	0.9294	-0.0084
8x8C	4.8	0.9243	0.9302	-0.0059
8x8D	4.8	0.9245	0.9307	-0.0062
8x8E	4.8	0.9152	0.9211	-0.0059
8x8F	4.5	0.9191	0.9245	-0.0054
9x9A	4.8	0.9290	0.9341	-0.0051
9x9B	4.8	0.9202	0.9330	-0.0128
9x9C	4.8	0.9203	0.9254	-0.0051
9x9D	4.8	0.9210	0.9254	-0.0044
9x9E	4.5	0.9157	0.9254	-0.0097
9x9G	4.8	0.9160	0.9211	-0.0051
10x10A	4.8	0.9311	0.9339	-0.0028
10x10B	4.8	0.9242	0.9332	-0.0090
10x10C	4.8	0.9253	0.9300	-0.0047
10x10F	4.7	0.9301	0.9335	-0.0034
10x10G	4.75	0.9403	0.9451	-0.0048

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.3  
 MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M FOR VARIOUS FUEL ROD  
 NUMBERS, OTHER STUDIES

Assembly Class 8x8B (4.8% Enrichment), variable fuel rod number				
Assembly Class	Maximum $k_{eff}$	Description	Reference	
8x8B	0.9256	Variable Rod Numbers	0.9294	-0.0038
Assembly Class 8x8D (4.8% Enrichment), variable fuel rod number and water rod number				
8x8D	0.9271	Variable Rod Numbers - 2L/2S	0.9307	-0.0036
8x8D	0.9256	Variable Rod Numbers - 4L	0.9307	-0.0051
8x8D	0.9275	Variable Rod Numbers - 4S	0.9307	-0.0032
8x8D	0.9259	Variable Rod Numbers - GE-9	0.9307	-0.0048
8x8D	0.9277	Variable Rod Numbers - 3L	0.9307	-0.0030
Assembly Class 9x9E/F (4.5% enrichment) With Alternate (Closer) Location of the Water Rods				
9x9E/F	0.9231	Closer Water Rods	0.9271	-0.0040



Table 6.III.4.4

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M  
FOR VARIOUS PART LENGTH ROD LENGTHS (10x10G, 4.75% Enrichment)

<b>Maximum <math>k_{eff}</math></b>	<b>Description</b>
0.9193	Full Length Rods Only
0.9279	Part Length Rods 25% length
0.9377	Part Length Rods 50% length
0.9439	Part Length Rods 75% length
0.9451	All Rods

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.5

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-68M  
FOR METAMIC POISON GAP TOLERANCE CALCULATION  
(10x10A, 4.8% Enrichment)

<b>Description</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
Reference	0.9339
Tolerance	0.9334

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.6

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M  
FOR ECCENTRIC FUEL ASSEMBLY POSITION  
(10x10A, 4.8% Enrichment)

<b>Position</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Delta-k</b>
In	0.9339	0.0074
Center	0.9265	reference
Out	0.9095	-0.0170

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.7

**MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-68M  
FOR EXTERNAL FLOODING**

Internal Water Density (%)	External Water Density (%)	7x7B (4.8%)	8x8F (4.5%)	9x9C (4.8%)	10x10A (4.8%)	10x10G (4.75%)
100	100	0.9243	0.9245	0.9254	0.9351	0.9451
100	70	0.9238	0.9250	0.9259	0.9353	0.9450
100	50	0.9235	0.9239	0.9249	0.9336	0.9456
100	20	0.9234	0.9245	0.9259	0.9342	0.9452
100	10	0.9234	0.9245	0.9257	0.9351	0.9446
100	05	0.9238	0.9247	0.9258	0.9346	0.9458
100	01	0.9230	0.9256	0.9261	0.9341	0.9459

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.8

**MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M  
FOR PARTIAL FLOODING**

10x10A (4.8%)		
Flooded Condition (% Full)	Maximum $k_{eff}$ , Horizontal Orientation	Maximum $k_{eff}$ , Vertical Orientation
25	0.7897	0.9122
50	0.9044	0.9290
75	0.9297	0.9326
100	0.9337	0.9337

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

TABLE 6.III.4.9

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M WITH LOW ENRICHED (3.3 wt%  $^{235}\text{U}$ ),  
CHANNELED, BWR FUEL

<b>Rod Array inside the Channel</b>	<b>Maximum <math>k_{eff}</math></b>
3x3	0.2045
6x6	0.7229
8x8	0.8900
9x9	0.9219
10x10	0.9248
11x11	0.9065
12x12	0.8689
13x13	0.8161
14x14	0.7562
16x16	0.6653
17x17	0.6449

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2

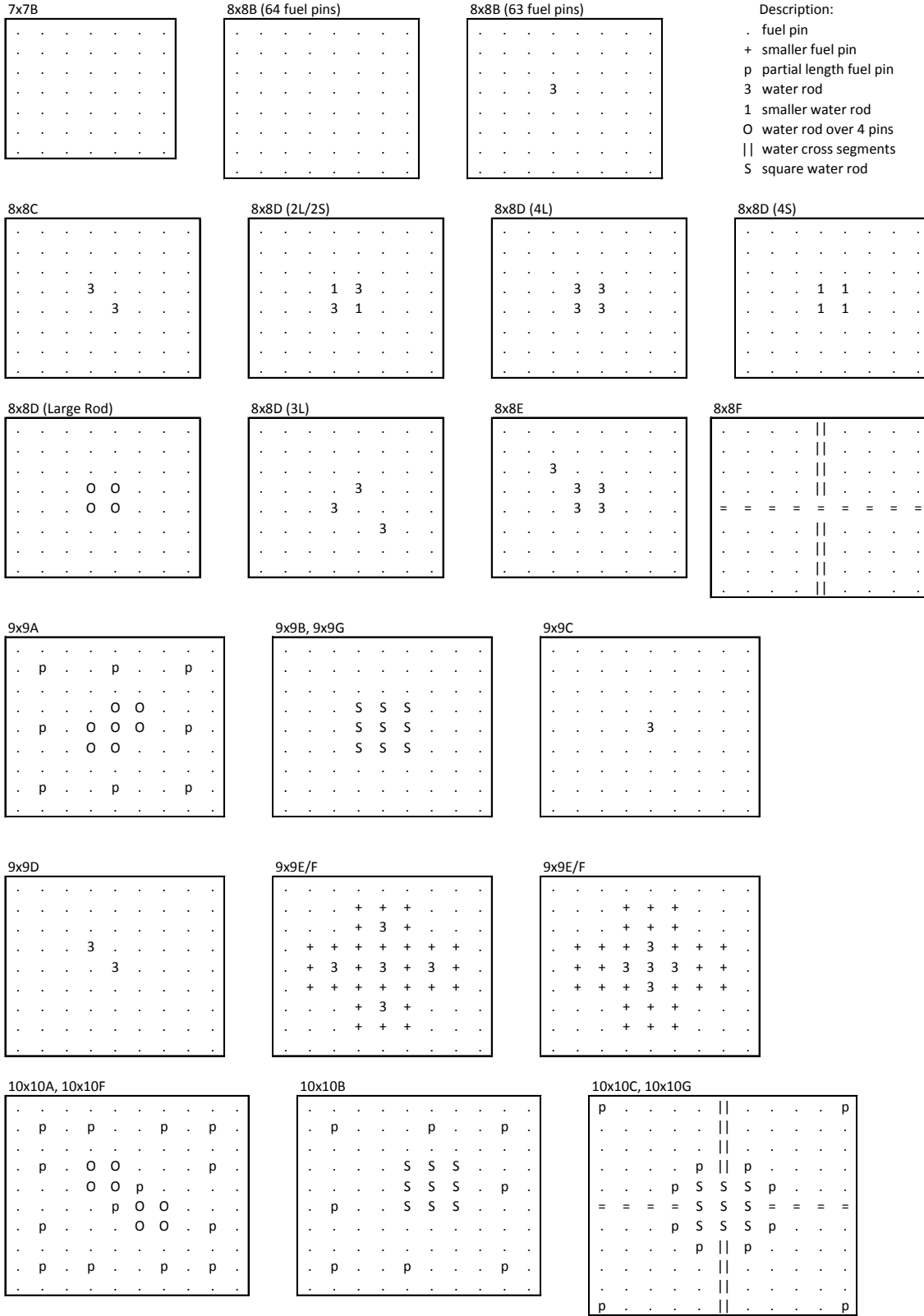
Table 6.III.4.10

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-68M WITH MIXTURE OF UNDAMAGED BWR FUEL  
AND LOW ENRICHED (3.3 wt%  $^{235}\text{U}$ ), CHANNЕLED BWR FUEL

Configuration	Rod Array	10x10A, 4.8 wt% $^{235}\text{U}$		10x10G, 4.75 wt% $^{235}\text{U}$	
		Maximum $k_{eff}$	Reactivity Effect	Maximum $k_{eff}$	Reactivity Effect
Undamaged Normal Fuel in all Cells	-	0.9339	Reference	0.9451	Reference
Checkerboard of CILC Fuel at 3.3 wt% $^{235}\text{U}$ and Undamaged Fuel	3x3	0.6218	-0.3121	0.6247	-0.3204
	6x6	0.8241	-0.1098	0.8281	-0.1170
	8x8	0.9110	-0.0229	0.9161	-0.0290
	9x9	0.9275	-0.0064	0.9329	-0.0122
	10x10	0.9297	-0.0042	0.9341	-0.0110
	11x11	0.9206	-0.0133	0.9264	-0.0187
	12x12	0.9054	-0.0285	0.9109	-0.0342
	13x13	0.8865	-0.0474	0.8913	-0.0538
	14x14	0.8666	-0.0673	0.8719	-0.0732
	16x16	0.8514	-0.0825	0.8563	-0.0888
17x17	0.8505	-0.0834	0.8561	-0.0890	

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Figure 6.III.4.1 Assembly Cross Sections



Part Length Rods only for 10x10G



### 6.III.5 CRITICALITY BENCHMARK EXPERIMENTS

Same as in Section 6.5

### 6.III.6 REGULATORY COMPLIANCE

Same as in Section 6.6

### 6.III.7 REFERENCES

- [6.III.1.1] NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington, D.C., January 1997.
- [6.III.1.2] 10CFR72.124, “Criteria for Nuclear Criticality Safety.”
- [6.III.1.3] USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 2 - July 1981.
- [6.III.1.4] “HI-STAR 100 AND HI-STORM 100 ADDITIONAL CRITICALITY CALCULATIONS”, Holtec Report HI-2012771 Rev.26 (proprietary)

## CHAPTER 7†: CONFINEMENT

### 7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STORM 100 System is provided by the MPC. The design of the HI-STORM 100 confinement boundary assures that there are no credible design basis events that would result in a radiological release to the environment. The HI-STORM 100 Overpack and HI-TRAC Transfer Cask are designed to provide physical protection for an MPC during normal, off-normal, and postulated accident conditions to assure that the integrity of the MPC confinement boundary is maintained. The inert atmosphere in the MPC and the passive heat removal capabilities of the HI-STORM 100 also assure that the SNF assemblies remain protected from degradation, which might otherwise lead to gross cladding ruptures during dry storage.

A detailed description of the confinement structures, systems, and components important to safety is provided in Chapter 2. The structural adequacy of the MPC is demonstrated by the analyses documented in Chapter 3. The physical protection of the MPC provided by the Overpack and the HI-TRAC Transfer Cask is demonstrated by the structural analyses documented in Chapter 3 and for off-normal and postulated accident conditions in Chapter 11. The heat removal capabilities of the HI-STORM 100 System are demonstrated by the thermal analyses documented in Chapter 4.

This chapter describes the HI-STORM 100 confinement boundary design and describes how the design satisfies the confinement requirements of 10CFR72 [7.0.1]. It also provides an evaluation of the MPC confinement boundary as it relates to the criteria contained in Interim Staff Guidance (ISG)-18 and ANSI N14.5-1997 [7.0.3] as justification for determining that leakage from the confinement boundary is not credible and, therefore, no confinement analysis is required.

This chapter is in compliance with NUREG-1536 except as noted in Table 1.0.2.

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

## 7.1 CONFINEMENT BOUNDARY

The primary confinement boundary against the release of radionuclides is the cladding of the individual fuel rods. The spent fuel rods are protected from degradation by maintaining an inert gas atmosphere (helium) inside the MPC and keeping the fuel cladding temperatures below the design basis values specified in Chapter 2.

The HI-STORM 100 confinement boundary consists of any one of the fully-welded MPC designs described in Chapter 1. Each MPC is identical from a confinement perspective so the following discussion applies to all MPCs. The confinement boundary of the MPC consists of:

- MPC shell
- bottom baseplate
- MPC lid (including the vent and drain port cover plates)
- MPC closure ring
- associated welds

The above items form a totally seal-welded vessel for the storage of design basis spent fuel assemblies.

The MPC requires no valves, gaskets or mechanical seals for confinement. Figure 7.1.1 shows an elevation cross-section of the MPC confinement boundary. All components of the confinement boundary are Important to Safety, Category A, as specified in Table 2.2.6. The MPC confinement boundary is designed and fabricated in accordance with the ASME Code, Section III, Subsection NB [7.1.1] to the maximum extent practicable. Chapter 2 provides design criteria for the confinement design. Section 2.2.4 provides applicable Code requirements. NRC-approved alternatives to specific Code requirements with complete justifications are presented in Table 2.2.15.

### 7.1.1 Confinement Vessel

The HI-STORM 100 System confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The MPC is designed, fabricated, inspected, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], including certain NRC-approved alternatives. The MPC shell and baseplate assembly and basket structure are delivered to the loading facility as one complete component. The MPC lid, vent and drain port cover plates, and closure ring are supplied separately and are installed following fuel loading. The MPC lid and closure ring are welded to the upper part of the MPC shell after fuel loading to provide redundant sealing of the confinement boundary. The vent and drain port cover plates are welded to the MPC lid after the lid is welded to the MPC. The welds forming the confinement boundary are described in detail in Section 7.1.3.

The MPC lid is made intentionally thick to minimize radiation exposure to workers during MPC closure operations, and is welded to the MPC shell. The vent and drain port cover plates are welded to the MPC lid following completion of MPC draining, moisture removal, and helium backfill activities to close the MPC vent and drain openings. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter. The combination of the welded MPC lid and closure ring form the redundant closure of the MPC.

Table 7.1.1 provides a summary of the design ratings for normal, off-normal and accident conditions for the MPC confinement vessel. Tables 1.2.2, 2.2.1, and 2.2.3 provide additional design basis information.

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.

Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by liquid penetrant method, volumetrically examined (or, if volumetric examination is not performed, multi-layer liquid penetrant examination must be performed), and pressure tested. If the MPC lid weld is acceptable, the vent and drain port cover plates are welded in place, examined by the liquid penetrant method and a leakage rate test is performed. Finally, the MPC closure ring is installed, welded and inspected by the liquid penetrant method. Chapters 8, 9, and 12 provide procedural guidance, acceptance criteria, and operating controls, respectively, for performance and acceptance of liquid penetrant examinations, volumetric examination, pressure testing and leakage rate testing of the field welds on the MPC.

After moisture removal, the MPC cavity is backfilled with helium. The helium backfill provides an inert atmosphere within the MPC cavity that precludes oxidation and hydride attack of the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and reduces the maximum fuel cladding temperatures. MPC inerting, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

#### 7.1.2 Confinement Penetrations

The MPC penetrations are designed to prevent the release of radionuclides under all normal, off-normal and accident conditions of storage. Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and for fuel cool-down and MPC flooding during unloading operations. No other confinement penetrations exist in the MPC. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain connectors allow the vent and drain ports to be operated like valves and prevent

the need to hot tap into the penetrations during unloading operations. The MPC vent and drain ports are sealed by cover plates that are seal welded to the MPC lid. No credit is taken for the seal provided by the vent and drain port caps. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, providing the redundant closure of the MPC vessel. The redundant closures of the MPC satisfy the requirements of 10CFR72.236(e) [7.0.1].

The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection.

### 7.1.3 Seals and Welds

The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], with certain NRC-approved alternatives. The MPC has no bolted closures or mechanical seals. Section 7.1.1 describes the design of the confinement vessel welds. The welds forming the confinement boundary are summarized in Table 7.1.2.

Confinement boundary welds are performed, inspected, and tested in accordance with the applicable requirements of ASME Section III, Subsection NB [7.1.1] with certain NRC-approved alternatives. The use of multi-pass welds, root pass, for multiple pass welds, and final surface liquid penetrant inspection, and volumetric examination essentially eliminates the chance of a pinhole leak through the weld. If volumetric examination is not performed, multi-layer liquid penetrant examination must be performed. Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The vent and drain port cover plate welds are helium leak tested in the field, providing added assurance of weld integrity. Additionally, a Code pressure test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity after fuel loading. The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld does not result in release of radioactive material to the environment. Table 9.1.4 provides a summary of the closure weld examinations and tests.

### 7.1.4 Closure

The MPC is a totally seal-welded pressure vessel. The MPC has no bolted closure or mechanical seals. The MPC's redundant closures are designed to maintain confinement integrity during normal conditions of storage, and off-normal and postulated accident conditions. There are no unique or special closure devices. Primary closure welds (lid-to-shell and vent/drain port cover plate-to-lid) are examined using the liquid penetrant technique to ensure their integrity. Additionally, the vent and drain port cover plate-to-MPC lid welds are helium leakage tested to be leaktight in accordance with ANSIN14.5-1997 [7.0.3]. A description of the MPC weld examinations is provided in Chapter 9.

Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required. Chapter 11 describes requirements for verifying the continued confinement capabilities of the MPC in the event of off-normal or accident conditions. As discussed in Section 2.3.3.2, no

instrumentation is required or provided for HI-STORM 100 System storage operations, other than normal security service instruments and TLDs.

#### 7.1.5 Damaged Fuel Container

The MPC is designed to allow for the storage of specified damaged fuel assemblies and fuel debris in a specially designed damaged fuel container (DFC). Fuel assemblies classified as damaged fuel or fuel debris as specified in Section 2.1.9 have been evaluated.

Damaged fuel assemblies and fuel debris will be loaded into stainless steel DFCs for storage in the HI-STORM 100 System. The DFCs that may be loaded into the MPCs are discussed in Section 2.1.3.

The DFC is designed to provide SNF loose component retention and handling capabilities. The DFC consists of a smooth-walled, welded stainless steel square container with a removable lid. The container lid provides the means of DFC closure and handling. The DFC is provided with stainless steel wire mesh screens in the top and bottom for draining, moisture removal and helium backfill operations. There are no other openings in the DFC. Section 2.1.9 specifies the fuel assembly characteristics for damaged fuel and fuel debris acceptable for loading in the MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F or MPC-68FF.

Since the DFC has screens on the top and bottom, the DFC provides no pressure retention function. The confinement function of the DFC is limited to minimizing the release of loose particulates within the sealed MPC. The confinement function of the MPC is not altered by the presence of the DFCs. The radioactive material available for release from the specified fuel assemblies are bounded by the design basis fuel assemblies analyzed herein.

#### 7.1.6 Design and Qualification of the MPC Lid-to-Shell Weld

The Holtec MPC lid-to-shell weld meets the criteria of NRC Interim Staff Guidance (ISG)-18 [7.1.2] such that leakage from the MPC lid-to-shell weld is not considered credible. Table 7.1.4 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one. In addition, because proper execution of the MPC lid-to-shell weld is vital to ensuring no credible leakage from the field-welded MPC, Holtec shall review the closure welding procedures for conformance to Code and FSAR requirements.

Table 7.1.1

## SUMMARY OF CONFINEMENT BOUNDARY DESIGN SPECIFICATIONS

<b>Design Condition</b>	<b>Design Pressure (psig)</b>	<b>Design Temperature* (°F)</b>
Normal	100	See Table 2.2.3
Off-Normal/Short-Term	110	
Accident	200	
*Temperature limits in Table 1.A.6 shall take precedence in case Duplex Stainless Steels are used for the fabrication of confinement boundary components.		

Table 7.1.2

## MPC CONFINEMENT BOUNDARY WELDS

<b>Confinement Boundary Welds</b>		
<b>MPC Weld Location</b>	<b>Weld Type†</b>	<b>ASME Code Category (Section III, Subsection NB)</b>
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port cover plates to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to closure ring radial	Partial Penetration Groove (field weld)	††
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C

† The tests and inspections for the confinement boundary welds are listed in Section 9.1.1.

†† This joint is governed by NB-5271 (liquid penetrant examination).



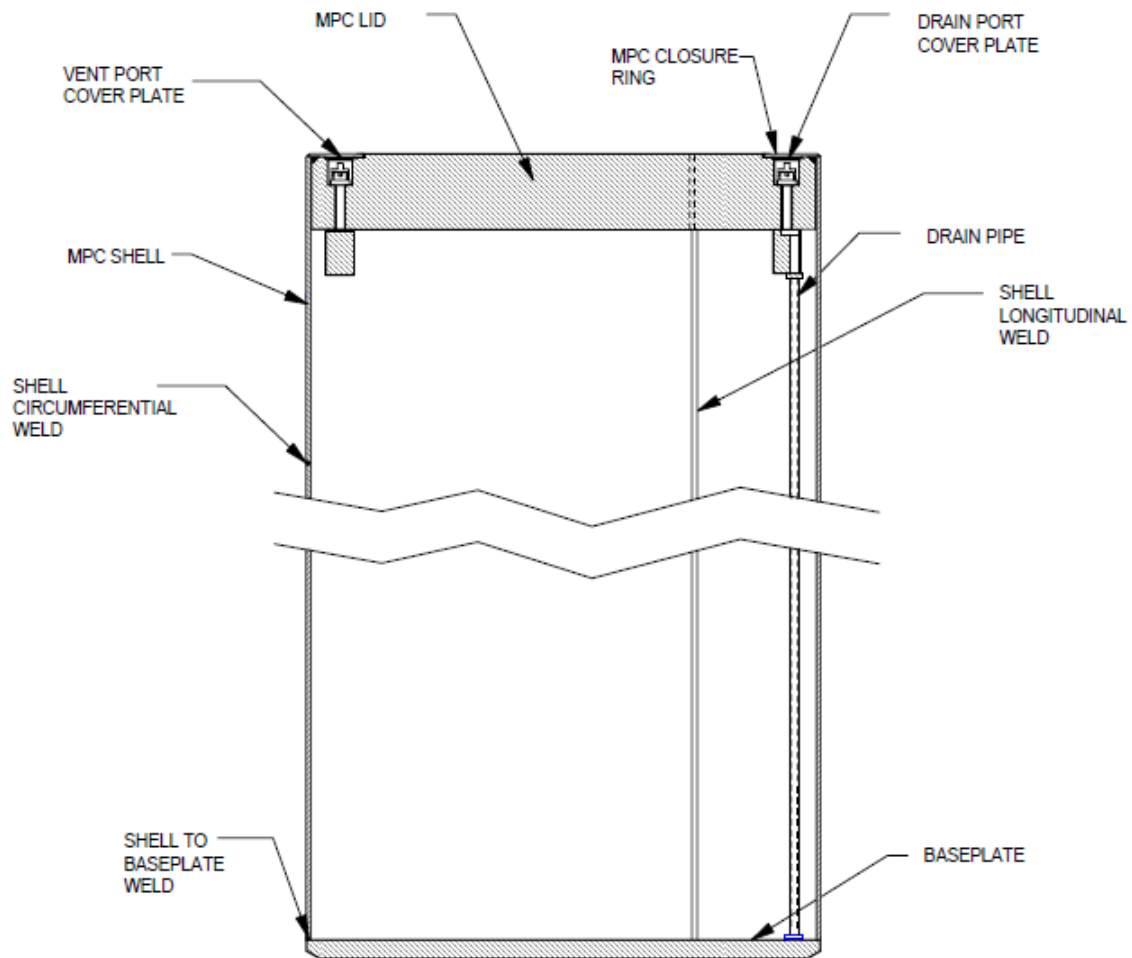
Table 7.1.3

TABLE DELETED

Table 7.1.4

## COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE FOR STORAGE

<b>DESIGN/QUALIFICATION GUIDANCE</b>	<b>HOLTEC MPC DESIGN</b>	<b>FSAR REFERENCE</b>
The canister is constructed from Alloy X type material	The MPC enclosure vessel is constructed entirely from Alloy X material. Alloy X is defined as Type 304, 304LN, 316, 316LN, or duplex stainless steel material. It should be noted that duplex steel material shall not be used for the fabrication of MPC baskets and internal components.	Section 1.2.1.1 and Appendix 1.A
The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3	The MPC lid-to-shell (LTS) closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8" of weld depth, which corresponds to the critical flaw size. A weld quality factor of 0.45 (45% of actual weld capacity) has been used in the stress analysis.	Section 9.1.1.1 and Tables 2.2.15 and 9.1.4.  HI-STAR FSAR Section 3.4.4.3.1.5 and Appendix 3.E (Docket 72-1008)
The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents, and natural phenomena	The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is design to withstand 45 g deceleration loadings and the cask system is analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g's.	Section 3.4.4.3 and Appendix 3.A.  HI-STAR FSAR Section 3.4.4.3
Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10 CFR 72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.	Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STORM CoC. Compliance with 10 CFR 72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.	Section 9.1.1.1 and Table 2.2.15  Section 13.0
Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.	The NRC has approved the Holtec quality assurance program under 10 CFR 71. That QA program approval has been adopted for activities governed by 10 CFR 72 as permitted by 10 CFR 72.140(d).	Section 13.0



**FIGURE 7.1.1: HI-STORM 100 SYSTEM CONFINEMENT BOUNDARY**

## 7.2 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

The MPC uses multiple confinement barriers provided by the fuel cladding and the MPC enclosure vessel to assure that there is no release of radioactive material to the environment. Chapter 3 shows that all confinement boundary components are maintained within their Code-allowable stress limits during normal and off-normal storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal and off-normal conditions of storage. Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Since the MPC confinement vessel remains intact, and the design bases temperatures and pressure are not exceeded, leakage from the MPC confinement boundary is not credible during normal and off-normal conditions of storage.

### 7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials, including fission gases, volatiles, fuel fines or crud, for postulated storage accident conditions. The analyses presented in Chapters 3 and 11 demonstrate that the MPC remains intact during all postulated accident conditions, including the associated increased internal pressure due to decay heat generated by the stored fuel. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], with certain NRC-approved alternatives as listed in Table 2.2.15. Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. In summary, there is no mechanistic failure that results in a breach of and associated leakage of radioactive material from the MPC confinement boundary.

7.4            REFERENCES

[7.0.1] 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

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

[7.0.3] ANSI N14.5-1997, American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment”.

[7.1.1] American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, Class 1 Components, 1995 Edition.

[7.1.2] Interim Staff Guidance 18, “The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation,” May 2003.

[7.2.1] Deleted

[7.2.2] Deleted.

[7.3.1] Deleted.

[7.3.2] Deleted.

[7.3.3] Deleted.

[7.3.4] Deleted.

[7.3.5] Deleted.

[7.3.6] Deleted.

[7.3.7] Deleted.

[7.3.8] Deleted.

[7.3.9] Deleted.

[7.3.10] Deleted.

[7.3.11] Deleted.

## **SUPPLEMENT 7.I**

### **CONFINEMENT EVALUATION OF THE HI-STORM 100U SYSTEM**

#### **7.I.1 INTRODUCTION**

This supplement is solely focused on providing an evaluation of confinement safety of the HI-STORM 100U System. The evaluation presented herein supplements those evaluations of the HI-STORM 100 and HI-STORM 100S overpacks contained in the main body of Chapter 7 of this FSAR.

#### **7.I.2 ACCEPTANCE CRITERIA**

The acceptance criteria for confinement evaluations for the HI-STORM 100U system are identical to the criteria for the HI-STORM 100 System analyzed in the main body of Chapter 7.

#### **7.I.3 EVALUATION**

The HI-STORM 100U System differs from the HI-STORM 100 System only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs are identical between the two systems.

In summary, as the MPC design is unchanged by the addition of the HI-STORM 100U VVM, the material presented in Section 7.1.6 certifying that the MPC meets the criteria of ISG-18, is valid for the HI-STORM 100U System. Therefore, leakage from the MPC confinement boundary is not considered credible and no confinement analysis is required.



## **SUPPLEMENT 7.III**

### **CONFINEMENT OF MPC-68M**

7.III.0 The main body of this chapter remains fully applicable for the HI-STORM 100 System using an MPC-68M except as indicated below since the MPC-68M fuel basket is used with the MPC enclosure vessel which is the confinement boundary of the system.

#### 7.III.1.5 Damaged Fuel Container

The MPC-68M is designed to allow for the storage of specified damaged fuel assemblies and fuel debris in a specially designed damaged fuel container (DFC). Section 2.III.1 specifies the fuel assembly characteristics for damaged fuel and fuel debris acceptable for loading in the MPC-68M.

## CHAPTER 8: OPERATING PROCEDURES<sup>†</sup>

### 8.0 INTRODUCTION:

This chapter outlines the loading, unloading, and recovery procedures for the HI-STORM 100 System for storage operations. The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance is met and the requirements of the CoC are met. The information provided in this chapter meets all requirements of NUREG-1536 [8.0.1].

Section 8.1 provides the guidance for loading the HI-STORM 100 System in the spent fuel pool. Section 8.2 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 8.3 provides the procedure for unloading the HI-STORM 100 System in the spent fuel pool. Section 8.4 provides the guidance for MPC transfer to the HI-STAR 100 Overpack for transport or storage. Section 8.4 can also be used for recovery of a breached MPC for transport or storage. Section 8.5 provides the guidance for transfer of the MPC into HI-STORM from the HI-STAR 100 transport overpack. Equipment specific operating details such as Vacuum Drying System, valve manipulation and Transporter operation are not within the scope of this FSAR and will be provided to users based on the specific equipment selected by the users and the configuration of the site.

The procedures contained herein describe acceptable methods for performing HI-STORM 100 loading and unloading operations. Unless otherwise stated, references to the HI-STORM 100 apply equally to the HI-STORM 100, 100S and 100S Version B. Users may alter these procedures to allow alternate methods and operations to be performed in parallel or out of sequence as long as the general intent of the procedure is met. In the figures following each section, acceptable configurations of rigging, piping, and instrumentation are shown. In some cases, the figures are artist's renditions. Users may select alternate configurations, equipment and methodology to accommodate their specific needs provided that the intent of this guidance is met and the requirements of the CoC are met. All rigging should be approved by the user's load handling authority prior to use. User-developed procedures and the design and operation of any alternate equipment must be reviewed by the Certificate holder prior to implementation.

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

Licenseses (Users) will utilize the procedures provided in this chapter, equipment-specific operating instructions, and plant working procedures and apply them to develop the site specific written, loading and unloading procedures.

The loading and unloading procedures in Section 8.1 and 8.3 can also be appropriately revised into written site-specific procedures to allow dry loading and unloading of the system in a hot cell or other remote handling facility. The Dry Transfer Facility (DTF) loading and unloading procedures are essentially the same with respect to loading removing moisture, and inerting, of the MPC. The dry transfer facility shall develop the appropriate site-specific procedures as part of the DTF facility license.

Tables 8.1.1 through 8.1.4 provide the handling weights for each of the HI-STORM 100 System major components and the loads to be lifted during various phases of the operation of the HI-STORM 100 System. Users shall take appropriate actions to ensure that the lift weights do not exceed user-supplied lifting equipment rated loads. Table 8.1.5 provides the HI-STORM 100 System bolt torque and sequencing requirements. Table 8.1.6 provides an operational description of the HI-STORM 100 System ancillary equipment along with its safety designation, where applicable. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in the Certificate of Compliance and as defined in Section 2.1.9 are loaded into the HI-STORM 100 System.

In addition to the requirements set forth in the CoC, users will be required to develop or modify existing programs and procedures to account for the operation of an ISFSI. Written procedures will be required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of items and components identified as Important to Safety, 10CFR72.48 [8.1.1] programs, specialized instrument calibration, special nuclear material accountability at the ISFSI, security modifications, fuel handling procedures, training and emergency response, equipment and process qualifications. Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. Users are also required to take necessary actions to prevent the fuel cladding from exceeding temperature limits during drying operations and during handling of the MPC in the HI-TRAC transfer cask. Section 4.5 of the FSAR provides requirements on the necessary actions, if any, based on the heat load of the MPC.

Table 8.1.7 summarizes some of the instrumentation used to load and unload the HI-STORM 100 System. Tables 8.1.8, 8.1.9, and 8.1.10 provide sample receipt inspection checklists for the HI-STORM 100 overpack, the MPC, and the HI-TRAC Transfer Cask, respectively. Users may develop site-specific receipt inspection checklists, as required for their equipment. Fuel handling shall be performed in accordance with written site-specific procedures.

## **Technical and Safety Basis for Loading and Unloading Procedures**

The procedures herein are developed for the loading, storage, unloading, and recovery of spent fuel in the HI-STORM 100 System. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present risks. The design of the HI-STORM 100 System, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize risks and mitigate consequences of potential events. To summarize, consideration is given in the loading and unloading systems and procedures to the potential events listed in Table 8.0.1.

The primary objective is to reduce the risk of occurrence and/or to mitigate the consequences of the event. The procedures contain Notes, Warnings, and Cautions to notify the operators to upcoming situations and provide additional information as needed. The Notes, Warnings and Cautions are purposely bolded and boxed and immediately precede the applicable steps.

In the event of an extreme abnormal condition (e.g., cask drop or tip-over event) the user shall have appropriate procedural guidance to respond to the situation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution and reporting to the appropriate regulatory agencies, as required.

Table 8.0.1  
OPERATIONAL CONSIDERATIONS

<b>POTENTIAL EVENTS</b>	<b>METHODS USED TO ADDRESS EVENT</b>	<b>COMMENTS/ REFERENCES</b>
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6. Procedural guidance is given for cask handling, inspection of lifting equipment, and proper engagement to the trunnions.	See Section 8.1.2.
Cask Tip-Over Prior to welding of the MPC lid	The Lid Retention System is available to secure the MPC lid during movement between the spent fuel pool and the cask preparation area.	See Section 8.1.5. See Figure 8.1.15.
Contamination of the MPC external shell	The annulus seal, pool lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water.	See Figures 8.1.13 and 8.1.14.
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.	See Figures 8.1.19-8.1.22.
Damage to fuel assembly cladding from oxidation	Fuel assemblies are never subjected to air or oxygen during loading and unloading operations.	See Section 8.1.5, and Section 8.3.3
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum Drying System is separate from pressurized gas and water systems.	See Figure 8.1.22 and 8.1.23.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to, and during welding or cutting activities. The space below the MPC lid shall be purged prior to, and during these activities.	See Section 8.1.5 and Section 8.3.3.

Table 8.0.1  
OPERATIONAL CONSIDERATIONS  
(CONTINUED)

<b>POTENTIAL EVENTS</b>	<b>METHODS USED TO ADDRESS EVENT</b>	<b>COMMENTS/ REFERENCES</b>
Excess dose from failed fuel assemblies	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.	See Figure 8.1.16 and Section 8.3.3.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.	See ALARA Notes and Warnings throughout the procedures.
Excess generation of radioactive waste	The HI-STORM system uses process systems that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Where possible, items are installed by hand and require no tools.	Examples: HI-TRAC bottom protective cover, bolt plugs in empty holes, pre-wetting of components.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.	See Section 8.1.4.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in stages to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. If the forced helium dehydration process used, the temperature of the gas exiting the demister is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations	<b>See Section 8.1.5</b>

Table 8.0.1  
OPERATIONAL CONSIDERATIONS  
(CONTINUED)

<b>POTENTIAL EVENTS</b>	<b>METHODS USED TO ADDRESS EVENT</b>	<b>COMMENTS/ REFERENCES</b>
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.	See Section 8.1.5.
Load Drop	Rigging diagrams and procedural guidance are provided for all lifts. Component weights are provided in Tables 8.1.1 through 8.1.4.	See Figures 8.1.6, 8.1.7, 8.1.9, 8.1.25 and 8.1.27. See Tables 8.1.1 through 8.1.4.
Over-pressurization of MPC during loading and unloading	Pressure relief valves in the water and gas processing systems limit the MPC pressure to acceptable levels.	See Figures 8.1.20, 8.1.21, 8.1.23 and 8.3.3.
Overstressing MPC lift lugs from side loading	The MPC is upended using the upending frame.	See Figure 8.1.6 and Section 8.1.2.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts.	See Section 8.1.7 for example. See Tables 8.1.1 through 8.1.4.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.	See Section 8.1.5 and Section 8.3.3.
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.	See Section 8.1.3 and Section 8.1.5.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during moisture removal and helium backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.	See Figure 8.1.11 and 8.1.16. See Section 8.3.3.

## 8.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

### 8.1.1 Overview of Loading Operations:

The HI-STORM 100 System is used to load, transfer and store spent fuel. Specific steps are performed to prepare the HI-STORM 100 System for fuel loading, to load the fuel, to prepare the system for storage and to place it in storage at an ISFSI. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC and/or HI-STORM may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications as long as the lift height restrictions are met (lift height restrictions apply only to suspended forms of transport). Users shall develop detailed written procedures to control on-site transport operations. Section 8.1.2 provides the general procedures for rigging and handling of the HI-STORM overpack and HI-TRAC transfer cask. Figure 8.1.1 shows a general flow diagram of the HI-STORM loading operations. The operations described below are envisioned to occur in a continuous fashion within a reasonable time frame. "Continuous fashion" is intended to be adherence to the natural flow of the loading procedure, from lowering the HI-TRAC into the spent fuel pool to placing the loaded HI-STORM onto the ISFSI pad, without extended delay. Delays due to shift change may be considered to be within the confines of continuous operation, while disruption of work for a full shift would be considered non-continuous operation.

Refer to the boxes of Figure 8.1.2 for the following description. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into HI-TRAC (Box 2). The annulus is filled with plant demineralized water<sup>†</sup> and the MPC is filled with either spent fuel pool water or plant demineralized water (borated as required) (Box 3). An inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC to prevent spent fuel pool water from contaminating the exterior surface of the MPC. HI-TRAC and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the optional Lid Retention System (Box 6). The lift yoke remotely engages to the HI-TRAC lifting trunnions to lift the HI-TRAC and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC from the spent fuel pool, the cask is removed from the spent fuel pool. If the Lid Retention System is being used, the HI-TRAC top lid bolts are installed to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

HI-TRAC is placed in the designated preparation area and the Lift Yoke and Lid Retention System (if utilized) are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The Temporary

<sup>†</sup> Users may substitute domestic water in each step where demineralized water is specified.



Shield Ring (if utilized) is installed and filled with water and the neutron shield jacket is filled with water<sup>1</sup> (if drained). The inflatable annulus seal is removed, and the annulus shield (if utilized) is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-TRAC during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the automated welding system (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. An ultrasonic or multi-layer PT examination is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The MPC Lid-to-Shell weld is then pressure tested followed by an additional liquid penetrant examination performed on the MPC Lid-to-Shell weld to verify structural integrity. To calculate the helium backfill requirements for the MPC (if the backfill is based upon helium mass or volume measurements), the free volume inside the MPC must first be determined. This free volume may be determined by measuring the volume of water displaced or any other suitable means.

Depending upon the burn-up or decay heat load of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system or forced helium dehydration system. For MPCs without high burn-up fuel and with sufficiently low decay heat, the vacuum drying system may be connected to the MPC and used to remove all liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and vacuum drying system lines. The internal pressure is reduced to below 3 torr and held for 30 minutes to ensure that all liquid water is removed.

For high-burn-up fuel or MPCs with high decay heat, or as an alternative for MPCs without high burn-up fuel and with lower decay heat, a forced helium dehydration system is utilized to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. The temperature of the gas exiting the system demister is maintained below 21°F for a minimum of 30 minutes to ensure that all liquid water is removed.

Following MPC moisture removal, the MPC is backfilled with a predetermined amount of helium gas. If the MPC contains high burn-up fuel and MPC heat load greater than the threshold heat load setting in Table 4.5.4, then a Supplemental Cooling System (SCS) is connected to the HI-TRAC annulus prior to helium backfill and is used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits (See Figure 2.C.1). The helium backfill ensures adequate heat transfer during storage, and provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes (for multi-pass welds) (Box

<sup>1</sup> Filled water jacket is relied in Section 4.5 thermal analysis of HI-TRAC in the dry and helium filled MPC condition.

10). The cover plate welds are then leak tested.

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield (if utilized) is removed and the remaining water in the annulus is drained. The Temporary Shield Ring (if utilized) is drained and removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination. HI-TRAC top lid<sup>4</sup> is installed and the bolts are torqued (Box 11). HI-TRAC surface dose rates are measured in accordance with the technical specifications. The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point on the MPC. MPC slings are installed between the MPC lift cleats and the lift yoke (Box 12).

If the HI-TRAC 125 is being used, the transfer lid is attached to the HI-TRAC as follows. The HI-TRAC is positioned above the transfer slide to prepare for bottom lid replacement. The transfer slide consists of an adjustable-height rolling carriage and a pair of channel tracks. The transfer slide supports the transfer step which is used to position the two lids at the same elevation and creates a tight seam between the two lids to eliminate radiation streaming. The overhead crane is shut down to prevent inadvertent operation. The transfer slide carriage is raised to support the pool lid while the bottom lid bolts are removed. The transfer slide then lowers the pool lid and replaces the pool lid with the transfer lid. The carriage is raised and the bottom lid bolts are replaced. The MPC lift cleats and slings support the MPC during the transfer operations. Following the transfer, the MPC slings are disconnected and HI-TRAC is positioned for MPC transfer into HI-STORM.

MPC transfer may be performed inside or outside the fuel building (Box 13). Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways (Box 14 and 15). The empty HI-STORM overpack is inspected and positioned with the lid removed. Vent duct shield inserts<sup>2</sup> are installed in the HI-STORM exit vent ducts. The vent duct shield inserts prevent radiation streaming from the HI-STORM Overpack as the MPC is lowered past the exit vents. If the HI-TRAC 100D or 125D is used, the mating device is positioned on top of the HI-STORM. The HI-TRAC is placed on top of HI-STORM. An alignment device (or mating device in the case of HI-TRAC 100D and 125D) helps guide HI-TRAC during this operation<sup>3</sup>. The MPC may be lowered using the MPC downloader, the main crane hook or other similar devices. The MPC downloader (if used) may be attached to the HI-TRAC lid or mounted to the overhead lifting device. The MPC slings are attached to the MPC lift cleats.

2 Vent duct shield inserts are only used on the HI-STORM 100.

3 The alignment guide may be configured in many different ways to accommodate the specific sites. See Table 8.1.6.

4 Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid

If used, the Supplemental Cooling System (SCS) will be disconnected from the HI-TRAC and the HI-TRAC annulus drained, prior to transfer of the MPC from the HI-TRAC to the HI-STORM. If the transfer doors are used (i.e. not the HI-TRAC 100D or 125D), the MPC is raised slightly, the transfer lid door locking pins are removed and the doors are opened. If the HI-TRAC 100D or 125D is used, the pool lid is removed and the mating device drawer is opened. Optional trim plates may be installed on the top and bottom of both doors (or drawer for HI-TRAC 100D and 125D) and secured using hand clamps. The trim plates eliminate radiation streaming above and below the doors (drawer). The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. The trim plates are removed, the doors (or drawer) are closed. The empty HI-TRAC must be removed with the doors open when the HI-STORM 100S is used to prevent interference with the lift cleats and slings. HI-TRAC is removed from on top of HI-STORM. The MPC slings and MPC lift cleats are removed. Hole plugs are installed in the empty MPC lifting holes to fill the voids left by the lift cleat bolts. The alignment device (or mating device with pool lid for HI-TRAC 100D and 125D) and vent duct shield inserts (if used) are removed, and the HI-STORM lid is installed. The exit vent gamma shield cross plates, temperature elements (if used) and vent screens are installed. The HI-STORM lid studs and nuts or lid closure bolts are installed. HI-STORM surface dose rates are measured in accordance with the technical specifications. The HI-STORM is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM Overpack and HI-TRAC transfer cask may be moved using a number of methods as long as the lifting equipment requirements are met. For sites with high seismic conditions, the HI-STORM 100A is anchored to the ISFSI. Once located at the storage pad, the inlet vent gamma shield cross plates are installed. Finally, the temperature elements and their instrument connections are installed (if used), and the air temperature rise testing (if required) is performed to ensure that the system is functioning within its design parameters.

### 8.1.2 HI-TRAC and HI-STORM Receiving and Handling Operations

**Note:**

HI-TRAC may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for HI-TRAC and HI-STORM handling. Site-specific procedures shall specify the required operational sequences based on the handling configuration at the sites.

1. Vertical Handling of HI-TRAC:
  - a. Verify that the lift yoke initial acceptance testing and any periodic recertifications are current, in accordance with ANSI N14.6.
  - b. Visually inspect the lifting device (lift yoke or lift links) and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Replace or repair damaged components as necessary.
  - c. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
  - d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

**Note:**

Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements.

**Warning:**

When lifting the loaded HI-TRAC with only the pool lid, the HI-TRAC should be carried as low as practicable. This minimizes the dose rates due to radiation scattering from the floor. Personnel should remain clear of the area and the HI-TRAC should be placed in position as soon as practicable.

- e. Raise HI-TRAC and position it accordingly.
2. Upending of HI-TRAC in the Transfer Frame:
- a. Position HI-TRAC under the lifting device. Refer to Step 1, above.
  - b. If necessary, remove the missile shield from the HI-TRAC Transfer Frame. See Figure 8.1.4.
  - c. Verify that the lift yoke initial acceptance testing and any periodic re-certifications are current, in accordance with ANSI N14.6.
  - d. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
  - e. Deleted.
  - f. Engage the lift yoke to the lifting trunnions. (The use of a ratchet strap or similar device to restrain the lift yoke arms is recommended during HI-TRAC Upending Operations.) See Figure 8.1.3.
  - g. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.
  - h. Slowly rotate HI-TRAC to the vertical position keeping all rigging as close to vertical as practicable. See Figure 8.1.4.
  - i. If used, lift the pocket trunnions clear of the Transfer Frame rotation trunnions.
3. Downending of HI-TRAC in the Transfer Frame:

**ALARA Warning:**

Due to the higher dose rates under the pool and transfer lid, use of auxiliary shielding should be considered when downending a loaded HI-TRAC.

- a. Position the Transfer Frame under the lifting device.
- b. Verify that the lift yoke initial acceptance testing and any periodic re-certifications are current, in accordance with ANSI N14.6.
- c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
- d. Deleted.

- e. Deleted.
  - f. Engage the lift yoke to the lifting trunnions. (The use of a ratchet strap or similar device to restrain the lift yoke arms is recommended during HI-TRAC Downending Operations.) See Figure 8.1.3.
  - g. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.
  - h. Position the pocket trunnions to receive the Transfer Frame rotation trunnions. See Figure 8.1.4 (Not used for HI-TRAC 100D and 125D).
  - i. Slowly rotate HI-TRAC to the horizontal position keeping all rigging as close to vertical as practicable.
  - j. Disengage the lift yoke.
4. Horizontal Handling of HI-TRAC in the Transfer Frame:

**ALARA Warning:**

Due to the higher dose rates under the pool and transfer lid, use of auxiliary shielding should be considered when handling a loaded HI-TRAC in horizontal orientation.

- a. Verify that the Transfer Frame is secured to the transport vehicle as necessary.
  - b. Downend HI-TRAC on the Transfer Frame per Step 3, if necessary.
  - c. If necessary, install the HI-TRAC missile Shield on the HI-STAR 100 Transfer Frame (See Figure 8.1.4).
5. Vertical Handling of HI-STORM:

**Note:**

The HI-STORM 100 Overpack may be lifted with a special lifting device that engages the overpack anchor blocks with threaded studs and connects to a cask transporter, crane, or similar equipment. The device is designed in accordance with ANSI N14.6.

- a. Visually inspect the HI-STORM lifting device for gouges, cracks, deformation or other indications of damage.
  - b. Visually inspect the transporter lifting attachments for gouges, cracks, deformation or other indications of damage.
  - c. If necessary, attach the transporter's lifting device to the transporter and HI-STORM.
  - d. Raise and position HI-STORM accordingly. See Figure 8.1.5.
6. Empty MPC Installation in HI-TRAC:

**Note:**

To avoid side loading the MPC lift lugs, the MPC must be upended in the MPC Upending Frame (or equivalent). See Figure 8.1.6.

- a. If necessary, rinse off any road dirt with water. Remove any foreign objects from the MPC internals.

- b. If necessary, upend the MPC as follows:
1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
  2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 8.1.6.
  3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 8.1.6.
  4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device) (See Figure 8.1.6).
  5. Raise the MPC in the Upending Frame.

**Warning:**

The Upending Frame corner should be kept close to the ground during the upending process.

6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.
  7. When the MPC approaches the vertical orientation, tension on the lower slings may be released.
  8. Place the MPC in a vertical orientation.
  9. Disconnect the MPC straps and disconnect the rigging.
- c. Install the MPC in HI-TRAC as follows:
1. Install the four point lift sling to the lift lugs inside the MPC. See Figure 8.1.7.
  2. Raise and place the MPC inside HI-TRAC.

**Note:**

An alignment punch mark is provided on HI-TRAC and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring. See Figure 8.1.8.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside HI-TRAC. Disconnect the MPC rigging or the MPC lift rig.

### 8.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

**Note:**

Receipt inspection, installation of the empty MPC in the HI-TRAC, and lower fuel spacer installation may occur at any location or be performed at any time prior to complete submersion in the spent fuel pool as long as appropriate steps are taken to prevent contaminating the exterior of the MPC or interior of the HI-TRAC.

**ALARA Note:**

A bottom protective cover may be attached to HI-TRAC pool lid bottom. This will help prevent imbedding contaminated particles in HI-TRAC bottom surface and ease the decontamination effort.

1. Place HI-TRAC in the cask receiving area. Perform appropriate contamination and security surveillances, as required.
2. If necessary, remove HI-TRAC Top Lid by removing the top lid bolts and using the lift sling. See Figure 8.1.9 for rigging.
  - a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.
  - b. Perform a radiological survey of the inside of HI-TRAC to verify there is no residual contamination from previous uses of the cask.
3. Disconnect the rigging.
4. Store the Top Lid and bolts in a site-approved location.
5. If necessary, configure HI-TRAC with the pool lid as follows:

**ALARA Warning:**

The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

- a. Inspect the seal on the pool lid for cuts, cracks, gaps and general condition. Replace the seal if necessary.
  - b. Remove the bottom lid bolts and store them temporarily.
  - c. Raise the empty HI-TRAC and position it on top of the pool lid.
  - d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - e. Install the pool lid bolts. See Table 8.1.5 for torque requirements.
  - f. If necessary, thread the drain connector pipe to the pool lid.
  - g. Store the HI-TRAC Transfer Lid in a site-approved location.
6. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

7. Install the MPC inside HI-TRAC and place HI-TRAC in the designated preparation area. See Section 8.1.2.

**Note:**

Upper fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Upper fuel spacer installation may occur any time prior to MPC lid installation.

8. Install the upper fuel spacers in the MPC lid as follows:

**Warning:**

Never work under a suspended load.

- a. Position the MPC lid on supports to allow access to the underside of the MPC lid.
  - b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Figure 8.1.10 and Table 8.1.5 for torque requirements.
  - c. Install threaded plugs in the MPC lid where and when spacers will not be installed, if necessary. See Table 8.1.5 for torque requirements.
9. At the user's discretion perform an MPC lid and closure ring fit test:

**Note:**

It may be necessary to perform the MPC installation and inspection in a location that has sufficient crane clearance to perform the operation.

- a. Visually inspect the MPC lid rigging (See Figure 8.1.9).
- b. At the user's discretion, raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid. See Figure 8.1.11.

**Note:**

The MPC Shell is relatively flexible compared to the MPC Lid and may create areas of local contact that impede Lid insertion in the Shell. Grinding of the MPC Lid below the minimum diameter on the drawing is permitted to alleviate interference with the MPC Shell in areas of localized contact. If the amount of material removed from the surface exceeds 1/8", the surface shall be examined by a liquid penetrant method (NB-2546). The weld prep for the Lid-to-Shell weld shall be maintained after grinding.

- c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. See Figure 8.1.12. Install the MPC lid. Verify that the MPC lid fit and weld prep are in accordance with the design drawings.

**ALARA Note:**

The closure ring is installed by hand. Some grinding may be required on the closure ring to adjust the fit.

- d. Install, align and fit-up the closure ring.
- e. Verify that closure ring fit and weld prep are in accordance with the fabrication drawings or the approved design drawings.



- f. Remove the closure ring, vent and drain port cover plates and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.
10. At the user's discretion, perform an MPC vent and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

**Note:**

Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used.

11. Install lower fuel spacers in the MPC (if necessary). See Figure 8.1.10.
12. Fill the MPC and annulus as follows:
- a. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

**Caution:**

Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

- b. Manually insert the inflatable annulus seal around the MPC. See Figure 8.1.13.
- c. Ensure that the seal is uniformly positioned in the annulus area.
- d. Inflate the seal.
- e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

**ALARA Note:**

Bolt plugs installed, and/or waterproof tape over empty bolt holes, reduce the time required for decontamination.

13. At the user's discretion, install HI-TRAC top lid bolt plugs and/or apply waterproof tape over any empty bolt holes.

**ALARA Note:**

Keeping the water level below the top of the MPC prevents splashing during handling.

14. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches below the top of the MPC shell. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements.
15. If necessary for plant crane capacity limitations, drain the water from the neutron shield jacket. See Tables 8.1.1 through 8.1.4 as applicable.
16. Place HI-TRAC in the spent fuel pool as follows:

**ALARA Note:**

The term "Spent Fuel Pool" is used generically to refer to the users designated cask loading location. The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

- a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-TRAC. See Figure 8.1.14.
- b. Verify spent fuel pool for boron concentration requirements in accordance with Tables 2.1.14 and 2.1.16.
- c. Engage the lift yoke to HI-TRAC lifting trunnions and position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

**ALARA Note:**

Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- d. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.
- e. When the top of the HI-TRAC reaches the elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.
- f. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged. Remove the lift yoke from the spent fuel pool while spraying the crane cables and yoke with plant demineralized water.
- g. Observe the annulus seal for signs of air leakage. If leakage is observed (by the steady flow of bubbles emanating from one or more discrete locations) then immediately remove the HI-TRAC from the spent fuel pool and repair or replace the seal.

#### 8.1.4 MPC Fuel Loading

**Note:**

An underwater camera or other suitable viewing device may be used for monitoring underwater operations.

**Note:**

When loading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading as specified in Section 2.1.9 have been selected for loading into the MPC.

**Caution:**

In accordance with the definition of “Undamaged Fuel,” some low-enriched channeled fuel must be shown to be without known or suspected grossly breached spent fuel rods. This determination can be made based on review of records, fuel sipping, or other method.

2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.
3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

8.1.5 MPC Closure**Note:**

The user may elect to use the Lid Retention System (See Figure 8.1.15) to assist in the installation of the MPC lid and lift yoke, and to provide the means to secure the MPC lid in the event of a drop accident during loaded cask handling operations outside of the spent fuel pool. The user is responsible for evaluating the additional weight imposed on the cask, lift yoke, crane and floor prior to use. See Tables 8.1.1 through 8.1.4 as applicable. The following guidance describes installation of the MPC lid using the lift yoke. The MPC lid may also be installed separately.

Depending on facility configuration, users may elect to perform MPC closure operations with the HI-TRAC partially submerged in the spent fuel pool. If opted, operations involving removal of the HI-TRAC from the spent fuel pool shall be sequenced accordingly.

1. Remove the HI-TRAC from the spent fuel pool as follows:
  - a. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.
  - b. Install the drain line to the underside of the MPC lid. See Figure 8.1.17.
  - c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage. See Figure 8.1.11 and 8.1.17.

**ALARA Note:**

Pre-wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- d. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented. See Figure 8.1.12.

- e. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

**Note:**

The outer diameter of the MPC lid will seat flush with the top edge of the MPC shell when properly installed. Once the MPC lid is installed, the HI-TRAC /MPC removal from the spent fuel pool should proceed in a continuous manner to minimize the rise in MPC water temperature.

- f. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.
- g. Engage the lift yoke to HI-TRAC lifting trunnions.
- h. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

**ALARA Note:**

Activated debris may have settled on the top face of HI-TRAC and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. Users are responsible for any water dilution considerations.

- i. Raise HI-TRAC until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Remove any activated or highly radioactive particles from HI-TRAC or MPC.
- j. Visually verify that the MPC lid is properly seated. Lower HI-TRAC, reinstall the lid, and repeat as necessary.
- k. Install the Lid Retention System bolts if the lid retention system is used.
- l. Continue to raise the HI-TRAC under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-TRAC reaches the same elevation as the reservoir, close the Annulus Overpressure System reservoir valve (if used). See Figure 8.1.14.

**Caution:**

Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action. Users shall initiate a course of action which involves initiating and maintaining forced water circulation at a required flow rate through the MPC. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements whenever water is added to the loaded MPC.

- m. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

**ALARA Note:**

Decontamination of HI-TRAC bottom should be performed using remote cleaning methods, covering or other methods to minimize personnel exposure. The bottom lid decontamination may be deferred to a convenient and practical time and location. Any initial decontamination should only be sufficient to preclude spread of contamination within the fuel building.

- n. Decontaminate HI-TRAC bottom and HI-TRAC exterior surfaces including the pool lid bottom. Remove the bottom protective cover, if used.
- o. If used, disconnect the Annulus Overpressure System from the HI-TRAC See Figure 8.1.14.
- p. Set HI-TRAC in the designated cask preparation area.

**Note:**

If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Users shall evaluate the cask weights to ensure that cask trunnion, lifting devices and equipment load limitations are not exceeded.

- q. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.
- r. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

**Warning:**

MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

- s. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.
- t. Perform decontamination and a dose rate/contamination survey of HI-TRAC.
- u. Prepare the MPC annulus for MPC lid welding as follows:

**ALARA Note:**

If the Temporary Shield Ring is not used, some form of gamma shielding (e.g., lead bricks or blankets) should be placed in the trunnion recess areas of the HI-TRAC water jacket to eliminate the localized hot spot.

- v. Decontaminate the area around the HI-TRAC top flange and install the Temporary Shield Ring, (if used). See Figure 8.1.18.

**ALARA Note:**

The water in the HI-TRAC-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

- w. Attach the drain line to the HI-TRAC drain port and lower the annulus water level approximately 6 inches.
2. Prepare for MPC lid welding as follows:

**Note:**

The following steps use two identical Removable Valve Operating Assemblies (RVOAs) (See Figure 8.1.16) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

**Note:**

The vent and drain ports are opened by pushing the RVOA operating rod down to engage the drive mechanism on the MPC port connection and turning the operating rod fully in the counter-clockwise direction. The operating rod will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the operating rod fully in the clockwise direction. The ports are closed when the operating rod cannot be turned further.

**Note:**

Steps involving preparation for welding may occur in parallel as long as precautions are taken to prevent contamination of the annulus.

- a. Clean the vent and drain ports to remove any dirt. Install the RVOAs (See Figure 8.1.16) to the vent and drain ports leaving caps open.

**ALARA Warning:**

Personnel should remain clear of the drain hoses any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

**Caution:**

Personnel shall ensure that the water level is not lowered below the top of the fuel cladding to avoid exposing the fuel to atmosphere to prevent oxidation and potential fuel damage.

- b. Attach the water pump to the drain port (See Figure 8.1.19) and lower the water level to keep moisture away from the weld region.
- c. Disconnect the water pump.
- d. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable seal
- e. Deflate and remove the inflatable annulus seal.

**ALARA Note:**

The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontaminated.

- f. Survey the MPC lid top surfaces and the accessible areas of the top three inches of the MPC.

**ALARA Note:**

The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

- g. Install the annulus shield. See Figure 8.1.13.

## 3. Weld the MPC lid as follows:

**ALARA Warning:**

Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

**ALARA Warning:**

It may be necessary to rotate or reposition the MPC lid slightly to achieve uniform weld gap and lid alignment. A punch mark is located on the outer edge of the MPC lid and shell. These marks are aligned with the alignment mark on the top edge of the HI-TRAC Transfer Cask (See Figure 8.1.8). If necessary, the MPC lid lift should be performed using a hand operated chain fall to closely control the lift to allow rotation and repositioning by hand. If the chain fall is hung from the crane hook, the crane should be tagged out of service to prevent inadvertent use during this operation. Continuous radiation monitoring is recommended.

- a. If necessary center the lid in the MPC shell using a hand-operated chain fall.

**Note:**

The MPC is equipped with lid shims that serve to close the gap in the joint for MPC lid closure weld.

- b. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

**ALARA Note:**

The AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

- c. Install the Automated Welding System (AWS) baseplate shield. See Figure 8.1.9 for rigging.
- d. If used, install the Automated Welding System Robot.

**Note:**

It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.

**Note:**

Combustible gas monitoring as described in Step 3e and the associated Caution block are required by the HI-STORM 100 CoC (CoC Appendix B (and B-100U), Section 3.8) and may not be deleted without prior NRC approval via CoC amendment.

**Caution:**

Oxidation of neutron absorber panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Additionally, radiolysis of the water may occur in high flux conditions creating additional combustible gases. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid welding operations per CoC Appendix B (and B-100U), Section 3.8. The space below the MPC lid shall be purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space.

- e. Perform combustible gas monitoring and purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.
  - f. Perform the MPC lid-to-shell weld and NDE with approved procedures (See 9.1 and Table 2.2.15).
  - g. Deleted.
  - h. Deleted.
  - i. Deleted.
  - j. Deleted.
4. Perform MPC Lid-to-Shell weld pressure testing as follows:

**ALARA Note:**

Testing is performed before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

- a. If performing a hydrostatic test, attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system and connect the pressurized water supply to the drain port. If performing a pneumatic test, attach the pressure supply and vent line to the vent port and route the vent line to a suitable radwaste connection. See Figure 8.1.20 for the pressure test arrangement.

**ALARA Warning:**

Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.



- b. If performing a hydrostatic test, fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements.
- c. Perform the pressure test of the MPC as follows:
1. Close the drain/vent valve and pressurize the MPC to minimum test pressure listed in Table 2.0.1 +5/-0 psig.
  2. Close the supply valve and monitor the pressure for a minimum of 10 minutes. The pressure shall not drop below the minimum test pressure during the performance of the test.
  3. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water (hydrostatic test) or helium using a bubble test solution (pneumatic test). The acceptance criterion is no observable leakage.
- d. Release the MPC internal pressure, disconnect the inlet line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.
1. Repeat the liquid penetrant examination on the MPC lid final pass.
- e. Repair any weld defects in accordance with the site's approved weld repair procedures. Re-perform the Ultrasonic (if necessary), PT, and pressure tests if weld repair is performed.
5. Drain the MPC as follows:

<b>Caution:</b>
This Caution block is required by the HI-STORM 100 CoC (CoC Appendix B, Section 3.4.10 and Appendix B-100U, Section 3.4.12) and may not be deleted without prior NRC approval via CoC amendment. To prevent the oxidation of the fuel the MPC interior shall be filled with helium or another suitable inert gas to avoid exposing the fuel to oxidizing agents while at elevated temperatures. Exposing fuel at elevated temperatures to oxidizing agents can lead to deleterious oxidation of the fuel.
<b>Caution:</b>
For MPCs above a threshold heat load (see Technical Specifications), vacuum drying is subject to a time limit starting at completion of this operational step and ending when helium backfill is initiated. Do not begin this step for such an MPC unless intending to proceed shortly thereafter to vacuum drying. In addition, users should ensure procedures for cycling are in place if time limits are applicable. The procedures should include the required backfilling pressure, hold times, and duration of subsequent vacuum drying cycles, understanding that the number of cycles is limited to ten per ISG-11 Rev. 3.

- a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20.

**ALARA Warning:**

Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

- b. Attach the water fill line to the drain port and fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the drain line.
- c. Disconnect the water fill and drain lines from the MPC leaving the vent port valve open to allow for thermal expansion of the MPC water.

**ALARA Warning:**

Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

- d. Attach a regulated helium or nitrogen supply to the vent port.
- e. Attach a drain line to the drain port shown on Figure 8.1.21.
- f. Deleted
- g. Verify the correct pressure on the gas supply.
- h. Open the gas supply valve and record the time at the start of MPC draining.

**Note:**

An optional warming pad may be placed under the HI-TRAC Transfer Cask to replace the heat lost during the evaporation process of MPC drying. This may be used at the user's discretion for older and colder fuel assemblies to reduce vacuum drying times.

- i. Start the warming pad, if used.

**Note:**

Users may continue to purge the MPC to remove as much water as possible.

- j. Drain the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve. See Figure 8.1.21.
- k. Deleted.
- l. Disconnect the gas supply line from the MPC.
- m. Disconnect the drain line from the MPC.

**Note:**

Vacuum drying or moisture removal using FHD (for high burn-up fuel or high decay heat) is performed to remove moisture and oxidizing gasses from the MPC. This ensures a suitable environment for long-term storage of spent fuel assemblies and ensures that the MPC pressure remains within design limits. The vacuum drying process described herein reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC. The moisture removal process limits bulk MPC temperatures by continuously circulating gas through the MPC. Section 8.1.5 Steps 6a through q are used for the vacuum drying method of drying and backfill. Section 8.1.5 Steps 7a through i are used for the FHD method of drying and backfill.

## 6. Dry and Backfill the MPC as follows (Vacuum Drying Method):

**Note:**

During vacuum drying, the annulus between the MPC and the HI-TRAC must be maintained full of water. Water lost due to evaporation or boiling must be replaced to maintain the water level. The vacuum drying process described herein reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC.

- a. Fill the annulus between the MPC and HI-TRAC with clean water. The water level must be within 6" of the top of the MPC.
- b. Attach the drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a. Other equipment configurations that achieve the same results may also be used.

**Note:**

The vacuum drying system may be configured with an optional fore-line condenser. Other equipment configurations that achieve the same results may be used.

**Note:**

To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

**Caution:**

For MPC heat loads greater than defined threshold limits, cyclic vacuum drying can be used. The method and acceptance criteria to determine the cycle time limits are specified in Section 4.5. Once vacuum drying time limit is reached for a cycle, MPC shall be backfilled with helium as described in Section 4.5 and vacuum drying cycle shall be repeated.

- c. Open the VDS suction valve and reduce the MPC pressure to below 3 torr. Refer to Section 4.5 for method to determine vacuum drying time and cycle limits. If MPC vacuum drying acceptance criteria are not met during allowable time, backfill the MPC cavity with helium to a pressure of  $\geq 0.5$  atm and reset the vacuum drying time (see Technical Specifications).

**Note:**

Helium used for drying shall be in accordance with the Technical Specification using 99.995% (minimum) purity. Other equipment configurations that achieve the same results may be used.

- d. Shut the VDS valves and verify a stable MPC pressure on the vacuum gage.

**Note:**

The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the vacuum drying system, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

- e. Perform the MPC drying pressure test in accordance with the technical specifications. If MPC vacuum drying acceptance criteria are not met during allowable time, backfill the MPC cavity with helium to a pressure of  $\geq 0.5$  atm and reset the vacuum drying time (see Technical Specifications) and restart vacuum drying.
- f. Close the vent and drain port valves.
- g. Disconnect the VDS from the MPC.
- h. Stop the warming pad, if used.
- i. Close the drain port RVOA cap and remove the drain port RVOA.

**Note:**

Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. Other equipment configurations that achieve the same results may be used.

- j. Set the helium bottle regulator pressure to the appropriate pressure.
- k. Purge the Helium Backfill System to remove oxygen from the lines.
- l. Attach the Helium Backfill System to the vent port as shown on Figure 8.1.23 and open the vent port.
- m. Slowly open the helium supply valve while monitoring the pressure rise in the MPC.

**Note:**

If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.

- n. Carefully backfill the MPC in accordance with the technical specifications
- o. If used, stop the water flow through the annulus between the MPC and HI-TRAC. Drain the water from the annulus
- p. Disconnect the helium backfill system from the MPC.
- q. Close the vent port RVOA and disconnect the vent port RVOA.
7. Dry and Backfill the MPC as follows (FHD Method):

**Note:**

Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. When using the FHD system to perform the MPC helium backfill, the FHD system shall be evacuated or purged and the system operated with 99.995% (minimum) purity helium.

**Note:**

MPC internal pressure during FHD drying only must be  $\leq 75$  psia, to comply with Technical Specification. The MPC internal pressure during backfill operations must be maintained below the MPC normal design pressure (100 psig) maximum.

**Caution:**

In the event of an FHD System failure, the MPC cavity pressure must be set to at least 72 psig to place the MPC in an acceptable condition.

- a. Attach the moisture removal system to the vent and drain port RVOAs. See Figure 8.1.22b. Other equipment configurations that achieve the same results may also be used.
  - b. Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.
  - c. Continue the monitoring and moisture removal until LCO 3.1.1 is met for MPC dryness.
  - d. Continue operation of the FHD system with the demoinsturizer on.
  - e. While monitoring the temperatures into and out of the MPC, adjust the helium pressure in the MPC to provide a fill pressure as required by the technical specifications.
  - f. Open the FHD bypass line.
  - g. Close the vent and drain port RVOAs.
  - h. Shutdown the FHD system and disconnect it from the RVOAs.
  - i. Remove the vent and drain port RVOAs.
8. Weld the vent and drain port cover plates as follows:

**Note:**

The process provided herein may be modified to perform actions in parallel.

- a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.
- b. Place the cover plate over the vent port recess.
- c. Deleted.

**Note:**

ASME Boiler and Pressure Vessel Code [8.1.3], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

- d. Weld cover plate and perform NDE on the cover plate with approved procedures (See 9.1 and Table 2.2.15)
  - e. Repair and weld defects in accordance with the site's approved code weld repair procedures.
  - f. Perform a helium leakage rate test on the cover plate welds. (See 9.1 and Table 2.2.15). Acceptance Criteria are defined in Technical Specification LCO 3.1.1.
  - g. Repair any weld defects in accordance with the site's approved code weld repair procedures.
  - h. Deleted.
  - i. Repeat for the drain port cover plate.
9. Perform a leakage test of the MPC vent and drain port cover plates as follows:

**Note:**

The leakage detector may detect residual helium in the atmosphere from the helium injection process. If the leakage tests detect a leak, the area should be blown clear with compressed air or nitrogen and the location should be retested.

**Note:**

The following process provides a high concentration of helium gas in the cavity. Other methods that ensure a high concentration of helium gas are also acceptable.

- a. If necessary, remove the cover plate set screws.
- b. Flush the cavity with helium to remove the air and immediately install the set screws recessed below flush with the top of the cover plate.
- c. Plug weld the recess above each set screw to complete the penetration closure welding.

**Note:**

ASME Boiler and Pressure Vessel Code [8.1.3], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

- d. Perform a liquid penetrant examination on the plug weld.

- e. Flush the area around the vent and drain cover plates with compressed air or nitrogen to remove any residual helium gas.
- f. Perform a helium leakage rate test of vent and drain cover plate welds in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [8.1.2]. The MPC Helium Leak Rate acceptance criteria is provided in the Technical Specification LCO 3.1.1.
- g. Repair any weld defects in accordance with the site's approved code weld repair procedures. Re-perform the leakage test as required.

10. Weld the MPC closure ring as follows:

**ALARA Note:**

The closure ring is installed by hand. No tools are required. Localized grinding to achieve the desired fit and weld prep are allowed.

- a. Install and align the closure ring. See Figure 8.1.8.
- b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE with approved procedures (See 9.1 and Table 2.2.15).
- c. Deleted.
- d. Deleted.
- e. Deleted.
- f. Deleted.
- g. Deleted.
- h. Deleted.
- i. Deleted.
- j. If necessary, remove the AWS. See Figure 8.1.7 for rigging.

8.1.6 PREPARATION FOR STORAGE

**ALARA Warning:**

Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.

**Caution:**

Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.

**Note:**

For an MPC containing high burn-up fuel assemblies and MPC heat load greater than the threshold heat load setting in Table 4.5.4, the Supplemental Cooling System is required to be operated within LCO 3.1.4 time limits following completion of backfill (see Section 4.5). In the event of a Supplemental Cooling System failure, a HI-TRAC in a horizontal orientation must be placed into a vertical orientation within 24 hours.

- 1. Remove the annulus shield (if used) and store it in an approved plant storage location

2. If use of the SCS is not required, attach a drain line to the HI-TRAC and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system.
3. Install HI-TRAC top lid as follows:

**Warning:**

When traversing the MPC with the HI-TRAC top lid using non-single-failure proof (or equivalent safety factors), the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

- a. Install HI-TRAC top lid. Inspect the bolts for general condition. Replace worn or damaged bolts with new bolts.
  - b. Install and torque the top lid bolts. See Table 8.1.5 for torque requirements.
  - c. Inspect the lift cleat bolts for general condition. Replace worn or damaged bolts with new bolts.
  - d. Install the MPC lift cleats and MPC slings. See Figure 8.1.24 and 8.1.25. See Table 8.1.5 for torque requirements.
  - e. Drain and remove the Temporary Shield Ring, if used.
4. Replace the pool lid with the transfer lid as follows (Not required for HI-TRAC 100D and 125D):

**ALARA Note:**

The transfer slide is used to perform the bottom lid replacement and eliminate the possibility of directly exposing the bottom of the MPC. The transfer slide consists of the guide rails, rollers, transfer step and carriage. The transfer slide carriage and jacks are powered and operated by remote control. The carriage consists of short-stroke hydraulic jacks that raise the carriage to support the weight of the bottom lid. The transfer step produces a tight level seam between the transfer lid and the pool lid to minimize radiation streaming. The transfer slide jacks do not have sufficient lift capability to support the entire weight of the HI-TRAC. This was selected specifically to limit floor loads. Users should designate a specific area that has sufficient room and support for performing this operation.

**Note:**

The following steps are performed to pretension the MPC slings.

- a. Lower the lift yoke and attach the MPC slings to the lift yoke. See Figure 8.1.25.
- b. Raise the lift yoke and engage the lift yoke to the HI-TRAC lifting trunnions.
- c. If necessary, position the transfer step and transfer lid adjacent to one another on the transfer slide carriage. See Figure 8.1.26. See Figure 8.1.9 for transfer step rigging.
- d. Deleted.
- e. Position HI-TRAC with the pool lid centered over the transfer step approximately one inch above the transfer step.



- f. Raise the transfer slide carriage so the transfer step is supporting the pool lid bottom. Remove the bottom lid bolts and store them temporarily.

**ALARA Warning:**

Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

- g. Lower the transfer carriage and position the transfer lid under HI-TRAC.
- h. Raise the transfer slide carriage to place the transfer lid against the HI-TRAC bottom lid bolting flange.
- i. Inspect the transfer lid bolts for general condition. Replace worn or damaged bolts with new bolts.
- j. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.
- k. Raise and remove the HI-TRAC from the transfer slide.
- l. Disconnect the MPC slings and store them in an approved plant storage location.

**Warning:**

HI-TRAC dose rates are measured to ensure they are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

**Note:**

HI-STORM receipt inspection and preparation may be performed independent of procedural sequence.

5. Perform the HI-TRAC surface dose rate measurements in accordance with the technical specifications. These dose rate measurements fulfill the requirement for the operational shielding effectiveness testing described in Section 9.1.5.2. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10 CFR 72.104(a).
6. Perform a HI-STORM receipt inspection and cleanliness inspection in accordance with a site-approved inspection checklist, if required. See Figure 8.1.27 for HI-STORM lid rigging.

**Note:**

MPC transfer may be performed in the truck bay area, at the ISFSI, or any other location deemed appropriate by the licensee. The following steps describe the general transfer operations (See Figure 8.1.28). The HI-STORM may be positioned on an air pad, roller skid in the cask receiving area or at the ISFSI. The HI-STORM or HI-TRAC may be transferred to the ISFSI using a heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function (See Figure 8.1.29) as long as the HI-TRAC and HI-STORM lifting requirements are not exceeded (See technical specifications). The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

8.1.7 Placement of HI-STORM into Storage

1. Position an empty HI-STORM module at the designated MPC transfer location. The HI-STORM may be positioned on the ground, on a de-energized air pad, on a roller skid, on a flatbed trailer or other special device designed for such purposes. If necessary, remove the exit vent screens and gamma shield cross plates, temperature elements and the HI-STORM lid. See Figure 8.1.28 for some of the various MPC transfer options.
  - a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.
  - b. Transfer the HI-TRAC to the MPC transfer location.

**Note:**

For most efficient heat rejection, the HI-TRAC transfer cask is envisaged to be routinely handled in the vertical orientation. However, architectural constraints at a plant, such as a low roll-up door opening or a low hung overhead duct work, may require the cask to be tilted or even downended for a short duration during this transfer step (as described in the notes in Subsection 8.1.2). In such a case, the continued thermal compliance of the cask's contents to ISG-11 Rev 3 will be verified by simulating the short term handling operation on the NRC-reviewed Fluent model and any possible adverse effect on the occupational dose shall be mitigated by use of suitably configured custom shielding.

2. De-energize the air pad or chock the vehicle wheels to prevent movement of the HI-STORM during MPC transfer and to maintain level, as required.

**ALARA Note:**

The HI-STORM vent duct shield inserts eliminate the streaming path created when the MPC is transferred past the exit vent ducts. Vent duct shield inserts are not used with the HI-STORM 100S.

3. Install the alignment device (or mating device for HI-TRAC 100D and 125D) and if necessary, install the HI-STORM vent duct shield inserts. See Figure 8.1.30.

**Caution:**

MPCs loaded with high burn-up fuel and MPC heat load greater than the threshold heat load defined in Table 4.5.4 require operation of the supplemental cooling system to prevent fuel cladding temperatures in excess of ISG-11 Rev. 3 limits. (See Section 4.5) All preparatory work related to the transfer should be completed prior to terminating the supplemental cooling operations.

4. If used, discontinue the supplemental cooling operations and disconnect the SCS. Drain water from the HI-TRAC annulus to an appropriate plant discharge point.
5. Position HI-TRAC above HI-STORM. See Figure 8.1.28.
6. Align HI-TRAC over HI-STORM (See Figure 8.1.31) and mate the overpacks.
7. If necessary, attach the MPC Downloader. See Figure 8.1.32.
8. Attach the MPC slings to the MPC lift cleats.
9. Raise the MPC slightly to remove the weight of the MPC from the transfer lid doors (or pool lid for HI-TRAC 100D and 125D and mating device)
10. If using the HI-TRAC 100D or 125D, unbolt the pool lid from the HI-TRAC.
11. Remove the transfer lid door (or mating device drawer) locking pins and open the doors (or drawer).

**ALARA Warning:**

MPC trim plates are used to eliminate the streaming path above and below the doors (or drawer). If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

12. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
13. Lower the MPC into HI-STORM.
14. Disconnect the slings from the MPC lifting device and lower them onto the MPC lid.
15. Remove the trim plates (if used), and close the doors (or mating device drawer).

**ALARA Warning:**

Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed due to radiation streaming.

**Note:**

It may be necessary, due to site-specific circumstances, to move HI-STORM from under the empty HI-TRAC to install the HI-STORM lid, while inside the Part 50 facility. In these cases, users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

16. Remove HI-TRAC from on top of HI-STORM.

**ALARA Note:**

The user may elect to partially close the mating device drawer on the loaded HI-STORM as a means to provide additional shielding for the top of the MPC. In this case, the mating device drawer shall remain open an adequate amount to equal at least the total area of all outlet vents of the HI-STORM. If the mating device uses an alignment ring, it shall be verified that the alignment ring does not block the annulus gap between the MPC and the HI-STORM.

17. Remove the MPC lift cleats and MPC slings and install hole plugs in the empty MPC bolt holes. See Table 8.1.5 for torque requirements.
18. Place HI-STORM in storage as follows:
  - a. Remove the alignment device (mating device with HI-TRAC pool lid for HI-TRAC 100D and 125D) and vent duct shield inserts (if used). See Figure 8.1.30.
  - b. Inspect the HI-STORM lid studs and nuts or lid closure bolts for general condition. Replace worn or damaged components with new ones.
  - c. If used, inspect the HI-STORM 100A anchor components for general condition. Replace worn or damaged components with new ones.
  - d. Deleted.

**Warning:**

Unless the lift is single failure proof (or equivalent safety factor) for the HI-STORM Lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

**Note:**

Shims may be used on the HI-STORM 100 lid studs. If used, the shims shall be positioned to ensure a radial gap of less than 1/8 inch around each stud. The method of cask movement will determine the most effective sequence for vent screen, lid, temperature element, and vent gamma shield cross plate installation.

- e. Install the HI-STORM lid and the lid studs and nuts or lid closure bolts. See Table 8.1.5 for bolting requirements. Install the HI-STORM 100 lid stud shims if necessary. See Figure 8.1.27 for rigging.
- f. Install the HI-STORM exit vent gamma shield cross plates, temperature elements (if used) and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34a.
- g. Remove the HI-STORM lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location. See Table 8.1.5 for torque requirements.

**Warning:**

HI-STORM dose rates are measured to ensure they are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

- h. Perform the HI-STORM surface dose rate measurements in accordance with the technical specifications. These dose rate measurements fulfill the requirement for the operational shielding effectiveness testing described in Section 9.1.5.2. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10 CFR 72.104(a).
- i. Secure HI-STORM to the transporter device as necessary.

**Note:**

The site-specific transport route conditions must satisfy the requirements of the technical specification.

- 19. Perform a transport route walkdown to ensure that the cask transport conditions are met.
- 20. Transfer the HI-STORM to its designated storage location at the appropriate pitch. See Figure 8.1.35.

**Note:**

Any jacking system shall have the provisions to ensure uniform loading of all four jacks during the lifting operation.

- a. If air pads were used, insert the HI-STORM lifting jacks and raise HI-STORM. See Figure 8.1.36. Remove the air pad.
- b. Lower and remove the HI-STORM lifting jacks, if used.
- c. For HI-STORM 100A overpack (anchored), perform the following:
  - 1. Inspect the anchor stud receptacles and verify that they are clean and ready for receipt of the anchor hardware.
  - 2. Align the overpack over the anchor location.
  - 3. Lower the overpack to the ground while adjusting for alignment.
  - 4. Install the anchor connecting hardware (See Table 8.1.5 for torque requirements).
- 21. Install the HI-STORM inlet vent gamma shield cross plates and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34.
- 22. Perform an air temperature rise test as follows for the first HI-STORM 100 System placed in service:

**Note:**

The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize. The purpose of this test is to confirm the initial performance of the HI-STORM 100 ventilation system.

- a. Measure the inlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average inlet air (or surface screen) temperature.

- b. Measure the outlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average outlet air (or surface screen) temperature.
  - c. Determine the average air temperature rise by subtracting the results of the average inlet screen temperature from the average outlet screen temperature.
  - d. Report the results to the certificate holder.
23. Users who load to CoC Amendment 5 or later with a heat load greater than 20kW must perform the following annular air flow thermal test or cite a test report that was performed and prepared by another User under any amendment. This test is required to be performed once per CoC Condition 9.
- a. The annular air flow thermal test shall be conducted at least 7 days after the HI-STORM is loaded in order for the overpack to establish thermal equilibrium.
  - b. The User or other qualified engineer shall calculate and record the actual heat load of the fuel stored in the HI-STORM.
  - c. To minimize the effects on the annular air flow, the test shall be performed when the weather is relatively dry and calm. Additionally, the test can be performed indoors to help minimize effects on air flow.
  - d. The ambient air temperature at the cask shall be recorded
  - e. For users loading to CoC Amendment 5 through 9, Revision 1, test data shall be collected for the annular flow at each cask outlet vents as follows (alternatively users may credit a test performed at another site under any amendment):
    - 1. The outlet vent screen and gamma shield shall be removed from one outlet vent. For indoor tests, all four vent screens and gamma shields may be removed before the lid is placed on the HI-STORM. If possible, gamma shields may be left in place for the test.
    - 2. A hot wire anemometer or similar flow measuring instrument shall be inserted into the annular space between the MPC and HI-STORM inner shell.
    - 3. The flow measuring instrument shall be positioned at least 6" below the top of the MPC and shall not significantly block the air flow.
    - 4. The instrument shall not be placed too close to the MPC or HI-STORM shells to avoid edge effects on the flow.
    - 5. The outlet gamma shield (if removed) and vent screen shall be re-installed. For indoor tests, all four gamma shields (if removed) and vent screens may remain uninstalled for the test.
    - 6. Measurements of the air flow shall be taken and recorded for a minimum of three places radially across the annular gap. If all four gamma shields and vents were removed at the same time, this step shall be repeated for each of the remaining three outlets.

7. The outlet vent screen and gamma shield shall be removed from the outlet vent (if in place) and the flow measuring instrument removed.
  8. The outlet gamma shield and vent screen shall be re-installed. If all four gamma shields and vent screens were removed at the same time in step 1, all four shall be re-installed.
- f. For users loading to CoC Amendment 10 or later, test data shall be collected for the annular flow at each cask inlet vent as follows (alternatively users may credit a test performed at another site under any amendment):
1. Test apparatus consisting of ducts through which all inlet air flow is channeled shall be installed at each HI-STORM inlet vent. The inlet ducts shall be designed to avoid adding significant air flow resistance to the HI-STORM cask.
  2. A hot wire anemometer or similar flow measuring instrument shall be used to measure air flow velocities at discrete locations across a single plane in each test apparatus duct.
  3. The measured air flow velocities shall be used to calculate the total mass air flow into each HI-STORM inlet vent.
- g. All test data shall be transmitted to the General Licensee for evaluation and validation of the thermal model.
- h. Users shall forward test and analysis results to the NRC in accordance with 10 CFR 72.4.

Table 8.1.1  
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS  
125-TON HI-TRAC\*\*

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case <sup>†</sup> Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid) <sup>††</sup>	245,040	245,040	245,040						1
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963						1
Empty HI-STORM 100S (Short) overpack (without lid) <sup>††</sup>	275,000	275,000	275,000						1
Empty HI-STORM 100S (Tall) overpack (without lid) <sup>††</sup>	290,000	290,000	290,000						1
HI-STORM 100S lid (without rigging. Add 1,000 lbs for 100S Version B Lid)	28,000	28,000	28,000						1
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	41,280	55,040	49,640	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	4,200	4,200	4,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Mating Device	15,000	15,000	15,000						
Empty HI-TRAC 125 (without Top Lid, neutron shield jacket water, or bottom lids)	117,803	117,803	117,803	1	1	1			1
Empty HI-TRAC 125D (without Top Lid, neutron shield jacket water, or bottom lids)	122,400	122,400	122,400	1	1	1			1
HI-TRAC 125 Top Lid	2,745	2,745	2,745			1			1
HI-TRAC 125D Top Lid	2,645	2,645	2,645			1			1
Optional HI-TRAC Lid Spacer (weight lbs/in thickness)	400	400	400						
HI-TRAC 125/125D Pool Lid(with bolts)	11,900	11,900	11,900	1	1				
HI-TRAC Transfer Lid (with bolts) (125 Only)	23,437	23,437	23,437			1			1
HI-TRAC 125 Neutron Shield Jacket Water	8,281	8,281	8,281		1	1			1
HI-TRAC 125 D Neutron Shield Jacket Water	9,000	9,000	9,000		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480			1	1		1

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated.

FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.2 for a description of each load handling case.

†† Short refers to both 100S-232 and 100S Version B-218. Tall refers to both 100S-243 and 100S Version B-229. Weights are based on 200 lb/cf concrete. Add an additional 1955 lbs. for the HI-STORM 100A overpack.



**TABLE 8.1.2**  
**ESTIMATED HANDLING WEIGHTS**  
**125-TON HI-TRAC\*\***

**Caution:**

The maximum weight supported by the 125-Ton HI-TRAC lifting trunnions cannot exceed 250,000 lbs. Users must take actions to ensure that this limit is not exceeded.

**Note:**

The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine their specific handling weights based on the MPC contents and the expected handling modes.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank empty)	232,641	241,059	240,302
2	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank full)	240,992	249,340	248,583
3	Loaded HI-TRAC 125 During Movement through Hatchway	237,893	246,311	246,227
1A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank empty)	237,238	245,656	244,899
2A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank full)	246,238	254,656	253,899
3A	Loaded HI-TRAC 125D During Movement through Hatchway	231,572	239,990	239,906
4	MPC during transfer operations	81,427	89,595	89,761
5A	Loaded HI-STORM 100 in storage (See Second Note to Table 8.1.1)	349,950	358,368	358,284
5B	Loaded HI-STORM 100S (Short) in storage (See Second Note to Table 8.1.1)	383,947	392,365	392,281
5C	Loaded HI-STORM 100S (Tall) in storage (See Second Note to Table 8.1.1)	398,947	407,365	407,281
6	Loaded HI-TRAC and transfer frame during on site handling	240,393	248,811	248,727

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.3  
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS  
100-TON HI-TRAC\*\*

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case <sup>†</sup> Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid) <sup>††</sup>	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (Short) overpack (without lid) <sup>††</sup>	275,000	275,000	275,000					1	
Empty HI-STORM 100S (Tall) overpack (without lid) <sup>††</sup>	290,000	290,000	290,000					1	
HI-STORM 100S lid (without rigging, add 1,000 lbs for 100S Version B Lid)	28,000	28,000	28,000						
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	41,280	55,040	49,640	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,200	3,200	3,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Empty HI-TRAC 100 (without Top Lid, neutron shield jacket water, or bottom lids)	84,003	84,003	84,003	1	1	1			1
HI-TRAC 100 Top Lid	1,189	1,189	1,189			1			1
HI-TRAC 100 Pool Lid	7,863	7,863	7,863	1	1				
HI-TRAC Transfer Lid (HI-TRAC 100 only)	16,686	16,686	16,686			1			1
Empty HI-TRAC 100D (without Top Lid, neutron shield jacket water, or pool lid)	84,204	84,204	84,204	1	1	1			1
HI-TRAC 100D Top Lid	1,239	1,239	1,239			1			1
HI-TRAC 100D Pool Lid	7,955	7,955	7,955	1	1	1			
Mating Device (HI-TRAC 100D only)	15,000	15,000	15,000						
HI-TRAC 100 Neutron Shield Jacket Water	7,583	7,583	7,583		1	1			1
HI-TRAC 100D Neutron Shield Jacket Water	7,800	7,800	7,800		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480					1	1

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.4 for a description of each load handling case.

† † Short refers to both 100S-232 and 100S Version B-218. Tall refers to both 100S-243 and 100S Version B-229. Weights are based on 200 lb/cf concrete. Add an additional 1955 lbs. for the HI-STORM 100A overpack.

Table 8.1.4  
ESTIMATED HANDLING WEIGHTS  
100-TON HI-TRAC\*\*

**Caution:**

The maximum weight supported by the 100-Ton HI-TRAC lifting trunnions cannot exceed 200,000 lbs. Users must take actions to ensure that this limit is not exceeded.

**Note:**

The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly and therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine the handling weights based on the contents to be loaded and the expected mode of operations.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC 100 removal from spent fuel pool (neutron tank empty)	193,804	202,222	201,465
2	Loaded HI-TRAC 100 removal from spent fuel pool (neutron tank full)	201,387	209,805	209,048
3	Loaded HI-TRAC 100 During Movement through Hatchway	193,608	202,026	201,942
1A	Loaded HI-TRAC 100D removal from spent fuel pool (neutron tank empty)	194,097	202,515	201,758
2A	Loaded HI-TRAC 100D removal from spent fuel pool (neutron tank full)	201,897	210,315	209,558
3A	Loaded HI-TRAC 100D During Movement through Hatchway	183,345	193,763	193,679
4	MPC during transfer operations	81,427	89,845	89,761
5A	Loaded HI-STORM 100 in storage (See Second Note to Table 8.1.1)	349,950	358,368	358,284
5B	Loaded HI-STORM 100S (Short) in storage (See Second Note to Table 8.1.1)	383,947	392,365	392,281
5C	Loaded HI-STORM 100S (Tall) in storage (See Second Note to Table 8.1.1)	398,947	407,365	407,281
6	Loaded HI-TRAC 100 and transfer frame during on site handling	197,588	206,006	205,922

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.5  
HI-STORM 100 SYSTEM TORQUE REQUIREMENTS

<b>Fastener<sup>†</sup></b>	<b>Torque (ft-lbs)<sup>††</sup></b>	<b>Pattern<sup>†††</sup></b>
HI-TRAC Top Lid Bolts <sup>†</sup>	Hand tight	None
HI-TRAC Pool Lid Bolts (36 Bolt Lid) <sup>†</sup>	Wrench tight	Figure 8.1.37
HI-TRAC Pool Lid Bolts (16 Bolt Lid) <sup>†</sup>	Wrench tight	Figure 8.1.37
100-Ton HI-TRAC Transfer Lid Bolts <sup>†</sup>	Wrench tight	Figure 8.1.37
125-Ton HI-TRAC Transfer Lid Bolts <sup>†</sup>	Wrench tight	Figure 8.1.37
MPC Lift Cleats Stud Nuts <sup>†</sup>	Wrench tight	None
MPC Lift Hole Plugs <sup>†</sup>	Hand tight	None
Threaded Fuel Spacers	Hand Tight	None
HI-STORM Lid Nuts <sup>†</sup>	Hand tight	None
HI-STORM 100S Lid Nuts and Lid Closure Bolts <sup>†</sup> (Temporary and Permanent Lids, Including Version B)	Hand Tight	None
Door Locking Pins	Hand Tight + 1/8 to 1/2 turn	None
HI-STORM 100 Vent Screen/Temperature Element Screws	Hand Tight	None
HI-STORM 100A Anchor Studs	55- 65 ksi tension applied by bolt tensioner (no initial torque)	None

<sup>†</sup> Studs and nuts shall be cleaned and inspected for damage or excessive thread wear (replace if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

<sup>††</sup> Unless specifically specified, torques have a +/- 5% tolerance.

<sup>†††</sup> No detorquing pattern is needed.

Table 8.1.6  
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure <sup>†</sup>	Description
Air Pads/Rollers	Not Important To Safety	8.1.29	Used for HI-STORM or HI-TRAC cask positioning. May be used in conjunction with the cask transporter or other HI-STORM 100 or HI-TRAC lifting device.
Annulus Overpressure System	Not Important To Safety	8.1.14	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Annulus Shield	Not Important To Safety	8.1.13	A shield that is placed at the top of the HI-TRAC annulus to provide supplemental shielding to the operators performing cask loading and closure operations.
Automated Welding System	Not Important To Safety	8.1.2b	Used for remote field welding of the MPC.
AWS Baseplate Shield	Not Important To Safety	8.1.2b	Provides supplemental shielding to the operators during the cask closure operations.
Bottom Lid Transfer Slide (Not used with HI-TRAC 100D and 125D)	Not Important To Safety	8.1.26	Used to simultaneously replace the pool lid with the transfer lid under the suspended HI-TRAC and MPC. Used in conjunction with the bottom lid transfer step.
Cask Transporter	Not Important to Safety unless used for MPC transfers	8.1.29a and 8.1.29b	Used for handling of the HI-STORM 100 Overpack and/or the HI-TRAC Transfer Cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function. May also be used for MPC transfers (examples are shown in Figure 8.1.28).

<sup>†</sup> Figures are representative and may not depict all configurations for all users.

**Table 8.1.6**  
**HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION**  
**(Continued)**

<b>Equipment</b>	<b>Important To Safety Classification</b>	<b>Reference Figure<sup>†</sup></b>	<b>Description</b>
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	8.1.9	Used for rigging such components such as the HI-TRAC top lid, pool lid, MPC lid, transfer lid, AWS, HI-STORM Lid and auxiliary shielding and the empty MPC.
Helium Backfill System	Not Important To Safety	8.1.23	Used for controlled insertion of helium into the MPC for pressure testing, blowdown and placement into storage.
HI-STORM 100 Lifting Jacks	Not Important To Safety	8.1.36	Jack system used for lifting the HI-STORM overpack to provide clearance for inserting or removing a device for transportation. .
Alignment Device	Not Important To Safety	8.1.31	Guides HI-TRAC into place on top of HI-STORM for MPC transfers. (Not used for HI-TRAC 100D and 125D)
HI-STORM Lifting Devices	Determined site-specifically based on type, location, and height of lift being performed. Lifting devices shall be provided in accordance with ANSI N14.6.	Not shown.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM 100 for cask handling. Does not include the crane hook (or other primary lifting device) device.
HI-STORM Vent Duct Shield Inserts	Important to Safety Category C .	8.1.30	Used for prevention of radiation streaming from the HI-STORM 100 exit vents during MPC transfers to and from HI-STORM. Not used with the HI-STORM 100S.
HI-TRAC Lid Spacer	Spacer Ring is Not-Important-To-Safety, Studs or bolts are I Important to Safety Category B	Not Shown	Optional ancillary which is used during MPC transfer operations to increase the clearance between the top of the MPC and the underside of the HI-TRAC top lid. Longer threaded studs (or bolts), supplied with the lid spacer, replace the standard threaded studs (or bolts) supplied with the HI-TRAC. The HI-TRAC lid spacer may ONLY be used when the HI-TRAC is handled in the vertical orientation or if HI-TRAC transfer lid is NOT used. The height of the spacer shall be limited to ensure that the weights and C.G. heights in a loaded HI-TRAC with the spacer do not exceed the bounding values found in Section 3.2 of the FSAR.
HI-TRAC Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC handling shall be provided in accordance with ANSI N14.6.	8.1.3	Used for connecting the crane (or other primary lifting device) to the HI-TRAC for cask handling. Does not include the crane hook (or other primary lifting device).

<sup>†</sup> Figures are representative and may not depict all configurations for all users.

Table 8.1.6  
 HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION (Continued)

Equipment	Important To Safety Classification	Reference Figure <sup>†</sup>	Description
HI-TRAC transfer frame	Not Important To Safety	8.1.4	A steel frame used to support HI-TRAC during delivery, on-site movement and upending/downending operations.
Cask Primary Lifting Device (Cask Transfer Facility)	Important to Safety. Quality classification of subcomponents determined site-specifically.	8.1.28 and 8.1.32	Optional auxiliary (Non-Part 50) cask lifting device(s) used for cask upending and downending and HI-TRAC raising for positioning on top of HI-STORM to allow MPC transfer. The device may consist of a crane, lifting platform, gantry system or any other suitable device used for such purpose.
Inflatable Annulus Seal	Not Important To Safety	8.1.13	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System	Important to Safety Status determined by each licensee. MPC lid lifting portions of the Lid Retention System shall meet the requirements of ANSI N14.6.	8.1.15, 8.1.17	Optional. The Lid Retention System secures the MPC lid in place during cask handling operations between the pool and decontamination pad.
MPC Lift Cleats	Important To Safety – Category A. MPC Lift Cleats shall be provided in accordance with of ANSI N14.6.	8.1.24	MPC lift cleats consist of the cleats and attachment hardware. The cleats are supplied as solid steel components that contain no welds. The MPC lift cleats are used to secure the MPC inside HI-TRAC during bottom lid replacement and support the MPC during MPC transfer from HI-TRAC into HI-STORM and vice versa. The ITS classification of the lifting device attached to the cleats may be lower than the cleat itself, as determined site-specifically.
Pressure Test System	Not Important to Safety	8.1.20	Used to pressure test the MPC lid-to-shell weld.
MPC Downloader	Important To Safety status determined site-specifically. MPC Downloader Shall meet the requirements of CoC, Appendix B, Section 3.5 for aboveground systems and Appendix A-100U, Section 5.5 for underground systems.	8.1.28 and 8.1.32	A lifting device used to help raise and lower the MPC during MPC transfer operations to limit the lift force of the MPC against the top lid of HI-TRAC. The MPC downloader may take several forms depending on the location of MPC transfer and may be used in conjunction with other lifting devices.
Deleted			
Deleted			

<sup>†</sup> Figures are representative and may not depict all configurations for all users.

Table 8.1.6  
 HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION (Continued)

Equipment	Important To Safety Classification	Reference Figure <sup>†</sup>	Description
Mating Device	Important-To-Safety – Category B	8.1.31	Used to mate HI-TRAC 100D and 125D to HI-STORM during transfer operations. Includes sliding drawer for use in removing HI-TRAC pool lid.
MPC Support Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	8.1.25	Used to secure the MPC to the lift yoke during HI-TRAC bottom lid replacement operations. Attaches between the MPC lift cleats and the lift yoke. Can be configured for different crane hook configuration.
MPC Upending Frame	Not Important to Safety	8.1.6	A steel frame used to evenly support the MPC during upending operations. and control the upending process.
Supplemental Cooling System	Important to Safety – Category B	2.C.1	A system used to circulate water or other coolant through the HI-TRAC annulus in order to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits during operations with the MPC in the HI-TRAC. Required only for MPC containing high burn-up fuel with a sufficiently high decay heat load as determined in accordance with Section 4.5. Calibration of the temperature instruments used to demonstrate heat removal from the HI-TRAC shall be performed in accordance with the requirements for Important to Safety Category B, the remaining components in the system are NITS.
MSLD (Helium Leakage Detector)	Not Important to Safety	Not shown	Used for helium leakage testing of the vent/drain port cover plate welds.
Deleted			
Temporary Shield Ring	Not Important To Safety	8.1.18	A water-filled tank that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.
Vacuum Drying (Moisture Removal) System	Not Important To Safety	8.1.22a	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Important to Safety – Category B	8.1.22b	Used for removal of residual moisture from the MPC following water draining. Calibration of the instrumentation used to confirm Tech Spec compliance shall be performed in accordance with the requirements for Important to Safety Category B, the remaining components of the system are NITS.
Vent and Drain RVOAs	Not Important To Safety	8.1.16	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Deleted			
Weld Removal System	Not Important To Safety	8.3.2b	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

<sup>†</sup> Figures are representative and may not depict all configurations for all users.



Table 8.1.7  
 HI-STORM 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND  
 UNLOADING OPERATIONS†

<b>Instrument</b>	<b>Function</b>
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors fluid flow rate during various loading and unloading operations.
Helium Mass Spectrometer Leakage Detector (MSLD)	Ensures leakage rates of welds are within acceptable limits.
Deleted	
Deleted	
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Deleted	
Temperature Surface Pyrometer	For HI-STORM vent operability testing.
Vacuum Gages	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Deleted	
Deleted	
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

† All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 8.1.8  
HI-STORM 100 SYSTEM OVERPACK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100 overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100 Overpack Lid:

1. Lid studs and nuts or lid closure bolts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device/ holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.

HI-STORM 100 Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be available, intact, and free of holes and tears in the fabric.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.
7. Anchor hardware, if used, shall be checked for general condition.

Table 8.1.9  
MPC INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:

1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Vent and Drain attachments shall be inspected for availability, thread condition operability and general condition.
4. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
5. Lower fuel spacers (if used) shall be inspected for availability and general condition.
6. Drain and vent port cover plates shall be inspected for availability and general condition.
7. Serial numbers shall be inspected for readability.
8. The MPC lid lift holes shall be inspected for thread condition.

MPC Main Body:

1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.

Table 8.1.10  
HI-TRAC TRANSFER CASK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-TRAC Top Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All Top Lid surfaces shall be relatively free of dents, scratches, gouges or other damage.

HI-TRAC Main Body:

1. The painted surfaces shall be inspected for corrosion, chipped, cracked or blistered paint.
2. The Top Lid bolt holes shall be inspected for dirt, debris and thread damage.
3. The Top Lid lift holes shall be inspected for thread condition.
4. Lifting trunnions shall be inspected for deformation, cracks, damage, corrosion, excessive galling, and, if applicable, damage to the locking plate and end plate, and presence or availability of locking plate and end plate retention bolts.
5. Pocket trunnion, if used, recesses shall be inspected for indications of overstressing (i.e., cracks, deformation, and excessive wear).
6. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
7. The nameplate shall be inspected for presence and general condition.
8. The neutron shield jacket shall be inspected for leaks.
9. Neutron shield jacket pressure relief valve shall be inspected for presence, and general condition.
10. The neutron shield jacket fill and drain plugs shall be inspected for presence, leaks, and general condition.
11. Bottom lid flange surface shall be clean and free of large scratches and gouges.

Table 8.1.10 (Continued)  
HI-TRAC OVERPACK INSPECTION CHECKLIST

HI-TRAC Transfer Lid (Not used with HI-TRAC 100D and 125D):

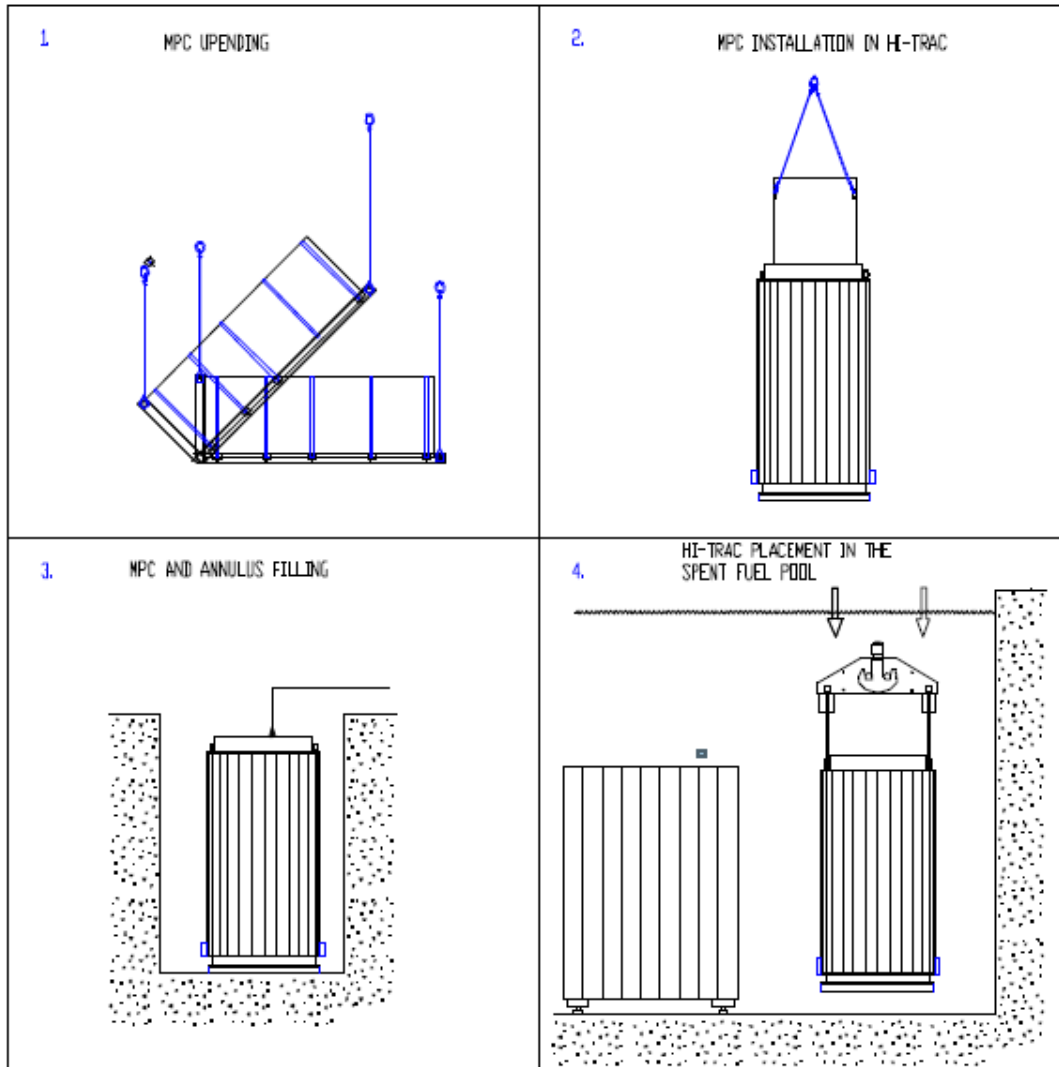
1. The doors shall be inspected for smooth actuation.
2. The threads shall be inspected for general condition.
3. The bolts shall be inspected for indications of overstressing (i.e., cracks, deformation, thread damage, excessive wear) and replaced as necessary.
4. Door locking pins shall be inspected for indications of overstressing (i.e., cracks, and deformation, thread damage, excessive wear) and replaced as necessary.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Lifting holes shall be inspected for thread damage.

HI-TRAC Pool Lid:

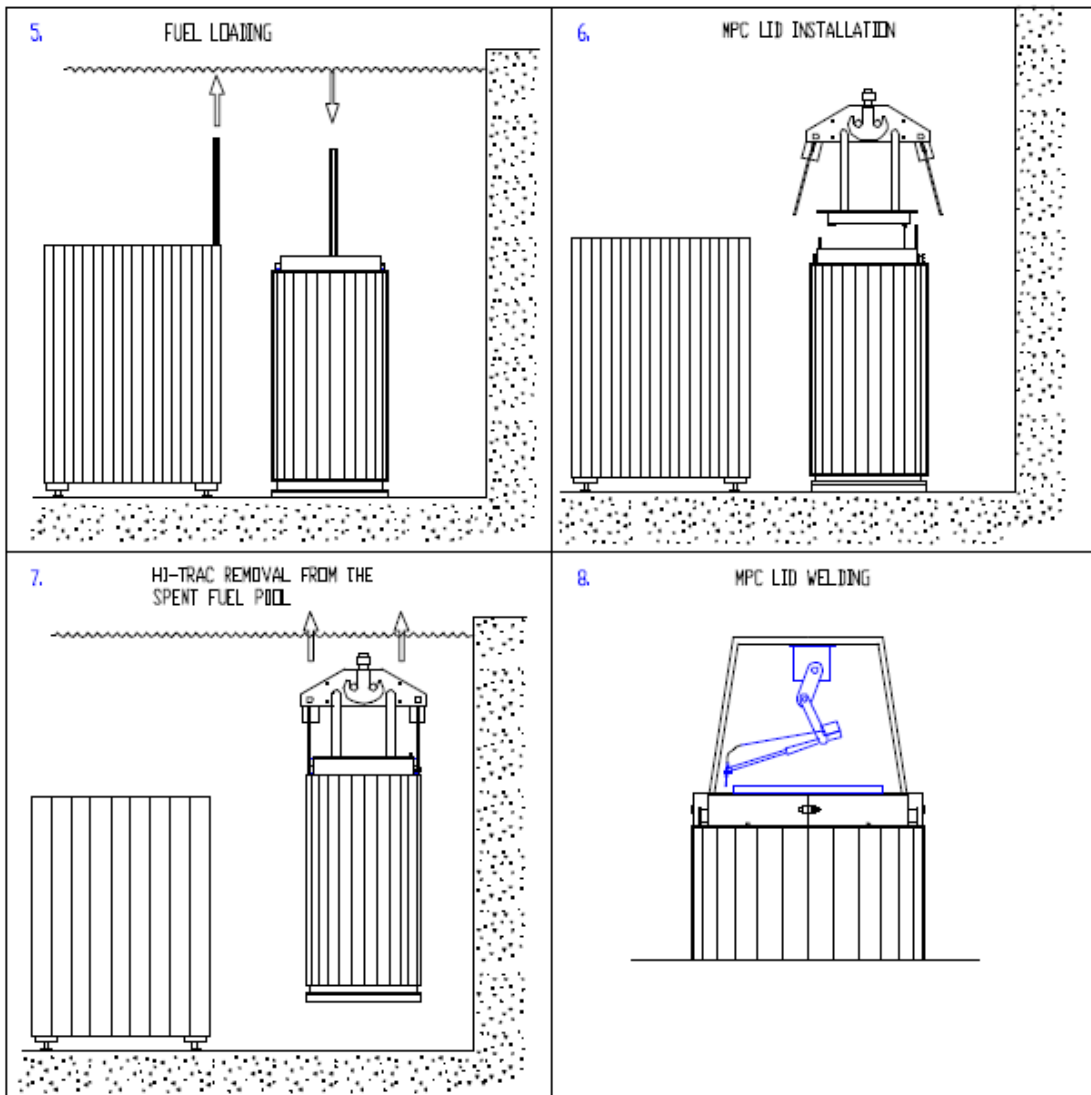
1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Threads shall be inspected for indications of damage.

<b>LOCATION: CASK RECEIVING AREA</b>	<b>BACKFILL MPC</b>
REMOVE HI-TRAC TOP LID	WELD VENT AND DRAIN PORT COVER PLATES & PERFORM NDE
CONFIGURE HI-TRAC WITH POOL LID	PERFORM LEAKAGE TEST ON CLOSURE PLATES
INSTALL MPC IN HI-TRAC	WELD MPC CLOSURE RING & PERFORM NDE
INSTALL UPPER FUEL SPACERS	DRAIN ANNULUS
INSTALL LOWER FUEL SPACERS	INSTALL HI-TRAC TOP LID
FILL MPC AND ANNULUS	PERFORM SURVEYS ON HI-TRAC
INSTALL ANNULUS SEAL	REMOVE TEMPORARY SHIELD RING
PLACE HI-TRAC IN SPENT FUEL POOL	REPLACE POOL LID WITH TRANSFER LID (Not Required for HI-TRAC 100D and 125D)
<b>LOCATION: SPENT FUEL POOL</b>	INSTALL MPC LIFT CLEAT
LOAD FUEL ASSEMBLIES INTO MPC	PERFORM SURVEYS OF POOL LID
PERFORM ASSEMBLY IDENTIFICATION VERIFICATION	PERFORM HI-TRAC SURFACE DOSE RATE MEASUREMENTS (PER TS)
INSTALL DRAIN LINE TO MPC LID	<b>LOCATION: CASK RECEIVING AREA</b>
ALIGN MPC LID AND LIFT YOKE TO DRAIN LINE	POSITION HI-STORM FOR MPC TRANSFER
INSTALL MPC LID	INSTALL ALIGNMENT DEVICE (MATING DEVICE FOR HI-TRAC 100D and 125D) & VENT DUCT SHIELD INSERTS
REMOVE HI-TRAC FROM SPENT FUEL POOL AND PLACE IN PREPARATION AREA	MATE OVERPACK AND TRANSFER CASK
<b>LOCATION: CASK PREPARATION AREA</b>	ATTACH MPC SLINGS
DECONTAMINATE HI-TRAC BOTTOM	OPEN TRANSFER LID DOORS (Remove pool lid and open mating device drawer for HI-TRAC 100D and 125D)
SET HI-TRAC IN CASK PREPARATION AREA	LOWER MPC INTO HI-STORM
FILL NEUTRON WATER JACKET	REMOVE HI-TRAC FROM ON TOP OF HI-STORM
MEASURE DOSE RATES AT MPC LID	REMOVE MPC LIFT CLEAT
DECONTAMINATE HI-TRAC AND LIFT YOKE	REMOVE ALIGNMENT DEVICE (Mating device for HI-TRAC 100D and 125D) & VENT DUCT SHIELD INSERTS
INSTALL TEMPORARY SHIELD RING	INSTALL HI-STORM LID
REMOVE INFLATABLE ANNULUS SEAL	PERFORM HI-STORM SURFACE DOSE RATE MEASUREMENTS (PER TS)
LOWER ANNULUS WATER LEVEL SLIGHTLY	<b>LOCATION: ISFSI</b>
SMEAR MPC LID TOP SURFACES	PLACE HI-STORM IN STORAGE
INSTALL ANNULUS SHIELD	INSTALL GAMMA SHIELD CROSS PLATES AND THERMOCOUPLES IN HI-STORM 100 OVERPACK EXIT VENTS
LOWER MPC WATER LEVEL	INSTALL HI-STORM VENT SCREENS
WELD MPC LID & PERFORM NDE	PERFORM THERMAL TESTING
DELETED	
RAISE MPC WATER LEVEL	
PERFORM MPC LID-TO-SHELL WELD PRESSURE TEST	
DELETED	
DRAIN MPC	
MEASURE VOLUME OF WATER DRAINED	
DRY MPC	
PERFORM MPC DRYNESS VERIFICATION TEST	

Figure 8.1.1: Loading Operations Flow Diagram

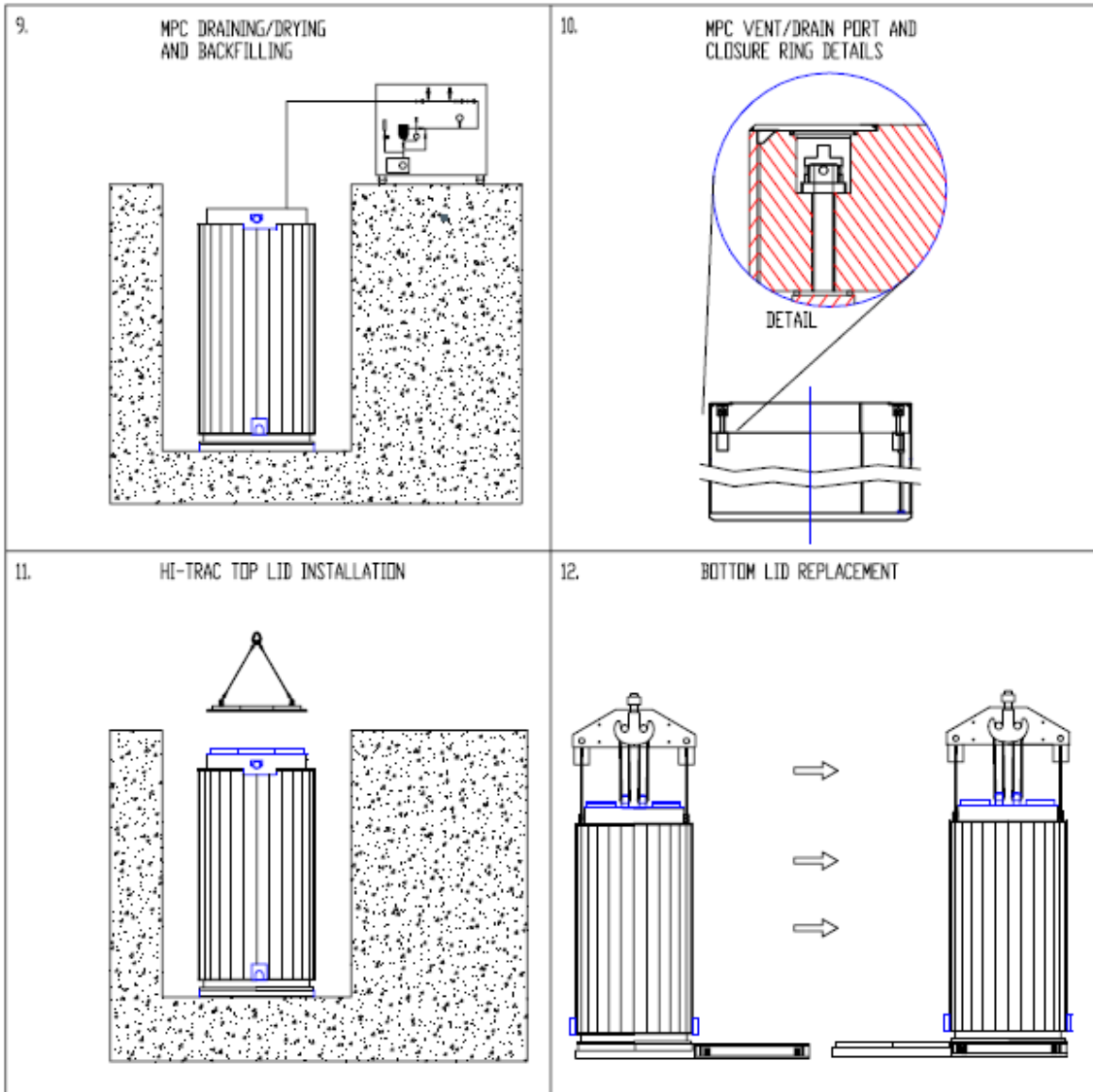


**Figure 8.1.2a: Major HI-STORM 100 Loading Operations**



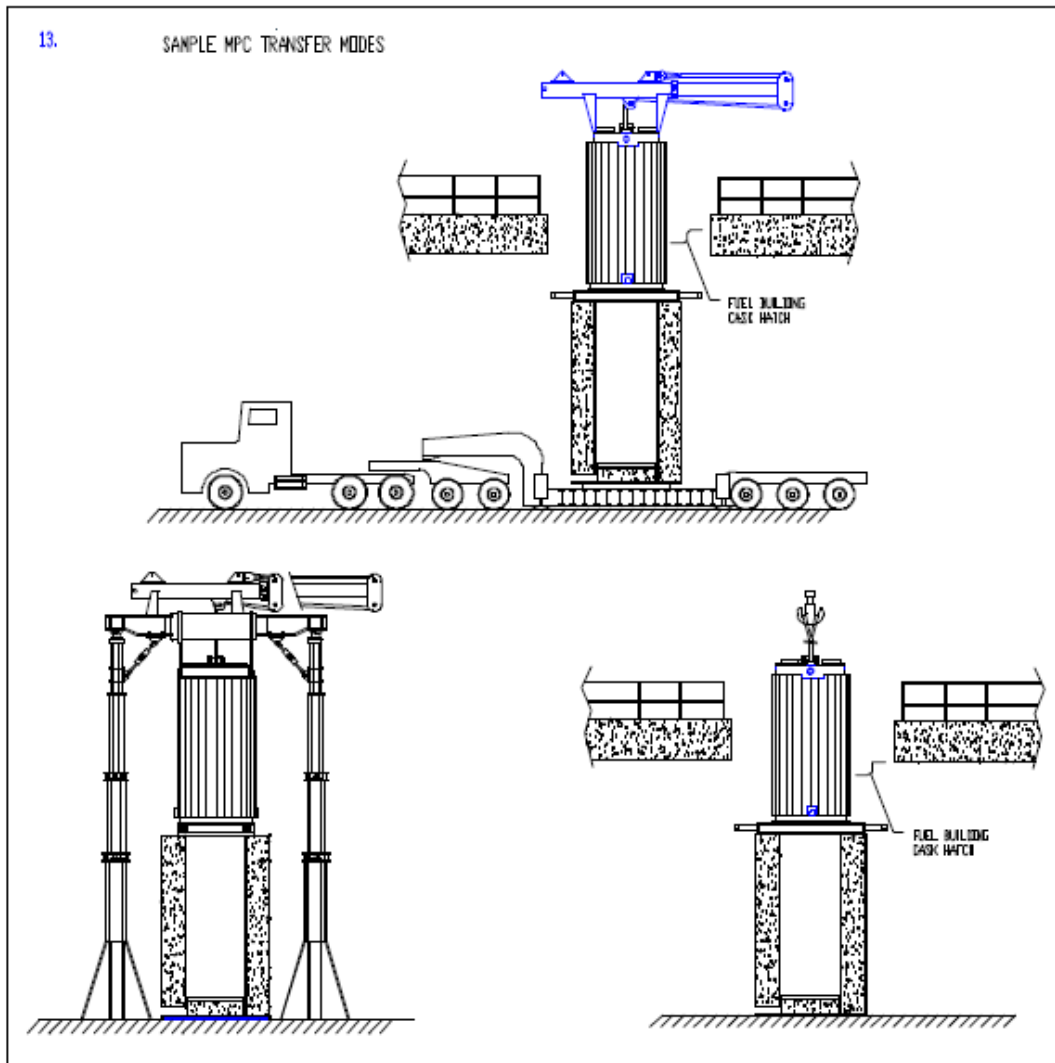
**Figure 8.1.2b: Major HI-STORM 100 Loading Operations**



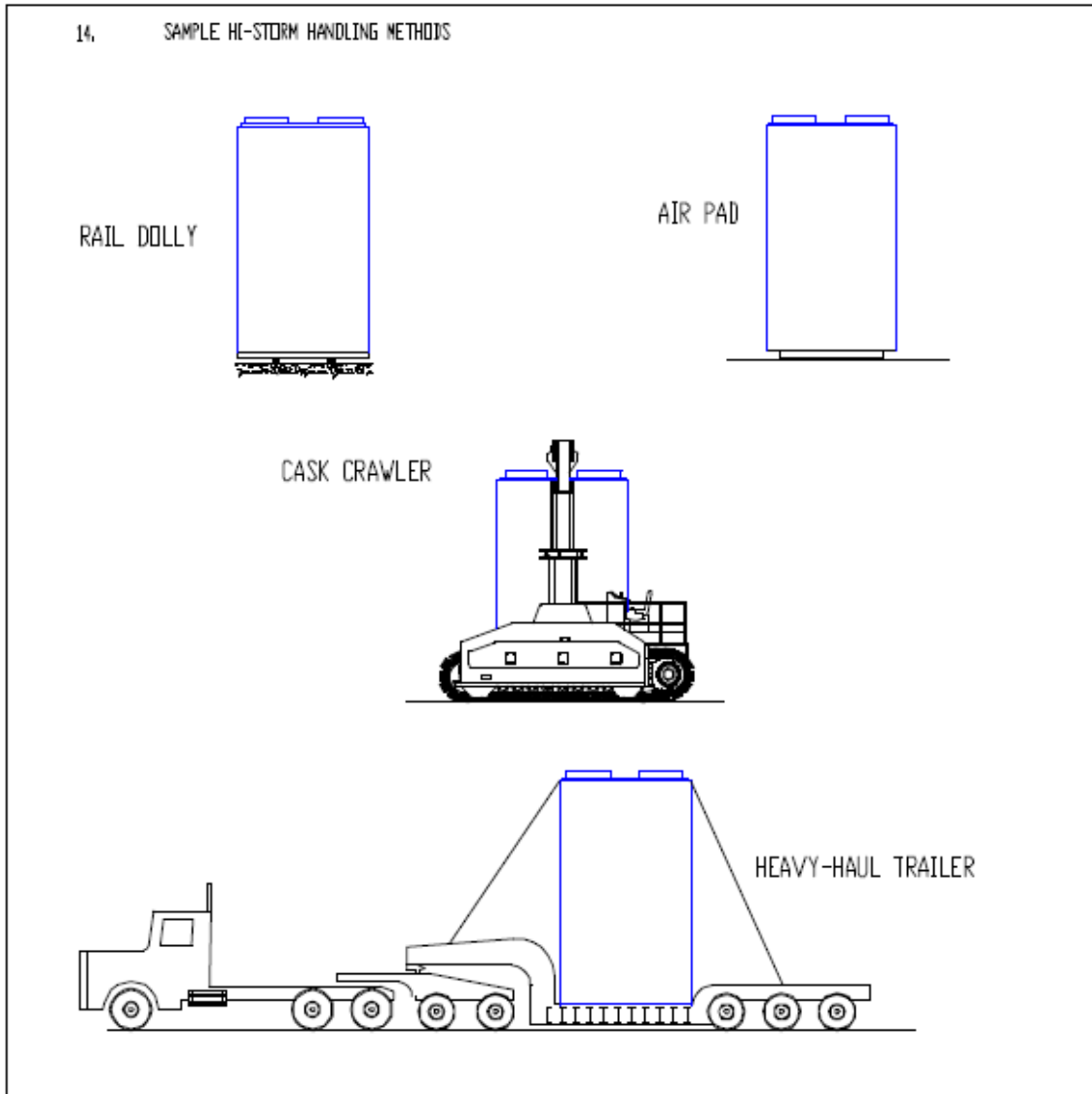


**Note: Bottom Lid Replacement is not required for HI-TRAC 100D and 125D**

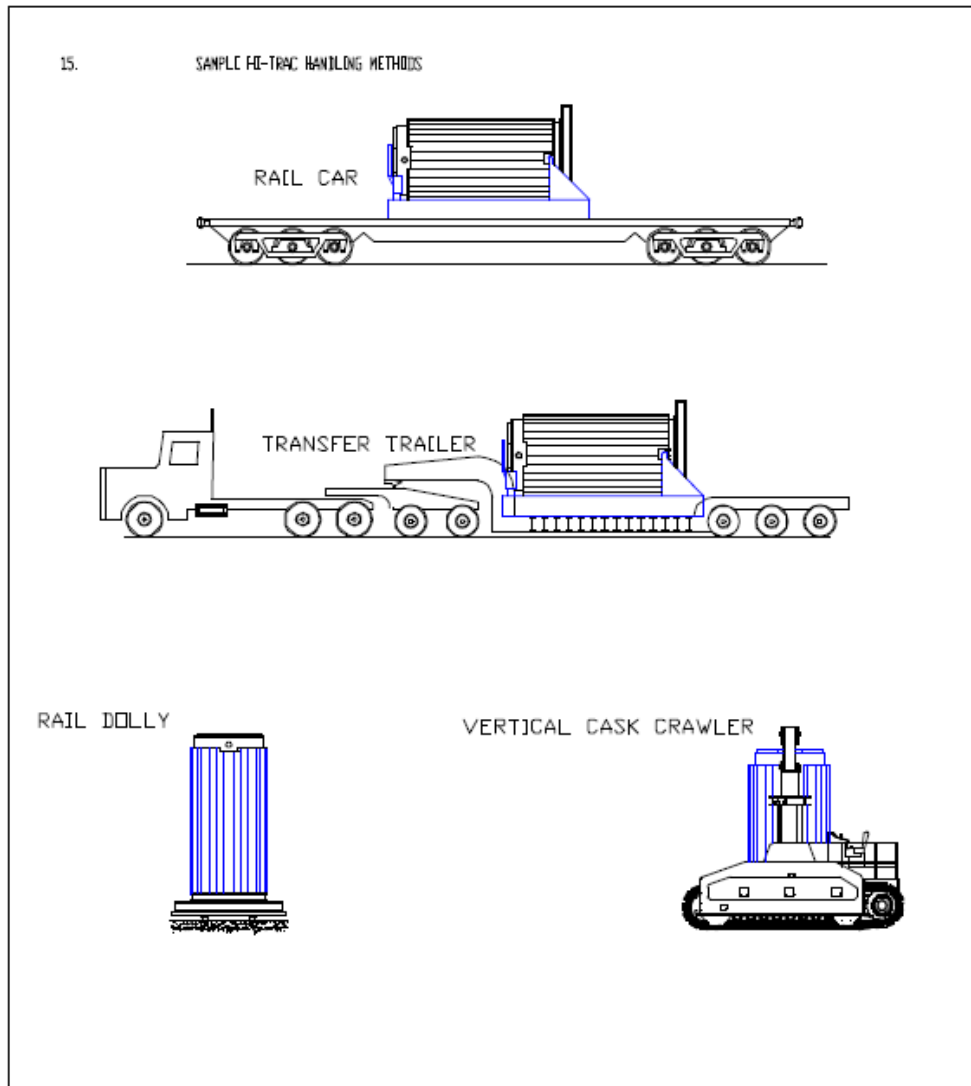
**Figure 8.1.2c: Major HI-STORM 100 Loading Operations**



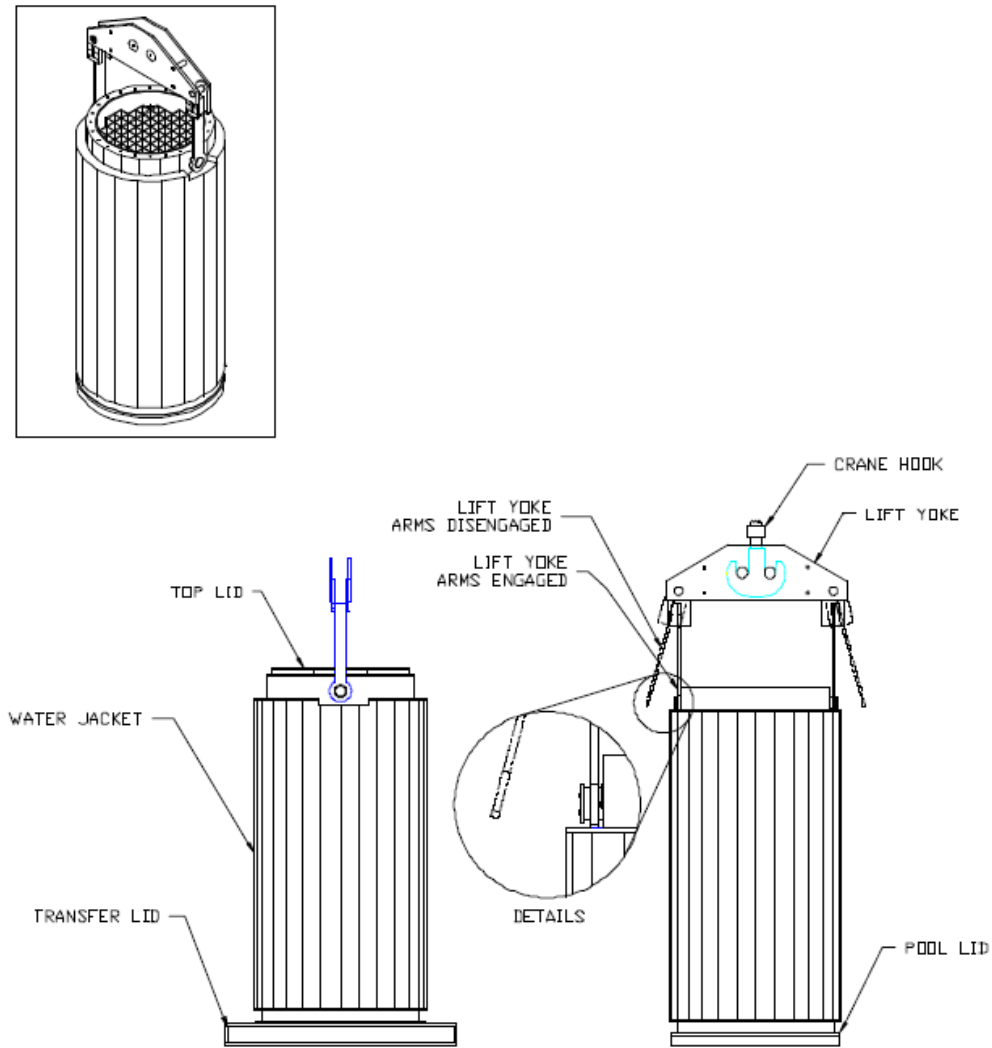
**Figure 8.1.2d: Major HI-STORM 100 Loading Operations (HI-TRAC with Transfer Lid Shown)**



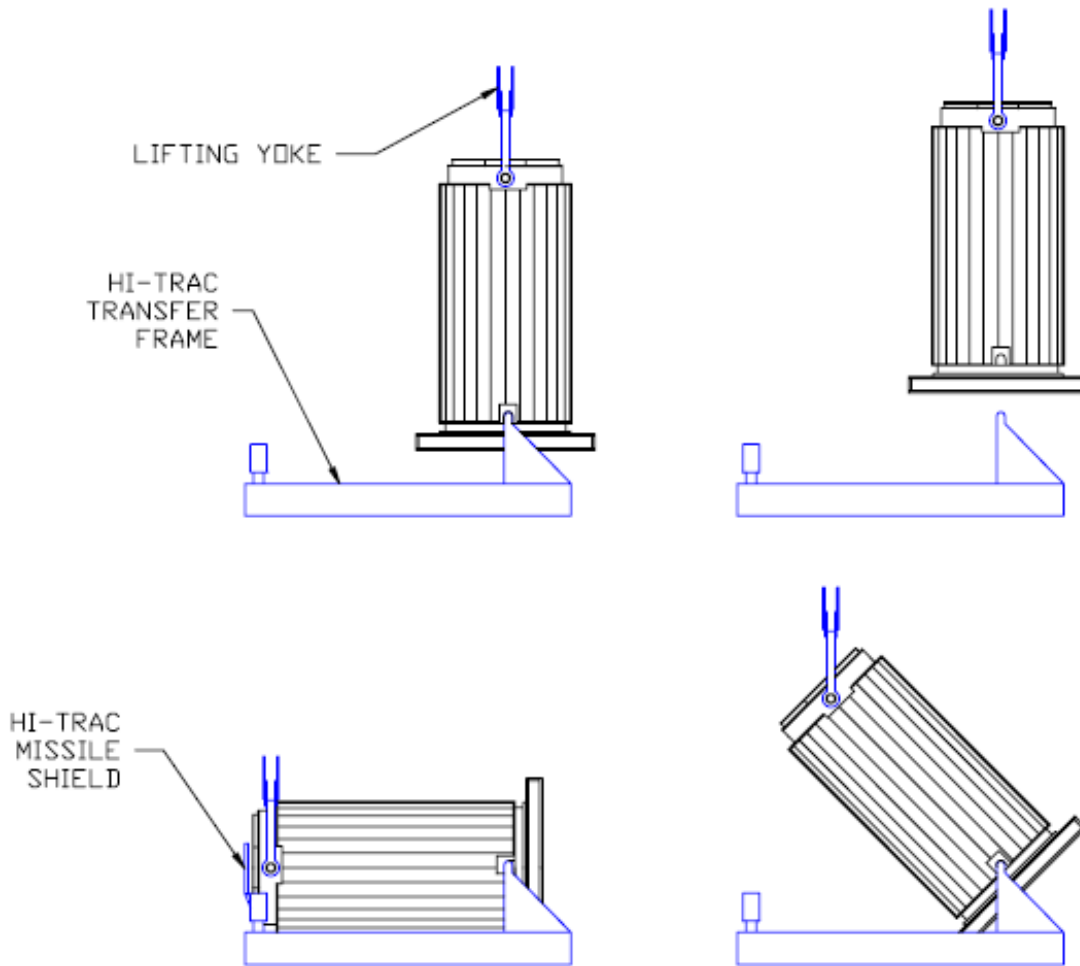
**Figure 8.1.2e: Example of HI-STORM 100 Handling Options**



**Figure 8.1.2f: Example of HI-TRAC Handling Options (Missile Shields Not Shown For Clarity)**

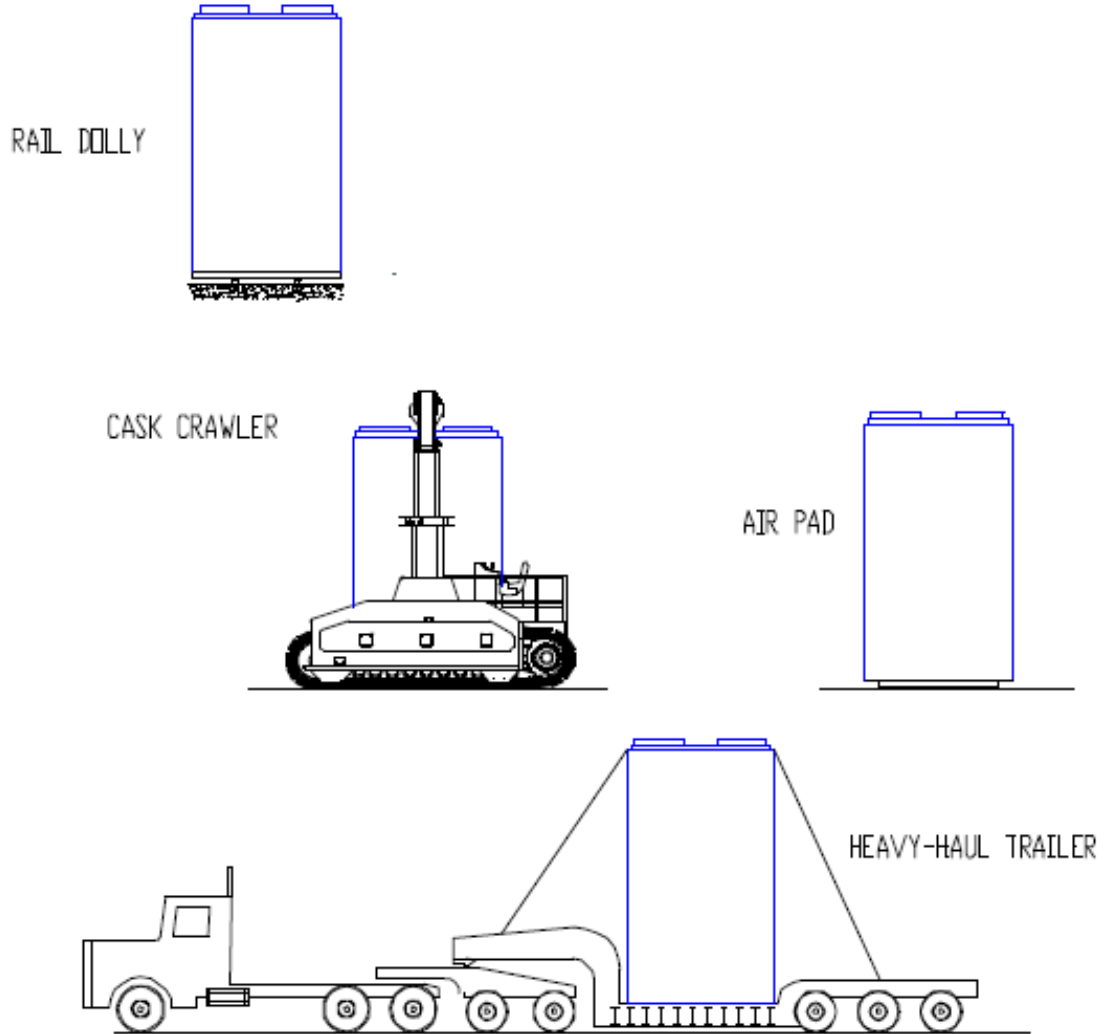


**Figure 8.1.3: Lift Yoke Engagement and Vertical HI-TRAAC Handling  
(Shown with the Pool Lid and the Transfer Lid)**

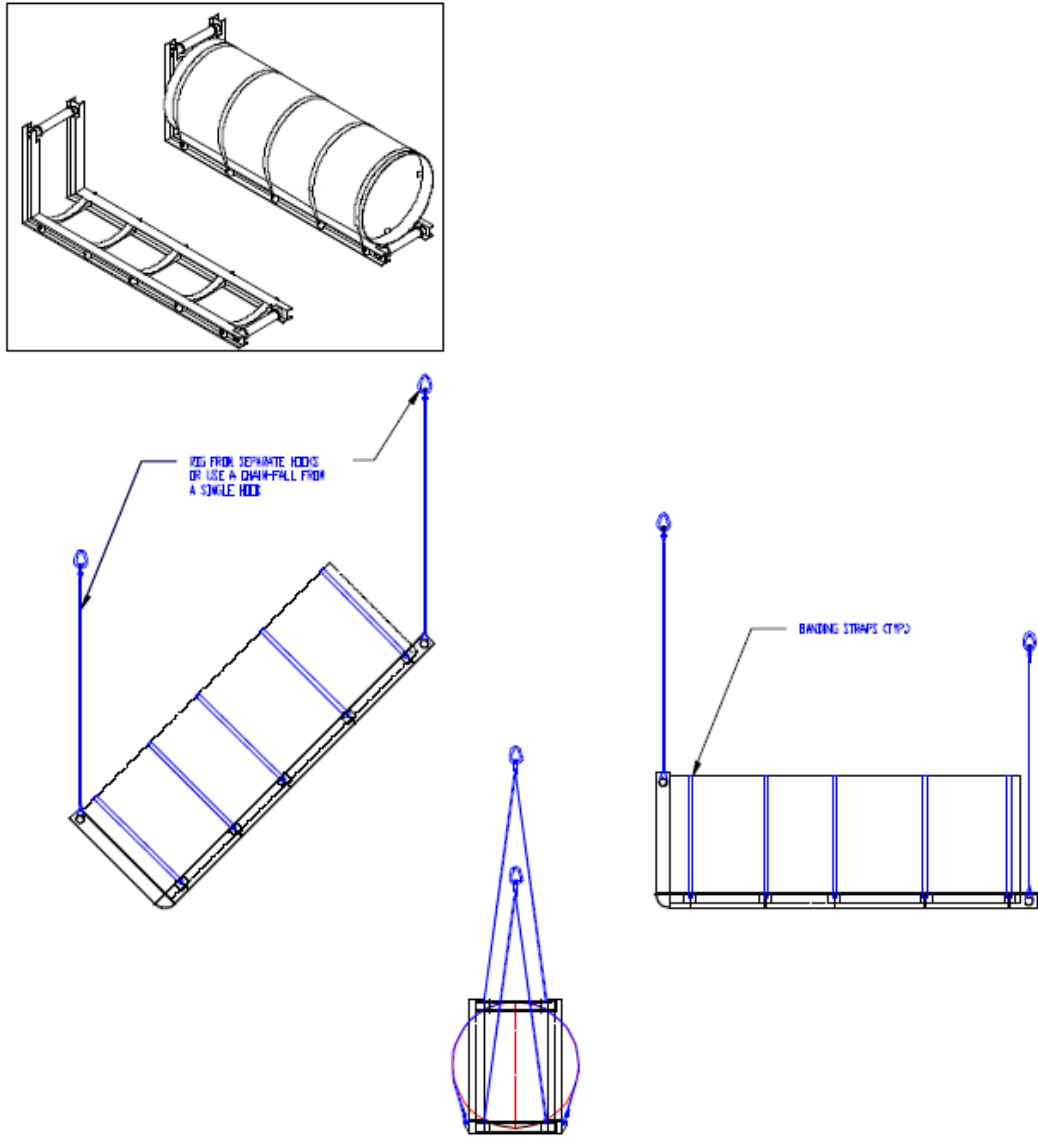


**Figure 8.1.4: HI-TRAC Upending/Downending in the Transfer Frame**

**(HI-TRAC with pocket trunnions shown, HI-TRAC 100D and 125D utilize separate upending frame without pocket trunnions)**

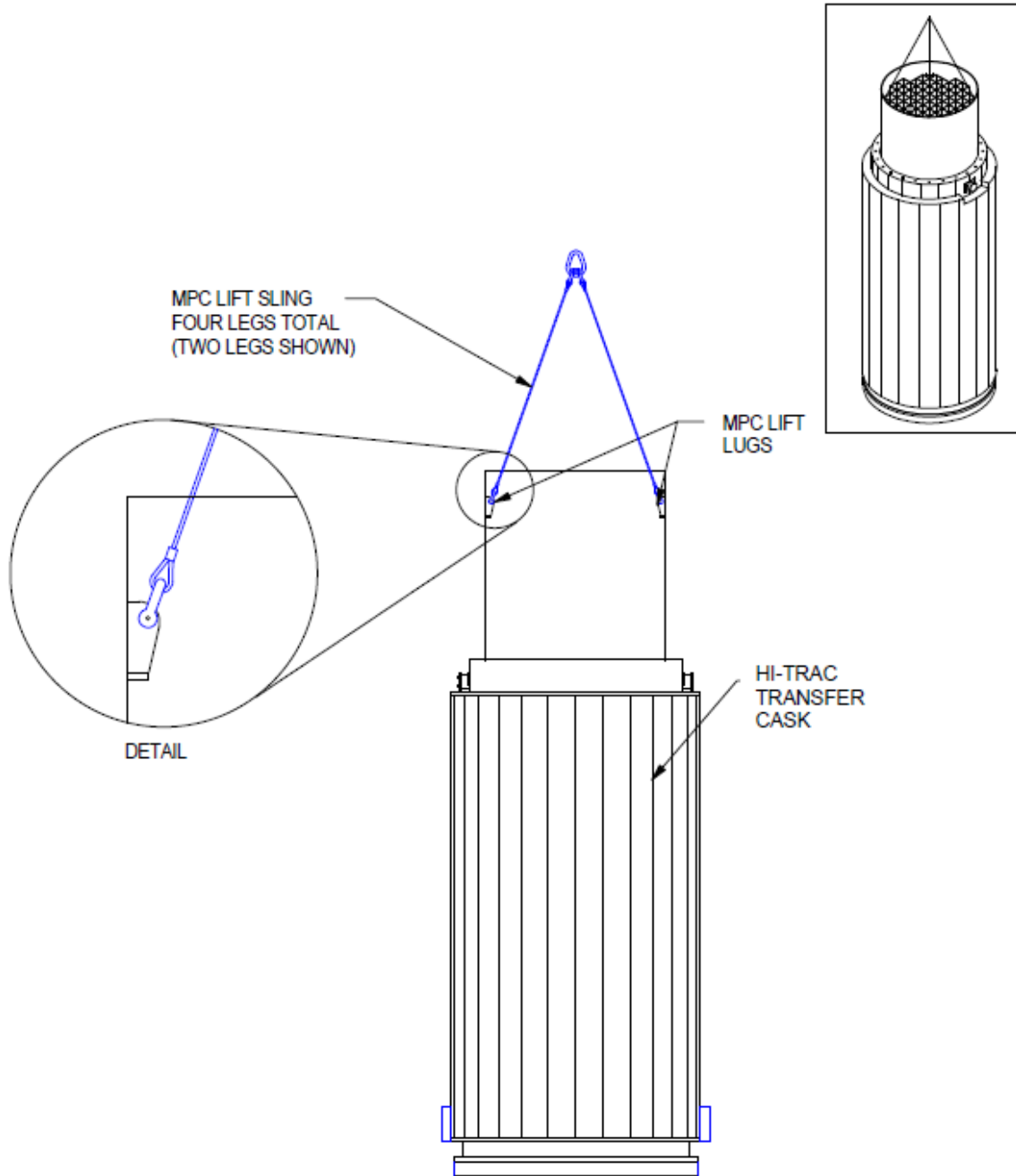


**Figure 8.1.5: HI-STORM Vertical Handling**

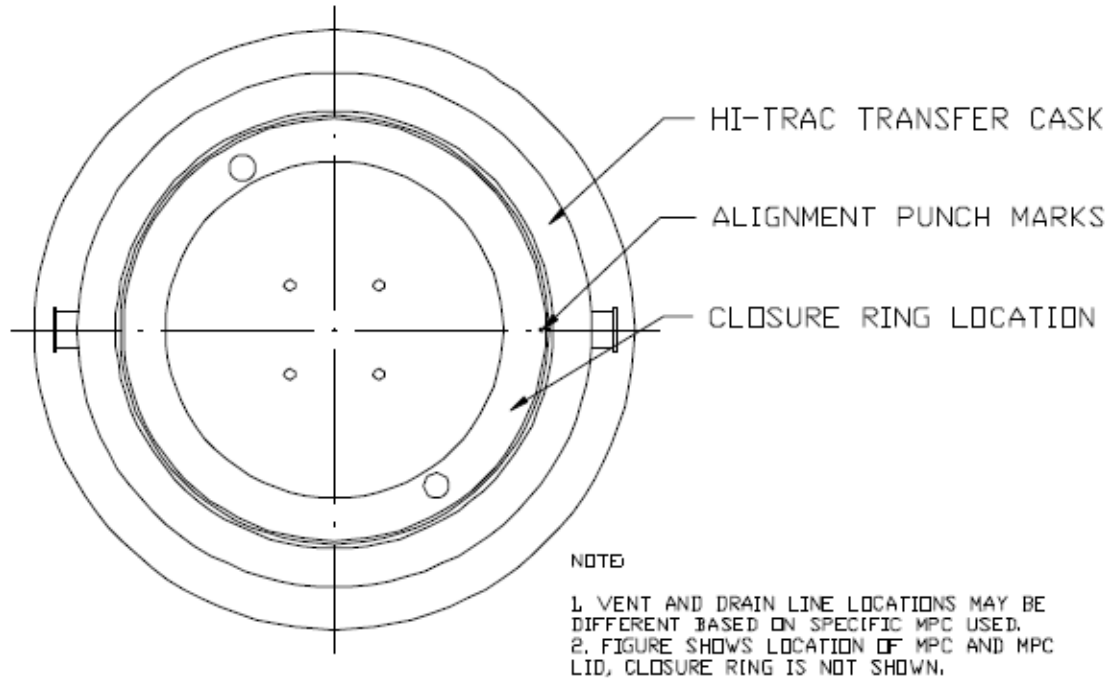


**Figure 8.1.6: MPC Upending in the MPC Upending Frame**

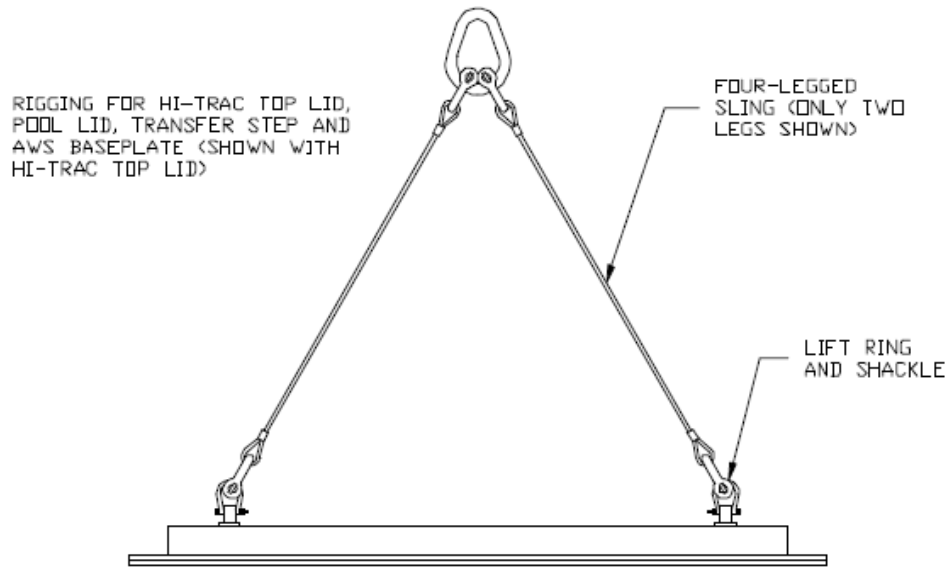




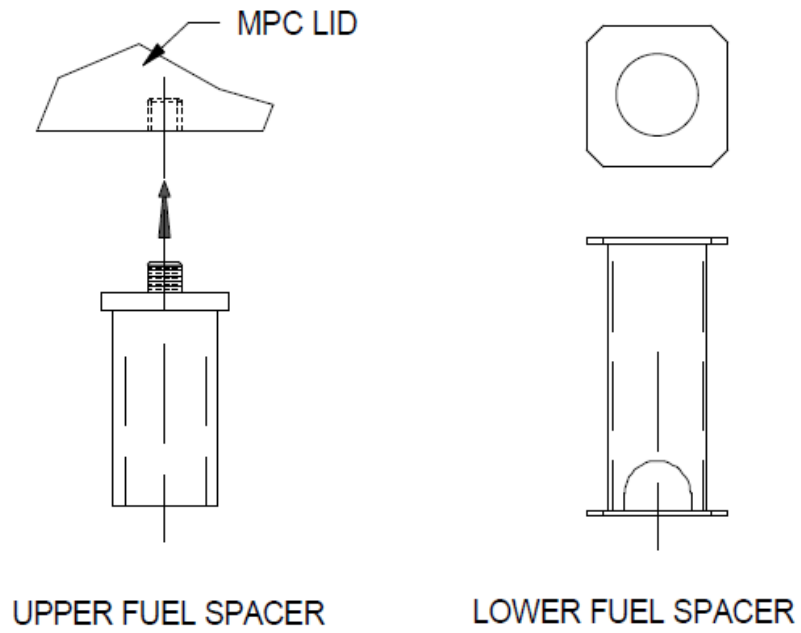
**Figure 8.1.7: MPC Rigging for Vertical Lifts**



**Figure 8.1.8: MPC Alignment in HI-TRAC**

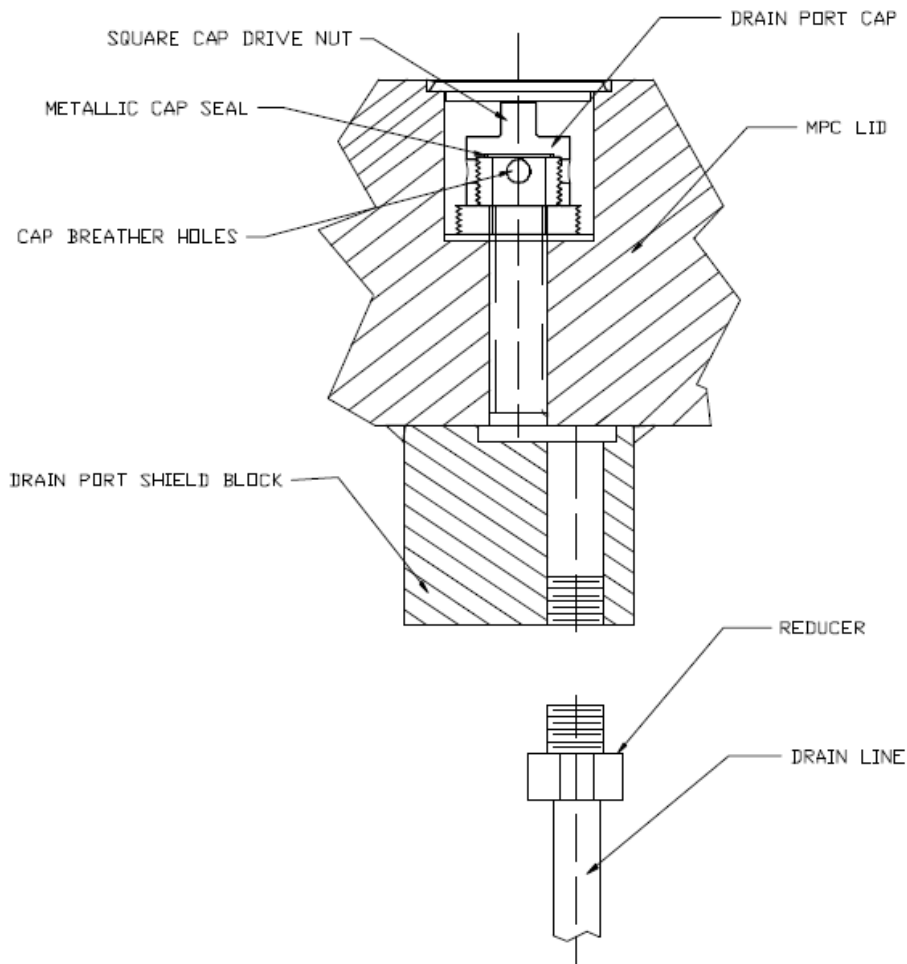


**Figure 8.1.9: MPC Lid AND HI-TRAC Accessory Rigging**



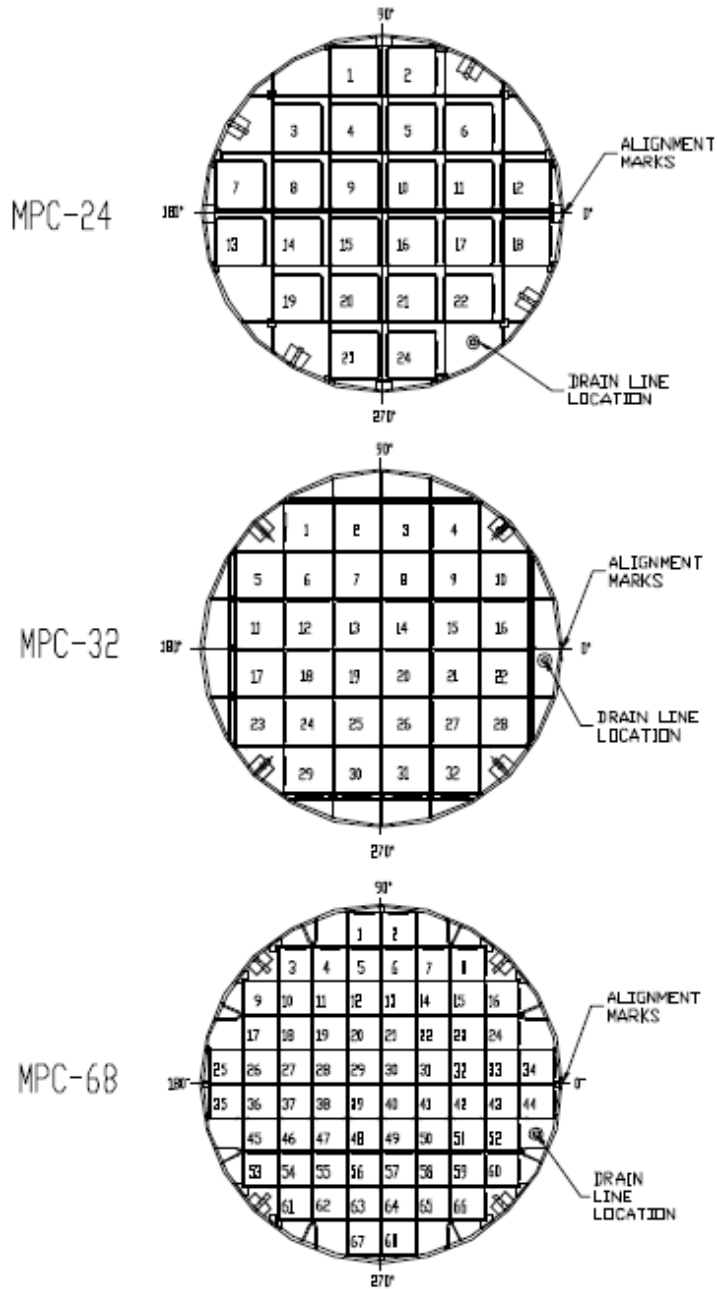
Note: Lengths are based on specific fuel assembly type to be stored.

**Figure 8.1.10: Fuel Spacers**

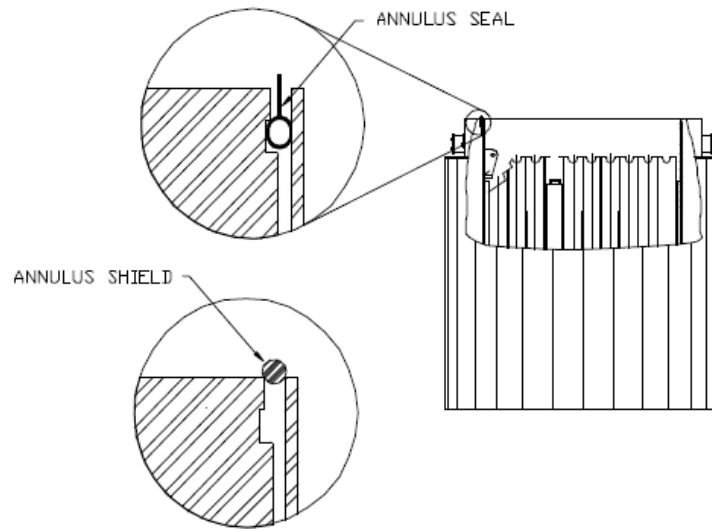


**Figure 8.1.11: Drain Port Details**

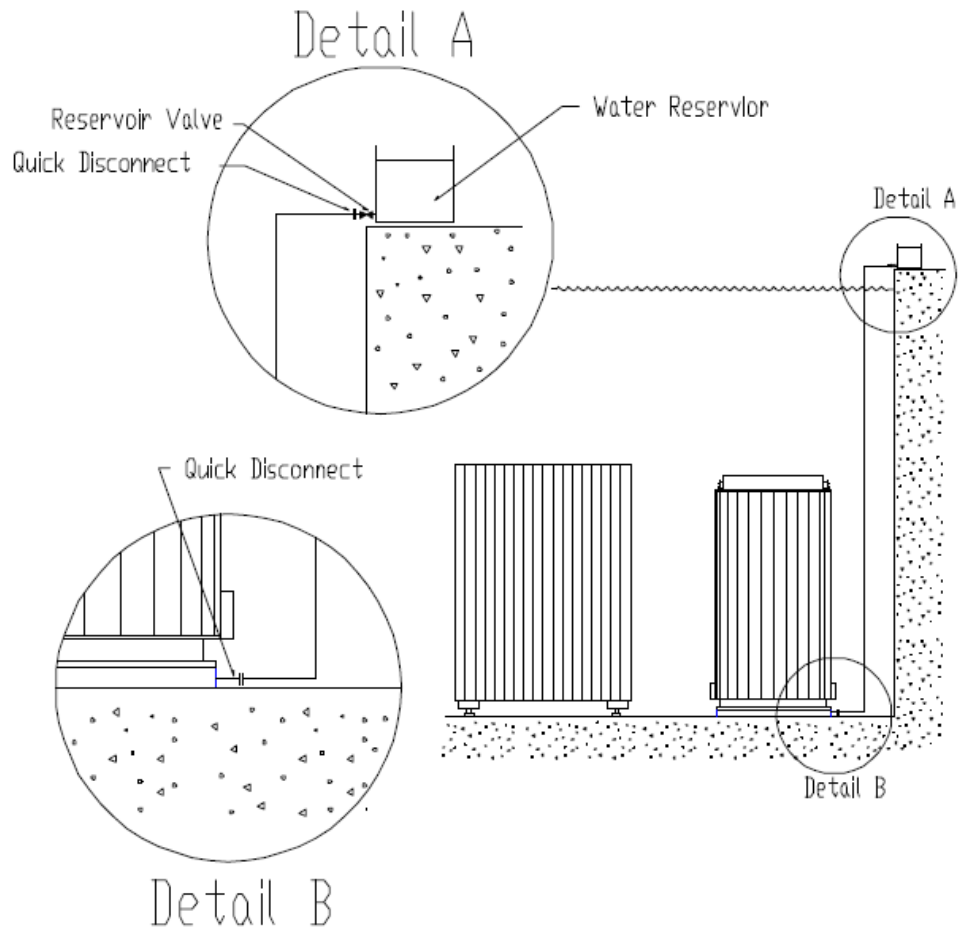
(Note – The Vent/Drain Connector may also have the optional configuration shown in the drawings in Section 1.5)



**Figure 8.1.12: Drain Line Positioning**

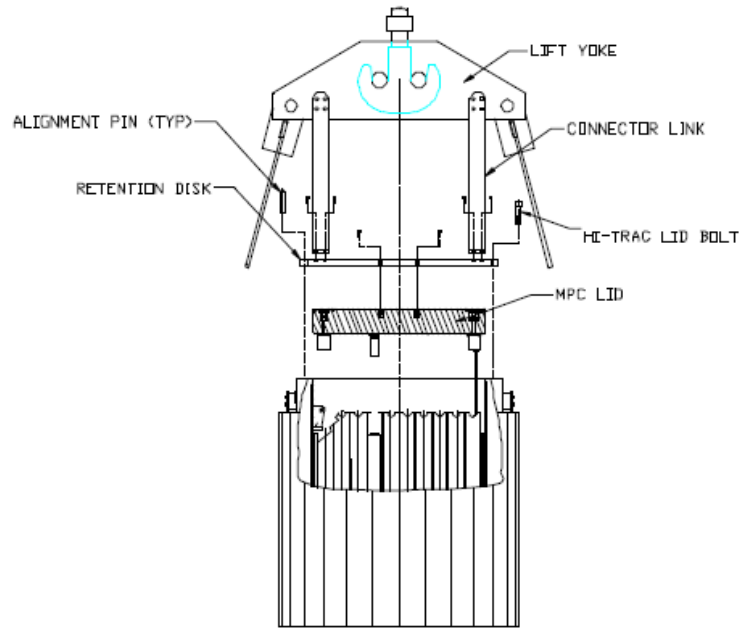
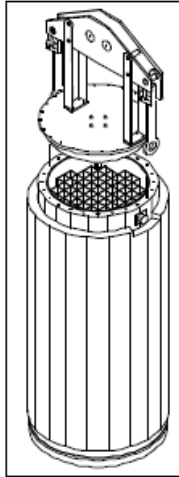


**Figure 8.1.13: Annulus Shield/Annulus Seal**

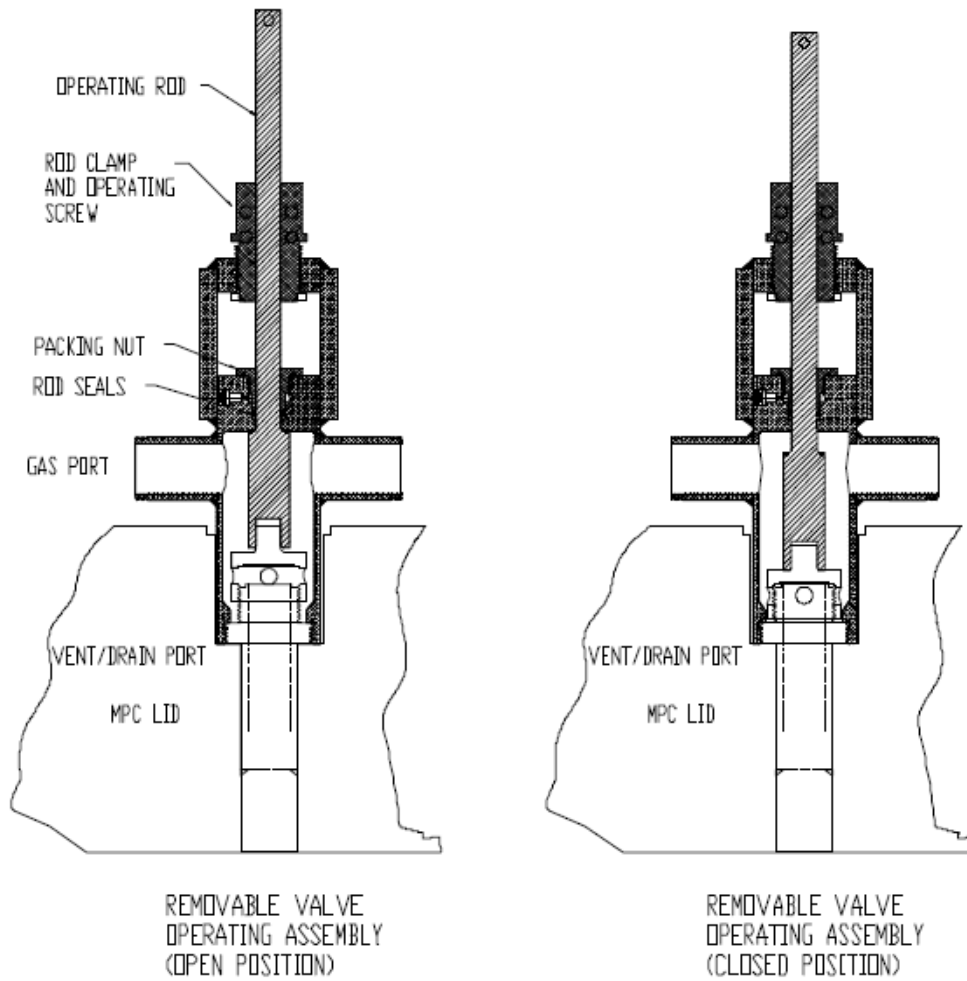


**Figure 8.1.14: Annulus Overpressure System**



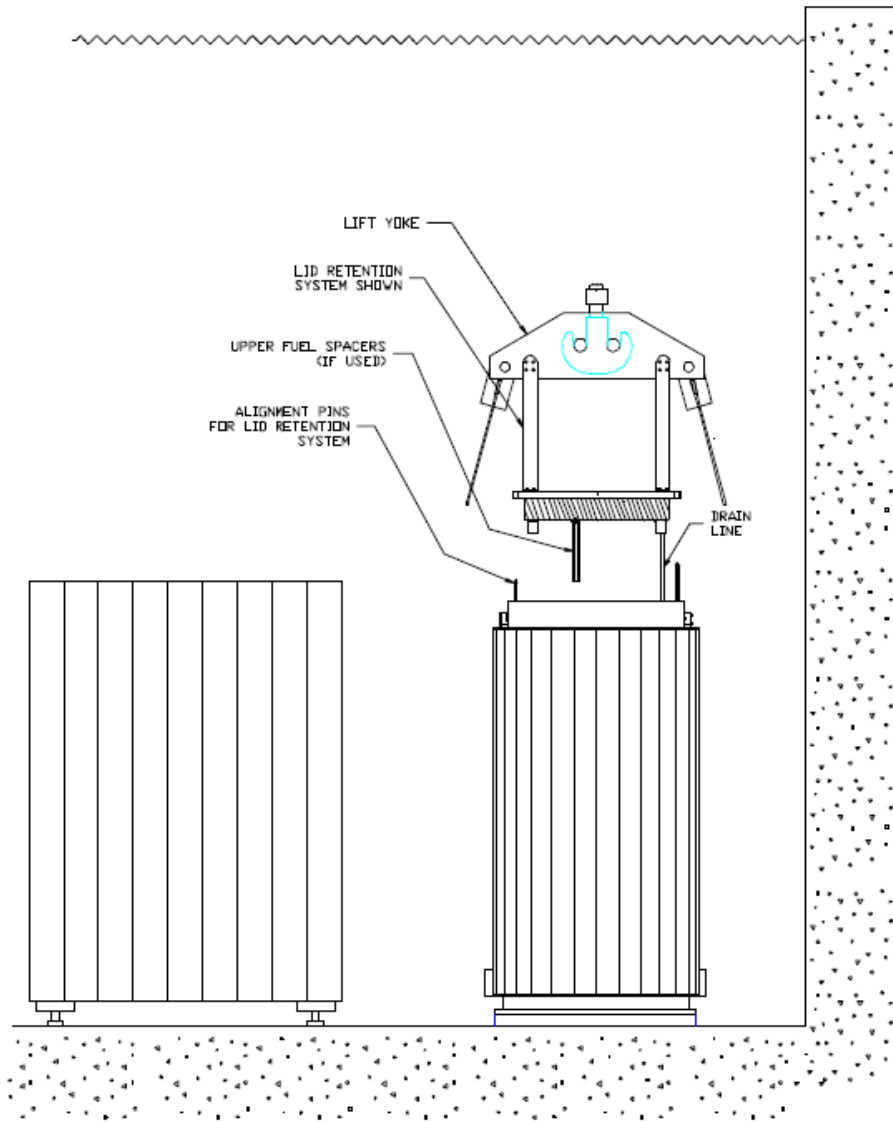


**Figure 8.1.15: HI-TRAC Lid Retention System in Exploded View**

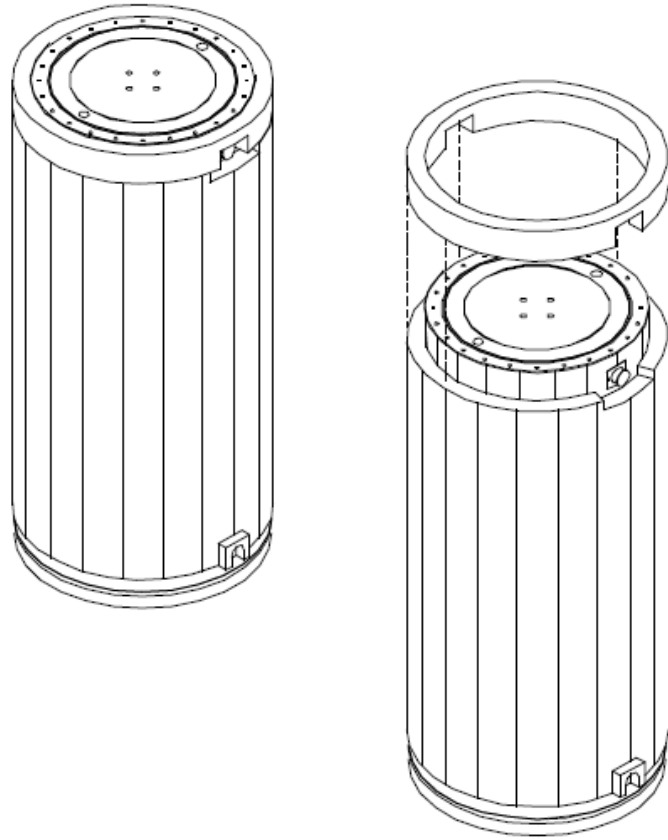


**Figure 8.1.16: MPC Vent and Drain Port RVOA Connector**

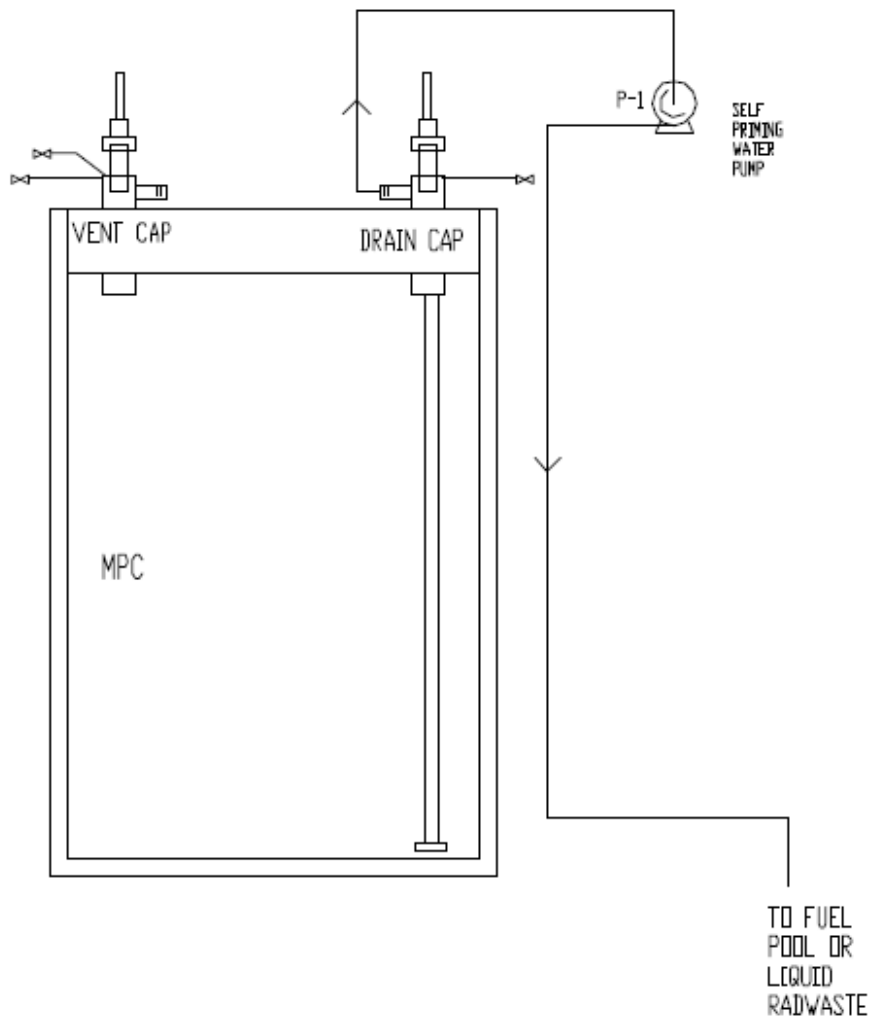
(Note – The Vent/Drain Connector may also have the optional configuration shown in the drawings in Section 1.5)



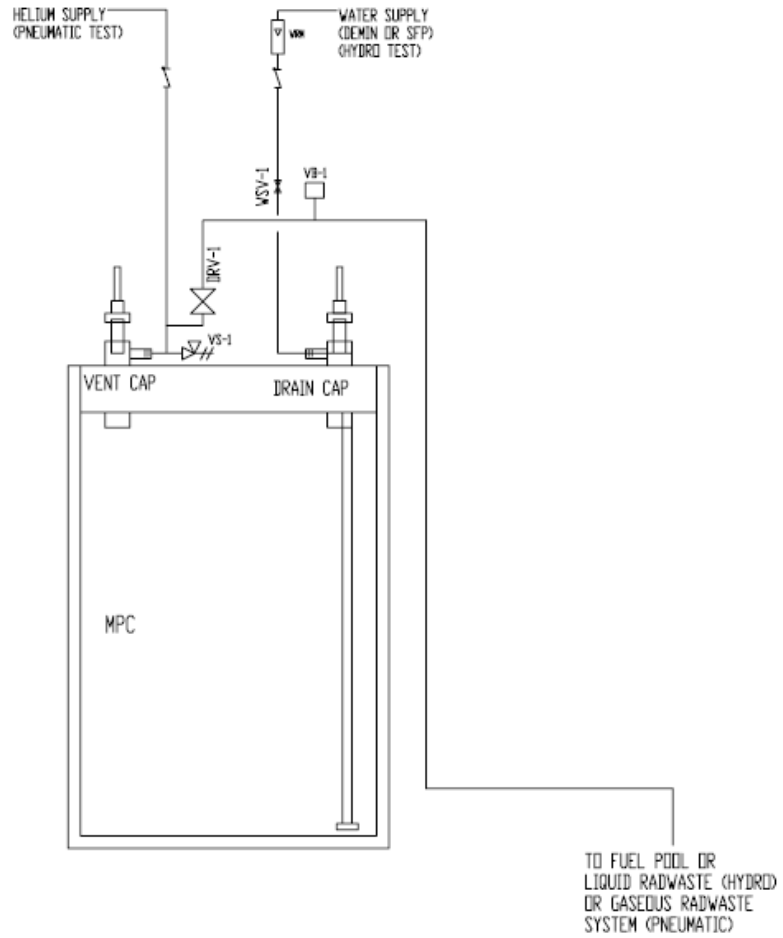
**Figure 8.1.17: Drain Line Installation**



**Figure 8.1.18: Temporary Shielding Ring**



**Figure 8.1.19: MPC Water Pump-Down for MPC Lid Welding Operations, Example P&I D**



**Figure 8.1.20: MPC Lid-to-Shell Pressure Testing, Example P&I D**

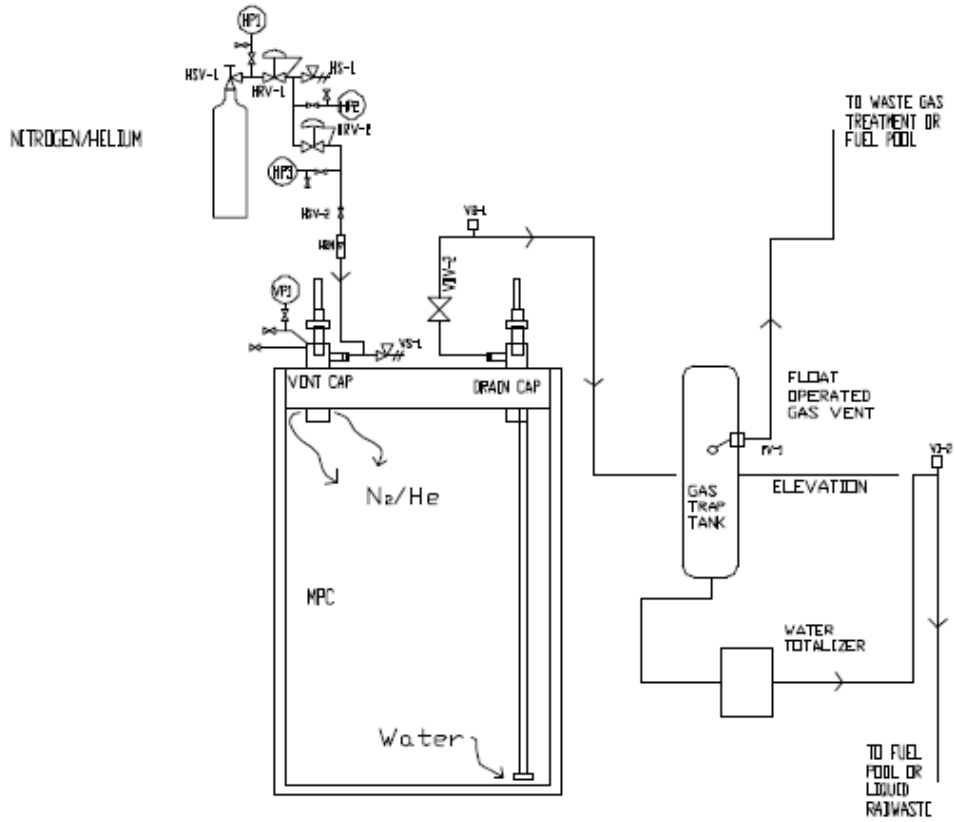
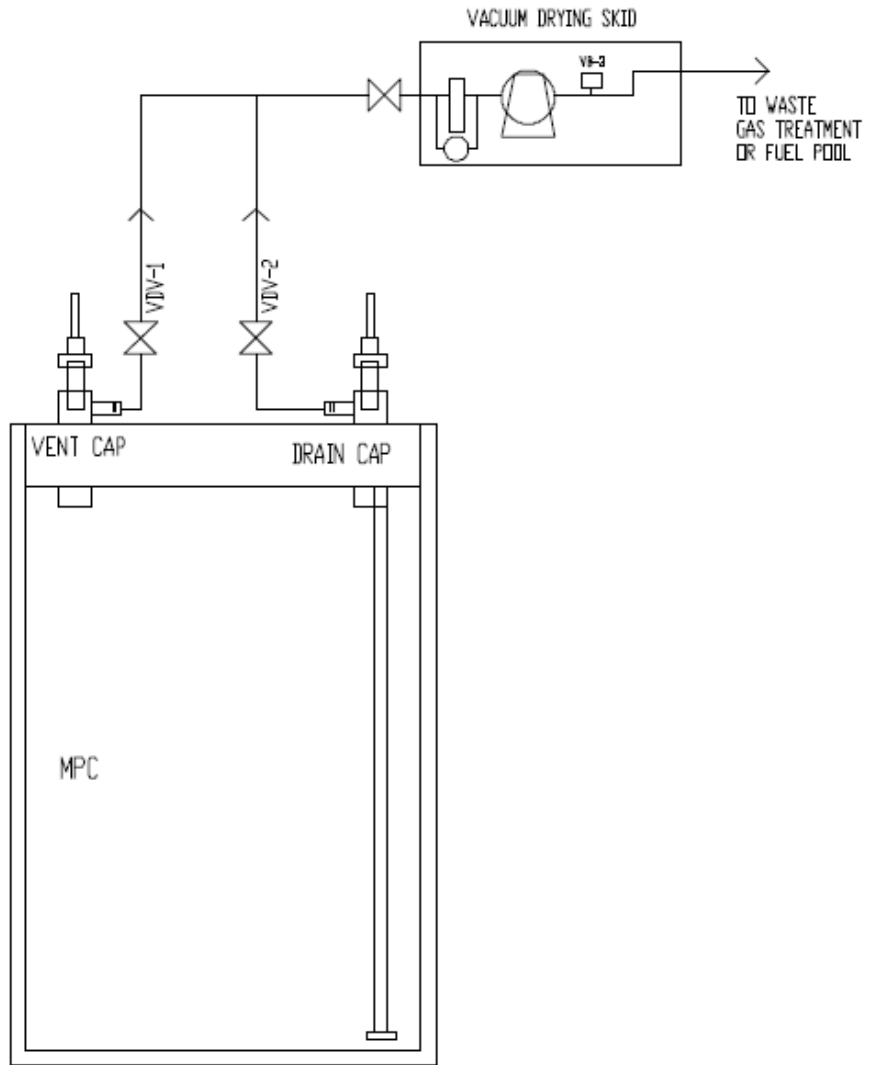
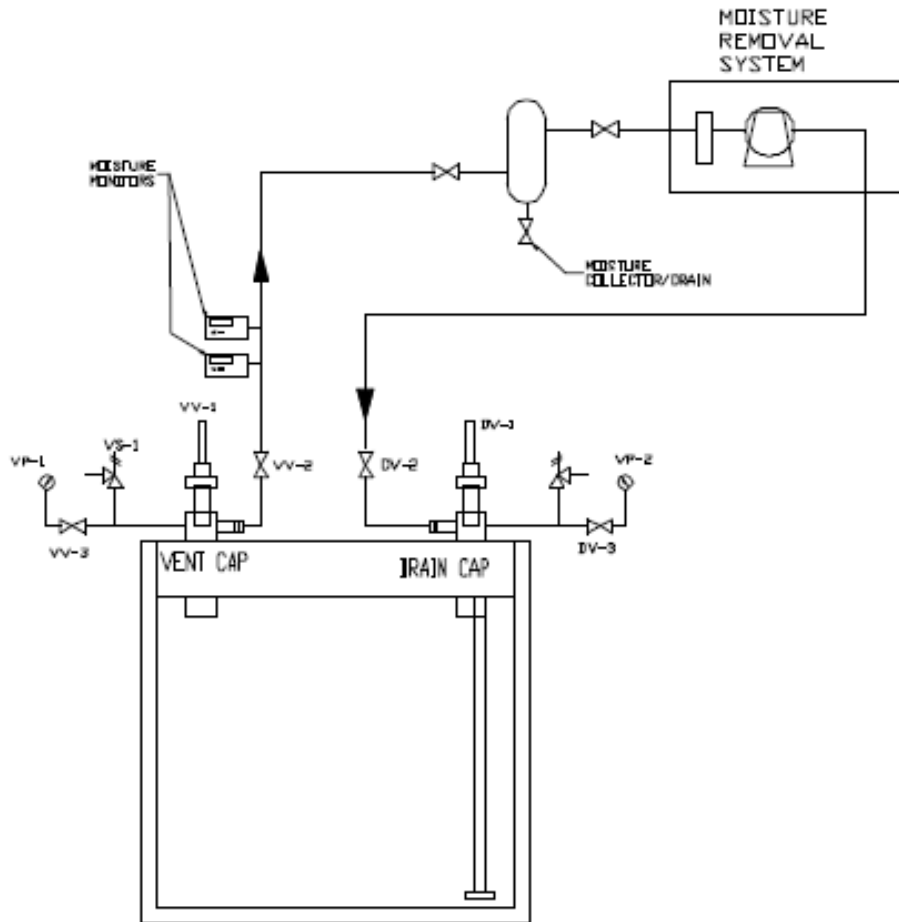


Figure 8.1.21: MPC Blowdown, Example P&I D

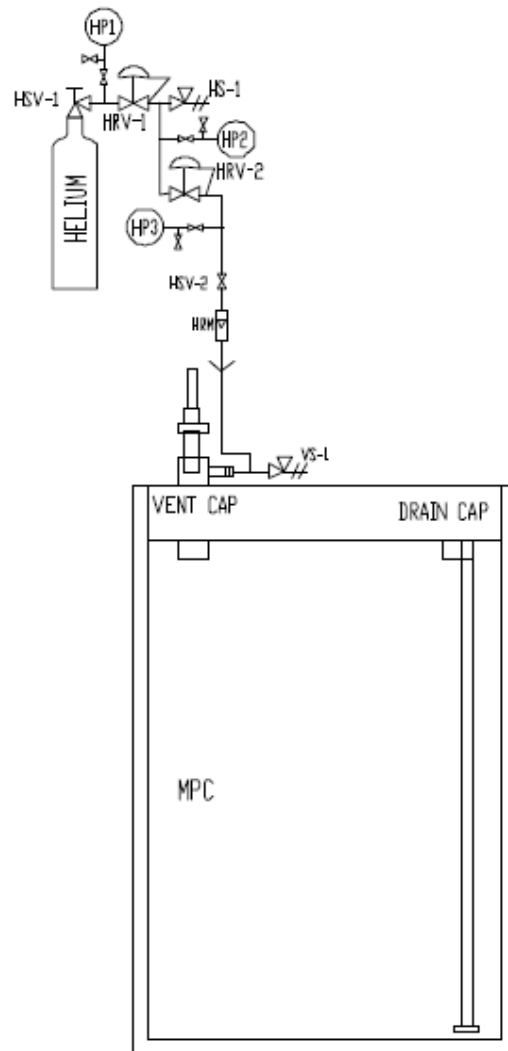


**Figure 8.1.22a: Vacuum Drying System, Example P&I D**

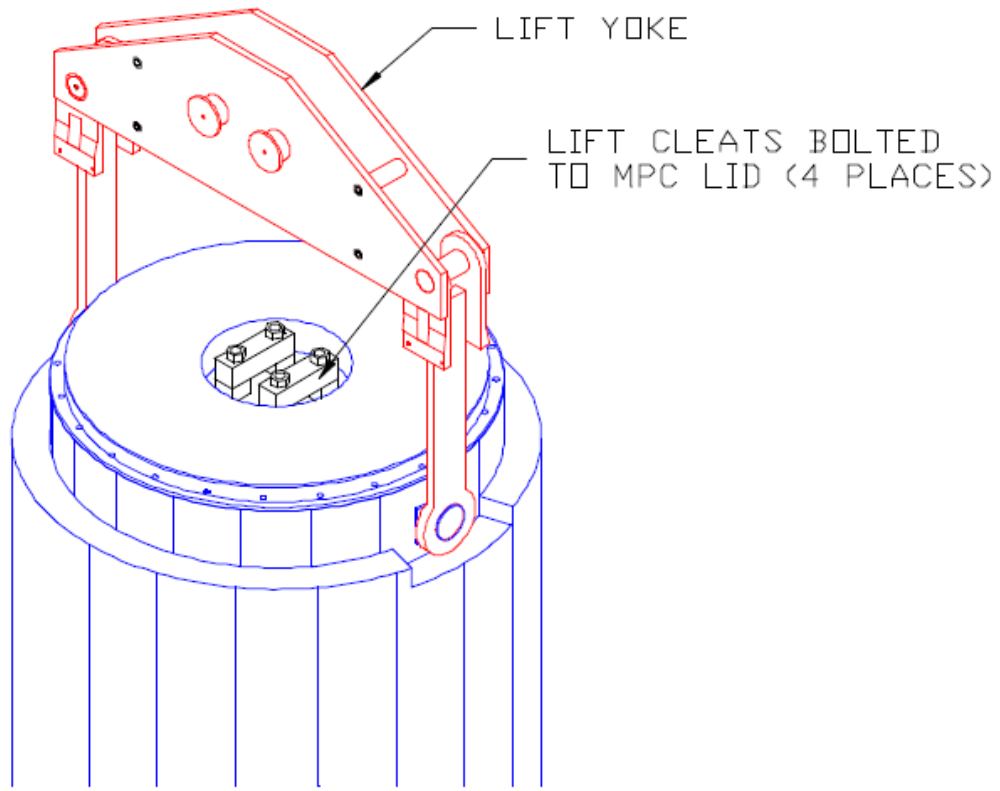




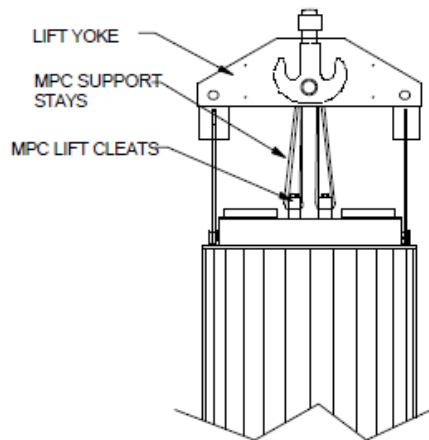
**Figure 8.1.22b: forced Helium Dehydration System, Example P&I D**



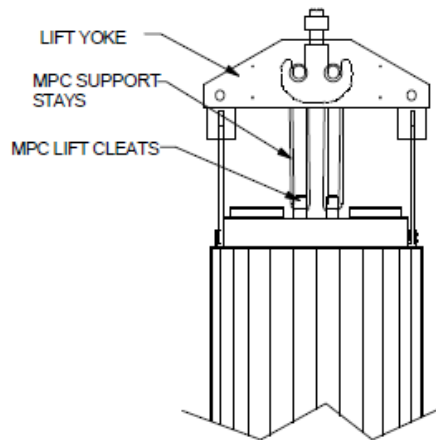
**Figure 8.1.23: Helium Backfill System, Example P&I D**



**Figure 8.1.24: MPC Lift Cleats**

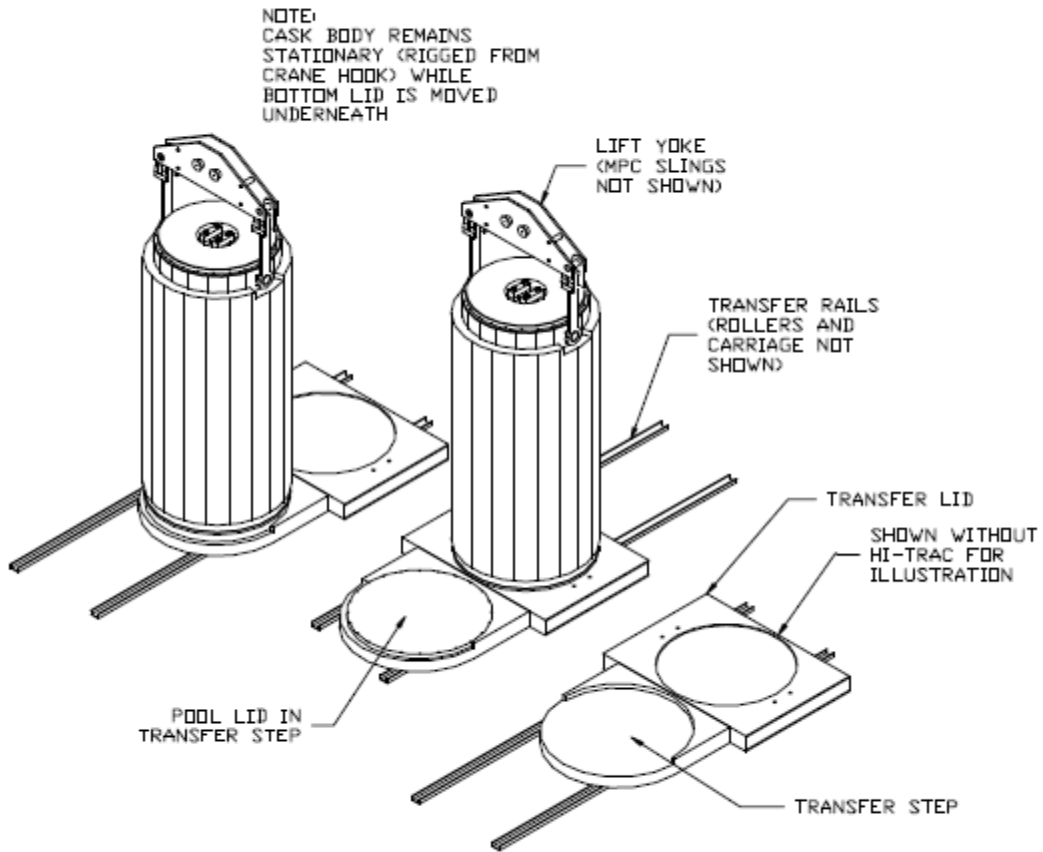


SINGLE-PIN  
ARRANGEMENT

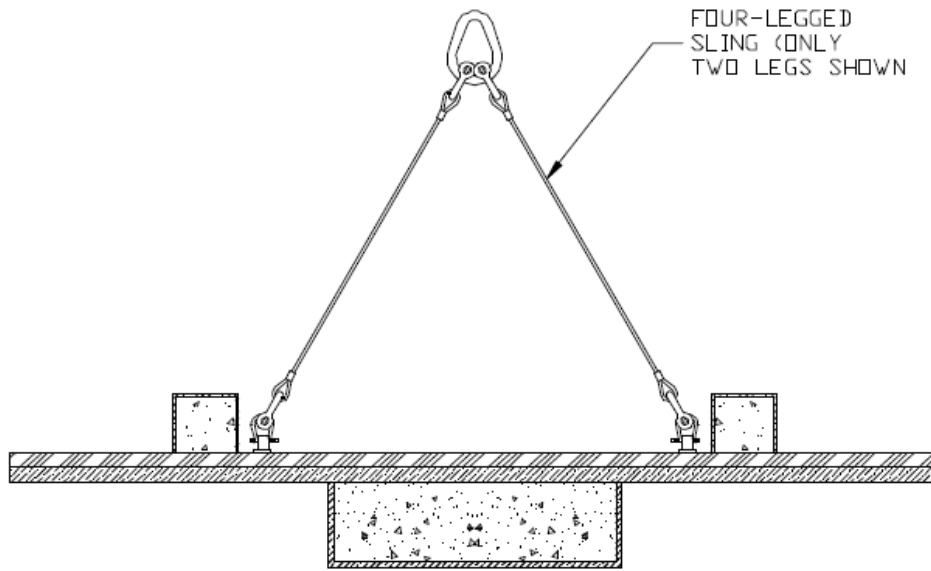


DOUBLE-PIN  
ARRANGEMENT

**Figure 8.1.25: MPC Slings**

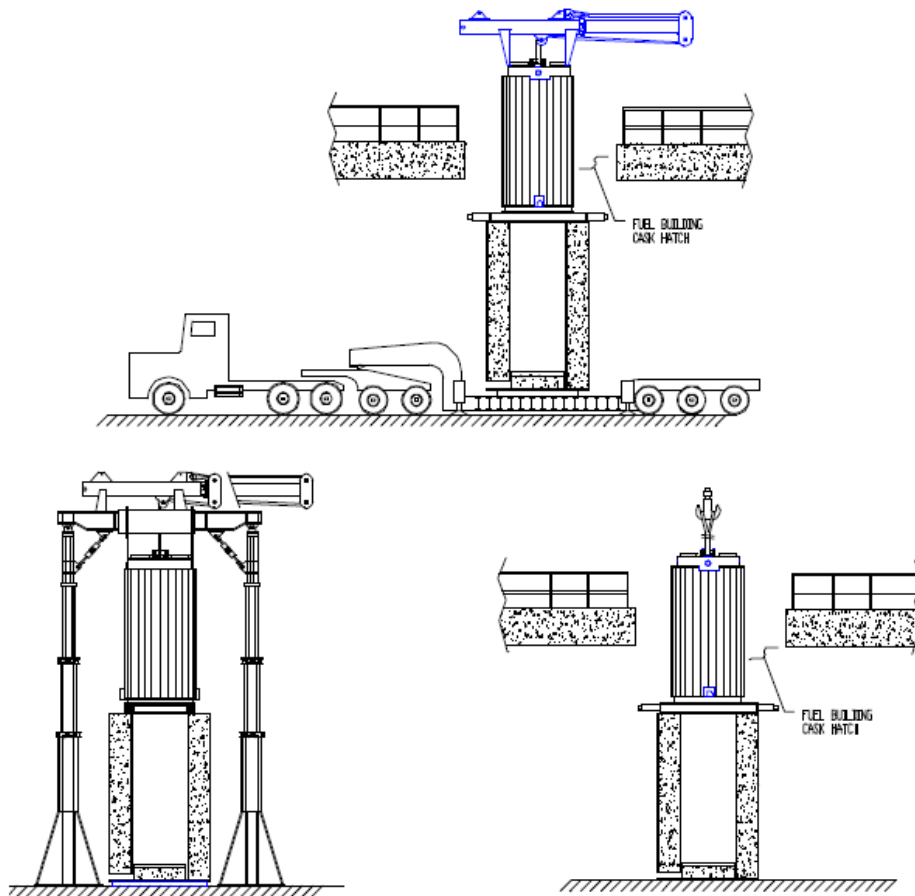


**Figure 8.1.26: HI-TRAC Bottom Lid Replacement**  
**(Not Required for HI-TRAC 100D and 125D)**

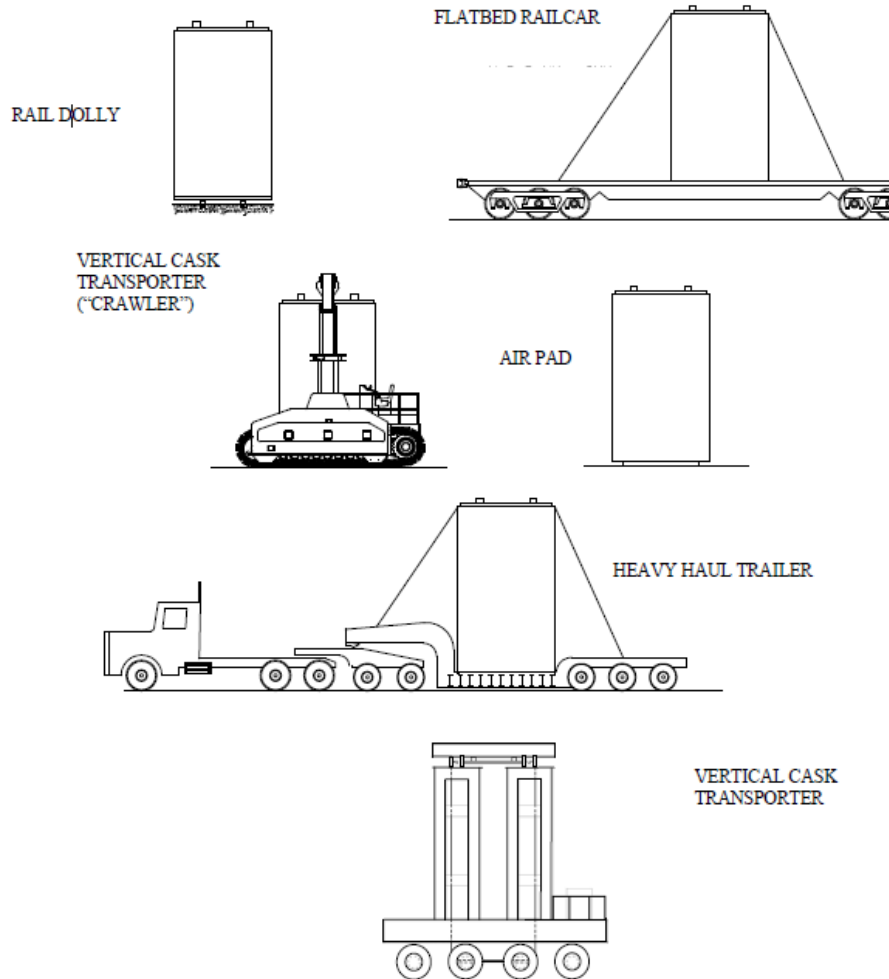


**Figure 8.1.27: HI-STORM Lid Rigging**

**(HI-STORM 100 Lid Shown)**

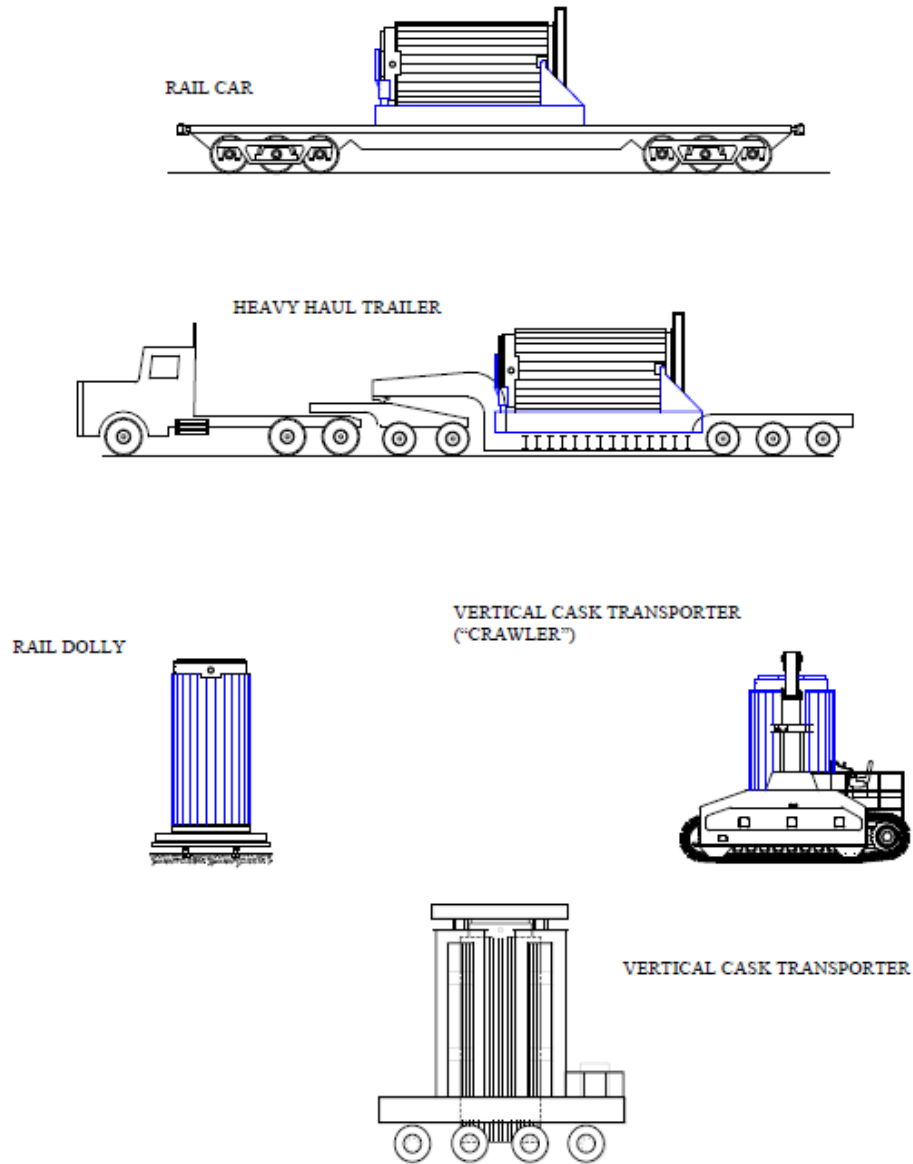


**Figure 8.1.28: Sample MPC Transfer Options  
(HI-TRAC with Transfer Lid Shown)**

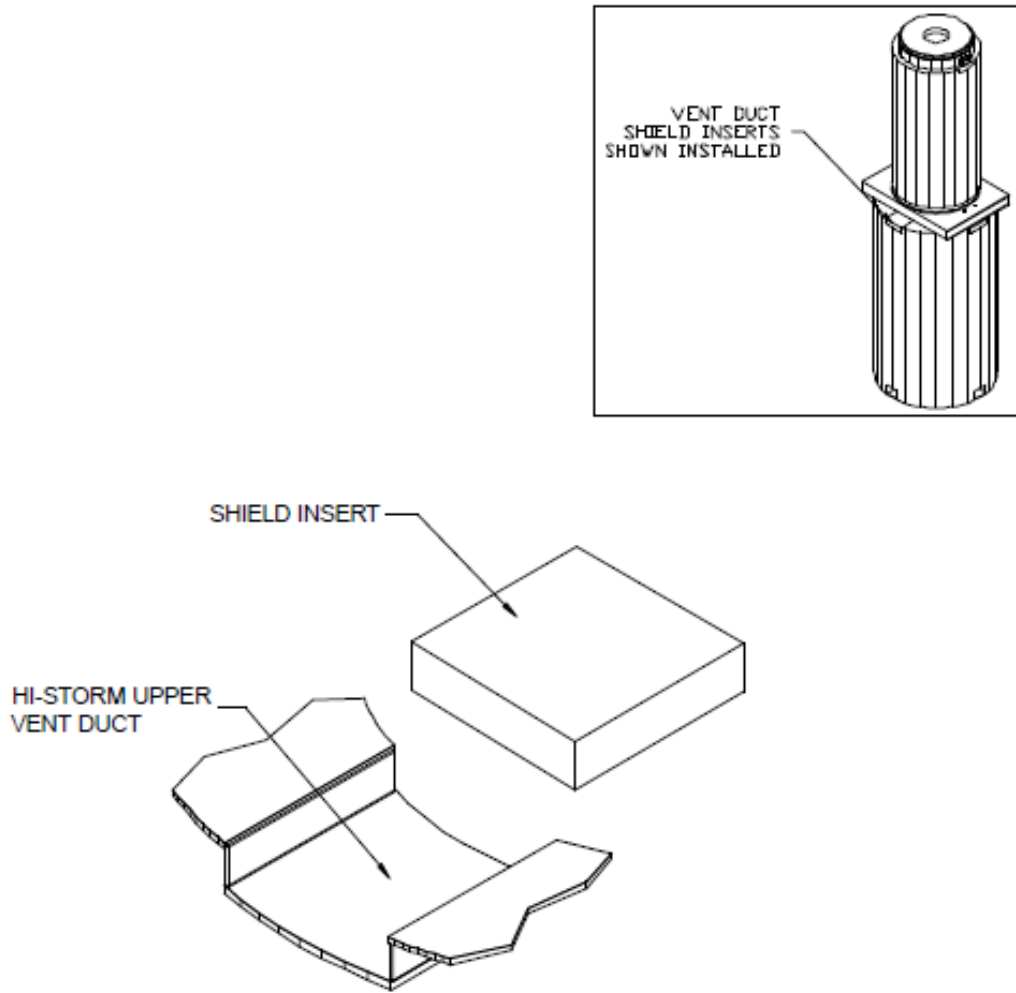


**Figure 8.1.29a: Sample HI-STORM Transfer Options**

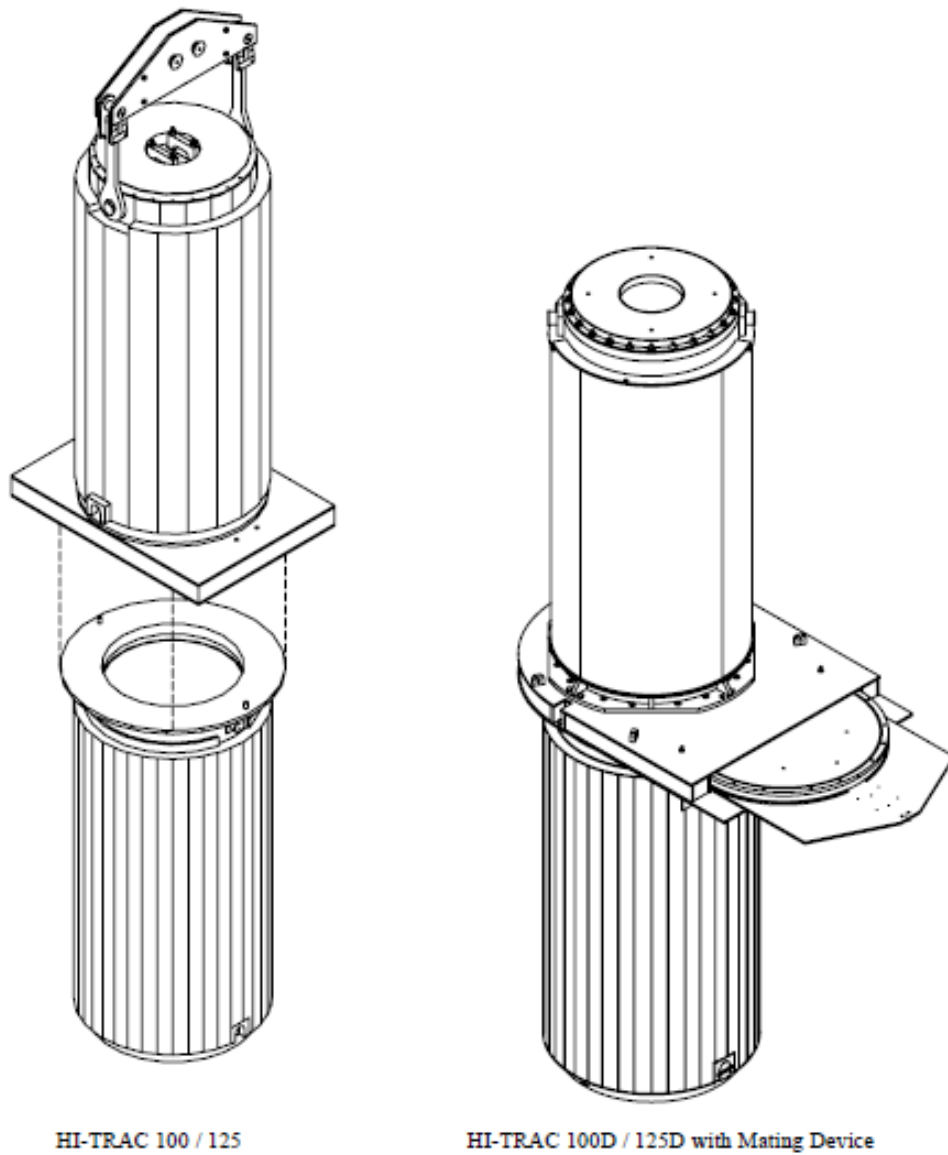




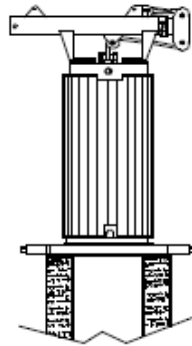
**Figure 8.1.29b: Sample HI-TRAC Transfer Options**



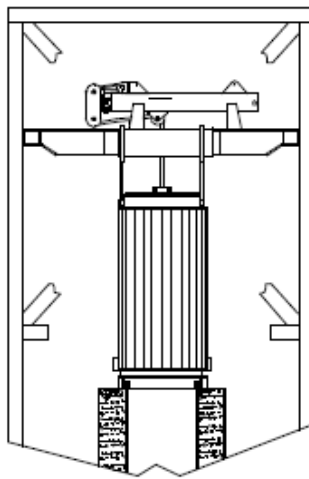
**Figure 8.1.30: Sample HI-STORM Vent Duct Shield Inserts  
(Not Required for HI-STORM 100S)**



**Figure 8.1.31: HI-TRAC Alignment Over HI-STORM**

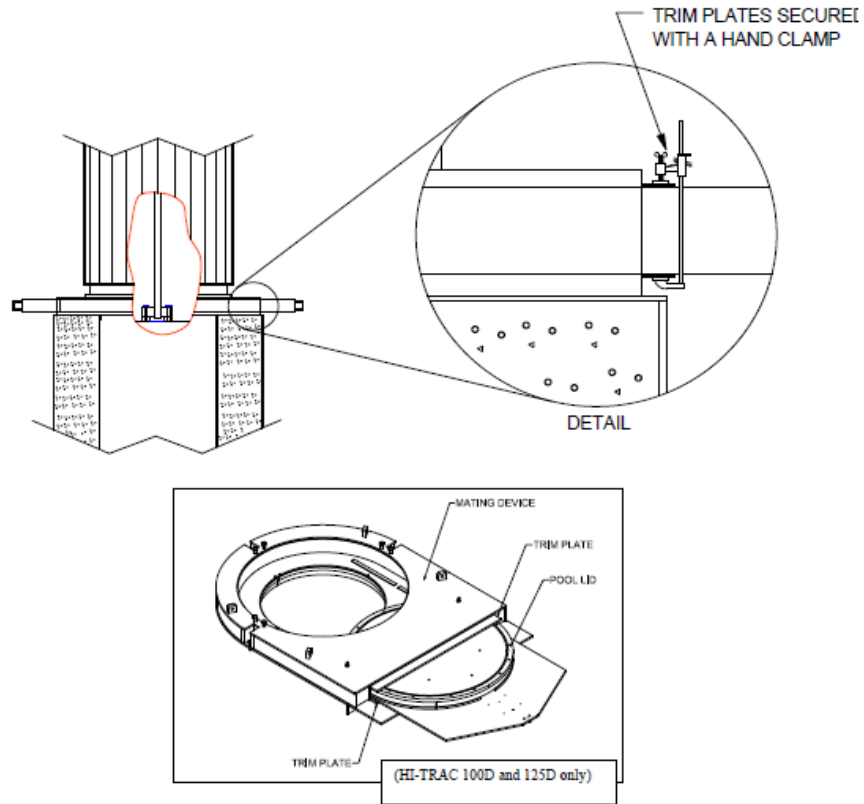


MOUNTED TO THE HI-TRAC LID

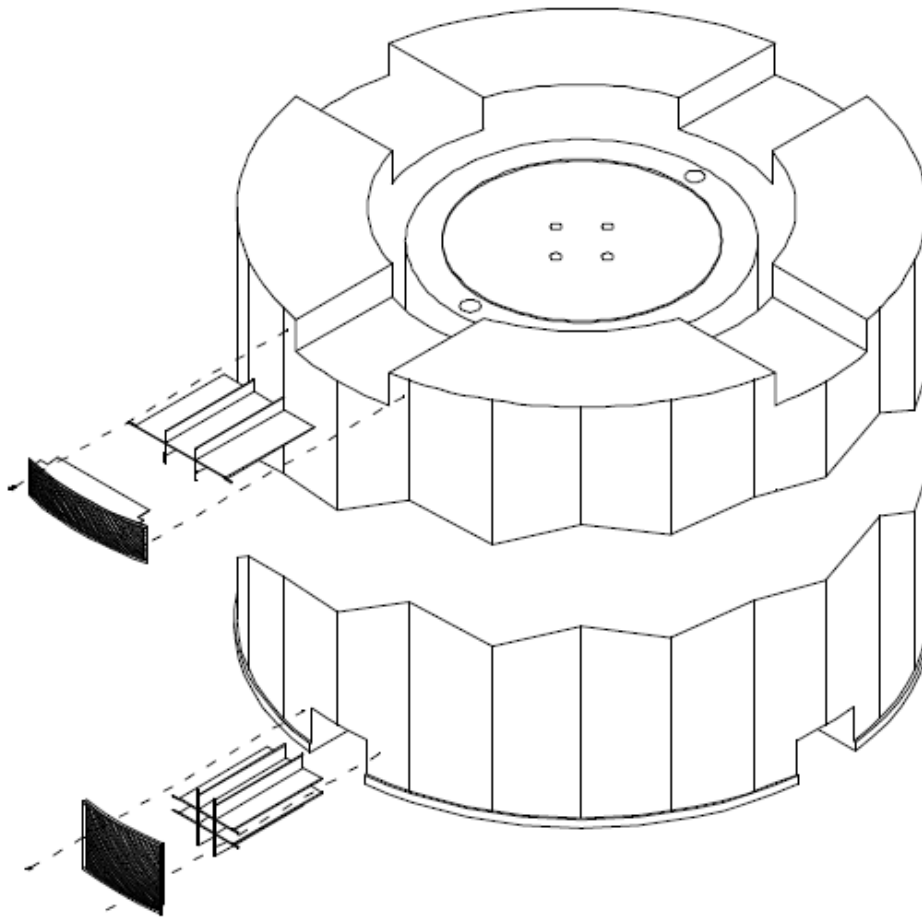


MOUNTED TO A CASK TRANSFER GANTRY

**Figure 8.1.32: Examples of an MPC Downloader**



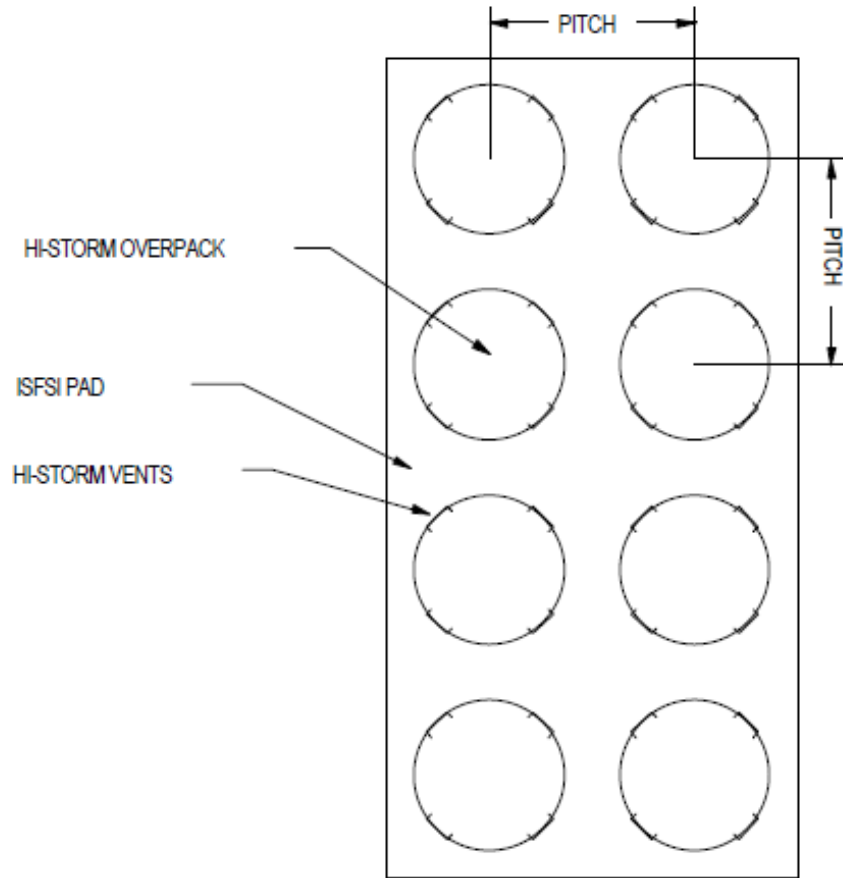
**Figure 8.1.33: Trim Plate Locations**



**Figure 8.1.34a: Typical HI-STORM Vent Screen and Gamma Shield Cross Plate Installation**

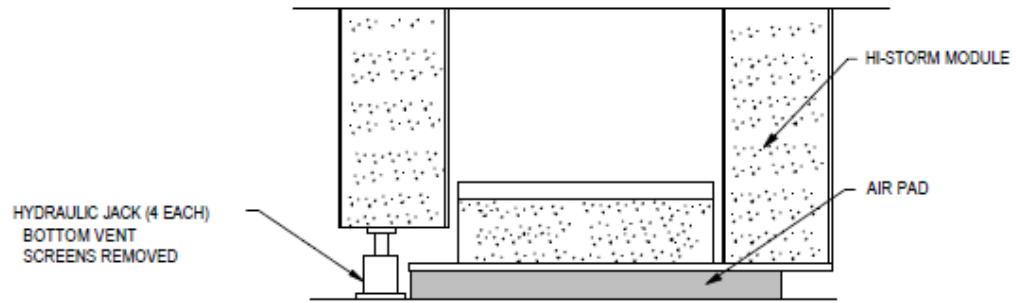
**Figure 8.1.34b**

**Intentionally Deleted**

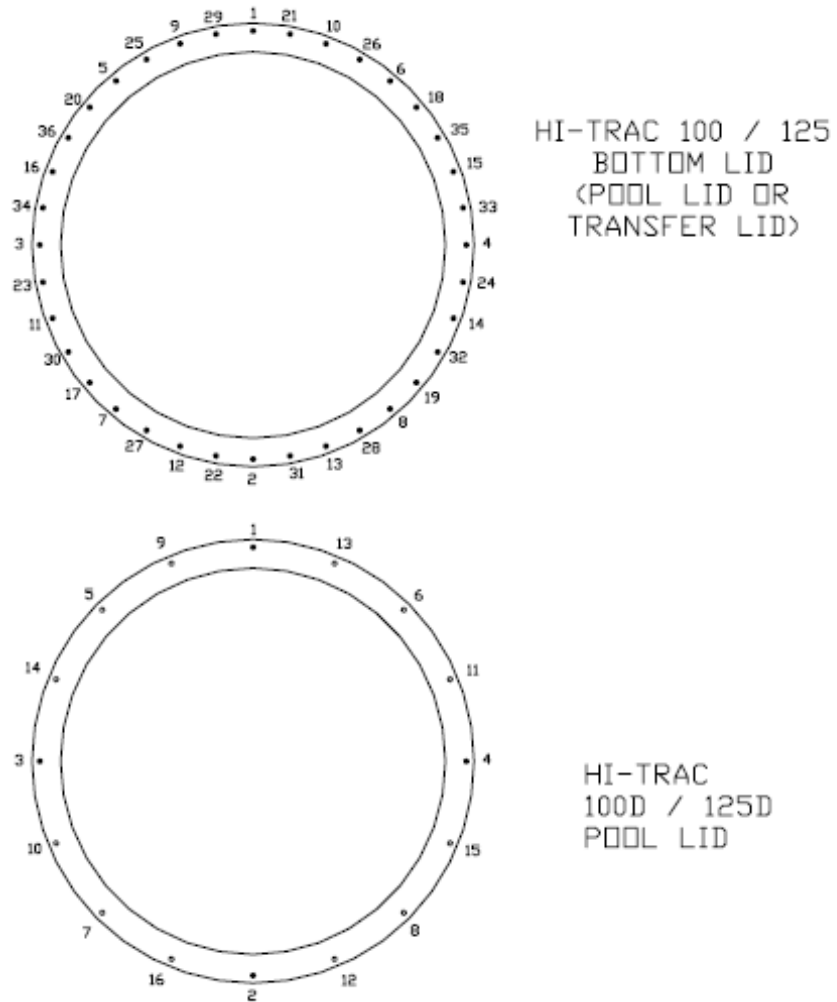


**Figure 8.1.35: HI-STORM Placement of the ISFSI Pad**





**Figure 8.1.36: HI-STORM Jacking**



**Figure 8.1.37: HI-TRAC Lid Bolt Torquing Pattern**

## 8.2 ISFSI OPERATIONS

The HI-STORM 100 System heat removal system is a totally passive system. Maintenance on the HI-STORM system is typically limited to cleaning and touch-up painting of the overpacks, repair and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM system. These include, but are not limited to, wind storms, heavy snow storms, fires inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM overpack, clear the debris, check the cask pitch, and/or replace damaged vent screens), perform the corrective action, verify that the system is operable (check ventilation flow paths and radiation). In the event of significant damage to the HI-STORM, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM overpack. If necessary, the procedures in Section 8.1 may be used to reposition a HI-STORM overpack for minor repairs and maintenance. In extreme cases, Section 8.3 may be used as guidance for unloading the MPC from the HI-STORM.

**Note:**

The heat removal system operability surveillance involves performing a visual examination on the HI-STORM exit and inlet vent screens to ensure that the vents remain clear or verifying the temperature rise from ambient to outlet is within prescribed limits. The metallic vent screens if damaged may allow leaves, debris or animals to enter the duct and block the flow of air to the MPC.

**ALARA Warning:**

Operators should practice ALARA principals when inspecting the vent screens. In most cases, binoculars allow the operator to perform the surveillance from a low dose area.

### 8.2.1 Perform the heat removal operability surveillance.

**Note:**

**CAUTION:** Some LCO requirements in the HI-STORM 100 CoC are based on individual system parameters (such as MPC total heat load). Sites should be aware of the variation in these requirements, and ensure procedures clearly identify how to implement these variations.

### 8.2.2 ISFSI Security Operations shall be performed in accordance with the approved site security program plan.

### 8.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

#### 8.3.1 Overview of HI-STORM 100 System Unloading Operations

**ALARA Note:**

The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. Figure 8.3.1 shows a flow diagram of the HI-STORM unloading operations. Figure 8.3.2 illustrates the major HI-STORM unloading operations.

Refer to the boxes of Figure 8.3.2 for the following description. The MPC is recovered from HI-STORM either at the ISFSI or the fuel building using the same methods as described in Section 8.1 (Box 1). The HI-STORM lid is removed, the vent duct shield inserts are installed, the alignment device (or mating device with pool lid for HI-TRAC 100D and 125D) is positioned, and the MPC lift cleats are attached to the MPC. The exit vent screens and gamma shield cross plates are removed as necessary. MPC slings are attached to the MPC lift cleat and positioned on the MPC lid. HI-TRAC is positioned on top of HI-STORM (Box 2) and the slings are brought through the HI-TRAC top lid. The MPC is raised into HI-TRAC, the HI-TRAC doors (or mating device drawer) are closed and the locking pins are installed. If the mating device and HI-TRAC 100D or 125D are used, the pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of HI-STORM. If the HI-TRAC 100D and 125D are not used, the HI-TRAC is positioned in the transfer slide and the transfer lid is replaced with the pool lid (Box 3) using the same method as with the loading operations.

If a Supplemental Cooling System (SCS) is required by the TS, it is connected to the HI-TRAC annulus following transfer from the HI-STORM to the HI-TRAC and used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits. HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC slings and MPC lift cleats are removed. The temporary shield ring is installed on the HI-TRAC upper section and filled with plant demineralized water. The HI-TRAC top lid is removed<sup>4</sup> (Box 4) and water is added to the annulus. If SCS is not being used, water is fed into the annulus through the drain port to cool the MPC shell. After filling the annulus with water the annulus and HI-TRAC top surfaces are covered to protect them from debris produced when removing the MPC lid. The weld removal system is installed (Box 7) and the MPC vent and drain ports are accessed (Box 5). The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas

<sup>4</sup> Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is filled with water (borated as required) at a controlled rate to avoid over pressurizing the MPC (Box 6) and the supplemental cooling system, if used, is terminated. The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place (Box 7).

The top surfaces of the HI-TRAC and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or lid retention system and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed (Boxes 8 and 9). All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris and crud (Box 10). HI-TRAC and MPC are returned to the designated preparation area (Box 11) where the MPC water is pumped back into the spent fuel pool or liquid radwaste facility. The annulus water is drained and the MPC and overpack are decontaminated (Box 12 and 13).

### 8.3.2 HI-STORM Recovery from Storage

**Note:**

The MPC transfer may be performed using the MPC downloader or the overhead crane.

**Note:**

The site-specific transport route conditions must satisfy the requirements of the technical specification.

1. Recover the MPC from HI-STORM as follows:
  - a. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met.
  - b. Transfer HI-STORM to the fuel building or site designated location for the MPC transfer.
  - c. Position HI-STORM under the lifting device.
  - d. Remove the HI-STORM lid nuts, washers and studs or lid closure bolts.
  - e. Remove the HI-STORM lid lifting hole plugs and install the lid lifting sling. See Figure 8.1.27.

**Note:**

The specific sequence for vent screen, temperature element, and gamma shield cross plate removal may vary based on the mode(s) or transport.

- f. Remove the HI-STORM exit vent screens, temperature elements and gamma shield cross plates. See Figure 8.1.34a.

**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) for the HI-STORM lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

- g. Remove the HI-STORM lid. See Figure 8.1.27.
  - h. Install the alignment device (or mating device with pool lid for HI-TRAC 100D and 125D) and vent duct shield inserts (HI-STORM 100 only). See Figure 8.1.30.
  - i. Deleted.
  - j. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings to the MPC lid. See Table 8.1.5 for torque requirements.
  - k. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for rigging. See Table 8.1.5 for torque requirements.
  - l. Deleted.
2. If necessary, configure HI-TRAC with the transfer lid (Not required for HI-TRAC 100D and 125D):

**ALARA Warning:**

The bottom lid replacement as described below may only be performed on an empty (i.e., no MPC) HI-TRAC.

- m. Position HI-TRAC vertically adjacent to the transfer lid. See Section 8.1.2.
  - n. Remove the bottom lid bolts and plates and store them temporarily.
  - o. Raise the empty HI-TRAC and position it on top of the transfer lid.
  - p. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - q. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.
3. At the site's discretion, perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

**Note:**

If the HI-TRAC is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

- 4. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.
- 5. Engage the lift yoke to the HI-TRAC lifting trunnions.

6. Align HI-TRAC over HI-STORM and mate the overpacks. See Figure 8.1.31.
7. If necessary, install the MPC downloader.
8. Remove the transfer lid (or mating device) locking pins and open the doors (mating device drawer).

**ALARA Warning:**

If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

9. At the user's discretion, install trim plates to cover the gap above and below the door (drawer for 100D and 125D). The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
10. Attach the ends of the MPC sling to the lifting device or MPC downloader. See Figure 8.1.32.

**Caution:**

Limitations for handling an MPC containing high burn-up fuel and total MPC heat load greater than the threshold heat load setting in Table 4.5.4 at the time of unloading in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance. The Supplemental Cooling System (SCS) is used to prevent fuel cladding temperatures from exceeding ISG-11 Rev. 3 limits. Operation of the SCS is initiated in accordance with the TS and continues until MPC re-flooding operations have commenced. Staging and check-out of the SCS shall be completed prior to transferring the MPC to the HI-TRAC.

11. Raise the MPC into HI-TRAC.
12. Verify the MPC is in the full-up position.
13. Close the HI-TRAC doors (or mating device drawer) and install the door locking pins.
14. For the HI-TRAC 100D and 125D, bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.
15. Lower the MPC onto the transfer lid doors (or pool lid for 100D and 125D).
16. Disconnect the slings from the MPC lift cleats.
17. If necessary, remove the MPC downloader from the top of HI-TRAC.
18. Remove HI-TRAC from the top of HI-STORM.

**Note:**

For the HI-TRAC 100 and HI-TRAC 125, operation of the SCS may need to be postponed until the pool lid is in place on the HI-TRAC. In any event, the supplemental cooling system shall be initiated in accordance with the TS.

19. Replace the transfer lid with the pool lid as follows (Not required for HI-TRAC 100D and 125D):
- a. Lower the lift yoke and attach the MPC slings between the lift cleats and the lift yoke. See Figure 8.1.25.
  - b. Engage the lift yoke to the HI-TRAC lifting trunnions.
  - c. Deleted.
  - d. Raise HI-TRAC and position the transfer lid approximately one inch above the transfer step. See Figure 8.1.26.
  - e. Raise the transfer slide carriage so the transfer carriage is supporting the transfer lid bottom. Remove the transfer lid bolts and store them temporarily.

**ALARA Warning:**

Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

- f. Lower the transfer carriage and position the pool lid under HI-TRAC.
- g. Raise the transfer slide carriage to place the pool lid against the HI-TRAC bottom lid bolting flange.
- h. Inspect the bottom lid bolts for general condition. Replace worn or damaged bolts with new bolts.
- i. Install the pool lid bolts. See Table 8.1.5 for torque requirements.
- j. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1) Continue operation of the SCS until MPC re-flooding operations have commenced.
- k. Raise and remove the HI-TRAC from the transfer slide.
- l. Disconnect the MPC slings and lift cleats.

**Warning:**

At the start of SCS operations, annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of filling the annulus. Water addition should be performed in a slow and controlled manner until water steam generation has ceased.

20. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1). Continue operation of the SCS until MPC re-flooding operations have commenced.

8.3.3 Preparation for Unloading:

1. Place HI-TRAC in the designated preparation area.



**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) for the HI-TRAC top lid, the top lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

2. Prepare for MPC cool-down.
3. Remove the top lid bolts and remove HI-TRAC top lid. See Figure 8.1.9 for rigging.

**Warning:**

At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Water addition should be performed in a slow and controlled manner until water steam generation has ceased.

4. Set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid.
5. Access the MPC as follows:

**ALARA Note:**

The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically vacuumed.

**ALARA Warning:**

Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

- a. Install bolt plugs and/or waterproof tape on the HI-TRAC top bolt holes.
  - b. Using the marked locations of the vent and drain ports, core drill the closure ring and vent and drain port cover plates.
6. Remove the closure ring section and the vent and drain port cover plates.

**ALARA Note:**

The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

7. Take an MPC gas sample as follows:

**Note:**

Users may select alternate methods of obtaining a gas sample.

- a. Attach the RVOAs (See Figure 8.1.16).
- b. Attach a sample bottle to the vent port RVOA as shown on Figure 8.3.3.
- c. Evacuate the RVOA and Sample Bottle.
- d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

- e. Close the vent port cap and disconnect the sample bottle.

**ALARA Note:**

The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

- f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.
- g. Deleted.

8. Fill the MPC cavity with water as follows:

**Caution:**

This Caution block is required by the HI-STORM 100 CoC (CoC Appendix B, Section 3.4.10 and CoC Appendix B-100U, Section 3.4.12) and may not be deleted without prior NRC approval via CoC amendment. To prevent the oxidation of the fuel the MPC interior shall be filled with helium or another suitable inert gas to avoid exposing the fuel to oxidizing agents while at elevated temperatures. Exposing fuel at elevated temperatures to oxidizing agents can lead to deleterious oxidation of the fuel.

- a. Open the vent and drain port caps using the RVOAs.
- b. Deleted.
- c. Deleted.
- d. Deleted.
- e. Deleted.
- f. Deleted.
- g. Deleted.
- h. Deleted.

**Caution:**

The introduction of water into the MPC may create steam. Re-flooding operations shall be closely controlled to insure that the internal pressure in the MPC does not exceed design limits. The water flow rate shall be adjusted to maintain the internal pressure below design limits.

- i. Prepare the MPC fill and vent lines as shown on Figure 8.1.20. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

**Note:**

When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.

- j. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. (Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.
- k. If used, cease operation of the SCS and remove the system from the HI-TRAC.

**Caution:**

Combustible gas monitoring as described this Caution block is required by the HI-STORM 100 CoC (CoC Appendix B and B-100U, Section 3.8) and may not be deleted without prior NRC approval via CoC amendment. Oxidation of neutron absorber panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid cutting operations. The space below the MPC lid shall be purged with inert gas prior to, and during MPC lid cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.

- l. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.
- m. Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge the gas space under the lid as necessary
- n. Remove the MPC lid-to-shell weld using the weld removal system. See Figure 8.1.9 for rigging.
- o. Vacuum the top surfaces of the MPC and HI-TRAC to remove any metal shavings.

## 9. Install the inflatable annulus seal as follows:

**Caution:**

Do not use any sharp tools or instruments to install the inflatable seal.

- a. Remove the annulus shield.
- b. Manually insert the inflatable seal around the MPC. See Figure 8.1.13.
- c. Ensure that the seal is uniformly positioned in the annulus area.

- d. Inflate the seal
  - e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.
10. Place HI-TRAC in the spent fuel pool as follows:
- a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.
  - b. Engage the lift yoke to HI-TRAC lifting trunnions, remove the MPC lid lifting hole plugs and attach the MPC lid slings or lid retention system to the MPC lid.
  - c. If the lid retention system is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - d. Install the lid retention system bolts if the lid retention system is used.

**ALARA Note:**

The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

- e. If used, fill the annulus overpressure system lines and reservoir with demineralized water and close the reservoir valve. Attach the annulus overpressure system to the HI-TRAC. See Figure 8.1.14.
- f. Position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

**ALARA Note:**

Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- g. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.
- h. When the top of the HI-TRAC reaches the elevation of the reservoir, open the annulus overpressure system reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.
- i. If the lid retention system is used, remove the lid retention bolts when the top of HI-TRAC is accessible from the operating floor.
- j. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.
- k. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.
- l. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.

- m. Disconnect the drain line from the MPC lid.
- n. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.
- o. Disconnect the lid retention system if used.

#### 8.3.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Vacuum the cells of the MPC to remove any debris or corrosion products.
3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

#### 8.3.5 Post-Unloading Operations

1. Remove HI-TRAC and the unloaded MPC from the spent fuel pool as follows:
  - a. Engage the lift yoke to the top trunnions.
  - b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.
  - c. Raise HI-TRAC until HI-TRAC flange is at the surface of the spent fuel pool.

**ALARA Warning:**

Activated debris may have settled on the top face of HI-TRAC during fuel unloading.

- d. Measure the dose rates at the top of HI-TRAC in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.
- e. Raise the top of HI-TRAC and MPC to the level of the spent fuel pool deck.
- f. Close the annulus overpressure system reservoir valve.
- g. Using a water pump, lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

**ALARA Note:**

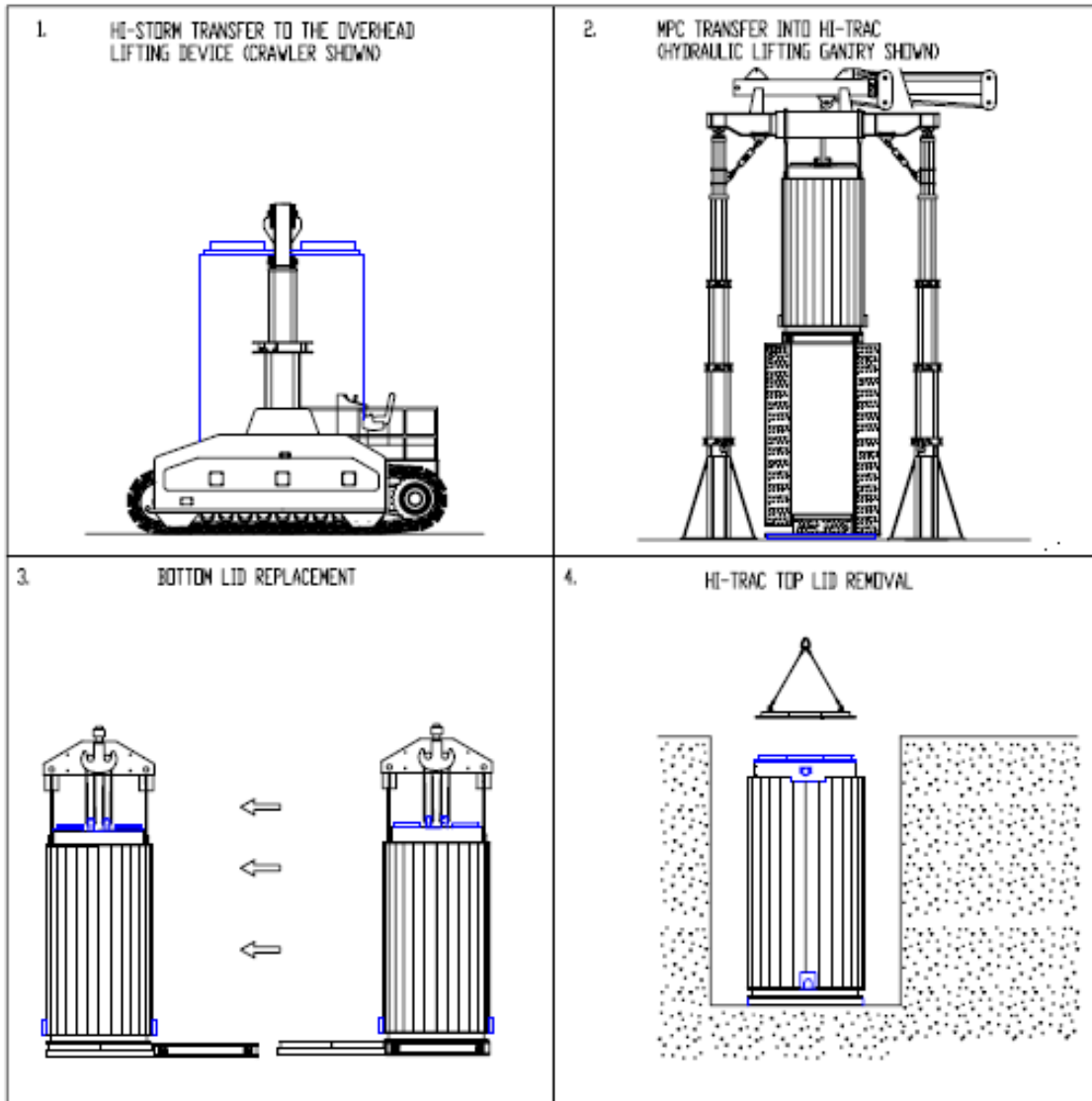
To reduce contamination of HI-TRAC, the surfaces of HI-TRAC and lift yoke should be kept wet until decontamination can begin.

- h. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water.
- i. Disconnect the annulus overpressure system from the HI-TRAC via the quick disconnect.
- j. Place HI-TRAC in the designated preparation area.
- k. Disengage the lift yoke.

1. Perform decontamination on HI-TRAC and the lift yoke.
2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.
3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.
4. Drain the water in the annulus area by connecting the drain line to the HI-TRAC drain connector.
5. Remove the MPC from HI-TRAC and decontaminate the MPC as necessary.
6. Decontaminate HI-TRAC.
7. Remove the bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.
8. Return any HI-STORM 100 equipment to storage as necessary.

<b>LOCATION: ISFSI</b>
RECOVER HI-STORM FROM STORAGE
<b>LOCATION: CASK RECEIVING AREA</b>
REMOVE HI-STORM EXIT VENT SCREENS AND CROSS PLATES
REMOVE HI-STORM LID
INSTALL HI-STORM ALIGNMENT DEVICE (OR MATING DEVICE) AND VENT DUCT SHIELD INSERTS
INSTALL MPC LIFT CLEATS
ATTACH MPC SLINGS
INSTALL TOP LID ON HI-TRAC
INSTALL MPC LIFT SLINGS
RAISE HI-TRAC AND MATE OVERPACKS
OPEN SHIELD DOORS (OR MATING DEVICE DRAWER)
ATTACH SLINGS TO LIFT DEVICE AND CLEAT
RAISE MPC INTO HI-TRAC
CLOSE HI-TRAC TRANSFER LID DOORS (HI-TRAC 100D and 125D - BOLT UP POOL LID)
REMOVE HI-TRAC FROM TOP OF HI-STORM
DISCONNECT MPC LIFT SLINGS
REPLACE TRANSFER LID WITH POOL LID (NOT FOR HI-TRAC 100D and 125D)
DISCONNECT MPC LIFT CLEAT
PLACE HI-TRAC IN DESIGNATED PREPARATION AREA
<b>LOCATION: CASK PREPARATION AREA</b>
REMOVE HI-TRAC TOP LID
FILL ANNULUS
INSTALL ANNULUS SHIELD
REMOVE MPC CLOSURE RING
REMOVE VENT PORT COVERPLATE WELD AND SAMPLE MPC GAS
FILL MPC CAVITY WITH WATER
REMOVE MPC LID TO SHELL WELD
INSTALL INFLATABLE SEAL
PLACE HI-TRAC IN SPENT FUEL POOL
<b>LOCATION: SPENT FUEL POOL</b>
REMOVE MPC LID
DISCONNECT DRAIN LINE
REMOVE SPENT FUEL ASSEMBLIES WASTE FROM MPC
VACUUM CELLS OF MPC
REMOVE HI-TRAC FROM SPENT FUEL POOL
<b>LOCATION: CASK PREPARATION AREA</b>
LOWER WATER LEVEL IN MPC
PUMP REMAINING WATER IN MPC TO SPENT FUEL POOL
REMOVE MPC FROM HI-TRAC
DECONTAMINATE HI-TRAC

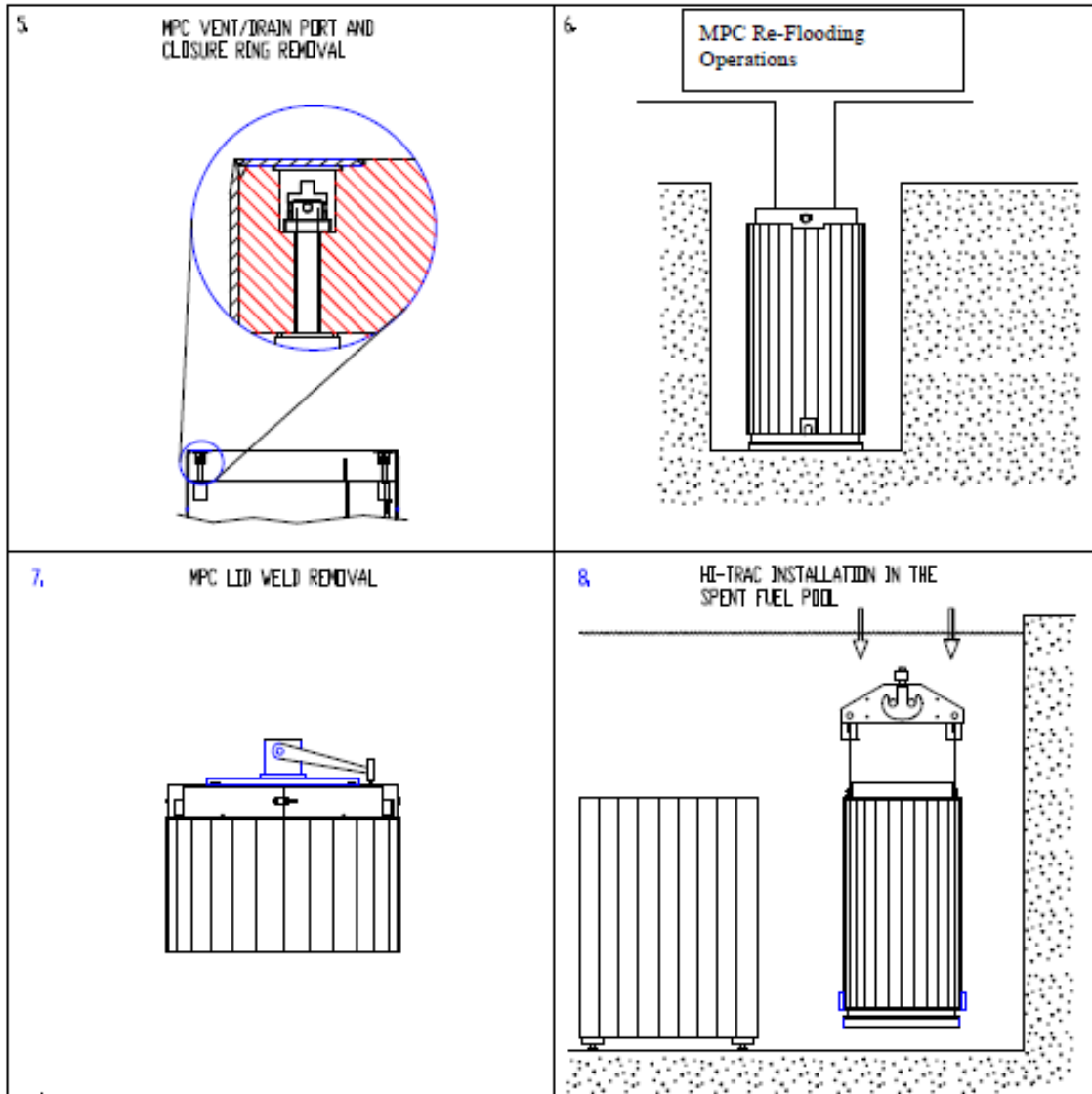
**Figure 8.3.1: Unloading Operations Flow Diagram**



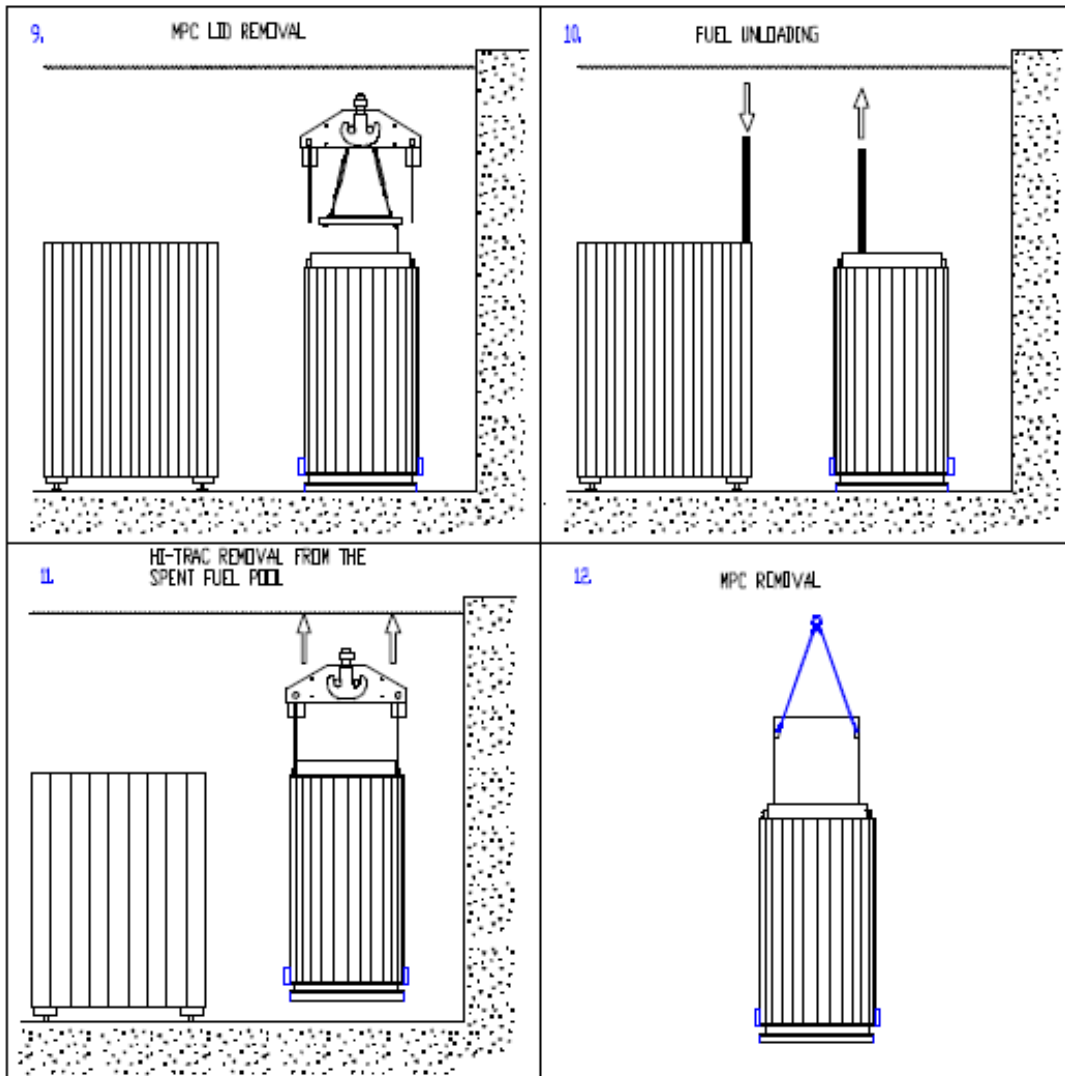
**Note: Bottom Lid Replacement Not Required for HI-TRAC 100D and 125D**

**Figure 8.3.2a: Major HI-STORM 100 Unloading Operations**

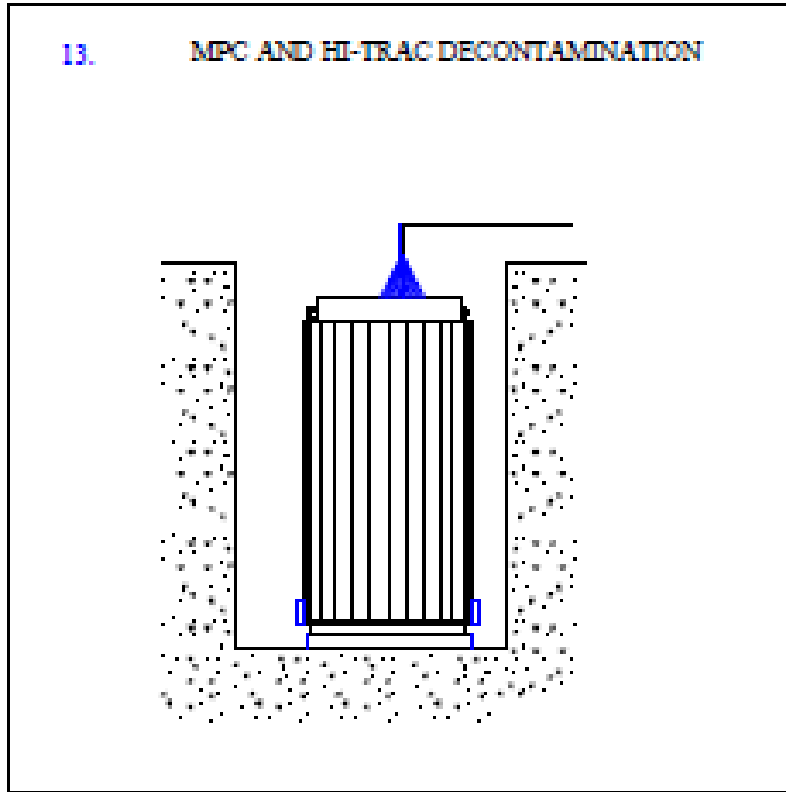




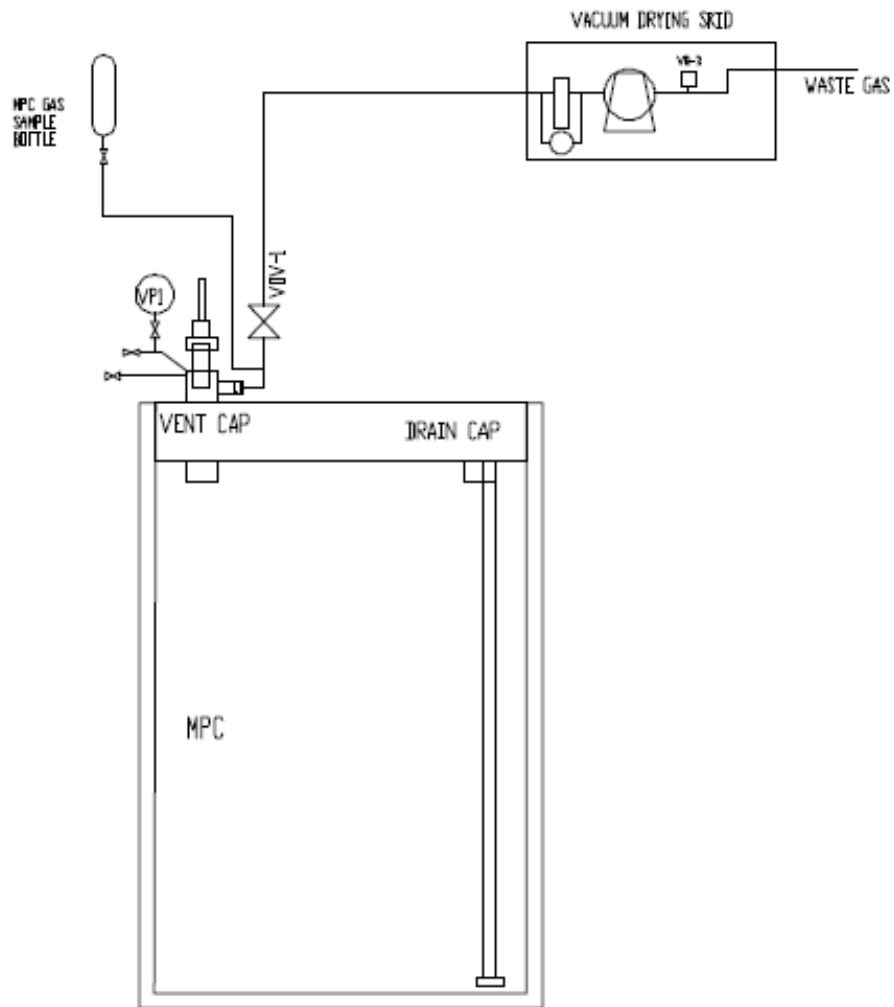
**Figure 8.3.2b: Major HI-STORM 100 Unloading Operations**



**Figure 8.3.2c: Major HI-STORM 100 Unloading Operations**



**Figure 8.3.2d: Major HI-STORM 100 Unloading Operations**



**Figure 8.3.3: MPC Gas Sampling in Preparation for Unloading**

**Figure 8.3.4: Deleted**

## 8.4 MPC TRANSFER TO A HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE

### 8.4.1 Overview of Operations

The MPC is recovered from storage and transferred into HI-TRAC using the same or similar method as described in Section 8.3. Once the MPC is inside HI-TRAC, the HI-STAR 100 is brought to the transfer location and positioned for receiving of the MPC. If used, the Temporary Shield Ring is installed and filled with water and the Transfer Collar is installed on the HI-STAR 100 Overpack. The Temporary Shield Ring reduces operator dose rates during MPC transfer operations. The Transfer Collar or mating device adapts the top surface of the HI-STAR 100 Overpack to mate with the bottom of HI-TRAC. The MPC may be lowered using the MPC Downloader, the main crane hook or similar device. The MPC slings and MPC lift cleats are attached to the MPC. The MPC is raised slightly, the transfer lid door (or mating device drawer) locking pins are removed and the doors are (drawer is) opened. The MPC is lowered into the HI-STAR. Following verification that the MPC is fully lowered, the MPC slings are disconnected and lowered onto the MPC lid. HI-TRAC is removed from on top of the HI-STAR 100 Overpack. The MPC lift cleat, slings, and the transfer collar/mating device are removed. Hole plugs are installed in the empty MPC lid bolt holes. The HI-STAR 100 Overpack is prepared for storage or transport in accordance with the Certificate of Compliance for storage or transport, as applicable.

### 8.4.2 Recovery from Storage

**Caution:**

Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.

1. Recover the MPC from storage and position it inside of HI-TRAC in accordance with Section 8.3.2.
2. Deleted.

### 8.4.3 MPC Transfer into the HI-STAR 100 Overpack

**Note:**

The following steps outline the HI-STAR 100 operating steps. Refer to the HI-STAR 100 System Final Safety Analysis Report (Docket No. 72-1008) and the HI-STAR 100 System Safety Analysis Report (Docket No. 71-9261) for HI-STAR 100 Overpack specific operations.

1. If necessary, remove the HI-STAR 100 closure plate and the removable shear ring segments. Perform a radiological survey of the inside of the HI-STAR 100 Overpack to verify there is no residual contamination from previous uses. If contamination levels are above specified limits, the HI-STAR 100 Overpack shall be decontaminated appropriately prior to use.
2. Discard any used metallic seals.
3. Perform a HI-STAR 100 receipt inspection in accordance with site-specific procedures.
4. Install the temporary shield ring on HI-STAR 100 and fill it with water, if used. See Figure 8.1.18.
5. Install the HI-STAR transfer collar (or mating device with the pool lid for HI-TRAC 100D and 125D). See Figure 8.4.1a (or Figure 8.4.1b).
6. Position HI-STAR adjacent to HI-TRAC.

**Note:**

Lifting of the loaded HI-TRAC shall be performed in accordance with the applicable lifting requirements.

7. Raise and align HI-TRAC over HI-STAR and mate the overpacks.

**Note:**

The MPC lift cleats and MPC slings are still installed from the previous operation.

8. Deleted.
9. Remove the transfer lid door (mating device drawer) locking pins and open the doors (drawer).

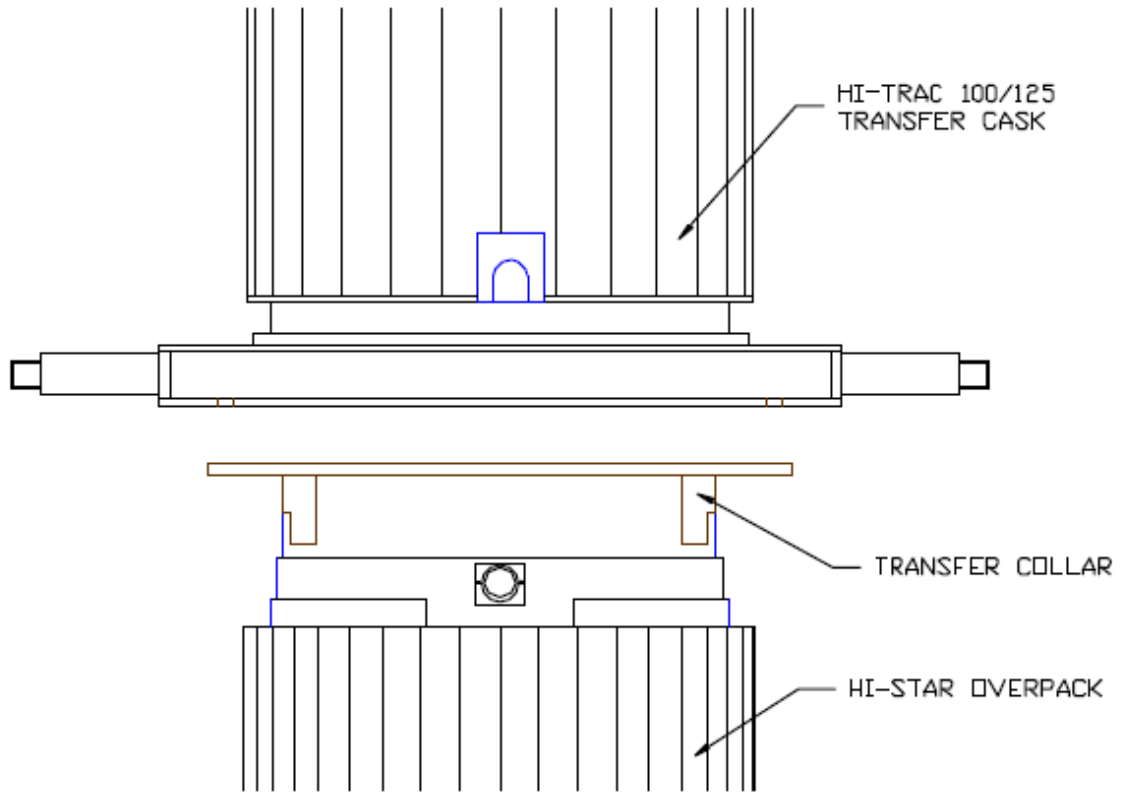
**ALARA Warning:**

If trim plates are not used, personnel should remain clear of the immediate door/drawer area during MPC downloading since there may be radiation streaming during MPC raising and lowering operations.

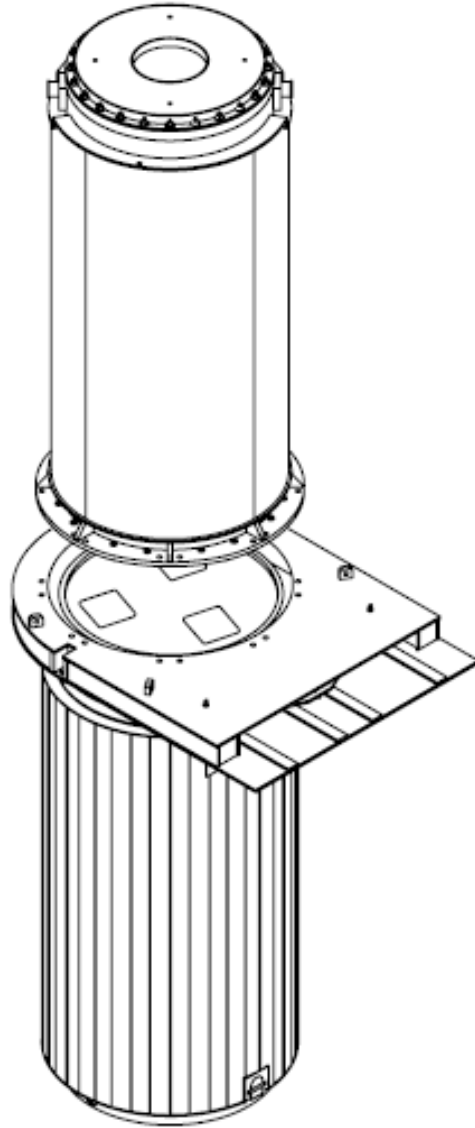
10. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
11. Lower the MPC into HI-STAR.
12. When the MPC is fully seated, disconnect the slings from the MPC lifting device and lower them on to the MPC lid.
13. Remove HI-TRAC from on top of HI-STAR 100 Overpack.
14. Remove the MPC lift cleat from the MPC and install hole plugs in the empty bolt holes. See Table 8.1.5 for torque requirements.
15. Remove the HI-STAR 100 transfer collar or mating device.
16. Drain and remove the temporary shield ring (if used) and store it in an approved plant storage location.

17. Complete HI-STAR preparation for transport in accordance with the HI-STAR 100 Safety Analysis Report (Docket 71-9261) and the Certificate of Compliance, or complete HI-STAR preparation for storage in accordance with the HI-STAR 100 Final Safety Analysis Report (Docket 72-1008) and the Certificate of Compliance, as applicable.





**Figure 8.4.1a: HI-STAR and HI-TRAC Mating**



**Figure 8.4.1b: HI-STAR and HI-TRAC 100D/125D Mating**

## 8.5 MPC TRANSFER INTO THE HI-STORM 100 OVERPACK DIRECTLY FROM TRANSPORT

### 8.5.1 Overview of Operations

HI-STAR 100 Dual-Purpose Cask System arrives at the receiving location and is surveyed for dose rates and contamination levels. The receiver reviews the shipping paperwork to ensure that the HI-STAR 100 Overpack met the internal contamination limits prior to transportation. The personnel barrier is removed, the impact limiters are removed, the tie-down is removed, and the HI-STAR 100 Overpack is upended. The HI-STAR 100 Overpack is positioned at the designated transfer area and the temporary shield ring is installed. The temporary shield ring reduces operator dose rates during MPC transfer operations. A gas sample is drawn from the annulus and analyzed. The gas sample provides an indication of MPC closure performance. The annulus is depressurized and the closure plate is removed. The transfer collar (mating device with pool lid for HI-TRAC 100 D and 125D) is installed and the MPC lift cleats are attached to the MPC. The transfer collar (mating device) is used to provide the mating surface on top of the HI-STAR 100 Overpack. The MPC slings are attached to the MPC lift cleat.

If the HI-TRAC 100D and 125D are not used, the HI-TRAC is configured with the transfer lid. The top lid<sup>5</sup> is installed, if necessary. HI-TRAC is raised and positioned on top of HI-STAR. The MPC slings are attached to the lifting device. The MPC is raised into HI-TRAC. The HI-TRAC doors/(mating device drawer) are closed and the locking pins are installed. For the HI-TRAC 100D and 125D, the pool lid is bolted on. HI-TRAC is raised and the HI-STAR 100 Overpack is removed from under HI-TRAC. The HI-STAR 100 Overpack is repositioned at the user's discretion.

HI-STORM is positioned for MPC receipt with the lid removed, the *alignment device (or mating device) positioned*, and the vent duct shield inserts installed in the exit vent ducts. HI-TRAC is raised and positioned on top of HI-STORM. For HI-TRAC 100D and 125D, the pool lid is unbolted. The locking pins are removed and the doors are opened (or mating device drawer opened for HI-TRAC 100D and 125D). The MPC is lowered into HI-STORM. The MPC slings are disconnected and lowered onto the MPC lid. HI-TRAC is raised and positioned at the site's discretion. The MPC lift cleat, slings, vent duct shield inserts, and transfer collar (or mating device) are removed and hole plugs are installed in the empty bolt holes. HI-STORM is prepared for storage and transferred to the ISFSI pad in the same manner as described in Section 8.1.

<sup>5</sup> Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

8.5.2 HI-STAR 100 SYSTEM Receipt and Preparation for MPC Transfer**Note:**

The following provides a general description of the HI-STAR 100 System operations. Refer to the HI-STAR 100 System Topical Safety Analysis Report (Docket 72-1008) and the Safety Analysis Report (Docket 71-9261) for HI-STAR-specific operations.

1. Review the shipping paperwork and verify that the HI-STAR 100 Overpack met the required internal contamination limits prior to transportation.
2. Measure the HI-STAR 100 dose rates in accordance with 10CFR20 [8.5.1].
3. Remove the personnel barrier.
4. Perform removable contamination surveys in accordance with 10CFR20 [8.5.1].
5. Remove the impact limiters.
6. Remove the tie-down.
7. Perform a visual inspection of the overpack for obvious signs of shipping damage.
8. Remove the removable shear ring segments from the overpack. (Approximate weight is 50 lbs each).
9. Transfer the HI-STAR 100 Overpack to the location for MPC transfer and position it vertically.
10. Install the temporary shield ring on the overpack top flange if used.

**ALARA Warning:**

Gas sampling is performed to assess the condition of the MPC confinement boundary. If a leak is discovered in the MPC boundary, the MPC may not be placed into HI-STORM. If no leak is detected, the annulus may be vented directly.

11. Perform gas sampling as follows:
  - a. Remove the overpack vent port cover plate and attach the backfill tool with a sample bottle attached. See Figure 8.5.1. Store the cover plate in a site-approved location.
  - b. Using a vacuum pump, evacuate the sample bottle and backfill tool.
  - c. Slowly open the vent port plug and gather a gas sample from the annulus. Reinstall the overpack vent port plug.
  - d. Evaluate the gas sample and determine the condition of the MPC confinement boundary.

12. If the confinement boundary is intact (i.e., no radioactive gas is measured) then vent the overpack annulus by removing the overpack vent port seal plug (using the backfill tool). Otherwise return the HI-STAR 100 to the spent fuel pool for MPC unloading in accordance with the HI-STAR 100 SAR.
13. Remove the closure plate bolts and remove the overpack closure plate. Store the closure plate on cribbing to protect the seal seating surfaces. Store the closure plate bolts in a site-approved location.
14. Install the HI-STAR 100 Seal Surface Protector.
15. Install the transfer collar (or mating device with pool lid for HI-TRAC 100D and 125D) on HI-STAR. See Figure 8.4.1.

**Note:**

The location of MPC transfer may be selected at the user's discretion.

16. Remove the MPC lift cleat hole plugs and install the MPC lift cleats. See Figure 8.1.24. See Table 8.1.5 for torque requirements.
17. Attach the MPC slings to the MPC lift cleat and lay them on the MPC lid.

**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) for the HI-TRAC top lid, the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

18. If necessary, install the HI-TRAC top lid. See Figure 8.1.9. See Table 8.1.5 for torque requirements.
19. If necessary, configure HI-TRAC with the transfer lid as follows (Not applicable for HI-TRAC 100D and 125D):

**ALARA Note:**

The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

- a. Position HI-TRAC vertically adjacent to the transfer lid.
- b. Remove the pool lid bolts and plates and store them in an approved plant storage location.
- c. Raise the empty HI-TRAC and position it on top of the transfer lid.
- d. Install the bottom lid bolts. See Table 8.1.5 for torque requirements.

20. Position HI-TRAC adjacent to HI-STAR.
21. Raise HI-TRAC above HI-STAR.
22. Align HI-TRAC over HI-STAR 100 and mate the overpacks. See Figure 8.1.31.
23. Remove the locking pins and open the doors or mating device drawer.

**ALARA Warning:**

If trim plates are not being used, personnel should remain clear of the door/drawer area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

24. At the users discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using clamps or any other method deemed suitable by the user. See Figure 8.1.33.

8.5.3 Perform MPC Transfer into HI-STORM 100

**Caution:**

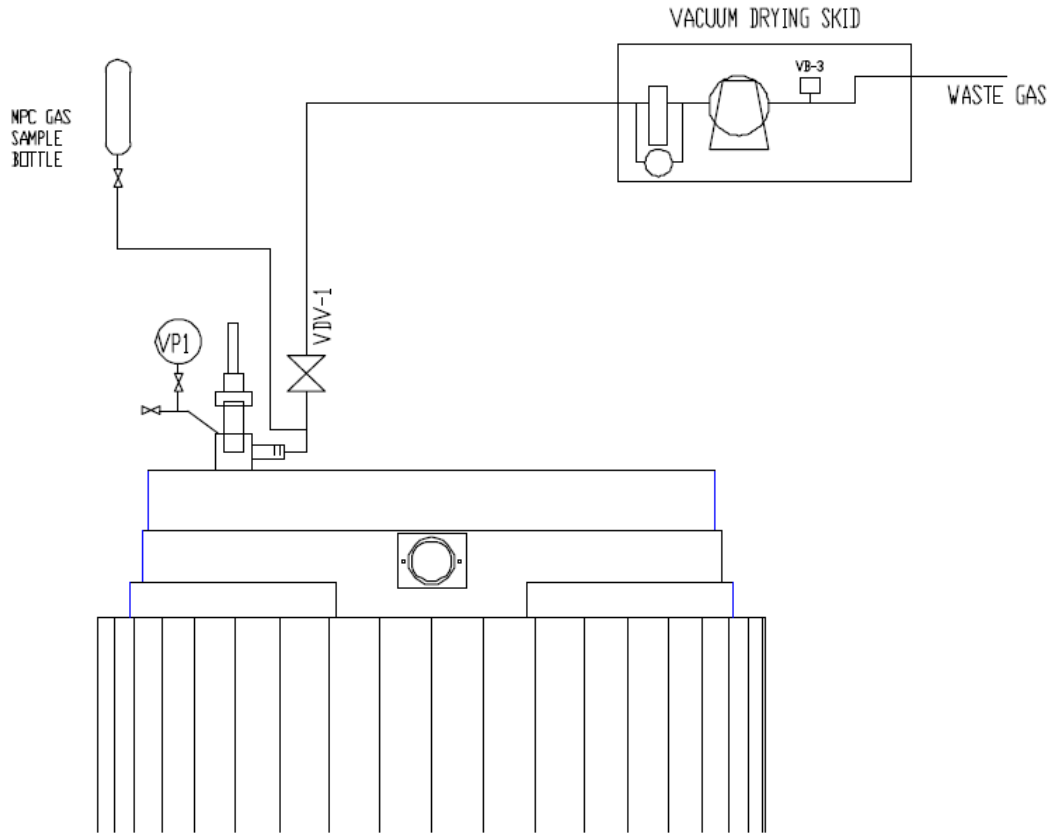
Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.

1. Raise the MPC into HI-TRAC by extending the MPC downloader.
2. Verify the MPC is in the full-up position.
3. Remove the trim plates (if used).
4. Close the HI-TRAC doors/drawer and install the locking pins.
5. For the HI-TRAC 100D and 125D, raise the pool lid and bolt it onto the HI-TRAC.
6. Raise HI-TRAC and remove the HI-STAR 100 Overpack from the operations area.
7. Transfer the MPC into HI-STORM in accordance with the steps provided in Section 8.1.

**Warning:**

HI-STORM dose rates are measured to ensure they are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

8. Perform the HI-STORM surface dose rate measurements in accordance with the technical specifications. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10 CFR 72.104(a).
9. Place HI-STORM in storage in accordance with the steps provided in Section 8.1.
10. Perform an air temperature rise test per Step 8.1.7.23 if required.



**Figure 8.5.1: HI-STAR Annulus Gas Sampling**



## 8.6 REFERENCES

- [8.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.
- [8.1.1] U.S. Code of Federal Regulations, Title 10 "Energy", Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,"
- [8.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.
- [8.1.3] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code".
- [8.5.1] U.S. Code of Federal Regulations, Title 10 " Energy", Part 20, "Standards for Protection Against Radiation,"

## SUPPLEMENT 8.I

### OPERATING PROCEDURES FOR THE HI-STORM 100U SYSTEM

#### 8.I.0 INTRODUCTION

The operations associated with the use of the HI-STORM 100U System, described in Supplement 1.I, are quite similar to the operations for all other variations of the HI-STORM 100 System. The following sections describe those operations that are, in any respect, unique to the HI-STORM 100U System and thus supplement the information presented in Chapter 8. Where practical, the section number used below directly references the corresponding section in Chapter 8. For example, Subsection 8.I.1.6 supplements or replaces the operations described in Subsection 8.1.6. The guidance provided in this supplement shall be used to develop the site-specific loading procedures for the HI-STORM 100U System, as described in the main body of Chapter 8.

#### 8.I.1 PROCEDURE FOR LOADING THE HI-STORM 100U SYSTEM IN THE SPENT FUEL POOL

##### 8.I.1.1 Overview of Loading Operations

The HI-STORM 100U System differs from the other variations of the HI-STORM 100 System in that the vertical ventilated module (VVM) is an integral part of the ISFSI and cannot be transported on site. The steps required to prepare and load the HI-STORM 100 System up to the point of MPC transfer to the HI-STORM VVM are described in Sections 8.1.2 to 8.1.5 of Chapter 8. For the HI-STORM 100U System, the MPC transfer is performed at the ISFSI after preparation of the MPC for storage operations within the HI-TRAC transfer cask. The loaded HI-TRAC transfer cask may be transported between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or other load handling equipment designed for such applications. The operational steps required to prepare, load the MPC, and transport it to the ISFSI using the HI-TRAC transfer cask are the same for both HI-STORM 100U and the aboveground HI-STORM for the case wherein the MPC transfer for long-term storage occurs at the ISFSI. The detailed operational steps presented in this supplement, therefore, start with the preparation and loading at the ISFSI.

Prior to MPC transfer at the ISFSI, the VVM lid is removed and the empty storage module is inspected. The mating device is positioned on top of the VVM. If used, the Supplemental Cooling System (SCS) is disconnected from the HI-TRAC, the HI-TRAC annulus is drained and the HI-TRAC transfer cask is placed on top of the mating device (Figure 8.I.1). The MPC may be downloaded using the vertical cask crawler, the MPC downloader attached to the transfer cask, or other suitable lifting device. The MPC lifting device is attached to the MPC. The pool lid is removed and the mating device drawer is opened. Optional temporary shielding may be installed, as guided by the licensee's radiation protection program, on or around the mating device. The MPC is lowered into the VVM (Figure 8.I.2). Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto or removed from the MPC lid. The temporary shielding is removed, if necessary. The HI-TRAC and pool lid may be reattached while on the VVM or may be removed and then reattached. HI-

TRAC is removed from on top of the VVM. The MPC lift cleats are removed. Plugs are installed in the empty MPC lifting holes to fill the voids left by the removal of the lift cleat studs. The mating device is removed and the VVM lid is installed. Finally, the temperature monitoring elements and their instrument connections, if used at the ISFSI, are installed, and post-loading performance verification is performed, as required.

#### 8.I.1.2 HI-STORM 100U Receiving Operations

Because the HI-STORM 100U VVM is cast as part of the ISFSI itself, there is no handling of the VVM. For an example of the rigging required to handle the HI-STORM 100U VVM Closure Lid, see Figure 8.I.3.

#### 8.I.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.3.

#### 8.I.1.4 MPC Fuel Loading

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.4.

#### 8.I.1.5 MPC Closure

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.5.

#### 8.I.1.6 Preparation for Storage

For the HI-STORM 100U System, these activities are essentially identical to those described in Subsection 8.1.6. However, for the HI-STORM 100U VVM, the receipt and cleanliness inspection should be carried out in accordance with a CoC Holder-approved inspection checklist and Table 8.I.1.

#### 8.I.1.7 Placement of MPC/HI-STORM 100U VVM into Storage

**Note:**

Because the HI-STORM 100U VVM is an integral part of the ISFSI, MPC transfer must take place at the ISFSI.

1. Perform a transport route walkdown to ensure that the cask transport conditions are met.
2. Remove the outlet vent cover if necessary. Remove the VVM lid. See Figure 8.I.3 for a lid rigging example and Subsection 3.I.2 for the bounding weight of the lid.

3. Inspect all vent and cavity locations in the VVM cavity for foreign objects. Remove any foreign objects.
4. Install the mating device on the VVM.
5. Transport the HI-TRAC transfer cask to the ISFSI location using a vertical cask crawler or other suitable transportation device.

**Caution:**

For MPCs with heat loads requiring supplemental cooling, all preparatory work related to the transfer should be completed prior to terminating supplemental cooling operations to prevent fuel cladding temperatures from exceeding the limits set forth in Chapter 2 of this FSAR.

6. If used, discontinue the supplemental cooling operations and disconnect the SCS. Drain the water from the HI-TRAC annulus to an appropriate plant discharge point.
7. Position the HI-TRAC transfer cask above the VVM.
8. Align HI-TRAC over the VVM and mate the casks using the mating device. See Figure 8.I.1.
9. Attach the MPC slings to the MPC lift cleats and the cask transporter or other suitable downloading device. See Figure 8.I.2.
10. Raise the MPC slightly to remove the weight of the MPC from the HI-TRAC pool lid.
11. Unbolt the pool lid from the HI-TRAC and lower the lid into the mating device.
12. Open the mating device drawer.

**ALARA Warning:**

Temporary shielding may be used to reduce personnel dose during transfer operations. If ALARA considerations dictate that temporary shielding not be used, personnel must remain clear of the immediate area around the mating device drawer during MPC downloading.

13. At the user's discretion, install temporary shielding to cover the gap above and below the mating device drawer.
14. Lower the MPC into the VVM.
15. Verify that the MPC is fully seated in the VVM
16. Disconnect the MPC slings from the downloading device and lower them onto the MPC lid or remove them from the MPC.
17. Remove the temporary shielding and close the mating device drawer.

**Note:**

The HI-TRAC pool lid may be reattached while positioned on the VVM or following HI-TRAC and mating device removal from the VVM.

18. Bolt the pool lid back onto the HI-TRAC.

**ALARA Warning:**

Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed to comply with ALARA requirements.

19. Remove the HI-TRAC transfer cask from on top of the VVM.
20. Open the mating device drawer and remove the MPC lift cleats and, if installed, the MPC slings.
21. Install hole plugs in the empty MPC bolt holes.
22. Close the mating device drawer and remove the mating device from on top of the VVM.

**Warning:**

Unless the lift is single failure proof (or equivalent safety factor) for the VVM lid, the lid shall be preferably kept less than 2 feet above the top surface of the VVM while over the MPC. This lift limit action is purely a defense-in-depth measure: the Closure Lid cannot fall and impact the MPC, as discussed in Supplement 1.I.

23. Install the VVM lid. See Figure 8.I.3 for lid rigging and Subsection 3.I.2 for the bounding lid weight.
24. Remove the VVM lid rigging equipment and re-install the outlet vent cover (if previously removed).
25. Install the HI-STORM temperature monitoring elements (if used).
26. Perform shielding effectiveness testing, as required.

## 8.I.2 ISFSI OPERATIONS

ISFSI operations for the HI-STORM 100U System are identical to the activities identified in Section 8.2. Additional operations related to the HI-STORM 100U System would include any periodic surveillance and maintenance activities required for the site specific cathodic protection system.

### 8.I.3 PROCEDURE FOR UNLOADING THE HI-STORM 100U SYSTEM IN THE SPENT FUEL POOL

#### 8.I.3.1 Overview of HI-STORM 100U System Unloading Operations

The MPC is recovered from the HI-STORM 100U VVM at the ISFSI using the same methodologies as described in Subsection 8.I.1, except that the order is basically reversed. The VVM temperature monitoring elements (if used) and lid are removed. The mating device is installed and the mating device drawer is opened. The MPC lift cleats are attached to the MPC. The MPC slings are attached to the MPC lift cleats and positioned on the MPC lid. The mating device drawer is closed and the HI-TRAC is positioned on top of the mating device and VVM. The pool lid is unbolted from the HI-TRAC and the mating device drawer is opened. The MPC slings are brought through the HI-TRAC top lid and connected to the lift device. The MPC is raised into HI-TRAC and the mating device drawer is closed. The pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of the VVM and transported to the preparation area. The remainder of the unloading operation is carried out in accordance with the operations for the standard HI-STORM 100 System (See Subsection 8.3.3).

### 8.I.3.2 Preparing HI-STORM 100U System and HI-TRAC for Recovery from Storage

1. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met for transporting the loaded HI-TRAC transfer cask.
2. Perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a written inspection checklist. Transport the HI-TRAC to the ISFSI using the cask transporter or other suitable device.
3. Remove the VVM temperature monitoring equipment (if used).
4. Remove the outlet vent cover if necessary. Remove the VVM lid, preferably keeping its height above the top of the CEC Flange to under 2 feet. See Figure 8.I.3 for a rigging example and Subsection 3.I.2 for the bounding lid weight.
5. Install the mating device on the VVM.
6. Open the mating device drawer.
7. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings on the MPC lid. See Table 8.1.5 for torque requirements.
8. Close the mating device drawer.
9. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for example rigging and Table 8.1.5 for torque requirements.
10. If previously drained, fill the neutron shield jacket with plant demineralized water or an approved antifreeze solution as necessary. Ensure that the fill and drain plugs are installed.
11. Align HI-TRAC over the mating device and VVM and mate the casks. See Figure 8.I.2.
12. Unbolt the pool lid from the HI-TRAC and lower into the mating device.
13. Open the mating device drawer.
14. At the user's discretion, install temporary shielding to cover the gap above and below the mating device drawer.
15. Raise the MPC slings up through the HI-TRAC and attach them to the lifting device. See Figure 8.I.1.

**ALARA Warning:**

If temporary shielding is not used, personnel should remain clear of the immediate drawer area during MPC downloading.

**Caution:**

Limitations for the handling of the loaded MPC in HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Chapter 4 for requirements on the use of the Supplemental Cooling System (SCS) to maintain fuel cladding temperatures below limits when MPCs with greater than threshold limits are transported in the HI-TRAC. Operation of the SCS normally continues until MPC cool-down and re-flooding operations have commenced. Staging and check-out of the SCS shall be completed prior to transferring the MPC to the HI-TRAC to minimize the time required to begin operations.

16. Raise the MPC into HI-TRAC.
17. Verify the MPC is in the full-up position.
18. Close the mating device drawer.
19. Bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.
20. Lower the MPC onto the pool lid.
21. Disconnect the slings from the lifting device and the MPC lift cleats.

**Note:**

Operation of the SCS will need to be postponed until the pool lid is in place on the HI-TRAC. Supplemental cooling shall begin, if required, by the provision in Chapter 4.

**Warning:**

At the start of SCS operations, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of filling the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased.

22. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1). Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.
23. Remove HI-TRAC from the top of the VVM.
24. Transport the HI-TRAC to the designated preparation area using the cask transporter or other suitable device.
25. Install the VVM lid and outlet vent cover (if previously removed) to prevent entry of foreign objects into the VVM.

### 8.I.3.3 Preparation for Unloading

Perform the balance of the unloading operations in accordance with Subsection 8.3.3 beginning at Step 2.

### 8.I.3.4 MPC Unloading

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.3.4.

8.I.3.5 Post-Unloading Operations

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.3.5.

8.I.4 MPC TRANSFER TO A HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE

For the HI-STORM 100U System, these activities are identical to those described in Section 8.4. MPC transfer to the HI-TRAC from the HI-STORM 100U VVM is addressed in Subsection 8.I.3 above as noted in Section 8.3.

8.I.5 MPC TRANSFER INTO THE HI-STORM 100U VVM DIRECTLY FROM TRANSPORT

For the HI-STORM 100U System, these activities are identical to those described in Section 8.5. MPC transfer to the HI-STORM 100U VVM from the HI-TRAC is addressed in Subsection 8.I.1 above as noted in Section 8.1.



Table 8.I.1  
HI-STORM 100U VVM INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100U VVM. Specific findings shall be brought to the attention of the project management for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100U VVM Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
3. The lid shall be inspected for the presence or availability of studs, nuts, and hole plugs.
4. Lid lifting points shall be inspected for dirt, debris, and general condition.
5. Vent openings shall be free from obstructions.
6. Vent screens shall be available, intact, and free of holes and tears.
7. Temperature monitoring elements, if used, shall be inspected for availability, function, calibration and provisions for mounting to the VVM outlet air passage.

HI-STORM 100U VVM Main Body:

1. Cooling passages shall be free from obstructions.
2. The interior cavity shall be free of debris, litter, tools, and equipment.
3. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.

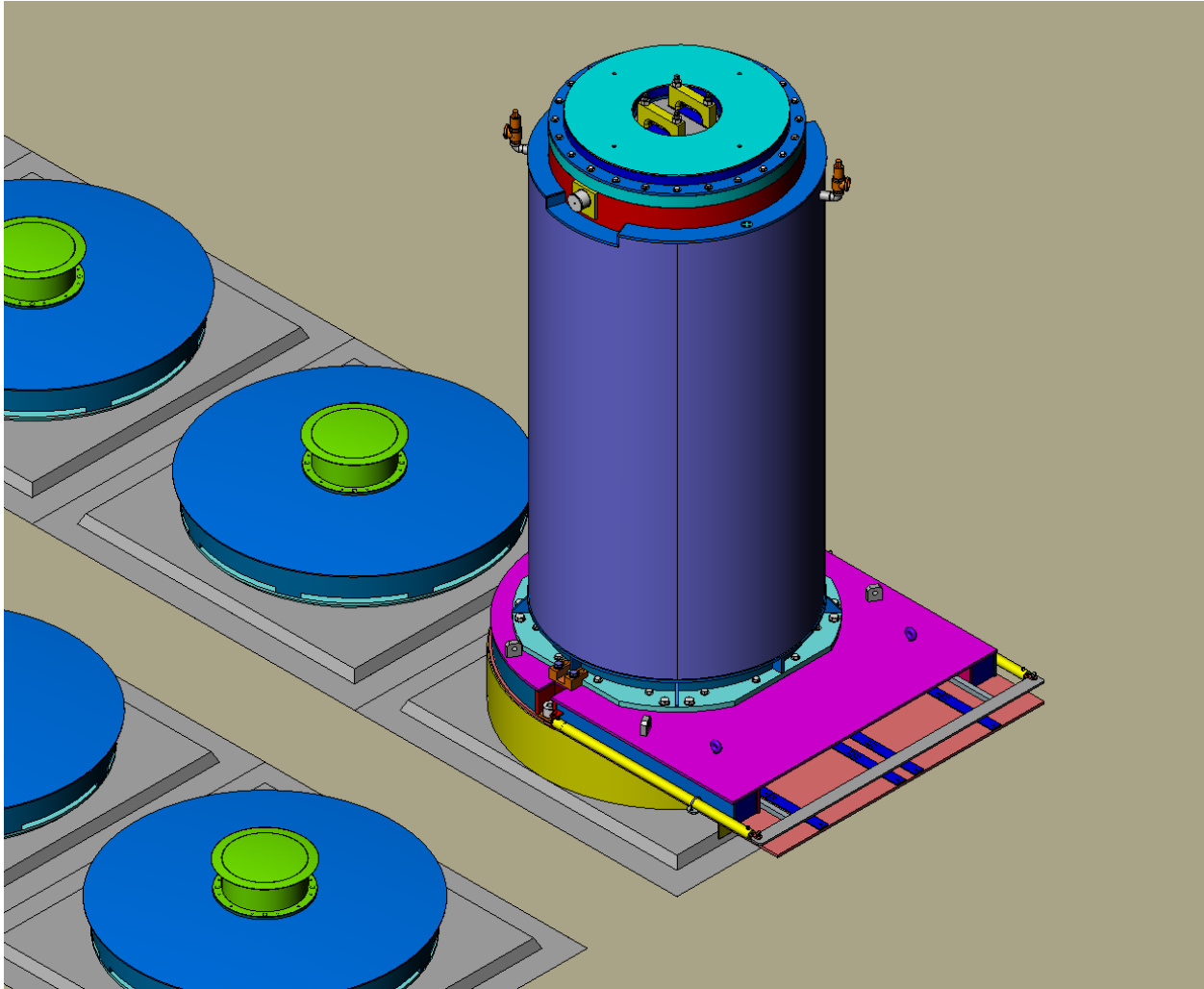


FIGURE 8.I.1: HI-TRAC ALIGNMENT AND PLACEMENT ON MATING DEVICE AND HI-STORM 100U\* VVM

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

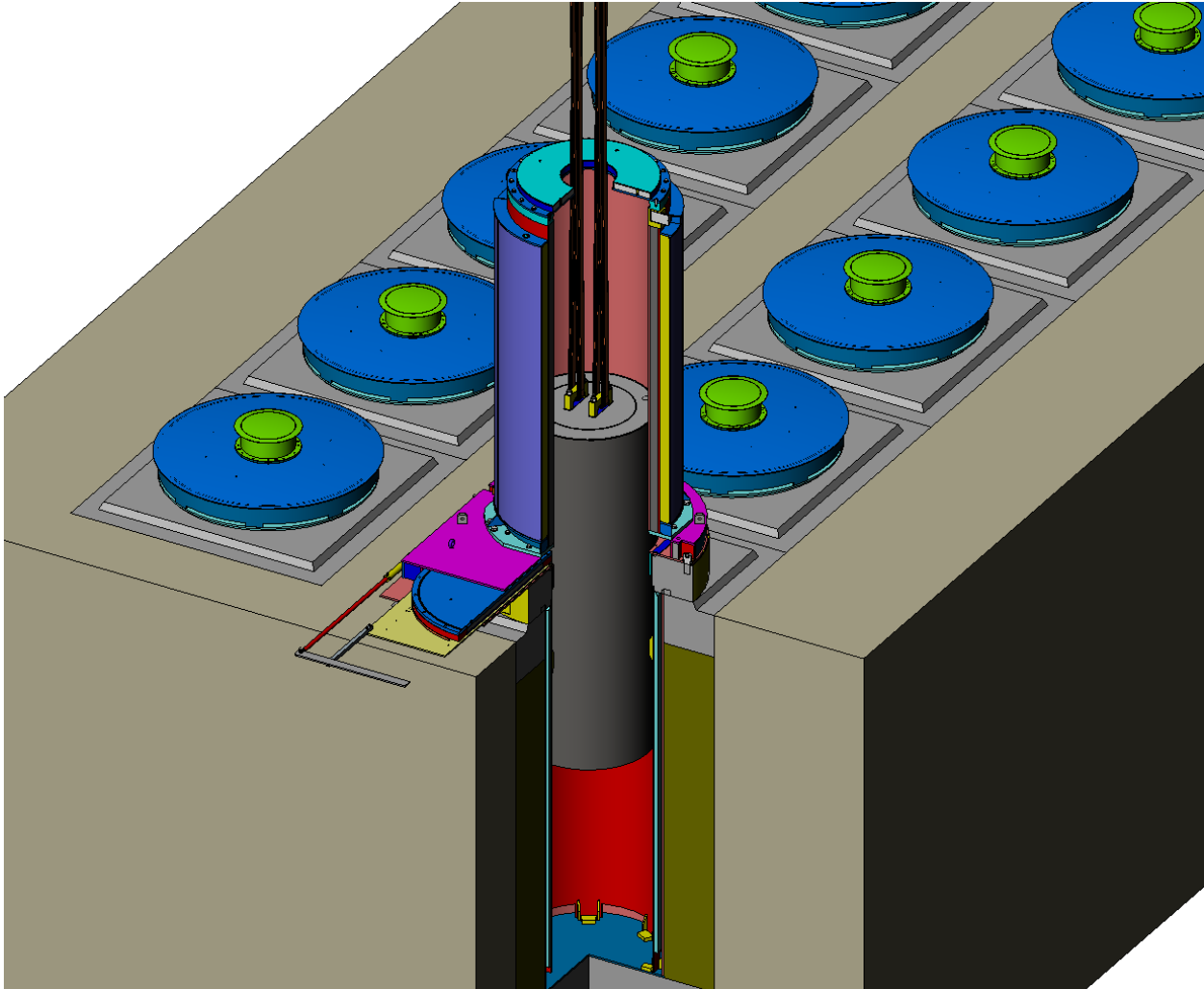


FIGURE 8.I.2: DOWNLOADING MPC INTO HI-STORM 100U\* VVM

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

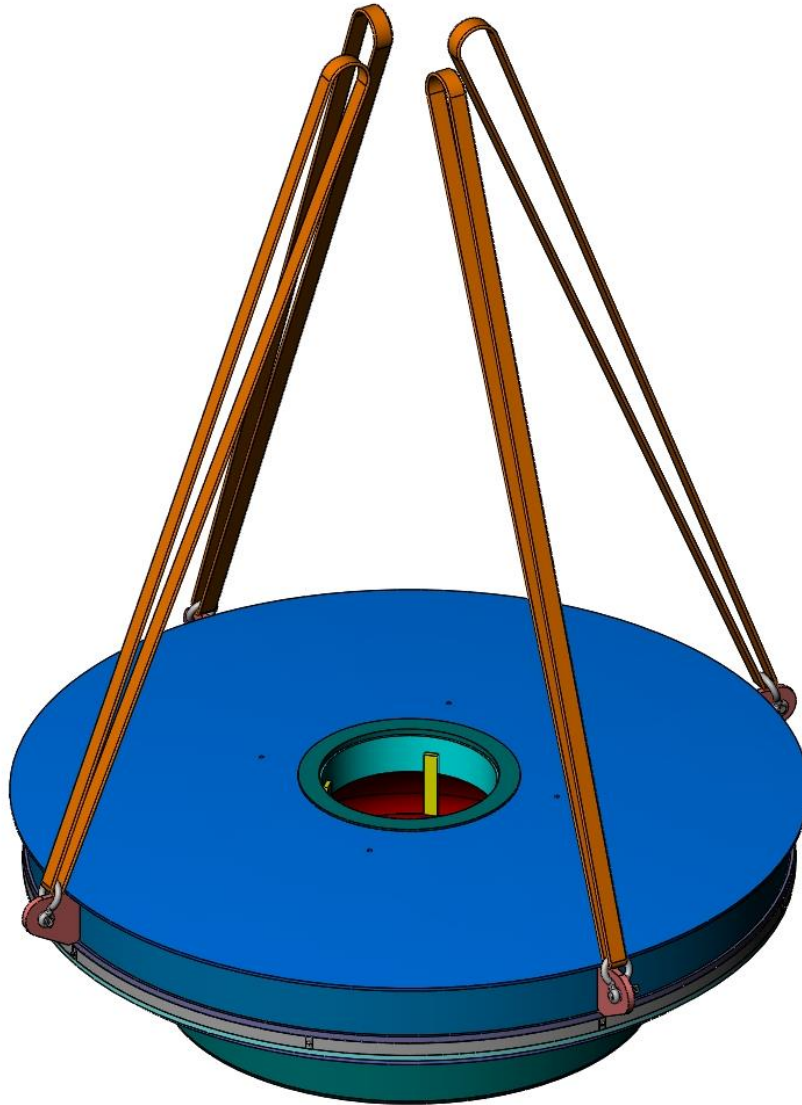


FIGURE 8.I.3: EXAMPLE RIGGING CONFIGURATION FOR THE HI-STORM 100U\* VVM LID

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

## SUPPLEMENT 8.III

### OPERATING PROCEDURES FOR THE MPC-68M

#### 8.III.0 INTRODUCTION

This supplement provides clarifications and limitations to the procedures for loading, unloading, and recovery presented in Chapter 8 as applicable to the HI-STORM 100 System with an MPC-68M. The procedure steps outlined in Chapter 8 for loading, unloading, and recovery, remain applicable. Also see Subsection 1.III.2.2 of Supplement 1.III.

#### 8.III.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

A supplemental cooling system is not required when loading an MPC-68M. Thermal analysis results in Supplement 4.III show that for all short-term operations, excluding vacuum drying of high burnup fuel at high heat loads, the fuel cladding temperatures remain below the temperature limits in ISG-11, Rev 3 [2.0.8] for moderate and high burnup fuel.

Thermal analysis results in Supplement 4.III show that during vacuum drying and drying using the forced helium dehydrator (FHD) system the fuel cladding temperatures remain below the temperature limits in ISG-11, Rev 3 for moderate and high burnup fuel under the following conditions.

- When using a vacuum drying system (VDS) with an MPC-68M containing high burnup fuel, the MPC must be loaded uniformly with a maximum allowable decay heat per fuel storage location not to exceed 0.426 kW (total cask heat load  $\leq$  29 kW).
- No limitations apply when using VDS with an MPC-68M containing all moderate burnup fuel up to design basis heat load.
- When using VDS to dry an MPC-68M with moderate or high burn up fuel at any heat load, the annulus space between the HI-TRAC and MPC must be filled with water. Flushing of the annulus space is not required.
- No limitations apply when using a forced helium dehydrator (FHD) system with MPC-68M containing moderate or high burnup fuel up to design basis heat load.

#### 8.III.2 ISFSI OPERATIONS

Same as Section 8.2.

**8.III.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL**

A supplemental cooling system is not required when unloading an MPC-68M. Thermal analysis results in Supplement 4.III show that for all short-term unloading operations the fuel cladding temperatures remain below the temperature limits in ISG-11, Rev 3 for moderate and high burnup fuel.

**8.III.4 MPC TRANSPORT TO A HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE**

Same as Section 8.4.

**8.III.5 MPC TRANSFER INTO THE HI-STORM 100 OVERPACK DIRECTLY FROM TRANSPORT**

Same as Section 8.5.

## CHAPTER 9<sup>†</sup>: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

### 9.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM 100 System, including the HI-TRAC transfer cask to verify that the structures, systems and components (SSCs) classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM 100 System fabricated, assembled, inspected, tested, and accepted for use under the scope of the HI-STORM 100 System CoC, except as noted herein.

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters ensure that the HI-STORM 100 System will maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions; will maintain subcriticality control; will properly transfer the decay heat of the stored radioactive materials; and that radiation doses will meet regulatory requirements.

Both pre-operational and operational tests and inspections are performed throughout HI-STORM 100 System operations to assure that the HI-STORM 100 System is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, pressure tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 8 identifies the tests and inspections. "Pre-operation", as referred to in this section, defines that period of time from receipt inspection of a HI-STORM 100 System until the empty MPC is loaded into a HI-TRAC transfer cask for fuel assembly loading.

The HI-STORM 100 System is classified as important to safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM 100 System shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance with a quality program commensurate with the particular SSC's graded quality category. Tables 2.2.6 and 8.1.6 provide the quality category for each major item or component of the HI-STORM 100 System and its ancillary equipment, respectively.

The acceptance criteria and maintenance program described in this chapter fully comply with the requirements of 10CFR72 [9.0.1] and NUREG-1536 [9.0.2], except as clarified in Table 1.0.2.

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

## 9.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM 100 System prior to and during loading of the system. These inspections and tests provide assurance that the HI-STORM 100 System has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the requirements of 10CFR72 [9.0.1].

Identification and resolution of noncompliances shall be performed in accordance with the Holtec International Quality Assurance Program as described in Chapter 13 of this FSAR, or the licensee's NRC-approved Quality Assurance Program.

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM 100 overpack, and the 100-ton HI-TRAC and 125-ton HI-TRAC transfer casks are listed in Tables 9.1.1, 9.1.2, and 9.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 8 and 12 provide operating guidance and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM 100 System has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR.

This section summarizes the test program required for the HI-STORM 100 System.

### 9.1.1 Fabrication and Nondestructive Examination (NDE)

The design, fabrication, inspection, and testing of the HI-STORM 100 System is performed in accordance with the applicable codes and standards specified in Tables 2.2.6 and 2.2.7 and on the Design Drawings. Additional details on specific codes used are provided below.

The following fabrication controls and required inspections shall be performed on the HI-STORM 100 System, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

1. Materials of construction specified for the HI-STORM 100 System are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by ASME Code [9.1.1] Section II (when applicable); the requirements of ASME Section III (when applicable); Holtec procurement specifications; and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to assure material traceability is maintained throughout fabrication. Materials for the confinement boundary (MPC baseplate, lid, closure ring, port cover plates and shell) shall also be inspected per the requirements of ASME Section III, Article NB-2500.
2. The MPC confinement boundary shall be fabricated and inspected in accordance



with ASME Code, Section III, Subsection NB, with alternatives as noted below. The MPC basket and basket supports (or basket shims) shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NG, with alternatives as noted below. Metal components of the HI-TRAC transfer cask and the HI-STORM overpack, as applicable, shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NF, Class 3 or AWS D1.1, as shown on the design drawings, with alternatives as noted below.

NOTE: NRC-approved alternatives to these Code requirements are discussed in FSAR Section 2.2.4.

3. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable ASME Section III Subsections (e.g., NB, NG, or NF, as applicable to the SSC). AWS code welding may be performed using welders and weld procedures that have been qualified in accordance with applicable AWS requirements or in accordance with ASME Code Section IX
4. Code welds shall be visually examined in accordance with ASME Code, Section V, Article 9 with acceptance criteria per ASME Code, Section III, Subsection NF, Article NF-5360, except the MPC fuel basket cell plate-to-cell plate welds, fuel basket support-to-canister welds, and basket shims-to-MPC fuel basket welds which shall have acceptance criteria to ASME Code Section III, Subsection NG, Article NG-5360, (as modified by the design drawings). Table 9.1.4 identifies additional nondestructive examination (NDE) requirements to be performed on specific welds, and the applicable codes and acceptance criteria to be used in order to meet the inspection requirements of the applicable ASME Code, Section III. Acceptance criteria for NDE shall be in accordance with the applicable Code for which the item was fabricated. These additional NDE criteria are also specified on the design drawings for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be reviewed and approved by Holtec in accordance with its QA program. NDE inspections of code welds shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [9.1.2] or other site-specific, NRC-approved program for personnel qualification. Non-code welds shall be examined and repaired in accordance with written and approved procedures.

5. The MPC confinement boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, UT, MT and/or PT, as applicable) to verify that it is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its confinement effectiveness.
6. ASME Code welds requiring weld repair shall be repaired in accordance with the requirements of the ASME Code, Section III, Article NB-4450, NG-4450, or NF-4450, as applicable to the SSC, and examined after repair in the same manner as the original weld.
7. Base metal repairs shall be performed and examined in accordance with the applicable fabrication Code.
8. Grinding and machining operations on the MPC confinement boundary shall be controlled through written and approved procedures and quality assurance oversight to ensure grinding and machining operations do not reduce base metal wall thicknesses of the confinement boundary beyond that allowed per the design drawings. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets Design Drawing requirements. A nonconformance shall be written for areas found to be below allowable base metal thickness and shall be evaluated and repaired per the applicable ASME Code, Subsection NB requirements.
9. Dimensional inspections of the HI-STORM 100 System shall be performed in accordance with written and approved procedures in order to verify compliance to design drawings and fit-up of individual components. All dimensional inspections and functional fit-up tests shall be documented.
10. Required inspections shall be documented. The inspection documentation shall become part of the final quality documentation package.
11. The HI-STORM 100 System shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.
12. Each cask shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

13. A documentation package shall be prepared and maintained during fabrication of each HI-STORM 100 System to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM 100 System or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, but not be limited to:
- Completed Shop Weld Records
  - Inspection Records
  - Nonconformance Reports
  - Material Test Reports
  - NDE Reports
  - Dimensional Inspection Report

#### 9.1.1.1 MPC Lid-to-Shell Weld Volumetric Inspection

1. The MPC lid-to-shell (LTS) weld shall be volumetrically or multi-layer liquid penetrant (PT) examined following completion of welding. If volumetric examination is used, the ultrasonic testing (UT) method shall be employed. Ultrasonic techniques (including, as appropriate, Time-of-Flight Diffraction, Focussed Phased Array, and conventional pulse-echo) shall be supplemented, as necessary, to ensure substantially complete coverage of the examination volume.
2. If volumetric examination is used, then a PT examination of the root and final pass of the LTS weld shall also be performed and unacceptable indications shall be documented, repaired and re-examined.
3. If volumetric examination is not used, a multi-layer PT examination shall be employed. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed. The 3/8 inch weld depth corresponds to the maximum allowable flaw size determined in Holtec Position Paper DS-213 [9.1.6].
4. The overall minimum thickness of the LTS weld has been increased by 0.125 inch over the size credited in the structural analyses, to provide additional structural capacity. A 0.625-inch J-groove weld was assumed in structural analyses in Chapter 3.
5. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection results, including relevant findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an

equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The inspection of the weld shall be performed by qualified personnel and shall meet the acceptance requirements of ASME Section III, NB-5350 for PT and NB-5332 for UT.

6. Evaluation of any indications shall include consideration of any active flaw mechanisms. However, cyclic loading on the LTS weld is not significant, so fatigue is not a factor. The LTS weld is protected from the external environment by the closure ring and the root of the LTS weld is dry and inert (He atmosphere), so stress corrosion cracking is not a concern for the LTS weld.
7. The volumetric or multi-layer PT examination of the LTS weld, in conjunction with other examinations and tests performed on this weld (PT of root and final layer, and pressure test); the use of ASME Section III acceptance criteria, and the additional weld material added to account for potential defects in the root pass of the weld, in total, provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The volumetric (or multi-layer PT) examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

#### 9.1.2 Structural and Pressure Tests

##### 9.1.2.1 Lifting Trunnions

Two trunnions (located near the top of the HI-TRAC transfer cask) are provided for vertical lifting and handling. The trunnions are designed in accordance with ANSI N14.6 [9.1.3] using a high-strength and high-ductility material (see Chapter 1). The trunnions contain no welded components. The maximum design lifting load of 250,000 pounds for the HI-TRAC 125 and HI-TRAC 125D and 200,000 pounds for the HI-TRAC 100 and HI-TRAC 100D will occur during the removal of the HI-TRAC from the spent fuel pool after the MPC has been loaded, flooded with water, and the MPC lid is installed. The high-material ductility, absence of materials vulnerable to brittle fracture, large stress margins, and a carefully engineered design to eliminate local stress risers in the highly-stressed regions (during the lift operations) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612 [9.1.4], the acceptance criteria for the lifting trunnions must be established in conjunction with other considerations applicable to heavy load handling.

Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure (including trunnion failure) is one:

- i. operator errors
- ii. rigging failure
- iii. lack of adequate inspection
- iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load (750,000 lbs for the HI-TRAC 125 and HI-TRAC 125D and 600,000 lbs for the HI-TRAC 100 and 100D) shall be applied for a minimum of 10 minutes. The accessible parts of the trunnions (areas outside the HI-TRAC cask), and the adjacent HI-TRAC cask trunnion attachment area shall then be visually examined to verify no deformation, distortion, or cracking occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-TRAC cask trunnion attachment areas shall require replacement of the trunnion and/or repair of the HI-TRAC cask. Following any replacements and/or repair, the load testing shall be performed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing shall be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section II requirements will provide further verification of the trunnion load capabilities. Test results shall be documented. The documentation shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above will provide adequate assurance against handling accidents.

#### 9.1.2.2 Pressure Testing

##### 9.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

All 125-ton and 100-ton HI-TRAC transfer cask water jackets shall be hydrostatically tested to 75 psig +3,-0 psig, and 71 psig +3, -0 psig, respectively, in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall be graduated over a range not less than 1.5 times nor more than 4 times the test pressure. Digital type pressure gages may be used without range

restriction provided the combined error due to calibration and readability does not exceed 1% of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Liquid penetrant (PT) or magnetic particle (MT) examination of accessible welds shall be performed in accordance with ASME Code, Section V, Articles 6 and 7, respectively, with acceptance criteria per ASME Code, Section III, Subsection NF, Articles NF-5350 and NF-5340, respectively. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

If a hydrostatic retest is required and fails, a nonconformance report shall be issued and a root cause evaluation and appropriate corrective actions taken before further repairs and retests are performed.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

#### 9.1.2.2.2 MPC Confinement Boundary

Pressure testing (hydrostatic or pneumatic) of the MPC confinement boundary shall be performed in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 and applicable sub-articles, when field welding of the MPC lid-to-shell weld is completed. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure. The MPC vent and drain ports will be used for pressurizing the MPC cavity. The loading procedures in FSAR Chapter 8 define the test equipment arrangement. The calibrated test pressure gage installed on the MPC confinement boundary shall have an upper limit of approximately twice that of the test pressure. Digital type pressure gages may be used without conforming to the upper limit restriction, provided that the combined error due to calibration and readability does not exceed 1% of the test pressure. Following completion of the required hold period at the test pressure, the surface of the MPC lid-to-shell weld shall be re-examined by liquid penetrant examination in accordance with ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. The performance and sequence of the test is described in FSAR Section 8.1 (loading procedures).

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the

leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

#### 9.1.2.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested include the components identified in Table 3.1.18 and applicable weld materials. Table 3.1.18 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as described in FSAR Appendix 1.D in accordance with written and approved procedures. Testing shall verify the composition, compressive strength, and density meet design requirements.

Concrete testing shall be performed for each lot of concrete. Concrete testing shall comply with Appendix 1.D.

Test results shall be documented and become part of the final quality documentation package.

#### 9.1.3 Leakage Testing

Leakage testing shall be performed in accordance with the requirements of ANSI N14.5 [9.1.5]. Testing shall be performed in accordance with written and approved procedures.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The acceptance criterion is “leaktight” as defined in ANSI N14.5.

The helium leakage test of the vent and drain port cover plate welds shall be performed using a helium mass spectrometer leak detector (MSLD). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criteria is met.

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are not required.

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

Leakage testing of the vent and drain port cover plates shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in FSAR Section 8.1 and the acceptance criteria are defined in the Technical Specifications in Appendix A to CoC 72-1014.

#### 9.1.4 Component Tests

##### 9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no fluid transport devices or rupture discs associated with the HI-STORM 100 System. The only valve-like components in the HI-STORM 100 System are the specially designed caps installed in the MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully-welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and are liquid penetrant examined and leakage tested to verify the MPC confinement boundary.

There are two pressure relief valves installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. These pressure relief valves are provided for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The pressure relief valves shall relieve at 60 psig and 65 psig.

##### 9.1.4.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM 100 System.

##### 9.1.5 Shielding Integrity

The HI-STORM overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM overpack concrete provides both neutron and gamma shielding. Additional neutron shielding is provided by the encased neutron absorber attached to the fuel basket cell surfaces inside the MPCs. The overpack's inner and outer steel shells, and the steel shield shell<sup>†</sup> provide radial gamma shielding. Concrete and steel plates provide axial neutron

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



and gamma shielding. A concrete ring attached to the top of the overpack lid provides additional gamma and neutron shielding in the axial direction. Steel gamma shield cross plates, installed in the overpack air inlet and outlet vents, provide additional shielding for radiation through the vent openings.

The HI-TRAC transfer cask uses three different materials for primary shielding. All HI-TRAC transfer cask designs include a radial steel-lead-steel shield and a steel-lead-steel pool lid design. The top lid in the HI-TRAC 125 and HI-TRAC 125D designs includes Holtite neutron shielding inside a steel enclosure. The HI-TRAC 100 and 100D top lids include only steel shielding. The HI-TRAC 125 transfer lid includes steel, lead, and Holtite, while the HI-TRAC 100 includes only steel and lead. The HI-TRAC 100D and 125D designs do not include a transfer lid. The water jacket, included in all transfer cask designs, provides radial neutron shielding. Testing requirements for the shielding items are described below.

#### 9.1.5.1 Fabrication Testing and Control

##### Holtite-A:

Neutron shield properties of Holtite-A are provided in Chapter 1, Section 1.2.1.3.2. Each manufactured lot of neutron shield material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration and neutron shield density (or specific gravity) meet the requirements specified in Chapter 1 and the Bill-of-Material. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package.

The installation of the neutron shielding material shall be performed in accordance with written and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

##### Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR Appendix 1.D and the design drawings prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 9.1.2.3 for concrete material testing requirements.

##### Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure voids are minimized. The lead shall be tested for chemical composition.

As an alternative to pouring molten lead, the HI-TRAC lead shielding may be installed as pre-cast sections, or lead sheets. If pre-cast sections or sheets are used, the design of the sections and the installations instructions shall minimize the gaps between adjacent lead sections and between the lead and the transfer cask walls to the extent practicable.

Steel:

Steel plates utilized in the construction of the HI-STORM 100 System shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

General Requirements for Shield Materials:

1. Test results shall be documented and become part of the quality documentation package.
2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed during fabrication and again after initial loading operations in accordance with Section 9.1.5.2 below and the operating procedures in Chapter 8.

#### 9.1.5.2 Shielding Effectiveness Tests

The effectiveness of the lead pours in the HI-TRAC transfer cask body shall be verified during fabrication by performing gamma scanning on all accessible surfaces of the cask in the lead pour region. The gamma scanning may be performed prior to, or after installation of the water jacket. The purpose of the gamma scanning test is to demonstrate that the gamma shielding of the transfer cask body is at least as effective as that of a lead and steel test block. For the test block, the steel thickness shall be equivalent to the minimum design thickness of steel in the transfer cask component and the lead thickness shall be 5 percent lower than the minimum design thickness of lead in the transfer cask component (see the Design Drawings for the design values). Data shall be recorded on a 6-inch by 6-inch (nominal) grid pattern over the surfaces to be scanned. Should the measured gamma dose rates exceed those established with the test block, the shielding of that transfer cask component shall be deemed unacceptable. Corrective actions should be taken, if practicable, and the testing re-performed until successful results are achieved. If physical corrective actions are not practicable, the degraded condition may be dispositioned with a written evaluation in accordance with applicable procedures to determine the acceptability of the transfer cask for service. Gamma scanning shall be performed in accordance with written and approved procedures. Dose rate measurements shall be documented and shall become part of the quality documentation package.

As an alternative to poured lead, the lead shielding inside the HI-TRAC transfer cask body may be made of individual lead sheets layered together. The minimum total thickness of lead sheets installed in the cask body shall be verified to meet the licensing drawing including tolerance. The effectiveness of each lead sheet shall be verified by visual examination. The visual examination includes absence of cracks, pores, inclusions, scratches, grooves, or other types of defects that could impair the gamma shielding function of the lead. Defects which exceed a depth of 10% of the plate thickness and an area greater than one square inch shall be rejected. Lead sheets thicker than 3/16" (nominal) shall be subject to UT testing. The UT testing will take place before the installation of the plates. The UT testing ensures that the plates are uniform internally. This is an accepted industry procedure for locating voids within the lead sheets in order to verify the shielding effectiveness of the sheets.

The effectiveness of the lead plates in the HI-TRAC pool lid (all transfer cask designs) and transfer lid (HI-TRAC 125 and 100 only) shall be verified during fabrication by performing a UT test of the lead plates. The UT testing will take place before the installation of the plates. The UT testing ensures that the plates are uniform internally. This is an accepted industry procedure for locating voids within the lead plate in order to verify the shielding effectiveness of the plate.

Following the first fuel loading of each HI-STORM 100 System (HI-TRAC transfer cask and HI-STORM storage overpack), a shielding effectiveness test shall be performed at the loading facility site to verify the effectiveness of the radiation shield. This test shall be performed after the HI-STORM overpack and HI-TRAC transfer cask have been loaded with an MPC containing spent fuel assemblies and the MPC has been drained, moisture removed, and backfilled with helium.

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM overpack and HI-TRAC. Measurements shall be taken at the locations specified in the Radiation Protection Program for comparison against the prescribed limits. The test is considered acceptable if the dose rate readings are less than or equal to the calculated limits. If dose rates are higher than the limits, the required actions provided in the Radiation Protection Program shall be completed. Dose rate measurements shall be documented and shall become part of the quality documentation package.

#### NOTE

Section 9.1.5.3 below (including Subsections 9.1.5.3.1 through 9.1.5.3.3) is incorporated into the HI-STORM 100 CoC by reference (CoC Appendix B, Section 3.2.8) and may not be deleted or altered in any way without prior NRC approval via CoC amendment. The text of this section is, therefore, shown in bold type to distinguish it from other text.

#### 9.1.5.3 Neutron Absorber Tests

Each plate of neutron absorber shall be visually inspected for damage such as scratches, cracks, burrs, peeled cladding, foreign material embedded in the surfaces, voids, delamination, and surface finish, as applicable.

#### 9.1.5.3.1 Boral (75% Credit)

After manufacturing, a statistical sample of each lot of neutron absorber shall be tested using wet chemistry and/or neutron attenuation testing to verify a minimum  $^{10}\text{B}$  content (areal density) in samples taken from the ends of the panel. The minimum  $^{10}\text{B}$  loading of the neutron absorber panels for each MPC model is provided in Table 2.1.15. Any panel in which  $^{10}\text{B}$  loading is less than the minimum allowed shall be rejected. Testing shall be performed using written and approved procedures. Results shall be documented and become part of the cask quality records documentation package.

#### 9.1.5.3.2 METAMIC<sup>®</sup> (90% Credit)

NUREG/CR-5661 identifies the main reason for a penalty in the neutron absorber B-10 density as the potential of neutron streaming due to non-uniformities in the neutron absorber, and recommends comprehensive acceptance tests to verify the presence and uniformity of the neutron absorber for credits more than 75%. Since a 90% credit is taken for METAMIC<sup>®</sup>, the following criteria must be satisfied:

- The boron carbide powder used in the manufacturing of METAMIC<sup>®</sup> must have small particle sizes to preclude neutron streaming
- The  $^{10}\text{B}$  areal density must comply with the limits of Table 2.1.15.
- The  $\text{B}_4\text{C}$  powder must be uniformly dispersed locally, i.e. must not show any particle agglomeration. This precludes neutron streaming.
- The  $\text{B}_4\text{C}$  powder must be uniformly dispersed macroscopically, i.e. must have a consistent concentration throughout the entire neutron absorber panel.
- The maximum  $\text{B}_4\text{C}$  content in METAMIC<sup>®</sup> shall be less than or equal to 33.0 weight percent.

To ensure that the above requirements are met the following tests shall be performed:

- All lots of boron carbide powder are analyzed to meet particle size distribution requirements.
- The following qualification testing shall be performed on the first production run of METAMIC<sup>®</sup> panels for the MPCs in order to validate the acceptability and consistency of the manufacturing process and verify the acceptability of the METAMIC<sup>®</sup> panels for neutron absorbing capabilities:

- 1) The boron carbide powder weight percent shall be verified by testing a sample from forty

different mixed batches. (A mixed batch is defined as a single mixture of aluminum powder and boron carbide powder used to make one or more billets. Each billet will produce several panels.) The samples shall be drawn from the mixing containers after mixing operations have been completed. Testing shall be performed using the wet chemistry method.

- 2) The  $^{10}\text{B}$  areal density shall be verified by testing a sample from one panel from each of forty different mixed batches. The samples shall be drawn from areas contiguous to the manufactured panels of METAMIC<sup>®</sup> and shall be tested using the wet chemistry method. Alternatively, or in addition to the wet chemistry tests, neutron attenuation tests on the samples may be performed to quantify the actual  $^{10}\text{B}$  areal density.
  - 3) To verify the local uniformity of the boron particle dispersal, neutron attenuation measurements of random test coupons shall be performed. These test coupons may come from the production run or from pre-production trial runs.
  - 4) To verify the macroscopic uniformity of the boron particle distribution, test samples shall be taken from the sides of one panel from five different mixed batches before the panels are cut to their final sizes. The sample locations shall be chosen to be representative of the final product. Wet chemistry or neutron attenuation shall be performed on each of the samples.
- During production runs, testing of mixed batches shall be performed on a statistical basis to verify the correct boron carbide weight percent is being mixed.
  - During production runs, samples from random METAMIC<sup>®</sup> panels taken from areas contiguous to the manufactured panels shall be tested via wet chemistry and/or neutron attenuation testing to verify the  $^{10}\text{B}$  areal density. This test shall be performed to verify the continued acceptability of the manufacturing process.

The measurements of  $\text{B}_4\text{C}$  particle size,  $^{10}\text{B}$  isotopic assay, uniformity of  $\text{B}_4\text{C}$  distribution and  $^{10}\text{B}$  areal density shall be made using written and approved procedures. Results shall be documented.

#### 9.1.5.3.3 Installation of the Neutron Absorber Panels

Installation of neutron absorber panels into the fuel basket shall be performed in accordance with written and approved instructions. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a neutron absorber panel in accordance with drawings in Chapter 1. These quality control processes, in conjunction with in-process manufacturing testing, provide the necessary assurances that the neutron absorber will perform its intended function. No additional testing or in-service monitoring of the neutron absorber material will be required.

#### 9.1.6 Thermal Acceptance Tests

The thermal performance of the HI-STORM 100 System, including the MPCs and HI-TRAC transfer casks, is demonstrated through analysis in Chapter 4 of the FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading. Following the loading and placement on the storage pad of the first HI-STORM System placed in service, the operability of the natural convective cooling of the HI-STORM 100 System shall be verified by the performance of an air temperature rise test. A description of the test is described in FSAR Chapter 8.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM 100 heat removal system.

#### 9.1.7 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in 10 CFR 72.236(k).

Table 9.1.1 MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>a) Examination of MPC components per ASME Code Section III, Subsections NB and NG, as defined on design drawings, per NB-5300 and NG-5300, as applicable.</p> <p>b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements.</p> <p>c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements.</p> <p>d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations.</p> <p>e) Cleanliness of the MPC shall be verified upon completion of fabrication.</p> <p>f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.</p>	<p>a) The MPC shall be visually inspected prior to placement in service at the licensee's facility.</p> <p>b) MPC protection at the licensee's facility shall be verified.</p> <p>c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.</p>	a) None.

Table 9.1.1 (continued)			
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	<p>a) Assembly and welding of MPC components shall be performed per ASME Code Section IX and III, Subsections NB and NG, as applicable.</p> <p>b) Materials analysis (steel, neutron absorber, etc.), shall be performed and records shall be kept in a manner commensurate with "important to safety" classifications.</p>	<p>a) None.</p>	<p>a) An ultrasonic (UT) examination or multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld shall be performed per ASME Section V, Article 5 (or ASME Section V, Article 2). Acceptance criteria for the examination are defined in Subsection 9.1.1.1 and in the Design Drawings.</p> <p>b) ASME Code NB-6000 pressure test shall be performed after MPC closure welding. Acceptance criteria are defined in the Code.</p>
Leak Tests	<p>a) Helium leakage testing of the MPC shell and MPC shell to baseplate welds is performed on the unloaded MPC.</p> <p>b) Helium leakage testing of the MPC base metals (shell, baseplate, lid) is performed.</p>	<p>a) None.</p>	<p>a) Helium leak rate testing shall be performed on the vent and drain port cover plate to MPC lid field welds and the cover plate base metals. See Technical Specification Bases in Chapter 12 for guidance on acceptance criteria.</p>



Table 9.1.1 (continued)			
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Criticality Safety	a) The boron content shall be verified at the time of neutron absorber material manufacture.  b) The installation of neutron absorber panels into MPC basket plates shall be verified by inspection.	a) None.	a) None.
Shielding Integrity	a) Material compliance shall be verified through CMTRs.  b) Dimensional verification of MPC lid thickness shall be performed.	a) None.	a) None.
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is to be tested during fabrication. <ul style="list-style-type: none"> <li>- MPC lid</li> <li>- vent/drain port cover plates</li> <li>- MPC closure ring</li> </ul> b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components shall be verified during pre-operation. <ul style="list-style-type: none"> <li>- MPC lid</li> <li>- MPC closure ring</li> <li>- vent/drain cover plates</li> </ul>	a) None.
Canister Identification Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 9.1.2 HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>Structural Steel Components:</p> <p>a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</p> <p>b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</p> <p>c) All welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</p> <p>d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations.</p> <p>Concrete Components: The following processes related to concrete components shall be implemented per ACI 349 as clarified in FSAR Appendix 1.D. Concrete testing shall be in accordance with Table 1.D.2. Activities shall be conducted in accordance with written and approved procedures.</p> <p>a) Assembly and examination. b) Materials verification. c) Mixing, pouring, and testing.</p>	<p>a) The overpack shall be visually inspected prior to placement in service.</p> <p>b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means.</p> <p>c) Overpack protection at the licensee's facility shall be verified.</p> <p>d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.</p>	<p>a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned .</p> <p>b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.</p>

Table 9.1.2 (continued)			
HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) No structural or pressure tests are required for the overpack during fabrication. b) Concrete compressive strength tests shall be performed per Appendix 1.D.	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per ACI-349 as clarified by FSAR Appendix 1.D, at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement. c) Verification of material composition shall be performed.	a) None	a) A shielding effectiveness test shall be performed after the initial fuel loading in accordance with Section 9.1.5.2 of this FSAR..

Table 9.1.2 (continued)			
HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM 100 System in accordance with the operating procedures in Chapter 8.  b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's discretion.  (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions.  (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading.	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</p> <p>b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</p> <p>c) All welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</p> <p>d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations</p> <p>e) Cleanliness of the transfer cask shall be verified upon completion of fabrication.</p> <p>f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.</p>	<p>a) The transfer cask shall be visually inspected prior to placement in service.</p> <p>b) Transfer cask protection at the licensee's facility shall be verified.</p> <p>c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.</p>	<p>a) Annual visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements. (See footnote for Table 9.2.1).</p>

Table 9.1.3 (continued)			
HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	<p>a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category.</p> <p>b) A load test of the lifting trunnions shall be performed during fabrication per ANSI N14.6.</p> <p>c) A pressure test of the neutron shield water jacket shall be performed during fabrication.</p>	<p>a) None.</p>	<p>a) Testing to verify continuing compliance of the lifting trunnions shall be performed per ANSI N14.6. (See footnote to Table 9.2.1).</p> <p>b) The set pressure of the relief valve on the neutron shield water jacket shall be verified by calibration annually. (See footnote to Table 9.2.1)</p>
Leak Tests	<p>a) None.</p>	<p>a) None.</p>	<p>a) None.</p>
Criticality Safety	<p>a) None.</p>	<p>a) None.</p>	<p>a) None.</p>
Shielding Integrity	<p>a) Material verifications (lead, Holtite-A, shell plates, etc.) shall be performed in accordance with the item's quality category. The required material certifications shall be obtained</p> <p>b) The placement of lead shall be controlled through written special process procedures.</p> <p>c) A shielding effectiveness test shall be performed during fabrication in accordance with Section 9.1.5.2 of this FSAR.</p>	<p>a) None</p>	<p>a) A shielding effectiveness test shall be performed after the initial fuel loading in accordance with Section 9.1.5.2 of this FSAR.</p>

Table 9.1.3 (continued)			
HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading..	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask components (top, in-pool, and transfer lids) shall be performed during fabrication.	a) Fit-up test of the transfer cask lifting trunnions with the transfer cask lifting yoke shall be performed.  b) Fit-up test of the transfer cask pocket trunnions with the horizontal transfer skid shall be performed.	a) Fit-up of the top, in-pool, and transfer lids shall be verified prior to use.

Table 9.1.4 HI-STORM 100 NDE REQUIREMENTS			
MPC			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
Shell longitudinal seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT or UT	ASME Section V, Article 2 (RT) ASME Section V, Article 5 (UT)	RT: ASME Section III, Subsection NB, Article NB-5320 UT: ASME Section III, Subsection NB, Article NB-5330
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350



Table 9.1.4 (continued)			
HI-STORM 100 NDE REQUIREMENTS			
MPC			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
Lid-to-shell	PT (root and final pass) and multi-layer PT (if UT is not performed).  PT (surface following pressure test)  UT (if multi-layer PT is not performed)	ASME Section V, Article 6 (PT)    ASME Section V, Article 5 (UT)	PT: ASME Section III, Subsection NB, Article NB-5350    UT: ASME Section III, Subsection NB, Article NB-5332
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug and lift lug baseplate	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350

Table 9.1.4 (continued)			
HI-STORM 100 NDE REQUIREMENTS			
HI-STORM OVERPACK			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
N/A	N/A	N/A	N/A
HI-TRAC TRANSFER CASK			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
HI-TRAC Body: Radial ribs and short ribs to outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Water jacket end plate-to-radial channel or enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid top plate-to-pool lid outer ring	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid bottom plate-to-pool lid outer ring	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS <b>HI-TRAC TRANSFER CASK</b>			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
HI-TRAC Body: Water jacket end plate-to-outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-outer shell longitudinal and circumferential welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Radial ribs and short ribs –to-enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Jacket drain pipe and couplings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-bottom flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued)			
HI-STORM 100 NDE REQUIREMENTS			
HI-TRAC TRANSFER CASK			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
HI-TRAC Body: Outer shell-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-outer and inner shells	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-outer shell (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Top lid welds except as noted on applicable drawings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-enclosure shell panel and radial rib (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued)			
HI-STORM 100 NDE REQUIREMENTS			
HI-TRAC TRANSFER CASK			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
HI-TRAC Body: Lower water jacket welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Gusset-to-baseplate, outer shell and water jacket bottom plate (HI-TRAC 100D and 125D only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lid intermediate plate and lead cover plate-to-lid top plate & lid bottom plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door top plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door end plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lead cover plate-to-lead cover side plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

## 9.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM 100 System Operations Manual, which shall be prepared and issued prior to the delivery and first use of the system to each user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal, and confinement performance; radiological safety, and proper handling of the system in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM 100 System is totally passive by design. There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from weathering effects in storage. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures no more demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STORM 100 System is provided in Table 9.2.1.

### 9.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC lifting trunnions and pocket trunnion recesses. The examination shall inspect for indications of overstress such as cracking, deformation, or wear marks. Repairs or replacement in accordance with written and approved procedures shall be required if unacceptable conditions are identified.

Testing to verify continuing compliance of the transfer cask trunnions shall be performed in accordance with ANSI N14.6-1993.

As described in FSAR Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

### 9.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the failure of the MPC confinement boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

### 9.2.3 Subsystem Maintenance

The HI-STORM 100 System does not include any subsystems, which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, helium backfill, and leakage testing systems. Rigging, remote welders, cranes, and lifting beams shall also be inspected prior to each loading campaign to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations with the HI-STORM 100 require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also FSAR Section 9.2.6.

### 9.2.4 Pressure Relief Valves

The pressure relief valves used on the water jackets for the HI-TRAC transfer cask shall be calibrated on an annual basis (or prior to the next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year) to ensure pressure relief settings are 60 +2/-0 psig and 65 +2/-0 psig, or replaced with factory-set relief valves.

### 9.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM overpack, HI-TRAC, and MPC degrade negligibly over time or as a result of usage.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.104(c) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC water jacket shall be verified during each loading campaign in accordance with the licensee's approved operations procedures.

The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. The use of Boral in similar nuclear applications is discussed in Chapter 1, and the long-term performance in a dry, inert gas atmosphere is evaluated in Chapter 3. A similar discussion is provided for METAMIC<sup>®</sup> neutron absorber material. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM 100 System.

9.2.6 Thermal

In order to assure that the HI-STORM 100 System continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.



Table 9.2.1

## HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

<b>Task</b>	<b>Frequency</b>
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM 100 Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC lifting trunnion and pocket trunnion recess visual inspection	Prior to each handling campaign
Testing to verify continuing compliance of HI-TRAC Lifting Trunnions	In accordance with ANSI N14.6-1993
HI-TRAC pressure relief valve calibration	Annually <sup>†</sup>
HI-TRAC internal and external visual inspection for compliance to design drawings	Annually <sup>†</sup>
HI-TRAC water jacket water level visual examination	During each handling campaign in accordance with licensee approved operations procedures
Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

<sup>†</sup> Or prior to next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year.

### 9.3 REGULATORY COMPLIANCE

Chapter 9 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STORM 100 System in accordance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STORM 100 System will provide assurance that the SSCs important to safety will perform their design function. The performance of the maintenance program by the licensee for each loaded HI-STORM 100 System will provide assurance for the continued safe long-term storage of the stored SNF.

The described acceptance criteria and maintenance programs can be summarized in the following evaluation statements:

1. Section 9.1 of this FSAR describes Holtec International's proposed program for pre-operational testing and initial operations of the HI-STORM 100 System. Section 9.2 describes the proposed HI-STORM 100 maintenance program.
2. Structures, systems, and components (SSCs) of the HI-STORM 100 System designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.2.6 and 8.1.6 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STORM 100 System and its ancillary equipment, respectively. Tables 2.2.6 and 2.2.7 present the applicable standards for their design, fabrication, and inspection of the HI-STORM 100 System components.
3. Holtec International will examine and test the HI-STORM 100 System to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 9.1 of this FSAR describes the MPC confinement boundary assembly, inspection, and testing.
4. Holtec International will mark the cask with a data plate indicating its model number, unique identification number, and empty weight.
5. It can be concluded that the acceptance tests and maintenance program for the HI-STORM 100 System are in compliance with 10CFR72 [9.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

#### 9.4 REFERENCES

- [9.0.1] U.S. Code of Federal Regulations, Title 10, "Energy", Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste".
- [9.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.
- [9.1.1] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 1995 Edition, including Addenda through 1997.
- [9.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.
- [9.1.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.
- [9.1.4] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.
- [9.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5, January 1997.
- [9.1.6] Holtec International Position Paper DS-213, "Acceptable Flaw Size in MPC Lid-to-Shell Welds", Revision 4.

## SUPPLEMENT 9.I

### ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM FOR THE HI-STORM 100U SYSTEM

#### 9.I.0 INTRODUCTION

This supplement addresses fabrication, inspection, test, and maintenance program for the HI-STORM 100U VVM. The HI-STORM 100U System does not require any changes to the acceptance criteria or maintenance requirements for the MPC or the HI-TRAC transfer cask. Therefore, the material on the fabrication, maintenance, testing, and inspection of all components contained in the main body of this chapter remains unchanged and applicable.

The VVM consists of a shop-fabricated CEC (Cavity Enclosure Container) installed below grade and a removable Closure Lid. The CEC is a welded shell-type structure made of low carbon steel plate and bar (or forging) stock. Likewise, the Closure Lid is made of welded and formed steel plates. However, unlike the CEC, the Closure Lid also contains shielding concrete.

By virtue of its underground configuration, the CEC is interfaced by the subgrade along its lateral surface by the top surface pad near its flanged upper region and by the support foundation along its bottom surface. The requirements on these interfacing bodies to the extent they are needed to enable the CEC to render its intended function are provided in Supplement 2.I. All requirements pertaining to the manufacturing, inspection, testing, and maintenance of the VVM SSCs are presented in this supplement to comply with the provisions of 10CFR71.24(p).

#### 9.I.1 ACCEPTANCE CRITERIA

The design, fabrication, inspection, and testing of the VVM is performed in accordance with the applicable codes and standards specified in Supplement 2.I and on the drawing. Acceptance criteria described in FSAR Section 9.1 for the overpack are also implemented for the HI-STORM 100U VVM, as applicable, and as further particularized below.

##### 9.I.1.1 Manufacturing of VVM Components

The manufacturing of VVM components shall be carried out in accordance with the CoC holder's NRC-approved QA program. All elements of the manufacturing cycle will be established to accord with the Important-to-Safety (ITS) designation of the specific part and the applicable provisions of the referenced codes and standards. The acceptance criteria for the manufactured components apply to each step of the manufacturing evolution, namely (a) supplier selection, (b) preparation of material procurement specifications, (c) preparation of the shop traveler and fabrication procedures, (d) fabrication activities such as forming, bending, plasma cutting, and welding, (e) in-process inspections, (f) final inspection, (g) packaging for shipment, and (h) assembling of the documentation package to serve as the archival evidence of adherence to the quality requirements.

### 9.I.1.2 Site Construction

Like the aboveground HI-STORM overpacks, the site construction activities on the VVM and its interfacing SSCs (namely, the foundation, the subgrade, and the top surface pad) shall be carried out to demonstrate compliance with the technical criteria set forth in this FSAR. The specific requirements, to ensure that the required *critical characteristics*<sup>1</sup> of the VVM and the interfacing SSCs are realized, are summarized below.

- a. All site construction processes shall be procedurized, reviewed, and approved in accordance with the CoC holder's NRC-approved QA program.
- b. Although the ITS designations apply to VVM components only, all interfacing SSCs (except for concrete encasement and cathodic protection system) shall also be designated as ITS Category C to ensure that the construction materials and processes will be subject to the necessary quality assurance regimen.
- c. Compliance with the requirements in this FSAR shall be demonstrated by appropriate testing and the results documented for archival reference. For example, the strength properties of the subgrade can be established using the classical "plate test".
- d. The shielding concrete shall comply with the applicable provisions of Appendix 1.D.
- e. The insulation installed on the Divider Shell shall be subject to visual examination to ensure that it is undamaged and properly secured.
- f. The coating on the exterior surface of the CEC shall be inspected for absence of damage before the surface becomes inaccessible.
- g. The impressed current cathodic protection system (ICCPS), if used, shall be installed in accordance with the applicable guidelines referenced in Supplement 3.I, Subsection 3.I.4.
- h. The concrete encasement, if used, shall be installed in accordance with the requirements of Supplement 3.I, Subsection 3.I.4.
- i. The dimensional compliance of the CEC, including its verticality, shall be inspected to establish compliance with the governing construction documents.
- j. The installation of the corrosion barriers shall be in accordance with written procedures.

<sup>1</sup> See Glossary for definition.

- k. All VVM surfaces that become inaccessible shall be photographed with sufficient resolution to provide a clear archive of their in-situ state at installation.

The governing processes for manufacturing control shall be prepared to be in complete compliance with this FSAR. For example, the dimensions of the HI-STORM 100U VVM steel components and the properties of the shielding concrete shall be verified to be in accordance with FSAR Supplement 2.I, Appendix 1.D (as applicable), and the drawings prior to concrete placement. The dimensional inspection and density measurements shall be documented.

The governing NDE references provided for the aboveground HI-STORM overpack in Table 9.1.4 also apply to the VVM, as do the requirements on the documentation package contents specified in paragraph 9.1.1(13).

In order to receive the Certificate-of-Compliance under the CoC holder's QA program, the manufacturing of the VVM components must meet all of the technical, quality control, procedural (quality assurance) and administrative requirements set forth in the manufacturing program.

Similarly, the low carbon steel plates, bars, and forgings, as applicable, used in the construction of the HI-STORM 100U VVM shall be dimensionally inspected to assure compliance with the requirements on the drawings. Test results shall be documented and become part of the quality documentation package. Dimensional inspections of the Closure Lid and its weight measurement after placement in the shielding concrete shall assure that the required amount of shielding material has been incorporated in the lid.

#### 9.I.1.3 Inspections and Testing

- i. Post-Construction Inspection:

Each as-built VVM shall be inspected for final acceptance before it is loaded with fuel. The following inspections define minimum acceptance requirements:

- a. The as-installed Divider Shell shall be inspected to ensure that it sits on the Bottom Plate and is flanked by lateral and circumferential guides located in the bottom region of the CEC. The top surface elevation and roundness of the Divider Shell shall be inspected to ensure it will not hinder installation of the Closure Lid.
- b. The Container Flange and Divider Shell gasket/seal bearing surfaces shall be inspected for their horizontal alignment (within specified tolerance).
- c. The Restraint Ring welded to the underside of the Closure Lid shall be checked for fitup with the Container Flange.
- d. The inlet and outlet air passages in the Closure Lid shall be inspected for absence of obstruction such as debris and extensive weld spatter.

- e. The results of the post-construction inspection shall be incorporated in the VVM's Documentation Package.
- f. The impressed current cathodic protection system (ICCPS), if used, shall be tested in accordance with the applicable guidelines referenced in Supplement 3.I, Subsection 3.I.4.

ii. Shielding Integrity and Effectiveness Test

The effectiveness tests performed during fabrication and operation of the HI-TRAC remains the same as in the aboveground system, as described in 9.1.5.2. The 100U VVM is unique however since the gamma and neutron shielding are provided by the concrete and steel components of the VVM as well by as by the surrounding subgrade and the top surface pad.

As shown in Supplement 5.I, the shielding performance of the HI-STORM 100U system is significantly better than its aboveground counterparts. Further, the long-term shielding effectiveness in the HI-STORM 100U system is also assured to an extremely high level of confidence by virtue of its physical configuration, choice of materials, and design embodiment, as described below:

- a. Absence of penetrations in the VVM body: The CEC has no penetrations and thus has no path that can serve as a conduit for radiation streaming. All penetrations are in the lid and are configured to maximize scattering of photons and neutrons.
- b. Axisymmetric penetrations in the lid: As shown in the drawings, all penetrations in the lid are axisymmetric. In particular, the air inlet passage is formed by narrowly separated steel plates that girdle the lid, allowing only infinitesimal amounts of neutrons and gamma radiation to the outside. Likewise, the outlet air passage is formed by a set of formed surfaces of revolution that preclude a direct "line-of-sight" from the fuel to the outside. Because these air passages are formed by welded steel shapes, they will remain invariant over time, making their shielding performance reliably constant over years of use.
- c. Aging of foundation, subgrade: Even though a very stiff support foundation is specified, some settlement of the foundation is expected. However, any settlement of the foundation would have no deleterious effect on the extent of shielding to the environment.

Second, while the subgrade is unloaded, except when a transporter is passing over it, some settlement of the subgrade and the above top surface pad is expected. Any subgrade settlement, however, will result in a corresponding compaction of its material, which will improve the subgrade's shielding capability. The settlement of the subgrade will not result in any new loading on the CEC structure (which is autonomously supported on the foundation) or the Closure Lid, which is

autonomously supported on the CEC structure. Any visible gap or crevice between the Container Flange and the surface pad should be filled with grout or caulking for both aesthetic purposes and enhancement of degradation and corrosion mitigation.

- d. Effect of Corrosion: It can be readily deduced from the VVM's design that the only surfaces that are not accessible for corrosion monitoring are the bottom face and outer surface of the CEC. As discussed in Supplement 3.I, corrosion mitigation measures for these surfaces are prescribed and expected to provide adequate corrosion mitigation for decades beyond the Design Life of the VVM. These corrosion mitigation measures, however, are chiefly required to preserve long-term structural, not shielding, integrity of the CEC because the effect of the CEC shell or Bottom Plate corrosion on the dose to the environment is ostensibly miniscule.

Thus, the potential for reduction in the shielding effectiveness of the HI-STORM 100U system due to corrosion is not a concern.

- e. Water Intrusion: Because the underground portion of the HI-STORM 100U VVM contains a thick steel shell structure without any penetrations, intrusion of ground water in the MPC storage cavity is not credible. At some sites, ground water may saturate the surrounding subgrade. A water-saturated subgrade, however, will be more effective (not less) in radiation attenuation and, therefore, seepage of ground water around the HI-STORM 100U vertical space will not threaten loss in its shielding integrity.
- f. Materials of Construction: The sole material of construction in the underground portion of the CEC is carbon steel (the optional concrete encasement is not credited in the shielding evaluation), which is a proven material of long-term shielding endurance under fluence levels that are orders of magnitude greater than those present in the CEC. The Closure Lid is comprised of low carbon steel that encases plain concrete. Concrete, like steel, is a proven durable material in a radiation environment that suffers negligible change in its shielding capability over long periods of use in a radiation environment. Therefore, material degradation induced reduction in the shielding effectiveness of the VVM is not a credible concern.

The long-term shielding effectiveness of the HI-STORM 100U is assured by virtue of its design configuration and its constituent materials.

However, to verify the VVM's shielding capability, operational neutron and gamma shielding effectiveness tests, (i.e. radiation surveys), shall be performed similar to those described in Subsection 9.1.5.2 for the aboveground systems. The tests will be performed after fuel loading using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the dose rates at the top of the VVM and at the vents, consistent with the



Radiation Protection Program. The tests are considered acceptable if the dose rates are less than or equal to the calculated limits. If dose rates are higher than the limits, the required actions provided in the Radiation Protection Program shall be completed. Dose rate measurements shall be documented and become a part of the quality documentation package.

iii. Thermal Acceptance Test

The thermal acceptance test is performed as described in Subsection 9.1.6 for the aboveground systems.

9.I.1.4 Cask Identification

The MPC and HI-TRAC transfer cask are identified as described in Subsection 9.1.7. See Subsection 9.I.3 for discussion of identification requirements for the VVM.

9.I.1.5 Storm Water Control Test

The VVM is designed to direct storm water and snow/ice melt-off away from the Closure Lid where the air passages are located. Any minor amount of moisture that may intrude into the MPC cavity due to wind-driven rain will evaporate in a short period of time due to the movement of heated air in the MPC storage cavity. To verify the effectiveness of the storm water drainage design, a one-time test shall be performed after construction of the first VVM to ensure that the design is effective in directing storm water away from the VVM to the ISFSI's drainage system. The VVM shall be subjected to a water spray that simulates exposure to rainfall of at least 2 inches per hour for at least one hour. At the conclusion of the water spray, the depth of the water (if any) in the bottom of the module cavity shall be measured. Any amount of water accumulation beyond wetting of the Bottom Plate indicates an inadequacy in rain diversion features of the VVM and shall be appropriately corrected. It should be noted that, as stated in Supplement 4.I, any amount of accumulated water in the CEC will not cause any component of the HI-STORM 100U System to exceed its regulatory limit as supported by Supplement 4.I. The only deleterious effect of water is the potential for increased preservative degradation (on the wetted surface of the Bottom Plate, CEC shell and other interfacing components) from prolonged exposure.

9.I.1.6 Inspection Criteria

The inspection and test criteria specified in FSAR Table 9.1.2 shall be implemented for the VVM to the extent they are applicable. No additional inspections or tests other than those specified in this supplement are required. Consistent with Table 9.1.4, no non-destructive examination (NDE) beyond visual examination is required for the VVM steel components.

## 9.I.2 MAINTENANCE PROGRAM

### 9.I.2.1 HI-STORM 100U System

The HI-STORM 100U System is totally passive by design and requires minimal normal maintenance to ensure it can perform its design functions. Periodic surveillance (via temperature monitoring or visual or camera-aided inspection of air passages) is required to ensure that the air passages in the lid are not blocked. Preventive or remedial painting of the exposed steel surfaces as part of the user's preventive maintenance program is recommended to mitigate corrosion. Such preventive maintenance activities are typical in scope and complexity to other standard maintenance activities at nuclear power plants. Maintenance activities described in FSAR Section 9.2 and Table 9.2.1 shall be implemented for other HI-STORM 100U System components, namely, the MPC and the HI-TRAC transfer cask, without any change.

The maintenance requirements on the VVM also parallel those for the aboveground HI-STORM overpack, but are slightly different. In the following, the essentials of the maintenance program for VVM are specified. Because the VVM is a passive structure and is largely protected from the weather due to its underground configuration, the maintenance requirements on it are rather minimal. Nevertheless, a carefully articulated preventive maintenance program is essential for a satisfactory service life.

In-service inspection for long-term interior and below-grade degradation shall be performed on a site-specific basis in accordance with Holtec prepared long-term maintenance guidelines and the licensee's preventive maintenance program. At most potential ISFSI sites, visual inspection of accessible areas of the VVM is expected to be sufficient to evaluate long-term degradation. The frequency of this visual in-service inspection should be in performed in accordance with Table 9.I.1. Additional in-service inspection activities may include more thorough inspections for foreign material accumulation, corrosion (CEC wall thinning) and insulation degradation as warranted by site-specific conditions. One example is ultrasonic inspection of the CEC using a remotely operated device to check for wall thinning. VVM Closure Lid removal or even MPC transfer into a HI-TRAC may be necessary for certain inspections. For certain inspections such as evaluating CEC wall thinning, a sampling of VVMs at a given ISFSI array may be used but should be commensurate with previous inspection results. The frequencies for the additional in-service inspection activities will be determined on a site-specific basis and may include allowance to increase or decrease the frequency commensurate with the results of previous inspections. In-service inspections for evaluating foreign material accumulation, corrosion (CEC wall thinning) and/or insulation degradation are not required if it is determined that the applicable degradation mechanisms do not exist. For defense in depth, VVM ISFSIs not employing impressed current cathodic protection shall be subject to a mandatory visual and UT inspection of at least one representative VVM to check for wall thinning and pitting of the CEC at an interval not to exceed 20 years. The inspections are performed on the inside surface of the CEC. The VVM chosen for inspection is not required to be in use or to have previously contained a loaded MPC. A VVM with a loaded MPC may be inspected using remote devices with the VVM lid removed. The oldest VVM or VVM considered to be most vulnerable to corrosion degradation shall be selected for inspection.

As is true for all components certified pursuant to this FSAR, the maintenance activities on the HI-STORM 100U VVM shall be performed in accordance with a written program that fulfills the requirements of the CoC holder's 10CFR72 Subpart G compliant QA program, the owner site's Safety Plan and corrective action program; and the system's Technical Specification.

Among the QA commitments are performance of maintenance by trained personnel by written procedures; and, written documentation of the maintenance work performed and of the results obtained. Table 9.I.1 provides a listing of the minimum maintenance activities on the HI-STORM 100U VVM.

#### 9.I.2.2 Subsystem Maintenance

No new ancillary equipment or sub-systems requiring periodic maintenance are required for use of the HI-STORM 100U System.

#### 9.I.2.3 Shielding

As discussed in the foregoing, the radiation shielding capacity of the HI-STORM 100U System is expected to degrade negligibly over time. Therefore, unless the VVM is subjected to an extreme environmental event that imparts stresses or temperatures beyond the design basis limits for the system (i.e., prolonged fire or impact from a beyond-the-design basis large energetic projectile) such that this event could potentially degrade the shielding effectiveness of the VVM, no shielding effectiveness tests beyond that required by the Radiation Protection Program are required over the life of the HI-STORM 100U System.

### 9.I.3 REGULATORY COMPLIANCE

The information presented in this supplement augments the material in the main body of the chapter to fulfill the regulatory requirements pertaining to the testing and maintenance of the HI-STORM 100U System, resolution of issues concerning adequacy and reliability, and cask identification. Section 9.3 of this FSAR provides the necessary information on all HI-STORM system components except the VVM. This section demonstrates the corresponding compliance information on the VVM.

- a. The program for pre-operational testing and initial operations, as required by 10CFR72.24(p) for the VVM, is provided in Section 9.I.1 herein.
- b. The maintenance protocol for the VVM, as specified in §72.236(g), is provided in Section 9.I.2 herein.
- c. The quality assurance requirements on the design, fabrication, on-site construction of the VVM commensurate with its ITS designation are invoked through Chapter 13 of this FSAR and are summarized in Section 9.I.1 herein, as called for in §72.122(a), §72.122(f), §72.128(a)(1) and §72.24(c).

- d. The provisions of §72.82(c) with respect to acceptance criteria and the appropriate test program to ensure compliance with the acceptance criterion are fulfilled by Section 9.I.1 herein.
- e. The quality requirements with respect to inspection, testing, and documentation, as set down in §72.212(b)(8), are provided in Section 9.I.1 herein.
- f. The VVM, except for its Closure Lid, does not require unique marking other than that required for fuel ISFSI configuration management because it is not a transportable structure. The module is essentially an integral portion of the ISFSI, not a separate cask structure that can be moved by the user. As such, the markings provided on the MPC alone are sufficient to meet the requirements of 10CFR72.236(k). Nevertheless, it is required that each VVM cavity be labeled with an identifier that is unique for the specific site. Further, because the Closure Lid will be subject to handling, consistent with defense-in-depth guidelines for handling heavy loads at nuclear plants, its unique identifier and its bounding weight shall be permanently marked on a readily visible location.

<b>Table 9.I.1: Maintenance Activities for the HI-STORM 100U VVM<sup>2</sup></b>			
	<b>Activity</b>	<b>Frequency</b>	<b>Comment</b>
1.	CEC cavity is visually inspected	Prior to MPC installation	To ensure that VVM internal components are properly aligned, the surface preservatives on all exposed surfaces are undamaged, the insulation on the Divider Shell is undamaged and the cavity is free of visible foreign material.
2.	Lid Examination	Prior to MPC installation	Ensure that the preservatives on the external surfaces are in good condition and the lid is free of dents and rust stains.
3.	Screen Inspection	Prior to installation of the flanged screen assembly and monthly when in use	Ensure that the screen is undamaged.
4.	Interfacing SSCs	Annually	Ensure that the top surface pad is free of visible cracks, the interface between the top surface pad and the Container Flange is grouted (or caulked) if necessary, the ISFSI drain system is functional, the ground water system (if used) is in working order, and that subgrade settlement counter measures are implemented. The cathodic protection system is routinely verified as operable in accordance with the Bases in Chapter 12 for guidance on acceptance criteria.
5.	Shielding Effectiveness Test	As required by the Radiation Protection Program	—

<sup>2</sup> This table provides maintenance activities for the “100U” VVM that parallel those for HI-STORM overpacks in Table 9.2.1.

<b>Table 9.I.1: Maintenance Activities for the HI-STORM 100U VVM<sup>2</sup></b>			
	<b>Activity</b>	<b>Frequency</b>	<b>Comment</b>
6.	ISFSI Settlement	Every five years	Confirm that the VVM settlement is within the range of the Plant's "best estimate". Implement countermeasures if the settlement is determined to be excessive by the CoC holder.
7.	VVM Air Temperature Monitoring System (if used)	Per Licensee's QA Program and manufacturer's recommendations	—
8.	VVM In-Service Inspection	Annually	Ensure that the vent screen assembly fasteners remain coated with preservative, the screen is undamaged, all visible external surfaces are corrosion free, and the air passages are not degraded.
9.	<p>Additional VVM In-Service Inspection for Long-Term Interior and Below-grade Degradation:</p> <p>a) Visual inspection of accessible areas for long-term degradation.</p> <p>b) Additional in-service inspection activities include inspection for foreign material accumulation, corrosion (CEC thinning) and insulation degradation</p> <p>c) Visual and UT inspection for VVMs <u>not</u> employing impressed current cathodic protection</p>	<p>a) Monthly visual inspection of accessible areas.</p> <p>b) Frequencies for additional in-service inspections are determined on a site-specific basis.</p> <p>c) Visual and UT inspection at an interval not to exceed 20 years</p>	<p>Inspection activities shall be commensurate with site-specific conditions. Under site conditions existing at most ISFSI sites, visual inspection of accessible areas is sufficient to determine the general condition of the system.</p> <p>See Subsection 9.I.2.1 and Technical Specifications for more information.</p>

## SUPPLEMENT 9.III

### ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM FOR THE MPC-68M

#### 9.III.0 INTRODUCTION

The main body of this chapter remains fully applicable for the HI-STORM 100 System using the MPC-68M model fuel basket (with existing MPC enclosure vessel) except as described below. Since the MPC-68M fuel basket is made entirely of Metamic-HT, the fabrication and inspection standards as well as the welding differ from the standard stainless steel baskets in the other MPC models. The MPC-68M model fuel basket is equipped with full length basket shims made entirely of aluminum alloy.

#### 9.III.1 ACCEPTANCE CRITERIA

##### 9.III.1.1 Fabrication and Non-Destructive Examination

The fabrication, inspection and testing on the MPC-68M fuel basket is performed in accordance with the following fabrication controls and required inspections and the drawing in Section 1.5.

1. Materials of construction specified for the MPC-68M fuel basket (including fuel basket and fuel basket shims) are identified in the drawing in Section 1.5. Metamic-HT panel material shall be fabricated and certified with supporting documentation as required in this FSAR supplement.
2. Structural welds shall be examined and repaired in accordance with written and approved procedures for welding Metamic HT.
3. Weld procedures and welder qualifications shall be in accordance with written and approved procedures for welding Metamic HT.
4. Non-structural welds requiring weld repair shall be repaired in accordance with written and approved procedures developed specifically for welding Metamic-HT or basket shims as applicable.

##### 9.III.1.2 Structural and Pressure Tests

Structural and pressure tests for the HI-STORM 100 System are discussed in the main body of this chapter.

##### 9.III.1.3 Material Testing

Qualification tests on Metamic-HT specimens drawn from production runs shall be performed in compliance with Table 9.III.1 requirements to ensure that the manufactured panels shall render their intended function. Testing shall be performed using written and approved procedures consistent with the test methods documented in Holtec's test report [1.III.3].

To ensure the above test requirements are met a sampling plan based on the MIL Standard 105E [9.III.1] is defined and incorporated in the Metamic-HT Manufacturing Manual.

Test results on all materials shall be documented and become part of the final quality documentation package.

#### 9.III.1.4 Component Tests

Component tests for the HI-STORM 100 System are discussed in the main body of this chapter.

#### 9.III.1.5 Shielding Integrity

The information in this section is moved to Section 9.III.2 below.

##### 9.III.1.5.1 Neutron Absorber Manufacturing Requirements

The information in this section is moved to Section 9.III.2.1 below.

##### 9.III.1.5.2 Neutron Absorber Manufacturing and Tests

The information in this section is moved to Section 9.III.2.2 below.

#### 9.III.2 Shielding Integrity

##### 9.III.2.1 Neutron Absorber Manufacturing Requirements

Essential characteristics of Metamic-HT are described in Supplement 1.III of this FSAR. The manufacturing of Metamic-HT is governed by a set of quality validated procedures contained in a Manufacturing Manual [1.III.4].

The key constituents of Metamic-HT, namely aluminum powder and Boron Carbide powder are procured under their respective purchasing specifications that define the required particle size distributions and set down the prohibited materials & impurities, as well as tolerable level of trace amounts of acceptable impurities.

As required by the procedures set down in its manufacturing manual [1.III.4], each panel of Metamic-HT neutron absorber material shall be visually inspected for damage such as scratches, cracks, burrs, presence of imbedded foreign materials, voids and discontinuities that could significantly affect its functional effectiveness.

Metamic-HT panels will be manufactured according to a Holtec purchase specification that incorporates all requirements set forth in this FSAR. The supplier of raw materials must be qualified under Holtec's quality program (or under Holtec subsidiary Nanotec's quality program)



for important to safety materials and components. The manufacturing of Metamic-HT is subject to all quality assurance requirements under Holtec International's NRC approved quality program.

The tests conducted on Metamic-HT to establish the compliance of the manufactured panels with Holtec's Purchasing Specification are intended to ensure that *critical characteristics* of the final product will meet the minimum guaranteed values (MGVs) in Supplement 1.III of this FSAR. The tests are performed at both the raw material and manufactured panels stages of production with the former serving as the insurer of the properties in the final product and the latter serving the confirmatory function. The testing is conducted for each lot of raw material and finished panels as prescribed in Table 9.III.1. A lot is defined as follows:

“Lot” means a population of an item that shares identical attributes that are central to the defining a critical performance or operational characteristics required of it. Thus, a lot of boron carbide powder procured to a certified Purchasing Specification used in the manufacturing of Metamic-HT is the bulk quantity of the powder that has the same particle size distribution. A lot of finished panels drawn from a powder mix and manufactured in an extrusion run have identical aluminum and boron carbide characteristics and the same extrusion conditions.

The following tests are performed (see Table 9.III.1):

(i) Testing and certification of powder material

- All lots of aluminum and boron carbide powder shall be certified to meet particle size distribution and chemistry requirements in [1.III.7].
- All lots of B4C shall be certified as containing Boron with the minimum isotopic B-10 per the boron carbide purchase specifications incorporated in the Manufacturing manual [1.III.4].
- Homogenized mixtures of Al powder(s) and boron carbide powder(s) from traceable lots, prepared for sintering and billet forming operations, shall have the minimum boron carbide wt% verified by wet chemistry testing of one sample from each lot of blended powders. The mixing/blending of the batch shall be controlled via approved procedures.

(ii) Testing of finished panels

The number of panels subject to testing shall be governed by Table 9.III.2. The panels that need to be tested per the statistical protocol of Table 9.III.2, hereafter referred to as test panels, shall be subject to the following evaluations:

- The Metamic-HT panels shall be tested for all mechanical properties in Table 1.III.2 in accordance with Table 9.III.2 sampling plan.

- The thickness of each panel will be measured using the procedure set down in [1.III.4]. The average measured value must meet the minimum basket wall requirements specified in the Licensing drawings in Section 1.5.
- One coupon from each Metamic-HT manufactured lot shall be subject to neutron attenuation testing to quantify the boron carbide content for compliance with the minimum requirement in the MPC-68M licensing drawings in Section 1.5 using written procedures.

(iii) Testing of Basket

- Metamic-HT basket welds shall be tested/inspected as stated in Section 9.III.1.1 using written procedures

Each neutron absorber plate shall be visually inspected for damage such as scratches, cracks, burrs, foreign material embedded in the surfaces, voids, and delaminations. Panels are also visually inspected for contamination on the surface as specified in the Manufacturing Manual [1.III.4]. Panels not meeting the acceptance criteria will be reworked or rejected. Unless basket is fabricated at the same factory manufacturing Metamic-HT, all panels shall be inspected before being shipped to the cask manufacturing facility where they may be subject to receipt inspection prior to installation.

### 9.III.3 MAINTENANCE PROGRAM

#### 9.III.3.1 Structural and Pressure Parts

Structural and pressure parts for the HI-STORM 100 System are discussed in the main body of this chapter.

#### 9.III.3.2 Leakage Tests

Leakage tests for the HI-STORM 100 System are discussed in the main body of this chapter.

#### 9.III.3.3 Subsystem Maintenance

Subsystem maintenance for the HI-STORM 100 System is discussed in the main body of this chapter.

#### 9.III.3.4 Pressure Relief Valves

Pressure relief valves for the HI-STORM 100 System are discussed in the main body of this chapter.

#### 9.III.3.5 Shielding

The Metamic-HT is not expected to degrade under normal long-term storage conditions. Therefore, no periodic verification testing of neutron poison material is required on the MPC-68M fuel basket.

#### 9.III.3.6 Thermal

Thermal for the HI-STORM 100 System is discussed in the main body of this chapter.

#### 9.III.3.7 Fuel Basket

No additional tests are required for the MPC-68M fuel basket. Long-term fuel basket integrity is ensured by fuel basket design and by extensive material testing (including the effects of creep and irradiation). The minimum guaranteed values (MGVs) provided in Table 1.III.2, the testing requirements in Table 9.III.1 and 9.III.2 along with the certification requirements in Table 9.III.3 ensure that the fuel basket will meet its long-term performance requirements.

#### 9.III.4 REFERENCES

[9.III.1] “Sampling Procedures and Tables for Inspection by Attributes”, Military Standard MIL-STD-105E, (10/5/1989).

Table 9.III.1  
Metamic-HT Testing Requirements

Item Tested	Property Tested For	Frequency of Test	Purpose of Test	Acceptance Criterion
B <sub>4</sub> C powder (raw material) (see note 1)	Particle size distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchase Specification [1.III.7]
	Purity	One sample per lot	To verify material supplier's data sheet	ASTM C-750
Al powder (raw material)	Particle Size Distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchase Specification [1.III.7]
	Purity	One sample per lot	To verify material supplier's data sheet	Must be 99% (min.) pure aluminum
B <sub>4</sub> C/Al Mix	B <sub>4</sub> C Content (by the wet chemistry method)	One sample per mixed/blended powders lot	To ensure B <sub>4</sub> C content requirements compliance	The B <sub>4</sub> C content in accordance with the drawing package in Section 1.5
Finished Metamic-HT panel	Thickness	Each Panel	To ensure fabricability of the basket	Per Holtec's Purchasing Specification [1.III.7]
	Width, straightness, camber, and bow	Per Sampling Plan (see note 3)	To ensure fabricability of the basket	Per Holtec's Purchasing Specification [1.III.7]
	Properties listed in Table 9.III.3	Per Sampling Plan (see note 2)	To ensure structural performance.	MGV per Table 9.III. 3
	B <sub>4</sub> C content (by neutron attenuation testing)	One coupon from each Metamic-HT manufactured lot	To ensure criticality safety	The B <sub>4</sub> C content in accordance with licensing drawing in Section 1.5

## Notes:

1. The B<sub>4</sub>C testing requirements apply if the raw material supplier is not in Holtec's Approved Vendor List.
2. Sampling Plan is included in the Metamic-HT Manufacturing Manual [1.III.4].
3. The sampling plan for the dimensional analysis of the finished Metamic-HT panels shall start at Tier No. 4 of Table 9.III.2 with samples coming from the first and last panel of every lot or operating shift, whichever comes first. If it is required to move up a Tier No. per the notes in Table 9.III.2, then the

required number of panels shall be tested in the beginning and end of each lot or operating shift, whichever comes first, and one sample shall be taken from each panel.

Table 9.III.2  
Tier System for Coupon Testing

Tier No.	Number of Extrusions Tested as a Percent of Number of Extrusions in the Lot	Number of Continuous Lots that Must Pass to Drop Down to the Next Tier
1	20	5
2	12.5	5
3	5	10
4	1	N/A
<p>Note 1: If a coupon fails with respect to any MGV property, then it may be replaced by two coupons from the extrusion that produced the failed coupon. If both of the replacement coupons pass the failed MGV property, then the lot can be accepted. If either of the replacement coupons is unsuccessful in meeting the failed MGV property, then the entire lot is rejected. As an alternative to rejecting the entire lot, testing of the failed MGV value on all extrusions within the lot is permitted to isolate acceptable panels.</p> <p>Note 2: Testing shall be moved up to the next tier if any MGV property fails in two consecutive lots.</p>		

Table 9.III.3 Minimum Guaranteed Values Required for Certification of Production Runs of Metamic-HT (All testing performed at ambient temperature.)		
	Property	MGV
1	Yield Strength, ksi	Table 1.III.2 for MGV values at ambient temperature
2	Tensile Strength, ksi	
3	Young's Modulus, ksi	
4	Area Reduction, %	

## CHAPTER 10: RADIATION PROTECTION<sup>†</sup>

This chapter discusses the design considerations and operational features that are incorporated in the HI-STORM 100 Storage System design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, transfer, and on-site dry storage. Occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections are provided. An off-site dose assessment for a typical ISFSI is also discussed. Since the determination of off-site doses is necessarily site-specific, similar dose assessments are to be prepared by the licensee, as part of implementing the HI-STORM 100 Storage System in accordance with 10CFR72.212 [10.0.1]. The information provided in this chapter meets all requirements of NUREG-1536.

### 10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

#### 10.1.1 Policy Considerations

The HI-STORM 100 has been designed in accordance with 10CFR72 [10.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [10.1.1] and the guidance provided in Regulatory Guides 8.8 [10.1.2] and 8.10 [10.1.3]. Licensees using the HI-STORM 100 System will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [10.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM 100 System, and be familiarized with the expected dose rates around the MPC, HI-STORM and HI-TRAC during all phases of loading, storage, and unloading operations. Chapter 12 provides dose rate limits at the HI-TRAC and HI-STORM surfaces to ensure that the HI-STORM 100 System is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings should be held with workers and radiological protection personnel prior to work on or around the system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities, will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [10.1.1] standards for radiation protection are met in accordance with the site's written commitments. Users can further reduce dose rates around the HI-STORM 100 System by preferentially loading longer-cooled and lower-burnup spent fuel assemblies in the periphery fuel storage cells of the MPC, and loading assemblies with shorter cooling times and higher burnups in the inner MPC fuel storage cell locations. Users can also further reduce the dose rates around the HI-TRAC by the use of temporary shielding. In some cases, users may opt to upgrade their existing crane to take advantage of the increased shielding capabilities of the 125-Ton HI-TRAC transfer cask (versus the 100-Ton HI-TRAC transfer cask). This decision should be based on a cost-benefit analysis. Temporary shielding and use of special tools to reduce dose is discussed in Section 10.1.4.

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



### 10.1.2 Design Considerations

Consistent with the design criteria defined in Section 2.3.5, the radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM 100 Storage System are as follows:

1. 10CFR72.104 [10.0.1] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements.
2. 10CFR72.106 [10.0.1] requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.
3. 10CFR20 [10.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.
4. Regulatory Position 2 of Regulatory Guide 8.8 [10.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM 100 Storage System as described below:
  - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [10.0.1]. Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a loaded HI-STORM 100 Storage cask is placed in an ISFSI. Due to the nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.
  - Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 or later in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM 100 System design include:
    - system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;

- system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;
  - system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;
  - system designs that minimize planned maintenance requirements;
  - system designs that minimize decontamination requirements at ISFSI decommissioning;
  - system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;
  - thick walled overpack that provides gamma and neutron shielding;
  - thick MPC lid which provides effective shielding for operators during MPC loading and unloading operations;
  - multiple welded barriers to confine radionuclides;
  - smooth surfaces to reduce decontamination time;
  - minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
  - capability of maintaining water in the MPC during welding to reduce dose rates;
  - capability of maintaining water in the transfer cask annulus space and water jacket to reduce dose rates during closure operations;
  - MPC penetrations located and configured to reduce streaming paths;
  - HI-STORM and HI-TRAC designed to reduce streaming paths;
  - MPC vent and drain ports with resealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;
  - use of a pool lid, annulus seal, and Annulus Overpressure System to prevent contamination of the MPC shell outer surfaces during in-pool activities;
  - temporary and auxiliary shielding to reduce dose rates around the HI-TRAC; and
  - low-maintenance design to reduce doses during storage operation.
- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.

- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM 100 Storage System is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this FSAR, and no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean water in the HI-TRAC transfer cask-MPC annulus and by using an inflatable annulus seal.
- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM 100 Storage System ISFSI since there are no radioactive systems at an ISFSI that could transport crud.
- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the HI-TRAC transfer cask is designed for ease of decontamination. In addition, an inflatable annulus seal is used to prevent fuel pool water from contacting and contaminating the exterior surface of the MPC.
- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.
- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins.
- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC shell, the primary confinement boundary. This material is resistant to the damaging effects of radiation and is well proven in the SNF cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

### 10.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STORM 100 System include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated welding system, lift yoke, transfer slide or mating device and moisture removal systems to reduce time operators spend in the vicinity of the loaded MPC;
- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;

- low fuel assembly lift-over height of the HI-TRAC maximizes water coverage over assemblies during fuel assembly loading;
- a water-filled neutron shield jacket allows filling after removal of the HI-TRAC from the spent fuel pool. This maximizes the shielding on the HI-TRAC without exceeding the crane capacity;
- descriptive operating procedures that provide guidance to reduce equipment contamination, obtain survey information, minimize dose and alert workers to possible changing radiological conditions;
- preparation and inspection of the HI-STORM and HI-TRAC in low-dose areas;
- MPC lid fit tests and inspections prior to actual loading to ensure smooth operation during loading;
- gas sampling of the MPC and HI-STAR 100 annulus (receiving from transport) to assess the condition of the cladding and MPC confinement boundary;
- HI-STORM vent temperature elements (See Chapter 12) allow remote monitoring of the vent operability surveillance;
- wetting of component surfaces prior to placement in the spent fuel pool to reduce the need for decontamination;
- decontamination practices which consider the effects of weeping during HI-TRAC transfer cask heat up and surveying of HI-TRAC prior to removal from the fuel handling building;
- a sequence of operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations.

#### 10.1.4 Auxiliary/Temporary Shielding

To minimize occupational dose during loading and unloading operations, a specially-designed set of auxiliary shielding is available. The HI-STORM 100 auxiliary shielding consists of the Automated Welding System Baseplate, the HI-TRAC Temporary Shield Ring, the annulus shield, HI-STORM vent shield insert, the HI-TRAC transfer step or mating device, and the shield panel trim plates. Additional supplemental shielding such as lead blankets and bricks or other such shielding may also be used to help reduce dose rates. Each auxiliary shield is described in Table 10.1.1, shown on Figure 10.1.1 and the procedures for utilization are provided in Chapter 8. Other embodiments of the temporary shielding may also be used. Table 10.1.2 provides the minimum requirements for use of the temporary shielding indicating optional and required shielding. Users shall evaluate the need for auxiliary and temporary shielding and use of special tooling to reduce the overall exposure based on an ALARA review of cask loading operations and the MPC contents.

Table 10.1.1  
HI-STORM 100 AUXILIARY AND TEMPORARY SHIELDS

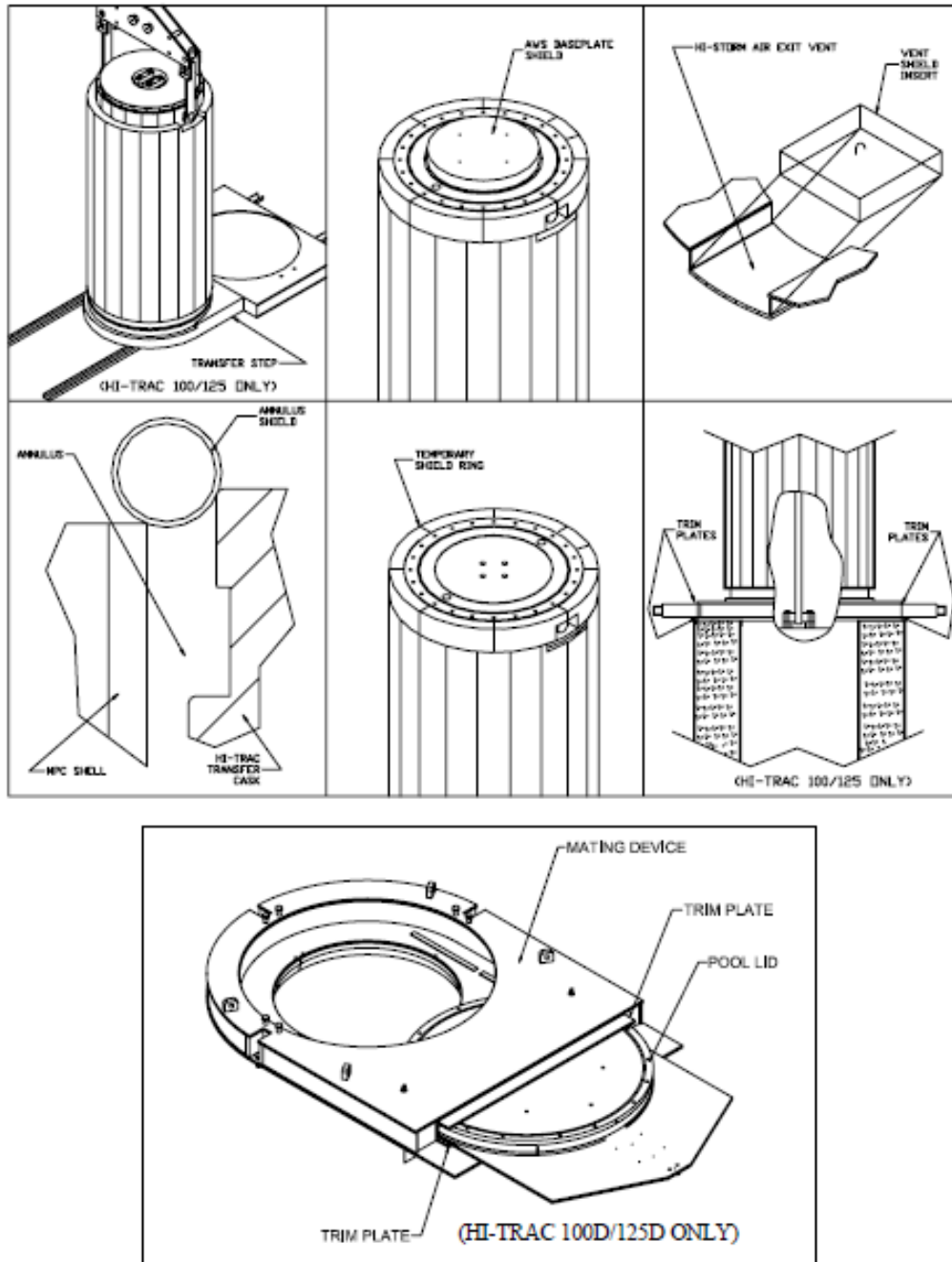
<b>Temporary Shield</b>	<b>Description</b>	<b>Utilization</b>
Automated Welding System Baseplate	Thick gamma and neutron shield circular plate that sits on the MPC lid. Plate is set directly on the MPC. Threaded lift holes are provided to assist in rigging.	Used during MPC closure and unloading operations in the cask preparation area to reduce the dose rates around the MPC lid. The design of the closure ring allows the baseplate shield to remain in place during the entire closure operation.
HI-TRAC Temporary Shield Ring	A water-filled tank that is placed atop of the HI-STAR or HI-TRAC neutron shield.	Used during MPC and HI-TRAC closure operations and MPC transfers into HI-STAR to reduce dose rates to the operators around the top flange of the HI-TRAC.
Annulus Shield	A shield that is seated between the MPC shell and the HI-TRAC.	Used during MPC closure operations to reduce streaming from the annulus.
HI-TRAC Transfer Step	A stepped block used to position the pool lid and transfer lid at the same elevation. The transfer step creates a tight seam between the two lids to eliminate streaming during bottom lid replacement.	Used during HI-TRAC 100 or 125 bottom lid replacement.
HI-TRAC Mating Device	Device used to remove HI-TRAC pool lid and to provide shielding during MPC transfer to HI-STORM..	Used during MPC transfer to the HI-STORM when used with HI-TRAC 100D or 125D
Shield Panel Trim Plates	Four steel plates approximately 0.25 inch by 3 inch by 80 inch that are placed at the ends of the transfer lid top and bottom plate and secured by clamps or other method deemed suitable by the user.	Used during MPC transfer to and from HI-TRAC to shield the small gap above and below the sliding doors on the transfer lid.
HI-STORM Vent Shield Inserts	Devices shaped to fit into the HI-STORM exit vents.	Used during MPC transfer to and from HI-STORM to eliminate the streaming path from the exit vents during MPC transfer operations.

Table 10.1.2  
HI-STORM 100 AUXILIARY AND TEMPORARY SHIELD REQUIREMENTS

<b>Auxiliary Shielding</b>	<b>Required for the 100-Ton HI-TRAC and HI-TRAC 100D</b>	<b>Required for the 125-ton HI-TRAC and HI-TRAC 125D</b>
<b>Temporary Shield Ring</b>	Note 1	Note 1
<b>Automated Welding System Baseplate Shield</b>	No	No
<b>Annulus Shield</b>	Note 1	Note 1
<b>Vent Duct Shield Inserts</b>	Note 2	Note 2
<b>Transfer Step</b>	Yes (Note 3)	Yes (Note 3)
<b>Trim Plates</b>	No	No
<b>Mating Device</b>	Yes (Note 4)	Yes (Note 4)

Notes:

1. Users shall determine the need for this temporary shielding based on the specific operations and the MPC contents.
2. Not required for the HI-STORM 100S Overpack.
3. Not used with the HI-TRAC 100D or 125D.
4. Used only with HI-TRAC 100D or 125D.



**FIGURE 10.1.1: HI-STORM 100 System Auxiliary/Temporary Shielding**

## 10.2 RADIATION PROTECTION DESIGN FEATURES

The development of the HI-STORM 100 System has focused on design provisions to address the considerations summarized in Sections 10.1.2 and 10.1.3. The intent has been to improve on past concrete-based dry storage system designs by developing HI-STORM 100 as a hybrid of current metal and concrete storage system technologies. The design is, therefore, an evolution in storage systems, which incorporates preferred features from concrete storage, canister-based systems while retaining several of the advantages of metal casks as well. This approach results in a reduction in the need for maintenance, in overall radiation levels, and in the time spent on maintenance, when compared with current concrete-based dry storage systems. The following specific design features ensure a high degree of confinement integrity and radiation protection:

- HI-STORM 100 has been designed to meet storage condition dose rates required by 10CFR72 [10.0.1] for three-year cooled fuel;
- HI-STORM 100 has been designed to accommodate a maximum number of PWR or BWR fuel assemblies to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site;
- HI-STORM 100 overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance to corrosion and other forms of degradation (e.g., erosion);
- HI-STORM 100 has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the confinement boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or credible accident conditions;
- HI-TRAC transfer cask utilizes a mating device or transfer step and other auxiliary shielding devices which reduce streaming paths and simplify operations;
- The pool lid maximizes available fuel assembly water coverage in the spent fuel pool.
- The transfer lid and mating device are designed for quick alignment with HI-STORM; and
- HI-STORM 100 has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel.



### 10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM system. This section uses the shielding analysis provided in Chapter 5 and the operations procedures provided in Chapter 8 to develop a dose assessment. The dose assessment is provided in Tables 10.3.1, 10.3.2, and 10.3.3.

The dose rates from the HI-STORM 100 overpack, MPC lid, HI-TRAC transfer cask, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the various loading and unloading operations. The dose rates are also calculated for the various conditions of the cask that may affect the dose rates to the operators (e.g., MPC water level, HI-TRAC annulus water level, neutron shield water level, presence of temporary shielding). The dose rates around the 100-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of 60,000 MWD/MTU and cooling of 3 years including BPRAs. The dose rates around the 125-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of 75,000 MWD/MTU and cooling of 5 years including BPRAs. The dose rates around the HI-STORM 100 overpack are based on 24 PWR fuel assemblies with a burnup of 60,000 MWD/MTU and cooling of 3 years. The selection of these fuel assembly types in all fuel cell locations bound all possible PWR and BWR loading scenarios for the HI-STORM System from a dose-rate perspective. The HI-STORM dose rates used in this chapter were calculated for the HI-STORM 100 S Version B. No assessment is made with respect to background radiation since background radiation can vary significantly by site. In addition, exposures are based on work being performed with the temporary shielding described in Table 10.1.2.

The choice of burnup and cooling times used in this chapter is extremely conservative. The bounding burnup and cooling time that resulted in the highest dose rates around the 100-ton, 125-ton HI-TRACs, and HI-STORM 100S Version B were used. In addition, including the source term from BPRAs increases the level of conservatism. The maximum dose rate due to BPRAs was used in this analysis. As stated in Chapter 5, using the maximum source for the BPRAs in conjunction with the bounding burnup and cooling time for fuel assemblies is very conservative as it is not expected that burnup and cooling times of the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values. This combined with the already conservative dose rates for the HI-TRACs and HI-STORMs results in an upper bound estimate of the occupational exposure. Users' radiation protection programs will assure appropriate temporary shielding is used based on actual fuel to be loaded and resulting dose rates in the field.

For each step in Tables 10.3.1 through 10.3.3, the operator work location is identified. These correspond to the locations identified in Figure 10.3.1. The relative locations refer to all HI-STORM Overpacks. The dose rate location points around the transfer cask and overpack were selected to model actual worker locations and cask conditions during the operation. Cask operators typically work at an arms-reach distance from the cask. To account for this, an 18-inch distance was used to estimate the dose rate for the worker. This assessment addresses only the operators that perform work on or immediately adjacent to the cask.

Justification for the duration of operations along with the corresponding procedure Sections from Chapter 8 are also provided in the tables. The assumptions used in developing time durations are based on mockups of the MPC, review of design drawings, walk-downs using other equipment to represent the HI-TRAC transfer cask and HI-STORM 100 overpack the HI-STAR 100 overpack and MPC-68 prototype, consultation with UST&D (weld examination) and consultation with cask operations personnel from Calvert Cliffs Nuclear Power Plant (for items such as lid installation and decontamination). In addition, for the shielding calculations, only the Temporary Shield Ring was assumed to be in place for applicable portions of the operations.

Tables 10.3.1a, 10.3.1b, and 10.3.1c provide a summary of the dose assessment for a HI-STORM 100 System loading operation using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and the HI-TRAC 125D respectively. Tables 10.3.2a, 2b, and 2c provide a summary of the dose assessment for HI-STORM 100 System unloading operations using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and HI-TRAC 125D respectively. Tables 10.3.3a, 3b, and 3c provide a summary of the dose assessment for transferring the MPC to a HI-STAR 100 overpack as described in Section 8.5 of the operating procedures using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and 125-ton HI-TRAC 125D, respectively. The HI-TRAC 100D was not specifically analyzed since, as stated in Section 5.4, the dose rates from the HI-TRAC 100 and 100D are similar at 1 meter from the overpack. In addition, the results for the HI-TRAC 125 and 125D indicate that there is only a small difference in the occupational exposure from using a mating device (HI-TRAC 125D) rather than transfer doors. Therefore, the use of the mating device in the HI-TRAC 100D does not result in occupational exposures significantly different than those presented for the 100-ton HI-TRAC.

### 10.3.1 Estimated Exposures for Loading and Unloading Operations

The assumptions used to estimate personnel exposures are conservative by design. The main factors attributed to actual personnel exposures are the age and burnup of the spent fuel assemblies and good ALARA practices. To estimate the dose received by a single worker, it should be understood that a canister-based system requires a diverse range of disciplines to perform all the necessary functions. The high visibility and often critical path nature of fuel movement activities have prompted utilities to load canister systems in a round-the-clock mode in most cases. This results in the exposure being spread out over several shifts of operators and technicians with no single shift receiving a majority of the exposure.

The total person-rem exposure from operation of the HI-STORM 100 System is proportional to the number of systems loaded. A typical utility will load approximately four MPCs per reactor cycle to maintain the current available spent fuel pool capacity. Utilities requiring dry storage of spent fuel assemblies typically have a large inventory of spent fuel assemblies that date back to the reactor's first cycle. The older fuel assemblies will have a significantly lower dose rate than the design basis fuel assemblies due to the extended cooling time (i.e., much greater than the values used to compute the dose rates). Users shall assess the cask loading for their particular fuel types (burnup, cooling time) to satisfy the requirements of 10CFR20 [10.1.1].

For licensees using the 100-Ton HI-TRAC transfer cask, design basis dose rates will be higher (than a corresponding 125-Ton HI-TRAC) due to the decreased mass of shielding. Due to the higher expected dose rates from the 100-Ton HI-TRAC, users may need to use the auxiliary shielding (See Table 10.1.2), and should consider preferential loading, and increased precautions (e.g., additional temporary or auxiliary shielding, remotely operated equipment, additional contamination prevention measures). Actual use of optional dose reduction measures must be decided by each user based on the fuel to be loaded.

### 10.3.2 Estimated Exposures for Surveillance and Maintenance

Table 10.3.4 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM modules or remote viewing methods instead of performing direct visual observation of the modules. Since security surveillances can be performed from outside the ISFSI, and since the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area, a dose rate of 3 mrem/hour is estimated. Although the HI-STORM 100 System requires only minimal maintenance during storage (e.g. touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Since most of the maintenance is expected to occur outside the actual cask array, a dose rate of 10 mrem/hour is estimated

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @ 2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @ 2 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
Deleted							
Deleted							SIMPLE ATTACHMENT NO TOOLS
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
FLUSH CAVITY WITH HELIUM AND INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
INSTALL MSLD OVER VENT PORT COVER PLATE	2	8A	1	37.9	1.3	1.3	INSTALLED BY HAND, NO TOOLS
INSTALL MSLD OVER DRAIN PORT COVER PLATE	2	8A	1	37.9	1.3	1.3	INSTALLED BY HAND, NO TOOLS
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES @2/MIN
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	354.2	106.3	106.3	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
<b>Section 8.1.7</b>							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	122.7	32.7	65.4	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	11.7	2.0	3.9	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>						<b>2063.1 PERSON-MREM</b>	



<b>Table 10.3.1b</b>							
<b>HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK</b>							
<b>ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)</b>							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON- MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	3	3.4	6.8	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	3	2.3	4.5	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	3	1.0	2.0	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	37.6	1.9	1.9	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	37.6	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	90.3	9.0	18.1	24 BOLTS @ 2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	837.0	118.6	118.6	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	554.8	92.5	92.5	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	1091.9	91.0	91.0	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	282.1	2.4	2.4	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	90.3	15.1	15.1	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	90.3	3.0	3.0	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	1091.9	36.4	36.4	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	69.8	7.0	14.0	24 BOLTS @ 2 BOLT/PERSON MINUTES

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
MEASURE DOSE RATES AT MPC LID	3	5A	1	69.8	3.5	3.5	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	1091.9	1874.4	1874.4	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	34.2	9.1	18.2	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	34.2	14.3	14.3	230 GAL @ 10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	282.1	2.4	2.4	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	34.2	1.1	1.1	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	34.2	1.1	1.1	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	34.2	2.9	2.9	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	34.2	3.4	3.4	30 SQ-FT @ 5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	34.2	1.7	1.7	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	34.2	0.6	0.6	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	34.2	1.1	1.1	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	34.2	11.4	34.2	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	34.2	6.8	13.7	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	34.2	11.4	22.8	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	34.2	4.6	9.1	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	34.2	131.1	131.1	MEASURED DURING WELD MOCKUP TESTING

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	34.2	5.7	5.7	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	34.2	1.1	1.1	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	34.2	25.7	25.7	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
Deleted							
Deleted							
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM () TO VENT AND DRAIN PORT RVOAs	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	79.5	2.0	2.0	SINGLE THREADED CONNECTION (1 RVOA)

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	79.5	2.0	2.0	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	79.5	2.7	2.7	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	79.5	1.3	1.3	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE VENT AND DRAIN COVER PLATE WELD	100	8A	1	79.5	132.5	132.5	MEASURED DURING WELD MOCKUP TESTING
FLUSH CAVITY WITH HELIUM AND INSTALL SET SCREWS	2	8A	1	79.5	2.7	2.7	4 SET SCREWS @2/MINUTE
PLUG WELD OVER ET SCREWS	8	8A	1	79.5	10.6	10.6	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
INSTALL MSLD OVER VENT PORT COVER PLATE	2	8A	1				INSTALLED BY HAND, NO TOOLS
INSTALL MSLD OVER DRAIN PORT COVER PLATE	2	8A	1				INSTALLED BY HAND, NO TOOLS
INSTALL AND ALIGN CLOSURE RING	5	8A	1	79.5	6.6	6.6	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	79.5	245.1	245.1	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	79.5	15.9	15.9	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	79.5	1.3	1.3	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	2190.1	36.5	36.5	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	79.5	13.3	26.5	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	79.5	15.9	15.9	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	363.8	151.6	303.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	79.5	1.3	1.3	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	363.8	24.3	24.3	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	363.8	24.3	48.5	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	900.9	225.2	225.2	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	2190.1	219.0	219.0	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	2190.1	657.0	657.0	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	363.8	24.3	48.5	INSTALLED BY HAND NO TOOLS
<b>Section 8.1.7</b>							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	900.9	300.3	600.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	35.2	52.8	158.4	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	35.2	1.2	2.3	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	692.0	115.3	230.7	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	692.0	46.1	92.3	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	692.0	46.1	92.3	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	122.7	32.7	65.4	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	11.7	2.0	3.9	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>						<b>6628.4 PERSON-MREM</b>	

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1.0	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @ 2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @ 2 BOLT/PERSON MINUTES

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE OF LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS



**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
Deleted							
Deleted							
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING
INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
Deleted							
Deleted							
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM A NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND, NO TOOLS
<b>Section 8.1.7</b>							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
ATTACH MPC LIFT SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE MATING DEVICE LOCKING PINS AND OPEN DRAWER	40	13B	2	118.5	79.0	158.0	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE MATING DEVICE	10	15A	1	43.9	7.3	7.3	3 BOLTS @ 2 MINUTES PER BOLT
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	122.7	32.7	65.4	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	11.7	2.0	3.9	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>							<b>2017.4 PERSON-MREM</b>

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	11.7	2.0	2.0	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	205.5	27.4	27.4	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	354.2	106.3	106.3	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	158.5	26.4	26.4	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH RE-FLOOD SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT RE-FLOOD LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON- MREM)	ASSUMPTIONS
<b>TOTAL</b>						<b>924.4 PERSON-MREM</b>	



**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	11.7	2.0	2.0	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	205.5	27.4	27.4	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	692.0	115.3	230.7	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	692.0	46.1	92.3	INSTALLED BY HAND NO FASTENERS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	363.8	60.6	60.6	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	692.0	46.1	92.3	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	35.2	11.7	23.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	35.2	52.8	52.8	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	35.2	11.7	23.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	900.9	300.3	600.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	363.8	24.3	48.5	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	2190.1	219.0	219.0	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	2190.1	657.0	657.0	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	363.8	60.6	60.6	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	900.9	225.2	225.2	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	79.5	8.0	8.0	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	34.2	1.1	1.1	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	2190.1	18.3	18.3	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	79.5	11.9	11.9	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	34.2	22.8	45.6	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	79.5	1.3	1.3	2 COVERS@2/MIN NO TOOLS

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH RVOAS	2	8A	1	79.5	2.7	2.7	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	79.5	0.7	0.7	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	79.5	0.7	0.7	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH RE-FLOOD SYSTEM TO RVOAs	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS X 2
DISCONNECT RE-FLOOD LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	34.2	5.7	5.7	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	79.5	1.3	1.3	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	34.2	5.7	11.4	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	282.1	9.4	9.4	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	282.1	9.4	9.4	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	69.8	2.3	2.3	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	69.8	14.0	14.0	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	282.1	2.4	2.4	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	282.1	47.0	47.0	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	837.0	118.6	118.6	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	90.3	18.1	18.1	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	3	1.0	2.0	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	3	1.0	2.0	CONSULTATION WITH CALVERT CLIFFS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON- MREM)	ASSUMPTIONS
<b>TOTAL</b>						<b>3275.0 PERSON-MREM</b>	

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	11.7	2.0	2.0	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	205.5	27.4	27.4	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
INSTALL MATING DEVICE WITH POOL LID	10	15A	1	43.9	7.3	7.3	3 BOLTS AT 2 MINUTES PER BOLT
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE MATING DEVICE DRAWER AND BOLT-UP POOL LID	36	13B	2	118.5	71.1	142.2	2 PINS@2MIN/PIN, 16 BOLTS @ 2MIN/BOLT
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH RE-FLOOD SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT RE-FLOOD LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS

<b>Table 10.3.2c</b>							
<b>HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK</b>							
<b>ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)</b>							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON- MREM)	ASSUMPTIONS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
<b>TOTAL</b>						<b>787.5 PERSON-MREM</b>	



**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	7.1	1.2	2.4	ALIGN AND POSITION REMOVE 4 SHACKLES
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	487.4	203.1	406.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	118.5	7.9	15.8	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	122.7	32.7	32.7	16POINTS@1 MIN

**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

<b>ACTION</b>	<b>DURATION (MINUTES)</b>	<b>OPERATOR LOCATION (FIGURE 10.3.1)</b>	<b>NUMBER OF OPERATORS</b>	<b>DOSE RATE AT OPERATOR LOCATION (MREM/HR)</b>	<b>DOSE TO INDIVIDUAL (MREM)</b>	<b>TOTAL DOSE (PERSON- MREM)</b>	<b>ASSUMPTIONS</b>
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	11.7	2.0	2.0	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	122.7	10.2	10.2	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASMT@1/MIN
<b>TOTAL</b>						<b>1068.3 PERSON-MREM</b>	

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	6.9	1.8	1.8	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	6.9	2.9	2.9	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	6.9	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	6.9	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	6.9	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	6.9	4.5	9.0	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	6.9	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	6.9	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	6.9	1.2	2.3	ALIGN AND POSITION REMOVE 4 SHACKLES
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	487.4	203.1	406.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	692.0	115.3	230.7	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	692.0	46.1	92.3	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	692.0	46.1	92.3	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	692.0	46.1	92.3	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	363.8	60.6	121.3	2 SLINGS@5/MIN
REMOVE TRIM PLATES	4	13B	2	692.0	46.1	92.3	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	122.7	32.7	32.7	16POINTS@1 MIN

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)**

<b>ACTION</b>	<b>DURATION (MINUTES)</b>	<b>OPERATOR LOCATION (FIGURE 10.3.1)</b>	<b>NUMBER OF OPERATORS</b>	<b>DOSE RATE AT OPERATOR LOCATION (MREM/HR)</b>	<b>DOSE TO INDIVIDUAL (MREM)</b>	<b>TOTAL DOSE (PERSON- MREM)</b>	<b>ASSUMPTIONS</b>
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	11.7	2.0	2.0	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	122.7	10.2	10.2	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASMT@1/MIN
<b>TOTAL</b>						<b>1633.3 PERSON-MREM</b>	

**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	6.7	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

<b>ACTION</b>	<b>DURATION (MINUTES)</b>	<b>OPERATOR LOCATION (FIGURE 10.3.1)</b>	<b>NUMBER OF OPERATORS</b>	<b>DOSE RATE AT OPERATOR LOCATION (MREM/HR)</b>	<b>DOSE TO INDIVIDUAL (MREM)</b>	<b>TOTAL DOSE (PERSON-MREM)</b>	<b>ASSUMPTIONS</b>
SURFACE PROTECTOR							
INSTALL MATING DEVICE ON HI-STAR	20	19B	2	7.1	2.4	4.7	ALIGN AND BOLT INTO PLACE
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	487.4	203.1	406.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE LOCKING PINS AND OPEN DRAWER	4	20B	2	118.5	7.9	15.8	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
RAISE THE POOL LID AND BOLT INTO PLACE ON HI-TRAC	32	20B	2	118.5	63.2	126.4	2 MINS/BOLT, 16 BOLTS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE THE MATING DEVICE	6	15A	1	43.9	4.4	4.4	3 BOLTS AT 2 MINUTES PER BOLTS
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS



**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)**

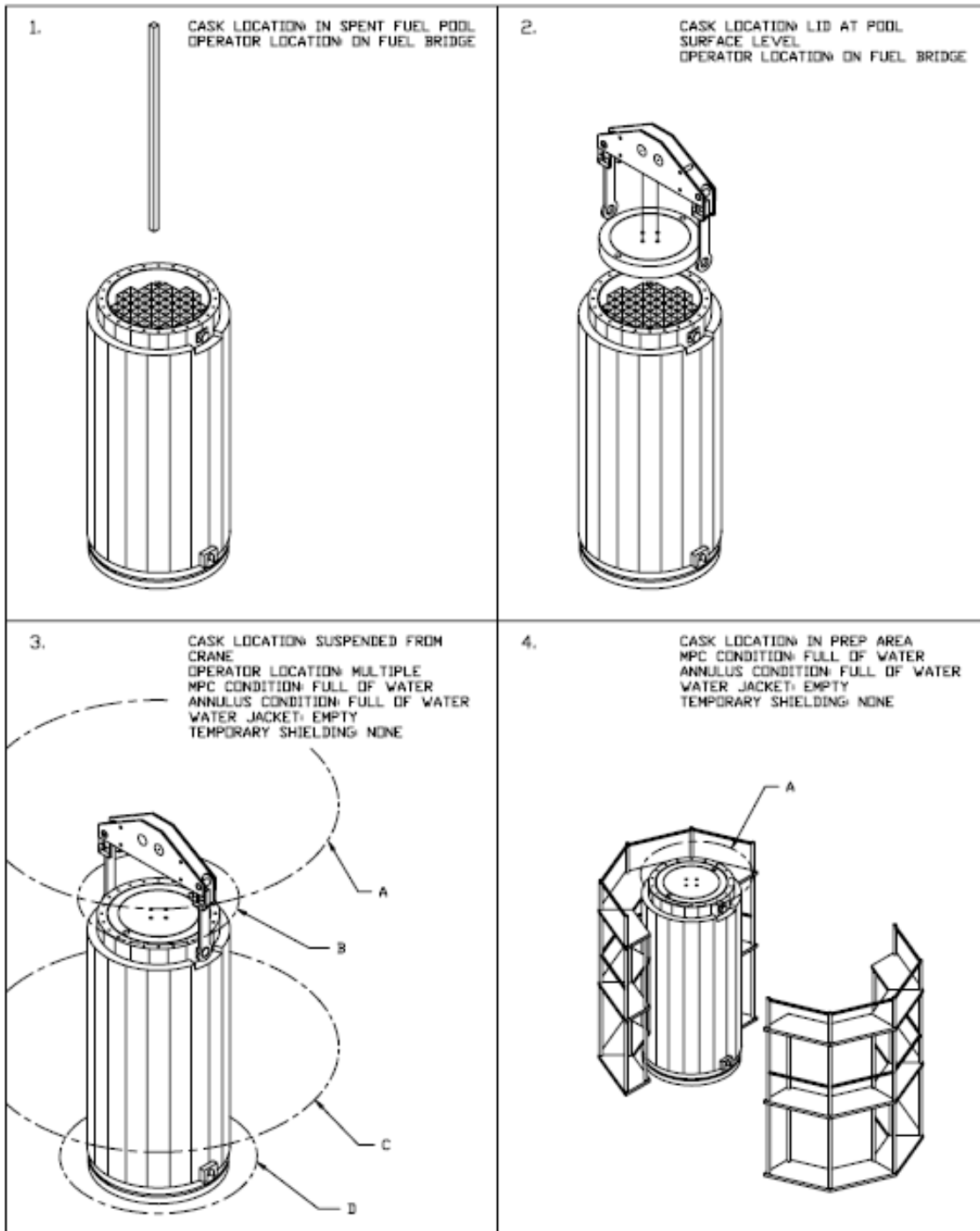
<b>ACTION</b>	<b>DURATION (MINUTES)</b>	<b>OPERATOR LOCATION (FIGURE 10.3.1)</b>	<b>NUMBER OF OPERATORS</b>	<b>DOSE RATE AT OPERATOR LOCATION (MREM/HR)</b>	<b>DOSE TO INDIVIDUAL (MREM)</b>	<b>TOTAL DOSE (PERSON- MREM)</b>	<b>ASSUMPTIONS</b>
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	122.7	32.7	32.7	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	11.7	2.0	2.0	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS @1/MIN
REMOVE AIR PAD	5	16D	1	122.7	10.2	10.2	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS @1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	205.5	27.4	27.4	8 MEASMT@1/MIN
<b>TOTAL</b>							<b>1198.6 PERSON-MREM</b>

Table 10.3.4  
ESTIMATED EXPOSURES FOR HI-STORM 100 SURVEILLANCE AND MAINTENANCE

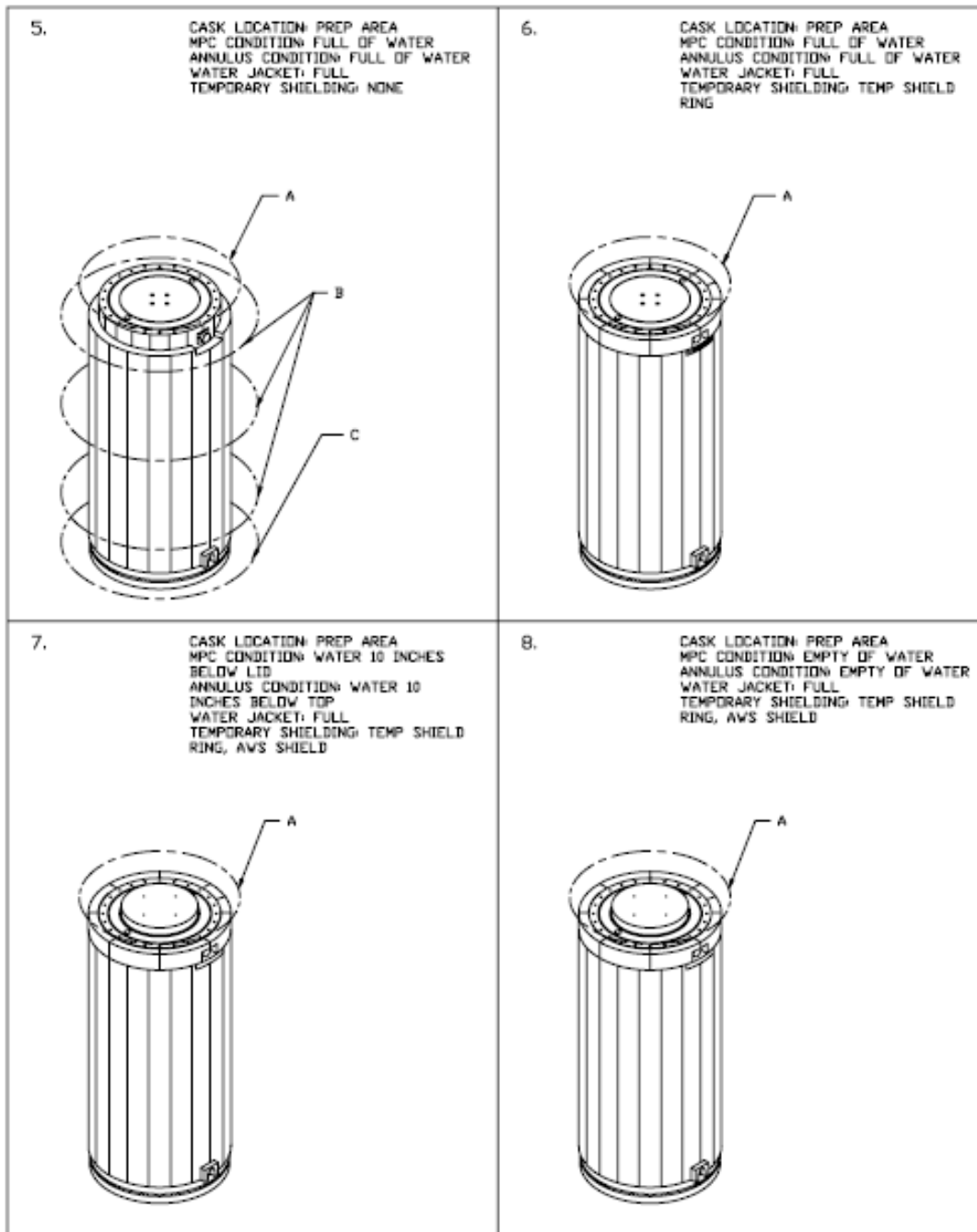
ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 10.3.1a, 10.3.1b, 10.3.1c, 10.3.2a, 10.3.2b, 10.3.2c, 10.3.3a, 10.3.3b, 10.3.3c and 10.3.4:

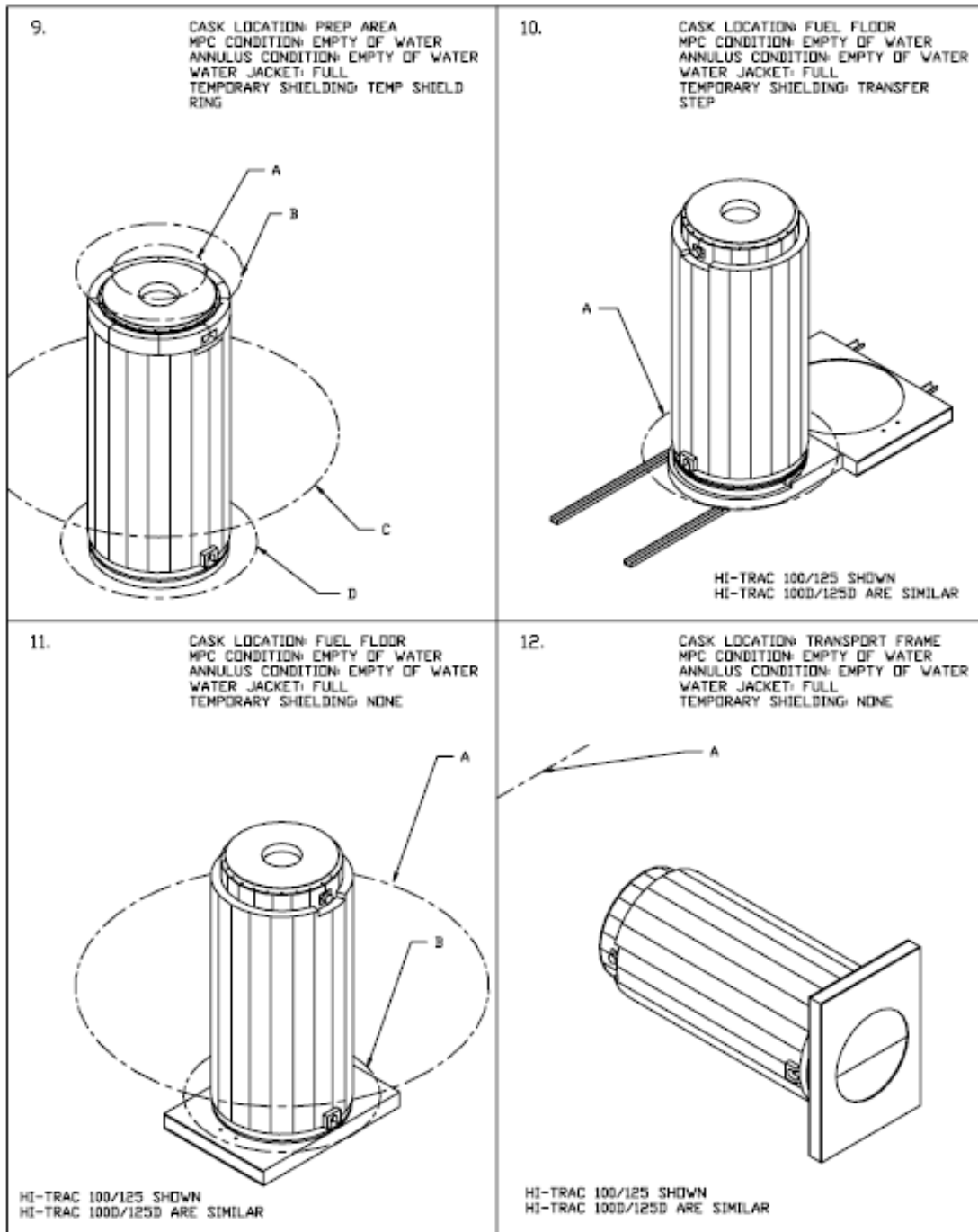
1. Refer to Chapter 8 for detailed description of activities.
2. Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.
3. HI-STAR 100 Operations assume that the cooling time is at least 10 years.



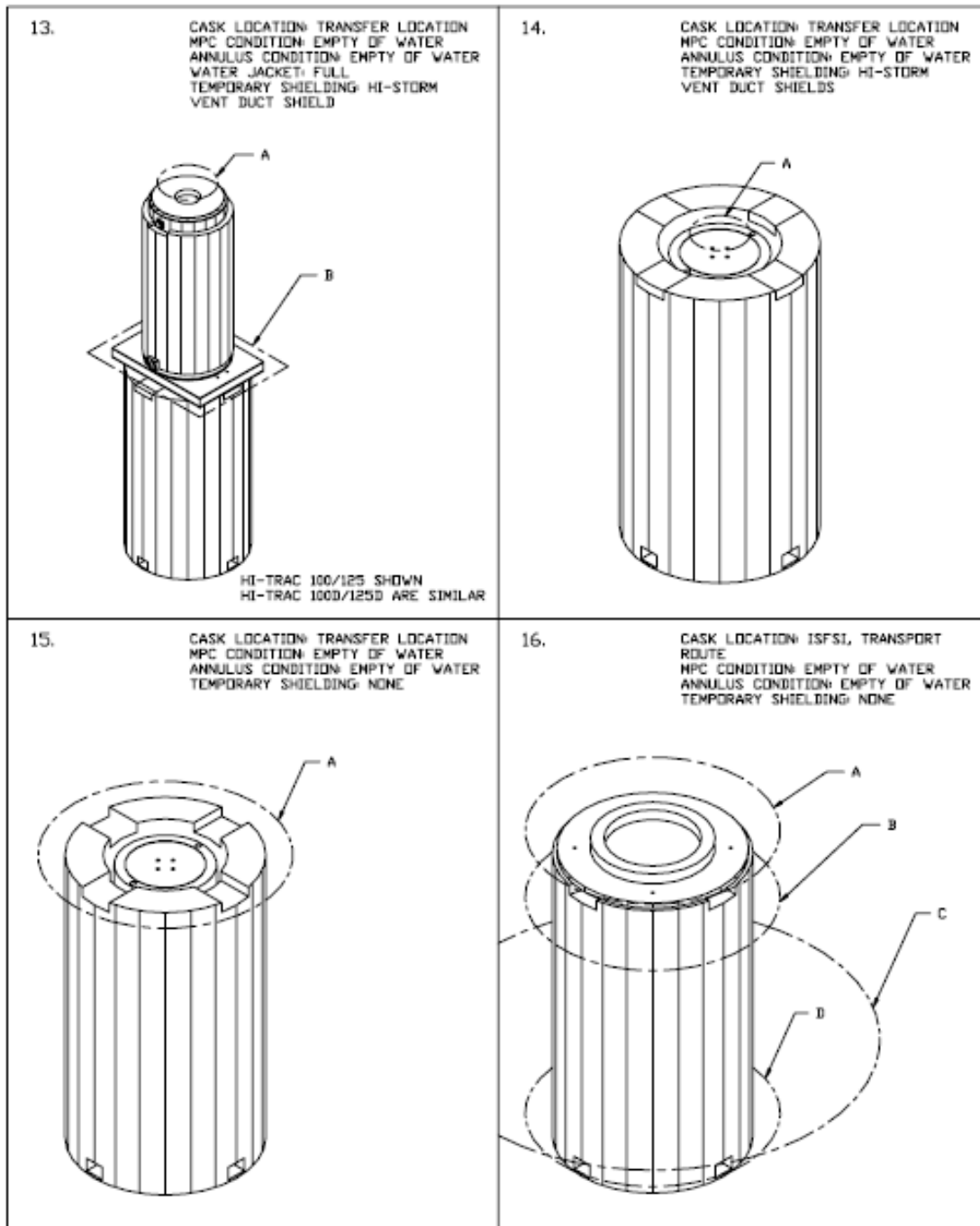
**FIGURE 10.3.1a: Operator Work Locations Used for Estimating Personnel Exposure**



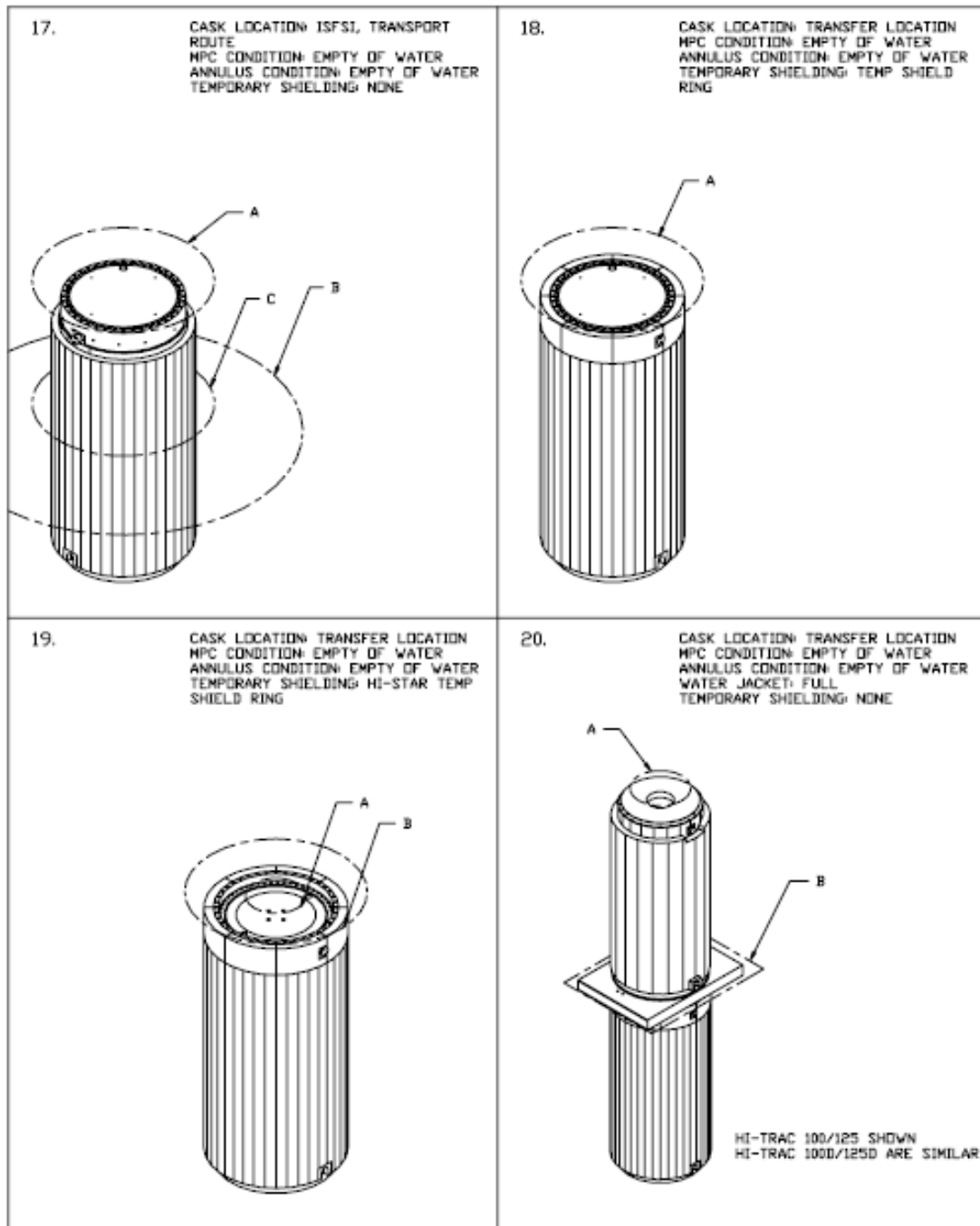
**FIGURE 10.3.1b: Operator Work Locations Used for Estimating Personnel Exposure**



**FIGURE 10.3.1c: Operator Work Locations Used for Estimating Personnel Exposure**



**FIGURE 10.3.1d: Operator Work Locations Used for Estimating Personnel Exposure**



**FIGURE 10.3.1e: Operator Work Locations Used for Estimating Personnel Exposure**

## 10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT

### 10.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [10.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STORM 100 System with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10CFR72.212 [10.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.9 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [10.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. As a summary of Chapter 5, Table 10.4.1 presents the annual dose results for a single overpack at 100 and 350 meters and a 2x5 array of HI-STORM 100 systems at 550 meters. These annual doses are based on a full array of design basis fuel with a burnup of 60,000 MWD/MTU and 3-year cooling. This burnup and cooling time combination conservatively bounds the allowable burnup and cooling times listed in Section 2.1.9. In addition, 100% occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, the casks were positioned on an infinite slab of soil to account for earth-shine effects. These results indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of 350 meters for a single cask and at 550 meters for a 2x5 array of HI-STORM 100 Systems containing design basis fuel. These results are presented only as an illustration to demonstrate that the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1]. Neither the distances nor the array configurations become part of the Technical Specifications. Rather, users are required to perform a site specific analyses to demonstrate compliance with 10CFR72.104[10.0.1] contributors and 10CFR20[10.1.1].

An additional contributor to the controlled area boundary dose is the loaded HI-TRAC transfer cask, if the HI-TRAC is to be used at the ISFSI outside of the fuel building. Table 10.4.2 provides dose rates at 100, 200, and 300 meters for a 100-ton HI-TRAC transfer cask loaded with design basis fuel. The 100-ton HI-TRAC dose rates bound the 125-ton HI-TRAC by large margins. Based on the short duration that the loaded HI-TRAC is used outside at the ISFSI, the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1] when worst-case design basis fuel is loaded in all fuel cell locations. However, users are required to perform a site specific analysis to demonstrate



compliance with 10CFR72.104[10.0.1] and 10CFR20[10.1.1] taking into account the actual site boundary distance and fuel characteristics.

Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

#### 10.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Section 11.1, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and off-normal handling of HI-TRAC) do not result in the degradation of the HI-STORM 100 System shielding effectiveness. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that of normal conditions.

#### 10.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 10.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 11 presents the results of the evaluations performed to demonstrate that the HI-STORM 100 System can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [10.0.1]. The accident events addressed in Chapter 11 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, confinement boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, partial blockage of MPC basket air inlets, and 100% blockage of air inlets.

The worst-case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-STORM overpack is the 30 day 100% blockage of air inlets event. A bounding volume of the neutron shield layer (concrete) is assumed to reach temperatures of at least 350°F in the cask body and lid. Material composition and density may be affected and therefore, the shielding effectiveness of this volume of concrete is degraded. However, even when considering worst case conservatism as discussed in Section 5.1.2, the dose rate at the controlled area boundary is bounded by the HI-TRAC loss of water in water jacket accident and does not exceed 10CFR72.106 limits.

Another accident evaluated in Section 11.2 for the loaded HI-STORM overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded. However, with over 25 inches of concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-TRAC transfer cask assumes that as a result of a fire, tornado missile, or handling accident, the all the water in the water jacket is lost. The shielding analysis of the 100-ton HI-TRAC transfer cask with complete loss of the water from the water jacket is discussed in Subsection 5.1.2. These results bound those for the 125-Ton HI-TRAC transfer cask by a large margin. The results in Subsection 5.1.2 show that there is sufficient time to implement and complete the corrective actions outlined in Chapter 11. Therefore, the dose requirement of 10CFR72.106 [10.0.1] is satisfied. Once again, this dose is calculated assuming design basis fuel in all fuel cell locations. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 10.4.1

**ANNUAL DOSE FOR ARRAYS OF HI-STORM 100S VERSION B OVERPACKS  
WITH DESIGN BASIS ZIRCALOY CLAD FUEL  
60,000 MWD/MTU AND 3-YEAR COOLING**

<b>Array Configuration</b>	<b>1 Cask</b>	<b>1 Cask</b>	<b>2x5 Array</b>
<b>Annual Dose (mrem/year)<sup>†</sup></b>	883.27	19.26	16.34
<b>Distance to Controlled Area Boundary (meters)<sup>††, †††</sup></b>	100	350	550

† 100% occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 3-year cooling as specified in the Section 2.1.9 is lower than the burnup analyzed for the design basis fuel used in this table.

Table 10.4.2  
DOSE RATE FOR THE 100-TON HI-TRAC TRANSFER CASK  
WITH DESIGN BASIS ZIRCALOY CLAD FUEL

<b>Fuel Burnup &amp; Cooling Time</b>	<b>100 Meters</b>	<b>200 Meters</b>	<b>300 Meters</b>
<b>60,000 MWD/MTU &amp; 3 Years</b>	0.91 mrem/hr	0.14 mrem/hr	0.04 mrem/hr

## 10.5 REFERENCES

- [10.0.1] *U.S. Code of Federal Regulations*, Title 10, "Energy" Part 72 "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."
- [10.1.1] *U.S. Code of Federal Regulations*, Title 10, "Energy" Part 20 "Standards for Protection Against Radiation."
- [10.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.
- [10.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May1997.

## SUPPLEMENT 10.I

### RADIATION PROTECTION

The HI-STORM 100U is a modular, underground vertical ventilated module (VVM) designed to accept all MPC models for storage at an ISFSI in lieu of above ground overpacks, like the HI-STORM 100 and HI-STORM 100S. As such, the radiological dose to plant personnel as well as members of the general public is well below those of the HI-STORM 100 and HI-STORM 100S when the MPC is in the overpack. Since the determination of off-site doses is necessarily site-specific, dose assessments similar to those described in Chapter 10 are to be prepared by the licensee as part of implementing the HI-STORM 100U System in accordance with 10CFR72.212 [10.0.1].

#### HI-STORM 100U Loading and Unloading Operations

The operations associated with the use of the HI-STORM 100U, described in Supplements 1.I and 8.I, are quite similar to the operations for all other variations of the HI-STORM 100 system. In both the aboveground and underground overpack, the MPC is transferred between the HI-TRAC and the overpack and in both cases the lid of the overpack is placed atop the overpack once the HI-TRAC is removed from the overpack. The only significant difference between the aboveground and underground overpack is the position of the HI-TRAC relative to ground level. For the aboveground overpack, the bottom of the HI-TRAC is approximately 18 feet above the ground and for the underground overpack, the bottom of the HI-TRAC is essentially at ground level. From an operations perspective, it will be easier to access the mating device and the pool lid bolts when the HI-TRAC is positioned atop the underground overpack rather than the aboveground overpack. In both cases, the same bolting and unbolting operations around the base of the HI-TRAC must be performed. Therefore, the estimated occupational dose for these scenarios is the same. The fact that the body of the HI-TRAC is closer to the ground when the underground overpack is being loaded will not affect the occupational dose rate since it is assumed that the workers not performing a task are positioned far enough away as to receive minimal dose.

Once the MPC transfer is complete and the HI-TRAC has been removed, the lid is placed on the overpack. For the underground overpack, this is a relatively simple operation of lifting the lid and placing it in the correct location. Unlike the aboveground overpack, the lid is not bolted to the body of the overpack. However, the outlet vent cover is installed on the overpack lid after the lid is placed upon the HI-STORM 100U, which installation requires bolting. Installation of the outlet vent cover places workers over the lid and adds some time to the operation. The duration of this operation can be estimated based on information provided in the tables in Section 10.3. Installation of the vent cover would be similar to the installation and alignment of the closure ring on top of the MPC. This activity is listed with an estimated duration of 5 min, for a single operator, in the tables in Section 10.3. Since the outlet vent cover is closer to the center of the lid than the closure ring, it is assumed here that two operators are required. There are four bolts, and bolt installation is typically listed in Section 10.3 to be performed at 2 bolts per minute, resulting in a duration of 2 minutes. Again, due to the location of those bolts, it is assumed that two

operators are necessary to perform this activity. In total, it is then conservatively estimated that it will take 10 minutes for two operators to perform the installation. The dose rate on top of the overpack lid is 31.53 mrem/hr (see Table 5.1.1), which translates to a dose to the individual of 5.26 mrem and a total dose of 10.51 person-mrem. This is a small increase (about 1 %) in the total dose when considering the entire MPC transfer into the HI-STORM system. However, it is recommended that the operators do not spend any unnecessary time on top of the lid to ensure/meet the ALARA principle. It should also be mentioned that actual occupational dose during loading vary widely depending on site specific conditions. Experience has shown that the dose rates are in general significantly lower than those estimated in Chapter 10 of this FSAR.

In conclusion, the operator dose rates will be similar to those described in Chapter 10 for the aboveground overpack. Therefore, occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections may be calculated using the information presented in the tables of Chapter 10 for the site-specific application of the HI-STORM 100U system. For the fuel loading/unloading, transportation, and storage operations utilizing the HI-STORM 100U, the dose information provided in Chapter 10 may be considered bounding.

### Excavation Activities

In the event it is desired to expand a loaded ISFSI utilizing the HI-STORM 100U design, excavation of material (i.e., soil) in the vicinity of the ISFSI is required. Radiation protection during excavation activities is achieved by prescribing a minimum proximity of any excavation activity to an existing operating HI-STORM 100U array. As described in 3.I.4.4 (iv), two scenarios may exist:

- (i) **No Retaining Wall Installed:** In this case, a minimum distance from the loaded ISFSI, called the Excavation Exclusion Zone (EEZ) is established based on the site specific subgrade and earthquake data using the methodology described in subsection 3.I.4.7. The EEZ ensures that the subgrade around the operating ISFSI will remain unaffected by the excavation during an earthquake. The required distance from the ISFSI for the EEZ is influenced by the properties of the subgrade and the strength of the DBE for the site. The EEZ must lie outside the Radiation Protection Space (RPS) defined as the spatial region around the ISFSI specified to provide radiation protection and safety from tornado missiles, shown on the licensing drawing.
- (ii) **Retaining Wall Installed:** If a retaining wall is installed at or beyond the Radiation Protection Space (RPS) per the licensing drawing and Table 2.I.2 then the EEZ boundary is at the retaining wall and excavation may occur up to the retaining wall as long as the depth of the excavation does not exceed the level of the bottom of the Support Foundation Pad. A retaining wall may be installed on any or all sides of the ISFSI. Site specific radiation protection measures for excavation activities need to include confirmation of the minimum soil properties along with the minimum distances between the excavation area and the loaded VVMs, as well as radiological monitoring of the excavation area.

Site specific evaluations also need to be performed to ensure that the radiation protection space boundary is maintained. Site specific accident scenarios (e.g., seismic conditions) will need to be accounted for in these evaluations. A general accident scenario evaluation, however, has been performed for the HI-STORM 100U design.

The impact of a tornado missile penetrating the soil creating a horizontal hole extending from the metal surface of the VVM to the outer surface of the soil was also considered. This evaluation, presented in Supplement 5.I, demonstrates that the dose at the site boundary is below the limit specified in 10 CFR 72.

### Normal Operation of Storage

During normal operation of storage, radiation will predominantly emanate from the inlet and outlet vents and the top of the lid. However, there are also some additional radiation streaming paths and scenarios that may have to be considered in the radiation protection program. The following two scenarios have been evaluated for the HI-STORM 100U design.

The first scenario evaluated address radiation streaming from a loaded VVM through an adjacent empty VVM. An empty VVM adjacent to a loaded VVM could potentially constitute a radiation streaming path since the soil providing shielding is limited between adjacent VVMs. Therefore, radiation passing through the soil to the unloaded VVM will have a path of less shielding and could contribute to occupational dose. This evaluation is presented in detail in Supplement 5.I, and concluded that there are no concerns about the dose rates contributing to occupational dose across the top of the empty VVM due to radiation streaming from the loaded neighboring VVM.

The second scenario concerns the soil access tube, or test station, that is part of the ICCPS design (see Figure 2.I.1) and could represent a potential streaming path. Therefore, radiation passing through the soil access tube could contribute to occupational dose. This evaluation is presented in detail in Supplement 5.I, and assumes a tube located about 5.5 feet from the center of the VVM with a diameter of 4 inches, that reaches down to the support foundation. With these dimensions, it is shown that there are no concerns about the dose rates contributing to occupational dose on the top of the soil access tube due to radiation streaming from a loaded VVM. However, if the tube is larger or located closer to the VVM, then the actual dimensions should be considered in the site specific dose rate calculations, and the result of the calculations should be considered in the site specific radiation protection program.



## **SUPPLEMENT 10.III**

### **RADIATION PROTECTION FOR THE MPC-68M**

The main body of this chapter remains fully applicable for the HI-STORM 100 System using the MPC-68M.

The MPC-68M basket is lighter than the generic MPC-68 basket. This is due to the lower density of the Metamic-HT panels as compared to stainless steel. The reduction in weight provides the user more flexibility to meet ALARA objectives. For example, additional temporary shielding may be used to further reduce the dose rates around the HI-TRAC when performing short term operations.

## CHAPTER 11<sup>†</sup>: ACCIDENT ANALYSIS

This chapter presents the evaluation of the HI-STORM 100 System for the effects of off-normal and postulated accident conditions. The design basis off-normal and postulated accident events, including those resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Subsections 2.2.2 and 2.2.3. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, confinement, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM 100 System are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and evaluations described therein.

Chapter 11 is in full compliance with NUREG-1536; no exceptions are taken.

### 11.1 OFF-NORMAL CONDITIONS

Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions, which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are listed in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM 100:

- Off-Normal Pressure
- Off-Normal Environmental Temperature
- Leakage of One MPC Seal Weld
- Partial Blockage of Air Inlets
- Off-Normal Handling of HI-TRAC Transfer Cask
- Malfunction of FHD System
- SCS Power Failure
- Off-Normal Loads<sup>‡</sup>

<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary. This chapter has been substantially revised in support of LAR#3 to enhance clarity of presentation and evaluation of results. Because of extensive editing a clean chapter is issued with this amendment.

<sup>‡</sup> Off-normal load combinations are defined in Chapter 2, Table 2.2.14 and evaluated in Chapter 3, Section 3.4.

## HI-TRAC Off-Normal Ambient Temperature

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events without affecting function, and are in compliance with the applicable acceptance criteria. The following subsections present the evaluation of the HI-STORM 100 System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.104(a) and 10CFR20.

### 11.1.1 Off-Normal Pressures

The sole pressure boundary in the HI-STORM 100 System is the MPC enclosure vessel. The off-normal pressure condition is specified in Subsection 2.2.2. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the MPC cavity under normal storage. The MPC internal pressure is evaluated with 10% of the fuel rods ruptured and 100% of the rods fill gas and 30% of the fission gases released to the cavity.

#### 11.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is low. Nonetheless, the event is postulated and evaluated.

#### 11.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure and, therefore, no monitoring is required.

#### 11.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off normal internal pressure is reported in Subsection 4.6.1 for the following conditions: limiting fuel storage scenario, tech. spec. maximum helium backfill and 10% rod rupture with 100% of rod fill gas and 30% of gaseous fission products released into the MPC cavity. The analysis shows that the MPC pressure remains below the design MPC internal pressure (Table 2.2.1).

It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would decrease the temperature rise, the calculated pressure is higher than would actually occur.

### Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

### Thermal

The MPC internal pressure for off-normal conditions is reported in Subsection 4.6.1. The design basis internal pressure for off-normal conditions used in the structural evaluation (Table 2.2.1) bounds the off-normal condition pressure. .

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100 System.

#### 11.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There is no corrective action requirement for off-normal pressure.

#### 11.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

### 11.1.2 Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (HI-STORM overpack) and 0 to 100°F (HI-TRAC transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM 100 System and must be evaluated against the allowable component design temperatures. The off-normal temperatures are evaluated against the off-normal condition temperature limits listed in Table 2.2.3.

#### 11.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

#### 11.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM overpack and MPC. Chapter 2 provides operational limitations to the use of the HI-TRAC transfer cask at temperatures of  $\leq 32^\circ\text{F}$  and prohibits use of the HI-TRAC transfer cask below  $0^\circ\text{F}$ .

#### 11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considers an environmental temperature of  $100^\circ\text{F}$  with insolation for a duration sufficient to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The Off-Normal ambient temperature condition is evaluated in Subsection 4.6.1. The results are in compliance with off-normal temperature and pressure limits in Tables 2.2.1 and 2.2.3.

The off-normal event considering an environmental temperature of  $-40^\circ\text{F}$  and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM overpack. The HI-STORM overpack and MPC are conservatively assumed to reach  $-40^\circ\text{F}$  throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is  $0^\circ\text{F}$  and the HI-TRAC is conservatively assumed to reach  $0^\circ\text{F}$  throughout the structure. For ambient temperatures from  $0^\circ$  to  $32^\circ\text{F}$ , antifreeze must be added to the demineralized water in the water jacket to prevent freezing. Chapter 3, Subsection 3.1.2, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100 System.

### Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 4.6.1, the resultant pressure is below the off-normal design pressure (Table 2.2.1). The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsection 3.1.2.

### Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Subsection 4.6.1 for the HI-STORM overpack and MPC. As can be seen from this table, all temperatures for the off-normal environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.2.3.

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100 System.

#### 11.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no corrective actions required for off-normal environmental temperatures.

#### 11.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

### 11.1.3 Leakage of One Seal

The HI-STORM 100 System has a reliable welded boundary to contain radioactive fission products within the confinement boundary. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, vent and drain port cover plates, and associated welds. The closure ring provides a redundant welded closure to the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. The welds on the vent and drain port cover plates are leakage tested. Additionally, the MPC lid weld is subjected to a pressure test to verify its integrity.

Section 7.1 provides a discussion as to how the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.

#### 11.1.3.1 Postulated Cause of Leakage of One Seal in the Confinement Boundary

There is no credible cause for the leakage of one seal in the confinement boundary. The conditions analyzed in Chapter 3 shows that the confinement boundary components are maintained within their Code-allowable stress limits under all normal and off-normal storage conditions. The MPC lid-to-shell weld meets the requirements of ISG-18, such that leakage from the confinement boundary is not considered credible. Therefore, there is no event that could cause leakage of one seal in the confinement boundary.

#### 11.1.3.2 Detection of Leakage of One Seal in the Confinement Boundary

The HI-STORM 100 System is designed such that leakage of one seal in the confinement boundary is not considered a credible scenario. Therefore, there is no requirement to detect leakage from one seal.

#### 11.1.3.3 Corrective Action for Leakage of One Seal in the Confinement Boundary

There is no corrective action required for the failure of one weld in the closure system of the confinement boundary. Leakage of one weld in the confinement boundary closure system is not a credible event.

#### 11.1.3.4 Radiological Impact of Leakage of One Seal in the Confinement Boundary

The off-normal event of the failure of one weld in the confinement boundary closure system has no radiological impact because leakage from the confinement barrier is not credible.

#### 11.1.4 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with debris screens on the inlet and outlet air ducts. These screens ensure the air ducts are protected from the incursion of foreign objects. There are four air inlet ducts 90° apart and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet ducts. As required by the design criteria presented in Chapter 2, it is conservatively assumed that two of the four air inlet ducts are blocked. The blocked air inlet ducts are assumed to be completely blocked with an ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the inherent thermal stability of the HI-STORM 100 System.

##### 11.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is conservatively assumed that the blocked air inlet ducts are completely blocked, although screens prevent foreign objects from entering the ducts. The screens are either inspected periodically or the outlet duct air temperature is monitored. It is, however, possible that blowing debris may block two air inlet ducts of the overpack.

##### 11.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet ducts will occur during the routine visual inspection of the screens or temperature monitoring of the outlet duct air. The frequency of inspection is based on an assumed complete blockage of all four air inlet ducts. There is no inspection requirement as a result of the postulated two inlet duct blockage, because the complete blockage of all four air inlet ducts is bounding.

##### 11.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

###### Structural

There are no structural consequences as a result of this off-normal event.

###### Thermal

The thermal analysis for the two air inlet ducts blocked off-normal condition is performed in Subsection 4.6.1. The analysis demonstrates that under bounding (steady-state) conditions, no system components exceed the off-normal temperature limits in Table 2.2.3.

###### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.



### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100 System.

#### 11.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet ducts does not affect the HI-STORM 100 System's ability to operate safely.

Periodic inspection of the HI-STORM overpack air duct screens is required. Alternatively, the outlet duct air temperature is monitored. The frequency of inspection is based on an assumed blockage of all four air inlet ducts analyzed in Section 11.2.

#### 11.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet ducts has no radiological impact because the confinement barrier is not breached and shielding is not affected.

#### 11.1.5 Off-Normal Handling of HI-TRAC

During unloading and/or downending of the HI-TRAC transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation.

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

This off-normal condition does not apply to the HI-TRAC 125D, which does not have lower pocket trunnions. Upending and downending of the HI-TRAC 125D is performed using an L-frame.

#### 11.1.5.1 Postulated Cause of Off-Normal Handling of HI-TRAC

If the cable of the crane handling the HI-TRAC is inclined from the vertical, it would possible to unload the upper lifting trunnions such that the lower pocket trunnions are supporting the total cask weight and the lifting trunnions are only preventing cask rotation.

#### 11.1.5.2 Detection of Off-Normal Handling of HI-TRAC

Handling procedures and standard rigging practice call for maintaining the crane cable in a vertical position by keeping the crane trolley centered over the lifting trunnions. In such an orientation it is not possible to completely unload the lifting trunnions without inducing rotation. If the crane cable were inclined from the vertical, however, the possibility of unloading the lifting trunnions would exist. It is therefore possible to detect the potential for this off-normal condition by monitoring the incline of the crane cable with respect to the vertical.

#### 11.1.5.3 Analysis of Effects and Consequences of Off-Normal Handling of HI-TRAC

If the upper lifting trunnions are unloaded, the lower pocket trunnions will support the total weight of the loaded HI-TRAC. The analysis of the pocket trunnions to support the applied load of one-half of the total weight is provided in Subsection 3.4.4. The consequence of off-normal handling of the HI-TRAC is that the pocket trunnions safely support the applied load.

#### Structural

The stress evaluations of the lower pocket trunnions are discussed in Subsection 3.4.4. All stresses are within the allowable values.

#### Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal handling of the HI-TRAC does not affect the safe operation of the system.

### 11.1.5.4 Corrective Action for Off-Normal Handling of HI-TRAC

The HI-TRAC transfer casks are designed to withstand the off-normal handling condition without any adverse effects. There are no corrective actions required for off-normal handling of HI-TRAC other than to attempt to maintain the crane cable vertical during HI-TRAC upending or downending.

### 11.1.5.5 Radiological Consequences of Off-Normal Handling of HI-TRAC

The off-normal event of off-normal handling of HI-TRAC has no radiological impact because the confinement barrier is not breached and shielding is not affected.

### 11.1.6 Malfunction of FHD System

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

#### 11.1.6.1 Postulated Cause of FHD Malfunction

Likely causes of FHD malfunction are (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in any helium leakage from the MPC or the FHD itself.

#### 11.1.6.2 Detection of FHD Malfunction

The FHD System is monitored during its operation. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

### 11.1.6.3 Analysis of Effects and Consequences of FHD Malfunction

#### Structural

The FHD System is required to be equipped with safety relief devices<sup>§</sup> to prevent the MPC structural boundary pressures from exceeding the design limits. Consequently there is no adverse effect.

#### Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit is 1058°F (Table 2.2.3). The FHD System malfunction event is evaluated assuming the following bounding conditions:

- 1) Steady state maximum temperatures have been reached
- 2) Design basis heat load
- 3) Standing column of air in the annulus
- 4) MPC's cavity pressure must be set to at least 72 psig

It is noted that operator action may be required to adjust the cavity pressure set point to ensure that condition 4 above is satisfied. This condition has been evaluated in Section 4.5. The steady state results are provided in Table 4.5.10. The results demonstrate that the peak fuel cladding temperatures remain below the off-normal limit.

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

#### Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, the structural boundary pressures cannot exceed the design limits.

#### Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

<sup>§</sup> The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.

Based on this evaluation, it is concluded that the FHD malfunction does not affect the safe operation of the HI-STORM 100 System.

#### 11.1.6.4 Corrective Action for FHD Malfunction

The HI-STORM 100 System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently no corrective action is required.

#### 11.1.6.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

#### 11.1.7 SCS Power Failure

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

##### 11.1.7.1 Postulated Cause of SCS Power Failure

The SCS is normally operated from an external source of power such as from site utilities or a feed from a heavy haul vehicle carrying the HI-TRAC. Occasional interruption in power supply is possible.

##### 11.1.7.2 Detection of SCS Power Failure

The HI-STORM 100 System is designed to withstand a power failure without affecting its ability to meet safety requirements. Consequently SCS monitoring and failure detection is not required.

##### 11.1.7.3 Analysis of Effects and Consequences of SCS Power Failure

The SCS System is required to be equipped with a backup power supply (See SCS specifications in Chapter 2, Appendix 2.C). This ensures uninterrupted operation of the SCS following a power failure. Consequently, a power failure does not affect SCS operation.

#### Structural

There is no effect on the structural integrity.

#### Thermal

There is no effect on thermal performance.

### Shielding

There is no effect on the shielding performance.

### Criticality

There is no effect on the criticality control.

### Confinement

There is no effect on the confinement function.

### Radiation Protection

As there is no effect on the shielding or confinement functions, there is no effect on occupational or public exposures.

Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.

#### 11.1.7.4 Corrective Action for SCS Power Failure

The HI-STORM 100 System is designed to withstand a power failure without an adverse effect on its normal operation. Consequently no corrective action is required.

#### 11.1.7.5 Radiological Impact of SCS Power Failure

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

#### 11.1.8 HI-TRAC Off-Normal Ambient Temperature

As evaluated in Section 4.5 the HI-TRAC is designed to conduct short term operations inside fuel handling building and in the outside environment under design basis ambient temperatures defined in Table 2.2.2. In an event where these temperatures are approached or exceeded an evaluation is required to support operations under greater than design ambient temperatures.

##### 11.1.8.1 Postulated Cause of Off-Normal Ambient Temperatures

The off-normal environmental temperature is postulated as an exceedance of design ambient temperatures caused by extreme weather conditions. To evaluate the effects of off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-TRAC temperatures to reach asymptotic steady state conditions. Because of the large

mass and thermal inertia of the HI-TRAC and the limited durations for the off-normal temperatures, this assumption is conservative.

#### 11.1.8.2 Detection of Off-Normal Ambient Temperatures

The HI-TRAC is designed with robust margins to withstand off-normal environmental temperatures without challenging short term safety limits. There is no requirement for detection of off-normal environmental temperatures for the HI-TRAC. To address potential exceedance of ambient temperatures at hot sites Appendix B of the HI-STORM Technical Specifications, Section 3.1 requires cask users to evaluate site ambient temperatures.

#### 11.1.8.3 Effects and Consequences of Off-Normal Ambient Temperatures

##### Structural

There is no effect on the structural performance of the system as a result of this off-normal event.

##### Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

##### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

##### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

##### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

##### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that off-normal environmental temperatures do not affect the safe operation of the HI-TRAC.

11.1.8.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-TRAC is designed with robust margins to withstand off-normal ambient temperatures. No corrective actions are necessary.

11.1.8.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal ambient temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.



## 11.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM 100 System or events postulated because their consequences may affect the public health and safety. Subsection 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM 100 System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. The following sections present the evaluation of the design basis postulated accident conditions and natural phenomena which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for postulated accident conditions are defined in Table 2.2.14. The load combinations include normal loads with the accident loads. The accident load combination evaluations are provided in Section 3.4.

### 11.2.1 HI-TRAC Transfer Cask Handling Accident

#### 11.2.1.1 Cause of HI-TRAC Transfer Cask Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-TRAC transfer cask is transported to the ISFSI in a vertical position. Unless the lifting device is designed in accordance with the criteria specified in Subsection 2.3.3, the height of the loaded overpack above the ground shall be limited to below the handling height limit determined in Chapter 3 to limit the inertia loading on the cask in a horizontal drop to less than 45g's. Although a handling accident is remote, a cask drop from the horizontal handling height limit is credible only if the lifting device is not designed in accordance with the criteria specified in Subsection 2.3.3. A vertical drop of the loaded HI-TRAC transfer cask is not a credible accident as the loaded HI-TRAC shall be transported and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 and a horizontal drop is precluded as HI-TRAC is transported in the vertical orientation. Nevertheless, for defense-in-depth a drop from a horizontal orientation is postulated and structural consequences evaluated. .

#### 11.2.1.2 HI-TRAC Transfer Cask Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded HI-TRAC in the horizontal position. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the HI-TRAC meets all structural requirements and there is no adverse effect on the confinement, thermal or subcriticality performance of the contained MPC. Limited localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket may occur as a result of the handling accident. The HI-TRAC top lid and transfer lid housing (pool lid for the HI-TRAC

125D) are demonstrated to remain attached by withstanding the maximum deceleration. The transfer lid doors (not applicable to HI-TRAC 125D) are also shown to remain closed during the drop. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1]. Therefore, demonstrating that the 45g limit for the HI-TRAC transfer cask is met ensures that the fuel cladding remains intact.

### Structural

The structural evaluation of the MPC for 45g's is provided in Section 3.4. As discussed in Section 3.4, the MPC stresses as a result of the HI-TRAC side drop, 45g loading, are all within allowable values.

As discussed above, the water jacket enclosure shell could be punctured which results in a loss of the water within the water jacket. Additionally, the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) are shown to remain in position under the 45g loading. Analysis of the lead in the HI-TRAC is performed in Appendix 3.F and it is shown that there is no appreciable change in the lead shielding.

### Thermal

The loss of the water in the water jacket causes the temperatures to increase due to a reduction in the thermal conductivity through the HI-TRAC water jacket. An analysis of the MPC in the HI-TRAC transfer cask temperatures with no water in the water jacket is presented in Subsection 4.6.2. The analysis results are below the short-term allowable fuel cladding and material temperatures limits for accident conditions.

### Shielding

The assumed loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded. As the structural analysis demonstrates that the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) remain in place, there is no change in the dose rates at the top and bottom of the HI-TRAC.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

## Radiation Protection

There is no degradation in the confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. The dose rates are provided in Chapter 5. Immediately after the drop accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit exposure. Based on a minimum distance to the controlled area boundary of 100 meters, the 10CFR72.106 dose rate requirements at the controlled area boundary (5 Rem limit) are not exceeded.

### 11.2.1.3 HI-TRAC Transfer Cask Handling Accident Dose Calculations

The handling accident could cause localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket as the neutron shield impacts the ground.

When the water jacket is impacted, the HI-TRAC transfer cask surface dose rate could increase. The HI-TRAC's post-accident shielding analysis presented in Chapter 5 assumes complete loss of the water in the water jacket and bounds the dose rates anticipated for the handling accident.

If the water jacket of the loaded HI-TRAC is damaged beyond immediate repair and the MPC is not damaged, the loaded HI-TRAC may be unloaded into a HI-STORM overpack, a HI-STAR overpack, or unloaded in the fuel pool. If the MPC is damaged, the loaded HI-TRAC must be returned to the fuel pool for unloading. Depending on the damage to the HI-TRAC and the current location in the loading or unloading sequence, personnel exposure is minimized by continuing to load the MPC into a HI-STORM or HI-STAR overpack. Once the MPC is placed in the HI-STORM or HI-STAR overpack, the dose rates are greatly reduced. The highest personnel exposure will result from returning the loaded HI-TRAC to the fuel pool to unload the MPC.

The analysis of the handling accident presented in Section 3.4 shows that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactive material from the confinement vessel. Any possible rupture of the fuel cladding will have no effect on the site boundary dose rates because the magnitude of the radiation source has not changed.

### 11.2.1.4 HI-TRAC Transfer Cask Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the HI-TRAC transfer cask and MPC to the maximum practical extent. As appropriate, place temporary shielding around the HI-TRAC to reduce radiation dose rates. Special handling procedures will be developed and approved by the ISFSI operator to lift and upright the HI-TRAC. Upon uprighting, the portion of the overpack not previously accessible shall be radiologically and visually inspected. If damage to the water jacket is limited to a local penetration or crushing, local repairs can be performed to the shell and the water replaced. If damage to the water jacket is extensive, the damage shall be repaired and re-tested in accordance with Chapter 9, following removal of the MPC.

If upon inspection of the damaged HI-TRAC transfer cask and MPC, damage of the MPC is observed, the loaded HI-TRAC transfer cask will be returned to the facility for fuel unloading in accordance with Chapter 8. The handling accident will not affect the ability to unload the MPC using normal means as the structural analysis of the 60g loading (HI-STAR Docket Numbers 71-9261 and 72-1008) shows that there will be no gross deformation of the MPC basket. After unloading, the structural damage of the HI-TRAC and MPC shall be assessed and a determination shall be made if repairs will enable the equipment to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the equipment for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

## 11.2.2 HI-STORM Overpack Handling Accident

### 11.2.2.1 Cause of HI-STORM Overpack Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-STORM overpack is lifted in the vertical orientation. The height of the loaded overpack above the ground shall be limited to below the vertical handling height limit determined in Chapter 3, unless the lifting device is designed in accordance with the criteria specified in Subsection 2.3.3. This vertical handling height limit will maintain the inertial loading on the cask in a vertical drop to 45g's or less. Although a handling accident is remote, a drop from the vertical handling height limit is a credible accident if the lifting device is not designed in accordance with the criteria specified in Subsection 2.3.3.

### 11.2.2.2 HI-STORM Overpack Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded overpack in the vertical orientation. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the HI-STORM 100 System meets all structural requirements and there are no adverse effects on the structural, confinement, thermal or subcriticality performance of the HI-STORM 100 System. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1].

#### Structural

The structural evaluation of the MPC under a 60g vertical load is presented in the HI-STAR FSAR and SAR [11.2.3 and 11.2.4] and it is demonstrated therein that the stresses are within allowable limits. The structural analysis of the HI-STORM overpack is presented in Section 3.4. The structural analysis of the overpack shows that the concrete shield attached to the underside of the overpack lid remains attached and air inlet ducts do not collapse.

### Thermal

As the structural analysis demonstrates that there is no adverse effect to the MPC or overpack, there is no effect on the thermal performance of the system as a result of this event.

### Shielding

As the structural analysis demonstrates that there is no adverse effect to the MPC or overpack, there is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no adverse effect on the criticality control features of the system as a result of this event.

### Confinement

There is no adverse effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the vertical drop of the HI-STORM Overpack with the MPC inside does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.2.3 HI-STORM Overpack Handling Accident Dose Calculations

The vertical drop handling accident of the loaded HI-STORM overpack will not cause any change of the shielding or breach of the MPC confinement boundary. Any possible rupture of the fuel cladding will have no affect on the site boundary dose rates because the magnitude of the radiation source has not changed. Therefore, the dose calculations are equivalent to the normal condition dose rates.

#### 11.2.2.4 HI-STORM Overpack Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, as required, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC is to be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the

MPC to return to service. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

### 11.2.3 Tip-Over

#### 11.2.3.1 Cause of Tip-Over

The analysis of the HI-STORM 100 System has shown that the overpack does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. It is highly unlikely that the overpack will tip-over during on-site movement because of the required low handling height or the use of a lifting device designed in accordance with the criteria specified in Subsection 2.3.3. The tip-over accident is stipulated as a non-mechanistic accident.

For the anchored HI-STORM designs (HI-STORM 100A and 100SA), a tip-over accident is not possible. As described in Chapter 2 of this FSAR, these system designs are not evaluated for the hypothetical tip-over. As such, the remainder of this accident discussion applies only to the non-anchored designs (i.e., the 100 and 100S designs only).

#### 11.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Section 3.4. The structural analysis provided in Appendix 3.A demonstrates that the resultant deceleration loading on the MPC as a result of the tip-over accident is less than the design basis 45g's. The analysis shows that the HI-STORM 100 System meets all structural requirements and there is no adverse effect on the structural, confinement, thermal, or subcriticality performance of the MPC. However, the side impact will cause some localized damage to the concrete and outer shell of the overpack in the radial area of impact.

#### Structural

The structural evaluation of the MPC presented in Section 3.4 demonstrates that under a 45g loading the stresses are well within the allowable values. Analysis presented in Chapter 3 shows that the concrete shields attached to the underside and top of the overpack lid remains attached. As a result of the tip-over accident there will be localized crushing of the concrete in the area of impact.

#### Thermal

The thermal analysis of the overpack and MPC is based on vertical storage. The thermal consequences of this accident while the overpack is in the horizontal orientation are bounded by the burial under debris accident evaluated in Subsection 4.6.2. Damage to the overpack will be limited as

discussed above. As the structural analysis demonstrates that there is no significant change in the MPC or overpack, once the overpack and MPC are returned to their vertical orientation there is no effect on the thermal performance of the system.

### Shielding

The effect on the shielding performance of the system as a result of this event is two-fold. First, there may be a localized decrease in the shielding thickness of the concrete in the body of the overpack. Second, the bottom of the overpack, which is normally facing the ground and not accessible, will now be facing the horizon. This orientation will increase the off-site dose rate. However, the dose rate limits of 10CFR72.106 are not exceeded.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

There is no effect on occupational or public exposures from radionuclide release as a result of this accident event since the confinement boundary integrity of the MPC remains intact.

Immediately after the tip-over accident a radiological inspection of the HI-STORM will be performed and temporary shielding shall be installed to limit exposure from direct radiation. Based on a minimum distance to the controlled area boundary of 100 meters, the 10CFR72.106 dose rate requirements at the controlled area boundary (5 Rem limit) are not exceeded.

Based on this evaluation, it is concluded that the accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.3.3 Tip-Over Dose Calculations

The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates from release of radioactivity.

The tip-over accident could cause localized damage to the radial concrete shield and outer steel shell where the overpack impacts the surface. The overpack surface dose rate in the affected area could

increase due to the damage. However, there should be no noticeable increase in the ISFSI site or boundary dose rate as a result of the localized damage on the side of the overpack.

The tip-over accident will also cause a re-orientation of the bottom of the overpack. As a result, radiation leaving the bottom of the overpack, which would normally be directed into the ISFSI pad, will be directed towards the horizon and the controlled area boundary. The dose rate at 100 meters from the bottom of the overpack, the minimum distance to the controlled area boundary, was calculated for the HI-STORM 100S Version B with an MPC-24 for an assumed accident duration of 30 days. The burnup and cooling time of the fuel was 60,000 MWD/MTU and 3 years, which is more conservative than the off-site dose analysis reported in Chapter 10, Table 10.4.1. This combination of overpack, MPC, burnup and cooling time is the same as that used in Chapters 5 and 10 for off-site dose calculations. The results presented below demonstrate that the regulatory requirements of 10CFR72.106 are easily met.

Distance	Dose Rate (mrem/hr)	Accident Duration	Total Dose (mrem)	10CFR72.106 Limit (mrem)
100 meters	2.36	720 hours or 30 days	1699.2	5000

#### 11.2.3.4 Tip-Over Accident Corrective Action

Following a tip-over accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the structural damage shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. If determined necessary, the MPC shall be returned to the facility for fuel unloading or transferred to either a HI-STAR or HI-STORM overpack in accordance with Chapter 8 for a duration that is determined to be appropriate. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs are required and will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

#### 11.2.4 Fire Accident

##### 11.2.4.1 Cause of Fire

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low, a conservative fire has been assumed and analyzed. The analysis shows that the



HI-STORM 100 System continues to perform its structural, confinement, thermal, and subcriticality functions.

#### 11.2.4.2 Fire Analysis

##### 11.2.4.2.1 Fire Analysis for HI-STORM Overpack

The possibility of a fire accident near an ISFSI is considered to be extremely remote due to an absence of combustible materials within the ISFSI and adjacent to the overpacks. The only credible concern is related to a transport vehicle fuel tank fire, causing the outer layers of the storage overpack to be heated by the incident thermal radiation and forced convection heat fluxes. The amount of combustible fuel in the on-site transporter is limited to a volume of 50 gallons.

#### Structural

As discussed in Section 3.4, there are no structural consequences as a result of the fire accident condition.

#### Thermal

Based on a conservative analysis discussed in Subsection 4.6.2, of the HI-STORM 100 System response to a hypothetical fire event, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM 100 System to cool the spent nuclear fuel within temperature limits (Table 2.2.3) during and after fire is not compromised.

#### Shielding

All the concrete remains below its short-term temperature limit. There is no adverse effect on the shielding function of the system as a result of this event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

#### Radiation Protection

Since there is a much localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.4.2.2 Fire Analysis for HI-TRAC Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 100-ton HI-TRAC is performed. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC.

##### Structural

As discussed in Section 3.4, there are no adverse structural consequences as a result of the fire accident condition.

##### Thermal

As discussed in Subsection 4.6.2, the MPC internal pressure and fuel temperature increases as a result of the fire accident. The fire accident MPC internal pressure and peak fuel cladding temperature are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).

The thermal analysis of the MPC in the HI-TRAC transfer cask under a fire accident is performed in Subsection 4.6.2. As can be concluded from the analysis, the temperatures for fuel cladding and components are below the accident temperature limits.

##### Shielding

The assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The assumed loss of all the Holtite in the 125-ton HI-TRAC lids results in an increase in the radiation dose rates at locations adjacent to the lids. The shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10CFR72.106 are not exceeded.

##### Criticality

There is no effect on the criticality control features of the system as a result of this event.

##### Confinement

There is no effect on the confinement function of the MPC as a result of this event, since the internal pressure does not exceed the accident condition design pressure and the MPC confinement boundary temperatures do not exceed the short-term allowable temperature limits.

## Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. Increases in the local dose rates adjacent to the water jacket are evaluated in Chapter 5. . Immediately after the fire accident a radiological inspection of the HI-TRAC shall be performed and temporary shielding shall be installed to limit exposure. .

### 11.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The high temperatures experienced by the HI-STORM overpack concrete are limited to the outermost layer and remain below the short term temperature limit. Therefore, there is no overall reduction in neutron shielding capabilities. The loaded HI-STORM overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC confinement boundary is not compromised and therefore, there is no release of airborne radioactive materials.

### 11.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC or HI-STORM overpack, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC as the water jacket pressure relief valves may open with resulting water loss and increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, install temporary shielding around the HI-TRAC. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC. If damage to the HI-TRAC is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced . If damage to the HI-TRAC water jacket or HI-TRAC body is widespread and/or radiological conditions require, the HI-TRAC shall be unloaded in accordance with Chapter 8, prior to repair.

If damage to the HI-STORM storage overpack as the result of a fire event is widespread and/or as radiological conditions require, the MPC shall be removed from the HI-STORM overpack in accordance with Chapter 8. However, the thermal analysis described herein demonstrates that the radial concrete remains below its design temperature. The HI-STORM overpack may be returned to service if there is no significant increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the visual inspection is satisfactory.

### 11.2.5 Partial Blockage of MPC Basket Vent Holes

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. The partial blockage of the MPC basket vent holes analyzes the effects on the HI-STORM 100 System due to the restriction of the vent openings.

#### 11.2.5.1 Cause of Partial Blockage of MPC Basket Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, dried, and backfilled with helium. There are only two possible sources of material that could block the MPC basket vent holes. These are the fuel cladding/fuel pellets and crud. Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture and that fuel cladding and fuel pellets would fall to block the basket vent holes. It is conceivable that a percentage of the crud deposited on the fuel rods may fall off of the fuel assembly and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. The HI-STORM 100 System maintains the peak fuel cladding temperature below the required long-term storage limits. All credible accidents do not cause the fuel assembly to experience an inertia loading greater than 60g's. Therefore, there is no mechanism for the extensive rupture of spent fuel rod cladding.

Crud can be made up of two types of layers, loosely adherent and tightly adherent. The SNF assembly movement from the fuel racks to the MPC may cause a portion of the loosely adherent crud to fall away. The tightly adherent crud is not removed during ordinary fuel handling operations. The MPC basket vent holes that act as the bottom plenum for the MPC internal thermosiphon are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (i.e., the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. Typically, BWR plants have more crud than PWR plants. Based on the maximum expected crud volume per fuel assembly provided in reference [11.2.2], and the area at the base of the MPC basket fuel storage cell, the maximum depth of crud at the bottom of the MPC-68 was determined. For the PWR-style MPC designs, 90% of the maximum crud volume was used to determine the crud depth. The maximum crud depths for each of the MPCs are listed in Table 2.2.8. The maximum amount of crud was assumed to be present on all fuel assemblies within the MPC. Both the tightly and loosely adherent crud was conservatively assumed to fall off of the fuel assembly. As can be seen by the values listed in the table, the maximum amount of crud depth does not block the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the available vent holes area is greater than that used in the thermal models.

#### 11.2.5.2 Partial Blockage of MPC Basket Vent Hole Analysis

The partial blockage of the MPC basket vent holes has no affect on the structural, confinement and thermal analysis of the MPC. There is no affect on the shielding analysis other than a slight increase of the gamma radiation dose rate at the base of the MPC due to the accumulation of crud. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible, and, therefore, the criticality analyses are not affected.

#### Structural

There are no structural consequences as a result of this event.

#### Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

#### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the partial blockage of MPC vent holes does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.5.3 Partial Blockage of MPC Basket Vent Holes Dose Calculations

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary. Therefore, there will be no effect on the site boundary dose rates because the magnitude of the radiation source has not changed. There will be no radioactive material release.

#### 11.2.5.4 Partial Blockage of MPC Basket Vent Holes Corrective Action

There are no consequences that exceed normal storage conditions. No corrective action is required for the partial blockage of the MPC basket vent holes.

## 11.2.6 Tornado

### 11.2.6.1 Cause of Tornado

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. Additionally, the transfer of the MPC from the HI-TRAC transfer cask to the overpack may be performed at the unsheltered ISFSI concrete pad. It is possible that the HI-STORM System (storage overpack and HI-TRAC transfer cask) may experience the extreme environmental conditions of a tornado.

### 11.2.6.2 Tornado Analysis

The tornado accident has two effects on the HI-STORM 100 System. The tornado winds and/or tornado missile attempt to tip-over the loaded overpack or HI-TRAC transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds, which attempt to penetrate the storage overpack or HI-TRAC transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC while being handled in the vertical orientation. During handling of the loaded HI-TRAC in the horizontal orientation, it is possible that the tornado missile and/or wind may cause the rollover of the loaded HI-TRAC on the transport vehicle unless the lifting device is designed in accordance with the requirements specified in Subsection 2.3.3. The horizontal drop handling accident for the loaded HI-TRAC, Subsection 11.2.1, evaluates the consequences of the loaded HI-TRAC falling from the horizontal handling height limit and consequently this bounds the effect of the roll-over of the loaded HI-TRAC on the transport vehicle.

### Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the storage overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or tornado missiles.

Analyses provided in Section 3.4 also shows that the tornado missiles do not penetrate the storage overpack or HI-TRAC transfer cask to impact the MPC. The result of the tornado missile impact on the storage overpack or HI-TRAC transfer cask is limited to damage of the shielding.

### Thermal

The thermal consequences are evaluated in Subsection 11.2.1.

### Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. Increases in the local dose rates as a result of the loss of water in the HI-TRAC water jacket are evaluated in Chapter 5. Immediately after the tornado accident a radiological inspection of the HI-TRAC shall be performed and temporary shielding shall be installed to limit the exposure to the public.

#### 11.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded storage overpack; damage the shielding materials of the overpack or HI-TRAC; or damage the MPC confinement boundary. There is no affect on the radiation dose as a result of the tornado winds. A tornado missile may cause localized damage in the concrete radial shielding of the storage overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

#### 11.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM 100 System to a tornado, the ISFSI operator shall perform a visual and radiological inspection of the overpack and/or HI-TRAC transfer cask. Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and repaired. Damage sustained by the HI-TRAC shall be inspected and repaired.

#### 11.2.7 Flood

##### 11.2.7.1 Cause of Flood

The HI-STORM 100 System will be located on an unsheltered ISFSI concrete pad. Therefore, it is possible for the storage area to be flooded. The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

#### 11.2.7.2 Flood Analysis

##### Structural

The flood accident affects the HI-STORM 100 overpack structural analysis in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood affects the MPC by applying an external pressure.

Section 3.4 provides the analysis of the flood water applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

##### Thermal

For a flood of sufficient magnitude to allow the water to come into contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a large flood will enter the annulus between the MPC and the overpack. The water would provide cooling that exceeds that available in the air filled annulus, due to water's higher thermal conductivity.

A smart flood condition that blocks the air flow but is not sufficient to allow water to come into contact with the MPC is bounded by the 100% inlet ducts blocked condition evaluated in Subsection 11.2.13.

##### Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water provides additional shielding that reduces radiation doses.

##### Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC



cavity and therefore the reactivity would be less than the loading condition in the fuel pool, which is presented in Section 6.1.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.7.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

#### 11.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM 100 System sustains no damage as a result of the flood. At the completion of the flood, exposed surfaces may need debris and adherent foreign matter removal.

### 11.2.8 Earthquake

#### 11.2.8.1 Cause of Earthquake

The HI-STORM 100 System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM 100 System, the ISFSI may experience an earthquake.

#### 11.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM 100 System. The objective is to determine the stability limits of the HI-STORM 100 System. Based on a static stability criteria, it is shown in Chapter 3 that the HI-STORM 100 System is qualified to seismic activity less than or equal to the values specified in Table 2.2.8. The analyses in Chapter 3 show that the HI-STORM 100 System will not tip over under the conditions evaluated. The seismic activity has no adverse thermal, criticality, confinement, or shielding consequences.

Some ISFSI sites will have earthquakes that exceed the seismic activity specified in Table 2.2.8. For these high-seismic sites, anchored HI-STORM designs (the HI-STORM 100A and 100SA) have been developed. The design of these anchored systems is such that seismic loads cannot result in tip-over or lateral displacement. Chapter 3 provides a detailed discussion of the anchored systems design.

### Structural

The sole structural effect of the earthquake is an inertial loading of less than 1g. This loading is bounded by the tip-over analysis presented in Subsection 11.2.3, which analyzes a deceleration of 45g's and demonstrates that the MPC allowable stress criteria are met.

### Thermal

There is no effect on the thermal performance of the system as a result of this event.

### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. If the overpack were to tip over, the resultant damage would be equal to that experienced by the tip-over accident analyzed in Subsection 11.2.3. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates or release of radioactivity.

#### 11.2.8.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have tipped-over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 11.2.3.4.

### 11.2.9 100% Fuel Rod Rupture

This accident event postulates that all the fuel rods rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity.

#### 11.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. There is no credible cause for 100% fuel rod rupture. This accident is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

#### 11.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no thermal, structural, criticality or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, which is being shielded, the shielding capability, or the criticality control features of the HI-STORM 100 System. The determination of the maximum accident pressure is provided in Chapter 4. The MPC design basis internal pressure bounds the pressure developed assuming 100% fuel rod rupture. The structural analysis provided in Chapter 3 evaluates the MPC confinement boundary under the accident condition design internal pressure.

#### Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

#### Thermal

The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.9. As can be seen from the values, the design basis accident condition MPC internal pressure (Table 2.2.1) used in the structural evaluation bounds the calculated value.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.9.3 100% Fuel Rod Rupture Dose Calculations

The MPC confinement boundary maintains its integrity. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. Therefore, there is no release of radioactive material or significant increase in radiation dose rates.

#### 11.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC confinement boundary is not damaged. The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

#### 11.2.10 Confinement Boundary Leakage

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials for postulated storage accident conditions. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The discussion contained in Chapter 7 demonstrates that MPC is designed, welded, tested and inspected to meet the guidance of ISG-18 such that leakage from the confinement boundary is considered non-credible.

##### 11.2.10.1 Cause of Confinement Boundary Leakage

There is no credible cause for confinement boundary leakage. The accidents analyzed in this chapter show that the MPC confinement boundary withstands all credible accidents. There are no man-made or natural phenomena that could cause failure of the confinement boundary restricting radioactive material release. Additionally, because the MPC satisfies the criteria specified in Interim Staff Guidance (ISG) 18, there is no credible leakage that would occur from the confinement boundary.

#### 11.2.10.2 Confinement Boundary Leakage Accident Corrective Action

The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel. No corrective actions are required.

#### 11.2.11 Explosion

##### 11.2.11.1 Cause of Explosion

An explosion within the bounds of an ISFSI is improbable since there are no explosive materials within the site boundary. An explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity the effects of an explosion on a reinforced structure are minimal. Explosions postulated to occur within or beyond the ISFSI boundary would require a site hazards evaluation under the provisions of 72.212 regulations by individual cask users.

##### 11.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external design pressure (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 11.2.7 and the tornado missile accident of Subsection 11.2.6, because explosive materials are not stored within close proximity to the casks. The bounding analysis shows that the MPC and the Overpack can withstand the effects of substantial accident external pressures without collapse or rupture.

#### Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

#### Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.11.3 Explosion Dose Calculations

The bounding external pressure load has no effect on the HI-STORM 100 overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM 100 System is experienced as a result of the explosion pressure load. The effects of explosion generated missiles on the HI-STORM 100 System structure is bounded by the analysis of tornado generated missiles.

#### 11.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by the explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM 100 System. Following an explosion, the ISFSI operator shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the concrete material may be replaced and the outer shell repaired.

#### 11.2.12 Lightning

##### 11.2.12.1 Cause of Lightning

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad. There is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

### 11.2.12.2 Lightning Analysis

The HI-STORM 100 System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM 100 System is hit with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, will provide a direct path to ground.

The MPC provides the confinement boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

#### Structural

There is no structural consequence as a result of this event.

#### Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

#### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STORM 100 System.

### 11.2.12.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the confinement boundary or shielding materials. Therefore, no further analysis is necessary.

#### 11.2.12.4 Lightning Accident Corrective Action

The HI-STORM 100 System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

#### 11.2.13 100% Blockage of Air Inlets

##### 11.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all four bottom inlets. Such blockage of the inlets may be postulated to occur as a result of a flood, blizzard snow accumulation, tornado debris, or volcanic activity.

##### 11.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet ducts is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass, and correspondingly large thermal capacity of the storage overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for a number of days. This accident condition is, however, a short duration event that will be identified and corrected by scheduled periodic surveillance at the ISFSI site depending on the cask heat load at the time of inspection. The temperature rise due to this accident event is small for heat loads much lower than design maximum heat load even if the condition persists for a substantial number of days. As evaluated in Sub-section 4.6.2.4, mandatory 30-day surveillance of casks is required under heat loads less than or equal to the threshold heat load specified in Table 4.6.8a at the time of inspection.

#### Structural

There are no structural consequences as a result of this event. However, since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, structural evaluation of a missile impact coincident with the 100% vent blockage event is evaluated in Section 3.4.8.1 to demonstrate safety of the system.

#### Thermal

A thermal analysis is performed in Subsection 4.6.2 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and component



temperatures remain below their temperature limits. The MPC internal pressure for this event is evaluated in Subsection 4.6.2 and is bounded by the design basis internal pressure for accident conditions (Table 2.2.1).

Since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, additional thermal evaluations are performed in Section 4.6.2.5 to demonstrate that the MPCs are safe at off-normal and accident conditions coincident with the 100% vent blockage event at threshold heat load.

### Shielding

For a short duration event there is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

For the duration of a 30-day event, this accident results in an increase in the radiation dose rates at 100 m. The shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10CFR72.106 are not exceeded.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100 System, if the blockage is removed in the specified time period.

#### 11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets 32-hour accident, the shielding capabilities of the HI-STORM 100 System are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

For the 100% blockage of air inlets 30-day accident, the material composition and density of portion of the concrete of the HI-STORM 100 System may be affected. 100 m dose rates will increase as a result of this event. Shielding analysis documented in Section 5.1.2 shows that the resulting dose rate is bounded by the HI-TRAC loss of water in water jacket accident condition and therefore is in compliance with the 10CFR72.106 limits.

#### 11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within the required action completion times. Upon detection of the complete blockage of the air inlet ducts, the ISFSI operator shall assign personnel to clear the blockage with mechanical and manual means as necessary. After clearing the overpack ducts, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air ducts of a cask with heat loads greater than the threshold heat load (Table 4.6.8a) at the time of occurrence for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to demonstrate adequate heat removal for the duration of the event. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and Overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain below their short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet ducts and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet ducts using pumps or fire-hoses or blowing air into the air outlet ducts using fans, to directly cool the MPC.

#### 11.2.14 Burial Under Debris

##### 11.2.14.1 Cause of Burial Under Debris

Burial of the HI-STORM System under debris is not a credible accident. During storage at the ISFSI, there are no structures over the casks. The minimum regulatory distance(s) from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM overpack air inlet ducts has already been considered in Subsection 11.2.13.

#### 11.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM System does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure encountered during the flood bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STORM System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limit during a burial under debris accident.

##### Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

##### Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised.

##### Shielding

There is no effect on the shielding performance of the system as a result of this event.

##### Criticality

There is no effect on the criticality control features of the system as a result of this event.

##### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

##### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100 System, if the debris is removed within the specified time. The 24-hour minimum duct inspection interval ensures that a burial under debris condition will be detected long before the allowable burial time is reached.

#### 11.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

#### 11.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures are not exceeded even for an extended duration of burial. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the performance of this corrective action.

#### 11.2.15 Extreme Environmental Temperature

##### 11.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

##### 11.2.15.2 Extreme Environmental Temperature Analysis

The accident condition considering an extreme environmental temperature (Table 2.2.2) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3.

#### Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

### Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in evaluation performed in Subsection 4.6.2. As concluded from this evaluation, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

### Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature and the dose calculations are equivalent to the normal condition dose rates.

#### 11.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

#### 11.2.16 Supplemental Cooling System (SCS) Failure

The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation. Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated herein.

##### 11.2.16.1 Cause of SCS Failure

Possible causes of SCS failure are: (a) Simultaneous loss of external and backup power, or (b) Complete loss of annulus water from an uncontrolled leak or line break.

##### 11.2.16.2 Analysis of Effects and Consequences of SCS Failure

###### Structural

See discussion under thermal evaluation below.

###### Thermal

In the event of a SCS failure due to (a), the following sequence of events occur:

- i) The annulus water temperature rises to reach its boiling temperature (~212°F).
- ii) A progressive reduction of water level and dryout of the annulus.

In the event of an SCS failure due to (b), a rapid water loss occurs and annulus is replaced with air. This condition is evaluated in Section 4.5.5.3.

###### Shielding

There is no adverse effect on the shielding effectiveness of the system.

###### Criticality

There is no adverse effect on the criticality control of the system.

###### Confinement

There is no adverse effect on the confinement function of the MPC. As discussed in the evaluations above, the structural boundary pressures are within design limits.

## Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.

### 11.2.16.3 SCS Failure Dose Calculations

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

### 11.2.16.4 SCS Failure Corrective Action

In the vertical orientation the HI-TRAC is designed to withstand an SCS failure without an adverse effect on its safety functions.

11.3        REFERENCES

- [11.2.1]      Chun et al., “Dynamic Impact Effects on Spent Fuel Assemblies,” Lawrence Livermore National Laboratory, UCID-21246, (October 1987).
  
- [11.2.2]      ESEERCO Project EP91-29 and EPRI Project 3100-02, “Debris Collection System for Boiling Water Reactor Consolidation Equipment,” B&W Fuel Company, (October 1995).
  
- [11.2.3]      Docket Number 72-1008, HI-STAR 100 System FSAR, Holtec Report HI-2012610, Revision 2.
  
- [11.2.4]      Docket Number 71-9261, HI-STAR 100 System SAR, Holtec Report HI-951251, Revision 10.



## SUPPLEMENT 11.I

### ACCIDENT EVALUATION FOR THE HI-STORM 100U SYSTEM

#### 11.I.0 INTRODUCTION

This supplement is focused on the off-normal and accident condition evaluations of the HI-STORM 100U vertical ventilated module (VVM). Only those events that are actually affected by the design of the overpack are discussed in detail herein. The reader is referred to the main body of Chapter 11 for discussions of any off-normal or accident conditions that are not dependent on the design of the storage overpack (i.e., MPC-only or HI-TRAC events).

The evaluations described herein parallel those of the HI-STORM 100 overpack contained in the main body of Chapter 11 of this FSAR. To ensure readability, the sections in this supplement are numbered to be directly analogous to the sections in the main body of the chapter. For example, the fire accident evaluation presented in Supplement Subsection 11.I.2.4 for the HI-STORM 100U is analogous to the evaluation presented in Subsection 11.2.4 of the main body of Chapter 11 for the HI-STORM 100. Tables and figures (if any) in this supplement, however, are labeled sequentially by section. If there is an analogous table or figure in the main body of Chapter 11, an appropriate notation is made in the supplement table or figure.

#### 11.I.1 OFF-NORMAL EVENTS

A general discussion of off-normal events is presented in Section 11.1 of the main body of Chapter 11. The following off-normal events are discussed in this supplement:

- Off-Normal Pressure
- Off-Normal Environmental Temperature
- Leakage of One MPC Seal Weld
- Partial Blockage of Air Inlets
- Off-Normal Handling of HI-TRAC Transfer Cask
- Malfunction of FHD System
- SCS Power Failure
- Off-Normal Wind
- HI-TRAC Off-Normal Ambient Temperature

The results of the evaluations presented herein demonstrate that the HI-STORM 100U System can withstand the effects of off-normal events without affecting its ability to perform its intended function, and is in compliance with the applicable acceptance criteria.

##### 11.I.1.1 Off-Normal Pressure

A discussion of this off-normal condition is presented in Subsection 11.1.1 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

#### Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The applicable pressure boundary stress limits are confirmed to bound the stresses resulting from the off-normal pressure.

#### Thermal

In 4.6.1 the MPC internal pressure under the conditions of 10% fuel rods ruptured, insolation and a limiting fuel storage configuration in an aboveground overpack is evaluated. This evaluation is bounding as the MPC temperatures in the 100U overpack are bounded by the aboveground overpack.

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation mentioned above, all stresses remain within allowable values, assuring confinement boundary integrity.

#### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.1.2 Off-Normal Environmental Temperatures

A discussion of this off-normal condition is presented in Subsection 11.1.2 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

### Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. However, as shown previously the resultant pressure is below the off-normal design pressure (Table 2.2.1). The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsections 3.1.2 and 3.I.1.

### Thermal

Supplement 4.I calculates bounding temperatures and pressures for the HI-STORM 100U under the elevated temperature condition. The calculated temperatures and pressures are reported in Table 4.I.5 and are below the off-normal limits (Tables 2.2.3, 2.I.8 and 2.2.1).

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM 100U overpack. The HI-STORM 100U overpack is conservatively assumed to reach -40°F throughout the structure. Chapter 3, Subsection 3.1.2 details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100U System.

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100U System.

#### 11.I.1.3 Leakage of One MPC Seal Weld

A discussion of this off-normal condition is presented in Subsection 11.1.3 of the main body of Chapter 11. The discussion presented therein is applicable in its entirety to an MPC in a HI-STORM 100U VVM as well.

#### 11.I.1.4 Partial Blockage of Air Inlets

A discussion of this off-normal condition is presented in Subsection 11.1.4 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

#### Structural

There are no structural consequences as a result of this off-normal event.

#### Thermal

Supplement 4.I calculates bounding temperatures for 50% blockage of the air inlets. The calculated bounding temperatures are reported in Table 4.I.6 and are below the MPC and VVM off-normal design temperatures (Tables 2.2.3 and 2.I.8). Additionally, the increased temperatures generate an elevated MPC internal pressure, also reported in Table 4.I.6, which is less than the off-normal design pressure (Table 2.2.1).

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

#### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.1.5 Off-Normal Handling of HI-TRAC

A discussion of this off-normal condition is presented in Subsection 11.1.5 of the main body of Chapter 11. The discussion presented therein remains completely applicable, as the design and method of operation of the HI-TRAC is the same as with the HI-STORM 100U.

#### 11.I.1.6 Failure of FHD System

A discussion of this off-normal condition is presented in Subsection 11.1.6 of the main body of Chapter 11. The discussion presented therein remains completely applicable for all MPCs.

#### 11.I.1.7 SCS Power Failure

A discussion of this off-normal condition is presented in Subsection 11.1.7 of the main body of Chapter 11. The discussion presented therein remains completely applicable to all MPCs.

#### 11.I.1.8 Off-Normal Wind

The HI-STORM 100U is designed for use at any site in the United States. Supplement 4.I evaluates the effects of off-normal wind (>0 and up to 15 MPH). The off-normal wind is postulated as a constant horizontal wind caused by extreme weather conditions (see Table 2.I.1). To determine the effects of the off-normal wind, it is conservatively assumed that these winds persist for a sufficient duration to allow the HI-STORM 100U System to reach thermal equilibrium. Because of the large mass of the HI-STORM 100U System with its corresponding large thermal inertia and the unlikely condition of a unidirectional wind for a long period of time, this assumption is conservative. The analyses presented in Supplement 4.I shows that the peak fuel cladding and material temperatures remains below the off-normal limits (Tables 2.2.3 and 2.I.8). Because the HI-STORM 100U System is designed to withstand the off-normal wind without any effect on its ability to maintain safe storage conditions, there is no requirement for detection of the off-normal wind.

#### Structural

There are no structural consequences as a result of this off-normal event.

#### Thermal

Supplement 4.I calculates peak fuel cladding temperatures for horizontal wind speeds of up to 15 miles per hour. The calculated temperatures (reported in Table 4.I.7) are below the off-normal limits (Table 2.2.3).

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal wind event does not affect the safe operation of the HI-STORM 100U System. The HI-STORM 100U System is designed to withstand the off-normal wind without any effect on its ability to maintain safe storage conditions. There are no corrective actions required for the off-normal wind. The off-normal wind has no radiological impact, and the confinement barrier and shielding integrity are not affected.

#### 11.I.1.9 HI-TRAC Off-Normal Ambient Temperature

A discussion of this off-normal condition is presented in Subsection 11.1.8 of the main body of Chapter 11. The discussion presented therein remains completely applicable as the design and method of operation of the HI-TRAC under the HI-STORM 100U is same as under the aboveground system.

#### 11.I.2 ACCIDENT EVENTS

A general discussion of accident events is presented in Section 11.1 of the main body of Chapter 11. The following accident events are discussed in this supplement section:

- HI-TRAC Transfer Cask Handling Accident
- HI-STORM 100U Overpack Handling Accident
- Tip-Over
- Fire Accident
- Partial Blockage of MPC Basket Vent Holes
- Tornado
- Flood
- Earthquake
- 100% Fuel Rod Rupture
- Confinement Boundary Leakage
- Explosion
- Lightning
- 100% Blockage of Air Inlets

Burial Under Debris  
Extreme Environmental Temperature  
SCS Failure

The results of the evaluations performed herein demonstrate that the HI-STORM 100U System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and is in compliance with the applicable acceptance criteria.

In addition to the above accidents events, identification of additional hazards during construction proximate to an operating ISFSI is treated in 11.I.2.17.

11.I.2.1 HI-TRAC Transfer Cask Handling Accident

A discussion of this accident condition is presented in Subsection 11.2.1 of the main body of Chapter 11. The discussion presented therein is applicable in its entirety, as the design and method of operation of the HI-TRAC is the same for the HI-STORM 100U.

11.I.2.2 HI-STORM Overpack Handling Accident

This accident event is not applicable to the HI-STORM 100U as this is an underground overpack surrounded by soil.

11.I.2.3 Tip-Over

This accident event is not applicable to the HI-STORM 100U. Due to the subterranean installation of the VVM with a surrounding subgrade for lateral support, tip-over is precluded.

11.I.2.4 Fire Accident

A discussion of this accident condition is presented in Subsection 11.2.4 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein. In addition, the discussion of the fire analysis for the HI-TRAC transfer cask presented therein remains completely applicable, as the design and method of operation of the HI-TRAC do not need to be changed for use with the HI-STORM 100U.

Structural

There are no structural consequences as a result of the fire accident condition.

Thermal

Supplement 4.I discusses the impact of a fire on the HI-STORM 100U System. As justified therein, the evaluation for the fire effects on an aboveground cask presented in Section 11.2 bound the effects on the HI-STORM 100U System. As described in Section 11.2, the effects of the fire do not cause

any system component or the contained fuel to exceed any design limit. As such, the results are bounding for the HI-STORM 100U System.

### Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: “The loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR.”.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

### Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.5 Partial Blockage of MPC Basket Vent Holes

A discussion of this accident condition is presented in Subsection 11.2.5 of the main body of Chapter 11. The discussion presented therein is applicable in its entirety to an MPC in a HI-STORM 100U VVM.

#### 11.I.2.6 Tornado

A discussion of this accident condition is presented in Subsection 11.2.6 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Because of its underground construction, the HI-STORM 100U is not affected by the tornado wind. The effect of tornado missiles propelled by high velocity winds that attempt to penetrate the exposed portions of the HI-STORM 100U must, however, be considered.



### Structural

Analyses presented in Supplement 3.I show that the impact of an intermediate tornado missile on the HI-STORM 100U closure lid does not result in the perforation of the lid or result in a structural collapse. The result of the tornado missile impact on the VVM is limited to localized damage of the shielding.

### Thermal

There are no thermal consequences as a result of the tornado beyond those discussed for the wind herein.

### Shielding

A tornado missile may cause localized damage to the HI-STORM 100U closure lid. As the HI-STORM 100U top is heavily shielded (a thick MPC lid backed up by a steel-concrete-steel top) the overall damage consequences (site boundary dose) are insignificant.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

### Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. A tornado missile may cause localized damage in the HI-STORM 100U closure lid. However, the damage will have a negligible effect on the site boundary dose.

Based on this evaluation, it is concluded that the tornado accident does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.7 Flood

A discussion of this accident condition is presented in Subsection 11.2.7 of the main body of Chapter 11. A description of the cause of this event is presented therein.

### Structural

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

### Thermal

The thermal consequences of flood are bounded by the 100% air inlets blockage accident (see Subsection 4.I.6.2).

### Shielding

There is no effect on the shielding performance of the system as a result of this accident event. The floodwater provides additional shielding which reduces radiation dose.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the spent fuel pool, which is presented in Section 6.1.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100U System.

### Flood Accident Corrective Action

The configuration of the VVM makes it uniquely suited to withstand a flooding event. Indeed, introducing water in the CEC is an effective method to lower the MPC contents' temperature. However, solid debris packed around the Divider Shell is an undesirable condition. Thus, while the thermal evaluations discussed in Supplement 4.I demonstrate that the HI-STORM 100U System will safely withstand a flood, corrective actions after such an event may be necessary. Periodic VVM air

temperature monitoring, required for the HI-STORM 100U System, will identify any blockage of the cooling passages that results in a non-normal thermal condition, including blockages due to a flood borne debris.

If the measured temperature rise exceeds the allowable value, then corrective actions to alleviate the condition will be required. To restore the system to a normal configuration, all flood water and any debris deposited by the receding water must be removed. The specific methods to be used are appropriately site specific and shall be addressed in the site emergence action plan. Examples of acceptable cleaning approaches include:

1. The MPC is removed from the VVM using the HI-TRAC transfer cask, allowing direct access to the interior of the VVM through both the inlet vents and the top of the module cavity. Water sprays and vacuuming is used to directly clean the VVM passages and surfaces.
2. Appropriate vacuuming equipment is inserted through the inlet ducts and down to the bottom plenum. Water is sprayed in through the outlet vents. Remote cameras are used to inspect the VVM cooling passages to identify debris and remove debris.

The adequacy of the cooling passages clearance operation is verified by visual inspection or, if the optional air temperature monitoring is used, the return of the air outlet temperatures to within allowable limits.

#### 11.I.2.8 Earthquake

A discussion of this accident condition is presented in Subsection 11.2.8 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

#### Structural

Because of its underground construction, the HI-STORM 100U VVM is inherently safe under seismic events. Analyses presented in Supplement 3.I show that the VVM will continue to render its intended function under a seismic event whose ZPAs are bounded by the values set forth in Supplement 2.I.

#### Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.9 100% Fuel Rod Rupture

A discussion of this accident condition is presented in Subsection 11.2.9 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

### Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

### Thermal

A bounding MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.9. The design basis accident condition MPC internal pressure (Table 2.2.1) used in the structural evaluation bounds the calculated value.

### Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.10 Confinement Boundary Leakage

A discussion of this accident condition is presented in Subsection 11.2.10 of the main body of Chapter 11. The discussion presented therein remains completely applicable to an MPC in a HI-STORM 100U VVM as well.

#### 11.I.2.11 Explosion

A discussion of this accident condition is presented in Subsection 11.2.11 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

### Structural

Because of its underground construction, the HI-STORM 100U and the MPC contained within are essentially shielded by the surrounding earth. Thus, no evaluation of the VVM or the contained MPC is required. The HI-STORM 100U closure lid is, however, aboveground and exposed to the explosion-induced pressure wave. Supplement 3.I includes an evaluation of the effect of the design-basis 10 psi pressure wave applied as a static pressure on the closure lid. This evaluation shows that the overpressure wave does not result in lid separation, and that all lid stresses are a fraction of the allowable limits.

### Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

### Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain well within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.12 Lightning

A discussion of this accident condition is presented in Subsection 11.2.12 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Because of its underground construction, the subterranean portion of the HI-STORM 100U would not be subjected to a direct lightning strike. The HI-STORM 100U closure lid is, however, aboveground and could be subjected to a direct strike. The closure lid is, however, a steel encased concrete structure just like on the aboveground casks. Thus, the discussion presented in Subsection 11.2.12 remains completely applicable to the exposed portions of the HI-STORM 100U System. Therefore, it is concluded that a lightning event will not prevent the VVM from rendering its intended function.

#### 11.I.2.13 100% Blockage of Air Inlets

A discussion of this accident condition is presented in Subsection 11.2.13 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

### Structural

There are no structural consequences as a result of this accident event.

### Thermal

Supplement 4.I calculates bounding temperatures for the 100% blockage of the air inlets. The calculated bounding temperatures after 24 hours of 100% blockage are reported in Table 4.I.9. The results are below the MPC and VVM accident temperature limits (Tables 2.2.3 and 2.I.8). Additionally, the increased temperatures generate an elevated MPC internal pressure, also reported in Table 4.I.9, which is less than the design basis accident pressure listed in Table 2.2.1.

### Shielding

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperatures do not exceed the accident temperature limit.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100 System, if the blockage is removed in the specified time period.

#### 11.I.2.14 Burial Under Debris

A discussion of this accident condition is presented in Subsection 11.2.14 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

### Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

### Thermal

Supplement 4.I discusses the impact of burial under debris on the HI-STORM 100U System. As explained therein, the evaluation for the effects of such an event on an aboveground cask presented in Section 11.2 bound the HI-STORM 100U.

### Shielding

There is no adverse effect on the shielding performance of the system as a result of this accident event.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100U System, if the debris is removed within the specified time period.

#### 11.I.2.15 Extreme Environmental Temperature

A discussion of this accident condition is presented in Subsection 11.2.15 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

### Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by the design-basis internal pressure and are well within the allowable values, as discussed in Section 3.4.

### Thermal



Supplement 4.I calculates bounding temperatures for the HI-STORM 100U under the extreme environmental temperature condition. The calculated bounding temperatures and pressures are reported in Table 4.I.8 and are below the MPC and VVM accident temperature and pressure limits (Tables 2.2.3, 2.I.8 and 2.2.1).

### Shielding

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100U System.

#### 11.I.2.16 Supplemental Cooling System (SCS) Failure

A discussion of this accident condition is presented in Subsection 11.2.16 of the main body of Chapter 11. The discussion presented therein remains completely applicable, as the design and method of operation of the SCS and the HI-TRAC is unchanged for use with the HI-STORM 100U System.

#### 11.I.2.17 Additional Hazards during Construction Proximate to the ISFSI

To protect an installed ISFSI from any site construction activity in its proximity, a certain minimum ground buffer distance beyond the edge of the perimeter of the VVM arrays shall be established. This distance, referred to as the Excavation Exclusion Zone (EEZ), is the minimum distance from the ISFSI where excavation can occur if a retaining wall was not installed at the ISFSI per the licensing drawing and Table 2.I.2. This distance is established based on soil conditions and the strength of the DBE as discussed in Section 3.I. If the user installs a retaining wall at or beyond the Radiation Protection Space (RPS) then the EEZ boundary is the retaining wall (see Section 2.I.2, item vi). If a

retaining wall is not installed, the EEZ boundary for a site is established using the methodology described in Section 3.I.4.

As is required for deploying casks certified under 10CFR72, Subpart L, every site modification that may potentially impact the continued operability of the ISFSI must be evaluated for acceptability under 10CFR72.212. A generic evaluation of the shielding consequences of digging a cavity adjacent to the RPS has been considered in Supplement 5.I of this FSAR. The analyses show that the dose at the edge of the cavity is below 0.2 mrem/hr, which is well below the customary limit that requires radiation posting at nuclear power plants.

Subsection 2.I.6 considers loadings from extreme environmental phenomena assuming that a deep cavity at the edge of the RPS perimeter with a retaining wall has been created as a part of site construction work and an accidental mechanical loading event across such cavity is credible. Analyses summarized in Subsection 3.I.4 show that the design basis projectiles (large, medium, or small), specified in Chapter 2 of this FSAR, applied in the most vulnerable location of the construction cavity, will fail to reach the CEC.

In addition to the generic analyses documented in this FSAR to validate the sufficiency of the RPS boundary, analyses of the consequences of any credible site specific loads or events during site construction work shall be performed with due consideration of the duration and nature of the site construction activity. The user's §72.212 evaluation program, used in considering ISFSI-proximate activities at aboveground ISFSIs, shall apply to the HI-STORM 100U installation as well without limitation.

To summarize, as discussed in Supplement 2.I and documented in the licensing drawing package in Section 1.5, and the technical specifications; an RPS has been established per Supplement 5.I with sufficient margin (ground buffer) against design basis projectiles analyzed in Supplement 3.I. An EEZ shall be established within which excavation activities cannot be performed. If the retaining wall is present at or beyond the RPS the EEZ boundary is located at the retaining wall. The RPS boundary and EEZ shall not be encroached upon during any site construction activity (this includes excavation). In addition to the generic analyses documented in this FSAR, site specific evaluation pursuant to §72.212 shall be performed for all other credible hazards that can be postulated during site construction. Administrative controls to guard against accidental human error in excavations (such as encroachment of the RPS) shall be addressed through written procedures consistent with the required controls needed for a safety significant activity within a Part 50 controlled area.

Subsection 2.I.2(iv) also requires the ISFSI owner to perform a seismic analysis of the ISFSI for the instance when the maximum amount of excavation of the area adjacent to the EEZ will exist to show that the RPS will not be encroached upon if a retaining wall is not used or be limited to excavation at a distance ten times the planned excavation depth from the ISFSI. The site's Design Basis Earthquake (DBE) will be used. PRA considerations shall not be used to diminish the strength of the seismic input. The Design Basis Seismic Model, described in 3.I.4, shall be used with appropriate representation of the construction cavity.

Because the actual projectiles for a specific ISFSI site are often different from the tornado borne missiles analyzed in Supplement 3.I herein, a site specific analysis of the effect of all credible missiles shall be performed assuming that the largest construction cavity adjacent to the ISFSI exists. PRA considerations shall not be used to rule out any missile that has been determined to be credible in the plant's FSAR.

Furthermore, the ISFSI owner shall implement ameliorative measures to prevent unacceptable damage to the ISFSI from any other credible adverse scenarios unique to a site that has not been considered in this FSAR. An example of such a measure is the installation of a berm to protect against environmental events such as soil erosion and mud slides. Such site specific design initiatives at any "100U" ISFSI, like its aboveground counterpart, are within the purview of the plant's §72.212 process.

## **SUPPLEMENT 11.III**

### **ACCIDENT ANALYSIS OF THE MPC-68M**

#### **11.III.0      INTRODUCTION**

The MPC-68M is canister which contains a Metamic-HT basket. The Metamic-HT basket has improved heat transfer as well as criticality control. This supplement presents the evaluation of the MPC-68M of the HI-STORM 100 System for the effects of off-normal and postulated accident conditions.

The supplemental information has been presented in the same format as the main body of Chapter 11. The basic changes that have been made on the MPC-68 are discussed in Supplement 1.III. The significant changes include the fuel basket and the precision extruded aluminum shims between the fuel basket and the Enclosure Vessel. The changes result in significant reduction in bending stresses under a canister slap-down event, improved heat transfer and criticality performance of the system.

All the off-normal conditions and the postulated accident conditions have been critically evaluated for the introduction of the MPC-68M. The structural integrity and the thermal performance of the HI-STORM 100 System with the MPC-68M are discussed in Supplement Sections 3.III and 4.III.

This Supplement is in full compliance with NUREG-1536; no exceptions are taken.

#### **11.III.1      OFF-NORMAL CONDITIONS**

All the off-normal conditions considered in FSAR Chapter 11 have been evaluated for MPC-68M. The Supplemental Cooling System (SCS) is not required as explained in Supplement Section 4.III. Therefore, the SCS power failure is not applicable. The thermal consequences of the off-normal conditions for the HI-STORM 100 system with the MPC-68M are presented below.

Supplement Section 4.III.6 provides details of the thermal evaluations for the off-normal conditions.

##### **11.III.1.1      Off-Normal Pressure**

The off-normal pressure of the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the cavity under normal storage as described in Sub-section 11.1.1. The normal storage temperature and pressure in the MPC-68M are reported in Section 4.III.4. FSAR Section 11.2 is applicable.

##### **11.III.1.2      Off-Normal Environmental Temperatures**

This off-normal condition is the same as that described in Sub-section 11.1.2. The off-normal environment temperatures are evaluated in Section 4.III.6. All the temperatures are below their respective temperature limit presented in Table 2.2.3. FSAR Section 11.1 is applicable.

11.III.1.3      Leakage of One Seal

FSAR Section 11.1.3 is applicable.

11.III.1.4      Partial Blockage of Air Inlets

This off-normal condition is the same as that described in Sub-section 11.1.4. Supplement Section 4.III.6 confirms that under bounding (steady state) conditions, all system components are below the off-normal temperature limits presented in Table 2.2.3. FSAR Section 11.1 is applicable.

11.III.1.5      Off-Normal Handling of HI-TRAC

FSAR Section 11.1.5 is applicable.

11.III.1.6      Malfunction of FHD System

FSAR Section 11.1.6 is applicable.

11.III.1.7      SCS Power Failure

As mentioned earlier the Supplementary Cooling System (SCS) is not necessary and therefore, the SCS power failure is not applicable for MPC-68M.

## 11.III.2 ACCIDENTS

All the accident conditions considered in the FSAR Chapter 11 have been evaluated for the HI-STORM 100 system with MPC-68M. It has been demonstrated that the consequences of the accident conditions are either the same or bounded by those presented in Chapter 11. Some additional discussions are provided below.

### 11.III.2.1 HI-TRAC Transfer Cask Handling Accident

Supplement Section 3.III.4 evaluated the potential consequences of the HI-TRAC drop events for a loaded MPC-68M inside the HI-TRAC and concluded that the drop analyses presented in the FSAR Subsection 3.4.9 are valid.

### 11.III.2.2 HI-STORM Overpack Handling Accident

Supplement Section 3.III.4 evaluated the potential consequences of the drop events for HI-STORM overpack carrying the loaded MPC-68M and concluded that although the resulting deceleration is higher than the maximum deceleration calculated in the FSAR Appendix 3.A, it is still less than the design basis vertical deceleration limit.

### 11.III.2.3 Tip-Over

Supplement Section 3.III.4 evaluated the potential consequences of the non-mechanistic tip-over of the HI-STORM overpack the loaded MPC-68M on to the ISFSI pad carrying. It is concluded that the resulting deceleration is less than the design basis deceleration limit. Also the total deformations of the basket walls during accident conditions are far below the limit imposed in the structural analysis in Supplement 3.III and analyzed in the criticality analysis in Supplement 6.III.

### 11.III.2.4 Fire Accident

It is concluded in the Supplement Section 4.III.6 that the thermal consequences of fire accident are similar to those presented in the FSAR Section 4.6. Therefore, FSAR Section 11.2 is applicable.

### 11.III.2.5 Partial Blockage of MPC Basket Vent Holes

The MPC-68M basket vent holes are located above the level of the maximum accumulation of debris. The blockage of the vent holes is not credible. The evaluation in the FSAR Section 11.2.5 can be considered as bounding.

### 11.III.2.6 Tornado

It is concluded in Supplement Section 3.III.4 that the tornado loads due to missile impact/wind and depressurization presented in FSAR Subsection 3.4.8 are bounding for the MPC-68M inside a HI-

STORM overpack. It is also concluded that the missile penetration analyses presented in Subsection 3.4.8 remain valid. Therefore, FSAR Section 11.2 is applicable.

#### 11.III.2.7 Flood

The flood accident is defined and evaluated in Sub-section 11.2.7. A smart flood condition that blocks the air flow but is not sufficient to allow water to come in contact with the MPC is bounded by the 100% inlet ducts blocked condition evaluation in Supplement Section 4.III.6. FSAR Section 11.2 is applicable.

#### 11.III.2.8 Earthquake

It is concluded in Supplement Section 3.III.4 that the seismic analyses presented in FSAR Chapter 3 bound the scenario of a loaded MPC-68M inside a HI-STORM overpack. Therefore, FSAR Section 11.2 is applicable.

#### 11.III.2.9 100% Fuel Rod Rupture

It is concluded in the Supplement Section 4.III.6 that the thermal consequences of the 100% fuel rod rupture are bounded by those presented in the FSAR Section 4.6. Therefore, FSAR Section 11.2 is applicable.

#### 11.III.2.10 Confinement Boundary Leakage

FSAR Section 11.2 is applicable.

#### 11.III.2.11 Explosion

It is concluded in Supplement Section 3.III.4 that the seismic analyses presented in FSAR Chapter 3 bound the scenario of a loaded MPC-68M inside a HI-STORM overpack. Therefore, FSAR Section 11.2 is applicable.

#### 11.III.2.12 Lightning

FSAR Section 11.2 is applicable.

#### 11.III.2.13 100% Blockage of Air Inlets

This accident condition is the same as that defined in Sub-section 11.2.13. The thermal consequences of 100% blockage of air inlets are presented in the Supplement Section 4.III.6. All the component temperatures and MPC cavity pressure are below their respective accident temperature and pressure limit. FSAR Section 11.2 is applicable.

11.III.2.14 Burial Under Debris

This accident condition is the same as that defined in Sub-section 11.2.14. The thermal consequences of burial under debris are presented in the Supplement Section 4.III.6. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised. FSAR Section 11.2 is applicable.

11.III.2.15 Extreme Environmental Temperature

This accident condition is the same as that defined in Sub-section 11.2.15. The thermal consequences of extreme environmental temperature are presented in the Supplement Section 4.III.6. As concluded from this evaluation, all temperatures are within the accident condition allowable values specified in Table 2.2.43. FSAR Section 11.2 is applicable.

11.III.2.16 Supplemental Cooling System (SCS) Failure

As discussed earlier the Supplemental Cooling System (SCS) is not required for the HI-STORM 100 system with the MPC-68M canister.



## **CHAPTER 12<sup>†</sup>: OPERATING CONTROLS AND LIMITS**

### **12.0 INTRODUCTION**

The HI-STORM 100 System provides passive dry storage of spent fuel assemblies in interchangeable MPCs with redundant multi-pass welded closure. The loaded MPC is enclosed in a single-purpose vertical ventilated module or overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STORM 100 System at an ISFSI. The information provided in this Chapter is in full compliance with NUREG-1536 [12.1.1].

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

## 12.1 PROPOSED OPERATING CONTROLS AND LIMITS

### 12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

12.1.1.1 This portion of the FSAR establishes the commitments regarding the HI-STORM 100 System and its use. Other 10CFR72 [12.1.2] and 10CFR20 [12.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [12.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM 100 System. The general license conditions governed by 10CFR72 [12.1.2] are not repeated with these Technical Specifications. Licensees are required to comply with all commitments and requirements.

12.1.1.2 The Technical Specifications provided in Appendix A and A-100U to CoC 72-1014 and the authorized contents and design features provided in Appendix B and B-100U to CoC 72-1014 are primarily established to maintain subcriticality, confinement boundary and intact fuel cladding integrity, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 12.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 12.1.2 provides the list of Technical Specifications for the HI-STORM 100 System.

Table 12.1.1  
HI-STORM 100 SYSTEM CONTROLS

Condition to be Controlled	Applicable Technical Specifications <sup>†</sup>
Criticality Control	3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) 3.1.4 Supplemental Cooling System
Shielding and Radiological Protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down 3.2.2 TRANSFER CASK Surface Contamination 5.4 Radioactive Effluent Control Program 5.7 Radiation Protection Program
Heat Removal Capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System 3.1.4 Supplemental Cooling System
Structural Integrity	3.1.5 Impressed Current Cathodic Protection System 5.5 Cask Transport Evaluation Program

<sup>†</sup> Technical Specifications are located in Appendix A to CoC 72-1014. Authorized contents are specified in FSAR Section 2.1.9

Table 12.1.2  
HI-STORM 100 SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION 1.1    Definitions 1.2    Logical Connectors 1.3    Completion Times 1.4    Frequency
2.0	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.1.4	Supplemental Cooling System
3.1.5	Impressed Current Cathodic Protection System
3.2.1	Deleted
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	Deleted
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINISTRATIVE CONTROLS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	Deleted
5.7	Radiation Protection Program
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements

## 12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR.

### 12.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM 100 Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM 100 System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important to Safety (overview);
4. HI-STORM 100 System Final Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STORM 100 Technical Specifications, Approved Contents, Design Features and other Conditions for Use;
8. HI-STORM 100 Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Operating Experience Reviews
11. HI-STORM 100 System and ISFSI Procedures, including
  - Procedural overview
  - Fuel qualification and loading
  - MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
  - MPC welding operations
  - HI-TRAC/overpack/VVM closure
  - Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, supplemental cooling (if used), and cooldown)

- MPC/HI-TRAC/overpack/VVM pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack/VVM for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack/VVM
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

### 12.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM 100 System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The dry run shall include, but is not limited to the following:

1. Receipt inspection of HI-STORM 100 System components.
2. Moving the HI-STORM 100 MPC/HI-TRAC into the spent fuel pool.
3. Preparation of the HI-STORM 100 System for fuel loading.
4. Selection and verification of specific fuel assemblies to ensure type conformance.
5. Locating specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.
6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.
7. Replacing the HI-TRAC pool lid with the transfer lid (HI-TRAC 100 and 125 only).
8. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup may be used).
9. HI-TRAC upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.
10. Placement of the HI-STORM 100 System at the ISFSI.
11. HI-STORM 100 System unloading, including cooling fuel assemblies, flooding the MPC cavity, and removing MPC welds (for which a mock-up may be used).

## 12. Installation and operation of the Supplemental Cooling System (if used).

### 12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STORM 100 System is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system to verify operability of the overpack heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

### 12.2.4 Limiting Conditions for Operation

Limiting Conditions for Operation specify the minimum capability or level of performance that is required to assure that the HI-STORM 100 System can fulfill its safety functions.

### 12.2.5 Equipment

The HI-STORM 100 System and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown in this FSAR that no credible condition or event prevents the HI-STORM 100 System from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

### 12.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM 100 System fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Section 2.1.9 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

### 12.2.7 Design Features

This section describes HI-STORM 100 System design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in this FSAR and in Appendix B to CoC 72-1014, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM 100 System is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 9.

### 12.2.8 MPC

- a. Basket material composition, properties, dimensions, and tolerances for criticality control.

- b. Canister material mechanical properties for structural integrity of the confinement boundary.
- c. Canister and basket material thermal properties and dimensions for heat transfer control.
- d. Canister and basket material composition and dimensions for dose rate control.

#### 12.2.9 HI-STORM Overpack/VVM

- a. HI-STORM overpack/VVM material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations, as applicable.
- b. HI-STORM overpack/VVM material thermal properties and dimensions for heat transfer control.
- c. HI-STORM overpack/VVM material composition and dimensions for dose rate control.

#### 12.2.10 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below execute the approach and equations described in Section 2.1.9.1 for determining allowable decay heat per storage location, burnup, and cooling time for the approved cask contents.

##### Example 1

In this example, a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern and will be stored in an aboveground HI-STORM system.

Step 1: Pick a value of X between 0.5 and 3. For this example X will be 2.8.

Step 2: Calculate  $q_{\text{Region2}}$  as described in Section 2.1.9.1.2:

$$q_{\text{Region2}} = (2 \times 34) / [(1 + (2.8)^{0.2075}) \times ((12 \times 2.8) + 20)] = 0.5668 \text{ kW}^\dagger$$

Step 3: Calculate  $q_{\text{Region1}}$  as described in Section 2.1.9.1.2:

$$q_{\text{Region1}} = X \times q_{\text{Region2}} = 2.8 \times 0.5668 = 1.5871 \text{ kW}$$

† Results are arbitrarily rounded to four decimal places.



Step 4: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 as described in Section 2.1.9.1.3.

Table 12.2.3 provides three hypothetical fuel assemblies in the 16x16A array/class that will be evaluated for acceptability for loading in the MPC-32 example above. The decay heat values in Table 12.2.3 are calculated by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is not acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.1 and 12.1.2. The solution is to develop another table using an enrichment of 3.0 wt.%  $^{235}\text{U}$  or less to determine this fuel assembly's suitability for loading in this MPC-32.

Fuel Assembly Number 2 is not acceptable for loading unless a unique maximum allowable burnup for a cooling time of 3.3 years is calculated by linear interpolation between the values in Table 12.2.1 for 3 years and 4 years of cooling. Linear interpolation yields a maximum burnup of 36,497 MWD/MTU (rounded down from 36,497.2), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.

Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment than those in Table 12.2.1, which is still below the actual initial enrichment of Fuel Assembly Number 3. Due to its relatively low total decay heat of 0.5 kW (fuel: 0.4, non-fuel hardware: 0.1), Fuel Assembly Number 3 may be stored in Region 1 or Region 2.

### Example 2

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern in an aboveground system. Assuming the same value 'X', the same maximum fuel storage location decay heats are calculated. The equation in Section 2.1.9.1.3 is executed for each fuel assembly using its exact initial enrichment to determine its maximum allowable burnup. Linear interpolation is used to further refine the maximum allowable burnup value between cooling times, if necessary.

Fuel Assembly Number 1: The calculated allowable burnup for 3.0 wt.%  $^{235}\text{U}$  and a decay heat value of 1.5871 kW ( $q_{\text{region1}}$ ) is 44,905 MWD/MTU at 4 years minimum cooling. Its decay heat is too high for loading in Region 2. Comparing the fuel assembly burnup and total decay heat of the contents<sup>†</sup> (fuel (1.01 kW) plus non-fuel hardware (0.5 kW)) to the calculated limits indicates that the fuel assembly, including the non-fuel hardware, is acceptable for storage in Region 1.

<sup>†</sup> The assumption is made that the non-fuel hardware meets burnup and cooling time limits in Table 2.1.25.

Fuel Assembly Number 2: The calculated allowable burnup for 3.2 wt.%  $^{235}\text{U}$  and a decay heat value of 1.5871 kW ( $q_{\text{region1}}$ ) is 32,989 MWD/MTU for 3 years cooling and 45,382 MWD/MTU for 4 years cooling. Linearly interpolating between these values for a cooling time of 3.3 years yields a maximum allowable burnup of 36,706 MWD/MTU and, therefore, the assembly is acceptable for storage in Region 1. This fuel assembly's decay heat is also too high for loading in Region 2.

Fuel Assembly Number 3: The calculated allowable maximum burnup for 4.3 wt.%  $^{235}\text{U}$  and a decay heat value of 0.5668 ( $q_{\text{Region2}}$ ) is 41,693 MWD/MTU for 18 years cooling. Comparing the fuel assembly burnup and total decay heat of the contents (fuel plus non-fuel hardware) against the calculated limits indicates that the fuel assembly and non-fuel hardware are acceptable for storage. Therefore, the assembly is acceptable for storage in Region 2. This fuel assembly would also be acceptable for loading in Region 1 (this conclusion is inferred, but not demonstrated).

### Example 3

In this example, a demonstration of the use of burnup versus cooling time tables for uniform fuel loading is provided. In this example it will be assumed that the MPC-68 is being loaded with array/class 9x9A fuel and will be stored in an aboveground HI-STORM system.

Step 1: CoC TS Appendix B Table 2.4-1 provides the heat load limit on each storage location ( $q_{\text{max}}$ ). For MPC-68 this is 0.5 kW.

Step 2: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments if the fuel being loaded varies significantly in initial enrichment. It is conservative to choose the lowest value of initial enrichment to generate the table.

In this example, two enrichments will be used: 3.0 and 4.5. Tables 12.2.4 and 12.2.5 show the burnup versus cooling time tables calculated for these enrichments for the respective  $q_{\text{max}}$ .

Table 12.2.6 provides three hypothetical fuel assemblies in the 9x9A array/class that will be evaluated for acceptability for loading in the MPC-68 example above. The decay heat values in Table 12.2.6 would be calculated by the user. The other information would be taken from the fuel assembly and reactor operating records.

All of the assemblies meet the per cell heat load limit of 0.5 kW.

Fuel Assembly Number 1 is acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.4 and the burnup is lower than that allowed for the cooling time of the assembly.

Fuel Assembly Number 2 is not acceptable for loading based on the current tables. The fuel assembly burnup is greater than allowed by Table 12.2.4, even with linear interpolation (30978

MWD/MTU). Fuel Assembly Number 2 may be acceptable for loading if a new table is created specifically for an initial enrichment of 3.5 wt% and the allowable burnup is greater than 35250.

Fuel Assembly Number 3 is acceptable for loading based on the allowable burnups in Table 12.2.5.

#### 12.2.11 Verifying Compliance with Total MPC Heat Load

Some operational steps and/or use of particular equipment are required if  $Q_{CoC}$  is above a certain value, e.g. 28.74 kW in the MPC-32. These include supplemental cooling, forced helium dehydration, helium backfill pressure, and surveillance requirements for LCO 3.1.2. These examples demonstrate the logic behind the decisions for these operational steps. Time to boil limits and vacuum drying are also considered in these examples.

Example 1:

Table 12.2.7 contains a proposed heat load pattern for loading a MPC-68 (non-duplex) into an aboveground HI-STORM 100 System. The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are  $\leq 0.50$  kW (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 18.917 kW.
4. The maximum heat load in any cell is 0.460 kW.
5.  $Q_{CoC}$ , as defined in Section 2.1.9.1.2 equation c is 31.280 kW.

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying because the  $Q_{CoC}$  heat load is greater than 30 kW (See FSAR Table 4.5.1).
2. Forced Helium Dehydration: The MPC should be dried using forced helium dehydration since the  $Q_{CoC}$  heat load exceeds the vacuum drying threshold heat loads (See FSAR Table 4.5.1).
3. Helium Backfill Pressure Range: The MPC should be backfilled to the higher pressure range given in the TS because the  $Q_{CoC}$  heat load exceeds the heat load in Table 3-2 of the CoC Appendix A and FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system would be required for on-site transport of High Burnup Fuel in the HI-TRAC after the MPC is dried, backfilled and sealed because the  $Q_{CoC}$  heat load exceeds the 90% design basis

threshold heat load in FSAR Table 2.1.26 and per cell limits in CoC Appendix B Table 2.4-1.

5. Heat Removal Surveillance (LCO 3.1.2): The user has 24 hours to clear blockage on the system containing this MPC since the  $Q_{CoC}$  heat load (assuming the pattern is at the time of inspection) exceeds the 28.152 kW ( $=0.414 \text{ kW} \times 68$ ) threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 18.917 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 18.917 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

#### Example 2

Table 12.2.8 contains a proposed heat load pattern for loading a MPC-32 (non-duplex). The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are  $\leq 1.062 \text{ kW}$  (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 17.471 kW.
4. The maximum heat load in any cell is 0.826 kW.
5.  $Q_{CoC}$ , as defined in Section 2.1.9.1.2 equation c is 26.432 kW.

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC can be dried using vacuum drying since the  $Q_{CoC}$  heat load is bounded by the threshold heat load  $Q_2$  in FSAR Table 4.5.1. The vacuum drying is time limited as  $Q_{CoC}$  exceeds threshold heat load  $Q_1$  in FSAR Table 4.5.1.
2. Forced Helium Dehydration: The MPC can be dried using forced helium dehydration but it is not required.
3. Helium Backfill Pressure Range: The MPC may be backfilled to either pressure range given in the TS because the  $Q_{CoC}$  heat load is bounded by the heat load limits in Table 3-2 of CoC Appendix A and FSAR Table 1.2.2.
4. Supplemental Cooling System: Since the maximum cell heat load is less than 90% of the design basis heat load per cell limit in FSAR Table 2.1.30, supplemental cooling

is not required for on-site transport in the HI-TRAC if the higher helium backfill range in FSAR Table 1.2.2 is used. However, the maximum cell heat load is higher than 90% of the design basis heat load per cell limit in FSAR Table 2.1.31, so supplemental cooling would be required if the MPC contains High Burnup Fuel and is backfilled to the lower helium backfill range in FSAR Table 1.2.2.

5. Heat Removal Surveillance (LCO 3.1.2): The user has 64 hours to clear blockage on the system containing this MPC since the  $Q_{CoC}$  heat load (assuming the pattern is at the time of inspection) is bounded by the 28.74 kW threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 17.471 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 17.471 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

### Example 3

Table 12.2.9 contains a proposed heat load pattern for loading a MPC-32 (non-duplex). The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

### General observations on this loading plan:

1. The heat loads do not meet the CoC limits for Uniform Loading, i.e. some cells are  $\geq 1.0625$  kW (See Table 2.1.26).
2. The X value that most closely meets this pattern (See Table 2.1.30) is 1.5 which means the inner locations cannot have a decay heat greater than 1.282 kW for each cell and the outer locations cannot have a decay heat greater than 0.854 kW for each cell. Note that the pattern also meets the criteria for any X value  $\geq 1.5$ .
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 20.697 kW.
4. The maximum heat load in any cell is 1.273 kW.
5. Since this MPC is loaded in a regionalized pattern,  $Q_{CoC}$ , as defined in Section 2.1.9.1.2 equation e is 32.484 kW. ( $12*1.282+20*0.855$ )

### Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying since the  $Q_{CoC}$  heat load under uniform loading (1.273 kWx32 equals 40.736 kW) exceeds the threshold heat loads in FSAR Table 4.5.1.
2. Forced Helium Dehydration: The MPC must be dried using forced helium dehydration only because vacuum drying is not permitted (see above) and regionalized loading  $Q_{CoC}$  is bounded by the design basis heat load in FSAR Table

- 4.5.1.
3. Helium Backfill Pressure Range: The MPC must be backfilled to the higher pressure range given in the TS because the  $Q_{CoC}$  heat load is greater than the limit given in CoC Appendix A Table 3-2 and FSAR Table 1.2.2. Also the inner region has cells that exceed the per cell heat load limit for lower pressure range given in CoC Appendix A Table 3-3 and FSAR Table 2.1.31.
  4. Supplemental Cooling System: A supplemental cooling system is not required for on-site transport of High Burnup Fuel in the HI-TRAC after the MPC is dried, backfilled and sealed because storage cell heat loads under regionalized storage do not exceed the 90% design basis threshold heat load in FSAR Table 2.1.30 for conditions of  $X \geq 2$ .
  5. Heat Removal Surveillance (LCO 3.1.2): The user has 24 hours to clear blockage on the system containing this MPC since the uniform loading  $Q_{CoC}$  heat load (assuming the pattern is at the time of inspection) exceeds the 28.74 kW threshold heat load in LCO 3.1.2.
  6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 20.697 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
  7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 20.697 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

Table 12.2.1

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment = 3.1 wt.% <sup>235</sup>U)  
 ( $q_{\text{Region 1}} = 1.5871 \text{ kW}$ ,  $q_{\text{Region 2}} = 0.5668 \text{ kW}$ )

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)</b>
≥3	32791	10896
≥4	45145	17370
≥5	53769	22697
≥6	59699	26615
≥7	63971	29386
≥8	67343	31437
≥9	68200	33000
≥10	68200	34271
≥11	68200	35384
≥12	68200	36322
≥13	68200	37189
≥14	68200	37980
≥15	68200	38773
≥16	68200	39512
≥17	68200	40234
≥18	68200	40908
≥19	68200	41620
≥20	68200	42324

Table 12.2.2

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment = 4.185 wt.% <sup>235</sup>U)  
 ( $q_{\text{Region 1}} = 1.5871 \text{ kW}$ ,  $q_{\text{Region 2}} = 0.5668 \text{ kW}$ )

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)</b>
≥3	34797	11101
≥4	47590	17870
≥5	56438	23272
≥6	62533	27157
≥7	66963	29907
≥8	68200	31935
≥9	68200	33510
≥10	68200	34785
≥11	68200	35927
≥12	68200	36894
≥13	68200	37790
≥14	68200	38593
≥15	68200	39419
≥16	68200	40191
≥17	68200	40937
≥18	68200	41643
≥19	68200	42363
≥20	68200	43094



Table 12.2.3

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE  
(Array/Class 16x16A)

<b>FUEL ASSEMBLY NUMBER</b>	<b>ENRICHMENT (wt. % <sup>235</sup>U)</b>	<b>FUEL ASSEMBLY BURNUP (MWD/MTU)</b>	<b>FUEL ASSEMBLY COOLING TIME (years)</b>	<b>FUEL ASSEMBLY DECAY HEAT (kW)</b>	<b>NON-FUEL HARDWARE STORED WITH ASSEMBLY</b>	<b>NFH DECAY HEAT (kW)</b>
1	3.0	37100	4.7	1.01	BPRA	0.5
2	3.2	35250	3.3	1.45	NA	NA
3	4.3	41276	18.2	0.4	BPRA	0.1

Table 12.2.4

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-68, Array/Class 9x9A, and Enrichment = 3.0 wt.%  $^{235}\text{U}$ )  
 ( $q_{\text{max}} = 0.5 \text{ kW}$ )

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP (MWD/MTU)</b>
$\geq 3$	27739
$\geq 4$	38536
$\geq 5$	46268
$\geq 6$	51583
$\geq 7$	55424
$\geq 8$	58303
$\geq 9$	60733
$\geq 10$	62798
$\geq 11$	64609
$\geq 12$	66331
$\geq 13$	68005
$\geq 14$	68200
$\geq 15$	68200
$\geq 16$	68200
$\geq 17$	68200
$\geq 18$	68200
$\geq 19$	68200
$\geq 20$	68200

Table 12.2.5

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-68, Array/Class 9x9A, and Enrichment =4.5 wt.% <sup>235</sup>U)  
 ( $q_{\max} = 0.5$  kW)

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP (MWD/MTU)</b>
≥3	30017
≥4	41399
≥5	49359
≥6	54839
≥7	58856
≥8	61932
≥9	64534
≥10	66802
≥11	68200
≥12	68200
≥13	68200
≥14	68200
≥15	68200
≥16	68200
≥17	68200
≥18	68200
≥19	68200
≥20	68200

Table 12.2.6

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE  
(Array/Class 9x9A)

<b>FUEL ASSEMBLY NUMBER</b>	<b>ENRICHMENT (wt. % <sup>235</sup>U)</b>	<b>FUEL ASSEMBLY BURNUP (MWD/MTU)</b>	<b>FUEL ASSEMBLY COOLING TIME (years)</b>	<b>FUEL ASSEMBLY DECAY HEAT (kW)</b>
1	3.0	37100	4.7	0.3
2	3.5	35250	3.3	0.495
3	4.5	41276	18.2	0.2

Table 12.2.7  
 Loading Map for Storage Cells in an MPC-68  
 Total Decay Heat (kW)

				0.171	0.170				
		0.175	0.171	0.186	0.183	0.180	0.176		
	0.171	0.300	0.313	0.315	0.310	0.309	0.303	0.169	
	0.172	0.312	0.450	0.456	0.449	0.444	0.302	0.176	
0.170	0.181	0.304	0.446	0.457	0.454	0.454	0.307	0.181	0.169
0.159	0.183	0.317	0.441	0.439	0.460	0.441	0.313	0.186	0.170
	0.174	0.315	0.440	0.437	0.445	0.443	0.311	0.181	
	0.161	0.303	0.304	0.316	0.320	0.301	0.302	0.182	
		0.166	0.173	0.181	0.183	0.182	0.178		
				0.169	0.155				

Table 12.2.8  
 Loading Map for Storage Cells in an MPC-32  
 Total Decay Heat (kW)

	0.370	0.394	0.392	0.387	
0.395	0.372	0.81	0.817	0.391	0.372
0.388	0.810	0.818	0.826	0.825	0.375
0.382	0.820	0.823	0.821	0.812	0.378
0.384	0.390	0.813	0.824	0.393	0.395
	0.373	0.372	0.375	0.374	

Table 12.2.9  
 Loading Map for Storage Cells in an MPC-32  
 Total Decay Heat (kW)

	0.310	0.349	0.329	0.378	
0.359	0.327	1.110	1.107	0.399	0.372
0.388	1.068	1.207	1.250	1.091	0.373
0.328	1.086	1.273	1.203	1.087	0.374
0.348	0.309	1.100	1.090	0.319	0.375
	0.337	0.327	0.357	0.367	

### 12.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM 100 System are provided in Appendix A to Certificate of Compliance 72-1014. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to CoC 72-1014. Bases applicable to the Technical Specifications are provided in FSAR Appendix 12.A. The format and content of the HI-STORM 100 System Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, “Writer’s Guide for the Restructured Technical Specifications” [12.3.1] was used as a guide in the development of the Technical Specifications and Bases.



## 12.4 REGULATORY EVALUATION

Table 12.1.2 lists the Technical Specifications for the HI-STORM 100 System. The Technical Specifications are detailed in Appendix A and A-100U to Certificate of Compliance 72-1014. The Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B and B-100U to CoC 72-1014.

The conditions for use of the HI-STORM 100 System identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and cask design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM 100 System will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

12.5        REFERENCES:

- [12.1.1]    U.S. Nuclear Regulatory Commission, NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, Final Report, January 1997.
- [12.1.2]    U.S. Code of Federal Regulations, Title 10, Energy, Part 72, Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."
- [12.1.3]    U.S. Code of Federal Regulations, Title 10, Energy, Part 20, Standards for Protection Against Radiation."
- [12.3.1]    Nuclear Management and Resources Council, Inc. – Writer’s Guide for the Restructured Technical Specifications, NUMARC 93-03, February 1993.

**HI-STORM 100 SYSTEM FSAR**

**APPENDIX 12.A**

**TECHNICAL SPECIFICATION BASES**

**FOR THE HOLTEC HI-STORM 100 SPENT FUEL STORAGE CASK SYSTEM**

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## B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

### BASES

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LCOs LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

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LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).

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LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the

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## BASES

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### LCO 3.0.2 (continued)

remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

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### LCO 3.0.3

This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

---

### LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM 100 System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

- a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to

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## BASES

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### LCO 3.0.4 (continued)

exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continuing with dry fuel storage activities for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the dry storage system. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

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### LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing to demonstrate:

- a. The equipment being returned to service meets the LCO; or
- b. Other equipment meets the applicable LCOs.

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## BASES

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LCO 3.0.5  
(continued)

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

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## B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

### BASES

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SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

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SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

- a. The systems or components are known to not meet the LCO, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the HI-STORM 100 System is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances

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## BASES

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SR 3.0.1  
(continued)

are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary dry storage cask system parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow dry fuel storage activities to proceed to a specified condition where other necessary post maintenance tests can be completed.

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SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension

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## BASES

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### SR 3.0.2 (continued)

to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

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### SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of HI-STORM 100 System conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

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## BASES

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### SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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### SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe conduct of dry fuel storage activities.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is

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## BASES

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### SR 3.0.4 (continued)

outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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## B 3.1 SFSC Integrity

### B 3.1.1 Multi-Purpose Canister (MPC)

#### BASES

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**BACKGROUND** A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain port cover plates and closure ring are installed and welded. Inspections are performed on the welds.

MPC cavity moisture removal using vacuum drying or forced helium dehydration is performed to remove residual moisture from the MPC cavity space after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel and by the heat added to the MPC from the optional warming pad, if used.

If forced helium dehydration is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

After the completion of drying, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

(continued)

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## BASES

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### BACKGROUND (continued)

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Backfilling the MPC with helium in the required quantity eliminates air inleakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage.

### APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC overpressurization evaluation.

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(continued)

## BASES (continued)

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LCO A dry, helium filled and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the multiple confinement boundaries. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.

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APPLICABILITY The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure both the confinement barriers and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored fuel.

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ACTIONS A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

### A.1

If the cavity vacuum drying pressure or demoinsturizer exit gas temperature limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

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**BASES**

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**ACTIONS**  
(continued)A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the MPC cavity vacuum drying acceptance criterion is not met during the allowable time, the Required Action ensures a sufficient quantity of helium within the MPC cavity to provide additional margin to the PCT limits. The Completion Time is sufficient to complete the corrective action commensurate with the safety significance of the CONDITION.

(continued)

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## BASES

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### ACTIONS (continued)

#### C.1

If the helium backfill quantity limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

#### C.2

Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition either by adding or removing helium or by demonstrating through analysis that all cask system limits will continue to be met. Since the quantity of helium estimated under Required Action C.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

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**BASES**

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**ACTIONS**  
(continued)D.1

If the helium leak rate limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM OVERPACK is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

D.2

Once the consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

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## BASES

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### ACTIONS (continued)

#### E.1

If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid (if required), perform fuel cooldown operations (if required), re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

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## BASES

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### SURVEILLANCE REQUIREMENTS SR 3.1.1.1 , SR 3.1.1.2, and SR 3.1.1.3

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. For moderate burnup fuel cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the gas temperature or dew point at the specified location reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demister exit temperature meeting the acceptance limit is an indication that the cavity is dry. For high burnup fuel and high decay heat load MPCs, the forced helium dehydration method of moisture removal must be used to provide necessary cooling of the fuel during drying operations. Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations.

Table 3-1 of Appendix A to the CoC provides the appropriate requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high) and the applicable short-term temperature limit. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11.

Having the proper quantity of helium in the MPC ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC.

To determine the proper quantity of helium in the MPC, if the cask heat load is less than or equal to the value in Table 3-2 of Appendix A to the CoC and the storage cell heat loads are less than or equal to the limits in either Table 3-3 of

(continued)

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## BASES

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### SURVEILLANCE REQUIREMENTS SR 3.1.1.1, SR 3.1.1.2 , and SR 3.1.1.3 (continued)

Appendix A to the CoC (regionalized) or Table 3-4 of Appendix A to the CoC (uniform), then the lower helium backfill pressure range in Table 3-2 item (i) can be used. The higher backfill pressure range in Table 3-2 item (ii) must be used if the cask heat load is greater than the value in Table 3-2 and the storage cell heat load is greater than the value in either Table 3-3 or Table 3-4. Note that the higher backfill pressure range in Table 3-2 item (ii) is just a subset of the wider range in item (i), and therefore can always be used as an option. The storage cell heat load limits specified in Table 3-3 and Table 3-4 for MPC-68/68F/68FF are also applicable to the MPC-68M, consistent with the analyses in the FSAR.

Meeting the helium leak rate limit ensures there is adequate helium in the MPC for long term storage and that there is no credible effluent dose from the cask.

All of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the cask design.

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REFERENCES	<ol style="list-style-type: none"> <li>1. FSAR Sections 1.2, 4.4, 4.5, 7.2, 7.3 and 8.1</li> <li>2. Interim Staff Guidance Document 11</li> <li>3. Interim Staff Guidance Document 18</li> <li>4. Deleted</li> </ol>
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## B 3.1 SFSC Integrity

### B 3.1.2 SFSC Heat Removal System

#### BASES

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**BACKGROUND** The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system that ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the air inlets. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the outlet air ducts at the top of the OVERPACK.

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**APPLICABLE SAFETY ANALYSIS** The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the air inlets are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.

Analyses have been performed for the half and complete obstruction of all the air inlets. Blockage of half of the air inlets reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits.

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(continued)

## BASES

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APPLICABLE  
SAFETY  
ANALYSIS  
(continued)

The complete blockage of all air inlets stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the duration of the analyzed event.

---

LCO

The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. Operability is defined as at least 50% of the inlet and outlet air ducts are available for air flow (i.e., unblocked). Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.

The intent of this LCO is to address those occurrences of air duct blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished quickly to restore the heat removal system to operable status (e.g., removal of loose debris).

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## BASES

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### LCO

(continued)

This LCO is not intended to address low frequency, unexpected Design Event III and IV class events such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.9 of Appendix B to the CoC.

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### APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed into its storage configuration, the heat removal system must be operable to ensure adequate dissipation of the decay heat from the fuel assemblies.

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### ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

#### A.1

Although the heat removal system remains operable, the blockage should be cleared expeditiously.

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## BASES

ACTIONS  
(continued)B.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within the Completion Time per the Table in the CoC for OVERPACKS containing MPCs with heat loads in excess of the heat loads in Table B.1-1 (below) at the time of inspection. This is a reasonable period of time to take action to remove the obstructions in the air flow path.

Table B.1-1 (Threshold* heat loads for HI-STORM 100 System Surveillance Frequency and Completion Time to restore heat removal system to operable status)		
<b>MPC Model(s)</b>	<b>Threshold Heat Load (per canister)</b>	<b>Threshold Heat Load (per assembly)</b>
24 (all variants)	18 kW	0.75 kW
68 (all variants)	18 kW	0.264 kW
32 (all variants)	16 kW	0.5 kW

Alternatively, for OVERPACKS containing MPCs with heat loads up to the thresholds in Table B.1-1 at the time of inspection, the system must be restored to operable status within twenty-four hours. Twenty-four hours is a reasonable period of time for these lower heat load systems since the temperature limits of the system components and fuel cladding are not exceeded and the event is not time limiting.

(continued)

**BASES**

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**ACTIONS**  
(continued)C.1

For MPCs with heat loads greater than the thresholds in Table B.1-1, if the heat removal system cannot be restored to operable status within Completion Time per the Table in the CoC, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the cask user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.

For MPCs with heat loads less than or equal to the threshold heat loads in Table B.1-1, there will be inconsequential temperature increase to the OVERPACK concrete if the system is not restored to operable status within 24 hours. If the heat removal system cannot be restored to operable status within 24 hours, the same actions as above are required.

(continued)

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**BASES**

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**ACTIONS****C.2.1**

In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented.

This Required Action must be complete within Completion Time per the Table in the CoC for all aboveground systems and in 16 hours for an underground system. These Completion Times are consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits after event initiation. For MPC heat loads up to the thresholds in Table B.1-1, system components temperatures do not exceed their 30-day accident temperature limits.

The Completion Time reflects the time to complete Required Action B.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all air ducts becoming simultaneously blocked by trash or debris.

(continued)

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## BASES

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### ACTIONS

#### C.2.2

In lieu of implementing Required Action C.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK. In this case, the requirements of CoC Appendix A, LCO 3.1.4 apply.

An engineering evaluation must be performed to determine if any concrete deterioration has occurred which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

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## BASES

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### ACTIONS

#### C.2.2 (continued)

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Times per the Table in the CoC reflect the Completion Time from Required Action C.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads and OVERPACK styles.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.1.2

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all air inlets and outlets are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Greater than 50% blockage of the total air inlet area or air outlet area renders the heat removal system inoperable and this LCO is not met. 50% or less blockage of the total air inlet area or air outlet area does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected air duct(s).

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## BASES

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### SURVEILLANCE REQUIREMENTS SR 3.1.2 (continued)

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the air outlets, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limit stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency for aboveground systems per the Completion Time Table in the CoC and 16 hours for underground systems is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts. For aboveground systems containing MPCs with heat loads less than or equal to the threshold heat loads in Table B.1-1 at the time of inspection, the surveillance frequency of 30 days is appropriate, since the system components and peak cladding temperature limits for 30-day accident are not exceeded and the event is not time limiting.

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REFERENCES	1.	FSAR Chapter 4
	2.	FSAR Sections 11.2.13 and 11.2.14
	3.	ANSI/ANS 57.9-1992

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## B 3.1 SFSC INTEGRITY

### B 3.1.3 MPC Cavity Reflooding

#### BASES

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**BACKGROUND** In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the cask preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The pressure in the MPC cavity is ensured to be less than the 100 psig design pressure. This is accomplished via direct measurement of the MPC gas pressure or via analysis.

After ensuring the MPC cavity pressure meets the LCO limit, the MPC is then re-flooded with water at a controlled rate and/or the pressure monitored to ensure that the pressure remains below 100 psig. Once the cavity is filled with water, the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and transfer cask are removed from the spent fuel pool and decontaminated.

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(continued)



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**BASES (continued)**


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**BACKGROUND (continued)**      Ensuring that the MPC cavity pressure is less than the LCO limit ensures that any steam produced within the cavity is safely vented to an appropriate location and eliminates the risk of high MPC pressure due to sudden generation of large steam quantities during re-flooding.

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**APPLICABLE SAFETY ANALYSIS**      The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Standard practice in the dry cask industry has historically been to directly reflood the cask with water. This standard practice is known not to induce fuel cladding failures.

The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by introducing water to the cavity in a controlled manner such that there is no sudden formation of large quantities of steam during MPC re-flooding. (Ref. 1).

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**LCO**      Determining the MPC cavity pressure prior to and during re-flooding ensures that there will be sufficient venting of any steam produced to avoid excessive MPC pressurization.

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**APPLICABILITY**      The MPC cavity pressure is controlled during UNLOADING OPERATIONS after the transfer cask and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the Cask Reflood LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS.

A note has been added to the APPLICABILITY for LCO 3.1.3 which states that the LCO is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concern for dry UNLOADING OPERATIONS.

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(continued)

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**BASES (continued)**

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**ACTIONS**

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

If the MPC cavity pressure limit is not met, actions must be taken to restore the parameters to within the limits before initiating or continuing re-flooding the MPC.

Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS      SR 3.1.3.1

The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal pressure prior to and during re-flooding the MPC there is sufficient steam venting capacity during MPC re-flooding.

The LCO must be met on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved. If the re-flood rate is limited to the bounding value given in FSAR Section 4.5 or calculated specifically for the MPC heat load then the MPC pressure must only be verified once prior to the re-flood.

If verifying the MPC pressure using direct measurement only the SR requires checks prior to the re-flood and every hour during re-flood. The direct measurement schedule is sufficient to prevent overpressurization of the MPC cavity as the rate of pressure rise is relatively slow compared to increase in re-flood rate.

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REFERENCES      1.      FSAR, Section 4.5, 8.3.2 and 8.3.3.

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### B 3.1 SFSC Integrity

#### B 3.1.4 Supplemental Cooling System

#### **BASES**

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**BACKGROUND** The Supplemental Cooling System (SCS) is an active cooling system that provides augmented heat removal from the MPC to ensure fuel cladding temperatures remain below the applicable limit during onsite transport operations in the TRANSFER CASK. The system is required for all MPCs meeting the burnup, heat load, and TRANSFER CASK orientation combinations specified in the Applicability of the LCO.

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#### **APPLICABLE SAFETY ANALYSIS**

For MPCs containing all moderate burnup fuel ( $\leq 45,000$  MWD/MTU), SCS operation is not required, because the fuel cladding temperature does not exceed the limit of 1058°F for moderate burnup fuel (Refs. 1, 2, and 3).

For MPC-68M containing high burnup fuel ( $>45,000$  MWD/MTU) up to design basis heat load, SCS operation is not required because the fuel cladding temperature does not exceed the limit of 752°F for high burnup fuel (Refs. 1, 2, and 3).

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(continued)

**BASES**

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**LCO**                    The Supplemental Cooling System must be operable if the MPC/TRANSFER cask assemblage meets one of the following conditions in the Applicability portion of the LCO in order to preserve the assumptions made in the thermal analysis.

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**APPLICABILITY**    This LCO is not applicable to MPC-68M.

The LCO is applicable within 4 hours after completion of MPC drying operations in accordance with LCO 3.1.1 (see NOTE below) or within 4 hours of transferring the MPC into the TRANSFER CASK if the MPC is to be unloaded, and the following conditions are met:

MPCs having one or more fuel assemblies with an average burnup greater than 45,000 MWD/MTU and either (i) MPC backfilled to higher helium backfill limits in Table 3-2 of Technical Specification Appendix B and having decay heat greater than 90% of the design basis heat load defined in Subsection 2.1.9.1 or (ii) MPC backfilled to lower helium backfill limits in Table 3-2 and having decay heat greater than 90% of heat load limits in Tables 3-3 or 3-4 of Tech. Spec. Appendix A.

Note: MPC drying operations are defined as the removal of moisture from the sealed canister and the use of the FHD to backfill the sealed canister with helium. MPC drying operations are complete when the partial pressure of the water vapor within the MPC is reduced to less than or equal to 3 torr, the MPC is backfilled in accordance with the TS, and the MPC is isolated from the FHD system.. The SCS is required to be operable within 4 hours from after backfilling.

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(continued)

## BASES

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### ACTIONS

#### A.1

If the SCS has been determined to be inoperable, the thermal analysis shows that the fuel cladding temperature would not exceed the short term temperature limit applicable to an off-normal condition, even with no water in the TRANSFER CASK-to-MPC annulus. Actions should be taken to restore the SCS to operable status in a timely manner. Because the thermal analysis is a steady-state analysis, there is an indefinite period of time available to make repairs to the SCS. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 7 days is considered appropriate and a reasonable amount of time to plan the work, obtain needed parts, and execute the work in a controlled manner.

#### B.1

If, after 7 days, the SCS cannot be restored to operable status, actions should be taken to remove the fuel assemblies from the MPC and place them back into the spent fuel pool storage racks. Thirty days is considered a reasonable time frame given that the MPC will be adequately cooled while this action is being planned and implemented, and certain equipment for this infrequent evolution (e.g., weld cutting machine) may take some time to acquire.

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## BASES

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### SURVEILLANCE REQUIREMENTS SR 3.1.4.1

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment, including during short-term evolutions such as on-site transportation in the TRANSFER CASK. The SCS is required to ensure adequate fuel cooling in certain cases. The SCS should be verified to be operable every two hours. This would involve verification that the coolant temperatures are within the design limits defined in Appendix 2.C. This is a reasonable Frequency given the typical oversight occurring during the on-site transportation evolution, the duration of the evolution, and the simple equipment involved.

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### AMBIENT TEMPERATURE REQUIREMENTS

Cask users are required in accordance with Design Features section in Tech. Specs. Appendix B to evaluate three day average ambient temperatures for compliance with FSAR Table 2.2.2 temperature limits under HI-TRAC transfer operations inside the part 50 structural boundary and outside of it. The three day average temperatures maybe computed from best available site data or based on local weather forecasts during scheduled periods of HI-TRAC movement or a suitably conservative method for bounding the three day average.

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- REFERENCES
1. FSAR Sections 4.5 and 4.III.5
  2. NRC Interim Staff Guidance 11, Rev. 3
  3. NRC Memorandum, C. Brown to M.W. Hodges, January 29, 2004
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## B 3.1 SFSC Integrity

### B 3.1.5 Impressed Current Cathodic Protection System (ICCPS)

#### **BASES**

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**BACKGROUND** The Impressed Current Cathodic Protection System (ICCPS) is an active corrosion mitigating system, which uses anodes plus a source of electric current that is forced to flow from the anodes to the Cavity Enclosure Container (CEC) to oppose corrosion currents. The user may elect to either extend an existing ICCPS to protect the installed ISFSI or establish an autonomous ICCPS. The ICCPS is designed on a site-specific basis for a particular ISFSI installation in accordance with the design criteria established in FSAR Supplement 3.I, Subsection 3.I.4.1.

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**APPLICABLE SAFETY ANALYSIS** The ICCPS provides reasonable assurance that corrosion will not promote degradation of the CEC Container Shell and Bottom Plate to the extent of challenging its structural function or initiate through-wall in-leakage of water into the storage cavity.

The ICCPS is required for all HI-STORM 100U VVM's that are installed in environments with aggressive corrosivity. FSAR Supplement 3.I, Subsection 3.I.4.1 provides the design criteria for an ICCPS and more details on the evaluation procedure to determine if the environment corrosivity is mild or aggressive.

The ICCPS is classified as an interfacing System, Structure or Component (SSC) to the HI-STORM 100U VVM and is designated as not important- to- safety. The ICCPS is not required to maintain subcriticality, control radiation doses to workers and public, assure fuel retrievability or maintain the confinement barrier.

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(continued)



## BASES

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LCO Although corrosion is expected to be a generally slow process even in the majority of subterranean environments determined to have aggressive corrosivity, the ICCPS will ensure corrosion will be kept to a minimum for the duration of storage operations. In order to be effective at mitigating degradation of the CEC due to corrosion, the ICCPS must be operable as soon as practicable after installation of the first HI-STORM 100U VVM and remain operable as much as practicable during storage operations.

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APPLICABILITY The LCO is applicable during STORAGE OPERATIONS. Once the first VVM to be protected by a particular ICCPS is installed the ICCPS must remain operable at all times to the extent practicable.

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ACTIONS A.1

If the ICCPS has been determined to be inoperable, Actions should be taken to restore the ICCPS to operable status in a timely manner. Because corrosion is an intrinsically slow process, there is sufficient time available to perform repairs and other corrective actions. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 6 months is considered appropriate and a reasonable amount of time to plan the work, obtain needed parts, and execute the work.

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(continued)

**BASES**

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**ACTIONS**  
(continued)A.2

In lieu of Action A.1, the user may opt to perform an engineering evaluation to establish reasonable assurance that CEC Container Shells and Bottom Plate are maintaining adequate integrity and have adequate corrosion reserve for continued storage operation for at least four more years or until the expiration of the license, whichever comes first. This action may include a more detailed review of the subterranean environment corrosivity, experience with corrosion of other buried systems at the site, past performance of the ICCPS and/or thickness measurements (e.g. ultrasonic inspection) of selected CEC Container Shells and Bottom Plates. Because corrosion is generally a slow process, there is sufficient time available to perform the engineering evaluation. In addition, the engineering evaluation if successful provides reasonable assurance that the CEC is experiencing minimal corrosion and will continue to perform its primary function during storage operations. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 1 year is considered appropriate and reasonable.

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(continued)

## BASES

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### ACTIONS (continued)

#### B.1

If after 8 years the ICCPS has been determined to be operable for less than 70% of the time since initial startup, Actions should be taken to establish reasonable assurance that CEC Container Shells and Bottom Plates are maintaining adequate integrity and have adequate corrosion reserve for continued storage operation of at least three more years or until the expiration of the license, whichever comes first. This action may include a more detailed review of the subterranean environment corrosivity, experience with corrosion of other buried systems at the site, past performance of the ICCPS and/or thickness measurements (e.g. ultrasonic inspection) of selected CEC Container Shells and Bottom Plates. Because corrosion is generally a slow process, there is sufficient time available to perform the engineering evaluation. In addition, the engineering evaluation, if successful, provides reasonable assurance that the CEC is experiencing minimal corrosion and will continue to perform its primary function during storage operations. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 1 year is considered appropriate and a reasonable amount of time to perform the Action. A 70% operable status is determined by engineering judgment to be the minimum duration of cathodic protection that will provide reasonable assurance the CEC will maintain its integrity during storage operations.

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(continued)

## BASES

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### ACTIONS (continued)

#### B.2

If an engineering evaluation does not establish adequate integrity of the CEC Container Shells and Bottom Plates, Actions should be taken to re-establish their integrity. This may entail performing thickness measurements (e.g. ultrasonic inspections) and repair of CEC components. Welding repairs require the transfer of the MPC into a transfer cask. Other types of repairs shall be evaluated to determine whether removal of the MPC from the VVM is necessary. Repairs with the potential to adversely affect safety functions shall be evaluated to ensure system functionality is maintained. If reasonable assurance is provided that the integrity of the CEC Container Shell and Bottom Plate has been adequately re-established, the initial start-up period shall be reset. Because corrosion is generally a slow process, there is sufficient time available to perform repairs. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 3 years ensures that there is sufficient time to perform the Action.

#### C.1

If an engineering evaluation cannot conclude adequate integrity of the CEC Container Shells and Bottom Plates or if repairs are not performed to re-establish integrity, Actions should be taken to transfer the MPC's from affected VVM's to unaffected VVM's or other approved and compatible casks. Because corrosion is generally a slow process there is sufficient time available to perform the MPC transfers, including supply of additional storage casks and storage pad(s) if needed. A Completion Time of 3 years is considered appropriate and a reasonable amount of time to perform the Action.

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**SURVEILLANCE  
REQUIREMENTS** SR 3.1.5.1

The long-term integrity of the CEC Container Shell and Bottom Plate is dependent on the ability of the ICCPS to cathodically protect all surfaces contacting the subgrade. The ICCPS should be verified to be operable one year after installation of the first VVM and every month thereafter. This may involve verification that the current and voltage requirements are in accordance with the site-specific design as well as ensuring that rectifiers are in working order. The surveillance intervals follow the guidance provided in the standards and publications listed in the design criteria provided in FSAR Supplement 3.I, Subsection 3.I.4.1. This is a reasonable Frequency given the generally slow process of corrosion. This surveillance is suspended for 1 year if Action A.2 is met.

SR 3.1.5.2

The long-term integrity of the CEC container shell and Bottom Plate is dependent on the ability of the ICCPS to cathodically protect all surfaces contacting the subgrade. The ICCPS should be verified to be operable for at least 70% of the time on a cumulative basis between 8 and 10 years after initial startup (whether or not initial startup was successful) to allow sufficient operating history. This verification will involve review of records that support the operable status of the ICCPS, which may include review of current and voltage measurements, and other site-specific design data. The surveillance intervals follow the guidance of engineering judgment. This is a reasonable Frequency given the generally slow process of corrosion.

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**REFERENCES** 1. FSAR Supplement 3.I, Subsection 3.I.4.1

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B 3.2 SFSC Radiation Protection

B 3.2.1 Deleted

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## B 3.2 SFSC Radiation Protection

## B 3.2.2 TRANSFER CASK Surface Contamination

**BASES**

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**BACKGROUND** A TRANSFER CASK is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the TRANSFER CASK may become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the TRANSFER CASK to the ISFSI, or prior to transferring the MPC into the OVERPACK, whichever occurs first, in order to minimize the radioactive contamination to personnel or the environment. This allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.

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**APPLICABLE SAFETY ANALYSIS** The radiation protection measures implemented during MPC transfer and transportation using the TRANSFER CASK are based on the assumption that the exterior surfaces of the TRANSFER CASKs have been decontaminated. Failure to decontaminate the surfaces of the TRANSFER CASKs could lead to higher-than-projected occupational doses.

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**LCO** Removable surface contamination on the TRANSFER CASK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm<sup>2</sup> from beta and gamma sources and 20 dpm/100 cm<sup>2</sup> from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the TRANSFER CASK loading process.

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(continued)

## BASES

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### LCO (continued)

Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose. LCO 3.2.2 requires removable contamination to be within the specified limits for the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC. The location and number of surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the MPC means the upper portion of the MPC external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed. The user shall determine a reasonable number and location of swipes for the accessible portion of the MPC. The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC, while implementing sound ALARA practices.

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### APPLICABILITY

The applicability is modified by a note that states that the LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING. This is consistent with the intent of this LCO, which is to ensure loose contamination on the loaded TRANSFER CASK and MPC outside the FUEL BUILDING is within limits. If the MPC transfer is performed inside the FUEL BUILDING the empty TRANSFER CASK remains behind and is treated like any other contaminated hardware under the user's Part 50 contamination control program.

Verification that the surface contamination is less than the LCO limit is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS, when the LCO is applicable. Measurement of surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.

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(continued)



## BASES (continued)

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**ACTIONS** A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. Subsequent TRANSFER CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

### A.1

If the removable surface contamination of a TRANSFER CASK or MPC, as applicable, that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the TRANSFER CASK or MPC and bring the removable surface contamination within limits. The Completion Time of 7 days is appropriate given that sufficient time is needed to prepare for, and complete the decontamination once the LCO is determined not to be met.

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**SURVEILLANCE REQUIREMENTS** SR 3.2.2.1

This SR verifies that the removable surface contamination on the TRANSFER CASK and/or accessible portions of the MPC is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification during LOADING OPERATIONS in order to confirm that the TRANSFER CASK or OVERPACK can be moved to the ISFSI without spreading loose contamination.

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**REFERENCES**

1. FSAR Sections 8.1.5 and 8.1.6.
2. NRC IE Circular 81-07.

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B 3.2

B 3.2.3 Deleted

### B 3.3 SFSC Criticality Control

#### B 3.3.1 Boron Concentration

#### BASES

**BACKGROUND** A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. The TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.

For those MPCs containing PWR fuel assemblies of relatively high initial enrichment, credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.

#### **APPLICABLE SAFETY ANALYSIS**

The spent nuclear fuel stored in the SFSC is required to remain subcritical ( $k_{eff} < 0.95$ ) under all conditions of storage. The HI-STORM 100 SFSC is analyzed to store a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-32 and MPC-32F, and for relatively high enrichment PWR fuel loaded in the MPC-24, -24E, and -24EF, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

(continued)

**BASES (continued)**

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**LCO**

Compliance with this LCO ensures that the stored fuel will remain subcritical with a  $k_{\text{eff}} \leq 0.95$  while water is in the MPC. LCOs 3.3.1.a and 3.3.1.b provide the minimum concentration of soluble boron required in the MPC water for the MPC-24, and MPC-24E/24EF, respectively, for MPCs containing all INTACT FUEL ASSEMBLIES. The limits are applicable to the respective MPCs if one or more fuel assemblies to be loaded in the MPC had an initial enrichment of U-235 greater than the value in Table 2.1-2 of Appendix B to the CoC for loading with no soluble boron credit.

LCO 3.3.1.e provides the minimum concentration of soluble boron required in the MPC water for the MPC-24E and MPC-24EF containing at least one DAMAGED FUEL ASSEMBLY or one fuel assembly classified as FUEL DEBRIS.

LCO 3.3.1.f provides the minimum concentration of soluble boron required in the MPC water for the MPC-32 and MPC-32F based on the fuel assembly array/class and the classification of the fuel as a DAMAGED FUEL ASSEMBLY or FUEL DEBRIS.

All fuel assemblies loaded into the MPC-24, MPC-24E, MPC-24EF, MPC-32, and MPC-32F are limited by analysis to maximum enrichments of 5.0 wt.% U-235.

The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes and fuel classifications (intact vs. damaged) are mixed in the same MPC. This ensures the assumptions pertaining to soluble boron used in the criticality analyses are preserved.

(continued)

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BASES

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APPLICABILITY The boron concentration LCO is applicable whenever an MPC-24, -24E, -24EF, -32, or -32F has at least one PWR fuel assembly in a storage location and water in the MPC. For the MPC-24 and MPC-24E/24EF, when all fuel assemblies to be loaded have initial enrichments less than the limit for no soluble boron credit as provided in CoC Appendix B, Table 2.1-2, the boron concentration requirement is implicitly understood to be zero.

During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water.

During UNLOADING OPERATIONS, the LCO is applicable when the MPC is re-flooded with water after helium cooldown operations. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is upon entering the Applicability.

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ACTIONS A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. If the boron concentration of water in the MPC is less than its limit, all activities LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions must be suspended immediately.

(continued)

## BASES

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### ACTIONS (continued)

#### A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately.

One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration be initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours prior to entering the Applicability of the LCO. For LOADING OPERATIONS, this means within four hours of loading the first fuel assembly into the cask.

For UNLOADING OPERATIONS, this means verifying the source of borated water to be used to re-flood the MPC within four hours of commencing re-flooding operations. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

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## BASES

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### SURVEILLANCE REQUIREMENTS

(continued)

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premise of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO

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### REFERENCES

1. FSAR Chapter 6.
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**HI-STORM 100 SYSTEM FSAR**

**APPENDIX 12.B**

**THIS APPENDIX HAS BEEN DELETED**



## **SUPPLEMENT 12.I**

### **OPERATING CONTROLS AND LIMITS FOR THE HI-STORM 100U SYSTEM**

#### **12.I.0 INTRODUCTION**

Operation of the HI-STORM 100 System with the underground HI-STORM 100U overpack is identical to operation of the HI-STORM 100 System with the aboveground overpacks up to the point where MPC transfer from the HI-TRAC transfer cask takes place. The loaded MPC is stored in the vertical orientation in the HI-STORM 100U for long-term storage operations. Operating controls and limits described in FSAR Chapter 12 are implemented for the HI-STORM 100U to the extent they are applicable. This supplement describes the unique operating controls and limits, including training requirements and technical specifications, required for the HI-STORM 100U.

#### **12.I.1 OPERATING CONTROLS AND LIMITS**

Like the aboveground overpacks, the HI-STORM 100U overpack is completely passive in performing its design functions. It provides for long-term interim storage of the spent fuel without any modifications to the MPC or HI-TRAC transfer cask designs used with the aboveground overpacks. The operating controls and limits established for the MPC and HI-TRAC transfer cask are not altered for use with the HI-STORM 100U.

Operating controls and limits pertaining to movement of the overpack do not apply to the HI-STORM 100U because it is an integral component of the ISFSI and is not able to be moved once installed. It is not required to establish a lift height limit or to evaluate a drop event because the HI-STORM 100U cannot be lifted. It is not required to measure dose rates on the side of the HI-STORM 100U because it is located underground and is, therefore, inaccessible by plant personnel and because side dose rates are limited by shielding from the surrounding soil and provide a negligible contribution to the dose rate at the controlled area boundary.

#### **12.I.2 TRAINING MODULES**

The classroom and dry run training programs described in FSAR Subsections 12.2.1 and 12.2.2 must be modified to address the differences between the use of an aboveground, moveable overpack and the belowground, immovable HI-STORM 100U. Specifically, procedures and dry runs must be created or modified to address the operational differences in using the HI-TRAC transfer cask and mating device with the HI-STORM 100U versus the aboveground overpack. These procedures and dry run training shall be based upon the operations of the HI-STORM 100U described in Supplement 8.I.

### 12.I.3 TECHNICAL SPECIFICATIONS

LCO 3.1.2, which requires periodic surveillance to ensure operability of the cask heat removal system, can be implemented with HI-STORM 100U more easily than for the aboveground overpacks because the outlet air ducts are located at approximately ground level. New LCO 3.1.5, for periodic confirmation of operation of the Impressed Current Cathodic Protection System, is added.

## **SUPPLEMENT 12.III**

### **OPERATING CONTROLS AND LIMITS FOR THE MPC-68M**

The main body of this chapter remains fully applicable for the HI-STORM 100 System using an MPC-68M with the exception that a supplemental cooling system is not required during loading/unloading operations of the MPC-68M in a HI-TRAC transfer cask.

## CHAPTER 13<sup>†</sup>: QUALITY ASSURANCE

### 13.0 QUALITY ASSURANCE PROGRAM

#### 13.0.1 Overview

This chapter provides a summary of the quality assurance program implemented by Holtec International for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the Company's dry storage/transport systems including the HI-STORM 100 system which includes the HI-TRAC transfer cask. This chapter is included in this FSAR to fulfill the requirements in 10 CFR 72.140 (c) (2) and 72.2(a)(1),(b).

Important-to-safety activities related to construction and deployment of the HI-STORM 100 system are controlled under the NRC-approved Holtec Quality Assurance Program. The Holtec QA program manual [13.0.2] is approved by the NRC [13.0.4] under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.

1. Holtec Quality Assurance Program Manual
2. Holtec Quality Assurance Procedures
3.
  - a. Holtec Standard Procedures
  - b. Holtec Project Procedures

Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide the assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

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

### 13.0.2 Graded Approach to Quality Assurance

Holtec International uses a graded approach to quality assurance on all safety-related or important-to-safety projects. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Subsection 13.0.1.

NUREG/CR-6407 [13.0.1] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STORM 100 system and HI-TRAC transfer cask. The ITS categories assigned to the HI-STORM 100 cask components are identified in Table 2.2.6. Quality categories for ancillary equipment are provided in Chapter 8 of this FSAR. Quality categories for other equipment needed to deploy the HI-STORM 100 system at a licensee's ISFSI are defined on a case-specific basis considering the component's design function using the guidelines of NUREG/CR-6407 [13.0.1].

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 system at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. These activities include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM 100 structures, systems, and components (SSCs) that are important-to-safety.

The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [13.0.3]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536 and has accordingly been added to the list of deviations in Table 1.0.2. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities.

13.1        REFERENCES

- [13.0.1]    NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.
- [13.0.2]    Holtec International Quality Assurance Program, Latest Approved Revision.
- [13.0.3]    NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.
- [13.0.4]    NRC QA Program Approval for Radioactive Material Packages No. 0784, Docket 71-0784.