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 ~~seven~~ 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. Exceptions to specific Code requirements with complete justifications are presented in Table 2.2.15.

7.1.1 Confinement Vessel

The HI-STORM 100 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, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1] to the maximum extent practicable. 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 at the loading site 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.

The design basis leakage rate for the MPC confinement boundary is provided in Table 7.1.1. The MPC shell and baseplate are helium leakage tested during fabrication in accordance with the requirements defined in Chapter 9. Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by liquid penetrant method (root and final), volumetrically examined (if volumetric examination is not performed, multi-layer liquid penetrant examination must be performed), helium leakage tested, and hydrostatically 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 (root and final), and a leakage rate test is performed. Finally, the MPC closure ring is installed, welded and inspected by the liquid penetrant method (root, if multiple pass, and final). Chapters 8, 9, and 12 provide procedural guidance, acceptance criteria, and Technical Specifications, respectively, for performance and acceptance of liquid penetrant examinations, volumetric examination, hydrostatic 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 which are seal welded to the MPC lid. No credit is taken for the seal provided by the vent and drain ports. 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] to the maximum extent practicable. 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] to the maximum extent practicable. 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. Welds are also helium leak tested, providing added assurance of weld integrity. Additionally, a hydrostatic test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity. 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 7.1.3 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 and leakage tested to ensure their integrity. 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. Section 11.2.1.4 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 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 the Approved Contents Section of Appendix B to the CoC have been evaluated.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel DFCs prior to placement in the HI-STORM 100 System. The DFCs that may be loaded into the MPCs are shown in Figures 2.1.1 through Figure 2.1.2c. 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. The screens are specified as a 250-by-250-mesh with an effective opening of 0.0024 inches. There are no other openings in the DFC. The CoC specifies the fuel assembly characteristics for damaged fuel acceptable for loading in the MPC-24E, MPC-24EF, *MPC-32F,* MPC-68, MPC-68F or MPC-68FF and for fuel debris acceptable for loading in the MPC-24EF, *MPC-32F,* 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 storage design basis leakage rates are 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.

Table 7.1.1

SUMMARY OF CONFINEMENT BOUNDARY DESIGN SPECIFICATIONS

Design Condition	Design Pressure (psig)	Design Temperature (°F)
Normal	100	MPC Lid: 550
		MPC Shell: 450
		MPC Baseplate: 400
Off-Normal	*110*~~100~~	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775
Accident	200	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775

Table 7.1.2

MPC CONFINEMENT BOUNDARY WELDS

Confinement Boundary Welds		
MPC Weld Location	**Weld Type†**	**ASME Code Category (Section III, Subsection NB)**
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

CLOSURE WELD EXAMINATIONS AND TESTS

Closure Weld Description	Inspections/Tests	ASME Acceptance Criteria
MPC Lid-to-Shell	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass Volumetric Examination of Weld (UT) or multi-layer PT Hydrostatic Test Post Hydrostatic Test - PT Helium Leakage Test	NF-5360 NB-5350 NB-5350 NF-5360 NB-5332 NB-6000 NB-5350 Sect. V and ANSI N14.5
Vent/Drain Cover Plate to MPC Lid	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass Helium Leakage Test	NF-5360 NB-5350 NB-5350 NF-5360 Sect. V and ANSI N14.5
Closure Ring Radial Welds	VT on Tack Welds PT Root Pass (if multiple pass) PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360
Closure Ring-to-MPC Shell	VT on Tack Welds PT Root Pass (if multiple pass) PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360
Closure Ring-to-MPC Lid	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360

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 storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal conditions of storage. Since the MPC confinement vessel remains intact, and the design bases temperatures and pressure are not exceeded, the design basis leakage rate is not exceeded during normal conditions of storage.

7.2.1 Release of Radioactive Material

The MPC is closed by the MPC lid, the vent and drain port cover plates, and the MPC closure ring. Weld examinations, including multiple surface examinations, volumetric examination, hydrostatic testing, and leakage rate testing on the MPC lid weld, and multiple surface examinations and leakage rate testing of the vent and drain port cover plate welds, assure the integrity of the MPC closure. The MPC is a strength-welded pressure vessel designed to meet the stress criteria of the ASME Code, Section III, Subsection NB [7.1.1]. The all-welded construction of the MPC with redundant closure provided by the fully welded MPC closure ring and extensive inspections and testing ensures that no release of fission gas or crud for normal storage and transfer conditions will occur. The above discussion notwithstanding, an analysis is performed in Section 7.2.7 to calculate the annual dose at 100 meters based on an assumed leakage rate of $5x10^{-6}$ atm-cm^3/sec under reference test conditions.

7.2.2 Pressurization of the Confinement Vessel

The loaded and sealed MPC is drained, removed of moisture, and backfilled with helium gas. This process provides a chemically non-reactive environment for storage of spent fuel assemblies. First, air in the MPC is displaced with water and then the water is displaced by helium or nitrogen gas during MPC blowdown. The MPC is then removed of all moisture, and backfilled with a predetermined mass of helium as specified in the Technical Specifications. Chapter 8 describes the steps of these processes and the Technical Specifications provide the acceptance criteria. This drying and backfilling process ensures that the resulting inventory of oxidizing gases in the MPC remains below 0.25% by volume, and that the MPC pressure is maintained within the design limitations. In addition, the MPC basket fluid contact areas are stainless steel alloy material or aluminum of extremely high corrosion and erosion resistance. The aluminum oxide layer on the aluminum components (e.g., heat conduction elements and Boral neutron absorption plates) ensures that there is no reaction during the short duration of exposure to the fuel pool water. Carbon steels are not employed in the construction of the MPCs. Therefore, no protective coatings which could interact with borated spent fuel pool water are used.

The only means of pressure increase in the MPC is from the temperature rise due to normal heat-up to normal operating temperatures and the release of backfill and fission gas contents from fuel rods

into the MPC cavity. Under the most adverse conditions of normal ambient temperature, full insolation, and design basis decay heat, the calculated pressure increase assuming 1% fuel rod failure is well below the system design pressure as shown in Chapter 4. For off-normal conditions of storage, failure of up to 10% of the fuel rods has been analyzed and would result in an MPC internal pressure below the value specified as the *off*-normal design pressure.

7.2.3 Confinement Integrity During Dry Storage

There is no credible mechanism or event that results in a release of radioactive material from the MPC under normal conditions. Since the MPC remains structurally intact and provides redundant welded closures as discussed above, the postulated leakage of radioactive material from the MPC will be limited to a leakage rate equivalent to the acceptance test criteria specified for the MPC helium leak tests. Leakage from the MPC during normal conditions of storage could result in the release of gaseous fission products, fines, volatiles and airborne crud particulates as discussed in Section 7.3.1. The conservative assumption is made that 2.5% of the fuel inventory is available for release under normal conditions of storage and 11.5% of the fuel inventory is available for release under off-normal conditions of storage. The maximum cavity internal operating pressure with *either 1% (normal conditions) or* 10% *(off-normal conditions)* fuel rod failure reported in Table 4.4.14 is bounded by the use of ~~an internal cavity~~ *the design* pressure*s* ~~of 101.4 psia (6.90 ATM)~~, which ~~is~~ *are* assumed as ~~an~~ initial conditions for th*ese*~~is~~ evaluation*s*.

The annual dose equivalent for the whole body, thyroid and other critical organs to an individual at the site boundary (100 meters) as a result of an assumed effluent release under normal and off-normal conditions of storage were determined. These doses were determined for each type of MPC. The ISFSI controlled area boundary must be at least 100 meters from the nearest loaded HI-STORM 100 System in accordance with 10CFR72.106(b) [7.0.1]. The doses are compared to the regulatory limits specified in 10CFR72.104(a) [7.0.1].

Confinement boundary welds performed at the fabricator's facility are inspected by volumetric and liquid penetrant examination methods as detailed in Section 9.1. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and MPC closure ring. The weld of the MPC lid-to-shell is liquid penetrant examined on the root and final pass, volumetrically (or multilayer liquid penetrant) examined, hydrostatically tested, and leak rate tested. The vent and drain port cover plates are liquid penetrant examined on the root and final pass and leak rate tested. The MPC closure ring welds are inspected by the liquid penetrant examination method. In Chapter 11, the MPC lid-to-shell weld is postulated to fail to confirm the safety of the HI-STORM 100 confinement boundary. The failure of the MPC lid weld is equivalent to the MPC drain or vent port cover weld failing. The MPC lid weld failure affects the MPC confinement boundary; however, no leakage will occur due to redundant sealing provided by the MPC closure ring.

7.2.4 Control of Radioactive Material During Fuel Loading Operations

The procedures for closure of the MPC, described in Section 8.1, are intended to assure that there is

no unintended release of gas, liquid, or solid materials from the MPC during dry storage. During MPC closure operations, the lines used for venting or draining are routed to the plant's spent fuel pool or radioactive waste processing systems. MPC closure operations are performed inside the plant's fuel building in a controlled and monitored environment.

Radioactive effluent handling during fuel loading and MPC draining, moisture removal, helium backfilling, and sealing operations is in accordance with the plant's 10CFR50 license and radioactive waste management system.

7.2.5 External Contamination Control

The external surface of the MPC is protected from contamination by preventing it from coming in contact with the spent fuel pool water. Prior to submergence in the spent fuel pool, an inflatable seal is installed at the top of the annulus formed between the MPC shell and the HI-TRAC transfer cask cavity. This annulus is filled with clean demineralized water and the seal is inflated. The inflated seal, backed by the demineralized water maintained at a slight positive pressure, is sufficient to preclude the entry of contaminated water into the annulus. These steps assure that the MPC surface is free of contamination that could become airborne during storage.

Additionally, following fuel loading operations and removal from the spent fuel pool, the upper end of the MPC shell is surveyed for loose surface contamination in accordance with the Technical Specifications contained in Chapter 12 of this FSAR.

7.2.6 Confinement Vessel Releasable Source Term

As discussed in Section 7.3.1, the source term used to evaluate the annual dose at the minimum controlled area boundary of 100 meters due to leakage from the MPC confinement boundary consists of gaseous fission products, fines, volatiles and airborne crud particulates. For storage of spent fuel assemblies with burnups in excess of 45 GWD/MTU the source term from the assumed rod breakage fractions of ISG-5 [7.2.2] must be augmented by the source term from 50% of the rods having peak cladding thicknesses greater than 70 micrometers. ISG-11 [7.2.1] recommends that for high burnup fuel assemblies to be classified as intact, no more than 3% of the rods may have peak cladding oxide thickness' greater than 70 micrometers and no more than 1% of the rods may have peak cladding oxide thickness' greater than 80 micrometers. Using Equation 7-0 below the fraction of the source term available for release may be determined:

Equation 7-0

$$F_R = F_B * (100\%) + F_{70} * P_S$$

where:

F_R is the percentage of the source term available for release,
F_B is the rod breakage fraction from ISG-5,

F_{70} is the percentage of rods that have peak cladding oxide thicknesses greater than 70 microns

P_S percentage of the source term for rods having peak cladding oxide thickness' greater than 70 microns that must be included in the total source term available for release.

Table 7.2.1 contains a summary of the values required for Equation 7-0 and the results for normal and off-normal conditions of storage. Additionally, a summary of the isotopes available for release is provided in Table 7.3.1.

7.2.7 Release of Contents Under Normal and Off-Normal Storage Conditions

7.2.7.1 Confinement Boundary Leakage Rate

The methodology presented in Section 7.3.3.1 was used to determine the leakage rate at the upstream conditions. Using the capillary diameter determined in Section 7.3.3.1, and the parameters for normal and off-normal conditions provided in Table 7.3.4, Equation 7-3 was solved for the leakage rate at the upstream conditions. The resultant normal ~~and off-normal~~ condition leakage rate *is calculated to be*~~,~~ -1.~~12~~*24*x10^{-5} cm^3/s ~~(at 581 K, 6.90 ATM) was calculated~~ *and the resultant off-normal condition leakage rate is calculated to be 1.47x10^{-5} cm^3/s.*~~.~~

7.2.7.2 ~~Percentage of Nuclides that Remain Airborne~~*Gravitational Settling*

The fines, volatiles and crud that are released from the fuel cladding to the cask cavity do not remain airborne inside the cask cavity for the entire duration of the normal/off-normal conditions. Therefore, credit is taken for gravitational settling of the fines, volatiles and crud in accordance with the methodology presented in reference [7.2.3].

~~In addition to the small fraction of fines that are released in the event of a cladding breach, only 10% of the fines released to the MPC cavity remain airborne long enough to be available for release from the MPC [7.3.11]. It is conservatively assumed that 100% of the volatiles, crud and gases remain airborne and available for release.~~

7.2.7.3 Fraction of Volume Released

The minimum free volume of each MPC design is presented in Tables 4.4.12, 4.4.13, 4.4.24, and 4.4.25[1]~~.~~. Using these volumes and the upstream normal and off-normal condition leakage rate ~~of 1.12x10^{-5} cm^3/s~~, the fraction of the volume released per second is calculated. For calculation of the doses from the MPC-24, MPC-24E and MPC-24EF the minimum free volume from the MPC-24E is used as it conservatively bounds the MPC-24 and MPC-24EF.

7.2.7.4 Release Fraction

The release fraction is that portion of the total radionuclide inventory that is released from the inside of the fuel rods to the MPC cavity. The release fractions provided in NUREG/CR-6487 [7.3.2] are

[1] *These volumes have been conservatively reduced by 1.67x10^5 cm^3.*

used. A summary of the release fractions is provided in Table 7.3.1.

7.2.7.5 Radionuclide Release Rate

The radionuclide release rate is the product of the quantity of isotopes available for release, the number of assemblies, the ~~percentage of nuclides that remain airborne~~*aerosol deposition factor*, the fraction of volume released, and the release fraction.

7.2.7.6 Atmospheric Dispersion Factor

For the evaluation of the dose at the controlled area boundary, the instantaneous χ/Q calculated for accident conditions (8.0×10^{-3} sec/m^3) was reduced to 1.6×10^{-4} sec/m^3 based on the long term nature of the release (1 year); the height of the release being essentially a ground level release (h_e = 0); all 16 compass directions (22.5 degree sectors) will be similarly affected due to the long term nature of the continuous release (over one year); the increase in average wind speeds (>1 m/s); and the additional effects of a reduction in atmospheric stability. Therefore, the χ/Q reduction factor of 50 used to correct the short term accident release χ/Q is conservative.

7.2.7.7 Dose Conversion Factors

Dose Conversion Factors (DCF) from EPA Federal Guidance Report No. 11, Table 2.1 [7.3.5] and EPA Federal Guidance Report No. 12, Table III.1 [7.3.6] were used for the analysis. ~~The DCFs are provided on the spread sheets included as Appendix 7.A.~~

7.2.7.8 Occupancy Time

An occupancy time of 8,760 hours is used for the analysis [7.0.2]. This conservatively assumes that the individual is exposed 24 hours per day for 365 days at the minimum controlled area boundary of 100 meters.

7.2.7.9 Breathing Rate

A breathing rate of 3.3×10^{-4} m^3/sec for a worker is used for the analysis [7.0.2]. This assumption is in accordance with the guidance provided in NUREG-1536 [7.0.2] for a worker.

7.2.8 Postulated Doses Under Normal and Off-Normal Conditions of Storage

The annual dose equivalent for the whole body, thyroid and other critical organs to an individual at the site boundary (100 meters) as a result of an assumed effluent release under normal and off-normal conditions of storage were determined. These doses are determined for each type of MPC and for each condition of storage (i.e., normal and off-normal). The postulated doses as a result of exposure to soil with ground surface contamination and soil contaminated to a depth of 15 cm were

also determined. The resultant doses were negligible compared to those resulting from submersion in the plume and are therefore not reported.

The doses were determined using spreadsheet software. The resultant doses are summarized for each MPC type in Tables 7.3.2 through Table 7.3.5 of the HI-STORM FSAR. ~~Example spread sheets used for the dose estimates are presented in Appendix 7.A.~~ Table 7.3.8~~9~~ compares the doses to the regulatory limits of 10CFR72.104(a).

7.2.8.1 Whole Body Dose

The annual dose equivalent to the whole body (ADE) is the sum of the inhaled committed effective dose equivalent (CEDE) and the deep dose equivalent to the whole body from submersion in the plume. The postulated doses were determined using spreadsheet software. ~~Example spread sheets are provided in Appendix 7.A.~~

The CEDE is the product of radionuclide release rate, the atmospheric dispersion factor, the occupancy time, the breathing rate, and the effective dose conversion factor.

The Deep Dose Equivalent is the product of the nuclide release rate, the atmospheric dispersion factor, the occupancy time, and the effective dose conversion factor.

7.2.8.2 Critical Organ Dose

The Annual Dose Equivalent (ADE) to the critical organ (or tissue) is the sum of the committed dose equivalent (CDE) to the critical organ or tissue from inhalation and the deep dose equivalent (DDE) to the organ or tissue from submersion in the plume. The postulated doses as a result of exposure to soil with ground surface contamination and soil contaminated to a depth of 15 cm were also determined. The resultant doses were negligible compared to those resulting from submersion in the plume and are therefore not reported.

The committed dose equivalent to the organ or tissue from inhalation is the product of *the* radionuclide release rate, the atmospheric dispersion factor, the occupancy time, the breathing rate, and the organ/tissue dose conversion factor. The deep dose equivalent to the organ or tissue from submersion in the plume is the product of the radionuclide release rate, the atmospheric dispersion factor, the occupancy time, and the organ/tissue dose conversion factor.

7.2.8.3 Site Boundary

The estimated annual dose equivalent for critical organs and the whole body at the minimum site boundary of 100 meters are presented in Tables 7.3.2 through 7.3.5. Since doses from any one MPC does not bound the doses from all other MPCs, bounding doses have been presented in Table 7.3.8 for BWR fuel (MPC-68, MPC-68F and MPC-68FF) and PWR fuel (MPC-24, MPC-24E, MPC-24EF, *MPC-32* and MPC-32*F*) separately. The doses from the MPC-68 bound the doses from all

casks containing BWR fuel and the doses from the MPC-32 bounds the doses from all casks containing PWR fuel. Additionally, Table 7.3.8 compares these bounding doses to the regulatory limits of 10CFR72.104(a).

7.2.9 Assumptions

The following presents a summary of assumptions for the normal condition confinement analysis of the HI-STORM 100 System.

- The distance from the cask to the site boundary is 100 meters.

- Under normal conditions of storage, 2.5% of the source term is available for release. Under off-normal conditions of storage, 11.5% of the source terms are available for release. This assumption is in accordance with ISG-5 [7.2.2], ISG-11 [7.2.1] and NUREG-1536 [7.0.2] for normal and off-normal storage conditions.

- Unchoked flow correlations were used as the unchoked flow correlations better approximate the true measured flow rate for the leakage rates.

- For conservatism, the upstream pressure at reference test conditions (inside of the MPC) is assumed to be 2 ATM and the down stream pressure (outside of the MPC) is assumed to be 1 ATM.

- The leak hole diameter is determined using reference test conditions rather than actual test conditions from Table 7.3.7. This is conservative, as it yields a larger leak hole diameter.

- The temperature at test conditions is assumed to be equal to a temperature, 212° F based on the maximum temperature achievable by the water in the MPC during performance of the leak test. This is conservative because the leak hole diameter computed from test conditions is larger.

- ~~Off~~ *The MPC pressure for* normal storage conditions ~~(i.e., MPC cavity at a pressure of 101.4 psia (6.90 ATM) at MPC cavity average temperature of 581 K)~~*is conservatively assumed to be equal to the design basis pressure. The MPC pressure for off-normal conditions is conservatively assumed to be larger than the design basis pressure. The off-normal temperature is postulated for this analysis as this temperature bounds the temperature under normal conditions of storage.* ~~are postulated for this analysis as these conditions bound the normal conditions of storage.~~

- The capillary length required for Equation 7-3 was conservatively chosen to be the MPC lid closure weld which is 1.9 cm.

- The majority of the activity associated with crud is due to ^{60}Co. This assumption follows

from the discussion provided in NUREG/CR-6487 [7.3.2].

- The normal and off-normal condition leakage rate persists for one year without a decrease in the rate or nuclide concentration.

- The individual at the site boundary is exposed for 8,760 hours [7.0.2]. This conservatively assumes that the individual is exposed 24 hours per day for 365 days.

- A breathing rate of 3.3×10^{-4} m^3/sec for a worker is used for the analysis [7.0.2]. This assumption is in accordance with the guidance provided in NUREG-1536 for a worker.

- All fuel stored in the MPC is of the design basis type with a bounding burnup and cooling time.

- Exposure to dose conversion factors for inhalation reported in EPA Federal Guidance Report No. 11, Table 2.1 [7.3.5] were selected by the most restrictive clearance class for each organ and each radionuclide.

- For conservatism, the maximum possible leakage rate under reference test conditions is assumed to be 7.5×10^{-6} atm-cm^3/s, which is 150% of the reference test leak rate of 5.0×10^{-6} atm-cm^3/s.

- *The MPC internal free volumes presented in Section 4.4 are conservatively reduced by 1.67×10^5 cm^3.*

Table 7.2.1
Parameters for Determining the Percentage of the Source Term Available for Release

MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68 and MPC-68FF

Parameter	Normal	Off-Normal
F_B	.01	.10
F_{70}	3.0%	3.0%
P_S	50%	50%
F_R	**2.5%**	**11.5%**

MPC-68F

Parameter	Normal	Off-Normal
F_R	**1.0%**	**10.0%**

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 normal, off-normal and 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] to the maximum extent practicable. In summary, there is no mechanistic failure that results in a breach of the MPC confinement boundary.

The above discussion notwithstanding, this section evaluates the consequences of a non-mechanistic postulated ground level breach of the MPC confinement boundary. This breach could result in the release of gaseous fission products, fines, volatiles and airborne crud particulates. The internal design accident pressure of 200 psig, as specified in Table 7.1.1, is conservatively increased in the analysis to 225 psig for this evaluation. The following doses to an individual at the site boundary (100 meters) as a result of an assumed effluent release under accident conditions of storage were determined: the committed dose equivalent (CDE) from inhalation and the deep dose equivalent (DDE) from submersion for critical organs and tissues (gonad, breast, lung, red marrow, bone surface, thyroid); the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from submersion for the whole body; the lens dose equivalent (LDE) for the lens of the eye; the shallow dose equivalent (SDE) from submersion for the skin; and the resulting Total Effective Dose Equivalent (TEDE) and Total Organ Dose Equivalent (TODE).

These doses were determined for each type of MPC. The ISFSI controlled area boundary must be at least 100 meters from the nearest loaded HI-STORM 100 System in accordance with 10CFR72.106(b) [7.0.1]. The doses are compared to the regulatory limits specified in 10CFR72.106(b) [7.0.1].

7.3.1 Confinement Vessel Releasable Source Term

In accordance with NUREG/CR-6487 [7.3.2], the following contributions are considered in determining the releasable source term for packages designed to transport irradiated fuel rods: (1) the radionuclides in the fuel rods, (2) the radionuclides on the surface of the fuel rods, and (3) the residual contamination on the inside surfaces of the vessel. NUREG/CR-6487 goes on to state that a radioactive aerosol can be generated inside a vessel when radioactive material from the fuel rods or from the inside surfaces of the container become airborne. The sources for the airborne material are (1) residual activity on the cask interior, (2) fission and activation-product activity associated with corrosion-deposited material (crud) on the fuel assembly surface, and (3) the radionuclides within the individual fuel rods. In accordance with NUREG/CR-6487, contamination due to residual activity on the cask interior surfaces is negligible as compared to crud deposits on the fuel rods themselves and therefore may be neglected. The source term considered for this calculation results from the spallation of crud from the fuel rods and from the fines, gases and volatiles which result from

cladding breaches. The methodology of NUREG/CR-6487 is conservatively applied to the storage confinement accident analysis as dry storage conditions are less severe than transport conditions.

The inventory for isotopes other than ^{60}Co is calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system as described in Section 5.2. The inventory for the MPC-24, MPC-24E, MPC-24EF, *MPC-32* and MPC-32*F* was conservatively based on the B&W 15x15 fuel assembly with a burnup of 75~~0~~,000 MWD/MTU, 3~~5~~ years of cooling time, and an enrichment of 5.0~~4.8~~%. The inventory for the MPC-68 and MPC-68FF was based on the GE 7x7 fuel assembly with a burnup of 65~~0~~,000 MWD/MTU, 5 years of cooling time, and 4.84% enrichment. The CoC limits the fuel assembly burnup below 60,000 MWD/MTU for both BWR and PWR fuel at 5 years of cooling time. This ensures that the inventory used in this calculation exceeds that of the fuel authorized for storage. The inventory for the MPC-68F was based on the GE 6x6 fuel assembly with a burnup of 30,000 MWD/MTU, 18 years of cooling time, and 1.8% enrichment. The CoC limits the burnup and cooling time of fuel (intact, damaged or debris) in an MPC-68F to a maximum of 30,000 MWD/MTU at a minimum of 18 years cooling time. Additionally, the MPC-68F was analyzed containing 67 GE 6x6 assemblies and a DFC containing 18 thorium rods. Finally, an Sb-Be source stored in one fuel rod in one assembly with 67 GE 6x6 assemblies was analyzed. The isotopes which contribute greater than 0.1% to the total curie inventory for the fuel assembly are considered in the evaluation as fines. The analysis also includes actinides as the dose conversion factors for these isotopes are in general, orders of magnitude greater than other isotopes (e.g., isotopes of plutonium, americium, curium, and neptunium were included regardless of their contribution to the inventory). A summary of the isotopes available for release is provided in Table 7.3.1.

7.3.2 Crud Radionuclides

The majority of the activity associated with crud is due to ^{60}Co [7.3.2]. The inventory for ^{60}Co was determined by using the crud surface activity for PWR rods (140×10^{-6} Ci/cm^2) and for BWR rods (1254×10^{-6} Ci/cm^2) provided in NUREG/CR-6487 [7.3.2] multiplied by the surface area per assembly (3×10^5 cm^2 and 1×10^5 cm^2 for PWR and BWR, respectively, also provided in NUREG/CR-6487). The source terms were then decay corrected (3~~5~~ years for the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F*, MPC-68 and MPC-68FF; 18 years for the MPC-68F) using the basic radioactive decay equation:

Equation 7-1:

$$A(t) = A_0 e^{-\lambda t}$$

where:

$A(t)$ is activity at time t [Ci]

A_0 is the initial activity [Ci]

λ is the $\ln 2/t_{1/2}$ (where $t_{1/2} = 5.272$ years for ^{60}Co)

t is the time in years (3~~5~~ years for the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F*,

MPC-68 and MPC-68FF; 18 years for the MPC-68F)

Total ^{60}Co crud is 140 μCi/cm^2 for PWR and 1254 μCi/cm^2 for BWR [7.3.2].

PWR
Surface area per Assy = 3.0E+05 cm^2
140 μCi/cm^2 x 3.0E+05 cm^2 = 42.0 Ci

BWR
Surface area per Assy = 1.0E+05 cm^2
1254 μCi/cm^2 x 1.0E+05 cm^2 = 125.4 Ci

$^{60}Co(t) = {}^{60}Co_0\, e^{-(\lambda t)}$, where $\lambda = \ln 2/t_{1/2}$, t = ~~53~~ *53* years (for the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F* and MPC-68), t = 18 years (MPC-68F), $t_{1/2}$ = 5.272 years for ^{60}Co [7.3.3]

MPC-24, MPC-24E, MPC-24EF ~~and MPC-32~~
MPC-32 and –32F
$^{60}Co(5) = 42.0\ Ci\ e^{-(\ln 2/5.272)(3\cancel{5})}$

$^{60}Co(5) = 2\cancel{8.31}\cancel{1.77}$ Ci

MPC-68 and MPC-68FF
$^{60}Co(5) = 125.4\ Ci\ e^{-(\ln 2/5.272)(3)}$
$^{60}Co(5) = 84.53\ Ci$ ~~$^{60}Co(5) = 125.4\ Ci\ e^{-(\ln 2/5.272)(5)}$~~
~~$^{60}Co(5) = 64.98$ Ci~~

MPC-68F

$^{60}Co(18) = 125.4\ Ci\ e^{-(\ln 2/5.272)(18)}$
$^{60}Co(18) = 11.76$ Ci

A summary of the ^{60}Co inventory available for release is provided in Table 7.3.1.

7.3.3 Release of Contents Under Non-Mechanistic Accident Conditions of Storage

7.3.3.1 Confinement Boundary Leakage Rate

The helium leak rate testing performed on the MPC confinement boundary verifies the helium leak rate under reference test conditions to be less than or equal to 5x10^{-6} atm-cm^3/s [1] as required by the Technical Specifications. As demonstrated by analysis, the MPC confinement boundary is not compromised as a result of normal, off-normal, and accident conditions. Based on the robust nature of the MPC confinement boundary, the NDE inspection of the welds, and the measurement of the helium leakage rate, there is essentially no leakage. However, it is conservatively assumed that the maximum possible leakage rate under reference test conditions from the confinement vessel is 7.5x10^{-6} atm-cm^3/s. The actual leakage test is performed at an elevated pressure (~~90 +10/ 0~~ *85*- psig *min*) to magnify the leakage rate. For purposes of determining the leak hole diameter, reference test condition parameters from Table 7.3.7 are used in Equation 7-2 and Equation 7-3 as it results in a larger leak hole diameter.

[1] According to ANSI N14.5 (1997), the mass-like leakage rate specified herein is often used in leakage testing. This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions.

Equation B-1 of ANSI N14.5 (1997) [7.3.8] is used to express this mass-like helium flow rate (Q_u) measured in atm-cm^3/s as a function of the upstream volumetric leakage rate (L_u) as follows:

Equation 7-2

$$Q_u = L_u * P_u \quad \text{atm-cm}^3\text{/sec (Equation B-1 from ANSI N14.5(1997))}$$

$$L_u = Q_u/P_u \quad \text{cm}^3\text{/sec}$$

where:

L_u is the upstream volumetric leakage rate [cm^3/s],
Q_u is the mass-like helium leak rate [atm cm^3/s], and
P_u is the upstream pressure [ATM]

The corresponding leakage rate at accident conditions is determined using the following methodology. For conservatism, unchoked flow correlations were used as the unchoked flow correlations better approximate the true measured flowrate for the leakage rates. Using the equations for molecular and continuum flow, Equation B-5 provided in ANSI N14.5-1997 [7.3.8], the corresponding capillary diameter, D, was calculated. For conservatism, the upstream pressure at reference test conditions (inside of the MPC) is assumed to be 2 ATM (minimum) and the down stream pressure (outside of the MPC) is assumed to be 1 ATM (at 298 K), therefore, the average pressure is 1.5 ATM. The evaluation was performed using the helium gas temperature at reference test conditions of both 70°F and 212°F. These temperatures are representative of the possible temperature of the helium gas in the confinement vessel during the helium leak test. The 212°F helium temperature is the upper bound because the water inside the MPC is shown not to boil in Chapter 4 as long as the "time-to-boil" time limit is not exceeded. From the two calculations using the two temperatures, it was determined that the higher temperature (212°F) results in a greater capillary diameter. The capillary length required for Equation 7-3 was conservatively chosen to be the minimum MPC lid closure weld, which is 1.9 cm. Table 7.3.6 provides a summary of the parameters used in the calculation.

Equation 7-3

$$L_u = [\frac{2.49x10^6 D^4}{a\,u} + \frac{3.81x10^3 D^3 \sqrt{\frac{T}{M}}}{a\,P_a}][P_u - P_d][\frac{P_a}{P_u}]$$

where:

L_u is the allowable leakage rate at the upstream pressure [cm^3/s],
a is the capillary length [cm],
T is the temperature [°K],
M is the gas molecular weight [g/mole] from ANSI N14.5, Table B1 [7.3.8],

u is the fluid viscosity for helium [cP] from Rosenhow and Hartnett [7.3.9]
P_u is the upstream pressure [ATM],
P_d is the downstream pressure [ATM], and
P_a is the average pressure; $P_a = (P_u + P_d)/2$ [ATM].
D is the capillary diameter [cm].

The capillary diameter (D) computed from the above equation is equal to 4.96×10^{-4} cm.

Using the capillary diameter determined above, and the parameters for accident conditions provided in Table 7.3.6, Equation 7-3 was solved for the leakage rate at the upstream conditions. The resultant hypothetical accident leakage rate~~,~~ *of* 1.95×10^{-5} cm^3/s ~~(at 843 K, 16.31 ATM)~~ was calculated.

7.3.3.2 ~~Percentage of Nuclides that Remain Airborne~~*Gravitational Settling*

~~In addition to the small fraction of fines that are released in the event of a cladding breach, only about 10% of the fines released to the MPC cavity remain airborne long enough to be available for release from the cask MPC [7.3.11]. It is conservatively assumed that 100% of the volatiles, crud and gases remain airborne and available for release.~~ *The fines, volatiles and crud that are released from the fuel cladding to the cask cavity do not remain airborne inside the cask cavity for the entire duration of the accident conditions. Therefore, credit is taken for gravitational settling of the fines, volatiles and crud in accordance with the methodology presented in reference [7.2.3].*

7.3.3.3 Fraction of Volume Released

The minimum free volume of each MPC design the confinement vessel is presented in Table 4.4.14, 4.4.13, 4.4.24, and 4.4.25[2]. Using these volumes and the upstream hypothetical accident leakage rate ~~of 1.25×10^{-5} cm^3/s~~, the fraction of the volume released per second is calculated. For the analysis of the MPC-24 and MPC-24E, the smaller of the two minimum free volumes was conservatively chosen.

7.3.3.4 Release Fraction

The release fraction is that portion of the total radionuclide inventory that is released from the cladding to the MPC cavity. The release fractions provided in NUREG/CR-6487 [7.3.2] are used. A summary of the release fractions is provided in Table 7.3.1.

7.3.3.5 Radionuclide Release Rate

The radionuclide release rate is the product of the quantity of isotopes available for release, the number of assemblies, the ~~percentage of nuclides that remain airborne~~*aerosol deposition factor*, the fraction of volume released, and the release fraction.

[2] *These volumes have been conservatively reduced by 1.67×10^5 cm^3.*

7.3.3.6 Atmospheric Dispersion Factor

The short-term accident condition atmospheric dispersion factor at 100 meters was determined using Regulatory Guide 1.145 [7.3.4]. In accordance with NUREG-1536 [7.0.2], the dispersion factor was determined on the basis of F-stability diffusion, a wind speed of 1 m/s, and plume meandering.

Reg Guide 1.145 [7.3.4] specifies that χ/Q be calculated using the following three equations. The values determined using Equations 7-4 and 7-5 should be compared and the higher value selected. This value should be compared with the value determined using Equation 7-6, and the lower value of these two should be selected as the appropriate χ/Q value. This methodology was used to determine the value for χ/Q.

Equation 7-4

$$\frac{\chi}{Q} = \frac{1}{U(\pi \sigma_y \sigma_z + A/2)}$$

Equation 7-5

$$\frac{\chi}{Q} = \frac{1}{U(3\pi \sigma_y \sigma_z)}$$

Equation 7-6

$$\frac{\chi}{Q} = \frac{1}{U \pi \Sigma_y \sigma_z}$$

where:

χ/Q is relative concentration, in sec/m^3,
π is 3.14159,
U is windspeed at 10 meters above plant grade, in m/sec,
σ_y is lateral plume spread, in meters, a function of atmospheric stability and distance (Figure 1, Reg Guide 1.145 [7.3.4]),
σ_z is vertical plume spread, in meters, a function of atmospheric stability and distance (Figure 2, Reg Guide 1.145 [7.3.4]),
Σ_y is lateral plume spread with meander and building wake effects, in m, = M σ_y, where M is determined from Figure 3, Reg Guide 1.145 [7.3.4], and
A is the smallest vertical-plane cross-sectional area of the structure (cross section of the MPC), m^2.

Equations 7-4 through 7-6 were solved using the parameters presented in Table 7.3.5. The

atmospheric dispersion factor, χ/Q, at 100 meters was selected in accordance with the methodology described above. The χ/Q value used to determine the dose is 8.0×10^{-3} sec/m^3. This short-term accident condition χ/Q is deemed conservative for an accident evaluation period of 30 days.

7.3.3.7 Dose Conversion Factors

Dose Conversion Factors (DCF) from EPA Federal Guidance Report No. 11, Table 2.1 [7.3.5] and EPA Federal Guidance Report No. 12, Table III.1 [7.3.6] were used for the analysis. ~~The DCFs are provided on the spread sheets included as Appendix 7.A.~~

7.3.3.8 Occupancy Time

An occupancy time of 720 hours (30 days) is used for the analysis [7.0.2]. This conservatively assumes that the individual is exposed 24 hours per day for 30 days at the minimum controlled area boundary of 100 meters. The accident event duration is considered conservative as any accident condition of storage resulting in the failure of 100% of the stored fuel rods would be detected by the routine security and surveillance inspections and corrective actions would be completed prior to the end of this 30-day period.

7.3.3.9 Breathing Rate

A breathing rate of 3.3×10^{-4} m^3/sec for a worker is used for the analysis [7.0.2]. This assumption is in accordance with the guidance provided in NUREG-1536 [7.0.2] for a worker.

7.3.4 Postulated Accident Doses

The following doses to an individual at the site boundary (100 meters) as a result of an assumed effluent release under accident conditions of storage were determined; the committed dose equivalent (CDE) from inhalation and the deep dose equivalent (DDE) from submersion for critical organs and tissues (gonad, breast, lung, red marrow, bone surface, thyroid); the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from submersion for the whole body; the lens dose equivalent (LDE) for the lens of the eye; the shallow dose equivalent (SDE) from submersion for the skin; and the resulting Total Effective Dose Equivalent (TEDE) and Total Organ Dose Equivalent (TODE). These doses are determined for each type of MPC. The postulated doses as a result of exposure to soil with ground surface contamination and soil contaminated to a depth of 15 cm were also determined. The resultant doses were negligible compared to the those resulting from submersion in the plume and are therefore not reported.

The doses were determined using spreadsheet software. The resultant doses are summarized for each MPC type in Tables 7.3.2 through Table 7.3.5 of the HI-STORM FSAR. ~~Example spread sheets used for the dose estimates are presented in Appendix 7.A.~~

7.3.4.1 Whole Body Dose (Total Effective Dose Equivalent)

The Total Effective Dose Equivalent is the sum of the inhaled committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) to the whole body from submersion in the plume. The postulated doses were determined using spreadsheet software. ~~Example spread sheets are provided in Appendix 7.A.~~

The CEDE is the product of radionuclide release rate, the atmospheric dispersion factor, the occupancy time, the breathing rate, and the effective dose conversion factor. The deep dose equivalent to the whole body from submersion is the product of the nuclide release rate, the atmospheric dispersion factor, the occupancy time, and the effective dose conversion factor.

7.3.4.2 Critical Organ Dose

The dose to the critical organ (or tissue) is the sum of the committed dose equivalent to the critical organ or tissue from inhalation and the deep dose equivalent to the organ or tissue from submersion in the plume. The postulated doses as a result of exposure to soil with ground surface contamination and soil contaminated to a depth of 15 cm were also determined. The resultant doses were negligible compared to ~~the~~ those resulting from submersion in the plume and are therefore not reported.

The committed dose equivalent to the organ or tissue from inhalation is the product of radionuclide release rate, the atmospheric dispersion factor, the occupancy time, the breathing rate, and the organ/tissue dose conversion factor. The deep dose equivalent to the organ or tissue from submersion in the plume is the product of the nuclide release rate, the atmospheric dispersion factor, the occupancy time, and the organ/tissue dose conversion factor.

The lens dose equivalent (LDE) as a result of submersion in the plume was estimated using guidance from Dr. James Turner in his book, *Atoms, Radiation, and Radiation Protection* [7.3.10]. Dr. Turner states that alpha particles and low-energy beta particles, such as those from tritium, cannot penetrate to the lens of the eye (at a depth of 3 mm). The discussion continues that many noble gases emit photons and energetic beta particles, which in turn must be considered in the dose estimate. Dr. Turner states that the dose-equivalent rate to tissues near the surface of the body (e.g., lens of the eye) is more than 130 times the dose-equivalent rate in the lung from gases contained in the lung. Using the accident condition of storage for the MPC-68 and the MPC-32 (which have the highest dose to the lung for BWR and PWR fuel respectively), the estimated dose to the lung from gases in the lung is $4.63\cancel{3.60}\times10^{-3}$ mrem and $4.88\cancel{5}.79\times10^{-3}$ mrem, respectively. Conservatively multiplying this value by 150, the estimated LDE is 0.695~~540~~ mrem for BWR fuel and 0.869~~732~~ mrem for PWR fuel. These estimated LDEs for BWR and PWR fuel are a small fraction of the 15 rem limit imposed by 10CFR72.106(b).

7.3.5 Site Boundary

The estimated accident doses at the controlled area boundary are highest for the accident condition of storage for the MPC-68 for BWR fuel and the MPC-32 for PWR fuel. The estimated TEDEs

~~(39.2mrem for BWR fuel and 29.1 mrem for PWR fuel)~~*presented in Table 7.3.8* are small fractions of the 5 rem whole body limit imposed by 10 CFR 72.106(b). The maximum estimated Total Organ Dose Equivalents (TODE) to the lung ~~and bone surface,~~ which are the highest critical organ doses *from*~~to~~ BWR and PWR fuel, ~~respectively, (205 mrem and 233 mrem, respectively)~~ are small fractions of the 50 rem critical organ limit imposed by 10 CFR 72.106(b). Additionally, the shallow dose equivalents to the skin ~~(0.303 mrem and 0.202 mrem)~~ are small fractions of the 50 rem shallow dose equivalent to skin or other extremity limit imposed by 10 CFR 72.106(b).

7.3.6 Assumptions

The following presents a summary of assumptions for the accident condition confinement analysis of the HI-STORM 100 System.

- The distance from the cask to the site boundary is 100 meters.
- 100% of the fuel rods have ruptured. This assumption is conservative because it results in the greatest potential release of radioactive material.
- Unchoked flow correlations were used as the unchoked flow correlations better approximate the true measured flowrate for the leakage rates associated with transportation packages.
- For conservatism, the upstream pressure at reference test conditions (inside of the MPC) is assumed to be 2 ATM and the down stream pressure (outside of the MPC) is assumed to be 1 ATM.
- The leak hole diameter is determined using reference test conditions rather than actual test conditions from Table 7.3.7. This is conservative, as it yields a larger leak hole diameter.
- The temperature at test conditions is assumed to be equal to an ambient reference temperature, 212° F based on the maximum temperature achievable by the water in the MPC during performance of the leak test. This is conservative because the leak hole diameter computed from test conditions is larger.
- Bounding accident conditions (i.e., MPC cavity pressure of 225 psig (which is above the design pressure of 200 psig) at peak cladding temperature limit (570° C)) are postulated for this analysis.
- The capillary length required for Equation 7-3 was conservatively chosen to be the MPC lid closure weld which is 1.9 cm.
- The majority of the activity associated with crud is due to ^{60}Co. This assumption follows from the discussion provided in NUREG/CR-6487 [7.3.2].
- The accident condition leakage rate persists for 30 days without a decrease in the rate or

nuclide concentration.

- The individual at the site boundary is exposed for 720 hours (30 days). This conservatively assumes that the individual is exposed 24 hours per day for 30 days.

- A breathing rate of 3.3 x 10^{-4} m^3/sec for a worker is used for the analysis [7.0.2]. This assumption is in accordance with the guidance provided in NUREG-1536 for a worker.

- All fuel stored in the MPC is of the design basis type with a bounding burnup and cooling time.

- Exposure to dose conversion factors for inhalation reported in EPA Federal Guidance Report No. 11, Table 2.1 [7.3.5] were selected by the most restrictive clearance class for each organ and each radionuclide.

- For conservativism, the maximum possible leakage rate at reference test conditions is assumed to be 7.5×10^{-6} atm-cm^3/s, which is 150% of the test leak rate of 5.0×10^{-6} atm-cm^3/s.

- *The MPC internal free volumes presented in Section 4.4 are conservatively reduced by 1.67×10^5 cm^3.*

Table 7.3.1
Isotope Inventory and Release Fraction

Nuclide	MPC-24, MPC-24E MPC-24EF, MPC-32 *MPC-32F* Ci/Assembly	MPC-68 MPC-68FF Ci/Assembly	MPC-68F Ci/Assembly	Release Fraction [7.3.2]
Gases				
^{3}H	*4.37~~3.68~~*E+02	1.59~~24~~E+02	1.78E+01	0.30
^{129}I	3.53~~31~~E-02	1.30~~12~~E-02	3.49E-03	0.30
^{85}Kr	*7.02~~5.86~~*E+03	2.57~~04~~E+03	2.37E+02	0.30
Crud				
^{60}Co	2.83~~18~~E+01	*8.45~~6.50~~*E+01	1.18E+01	0.15 normal/off-normal 1.0 accident
Volatiles				
^{90}Sr	*7.02~~6.32~~*E+04	2.65~~24~~E+04	4.29E+03	2.0E-04
^{106}Ru	*6.41~~1.59~~*E+04	*1.98~~4.71~~*E+04~~3~~	2.30E-01	2.0E-04
^{134}Cs	*8.63~~4.04~~*E+04	*2.79~~1.18~~*E+04	3.16E+01	2.0E-04
^{137}Cs	*1.10~~9.82~~*E+05~~4~~	*4.05~~3.35~~*E+04	7.21E+03	2.0E-04
Fines				
^{241}Pu	*9.65~~8.53~~*E+04	*3.01~~2.58~~*E+04	5.16E+03	3.0 E-05
^{90}Y	*7.02~~6.32~~*E+04	2.65~~24~~E+04	4.29E+03	3.0 E-05
^{147}Pm	*4.46~~2.63~~*E+04	*1.61~~9.63~~*E+04~~3~~	1.18E+02	3.0 E-05
^{144}Ce	*4.78~~8.14~~*E+04~~3~~	*1.44~~2.48~~*E+04~~3~~	--	3.0 E-05
^{144}Pr	*4.78~~8.14~~*E+04~~3~~	*1.44~~2.48~~*E+04~~3~~	--	3.0 E-05
^{154}Eu	*7.48~~5.90~~*E+03	*2.41~~1.74~~*E+03	1.44E+02	3.0 E-05
^{244}Cm	1.30~~01~~+04	*3.99~~2.38~~*E+03	2.17E+02	3.0 E-05
^{238}Pu	*6.65~~5.81~~*E+03	*2.10~~1.58~~*E+03	2.50E+02	3.0 E-05
^{125}Sb	*3.98~~2.30~~*E+03	*1.43~~7.91~~*E+02	--	3.0 E-05
^{155}Eu	*2.39~~1.65~~*E+03	*8.58~~5.41~~*E+02	--	3.0 E-05

Table 7.3.1
(continued)

Isotope Inventory and Release Fractions

Nuclide	MPC-24, MPC-24E MPC-24EF, MPC-32 *MPC-32F* Ci/Assembly	MPC-68 MPC-68FF Ci/Assembly	MPC-68F Ci/Assembly	Release Fraction [7.3.2]
^{241}Am	*6.349.00*E+02	2.*0578*E+02	2.52E+02	3.0 E-05
^{125m}Te	*9.735.61*E+02	*3.511.93*E+02	--	3.0 E-05
^{240}Pu	4.*105*E+02	1.*650*E+02	6.81E+01	3.0 E-05
^{151}Sm	3.*5738*E+02	*1.079.76*E+02 1	--	3.0 E-05
^{239}Pu	2.064E+02	6.730E+01	2.95E+01	3.0 E-05
^{137m}Ba	*1.059.27*E+054	3.*8217*E+04	6.81E+03	3.0 E-05
^{106}Rh	*6.411.59*E+04	*1.984.71*E+04 3	--	3.0 E-05
^{144m}Pr	*6.701.14*E+02	*2.023.48*E+02 1	--	3.0 E-05
^{243}Am	*5.464.87*E+01	1.*8036*E+01	3.30E+00	3.0 E-05
^{242}Cm	*6.103.23*E+021	*2.009.82*E+02 0	7.71E-01	3.0 E-05
^{243}Cm	*4.263.63*E+01	*1.329.79*E+01	1.54E+00	3.0 E-05
^{239}Np	*5.464.87*E+01	1.*8036*E+01	3.30E+00	3.0 E-05
^{237}Np	*4.153.88*E-01	1.*3718*E-01	2.72E-02	3.0 E-05
^{242}Pu	*3.092.85*E+00	*1.119.04*E+-0*04*	3.06E-01	3.0 E-05
^{242}Am	*9.248.72*E+00	2.*9971*E+00	9.31E-01	3.0 E-05

^{242m}Am	*9.28~~8.76~~*E+00	*3.00~~2.72~~*E+00	9.35E-01	3.0 E-05
^{242m}Am	--	*3.35E+02*	--	*3.0 E-05*

Note: The isotopes which contribute greater than 0.1% to the total curie inventory for the fuel assembly are considered in the evaluation as fines. The analysis also includes actinides as the dose conversion factors for these isotopes are in general, orders of magnitude greater than other isotopes (e.g., isotopes of plutonium, americium, curium, and neptunium were included regardless of their contribution to the inventory).

Table 7.3.2

MPC-24, MPC-24E and MPC-24EF
Postulated Doses
To An Individual at the Controlled Area Boundary (100 meters)
As a Result of an Assumed Effluent Release

Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	1.56E-04	1.07E-04	8.09E-03	1.04E-03	4.91E-03	2.23E-04
DDE	6.02E-06	6.89E-06	5.89E-06	5.64E-06	1.11E-05	6.08E-06
ADE	1.62E-04	1.14E-04	8.10E-03	1.05E-03	4.92E-03	2.29E-04

	Skin/Extremity
SDE	5.96E-04

	Whole Body
CEDE	1.23E-03
DDE	6.13E-06
ADE	1.24E-03

Off-Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	8.41E-04	5.39E-04	4.33E-02	5.63E-03	2.67E-02	1.18E-03
DDE	3.20E-05	3.66E-05	3.13E-05	2.99E-05	5.92E-05	3.23E-05
ADE	8.73E-04	5.76E-04	4.33E-02	5.66E-03	2.68E-02	1.21E-03

	Skin/Extremity
SDE	3.25E-03

	Whole Body
CEDE	6.56E-03
DDE	3.25E-05
ADE	6.59E-03

Accident Conditions [mrem/30 days]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	1.81E-01	2.62E-02	2.90E+00	1.05E+00	1.10E+01	5.65E-02
DDE	1.53E-03	1.75E-03	1.50E-03	1.43E-03	2.83E-03	1.55E-03
TODE	1.83E-01	2.80E-02	2.90E+00	1.05E+00	1.10E+01	5.81E-02

	Skin/Extremity
SDE	1.54E-01

	Whole Body
CEDE	8.61E-01
DDE	1.56E-03
TEDE	8.63E-01

Table 7.3.3

MPC-32 *and MPC-32F*
Postulated Doses
To An Individual at the Controlled Area Boundary (100 meters)
As a Result of an Assumed Effluent Release

Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	2.19E-04	1.49E-04	1.13E-02	1.45E-03	6.86E-03	3.12E-04
DDE	8.42E-06	9.63E-06	8.24E-06	7.89E-06	1.55E-05	8.51E-06
ADE	2.27E-04	1.59E-04	1.13E-02	1.46E-03	6.88E-03	3.21E-04

	Skin/Extremity
SDE	8.33E-04

	Whole Body
CEDE	1.72E-03
DDE	8.57E-06
ADE	1.73E-03

Off-Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	1.18E-03	7.54E-04	6.06E-02	7.88E-03	3.74E-02	1.65E-03
DDE	4.47E-05	5.12E-05	4.37E-05	4.18E-05	8.28E-05	4.52E-05
ADE	1.22E-03	8.05E-04	6.06E-02	7.92E-03	3.75E-02	1.70E-03

	Skin/Extremity
SDE	4.54E-03

	Whole Body
CEDE	9.17E-03
DDE	4.55E-05
ADE	9.22E-03

Accident Conditions [mrem/30 days]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	2.52E-01	3.66E-02	4.04E+00	1.47E+00	1.54E+01	7.89E-02
DDE	2.14E-03	2.45E-03	2.09E-03	2.00E-03	3.96E-03	2.16E-03
TODE	2.54E-01	3.91E-02	4.04E+00	1.47E+00	1.54E+01	8.11E-02

	Skin/Extremity
SDE	2.15E-01

	Whole Body
CEDE	1.20E+00
DDE	2.18E-03
TEDE	1.20E+00

Table 7.3.4
MPC-68 and MPC-68FF
Postulated Doses
To An Individual at the Controlled Area Boundary (100 meters)
As a Result of an Assumed Effluent Release

Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	1.80E-04	1.93E-04	1.04E-02	1.22E-03	5.07E-03	3.15E-04
DDE	8.33E-06	9.51E-06	8.20E-06	7.91E-06	1.47E-05	8.46E-06
ADE	1.88E-04	2.03E-04	1.04E-02	1.23E-03	5.08E-03	3.23E-04

	Skin/Extremity
SDE	6.71E-04

	Whole Body
CEDE	1.59E-03
DDE	8.49E-06
ADE	1.60E-03

Off-Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	8.82E-04	6.61E-04	4.95E-02	6.27E-03	2.74E-02	1.37E-03
DDE	3.75E-05	4.29E-05	3.67E-05	3.52E-05	6.87E-05	3.79E-05
ADE	9.20E-04	7.04E-04	4.95E-02	6.31E-03	2.75E-02	1.41E-03

	Skin/Extremity
SDE	3.65E-03

	Whole Body
CEDE	7.40E-03
DDE	3.81E-05
ADE	7.44E-03

Accident Conditions [mrem/30 days]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	1.78E-01	3.59E-02	3.24E+00	1.05E+00	1.07E+01	6.91E-02
DDE	1.87E-03	2.14E-03	1.84E-03	1.77E-03	3.40E-03	1.90E-03
TODE	1.80E-01	3.80E-02	3.24E+00	1.05E+00	1.07E+01	7.10E-02

	Skin/Extremity
SDE	1.73E-01

	Whole Body
CEDE	8.93E-01
DDE	1.91E-03
TEDE	8.95E-01

Table 7.3.5

MPC-68F
Postulated Doses
To An Individual at the Controlled Area Boundary (100 meters)
As a Result of an Assumed Effluent Release

Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	9.75E-06	1.60E-05	6.86E-04	8.11E-05	3.23E-04	1.45E-05
DDE	4.80E-07	5.45E-07	4.77E-07	4.65E-07	7.89E-07	4.90E-07
ADE	1.02E-05	1.65E-05	6.86E-04	8.16E-05	3.24E-04	1.50E-05

	Skin/Extremity
SDE	2.50E-05

	Whole Body
CEDE	1.08E-04
DDE	4.90E-07
ADE	1.08E-04

Off-Normal Conditions [mrem/yr]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	8.00E-05	5.28E-05	5.55E-03	8.33E-04	3.73E-03	5.12E-05
DDE	2.90E-06	3.32E-06	2.84E-06	2.72E-06	5.32E-06	2.93E-06
ADE	8.29E-05	5.61E-05	5.55E-03	8.36E-04	3.74E-03	5.41E-05

	Skin/Extremity
SDE	2.93E-04

	Whole Body
CEDE	8.36E-04
DDE	2.95E-06
ADE	8.39E-04

Accident Conditions [mrem/30 days]

	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid
CDE	2.63E-02	3.40E-03	4.30E-01	1.63E-01	1.67E+00	3.25E-03
DDE	1.70E-04	1.93E-04	1.66E-04	1.60E-04	3.06E-04	1.71E-04
TODE	2.65E-02	3.59E-03	4.30E-01	1.63E-01	1.67E+00	3.42E-03

	Skin/Extremity
SDE	1.60E-02

	Whole Body
CEDE	1.28E-01
DDE	1.72E-04
TEDE	1.28E-01

Table 7.3.6
χ/Q Parameters

Parameter	Value	Reference
U	1 m/s	NUREG-1536 [7.0.2]
σ_y	4.0 m	Figure 1, Reg Guide 1.145 [7.3.4]
σ_z	2.5 m	Figure 2, Reg Guide 1.145 [7.3.4]
$\Sigma_y = M\ \sigma_y$	16	M is determined from Figure 3, Reg Guide 1.145 [7.3.4]
A	8.41 m^2	Chapter 1, Section 1.5

Table 7.3.7

Parameters for Test, Normal/Off-Normal and Hypothetical Accident Conditions

Parameter	Reference Test	Actual Test	Normal/Off-Normal[1]	Hypothetical Accident
P_u	2 ATM (min)	6.78 ATM (min)	*7.80/9.*~~56.9~~0 ATM	16.31 ATM
P_d	1 ATM	1 ATM	1 ATM	1 ATM
T	373 K	373 K	581 K	843 K
M	4 g/mol	4 g/mol	4 g/mol	4 g/mol
u (helium)	0.0231 cP	0.0231 cP	0.0309 cP	0.0397 cP
a	1.9 cm	1.9 cm	1.9 cm	1.9 cm

[1]The values in this column, *with the exception of the pressure* -are for the off-normal condition~~. They uniformly bound the normal condition values.~~, *as they bound the normal condition values. Pressure values are given for both normal and off-normal conditions and correspondingly used to determine the leakage rate for each condition.*

Table 7.3.8

Postulated Bounding Doses Compared to Regulatory Limits
To An Individual at the Controlled Area Boundary (100 meters)
As a Result of an Assumed Effluent Release

	BWR	PWR	Regulatory Limit
10CFR72.104(a) - Normal			
Whole body ADE	*1.60E-03*~~0.515~~ mrem	*1.73E-03*~~0.159~~ mrem	25 mrem
Thyroid ADE	*3.23E-04*~~0.129~~ mrem	*3.21E-04*~~0.022~~ mrem	75 mrem
Critical Organ ADE (Max)	*1.04E-02*~~2.95~~ mrem	*1.13E-02*~~0.900~~ mrem	25 mrem
10CFR72.104(a) - Off-normal			
Whole body ADE	*7.44E-03*~~0.735~~ mrem	*9.22E-03*~~0.486~~ mrem	25 mrem
Thyroid ADE	*1.41E-03*~~0.136~~ mrem	*1.70E-03*~~0.033~~ mrem	75 mrem
Critical Organ ADE (Max)	4.*95E-02*~~.07~~ mrem	*6.06E-02*~~4.08~~ mrem	25mrem
10CFR72.106(b) - Accident			
TEDE	*0.895*~~39.2~~ mrem	*1.20*~~29.1~~ mrem	5 rem
TODE=DDE+CDE (Max)	*10.7*~~216~~ mrem	*15.4*~~254~~ mrem	50 rem
LDE	0.*869*~~540~~ mrem	0.*695*~~732~~ mrem	15 rem
SDE	0.*173*~~303~~ mrem	*0.215*~~0.212~~ mrem	50 rem

ADE: Annual Dose Equivalent
TEDE: Total Effective Dose Equivalent
TODE: Total Organ Dose Equivalent
DDE: Deep Dose Equivalent
CDE: Committed Dose Equivalent
LDE: Lens Dose Equivalent
SDE: Shallow Dose Equivalent

7.4 REFERENCES

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

[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.

[7.2.1] Interim Staff Guidance-11, Revision 1, "Transportation and Storage of Spent Fuel Having Burnups in Excess of 45GWD/MTU", May 16, 2000.

[7.2.2] Interim Staff Guidance-5, Revision 1, "Normal, Off-Normal, and Hypothetical Dose Estimate Calculations", June 18, 1999.

[7.2.3] Schaperow, Jason H. "Best-Estimate Offsite Dose from Dry Storage Cask Leakage," USNRC, SMSAB-00-03, June 2000.

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[7.3.6] U.S. EPA, Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil*, EPA 402-R-93-081, 1993.

[7.3.7] International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1; Pergamon

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[7.3.8] ANSI N14.5-1997. "American National Standard for Radioactive Material Leakage Tests on Packages for Shipment."

[7.3.9] Rosenhow, W.M.. and Hartnett, J.P., *Handbook of Heat Transfer*, McGraw Hill Book Company, New York, 1973.

[7.3.10] Turner, James E. *Atoms, Radiation , and Radiation Protection*, McGraw Hill Book Company, New York, 1992.

[7.3.11] Rashid, Y.R., et al, "An Estimate of the Contribution of Spent Fuel Products to the Releaseable Source Term in Spent Fuel Transportation Casks," SAND88-2778C, Sandia National Laboratories, 1988

APPENDIX 7.A

DELETED

~~EXAMPLE DOSE CALCULATIONS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS OF STORAGE~~

~~MPC-32, Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-32, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-32, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-32, Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-32, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-32, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-68, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-68, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-68, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-68, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

NO CHANGES REQUIRED IN CHAPTER 8

HI-TRAC. Measurements shall be taken at the locations specified in the Technical Specifications in Appendix A to CoC 72-1014 and, if necessary, average dose rates computed for comparison against the prescribed limits. The results of the dose rate measurements shall be compared to the limits specified in the Technical Specifications. The test is considered acceptable if the dose rate readings are less than or equal to limits in the Technical Specifications. If dose rates are higher than the limits, the Required Actions provided in the Technical Specifications shall be completed. Dose rate measurements shall be documented and shall become part of the quality documentation package.

9.1.5.3 Neutron Absorber Tests

Each plate of ~~Boral~~ *neutron absorber* shall be visually inspected by the manufacturer for damage (e.g., scratches, cracks, burrs, and peeled cladding, *as applicable*) and foreign material embedded in the surfaces. In addition, the MPC fabricator shall visually inspect the ~~Boral~~ *neutron absorber* plates on a lot sampling basis. The sample size shall be determined in accordance with MIL-STD-105D or equivalent. The selected ~~Boral~~*neutron absorber* plates shall be inspected for damage such as inclusions, cracks, voids, delamination, and surface finish, *as applicable*.

After manufacturing, a statistical sample of each lot of ~~Boral~~*the neutron absorber* shall be tested using *a proven method, such as* wet chemistry, *spectragraphy,* ~~and/~~or neutron attenuation ~~techniques~~ *testing* to verify a minimum ^{10}B content (areal density) at the ends of the panel. The minimum ^{10}B loading of the ~~Boral~~*neutron absorber* panels for each MPC model is provided in Table 2.1.15. Any panel in which ^{10}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.

Installation of ~~Boral~~*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 ~~Boral~~*Neutron absorber* panel in accordance with Design Drawings in Chapter 1. These quality control processes, in conjunction with ~~Boral~~*in-process* manufacturing testing, provide the necessary assurances that the ~~Boral~~*neutron absorber* will perform its intended function. No additional testing or in-service monitoring of the ~~Boral~~*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 Design 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

CHAPTER 10: RADIATION PROTECTION†

10.0 OVERVIEW

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.

The description of the shielding analysis in Chapter 5 addresses the impact of the latest approved cask contents on the shielding effectiveness of the cask system. The occupational exposures estimated in this chapter are reference values based on the generic operating guidelines contained in Chapter 8. The radiation source term used in these estimates is consistent with the authorized contents for the initial certification of the HI-STORM 100 System in May 2000 and as modified by CoC Amendment 1. These dose estimates are not revised as the authorized contents of the cask system are altered further and are not to be considered bounding values by users. Each cask user must uniquely evaluate the expected and actual occupational exposures and establish appropriate protective measures in accordance with their radiation protection program, based on the site specific cask contents, ISFSI arrangement, and detailed plant operating procedures.

Not maintaining these dose estimates current with the latest authorized contents is based on the following:

- *Chapter 5 is maintained current with regard to the impact of cask contents changes and design changes that may affect the shielding effectiveness of the cask system.*
- *There are no regulatory limits on dose rates during dry storage cask loading, unloading, and storage conditions.*
- *Each user of the cask system is required by 10 CFR 72.212 to demonstrate compliance with the controlled area annual dose limits of 10 CFR 72.104 on a site-specific basis.*
- *Each user of the cask system is required to establish and maintain a radiation protection program in accordance with Section 5.0 of Appendix A to the HI-STORM 100 System CoC, and to monitor occupational radiation exposure to demonstrate compliance with the occupational radiation dose limits of 10 CFR 20 on a site-specific basis.*

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG 1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

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 ~~200~~ *250* meters and a 2x5 array of HI-STORM 100 systems at ~~350~~ *450* meters. These annual doses are based on a full array of design basis fuel with a burnup of ~~52,500~~ *47,500* MWD/MTU and ~~5~~ *5*-year cooling. This burnup and cooling time combination conservatively bounds the allowable burnup and cooling times listed in *Appendix B to the CoC* ~~the Technical Specifications~~. 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 ~~200~~ *250* meters for a single cask and at ~~350~~ *450* 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.

A minor contributor to the minimum controlled area boundary is the normal storage condition leakage from the welded MPC. Although leakage is not expected, Section 7.2 provides an analysis for the annual dose equivalent based on a continuous leak from the MPC. The annual dose equivalent to an individual at the minimum controlled area boundary based on the assumed leakage rate and continuous occupancy is presented in Table 7.3.8. 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.

However, the annual dose at the controlled area boundary as a result of an assumed effluent release under off-normal conditions is different than that under normal conditions. Under off-normal conditions, 10% of the fuel rods are assumed to have been breached, in lieu of 1% of the fuel rods for normal conditions. The resulting annual dose equivalent to an individual at the minimum controlled area boundary, based on the assumed leakage rate and continuous occupancy, is presented in Table 7.3.8. The analysis to determine the off-normal dose at the controlled area boundary is described in Section 7.2.

10.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies that 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.

Subsection 7.3 *and Table 7.3.8* demonstrates that the resultant doses for a non-mechanistic postulated breach of the MPC confinement boundary at the regulatory

minimum site boundary distance of 100 meters is presented ~~in Table 7.3.8~~ within the regulatory limits specified in 10CFR72.106 [10.0.1].

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 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 Section 5.1.2. These results bound those for the 125-Ton HI-TRAC transfer cask by a large margin. The results in that section show that the resultant dose rate at the 100-meter controlled area boundary would be approximately *3.85* ~~1.47~~ mrem/hour for the loaded HI-TRAC transfer cask during the accident condition. At the calculated dose rate, it would take approximately *54* ~~141~~ days for the dose at the controlled area boundary to reach 5 rem. This length of time is sufficient 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 100 OVERPACKS
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
~~52,500~~ *47,500* MWD/MTU AND ~~5~~-*3* YEAR COOLING

Array Configuration	**1 Cask**	**1 Cask**	**2x5 Array**
Annual Dose (mrem/year)†	*290.3* ~~130.0~~	*22.83* ~~20.19~~	*15.52* ~~18.64~~
Distance to Controlled Area Boundary (meters)††, †††	100	*250* ~~200~~	*450* ~~350~~

† 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 5-year cooling as specified in the Technical Specifications 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

Fuel Burnup & Cooling Time	**100 Meters**	**200 Meters**	**300 Meters**
43,500 ~~**42,500**~~ **MWD/MTU &** ***3*** ~~**5**~~ **Years**	*0.91* ~~0.42~~ mrem/hr	*0.14* ~~0.06~~ mrem/hr	*0.37* ~~0.02~~ mrem/hr
70,000 ~~**52,500**~~ **MWD/MTU & 10 Years**	*0.36* ~~0.26~~ mrem/hr	*0.06* ~~0.04~~ mrem/hr	*0.02* ~~0.01~~ mrem/hr

CHAPTER 11†: 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 Sections 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

During normal storage operations of the HI-STORM 100 System it is possible that an off-normal situation could occur. 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 Pressures
Off-Normal Environmental Temperatures
Leakage of One MPC Seal Weld
Partial Blockage of Air Inlets
Off-Normal Handling of HI-TRAC Transfer Cask

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.

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† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

The 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 sections 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 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for off-normal conditions are defined in Table 2.2.14. The load combinations include both normal and off-normal loads. The off-normal load combination evaluations are discussed in Section 11.1.5.

11.1.1 Off-Normal Pressures

The sole pressure boundary in the HI-STORM 100 System is the MPC internal pressure boundary. The off-normal pressure condition is specified in Section 2.2.2.1. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature obtained with maximum decay heat load design basis fuel. The maximum off-normal environmental temperature is 100°F with full solar insolation. *In accordance with NUREG 1536, failure of 10% fuel rods is combined with off-normal temperatures.* ~~The MPC internal pressure is further increased by the conservative assumption that 10% of the fuel rods rupture and 100% of the fill gas, and 30% of the fission gases are 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

Chapter 4 calculates the MPC internal pressure with an ambient temperature of 80°F, 10% fuel rods ruptured, full insolation, and maximum decay heat, and reports the maximum value of ~~75.0 psig~~ ~~99.8~~ *99.8 psig* in Table 4.4.14 at an average temperature of ~~513.6~~ *549.2* °K. Using this pressure, the off-normal temperature of 100°F (ΔT of 20°F or 11.1°K), and the ideal gas law, the off-normal resultant pressure is calculated to be below the normal condition MPC internal design pressure.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(99.8 psig + 14.7)(549.2 + 11.1^o K)}{549.2^o K}$$

$$P_2 = 116.8 psia \quad or \quad 102.1 psig$$

This calculated off-normal pressure is bounded by the ~~The~~ off-normal MPC internal design pressure ~~of 100 psig (~~ *reported in* Table 2.2.1~~)~~ ~~has been established to bound the off-normal condition~~. Therefore, no additional analysis is required.

Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is equivalent to the evaluation at normal internal pressures, since the normal design pressure was set at a value, which would encompass the off-normal pressure. Therefore, the resulting stresses from the off-normal condition are equivalent to that of the normal condition and are well within the short-term allowable values, as discussed in Section 3.4.

Thermal

The MPC internal pressure for off-normal conditions is calculated as presented above. As can be seen from the value above, the ~~100 psig~~ design basis internal pressure for off-normal conditions used in the structural evaluation bounds the calculated value above.

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. This off-normal event is of a short duration, therefore the resultant temperatures are evaluated against the accident condition temperature limits as 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 ≤32°F and prohibits use of the HI-TRAC transfer cask below 0°F.

11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considering an environmental temperature of 100°F for a duration ~~sufficient to~~*sufficient to* reach thermal equilibrium is evaluated with respect to design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 100°F environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Subsection 4.4. These temperatures are conservatively calculated at an environmental temperature of 80°F. The maximum off-normal environmental temperature is 100°F, which is an increase of 20°F. Conservatively bounding temperatures for all MPC designs (Table 1.2.1) are calculated to be as listed in Table 11.1.1. As illustrated by the table, all the maximum off-normal temperatures are below the short-term condition design basis temperatures. The maximum temperatures are the peak values and are based on the conservative assumptions applied in this analysis. The component temperatures for the HI-TRAC listed in Table 4.5.2 are all based on the maximum off-normal environmental temperature. The off-normal environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term temperature limits. Therefore, all the HI-STORM 100 System maximum off-normal temperatures meet the design requirements.

Additionally, the off-normal environmental temperature generates a pressure that is evaluated in Subsection 11.1.1. The off-normal MPC cavity pressure is less than the design basis pressure listed in Table 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 overpack. The HI-STORM overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is 0°F and the HI-TRAC is conservatively assumed to reach 0°F throughout the structure. For ambient temperatures from 0° to 32°F, a 25% ethylene glycol solution is added to the demineralized water in the water jacket to prevent freezing. Chapter 3, Subsection 3.1.2.3, 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 11.1.1.3, the resultant pressure is well below the design

pressure ~~of 100 psig~~ used in the structural analysis. The effect of the lower off-normal thermal conditions (i.e., -40°F) results in an evaluation of the potential for brittle fracture that is discussed in Section 3.1.2.3.

Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Table 11.1.1 for the HI-STORM overpack and MPC. As can be seen from this table, all temperatures for off-normal conditions are within the short-term allowable values described 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, and vent and drain port cover plates. 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 MPC lid, and vent and drain port covers are leakage tested. Additionally, the MPC lid weld is subjected to a hydrostatic test to verify its integrity.

The MPC lid-to-MPC shell weld is postulated to fail to confirm the safety of the HI-STORM 100 confinement boundary. The failure of the MPC lid weld is equivalent to the MPC drain or vent port cover weld failing. The MPC lid-to-shell weld has been selected because it is the main closure weld performed in the field for the MPC. It is extremely unlikely that the weld examination, helium leakage testing and hydrostatic testing would fail to detect a poorly welded closure plate. The MPC lid weld failure affects the MPC confinement boundary; however, no leakage will occur.

11.1.3.1 Postulated Cause of Leakage of One Seal in the Confinement Boundary

Failure of the MPC confinement boundary is highly unlikely. The MPC confinement boundary is shown to withstand all normal, off-normal, and accident conditions. There are no credible conditions that could damage the integrity of the MPC confinement boundary. The MPC lid-to-MPC shell weld is liquid penetrant inspected on the root and final pass, volumetrically inspected or liquid penetrant inspected on multiple passes, hydrostatically tested, and helium leak tested. The initial integrity of the closure welds will be maintained throughout the design life because the MPC is stored within the HI-STORM ~~overpack which~~*overpack, which* provides physical protection and a weather shield. Failure of the MPC lid-to-MPC shell weld would require all of the following:

1. Improper weld by a qualified welding machine or welder using approved welding procedures.

2. Failure to detect the unacceptable indication during the liquid penetrant or volumetric inspections performed by a qualified inspector in accordance with approved procedures.

3. Failure of the qualified leakage test equipment to detect the leak in accordance with approved procedures.

4. Failure to detect the unacceptable leak during the hydrostatic test performed by qualified personnel in accordance with approved procedures.

The evaluation of the failure of the MPC lid-to-MPC shell weld has been postulated to demonstrate the safety of the HI-STORM 100 confinement system and cannot be derived from a credible loading condition.

11.1.3.2 Detection of Leakage of One Seal in the Confinement Boundary

The HI-STORM 100 System is designed to withstand the leakage of one field weld in the confinement boundary without any effects on its ability to meet its safety requirements. As the HI-STORM 100 System can withstand the failure of one field weld with no leakage, there is no requirement to detect leakage from one seal.

11.1.3.3 Analysis of Effects and Consequences of Leakage of One Seal in the Confinement Boundary

If the MPC lid-to-MPC shell weld were to fail, the MPC closure ring will retain the design pressure. The analysis of the MPC closure ring's ability to retain the design pressure is provided in Appendix 3.E of the HI-STAR TSAR Docket Number 72-1008. The consequences of the MPC lid-to-MPC shell weld failure are that the MPC closure ring maintains the integrity of the confinement boundary.

Structural

The stress evaluation of the closure ring is discussed in Appendix 3.E. 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 leakage of one seal event does not affect the safe operation of the HI-STORM 100 System.

11.1.3.4 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 does not affect the HI-STORM 100 System's ability to operate safely.

11.1.3.5 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 the confinement barrier is not breached and shielding is not affected.

11.1.4 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with fine mesh 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.

An additional evaluation is performed with three of the four air inlet ducts. While not required by the HI-STORM System design criteria, this additional evaluation is performed as a parametric study of the effects of incremental duct blockage. The purpose of the parametric study is to demonstrate the robustness of the HI-STORM System design beyond the design basis.

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 mesh screens prevent foreign objects from entering the ducts. The mesh screens are either inspected periodically or the outlet duct air temperature is monitored as specified by Technical Specifications in Appendix A to the CoC. It is, however, possible that blowing debris may block two air inlet ducts of the overpack. As already stated, the blockage of three inlet ducts is evaluated only to demonstrate the limited effects of additional incremental duct blockage.

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 mesh screens or temperature monitoring of the outlet duct air as required and specified by Technical Specifications in Appendix A to the CoC. 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

Evaluations for two inlet ducts ~~and three inlet~~ ducts blocked ~~are~~ *is* evaluated for the ~~MPC-32 at its maximum~~ *highest design basis* decay heat load *(for MPC-68)*. ~~Only the MPC-32 is evaluated because it has the highest decay heat load of all MPC designs (Table 1.2.1).~~ The largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is ~~25~~ *36*°F[†], for the MPC shell. ~~The largest temperature rise of the MPC or its contents as a result of the blockage of three air inlet ducts (performed as a parametric study of incremental duct blockage only) is 81°F, also for the MPC shell.~~ Conservatively adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of two air inlet ducts are provided in Table 11.1.2. ~~Following this same procedure of adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of three air inlet ducts are included in the same table for comparison purposes.~~ These values are based on full insolation and an ambient temperature of 80°F. The analysis method for the blockage of two ~~and three~~ of the air inlet ducts is conservative with respect to the analysis method for the normal condition. As a result of the air inlet duct blockages, the head loss is increased and the airflow is decreased thereby increasing component temperatures.

As stated above, the largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is ~~25~~*36* °F, for the MPC shell. A bounding MPC internal pressure as a result of this calculated temperature increase is computed, based on initial conditions presented previously in Subsection 11.1.1.3, as follows:

$$P_2 = P_1 \frac{T_1 + \Delta T}{T_1}$$

where:

P_2 = Bounding MPC Cavity Pressure (psia)
P_1 = Initial MPC Cavity Pressure (~~89.74~~ *107.4* psia)
T_1 = Initial MPC Cavity Average Temperature (~~513.6~~ *549.2*°K)
ΔT = Bounding MPC Temperature Rise (~~25~~ *36* °F or ~~19~~ ~~.9~~ *20*°K)

[†] *The temperature rise is conservatively calculated by prorating the previously reported 25 °F temperature rise for the 28.74 KW decay heat load to the current design basis of 41.22 KW.*

Substituting these values into the equation above, the bounding MPC internal pressure is obtained as:

$$\cancel{P_2 = 89.7 \times \frac{513.6 + 13.9}{513.6} = 92.1 psia = 77.4 psig}$$

$$P_2 = (107.4)(549.2+20)/(549.2) = 111.3 \text{ psia or } 96.6 \text{ psig}$$

The off-normal MPC internal design pressure ~~of 100 psig~~*as reported in* ~~(~~Table 2.2.1~~)~~ has been established to bound this partial inlet duct blockage condition.

~~Although it is a beyond the design basis condition, the bounding pressure rise for the three blocked air inlet ducts condition can be determine in the same manner. As stated above, the bounding temperature rise for this condition is 81°F (44.9°K), and the corresponding bounding MPC internal pressure is 97.5 psia (82.8 psig). This parametric evaluation demonstrates the insensitivity of the MPC internal pressure to incremental duct blockage, as the relatively large incremental flow area reduction increases the pressure by only 5.4 psi.~~

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

Using the methodology and model discussed in Section 4.4, the thermal analysis for the two air inlet ducts blocked off-normal condition is performed. The analysis demonstrates that under steady-state conditions, no system components exceed the short-term allowable temperatures in Table 2.2.3.

~~The parametric study of incremental duct blockage, performed by evaluating a three air inlet ducts blocked condition, demonstrates the insensitivity of the system to relatively large incremental flow area reductions. This beyond the design basis condition results in relatively small temperature increases and temperatures well below the short-term allowable temperatures in Table 2.2.3, even though no such requirement exists.~~

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.

Inspection of the HI-STORM overpack air duct screen covers is required with the frequency specified by Technical Specifications in Appendix A to the CoC or, 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 Subsection 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 upending and/or downending of the HI-TRAC transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation.

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

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 Appendices 3.AA and 3.AI of this FSAR. 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 Appendices 3.AA and 3.AI. 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 Off-Normal Load Combinations

Load combinations for off-normal conditions are provided in Table 2.2.14. The load combinations include normal loads with the off-normal loads. The load combination results are shown in Section 3.4 to meet all allowable values.

Table 11.1.1

MAXIMUM TEMPERATURES CAUSED BY OFF-NORMAL ENVIRONMENTAL TEMPERATURES

Location	**Temperature [°F]**	**Design Basis Limits [°F]**
Fuel Cladding	711 (PWR) 760 (BWR)	1058 short-term
MPC Basket	740	950 short-term
MPC Shell	~~371~~ *436*	775 short-term
Overpack Air Outlet	~~226~~ *261*	N/A
Overpack Inner Shell	~~219~~ *260*	350 short-term (overpack concrete)
Overpack Outer Shell	~~165~~ *179*	350 short-term (overpack concrete)

Table 11.1.2

BOUNDING† TEMPERATURES CAUSED BY PARTIAL BLOCKAGE OF AIR INLET DUCTS [°F]

Temperature Location	**No Blockage of Inlet Ducts**	**Partial Blockage of Inlet Ducts**		**Off-Normal Design Basis**
		2 Ducts Blocked	**~~3 Ducts Blocked~~**	
Fuel Cladding	740	~~765~~ *776*	~~821~~	1058 short-term
MPC Basket	720	~~745~~ *756*	~~801~~	950 short-term
MPC Shell	~~351~~ *416*	~~376~~ *452*	~~432~~	775 short-term
Overpack Air Outlet	~~206~~ *241*	~~231~~ *277*	~~287~~	N/A
Overpack Inner Shell	~~199~~ *240*	~~224~~ *276*	~~280~~	350 short-term (overpack concrete)
Overpack Outer Shell	~~145~~ *159*	~~170~~ *185*	~~226~~	350 short-term (overpack concrete)

† The bounding temperatures presented in this table are obtained by adding the maximum temperature rise of any cask component to the normal condition temperatures of every cask component.

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. Section 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 can be transported to the ISFSI in the vertical or horizontal position. The loaded HI-TRAC transfer cask is typically transported by a heavy-haul vehicle that cradles the HI-TRAC horizontally or by a device with redundant drop protection that holds the HI-TRAC vertically. The height of the loaded overpack above the ground shall be limited to below the horizontal handling height limit determined in Chapter 3 and specified by the Technical Specifications in Appendix A to the CoC 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 a credible accident. 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 2.3.3.1 as required by the Technical Specification.

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-STORM 100 System 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 are

demonstrated to remain attached by withstanding the maximum deceleration. The transfer lid doors 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, and transfer lid doors 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 slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

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 Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded. As the structural analysis demonstrates that the HI-TRAC top lid, transfer lid, and transfer lid doors 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 rate at 1 meter from the water jacket after the water is lost is calculated in Table 5.1.10. Immediately after the drop accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public. Based on a minimum distance to the controlled area boundary of 100 meters, the dose rate at the controlled area boundary will be approximately 1.47 mrem/hr (Section 5.1.2). Therefore, it is evident, based on the short duration of the accident, that the requirements of 10CFR72.106 (5 Rem) will not be 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 Section 5.1.2 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 simply 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, less personnel exposure may be received 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.

As a result of the loss of water from the water jacket, the dose rates at 1 meter adjacent to the water jacket mid-height increase (Table 5.1.10). Increasing the personnel exposure for each task affected by the increased dose rate adjacent to the water jacket by the ratio of the one meter dose rate increase results in a cumulative dose of less than 5.0 person-rem, for the 125-ton HI-TRAC or 100-ton HI-TRAC. Using the ratio of the water jacket mid-height dose rates at one meter is very conservative. Dose rate at the top and bottom of the HI-TRAC water jacket would not increase as much as the peak mid-height dose rates. In the determination of the personnel exposure, dose rates at the top and bottom of the loaded HI-TRAC are assumed to remain constant.

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 and specified by the Technical Specifications in Appendix A to the CoC. 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.

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 TSAR and SAR [11.2.6 and 11.2.7] 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 change in 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 change in 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 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 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 low handling height limit. 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 11.2.14. 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 limited to a localized decrease in the shielding thickness of the concrete.

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 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 accident pressure does not affect the safe operation of the HI-STORM 100 System.

11.2.3.3 Tip-Over Dose Calculations

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, because the affected areas will be small and localized. 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.

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 will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC shall 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 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 due to the lack of combustible materials at the ISFSI, 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 based on a Technical Specification in Appendix A to the CoC.

With respect to fire accident thermal analysis, NUREG-1536 (4.0,V,5.b) states:

> "Fire parameters included in 10 CFR 71.73 have been accepted for characterizing the heat transfer during the in-storage fire. However, a bounding analysis that limits the fuel source thus limits the length of the fire (e.g., by limiting the source of the fuel in the transporter) has also been accepted."

Based on this NUREG-1536 guidance, the fire accident thermal analysis is performed using the 10 CFR 71.73 parameters and the fire duration is determined from the limited fuel volume of 50 gallons. The entire transient evaluation of the storage fire accident consists of three parts: (1) a bounding steady-state initial condition, (2) the short-duration fire event, and (3) the post-fire temperature relaxation period.

As stated above, the fire parameters from 10 CFR 71.73 are applied to the HI-STORM fire accident evaluation. 10 CFR 71 requirements for thermal evaluation of hypothetical accident conditions specifically define pre- and post-fire ambient conditions, specifically:

> "the ambient air temperature before and after the test must remain constant at that value between -29°C (-20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration."

The ambient air temperature is therefore set to 100°F both before (bounding steady state) and after (post-fire temperature relaxation period) the short-duration fire event.

During the short-duration fire event, the following parameters from 10CFR71.71(c)(4) are applied:

1. Except for a simple support system, the cask must be fully engulfed. The ISFSI pad is a simple support system, so the fire environment is not applied to the overpack baseplate. By fully engulfing the overpack, additional heat transfer surface area is conservatively exposed to the elevated fire temperatures.

2. 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.

3. The average flame temperature must be at least 800°C (1475°F). 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 temperature.

4. 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.

5. 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 [11.2.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft^2×°F) is applied to exposed overpack surfaces during the short-duration fire.

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.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width, the fuel ring surrounding the overpack covers 147.6 ft^2 and has a depth of 0.54 in. From this depth and a linear fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.622 minutes (217 seconds). The linear fuel consumption rate of 0.15 in/min is the smallest value given in a Sandia Report on large pool fire thermal testing [11.2.2]. Use of the minimum linear consumption rate conservatively maximizes the duration of the fire.

It is recognized that the ventilation air in contact with the inner surface of the HI-STORM overpack with design-basis decay heat under maximum normal ambient temperature conditions varies between 80°F at the bottom and ~~206~~ *241*°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy only 1.25% 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 substantially elevated 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:

$$q_F = h_{fc}(T_A - T_S) + 0.1714 \times 10^8 \varepsilon [(T_A + 460)^4 - (T_S + 460)^4]$$

where:

q_F =Surface Heat Input Flux (Btu/ft^2-hr)
h_{fc} = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft^2-hr-°F)
T_A = Fire Condition Temperature (1475°F)
T_S = Transient Surface Temperature (°F)
ε = 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 [11.2.2].

After the fire event, the ambient temperature is restored to 100°F 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:

$$q_S = h_S(T_S - T_A) + 0.1714 \times 10^8 \varepsilon [(T_S + 460)^4 - (T_A + 460)^4]$$

where:

q_S =Surface Heat Loss Flux (Btu/ft^2-hr)
h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_S = Transient Surface Temperature (°F)
T_A = Ambient Temperature (°F)
ε = Surface Emissivity

In the post-fire temperature relaxation phase, the surface heat transfer coefficient (h_S) is determined by the following equation:

$$h_S = 0.19 \times (T_A - T_S)^{1/3}$$

where:

h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_A = External Air Temperature (°F)
T_S = Transient Surface Temperature (°F)

As discussed in Subsection 4.5.1.1.2, this equation is appropriate for turbulent natural convection from vertical surfaces. For the same conservative value of the Z parameter assumed earlier (2.6×10^5)

and the HI-STORM overpack height of approximately 19 feet, the surface-to-ambient temperature difference required to ensure turbulence is 0.56 °F.

A two-dimensional, axisymmetric model was developed for this analysis. Material thermal properties used were taken from Section 4.2. An element plot of the 2-D axisymmetric ANSYS model is shown in Figure 11.2.1. 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.

To maximize the rate of heat input from the fire to the HI-STORM overpack, the overpack temperature at the start of fire is understated and the overpack inside surface instantaneously elevated to a high temperature (300°F). This maximizes the fire-to-overpack temperature differential and therefore the heat input to the overpack is overstated. As a result, the overpack interior temperature rise is exaggerated in the ANSYS model. ~~Based on the results of the analysis, the maximum temperature increases at several points near the overpack mid-height are summarized in Table 11.2.2 along with the corresponding peak temperatures.~~ Temperature profiles through the storage overpack wall thickness near the mid-height of the cask are included in Figures 11.2.2 through 11.2.4. A plot of temperature versus time is shown in Figure 11.2.5 for several points through the overpack wall, near the mid-height of the cask. The temperature profile plots (Figures 11.2.2 through 11.2.4) each contain profiles corresponding to time "snapshots". Profiles are presented at the following times: 1 minute (60 seconds), 2 minutes (120 seconds), 3.622 minutes (217 seconds – end of fire), 10 minutes (600 seconds), 20 minutes (1200 seconds), 40 minutes and 90 minutes. *Based on the results of the analysis, the maximum temperature at several overpack locations are summarized in Table 11.2.2 along with peak fuel temperatures.*
The primary shielding material in the storage overpack is concrete, which can suffer a reduction in neutron shielding capability at sustained high temperatures due to a loss of water. As shown in Figure 11.2.5, less than 1 inch of the concrete near the outer overpack surface exceeds the material short-term temperature limit. This condition is addressed specifically in NUREG-1536 (4.0,V,5.b), which states:

> "The NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire."

These results demonstrate that the fire accident event does not substantially affect the HI-STORM overpack. Only localized regions of concrete are exposed to temperatures in excess of the allowable short-term temperature limit. No portions of the steel structure exceed the allowable temperature limits.

Having evaluated the effects of the fire on the overpack, we must 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)
ρ = Overpack Density (lb/ft^3)
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^3) 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:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 127.7 hrs$$

One-tenth of this time constant is approximately 12.8 hours (766 minutes), substantially longer than the fire duration of 3.622 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The temperature of the MPC is therefore increased by the contained decay heat only.

Table 4.5.5 lists lower-bound thermal inertia values for the MPC and the contained fuel assemblies of 4680 Btu/°F and 2240 Btu/°F, respectively. Applying an upper-bound decay heat load of ~~28.74~~ *41.22*-kW (~~98,090~~- *140,685* Btu/hr) for the 3.622 minute (0.0604 hours) fire duration results in the contained fuel assemblies heating up by only:

$$\Delta T_{fuel} = \frac{140685 \times 0.0604}{4680 + 2240} = 1.24°F$$

This is a negligible increase in the fuel temperature. Consequently, the impact on the MPC internal helium pressure will be negligible as well. 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 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 design temperature limits during post-fire temperature relaxation is not compromised.

Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

Thermal

As discussed above, the MPC internal pressure increases a negligible amount and is bounded by the 100% fuel rod rupture accident in Section 11.2.9. As shown in Table 11.2.2, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.

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." Less than one-inch of the concrete (less than 4% of the total overpack radial concrete section) exceeds the short-term temperature limit.

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 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 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. 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. All of these parameters are conservatively bounded by the values in Table 11.2.3, which are used for the fire transient analysis.

Using the values stated in Table 11.2.3, a bounding cask temperature rise of 5.509°F per minute is determined from the combined radiant and forced convection fire and decay heat inputs to the cask *at a reference heat load of 28.74 kw. Prorating the rate of temperature rise by the ratio of design maximum heat load (41.22 kw) and reference heat load(28.74 kw), a conservative upperbound to the rate of temperature rise (R = 7.9°F/min) is established.* During the handling of the HI-TRAC transfer cask, the transporter is limited to a maximum of 50 gallons, in accordance with a Technical Specification in Appendix A to the CoC. The duration of the 50-gallon fire *(tf)* is 4.775 minutes. Therefore, *the temperature rise (ΔT_f) computed as the product of R and tf is 37.7°F. Because the cladding temperature at the start of fire is below the short term temperature limit in excess of 150°F,* the fuel cladding *temperature limit is not exceeded (see Table 11.2.5).* ~~will not exceed the short-term fuel cladding temperature limit (see Table 11.2.5).~~

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 valve 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.

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. *Employing a conservatively bounding temperature rise (ΔT_f = 43.2°F) and an initial steady state condition for the hottest canister (MPC-68, design maximum heat load (41.22 kw) and 100°F ambient), the accident pressure is computed to be 112.7 psig which is substantially below the accident design limit [Table 2.2.1].* ~~Table 11.2.4 provides the MPC maximum internal pressures as a result of the HI-TRAC fire accident. The values presented in Table 11.2.4 are determined using a bounding temperature rise of 43.2°F, instead of the calculated 26.3°F temperature rise, and are therefore conservative.~~ Table 11.2.5 provides a summary of the loaded HI-TRAC bounding maximum temperatures for the hypothetical fire accident condition.

Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

Thermal

As discussed above, the MPC internal pressure increases as a result of the fire accident, but the internal pressure, ~~conservatively including a non-mechanistic 100% fuel rod rupture, is shown in Table 11.2.4 to be~~ *is substantially* less than the ~~accident condition~~ MPC internal *accident* design pressure ~~of 200 psig (~~Table 2.2.1~~)~~. As shown in Table 11.2.5, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.

The loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8 based on an assumed start at normal on-site transport conditions. As can be seen from the values in the table, the temperatures ~~increase by less than 20°F. Therefore, if the temperatures presented in Table 11.2.5 were increased by 20°F to account for the decrease in conductivity of the water jacket, the resultant temperatures will still be~~ *are* well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

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 analysis results presented in Section 5.1.2 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, 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. There are increases in the local dose rates adjacent water jacket. HI-TRAC dose rates at 1 meter and 100

meters from the water jacket, after the water is lost, have already been reported in Subsection 11.2.1.2. Immediately after the fire accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

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 elevated temperatures experienced by the HI-STORM overpack concrete shield is limited to the outermost layer. Therefore, any corresponding reduction in neutron shielding capabilities is limited to the outermost layer. The slight increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate, due to the limited amount of the concrete shielding with reduced effectiveness and the negligible neutron dose rate calculated for normal conditions at the site boundary. 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 pressure relief valves may have opened and water loss from the water jacket may have occurred resulting in an 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 by adding water at pressure. 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 only the outermost layer of the radial concrete exceeds its design temperature. The HI-STORM overpack

may be returned to service if there is no 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, vacuum 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 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 [11.2.5], 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 (see Table 1.2.1), 90% of the maximum crud volume was used to determine the crud depth. The maximum crud depths calculated for each of the MPCs is 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 totally block any

of 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 2.3.3.1. 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. 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 loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

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 Section 5.1.2 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. There are increases in the local dose rates adjacent water jacket as a result of the loss of water in the HI-TRAC water jacket. HI-TRAC dose rates at 1 meter and 100 meters from the water jacket, after the water is lost, have already been reported in Subsection 11.2.1.2. Immediately after the tornado accident a radiological inspection of the HI-TRAC will 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

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.

Structural

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 actually provide cooling that exceeds that available in the air filled annulus, due to water's higher thermal conductivity, density and heat capacity, and the forced convection coefficient associated with flowing water. Since the flood water temperature will be within the off-normal temperature range specified in Table 2.2.2, the thermal transient associated with the initial contact of the floodwater will be bounded by the off-normal operation conditions.

For a smaller flood that blocks the air inlet ducts but is not sufficient to allow water to come into contact with the MPC, a thermal analysis is included in Subsection 11.2.13 of this FSAR.

Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water acts as a radiation shield and will reduce the 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, the exterior and interior of the overpack, and the exterior of the MPC shall be cleaned to maintain the proper air flow and emissivity.

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 Section 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 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 4.4.14. As can be seen from the values, the ~~200 psig~~ design basis accident condition MPC internal pressure 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 confinement boundary leakage accident assumes simultaneous rupture of 100% of the fuel rods and the release of the available radioactive gas inventory to the environment at a rate based on 150% of the maximum leak rate under reference conditions.

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. The release is analyzed to demonstrate the safety of the HI-STORM 100 System.

11.2.10.2 Confinement Boundary Leakage Analysis

The following is the basis for the conservative analysis of the confinement boundary leakage accident.

1. All the fuel stored in the MPC has been cooled for ~~5~~*3* years. The PWR fuel type is the B&W 15×15 at ~~4.8~~*5.0* % enrichment with a burnup of ~~70~~*75*,000 MWD/MTU. The BWR fuel type is the GE 7×7 at 4.*8*4% enrichment with a burnup of ~~60~~*70*,000 MWD/MTU. These fuel characteristics bound the design basis fuel for the HI-STORM 100 System.

2. One hundred percent of all the fuel rods are assumed to rupture.

3. The releasable source term and release fractions are in accordance with NUREG-1536, ISG-5 and ISG-11.

4. *Credit is taken for the gravitational settling of fines, volatiles and cruds.*

54. The maximum possible leakage rate of radionuclides to the environment is based on the helium leak rate under reference test conditions from the Technical Specification in Appendix A to the CoC.

Chapter 7 presents an evaluation of the consequences of a non-mechanistic postulated ground-level breach of the MPC confinement boundary under hypothetical accident conditions of storage. The resulting Total Effective Dose Equivalent (TEDE) and other dose equivalents at a downstream distance of 100 meters are evaluated for each MPC type.

Structural

There are no structural consequences of the loss of confinement accident.

Thermal

Since this event is a non-mechanistic assumption, there are no realistic thermal consequences. As discussed in the Technical Specifications in Appendix A to the CoC, the leak test rate would result in a negligible loss of helium fill gas over the design life of the MPC, which would have an inconsequential effect on thermal performance.

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

This event is based upon an assumed instantaneous breach of the confinement.

Radiation Protection

The postulated release will result in an increase in dose to the public. The analysis of this event is provided in Section 7.3. As shown therein, the postulated breach results in dose rates to the public less than the limit established by 10CFR72.106(b) for the site boundary.

11.2.10.3 Confinement Boundary Leakage Dose Calculations

10CFR72.106 requires that any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 Rem to the whole body or any organ from any design basis accident. The maximum whole body dose contribution as a result of the instantaneous leak accident is calculated in Chapter 7 (Table 7.3.8). The maximum doses as a result of the confinement boundary leak accident is calculated in Chapter 7 (Table 7.3.8). Both values are well below the regulatory limit of 5 Rem.

11.2.10.4 Confinement Boundary Leakage Accident Corrective Action

A detected breached MPC will need to be repaired or the fuel removed and placed into a new MPC. First, the breached MPC must be returned to the facility in accordance with the procedures provided in Chapter 8. If the leak can be detected and repaired, and testing can be performed to verify the integrity of the confinement boundary, the MPC may be placed back into service. Otherwise, the MPC should be unloaded in accordance with the procedures provided in Chapter 8.

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. The fuel available for the explosion would be limited and therefore, any explosion would be limited in size. Any explosion stipulated to occur beyond the site boundary would have a minimal effect on the HI-STORM 100 System.

11.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external pressure of 60 psig (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 will not be stored within close proximity to the casks. The HI-STORM Overpack does not experience the 60 psi external pressure since it is not a sealed vessel. However, a pressure differential of 10.0 psi (Table 2.2.1) is applied to the overpack. Section 3.4 provides the analysis of the accident external pressure on the MPC and overpack. The analysis shows that the MPC can withstand the effects of the accident condition external pressure, while conservatively neglecting the MPC internal pressure.

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.

The lightning accident shall have no adverse consequences on thermal, criticality, confinement, shielding, or structural performance of the HI-STORM 100 System.

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 (in excess of 170,000 lbs), it is expected that a significant temperature rise is only possible if the completely 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. Thus, the worst possible scenario is a complete loss of ventilation air during the scheduled surveillance time interval in effect at the ISFSI site.

It is noted that there is a large thermal margin, between the maximum calculated fuel cladding temperature with design-basis fuel decay heat (Tables 4.4.9, 4.4.10, 4.4.46 and 4.4.27) and the short-

term fuel cladding temperature limit (1058°F), to meet the transient short-term fuel cladding temperature excursion. In other words, the fuel stored in a HI-STORM system can heat up by over 300°F before the short-term peak temperature limit is reached. The concrete in the overpack and the MPC and overpack structural members also have significant, margins between their calculated maximum long-term temperatures and their short-term temperature limits, with which to withstand such extreme hypothetical events.

To rigorously evaluate the minimum time available before the short-term temperature limits of either the concrete, structural members or fuel cladding are exceeded, a transient thermal model of the HI-STORM System is developed. The HI-STORM system transient model with all four air inlet ducts completely blocked is created as an axisymmetric finite-volume (FLUENT) model. With the exceptions of the inlet air duct blockage and the specification of thermal inertia properties (i.e., density and heat capacity), the model is identical to the steady-state models discussed in Chapter 4 of this FSAR. ~~The model includes the lowest MPC thermal inertia of any MPC design.~~ *For conservatism, the MPC basket and fuel thermal inertia is ignored.*

In the ~~first step of the~~ transient solution, the decay heat load is set equal to ~~22.25~~ *41.22* kW and the MPC internal convection (i.e., thermosiphon) is ~~suppressed~~ *included*. This evaluation provides the peak temperatures of the fuel cladding, the MPC confinement boundary and the concrete overpack shield wall, all as a function of time. *The hottest canister (MPC-68) is employed in the blocked ducts evaluation.* ~~Because the MPC with the lowest thermal inertia is used in the analysis, the temperature rise results obtained from evaluation of this transient model, therefore, bound the temperature rises for all MPC designs (Table 1.2.1) under this postulated event.~~

The results of the blocked duct thermal transient evaluation are ~~presented in Figure 11.2.7 and~~ *provided in* Table *11.2.10* ~~11.2.9~~. ~~Figure 11.2.7 presents the temperature rise as a function of time after complete air inlet duct blockage for the following:~~

~~i. Fuel Cladding at the Location of Initial Maximum Temperature~~
~~ii. MPC Shell at the Location of Initial Maximum Temperature~~
~~iii. Overpack Inner Concrete at the Active Fuel Axial Mid-Height~~
~~iv. Overpack Inner Concrete at the Location of Initial Maximum Temperature~~
~~v. Overpack Outer Concrete at the Active Fuel Axial Mid-Height~~
~~vi. Overpack Outer Concrete at the Location of Initial Maximum Temperature~~

The concrete section average (i.e., through thickness) temperature remains *substantially* below the short-term temperature limit through ~~72~~ *33* hours of blockage. Both the fuel cladding and the MPC confinement boundary temperatures remain below their respective short-term temperature limits. ~~at 72 hours, the fuel cladding by over 150°F and the confinement boundary by almost 175°F.~~ Table ~~11.2.9~~ *11.2.10* summarizes the *maximum* temperatures ~~at several points in the HI-STORM System at 33 hours and 72 hours after complete inlet air duct blockage~~. These results establish the design-basis minimum surveillance interval (i.e., 24 hours per Technical Specifications in Appendix A to the CoC) for the duct screens.

~~Incorporation of the MPC thermosiphon internal natural convection, as described in Chapter 4, enables the maximum design basis decay heat load to rise to about 29 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak MPC closure plate outer surface temperature, for example, is computed to be about 450°F in the thermosiphon enabled solution compared to about 210°F in the thermosiphon suppressed solution, with both solutions computing approximately the same peak clad temperature. In the 100% inlet duct blockage condition, the heated MPC closure plate and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and by virtue of the fact that thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limit for reaching the short-term peak fuel cladding temperature limit (72 hours) remains applicable.~~

~~It should be noted that the rupture of 100% of the fuel rods and the subsequent release of the contained rod gases has a significant positive impact on the MPC internal thermosiphon heat transport mechanism. The increase in the MPC internal pressure accelerates the thermosiphon, as does the introduction of higher molecular weight gaseous fission products. The values reported in Table 11.2.9 do not reflect this improved heat transfer and will actually be lower than reported. Crediting the increased MPC internal pressure only and neglecting the higher molecular weights of the gaseous fission products, the MPC bulk average gas temperature will be reduced by approximately 34.5°C (62.1°F).~~

Under the complete air inlet ducts blockage accident condition, it must be demonstrated that the MPC internal pressure does not exceed its design-basis accident limit during this event. Chapter 4 presented the MPC internal pressure calculated at an ambient temperature of 80°F, ~~100% fuel rods ruptured,~~ full insolation, and maximum decay heat. This calculated pressure is ~~174.8~~ 107.4 psia, as reported in Table 4.4.14, at an average temperature of ~~513.6~~ *549.2* °K. Using this pressure, a bounding increase in the MPC cavity temperature of ~~184~~ *329* °F (~~102.2~~ *182.8* °K, maximum of MPC shell or fuel cladding temperature rise 33 hours after blockage of all four ducts, see Table 11.2.*10* ~~9~~), ~~the reduction in the bulk average gas temperature of 34.5°C~~, and the ideal gas law, the resultant MPC internal pressure is calculated below.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(107.4\,\text{psi}\,a)\,(549.2°\,\text{K} + 182.8°\,\text{K})}{549.2°\,\text{K}}$$

$$P_2 = 143.1\,\text{psia or } 128.4\,\text{psig}$$

The accident MPC internal design pressure, ~~of 200 psig (~~ *as reported in* Table 2.2.1, bounds the resultant pressure calculated above. Therefore, no additional analysis is required.

Structural

There are no structural consequences as a result of this event.

Thermal

Thermal analysis is performed to determine the time until the concrete section average and peak fuel cladding temperatures approach their short-term temperature limits. At the specified time limit, both the concrete section average and peak fuel cladding temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is calculated as presented above. As can be seen from the value above, the ~~200 psig~~ design basis internal pressure for accident conditions used in the structural evaluation bounds the calculated value above.

~~To demonstrate the robustness of the HI-STORM System design, the results of the parametric study of incremental duct blockage performed in Subsection 11.1.4 are examined again. Even with three air inlet ducts completely blocked, as shown in Table 11.1.2, large steady-state margins against the short-term temperature limits exist for all system components and the fuel cladding of the stored assemblies. Both the peak fuel cladding and overpack concrete section average temperatures, which approach their limiting temperatures under the 100% blockage condition, with a single open duct are approximately 240°F and 100°F, respectively, less than their respective short-term temperature limits. These results show that only a relatively small amount of the total air inlet duct area, on the order of 25% or less, must remain open to prevent exceeding system short-term temperature limits under steady-state conditions.~~

Shielding

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.

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. The Technical Specifications in Appendix A to the CoC specify the time interval to ensure that the blockage duration cannot exceed the time limit calculated herein.

11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets 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.

11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet ducts accident shows that the overpack concrete section average and fuel cladding peak temperatures remain substantially below their short-term temperature limits if the blockage is cleared within 72 hours. 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. Per the Technical Specifications in Appendix A to the CoC, visual inspection of the duct screens is specified on a frequency of 24 hours, or air outlet temperature monitoring is required. Therefore, an undetected blockage event could not exceed 24 hours.

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. A measured temperature difference between the ambient air and the exit air that exceeds the design-basis maximum air temperature rise, calculated in Section 4.4.2, will indicate blockage of the overpack air ducts.

For an accident event that completely blocks the inlet or outlet air ducts, a site-specific evaluation or analysis may be performed to demonstrate that adequate heat removal is available for the duration of the event. Adequate heat removal is defined as overpack concrete section average and fuel cladding temperatures remaining below their short term temperature limits. For those events where an evaluation or analysis is not performed or is not successful in showing that fuel cladding temperatures remain below the short term temperature limit, 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. Another example of supplemental cooling, for sufficiently low decay heat loads, would be to remove the overpack lid to increase free-surface natural convection.

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 of 100 meters 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.

To demonstrate the inherent safety of the HI-STORM 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 System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short term design fuel cladding temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, and the spent nuclear fuel decay heat generation.

As stated in Subsection 11.2.13.2, there is a margin of over 300°F between the maximum calculated fuel cladding temperature and the short-term fuel cladding temperature limit. If a highly conservative 150°F is postulated as the permissible fuel cladding temperature rise for the burial under debris scenario, then a curve representing the relationship between the time required and decay heat load can be constructed. This curve is shown in Figure 11.2.6. In this figure, plots of the burial period at different levels of heat generation in the MPC are shown based on a 150°F rise in fuel cladding temperature resulting from transient heating of the HI-STORM System. Using the values stated in Table 11.2.6, the allowable time before the cladding temperatures meet the short-term fuel cladding temperature limit can be determined using:

$$\Delta t = \frac{m \times c_p \times \Delta T}{Q}$$

where:

Δt = Allowable Burial Time (hrs)
m = Mass of HI-STORM System (lb)
c_p = Specific Heat Capacity (Btu/lb×°F)
ΔT = Permissible Fuel Cladding Temperature Rise (150°F)
Q = Total Decay Heat Load (Btu/hr)

The allowable burial time as a function of total decay heat load (Q) is presented in Figure 11.2.6.

The MPC cavity internal pressure *(P2)* under this accident scenario *is computed below by the Ideal Gas Law for the hottest canister (MPC-68) as:*

P1 = 107.4 psia (Initial MPC pressure under normal storage condition at design maximum heat load)
T1 = 549.2°K (average temperature of Helium in MPC cavity)
ΔT = 150°F (83.3°K) (maximum temperature rise)

P2 = P1(T1+ΔT)/T1*
=107.4(549.2+83.3)/549.2*
= 123.7 psia (109 psig)

The computed pressure is substantially below the accident design pressure limit (Table 2.2.1). Therefore confinement integrity is maintained under this postulated accident.
~~is bounded by the calculated internal pressure for the hypothetical 100% air inlets blockage previously evaluated in Subsection 11.2.13.2.~~

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

With the cladding temperature rise limited to 150ºF, the corresponding pressure rise, bounded by the calculations in Subsection 11.2.13.2,demonstrates large margins of safety for the MPC vessel structural integrity. Consequently, cladding integrity 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 (Figure 11.2.6). The 24-hour minimum duct inspection interval specified in the Technical Specification in Appendix A to the CoC 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 will not exceed the short term limit if the debris is removed within 45 hours. 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 environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 125°F environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Section 4.4. These temperatures are conservatively calculated at an environmental temperature of 80°F. The extreme environmental temperature is 125°F, which is an increase of 45°F. Conservatively bounding temperatures for all the MPC designs are obtained and reported in Table 11.2.7. As illustrated by the table, all the temperatures are well below the accident condition design basis temperatures. The extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STORM 100 System extreme environmental temperatures meet the design requirements.

Additionally, the extreme environmental temperature generates a pressure that is bounded by the pressure calculated for the complete inlet duct blockage condition because the duct blockage condition temperatures are much higher than the temperatures that result from the extreme environmental temperature. As shown in Subsection 11.2.13.2, the accident condition pressures are below the accident limit specified in Table 2.2.1.

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 Table 11.2.7. As can be seen from this table, 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.

Table 11.2.1

INTENTIONALLY DELETED

Table 11.2.2

HI-STORM 100 OVERPACK MAXIMUM TEMPERATURES[2]
AS A RESULT OF THE HYPOTHETICAL FIRE CONDITION

Material/Component	**Initial[†] Condition (°F)**	**During Fire (°F)**	**Post-Fire[††] Cooldown (°F)**
Fuel Cladding	~~691 (MPC-24)~~ ~~691 (MPC-24E)~~ ~~691 (MPC-32)~~ ~~740 (MPC-68)~~ *740*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~ *741*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~ *741*
MPC Fuel Basket	~~650 (MPC-24)~~ ~~650 (MPC-24E)~~ ~~660 (MPC-32)~~ ~~720 (MPC-68)~~ *720*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~ *721*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~ *721*
Overpack Inner Shell	~~195~~ *240*	300	~~195~~ *300*
Overpack Radial Concrete Inner Surface	~~195~~ *240*	~~281~~ *300*	~~282~~ *300*
Overpack Radial Concrete Mid-Surface	~~173~~ *199*	~~173~~ *199*	~~184~~ *210*
Overpack Radial Concrete Outer Surface	~~157~~ *159*	~~529~~ *531*	~~530~~ *532*
Overpack Outer Shell	~~157~~ *159*	~~570~~ *572*	~~570~~ *572*

2 Bounding temperatures reported for the hottest canister (MPC-68).

~~‡ Bounding 195°F uniform inner surface and 157°F uniform outer surface temperatures assumed.~~

†† Maximum temperature during post-fire cooldown.

Table 11.2.3

SUMMARY OF INPUTS FOR HI-TRAC FIRE ACCIDENT HEAT-UP

Minimum Weight of Loaded HI-TRAC with Pool Lid (lb)	180,436
Lower Heat Capacity of Carbon Steel (Btu/lbm·°R)	0.1
Heat Capacity UO_2 (Btu/lbm·°R)	0.056
Heat Capacity Lead (Btu/lbm·°R)	0.031
~~Maximum~~ *Reference* Decay Heat (kW)	28.74
Total Fuel Assembly Weight (lb)	40,320
Lead Weight (lb)	52,478
Water Weight (lb)	7,595

Table 11.2.4

[INTENTIONALLY DELETED]

~~BOUNDING HI-TRAC HYPOTHETICAL~~

~~FIRE CONDITION PRESSURES~~[‡]

~~Condition~~	~~Pressure (psig)~~			
	~~MPC-24~~	~~MPC-24E~~	~~MPC-32~~	~~MPC-68~~
~~Without Fuel Rod Rupture~~	~~79.8~~	~~79.8~~	~~79.8~~	~~79.8~~
~~With 100% Fuel Rod Rupture~~	~~158.9~~	~~159.3~~	~~191.1~~	~~126.6 (72.48-488)~~

† The reported pressures are based on temperatures that exceed the calculated maximum temperatures and are therefore slightly conservative.

Table 11.2.5

SUMMARY OF BOUNDING MPC PEAK TEMPERATURES
DURING A HYPOTHETICAL HI-TRAC FIRE ACCIDENT CONDITION

Location	**Initial Steady State Temperature [°F]**	**Bounding Temperature Rise [°F]**	**Hottest MPC Cross Section Peak Temperature [°F]**
Fuel Cladding	~~872~~ *871*	~~26.3~~ *43.2*	~~898.3~~ *914.2*
Basket Periphery	~~600~~ *675*	~~26.3~~ *43.2*	~~626.3~~ *718.2*
MPC Shell	~~455~~ *539*	~~26.3~~ *43.2*	~~481.3~~ *582.2*

Table 11.2.6

SUMMARY OF INPUTS FOR ADIABATIC CASK HEAT-UP

Minimum Weight of HI-STORM 100 System (lb) (overpack and MPC)	300,000
Lower Heat Capacity of Carbon Steel (BTU/lb/°F)	0.1
Initial Uniform Temperature of Cask (°F)	740†
Bounding Decay Heat (kW)	~~28.74~~ *41.22*

† The cask is conservatively assumed to be at a uniform temperature equal to the maximum fuel cladding temperature.

Table 11.2.7

MAXIMUM TEMPERATURES CAUSED BY EXTREME ENVIRONMENTAL TEMPERATURES† [°F]

Location	**Temperature**	**Accident Temperature Limit**
Fuel Cladding	736 (PWR) 785 (BWR)	1058
MPC Basket	765	950
MPC Shell	~~396~~ *461*	775
Overpack Air Exit	~~251~~ *286*	N/A
Overpack Inner Shell	~~244~~ *285*	350 (overpack concrete)
Overpack Outer Shell	~~190~~ *204*	350 (overpack concrete)

~~† Conservatively bounding temperatures reported include a hypothetical rupture of 10% of the fuel rods.~~

Table 11.2.8

MAXIMUM TEMPERATURES CAUSED BY LOSS OF WATER FROM THE HI-TRAC WATER JACKET [°F]

Temperature Location	**Normal**	**Calculated Without Water in Water Jacket**	**Accident Condition Design Temperature**
Fuel Cladding	~~872~~ *871*	~~888~~ *891*	1058 ~~short-term~~
MPC Basket	~~852~~ *851*	~~868~~ *871*	950 ~~short-term~~
MPC Basket Periphery	~~600~~ *675*	~~612~~ *695*	950 ~~short-term~~
MPC Shell	~~455~~ *539*	~~466~~ *561*	775 ~~short-term~~
HI-TRAC Inner Shell	~~322~~ *401*	~~342~~ *423*	~~400 long-term~~ 600 ~~short-term~~
HI-TRAC Water Jacket Inner Surface	~~314~~ *371*	~~334~~ *401*	~~350 long-term~~ *700*
HI-TRAC Enclosure Shell Outer Surface	~~224~~ *255*	~~222~~ *253*	~~350 long-term~~ *700*
Axial Neutron Shield†	~~258~~ *271*	~~261~~ *275*	~~300 long-term~~ *NA*

~~Note: Where it can be shown that the temperatures are below the normal long-term condition limits, the calculated temperatures are compared to the normal long-term temperature limits for conservatism. The corresponding short-term temperature limits are higher temperatures as presented in Table 2.2.3.~~

† Local maximum section temperature.

Table 11.2.9

[INTENTIONALLY DELETED]

~~SUMMARY OF BLOCKED AIR INLET DUCT EVALUATION RESULTS~~

	~~Max. Initial Steady-State Temp.~~[‡] ~~(°F)~~	~~Temperature Rise (°F)~~		~~Transient Temperature (°F)~~		~~Short-Term Temperature Limit (°F)~~
		~~at 33 hrs~~	~~at 72 hrs~~	~~at 33 hrs~~	~~at 72 hrs~~	
~~Fuel Cladding~~	~~740~~	~~101~~	~~160~~	~~841~~	~~900~~	~~1058~~
~~MPC Shell~~	~~351~~ ~~*416*~~	~~184~~	~~250~~	~~535~~	~~601~~	~~775~~
~~Overpack Inner Shell #1~~[‡‡]	~~199~~ ~~*240*~~	~~113~~	~~174~~	~~312~~	~~373~~	~~600~~
~~Overpack Inner Shell #2~~[‡‡‡]	~~155~~	~~193~~	~~286~~	~~348~~	~~441~~	~~600~~
~~Overpack Outer Shell~~	~~145~~ ~~*159*~~	~~14~~	~~40~~	~~159~~	~~185~~	~~600~~
~~Concrete Section Average~~	~~172~~	~~79~~	~~141~~	~~251~~	~~313~~	~~350~~

~~‡ Conservatively bounding temperatures reported includes a hypothetical rupture of 10% of the fuel rods.~~

~~‡‡ Coincident with location of initial maximum.~~

~~‡‡‡ Coincident with active fuel axial mid-height.~~

Table 11.2.10

BOUNDING MAXIMUM BLOCKED AIR INLET DUCTS TEMPERATURES

Item	*Initial Steady State Condition [°F]*	*After 33 hours of blockage [°F]*	*Accident Temperature Limit (°F)*
Fuel Cladding	*740*	1042	*1058*
MPC Shell	*416*	745	*775*
Overpack Inner Shell	*240*	*571*	*600*
Concrete Section Average	*180*	301	*350*

Figure 11.2.7
[INTENTIONALLY DELETED]

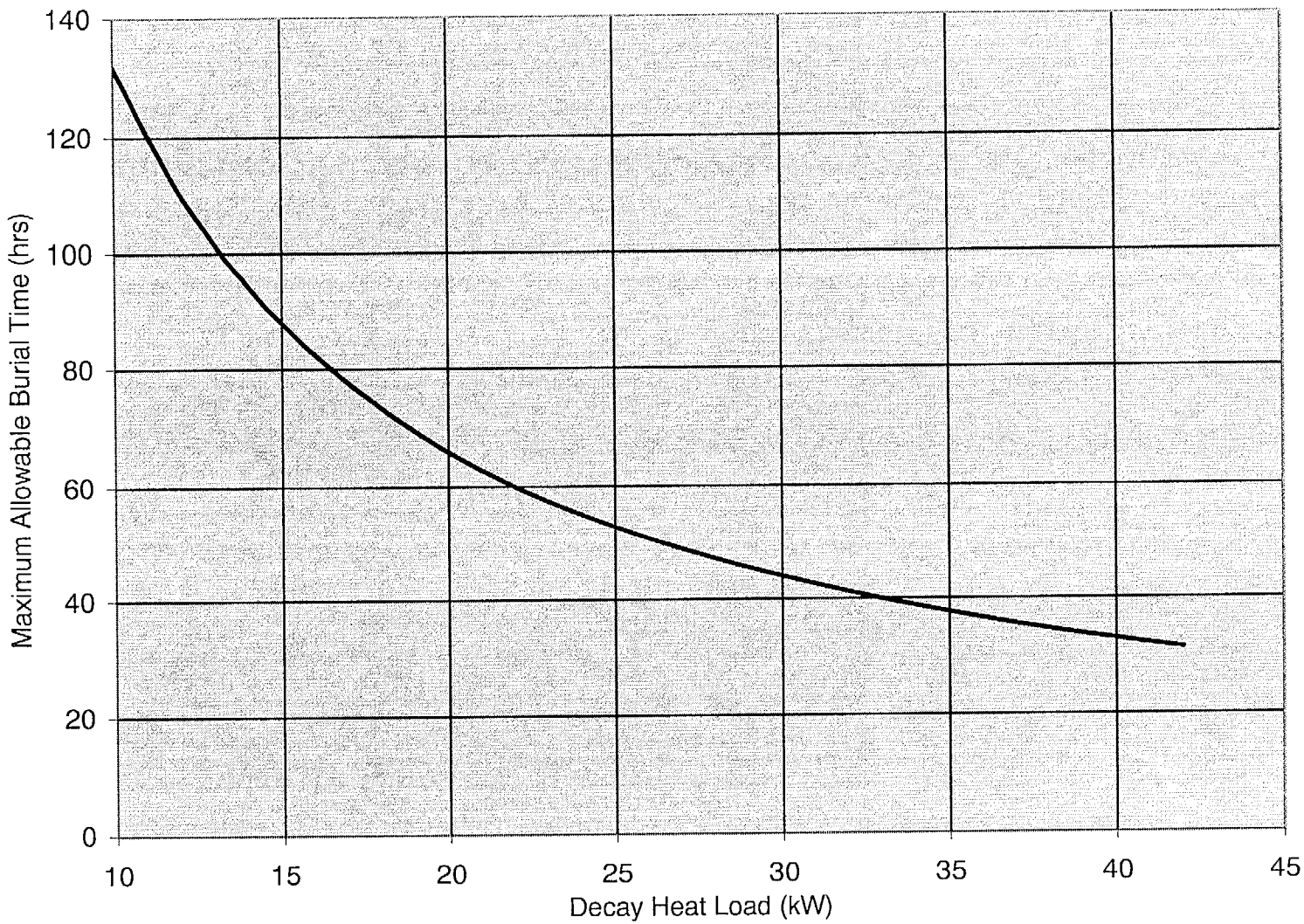

Figure 11.2.6: ALLOWABLE BURIAL UNDER DEBRIS TIME VERSUS DECAY HEAT LOAD.

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 to CoC 72-1014 and the authorized contents and design features provided in Appendix B 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**[1]
Criticality Control	Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features 3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) 5.6 Fuel Cladding Oxide Thickness Evaluation Program
Shielding and Radiological Protection	Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features 3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down ~~3.2.1 TRANSFER CASK Average Surface Dose Rates~~ ~~3.2.2 TRANSFER CASK Surface Contamination~~ ~~3.2.3 OVERPACK Average Surface Dose Rates~~ *5.7 Radiation Protection Program*
Heat Removal Capability	Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features 3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural Integrity	3.5 Cask Transfer Facility (CTF) (CoC 72-1014, Appendix B – Design Features) 5.5 Cask Transport Evaluation Program

[1] Technical Specifications are located in Appendix A to CoC 72-1014

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. Refer to Appendix B to CoC 72-1014 for fuel specifications.
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
~~3.2.1~~	~~TRANSFER CASK Average Surface Dose Rates~~
~~3.2.2~~	~~TRANSFER CASK Surface Contamination~~
~~3.2.3~~	~~OVERPACK Average Surface Dose Rates~~
3.3.1	Boron Concentration
Table 3-1	MPC Model-Dependent Limits
4.0	Not Used. Refer to Appendix B to CoC 72-1014 for design features.
5.0	ADMINSTRATIVE CONTROLS AND PROGRAMS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	Fuel Cladding Oxide Thickness Evaluation Program
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 for the HI-STORM 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR. In addition to the controls and limits provided in the Technical Specifications contained in Appendix A to Certificate of Compliance 72-1014 and the Approved Contents and Design Features in Appendix B to Certificate of Compliance 72-1014, the licensee shall ensure that the following training and dry run activities are performed.

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 (IFSI). 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 closure
- Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, and cooldown)
- MPC/HI-TRAC/overpack 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 for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- 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 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.

8. MPC welding, NDE inspections, hydrostatic testing, draining, moisture removal, helium backfilling and leakage testing (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).

12.2.3 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 in Appendix A to CoC 72-1014 and the Authorized Contents and Design Features in Appendix B to CoC 72-1014 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 100 Overpack

a HI-STORM 100 overpack 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.

b. HI-STORM 100 overpack material thermal properties and dimensions for heat transfer control.

c. HI-STORM 100 overpack material composition and dimensions for dose rate control.

12.2.10 Decay Heat and Burnup Limits for Fuel Storage

12.2.10.1 Uniform Fuel Storage

Section 2.4.1 of Appendix B to the HI-STORM 100 System CoC provides tabular limits for fuel assembly decay heat and burnup as a function of cooling time. Cask users must ensure that each fuel assembly to be loaded into the HI-STORM 100 System using the uniform loading option meets the limits provided in the tables contained in CoC Appendix B, Section 2.4.1. These burnup and decay heat limits as a function of cooling time are in addition to other restrictions in Appendix B to the CoC regarding fuel

selection, including fuel dimensions, enrichment, uranium mass, and the presence of non-fuel hardware.

12.2.10.2 Regionalized Fuel Storage

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Table 12.2.2 shows limitations on minimum and maximum allowable burnups as follows:

- *The calculated maximum allowable burnup must be greater than or equal to 20,000 MWD/MTU. If the calculated allowable maximum burnup is less than 20,000 MWD/MTU, then storage of fuel is not permitted at that cooling time. The next highest minimum cooling time yielding a maximum allowable burnup ≥ 20,000 MWD/MTU shall be used. A fuel assembly with an actual burnup less than 20,000 MWD/MTU may be stored, but it must have the longer cooling time.*

- *The maximum permitted actual burnup for any Zircaloy-2 or Zircaloy-4 clad PWR or BWR fuel assembly is 75,000 MWD/MTU and 70,000 MWD/MTU, respectively. Any calculated maximum allowable burnup above these values shall be reduced to these values. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the calculated value, whichever is less.*

Table 12.2.1

Example Allowable Decay Heat Values for Regionalized Storage in MPC-32
($q_{Region\ 2}$ = 0.750 kW)

MINIMUM COOLING TIME (yr)	*MAXIMUM DECAY HEAT PER FUEL ASSEMBLY IN REGION 1 ($q_{Region\ 1}$, kW)*
3	1.636
4	1.636
5	1.636
6	1.546
7	1.425
8	1.404
9	1.380
10	1.354
11	1.350
12	1.340
13	1.331
14	1.322
15	1.314

Table 12.2.2

Example Allowable Burnup Values for Regionalized Storage in MPC-32 (17x17A Fuel, $q_{Region\ 2}$ = 0.750 kW)

MINIMUM COOLING TIME (yr)	MAXIMUM BURNUP PER FUEL ASSEMBLY IN REGION 1 (MWD/MTU)	MAXIMUM BURNUP PER FUEL ASSEMBLY IN REGION 2 (MWD/MTU)
3	35,904	Not Permitted
4	50,681	24,360
5	61,859	31,630
6	66,972	36,865
7	68,274	40,547
8	74,464	43,328
9	75,000	45,531
10	75,000	47,231
11	75,000	48,836
12	75,000	50,136
13	75,000	51,459
14	75,000	56,343
15	75,000	53,654
16	NC	54,812
17	NC	55,854
18	NC	56,884
19	NC	57,987
20	NC	58,997

NOTE: "NC" means not calculated.

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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 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 dose rates are measured and 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 moisture removal is performed. The MPC cavity is backfilled with helium. Additional dose rates are measured and the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.

MPC cavity moisture removal using vacuum drying or forced helium recirculation is performed to remove residual moisture from the MPC fuel cavity 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 helium recirculation 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.

(continued)

BASES

BACKGROUND (continued)

After the completion of moisture removal, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Providing a helium pressure in the required range at room temperature (70°F), 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. Providing helium in the required density range accomplishes the same function.

In-leakage of air could be harmful to the fuel. Prior to moving the SFSC to the storage pad, the MPC helium leak rate is determined to ensure that the fuel is confined.

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.

(continued)

BASES (continued)

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.

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.

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 demoisturizer 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)

BASES

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 helium backfill density or *pressure* 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.

(continued)

BASES

ACTIONS (continued)

B.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. Since the quantity of helium estimated under Required Action B.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.

C.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 calculated in the FSAR confinement analyses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS (continued)

C.2

Once the cause and 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 off-site 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.

D.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, perform fuel cooldown operations, 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1, SR 3.1.1.2, and SR 3.1.1.3

SR 3.1.1.1 is modified by a note that states, in addition to the requirements of SR 3.1.1.1 for high burnup fuel, MPCs with heat loads in excess of a certain value shall be dried using the helium *recirculation method. The basis for this note is that, if vacuum drying were used for higher heat load MPCs, it would need to be completed in a relatively short period of time to avoid exceeding the short term peak fuel cladding temperature limit. Applying a time limit that is too restrictive could inhibit the ability to dry the MPC in a normal time frame. The helium recirculation method of moisture removal continuously cools the fuel while removing moisture, thereby eliminating the need to establish a time limit, allowing completion of the moisture removal process in a deliberate, controlled manner.*

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1, SR 3.1.1.2, and SR 3.1.1.3 (continued)

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 demoisturizer exit temperature reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. For high burnup fuel, the forced helium recirculation 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.

Having the proper helium backfill density or pressure ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC. Meeting the helium leak rate limit ensures there is adequate helium in the MPC for long term storage and the leak rate assumed in the confinement analyses remains bounding for off-site dose.

The leakage rate acceptance limit is specified in units of atm-cc/sec. This is a mass-like leakage rate as specified in ANSI N14.5 (1997). This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions. This allows the leakage rate as measured by a mass spectrometer leak detector (MSLD) to be compared directly to the acceptance limit without the need for unit conversion from test conditions to standard, or reference conditions.

All three 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.

REFERENCES

1. FSAR Sections 1.2, 4.4, 4.5 7.2, 7.3 and 8.1

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES

BACKGROUND

The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system which 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 four inlet air ducts at the bottom of the OVERPACK. 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 four outlet air ducts at the top of the OVERPACK.

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 four inlet and four outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.

Analyses have been performed for the complete obstruction of two, three, and four inlet air ducts. Blockage of two inlet air ducts 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.

Blockage of three inlet air ducts further reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this accident condition, no SFSC components exceed the short term temperature limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The complete blockage of all four inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler inner shell of the 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 72-hour duration of the analyzed event. Therefore, the limiting component is assumed to be the fuel cladding. O

LCO

The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. 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 within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris).

(continued)

BASES

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.

APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate heat transfer of the decay heat away from the fuel assemblies.

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

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

(continued)

BASES

ACTIONS
(continued)

B.1

If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, ~~Surveillance Requirement (SR) 3.2.3.1 is~~*dose rates are* required to be ~~performed~~*measured in accordance with the Radiation Protection Program* to determine the effectiveness of the radiation shielding provided by the concrete. This ~~SR~~ *Action* must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of whether the concrete is providing adequate 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.

B.2.1

In addition to Required Action B.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 B.2.2 is being implemented.

This Required Action must be complete in 48 hours. The Completion Time reflects a conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short term temperature limit for more than 72 hours. It is also unlikely that an unforseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.

(continued)

BASES

ACTIONS
(continued)

B.2.2

In lieu of implementing Required Action B.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.

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

(continued)

BASES

ACTIONS

B.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 Time of 48 hours reflects a conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short term temperature limit for more than 72 hours. It is also unlikely that an unforseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

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. There are two options for implementing SR 3.1.2.1, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all four inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Complete blockage of any one or more inlet or outlet air ducts renders the heat removal system inoperable and this LCO not met. Partial blockage of one or more inlet or outlet air ducts 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 duct(s).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet 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 limits 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 of 24 hours 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.

REFERENCES

1. FSAR Chapter 4
2. FSAR Sections 11.2.13 and 11.2.14
3. ANSI/ANS 57.9-1992

B 3.1 SFSC INTEGRITY

B 3.1.3 Fuel Cool-Down

BASES

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 *bulk helium temperature in the* MPC *cavity* is *ensured to be less than or equal to 200°F. This is accomplished via direct measurement of the MPC gas exit temperature or any other appropriate means based on a thermal evaluation of the particular MPC to be unloaded, considering its contents and the duration of time the MPC has been loaded. It is possible that thermal evaluation may determine that the bulk gas temperature is already within the LCO limit due to low decay contents and/or an extended time since loading, in which case, no additional action is required.* ~~attached to the Cool-Down System. The Cool-Down System is a closed-loop forced ventilation gas cooling system that cools the fuel assemblies by cooling the surrounding helium gas.~~

After ensuring the MPC cavity bulk helium temperature meets the LCO limit, ~~Following fuel cool-down,~~ the MPC is then re-flooded with water and 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.

Ensuring that ~~Reducing~~ the *bulk helium temperature is less than the LCO limit* ~~fuel cladding temperatures~~ significantly reduces the temperature gradients across the *fuel* cladding thus minimizing thermally-induced stresses on the cladding during MPC re-flooding. Reducing the MPC internal temperatures eliminates the risk of high MPC pressure due to sudden generation of *large* steam *quantities* during re-flooding. *The LCO limit of 200°F for bulk helium temperature eliminates the potential for gross steam generation during re-flooding.*

(continued)

BASES

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 minimizing thermally-induced stresses to the cladding.

This is accomplished during the unloading operations by lowering the MPC ~~internal~~ *cavity bulk helium* temperatures prior to MPC re-flooding. The *i*Integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by reducing the MPC internal temperatures such that there is no sudden formation of *large quantities of* steam during MPC re-flooding. (Ref. 1).

LCO

~~Monitoring~~ *Determining* the ~~circulating~~ MPC ~~gas exit~~*cavity bulk helium* temperature *prior to re-flooding* ensures that there will be no large thermal gradient across the fuel assembly cladding during re-flooding which could be potentially harmful to the cladding. The temperature limit specified in the LCO was selected to ensure that the MPC *cavity bulk helium temperature is sufficiently low to preclude high thermal stresses in the fuel cladding during* ~~gas exit temperature will closely match the desired fuel cladding temperature prior to~~ re-flooding *of* the MPC. The temperature was selected to be lower than the boiling temperature of water with an additional margin.

For the purposes of this LCO, "bulk helium temperature" is defined as the spatial average of the helium temperature in the MPC cavity. The bulk helium temperature will be between the peak and lowest fuel cladding temperature present in the basket.

(continued)

BASES

APPLICABILITY

The MPC *cavity bulk* helium ~~gas exit~~ temperature is measured 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 Fuel Cool-Down 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 Applicability 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 concerning for dry UNLOADING OPERATIONS.

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 MPC *cavity bulk* helium ~~gas exit~~ temperature limit is not met, actions must be taken to restore the parameters to within the limits before re-flooding the MPC. Failure to successfully complete fuel cool-down could have several causes, such as failure of the cool down system, inadequate cool down, or clogging of the piping lines. The Completion Time is sufficient to determine and correct most failure mechanisms and proceeding with activities to flood the MPC cavity with water are prohibited.

(continued)

BASES

ACTIONS (continued)

~~A.2~~

~~If the LCO is not met, in addition to performing Required Action A.1 to restore the gas temperature to within the limit, the user must ensure that the proper conditions exist for the transfer of heat from the MPC to the surrounding environs to ensure the fuel cladding remains below the short term temperature limit. If the TRANSFER CASK is located in a relatively open area such as a typical refuel floor, no additional actions are necessary. However, if the TRANSFER CASK is located in a structure such as a decontamination pit or fuel vault, additional actions may be necessary depending on the heat load of the stored fuel.~~

~~Three acceptable options for ensuring adequate heat transfer for a TRANSFER CASK located in a pit or vault are provided below, based on an MPC loaded with fuel assemblies with design basis heat load in every storage location. Users may develop other alternatives on a site-specific basis, considering actual fuel loading and decay heat generation.~~

~~1. Ensure the annulus between the MPC and the TRANSFER CASK is filled with water. This places the system in a heat removal configuration which is bounded by the FSAR thermal evaluation of the system considering a vacuum in the MPC. The system is open to the ambient environment which limits the temperature of the ultimate heat sink (the water in the annulus) and, therefore, the MPC shell to 212° F.~~

~~2. Remove the TRANSFER CASK from the pit or vault and place it in an open area such as the refuel floor with a reasonable amount of clearance around the cask and not near a significant source of heat.~~

(continued)

BASES

ACTIONS

~~A.2 (continued)~~

~~3. Supply nominally 1000 SCFM of ambient (or cooler) air to the space inside the vault at the bottom of the TRANSFER CASK to aid the convection heat transfer process. This quantity of air is sufficient to limit the temperature rise of the air in the cask-to-vault annulus to approximately 60° F at design basis maximum heat load while providing enhanced cooling of the cask by the forced flow.~~

~~Twenty-two (22) hours is an acceptable time frame to allow for completion of Required Action A.2 based on a thermal evaluation of a TRANSFER CASK located in a pit or vault. In such a configuration, passive cooling mechanisms will be largely diminished. Eliminating 90% of the passive cooling mechanisms with the cask emplaced in the vault, the thermal inertia of the cask (approximately 20,000 Btu/° F) will limit the rate of temperature rise with design basis maximum heat load to approximately 4.5 degrees F per hour. Thus, the fuel cladding temperature rise in 22 hours will be less than 100° F. Large short-term temperature margins exist to preclude any cladding integrity concerns under this temperature rise.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on the material condition of the fuel assembly cladding. By minimizing thermally-induced stresses across the cladding the integrity of the fuel assembly cladding is maintained. The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal temperature prior to re-flooding the MPC there is ~~no~~ *minimal* formation of steam during MPC re-flooding.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 (continued)

The MPC *cavity bulk* helium ~~exit gas~~ temperature limit ensures that there will be no large thermal gradients across the fuel assembly cladding during MPC re-flooding and ~~no~~*minimal* formation of steam which could potentially overpressurize the MPC.

The SR is met in one of two ways. The temperature of the gas exiting the MPC may be measured directly. Alternatively, a thermal evaluation may be performed, consistent with the methodology in the HI-STORM FSAR, to determine the MPC bulk helium temperature in the canister designated for unloading. This evaluation may consider the particular characteristics of the MPC, such as fuel cooling time, presence of NON-FUEL HARDWARE, and ambient conditions in determining the bulk helium temperature. If the MPC cavity bulk helium temperature LCO is shown to be met by this evaluation, no further actions are required and MPC unloading may proceed. If the LCO is shown not to be met by the thermal evaluation, appropriate means shall be used to cool the MPC cavity until the LCO is met (via direct measurement of the helium gas exit temperature or by an evaluation that includes the cooling process). When the LCO is met, unloading may proceed.

The LCO must be met ~~Fuel cool-down must be performed successfully~~ on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved.

REFERENCES

1. FSAR, Sections 4.4.1, 4.5.1.1.4, and 8.3.2.

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 dose rates are measured and 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 vacuum drying is performed. The MPC cavity is backfilled with helium. Additional dose rates are measured and 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 stored 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)

LCO

Compliance with this LCO ensures that the stored fuel will remain subcritical with a $k_{eff} \leq 0.95$ 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 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.~~c~~*f* provides the minimum ~~boron~~ 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.* ~~if one or more to fuel assemblies to be loaded had an initial enrichment less than or equal to 4.1 wt.% U-235. LCO 3.3.1.d provides the minimum boron concentration required in the MPC water for the MPC-32 if one or more to fuel assemblies to be loaded had an initial enrichment greater than 4.1 wt.% U-235.~~

All fuel assemblies loaded into the MPC-24, MPC-24E, MPC-24EF, ~~and~~ MPC-32*, and MPC-32F* are limited by analysis to maximum enrichments of 5.0 wt.% U-235.

(continued)

BASES

APPLICABILITY

The boron concentration LCO is applicable whenever an MPC-24, -24E, -24EF, ~~or~~ -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.

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

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.

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 of 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.

(continued)

BASES

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

REFERENCES

1. FSAR Chapter 6.

CHAPTER 13†: QUALITY ASSURANCE

13.0 ~~INTRODUCTION~~*QUALITY ASSURANCE PROGRAM*

13.0.1 Overview

This chapter provides a summary of the quality assurance program implemented for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the HI-STORM 100 System and HI-TRAC transfer cask designated as important to safety.

~~Table 2.2.6 identifies the structures, systems and components (SSCs) of the HI-STORM 100 System and HI-TRAC transfer cask that are considered important to safety. Table 8.1.6 identifies the ancillary equipment needed for handling and loading operations that has been designated as important to safety.~~

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 (References [13.0.2 and 13.0.4]). The Holtec Quality Assurance Program satisfies the requirements of 10 CFR 72, Subpart G. Pursuant to 10 CFR 72.140(d), this QA program, approved under Subpart H to 10 CFR 71, may be applied to spent fuel storage cask activities provided the additional recordkeeping requirements of 10 CFR 72.174 are also met. The Holtec QA program meets these additional recordkeeping requirements.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, as described 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.

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† 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~~ *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 (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

13.0.2 Graded Approach to Quality Assurance

For the HI-STORM 100 System, a graded approach to quality assurance is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Section 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 categories assigned to the cask components are identified in Table 2.2.6. Quality categories for ancillary equipment are provided in Table 8.1.6 on a generic basis. Quality categories for other equipment used to deploy the HI-STORM 100 System at a licensee's ISFSI are defined on a case-specific based on site-specific needs and the component's design function.

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 System on a site-specific independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may 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 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, this constitutes a deviation from NUREG-1536 and has been added to the list of deviations in Table 1.0.3. 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 due to maintenance to the two program descriptions.

13.1 ~~GRADED APPROACH TO QUALITY ASSURANCE~~

This section intentionally deleted.

~~For the HI-STORM 100 System and HI-TRAC transfer cask, a graded approach to quality is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents.~~

~~NUREG/CR-6407 [13.1.1] provides descriptions of quality categories A, B and C. These descriptions are provided below.~~

~~Category A:~~ ~~Category A items include structures, systems, and components whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.~~

~~Category B:~~ ~~Category B items include structures, systems, and components whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.~~

~~Category C:~~ ~~Category C items include structures, systems, and components whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.~~

~~Using these descriptions along with the quality category assignments from NUREG/CR-6407 [13.1.1], Holtec International has assigned a quality category to each individual component of the HI-STORM 100 System and HI-TRAC transfer cask. The categories are identified in Table 2.2.6.~~

~~Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 System on a site-specific independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may 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 which are important to safety. Regardless of the provisions of the procurement contract, the quality requirements set forth in this document constitute the minimum set of acceptable bases. Activities performed in the course of the previous and ongoing work effort on HI-STORM 100 comply with Holtec International's quality assurance program. Holtec International's QA program was developed to meet Nuclear Regulatory Commission (NRC) requirements delineated in 10CFR50, Appendix B, and has been expanded to include provisions of 10CFR71, Subpart H and 10CFR72, Subpart G, for structures, systems, and components designated as important to safety. A topical report [13.1.2] on the Holtec International QA program has been previously submitted to the~~

~~NRC. Quality Assurance Program Approval for Radioactive Packages No. 0784 was issued by the NRC. This quality assurance program also applies to the design, material procurement, fabrication, inspection, testing, handling, and repair of the HI-STORM 100 System, including the HI-TRAC transfer cask.~~

~~The quality assurance program described in this chapter fully complies with the requirements of 10CFR72 Subpart G, and NUREG-1536 [13.1.3].~~

13.6 REFERENCES

[13.~~1~~*0*.1] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[13.~~1~~*0*.2] Holtec International Quality Assurance Program, *Revision 13.* ~~Topical Report for 10CFR71, Subpart H and 10CFR72, Subpart G, Holtec International Report HI-941152, Rev. 2 (8/4/94)~~.

[13.~~1~~*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, Revision 3.

~~[13.3.1] NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities"~~

~~[13.3.2] U.S. Code of Federal Regulations, Title 10, "Energy", Part 21, "Reporting of Defects and Noncompliance."~~

U.S. Nuclear Regulatory Commission
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Document ID 5014452
Attachment 5

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Brian Gutherman, being duly sworn, depose and state as follows:

(1) I am Licensing Manager of Holtec International and have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is contained in enclosed with Holtec Letter No. 5014452, appropriately identified as confidential information:

- Certain proposed changes to CoC 72-1014, Appendix B, Section 2.4.2.
- Certain proposed changes to HI-STORM FSAR Section 4.4.1.1.9.
- Certain proposed changes to HI-STORM FSAR Section 12.2.10.

This information is considered proprietary to Holtec International.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

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(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have

been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information at this time would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10CFR2.790

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

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Attachment 5

AFFIDAVIT PURSUANT TO 10CFR2.790

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Mr. Brian Gutherman, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 28th day of February, 2002.

Brian Gutherman
Holtec International

Subscribed and sworn before me this 28th day of February, 2002.

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005