9.4 PROCEDURE FOR UNLOADING THE HI-STORM FW FUEL IN THE SPENT FUEL POOL

9.4.1 Overview of HI-STORM FW System Unloading Operations

ALARA Note:
The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM FW system unloading procedures describe the general actions necessary to prepare the MPC for unloading, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC VW and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. The principal operational steps are summarized below.

The MPC is recovered from HI-STORM FW either at the ISFSI or the fuel building using the same methods as described in Section 9.2 (in reverse order). The HI-STORM FW lid is removed and the mating device is positioned on the HI-STORM FW. MPC slings are attached to the MPC lift attachment and positioned on the MPC lid. HI-TRAC VW is positioned on top of HI-STORM FW and the slings are brought through the top of the HI-TRAC VW. The MPC is raised into HI-TRAC VW, the mating device drawer is closed, and the bottom lid is bolted to the HI-TRAC VW. The HI-TRAC VW is removed from on top of HI-STORM FW.

HI-TRAC VW and its enclosed MPC are returned to the designated preparation area and the MPC lift rigging is removed. Water is added into the annulus space between the MPC and HI-TRAC VW, if required. The annulus and HI-TRAC VW top surfaces are covered to protect them from debris produced when removing the MPC lid weld. The weld removal system is installed and the MPC vent and drain ports are accessed. The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is filled with water (borated as required) at a controlled rate to avoid over-pressuring the MPC. The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place.

The top surfaces of the HI-TRAC VW and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC VW lift blocks (for HI-TRAC VW Version P, the lift yoke is engaged to the HI-TRAC VW lifting trunnions). If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC VW is placed in the spent fuel pool and the MPC lid is removed. All

fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are cleared of any assembly debris and crud. HI-TRAC VW and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and overpack are decontaminated.

9.4.2 HI-STORM FW Recovery from Storage

1. Recover the MPC from HI-STORM FW as follows:

   a. Perform a transport route walkdown to ensure that the cask transport conditions are met.

   b. Transfer HI-STORM FW to the fuel building or site designated location for the MPC transfer.

   c. Position HI-STORM FW under the lifting device.

   d. Remove the HI-STORM FW lid.

   e. Install the mating device with bottom lid on top of the HI-STORM FW.

   f. Remove the MPC lift attachment plugs and install the MPC lift rigging to the MPC lid.

2. At the site's discretion, perform a HI-TRAC VW receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

Note:
If the HI-TRAC VW is expected to be operated in an environment below 32 ^{o}F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

3. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.

4. Engage the lift yoke to HI-TRAC VW.

5. Align HI-TRAC VW over HI-STORM FW and mate the overpacks.

6. Unbolt the bottom lid and open the mating device drawer.

7. Attach the ends of the MPC sling to the lifting device.

8. Raise the MPC into HI-TRAC VW.

9. Verify the MPC is in the full-up position.

10. Close the mating device.

11. Bolt the bottom lid to the HI-TRAC VW*.

12. Lower the MPC onto the bottom lid.

13. Disconnect the MPC lift rigging from the MPC lid.

14. Remove HI-TRAC VW from the top of the HI-STORM FW.

9.4.3 Preparation for Unloading

1. Prepare for MPC cool-down as follows:

Warning:
At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus fill. Users may also elect the source of water for the annulus. Water addition should be performed in a slow and controlled manner until water steam generation has ceased.

2. If necessary, set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC VW top surfaces to protect them from debris produced when removing the MPC lid weld.

3. Access the MPC as follows:

ALARA Note:
The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically removed.

ALARA Warning:
Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

   a. Using the marked locations of the vent and drain ports, core drill the closure ring and port cover plates.

   b. Remove the closure ring sections and the vent and drain port cover plates.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

* Upon installation, studs, nuts, and threaded plugs shall be cleaned and inspected for damage or excessive thread wear (replaced if necessary) and coated with a light layer of Loctite N-5000 High Purity Anti-Seize (or equivalent).

4. Take an MPC gas sample as follows:

Note:
Users may select alternate methods of obtaining a gas sample.

a. Attach the RVOAs.

b. Attach a sample bottle to the vent port RVOA.

c. Evacuate the RVOA and Sample Bottle.

d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

5. Fill the MPC cavity with water as follows:

Caution:
The MPC interior shall be filled with helium or another suitable inert gas to avoid exposing the fuel to oxidizing agents while at elevated temperatures. Exposing fuel at elevated temperatures to oxidizing agents can lead to deleterious oxidation of the fuel.

a. Open the vent and drain port caps using the RVOAs.

Caution:
The introduction of water into the MPC may create water vapor. Re-flooding operations shall be closely controlled to ensure that the internal pressure in the MPC does not exceed design limits. The water flow rate shall be adjusted to maintain the internal pressure below design limits. See LCO 3.1.3 and SAR section 4.5.5.

Caution:
To mitigate unfavorable thermal shocking of the fuel cladding during re-flooding operations the re-flood water shall be at a temperature $\geq$80°F. See Section 3.4.4 for related fuel cladding evaluations.

b. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

Note:
When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC. Testing must be completed within four hours prior to unloading and every 48 hours after in accordance with the LCO until all the fuel is removed from the MPC. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

c. Attach the water fill line from a water source with water temperature ≥80°F to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. (Refer to LCO 3.3.1 for boron concentration requirements). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

d. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.

Caution:
A radiolysis of water may occur in high flux conditions inside the MPC creating combustible gases. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid removal operations. The space below the MPC lid shall be purged with inert gas prior to, and during MPC lid removal operations, including grinding, and other hot work, to provide additional assurance that flammable gas concentrations will not develop in this space.

e. Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge the gas space under the lid as necessary.

f. Remove the MPC lid-to-shell weld using the weld removal system.

g. Remove any metal shavings from the top surfaces of the MPC and HI-TRAC VW.

6. Install the inflatable annulus seal.

7. Place HI-TRAC VW in the spent fuel pool as follows:

   a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.

   b. Engage the lift yoke to HI-TRAC VW lifting blocks, remove the MPC lid lifting plugs and attach the MPC lid slings.

c. Position HI-TRAC VW into the spent fuel pool in accordance with site-approved rigging procedures.

d. Disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

e. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel.

f. Disconnect the drain line from the MPC lid.

g. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid.

9.4.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Remove any debris or corrosion products from the MPC cells.

9.4.5 Post-Unloading Operations

1. Remove HI-TRAC VW and the unloaded MPC from the spent fuel pool as follows:

a. Engage the lift yoke to the HI-TRAC VW lift blocks.

b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lift blocks.

c. Raise HI-TRAC VW until HI-TRAC VW flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of HI-TRAC VW during fuel unloading.

d. Measure the dose rates at the top of HI-TRAC VW in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

e. Raise the top of HI-TRAC VW and MPC to the level of the spent fuel pool deck.

f. Close the annulus overpressure system reservoir valve, if used.

g. Lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of HI-TRAC VW, the surfaces of HI-TRAC VW and lift yoke should be kept wet until decontamination can begin.

h. Remove HI-TRAC VW from the spent fuel pool under the direction of radiation protection personnel.

i. Disconnect the annulus overpressure system from the HI-TRAC VW.

j. Place HI-TRAC VW in the designated preparation area.

k. Disengage the lift yoke.

l. Perform decontamination on HI-TRAC VW and the lift yoke.

m. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

n. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

o. Drain the water in the annulus area by connecting the drain line to the HI-TRAC VW drain connector.

p. Remove the MPC from HI-TRAC VW and decontaminate the MPC as necessary.

q. Decontaminate HI-TRAC VW.

r. Return any HI-STORM FW equipment to storage as necessary.

9.5 REFERENCES

[9.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[9.1.1] U.S. Code of Federal Regulations, Title 10 "Energy", Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,"

[9.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

[9.1.3] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code".

[9.5.1] U.S. Code of Federal Regulations, Title 10 " Energy", Part 20, "Standards for Protection Against Radiation,"

CHAPTER 10†: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

10.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM FW system (overpack, MPC and transfer cask) to verify that the structures, systems and components (SSCs) classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM FW system fabricated, assembled, inspected, tested, and accepted for use under the purview of the HI-STORM FW system CoC.

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters ensure that the HI-STORM FW system will maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions; will maintain subcriticality control; will reject the decay heat of the stored radioactive materials to the environment by passive means and maintain radiation doses within regulatory limits.

Both pre-operational and operational tests and inspections are performed throughout HI-STORM FW system operations to assure that the HI-STORM FW system is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, pressure tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 9 identifies the tests and inspections. "Pre-operation" as referred to in this chapter defines that period of time from receipt inspection of a HI-STORM FW system until the empty MPC is loaded into a HI-TRAC transfer cask for fuel assembly loading.

The HI-STORM FW system is classified as important-to-safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM FW system shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance with a quality program commensurate with the particular SSC's graded quality category. The licensing drawings identify all important to safety subcomponents of the HI-STORM FW system.

The acceptance criteria and maintenance program described in this chapter comply with the requirements of 10CFR72 [10.0.1] and NUREG-1536 [10.0.2] to the maximum extent possible, as described in Chapter 1.

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

The acceptance test requirements on the manufactured welds in the HI-STORM FW system are contained in the component licensing drawings in Section 1.5. Additional details on the requirements in the drawings are provided in this chapter, which will be incorporated in the shop manufacturing documents (viz., weld procedures, shop travelers, inspection procedures, and fabrication procedures) to ensure full compliance with this FSAR.

10.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM FW system prior to and during loading of the system. These inspections and tests provide the assurance that the HI-STORM FW system has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the provisions of 10CFR72 [10.0.1].

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM FW overpack, and the HI-TRAC VW transfer casks are listed in Tables 10.1.1, 10.1.2, and 10.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 9 and 13 provide operating guidance and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM FW system has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR. Identification and resolution of manufacturing noncompliances, if any, shall be performed in accordance with the Holtec International Quality Assurance Program approved by the USNRC.

The contents of this chapter related to welding non-destructive examination are presented in the drawing package in Section 1.5. Likewise, the material on testing and maintenance of system components in this FSAR governs the content of the daughter documents such as the Manufacturing Manual and O&M Manual for the system components used in the manufacturing and long-term maintenance of the system components, respectively.

10.1.1 Fabrication and Nondestructive Examination (NDE)

This subsection summarizes the test program required for the HI-STORM FW system.

10.1.1.1 Fabrication Requirements

The following fabrication controls and required inspections shall be performed on the HI-STORM FW system, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

i. Materials of construction specified for the HI-STORM FW system are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by the ASME Code [10.1.1] Section II (where applicable),

the requirements of ASME Section III (where applicable), Holtec procurement specifications, and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to ensure that material traceability is maintained throughout fabrication. Materials for the Confinement Boundary (MPC baseplate, lid, closure ring, port cover plates and shell) shall also be procured in compliance with the requirements of ASME Section III, Article NB-2500.

ii. The MPC Confinement Boundary shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NB to the extent practicable, as explained in this chapter.

iii. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable code (such as ASME Section III Subsection NB for the Confinement Boundary.

iv. Code welds shall be visually examined in accordance with ASME Code, Section V, Article 9. The acceptance criteria for the welds shall be based on the ASME Codes provided in Table 10.1.5. These additional NDE criteria are also specified on the licensing drawings in Section 1.5 for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be subject to review and approval by Holtec in accordance with the Company's QA program prior to use. NDE inspections of code welds shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [10.1.2] or other site-specific, NRC-approved program for personnel qualification.

v. The MPC confinement boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, RT, UT, MT and/or PT, as applicable) to verify that it is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its confinement effectiveness.

vi. Repair of confinement boundary welds shall conform to the requirements of the ASME Code, Section III, Article NB-4450.

vii. Base metal repairs shall be performed and examined in accordance with the applicable reference code set down in Table 10.1.5.

viii. Grinding and machining operations on the MPC Confinement Boundary shall be controlled through written and approved procedures to ensure grinding and machining operations do not reduce local base metal wall thicknesses of the Confinement Boundary below allowable limits. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved

procedures to verify base metal thickness meets the applicable requirements.

ix. Optional peening of the MPC Confinement Boundary and/or MPC Confinement Boundary welds shall be controlled through written and approved procedures to ensure the efficacy of the peening process and the integrity of the Confinement Boundary. Procedure qualification for the peening process shall be performed in accordance with Section 10.1.1.5.

x. Non-structural tack welds that do not become an integral part of a weld are not required to be removed. Non-structural tack welds that do not become an integral part of a permanent weld shall be examined by an approved visual examination procedure.

xi. The HI-STORM FW system shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.

xii. Each cask shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

xiii. A documentation package shall be prepared and maintained during fabrication of each HI-STORM FW system to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM FW system or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, as applicable, but not be limited to:

- Completed Shop Weld Records
- Inspection Records
- Nonconformance Reports
- Material Test Reports
- NDE Reports
- Dimensional Inspection Report

10.1.1.2 MPC Lid-to-Shell Weld Inspection

i. The MPC lid-to-shell (LTS) weld shall be examined using a progressive multi-layer liquid penetrant (PT) examination during welding.

ii. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed as specified in the drawing package in Section 1.5.

The inspection results, including relevant findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. Mapping is considered an equivalent record which contains the type, size and location of the relevant indications discovered during weld examination. The documentation of relevant indications should be taken during the final interpretation period described in ASME Section V, Article 6, T-676.

The multi-layer PT examination of the LTS weld, in conjunction with other examinations and tests performed on this weld, shall use ASME Section III acceptance criteria (see Table 10.1.4) which provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The multi-layer PT examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

10.1.1.3 Visual Inspections and Measurements

The HI-STORM FW system components shall be assembled in accordance with the licensing drawing package in Section 1.5. The drawings provide dimensional tolerances that define the limits on the dimensions used in licensing basis analysis. Fabrication drawings provide additional dimensional tolerances necessary to ensure fit-up of parts. Visual inspections and measurements shall be made and controls shall be exercised to ensure that the cask components conform to the dimensions and tolerances specified on the licensing and fabrication drawings. These dimensions are subject to independent confirmation and documentation in accordance with the Holtec QA program approved in NRC Docket No. 71-0784.

The following shall be verified as part of visual inspections and measurements:

- Visual inspections and measurements shall be made to ensure that the systems' effectiveness is not significantly reduced as a result of manufacturing deviations. Any *important-to-safety* component found to be under the specified minimum thickness shall be justified under the rules of 10CFR72.48 or repaired or replaced, as appropriate.
- Visual inspections shall be made to verify that neutron absorber panels and basket shims are present as required by the MPC basket design.
- The system components shall be inspected for cleanliness and preparation for shipping in accordance with written and approved procedures.

The visual inspection and measurement results for the HI-STORM FW system shall become part of the final quality documentation package.

10.1.1.4 Weld Examination

The examination of the HI-STORM FW system welds shall be performed in accordance with the drawing package in Section 1.5 and the applicable codes and standards.

All code weld inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A. All required inspections, examinations, and tests shall become part of the final quality documentation package.

The following specific weld requirements shall be followed in order to verify fabrication in accordance with the provisions of this FSAR.

1. Confinement Boundary welds including any attachment welds (and temporary welds to the Confinement Boundary) shall be examined in accordance with ASME Code Section V, with acceptance criteria per ASME Code Section III, Subsection NB, Article NB-5300. Examinations, Visual (VT), Radiographic (RT), and Liquid Penetrant (PT), apply to these welds as defined by the code. These welds shall be repaired in accordance with the requirements of the ASME Code Section III, Article NB-4450 and examined after repair in the same manner as the original weld.

2. Basket welds, although they are conservatively not credited in structural analysis, shall be examined and repaired in accordance with NDE specified in the drawing package and with written and approved procedures developed specifically for welding Metamic-HT with acceptance criteria per ASME Section V, Article 1, Paragraph T-150 (2007 Edition). The basket welds, made by the Friction Stir Weld process, are classified as Category C per NG-3351.3 and belonging to Type III (by virtue of being corner joint with a thru-thickness "stir zone") in Table NG-3352-1. These weld requirements are not applicable to welds identified as NITS on the drawing package.

3. Non-code welds shall be examined and repaired in accordance with written and approved procedures as defined in the system Manufacturing Manual.

10.1.1.5 MPC Surface Peening Procedure Qualification

Peening shall be performed using a QA validated procedure that is qualified to deliver the expected surface enhancement on a repeatable basis while maintaining the structural integrity of the MPC. The essential variables of the process shall be identified by the peening service provider and their acceptable range identified in the procedure.

Testing shall be performed to confirm that the peening process maintains all MPC safety functions. Testing shall use sample coupons that simulate the surface condition of the Confinement Boundary and/or MPC Confinement Boundary welds that are to be peened. Table 10.1.9 specifies the required testing that must precede the first peening operation carried out on a production MPC to confirm the efficacy of the peening procedure for a specific MPC fabrication

condition. The MPC fabrication condition is defined as the combination of the following parameters:

- Material type
- Material thickness
- Weld method and weld preparation area
- Weld orientation (axial / circumferential / axial and circumferential)
- Rolling method and welding/rolling sequence
- Peening process and essential variables

The tests in Table 10.1.9 shall be repeated to requalify the peening procedure for any change to the MPC fabrication condition. Alternatively, a technical evaluation may be performed to justify acceptance of a change to the MPC fabrication condition without retesting. Tests may be performed on separate and independent coupons for different conditions. All coupons shall be manufactured and tested per an approved procedure and all results shall be documented in a technical report and saved as a company record.

10.1.2 Structural and Pressure Tests

10.1.2.1 Lifting Locations

The lifting of all HI-STORM FW components (except the HI-TRAC VW Version P and HI-STORM FW Version XL lid) is engineered to occur though threaded couplings integral to the strongest part in the component. Thus, as shown in the HI-TRAC VW drawings (Section 1.5) the threaded connection is located in the top forging. These lift locations are accordingly referred to as *tapped anchor locations* (TAL). The TALs to lift the MPCs (in all Holtec designs) is located in the top lid (thickest part) and those for the HI-STORM FW overpack are welded to the radial connector plates (in all HI-STORM models).

Because the TALs are integral to the component, they possess high ductility and shall meet the requirements as shown in Paragraph 3.4.3.1 and Table 2.2.6.

Two trunnions (located on the HI-TRAC VW Version P top flange) are provided for vertical lifting and handling. The trunnions are designed in accordance with NUREG 0612 using a high strength and high ductility material. The trunnions contain no welded components. The maximum design lifting load for the HI-TRAC VW Version P will occur during removal of the HI-TRAC from the spent fuel pool after the MPC has been loaded, flood with water, and the MPC lid is installed. The high-material ductility, absence of materials vulnerable to brittle fracture, large stress margins, and a carefully engineered design to eliminate local stress risers in the high-stressed regions (during the lift operations) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612, the acceptance criteria for the lifting trunnions must be established in conjunction with other considerations applicable to heavy load handling.

Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure is one:

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

Each TAL will be subjected to a dimensional test in the shop using go/no-go gauges to ensure that the threads meet the dimensional requirements. As an alternative to the thread gauge test, the threads may be proof-tested using a torque test to simulate a load equal to three times the design load. Furthermore, the thread in the TAL shall be visually inspected in accordance with a written procedure to ensure absence of burrs, undercuts, and other stress raisers.

The acceptance testing of the TALS in the manner described above will provide adequate assurance against handling accidents.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load shall be applied for a minimum of 10 minutes. The accessible parts of the trunnions (areas outside the HI-TRAC VW Version P cask), and the adjacent HI-TRAC VW Version P cask trunnion attachment area shall then be visually examined to verify no deformation, distortion, or cracking occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-TRAC VW Version P cask trunnion attachment areas shall require replacement of the trunnion and/or repair of the HI-TRAC VW Version P cask. Following any replacements and/or repair, the load testing shall be performed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing shall be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section II requirements will provide further verification of the trunnion load capabilities. Test results shall be documented. The documentation shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above will provide adequate assurance against handling accidents.

10.1.2.2 Pressure Testing

10.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

All HI-TRAC transfer cask water jackets shall be hydrostatically tested in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

10.1.2.2.2 MPC Confinement Boundary

Pressure testing (hydrostatic or pneumatic) of the MPC Confinement Boundary shall be performed to verify the lid-to-shell field weld in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 and applicable sub-articles, when field welding of the MPC lid-to-shell weld is completed. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure. The calibrated test pressure gage installed on the MPC Confinement Boundary shall have an upper limit of approximately twice that of the test pressure. The MPC vent and drain ports will be used for pressurizing the MPC cavity. Water shall be pumped into the MPC drain port until water only is flowing from the MPC vent port. The MPC vent port is then closed and the pressure is increased to the test pressure. While the MPC is under pressure, the MPC lid-to-shell weld shall be examined for leakage. If any leaks are observed, the pressure shall be released and the weld shall be repaired in accordance with the requirements of ASME Code, Section III, Subsection NB. Following completion of the required hold period at the test pressure, the pressure shall be released and the surface of the MPC lid-to-shell weld shall be re-examined by liquid penetrant examination in accordance with ASME

Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

10.1.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested are identified in Table 3.1.9 and applicable weld materials. Table 3.1.9 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as set down in Chapter 1.D of the HI-STORM 100 FSAR (Docket 72-1014) [10.1.6] in accordance with written and approved procedures. Testing shall verify the compressive strength and density meet design requirements. Tests required shall be performed at a frequency as defined in the applicable ACI code.

Qualification tests on Metamic-HT coupons drawn from production runs shall be performed in compliance with Table 10.1.6 requirements to ensure that the manufactured panels shall render their intended function. Testing shall be performed using written and approved procedures consistent with the test methods documented in Holtec's test report [10.1.7]. To ensure the above test requirements are met a sampling plan based on the MIL Standard 105E [10.1.8] is defined and incorporated in the Metamic-HT Manufacturing Manual's Shop Operating Procedure HTSOP-108.

Test results on all materials shall be documented and become part of the final quality documentation package.

10.1.4 Leakage Testing

Leakage testing shall be performed in accordance with written and approved procedures and the leakage test methods and procedures of ANSI N14.5 [10.1.5], as follows.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The acceptance criterion is "leaktight" as defined in ANSI N14.5. The helium leakage test of the vent and drain port cover plate welds shall be performed using a helium mass spectrometer leak detector (MSLD). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are not required. Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in Chapter 9 of this FSAR and the acceptance criteria are defined in the Technical Specifications for the HI-STORM FW system.

10.1.5 Component Tests

10.1.5.1 Valves, Pressure Relief Devices, and Fluid Transport Devices

There are no fluid transport devices associated with the HI-STORM FW system. The only valve-like components in the HI-STORM FW system are the specially designed caps installed in the MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully-welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and are liquid penetrant examined and leakage tested to verify the MPC Confinement Boundary.

There are multiple pressure relief devices installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. One is provided for venting air and water due to pressure build-up from thermal expansion of the water in the water jacket. The other relief devices are provided for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The set pressures for the pressure relief devices are listed on the HI-TRAC VW drawings in Section 1.5.

10.1.5.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM FW system.

10.1.6 Shielding Integrity

The HI-STORM FW overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM FW overpack concrete provides both neutron and gamma shielding. The overpack's inner and outer steel shells, and the steel shield shell, provide radial gamma shielding. Concrete and steel plates provide axial neutron and gamma shielding.

The HI-TRAC VW transfer cask uses three different materials for primary shielding. All HI-TRAC VW transfer cask designs include a radial steel-lead-steel shield and a removable steel bottom lid. Testing requirements on shielding materials are presented below.

Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR drawings in Section 1.5 prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 10.1.3 for concrete material testing requirements.

Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure that voids are minimized. The lead shall be examined to preclude macrovoids (through holes) in the material using written and qualified procedures.

The lead shall be installed in such a manner that there are no macro-voids (through holes) and that the cask is not subjected to a severe thermal cycle.

Steel:

Steel plates utilized in the construction of the HI-STORM FW system shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

General Requirements for Shield Materials:

1. Test results for concrete density and lead examinations for macrovoids, as applicable, shall be documented and become part of the quality documentation package.

2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed after initial loading operations in accordance with description below and the operating procedures in Chapter 9.

10.1.6.1 Shielding Effectiveness Tests

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures at the host plant site. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM FW overpack and HI-TRAC VW. Measurements shall be taken at the locations specified in the Radiation Protection Program for comparison against the prescribed limits. The test is considered acceptable if the dose rate readings are less than or equal to the calculated limits. If dose rates are higher than the limits, the required actions provided in the Radiation Protection Program shall be carried out. Dose rate measurements shall be documented and shall become part of the quality record of the loaded cask.

10.1.6.2 Neutron Absorber Manufacturing Requirements

Essential characteristics of Metamic-HT are described in Chapter 1 of this FSAR. As described in Chapter 1, Metamic-HT is made from high purity aluminum using a powder metallurgy process that results in pinning of the materials grain boundaries by dispersoids of nanoparicles of aluminum oxide. The manufacturing of Metamic-HT is governed by a set of quality validated Holtec Standard procedures contained in the Metamic-HT Manufacturing Manual [1.2.7].

The key constituents of Metamic-HT, namely aluminum powder and Boron Carbide powder are procured under their respective purchasing specifications that define the required particle size distributions and set down the prohibited materials & impurities, as well as tolerable level of impurities. The supplier of raw materials must be qualified under Holtec's quality program for important to safety materials and components or the material shall be commercially dedicated by Holtec in accordance with the Holtec Quality Assurance program.

A description of the manufacturing processes for Metamic-HT is presented in the Metamic-HT Sourcebook [1.2.6] and implemented in the Metamic-HT Manufacturing Manual [1.2.7].

As required by the procedures set down in its manufacturing manual [1.2.6], each panel of Metamic-HT neutron absorber material shall be visually inspected for damage such as scratches, cracks, burrs, presence of imbedded foreign materials, voids and discontinuities that could significantly affect its functional effectiveness.

Metamic-HT panels will be manufactured according to a Holtec purchase specification that incorporates all requirements set forth in this FSAR. The manufacturing of Metamic-HT is subject to all quality assurance requirements under Holtec International's NRC approved quality program.

The tests conducted on Metamic-HT to establish the compliance of the manufactured panels with Holtec's Purchasing Specification are intended to ensure that *critical characteristics* of the final product will meet the minimum guaranteed values (MGVs) set forth in this FSAR (Table 1.2.8a). The tests are performed at both the raw material and manufactured extrusion/panel stages of

production with the former serving as the insurer of the properties in the final product and the latter serving the confirmatory function.

The testing is conducted for each lot of raw material and finished panels as prescribed in Table 10.1.6. A lot is defined as follows:

> "Lot" means a population of an item that shares identical attributes that are central to defining a critical performance or operational characteristic required of it. Thus, a lot of boron carbide powder procured to a certified Purchasing Specification used in the manufacturing of Metamic-HT is the bulk quantity of the powder that has the same particle size distribution. A lot of finished panels drawn from a powder mix and manufactured in an extrusion run have identical aluminum and boron carbide characteristics and the same extrusion conditions.

The following tests are performed (see Table 10.1.6):

(i) Testing and certification of powder material

- All lots of aluminum and boron carbide powder shall be certified to meet particle size distribution and chemistry requirements in [1.2.8].
- All lots of B_4C shall be certified as containing Boron with the minimum isotopic B-10 per the boron carbide purchase specifications incorporated in the Manufacturing manual [1.2.7].
- Homogenized mixtures of Al powder(s) and boron carbide powder(s) from traceable lots, prepared for sintering and billet forming operations, shall have the minimum boron carbide wt% verified by wet chemistry testing of one sample from each lot of blended powders. The mixing/blending of the batch shall be controlled via approved procedures.

(ii) Testing of finished panels

The number of panels subject to testing shall be governed by Table 10.1.7. The panels that need to be tested per the statistical protocol of Table 10.1.7, hereafter referred to as test panels, shall be subject to the following evaluations:

- The Metamic-HT panels shall be tested for all mechanical properties in Table 1.2.8a in accordance with Table 10.1.7 sampling plan.
- The thickness, width, straightness, camber, and bow of the first and last panel of each lot or operating shift, whichever comes first, will be measured using the procedure set down in [1.2.7]. The average measured thickness value must meet the minimum basket wall requirements specified in the Licensing drawings in Section 1.5.
- One coupon from the test panel shall be subject to neutron attenuation testing to quantify the boron carbide content for compliance with the minimum requirement in Table 1.2.2 using written procedures.

(iii) Testing of Basket

- Metamic-HT basket welds shall be tested/inspected as stated in Section 10.1.1.4 using written procedures.

Each neutron absorber plate shall be visually inspected for damage such as scratches, cracks, burrs, foreign material embedded in the surfaces, voids, and delaminations. Panels are also visually inspected for contamination on the surface as specified in the Manufacturing Manual [1.2.7]. Panels not meeting the acceptance criteria will be reworked or rejected. Unless basket is fabricated at the same factory manufacturing Metamic-HT, all panels shall be inspected before being shipped to the cask manufacturing facility where they may be subject to receipt inspection prior to installation.

FSW Procedure Qualification, Welder Operator Qualification and Welded Coupon Test:

A. Procedure qualification and welder operator qualification of the Friction Stir Welding (FSW) process shall meet the following requirements:

   - The Procedure Qualification Record (PQR) shall meet the essential variable requirements of QW-267.

   - The Weld Procedure Specification (WPS) shall meet the essential variable requirements of QW-267, QW-361.1(e) and QW-361.2.

   - Welder operator performance qualifications shall meet the essential variable requirements of QW-361.2.

   - Welder operator may be qualified by volumetric NDE of a test coupon; or a coupon from their initial production welding within the limitations of QW-304 and QW-305; or by bend tests taken from a test coupon.

   - All welding by FSW process shall meet applicable requirements of ASME Section IX, 2013 Edition [8.1.1].

B. Procedure qualification of the Friction Stir Welding process may be accomplished by tensile testing the appropriate number of coupons per ASME Section IX (2007) and achieving a nominal 60% of Metamic-HT MGV tensile strength. Verification of weld soundness is performed by visual examination, radiography and bend testing per approved written procedures (bend testing emulates ASME Section IX). Bend test qualification of a representative weld sample emulating ASME Section IX paragraph QW 160 at a bend radius that produces at least 150% of the average tensile strain developed in the friction stir welded joint under the hypothetical free drop accident condition. The bend radius shall be recorded on the PQR. The bend test sample must meet the

acceptance criteria of Section IX QW-163 and visual examination acceptance criteria of ASME Section III Subsection NG 5362 with any additional requirements per Holtec approved written procedure. In addition, at least one welded coupon from the population of Metamic-HT production panels used for manufacturing each fuel basket type must pass the criteria provided herein. The results shall be documented and saved in the company's electronic database.

10.1.7 Thermal Acceptance Tests

The thermal performance of the HI-STORM FW system, including the MPCs and HI-TRAC transfer cask, is demonstrated through analysis in Chapter 4 of this FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading.

The first manufactured MPC, either MPC-37 or MPC-89, will be thermally tested using an approved QA controlled Holtec procedure [10.1.9]. The following are the basic steps of this procedure.

1. The MPC will be arrayed in the vertical orientation on the test pad with interface insulation to minimize heat loss from the bottom.
2. Twelve storage cells (three in each quadrant) will be loaded with bayonet electric heaters each calibrated to deliver one kilowatt heat uniformly over its length. The heaters will be situated co-axially within each storage cell. Thus the heat generation in the MPC shall be quadrant-symmetric.
3. The top of the MPC shall be enclosed by an insulated lid. Calibrated thermocouples will be fastened to selected cell walls in each quadrant in a symmetric manner.
4. The test will be run for a sufficiently long time such that steady state conditions are reached. The ambient temperature and the thermocouple readings will be taken as specified in the test procedure.
5. The test condition will be simulated on the design basis FLUENT model of the MPC in Chapter 4 and the temperatures at all of the thermocouple locations predicted by FLUENT will be compared with the test data.
6. The amounts by which the FLUENT temperatures exceed the corresponding measured temperatures (positive margin) collectively define the margin of conservatism in the FSAR analysis model. A negative margin will warrant an immediate report to the NRC and appropriate licensing action pursuant to Holtec's QA program.

Following the loading and placement on the storage pad of the first HI-STORM system placed in service as specified in CoC Condition #8, the operability of the natural convective cooling of the HI-STORM FW system shall be verified by the performance of an air mass flow rate test. A description of the test is described in Chapter 9.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM FW heat removal system.

10.1.8 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in 10 CFR 72.236(k).

Table 10.1.1
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of MPC code welds per ASME Code Section III, Subsection NB, as defined on design drawings, per NB-5300, as applicable. b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements. c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements. d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. Acceptance criteria for non-code welds are defined on the drawings. e) Cleanliness of the MPC shall be verified upon completion of fabrication. f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.	a) The MPC shall be visually inspected prior to placement in service at the licensee's facility. b) MPC protection at the licensee's facility shall be verified. c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.	a) None.

Table 10.1.1 (continued)
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of MPC components is performed per ASME Code Section IX and III, Subsection NB, as applicable. b) Materials analysis (steel, neutron absorber, etc.), is performed and records are kept in a manner commensurate with "important to safety" classifications.	a) None.	a) A multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld is performed per ASME Section V, Article 2. Acceptance criteria for the examination are defined in Subsection 10.1.1, and in the Licensing Drawings. b) ASME Code NB-6000 pressure test is performed after MPC closure welding. Acceptance criteria are defined in the Code.
Leak Tests	a) Helium leakage testing of the MPC base metal (shell, baseplate and MPC lid),MPC shell to baseplate welds and MPC shell to shell welds is performed on the unloaded MPC. Acceptance criterion is in accordance with "leaktight" definition in ANSI N14.5.	a) None.	a) Helium leakage testing is performed on the vent and drain port cover plates to MPC lid field welds. See Technical Specification for guidance on acceptance criteria.
Criticality Safety	a) The boron content is verified at the time of neutron absorber material manufacture. b) The installation of MPC cell panels is verified by inspection.	None.	None.
Shielding Integrity	a) Material compliance is verified through CMTRs. b) Dimensional verification of MPC lid thickness is performed.	None.	None.

Table 10.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Thermal Acceptance	a)None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is verified during fabrication. - MPC lid - vent/drain port cover plates - MPC closure ring b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components is verified during pre-operation. -MPC lid -MPC closure ring -vent/drain cover plates	a) None.
Canister Identification Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 10.1.2
HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	Structural Steel Components: a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All structural weld s requiring PT examination as shown on the Licensing Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All structural welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using ANSI NDE symbols and/or notations. Concrete Components: The following processes related to concrete components shall be implemented in accordance with the provisions of Appendix 1.D of [10.1.6]. Concrete testing shall be in accordance with Table 1.D.1. Activities shall be conducted in accordance with written and approved procedures. a) Assembly and examination. b) Mixing, pouring, and testing.	a) The overpack shall be visually inspected prior to placement in service. b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means. c) overpack protection at the licensee's facility shall be verified. d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.	a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned. b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.

Table 10.1.2 (continued) HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. b) Concrete compressive strength tests shall be performed per Appendix 1.D of [10.1.6].	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per Appendix 1.D of [10.1.6], at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement.	a) None	a) A shielding effectiveness test shall be performed after the initial fuel loading.

Table 10.1.2 (continued)
HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM FW system in accordance with the operating procedures in Chapter 9. b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's option. (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions. (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading.	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.

Table 10.1.3
HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All structural welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All structural welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using standard ANSI NDE symbols and/or notations e) Cleanliness of the transfer cask shall be verified upon completion of fabrication. f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.	a) The transfer cask shall be visually inspected prior to placement in service. b) Transfer cask protection at the licensee's facility shall be verified. c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.	a) Visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements.

Table 10.1.3 (continued) HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. b) A pressure test of the neutron shield water jacket shall be performed upon completion of fabrication. c) A load test of the HI-TRAC VW Version P lifting trunnions shall be performed during fabrication per ANSI N14.6.	a) None.	a) Testing to verify continuing compliance of the HI-TRAC VW Version P lifting trunnions shall be performed per ANSI N14.6. .
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) None.	a) None.	a) None.
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading.	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask bottom lid shall be performed during fabrication.	a) Fit-up test of the HI-TRAC VW Version P lifting trunnions with the lift yoke shall be performed..	a) Fit-up of the bottom lid shall be verified prior to use.

Table 10.1.4
HI-STORM FW MPC NDE REQUIREMENTS

Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
Shell longitudinal seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Lid-to-shell	PT (root and final pass) and multi-layer PT. PT (surface following pressure test)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug and lift lug baseplate	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350

Table 10.1.5

REFERENCE ASME CODES FOR CODE WELD INSPECTIONS AND INSPECTION CRITERIA OF HI-STORM FW COMPONENTS

Component	Applicable Reference Code for Inspection Criteria	Applicable Code for Inspection Process
MPC Confinement Boundary	ASME Section III Subsection NB	Section V
HI-STORM FW Overpack Steel Weldment	ASME Section III Subsection NF for Class 3 Structures	Section V
HI-TRAC VW Transfer Cask (Steel Weldment)	ASME Section III Subsection NF for Class 3 Structures	Section V

Table 10.1.6
Metamic-HT Testing Requirements

	Item Tested	Property Tested For	Frequency of Test	Purpose of Test	Acceptance Criterion
i.	B_4C powder (raw material) (see note 1)	Particle size distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification [1.2.7]
		Purity	One sample per lot	To verify material supplier's data sheet	ASTM C-750
ii.	Al Powder (raw material)	Particle Size Distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification [1.2.8]
		Purity	One sample per lot	To verify material supplier's data sheet	Must be 99% (min.) pure aluminum
iii.	B_4C/Al Mix	B_4C Content (by the wet chemistry method)	One sample per mixed/blended powders lot	To ensure wt.% B_4C requirements compliance	The weight density of B_4C must meet the minimum wt% specification in Table 1.2.2.
iv.	Finished Metamic-HT panel	Thickness and width, straightness, camber and bow	Per Sampling Plan Table 10.1.7 (see Note 3)	To ensure fabricability of the basket	Per Holtec's Purchasing Specification [1.2.8]
		Mechanical Properties, (See Table 10.1.8)	Per Sampling Plan Table 10.1.7 (see Note 2)	To ensure structural performance.	MGV per Table 1.2.8a
		B-10 areal density (by neutron attenuation)	One coupon from each Metamic-HT manufactured lot	To ensure criticality safety	The B_4C content must meet the minimum wt% specification in Table 1.2.2.

Notes:

1. The B_4C testing requirements apply if the raw material supplier is not in Holtec's (Or Nanotec's) Approved Vendor List.
2. Sampling Plan is included in the Metamic-HT Manufacturing Manual [1.2.7].
3. The sampling plan for the dimensional analysis of the finished Metamic-HT panels shall start at Tier No. 4 of Table 10.1.7 with samples coming from the first and last panel of every lot or operating shift, whichever comes first. If it is required to move up a Tier No. per the notes in Table 10.1.7, then that many panels shall be tested in the beginning and end of each lot or operating shift, whichever comes first, and one sample shall be taken from each panel.

Table 10.1.7 Tier System for Coupon Testing		
Tier No.	Number of Extrusions Tested as a Percent of Number of Extrusions in the Lot	Number of Continuous Lots that Must Pass to Drop Down to the Next Tier
1	20	5
2	12.5	5
3	5	10
4	1	N/A
Note 1: If a coupon fails with respect to any MGV property, then it may be replaced by two coupons from the extrusion that produced the failed coupon. If both of the replacement coupons pass the failed MGV property, then the lot can be accepted. If either of the replacement coupons is unsuccessful in meeting the failed MGV property, then the entire lot is rejected. As an alternative to rejecting the entire lot, testing of the failed MGV value on all extrusions within the lot is permitted to isolate acceptable panels. Note 2: Testing shall be moved up to the next tier if any MGV property fails in two consecutive lots.		

Table 10.1.8
Minimum Guaranteed Values Required for Certification of Production Runs of Metamic-HT
(All testing performed at ambient temperature.)

	Property	MGV
1	Yield Strength, ksi	See Table 1.2.8a for MGV values
2	Tensile Strength, ksi	
3	Young's Modulus, ksi	
4	Area Reduction, %	

Table 10.1.9
Peening Procedure Qualification Testing Requirements

Test	Test Direction	Acceptance Criterion
Residual Stress Testing (Note 1)	Direction of weld	Residual stresses shall have an insignificant effect on the computed safety factors in Chapter 3. For purposes of this determination, an insignificant loss of safety margin with reference to an acceptance criterion is defined as the estimated reduction that is no more than one order of magnitude below the available margin reported in the FSAR.
	Perpendicular to direction of weld	

Notes:

1. Testing shall be repeated for any change in MPC fabrication condition. MPC fabrication conditions are listed in Paragraph 10.1.1.5.

10.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM FW system Operations and Maintenance Manual, which shall be prepared and issued prior to the first use of the system by a user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal, and confinement performance, radiological safety, and proper handling of the system in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM FW system is totally passive by design: There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from the effects of weather. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures that are far less demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STORM FW system is provided in Table 10.2.1.

10.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC TALs and the bottom lid bolts* and bolt holes. The examination shall inspect for indications of overstress such as cracks, deformation, wear marks, and missing or damaged threads. Repairs or replacement in accordance with written and approved procedures shall be required if an unacceptable condition is identified.

As described in Chapters 7 and 12 of this FSAR, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

10.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 12, there are no credible normal, off-normal, or accident events which can

* Upon installation, studs, nuts, and threaded plugs shall be cleaned and inspected for damage or excessive thread wear (replaced if necessary) and coated with a light layer of Loctite N-5000 High Purity Anti-Seize (or equivalent).

cause the failure of the MPC Confinement Boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

10.2.3 Subsystem Maintenance

The HI-STORM FW system does not include any subsystems, which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, forced helium drying, helium backfill, and leakage testing systems per their O&M manuals. Rigging, remote welders, cranes, and lifting beams shall also be inspected prior to each loading campaign to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations with the HI-STORM FW require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also Subsection 10.2.6.

10.2.4 Pressure Relief Devices

The pressure relief devices used on the water jackets for the HI-TRAC VW transfer cask shall be calibrated as specified in the HI-TRAC VW O&M Manual to ensure pressure relief settings are accurate prior to the cask's use.

10.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM FW overpack, HI-TRAC VW, and MPC are not subject to measurable degradation over time or as a result of usage.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.104(c) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC VW water jacket shall be verified during each loading campaign in accordance with the licensee's approved operations procedures.

The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM FW system.

10.2.6 Thermal

In order to assure that the HI-STORM FW system continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 10.2.1

HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	Frequency
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM FW Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC TAL or Lifting Trunnion visual inspection	Prior to each handling campaign
HI-TRAC bottom lid bolts and bolt holes	Prior to each handling campaign
HI-TRAC pressure relief device calibration	Per the device manufacturer's recommendation.
HI-TRAC internal and external visual inspection for compliance with design drawings	Annually†
HI-TRAC water jacket water level visual examination	During each handling campaign in accordance with licensee approved operations procedures
Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

† Or prior to next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year.

10.3 REGULATORY COMPLIANCE

Chapter 10 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STORM FW system in conformance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STORM FW system will provide the assurance that the SSCs important to safety will perform their intended function without limitation. The performance of the maintenance program by the licensee for each loaded HI-STORM FW system will provide the assurance for the continued safe long-term storage of the stored SNF.

The described acceptance criteria and maintenance programs can be summarized in the following evaluation statements:

1. Section 10.1 of this FSAR describes Holtec International's proposed program for pre-operational testing and initial operations of the HI-STORM FW system. Section 10.2 describes the proposed HI-STORM FW system's maintenance program.

2. Structures, systems, and components (SSCs) of the HI-STORM FW system designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with their safety category. The licensing drawings in Section 1.5 and Table 9.2.1 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STORM FW system and its ancillary equipment, respectively. Tables 1.2.6 and 1.2.7 present the applicable standards for their design, fabrication, and inspection of the HI-STORM FW system components.

3. Holtec International will examine and test the HI-STORM FW system to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 10.1 of this FSAR describes the MPC Confinement Boundary assembly, inspection, and testing.

4. Each cask shall bear a nameplate indicating its model number, unique identification number, and empty weight.

5. It can be concluded that the acceptance tests and maintenance program for the HI-STORM FW system are in compliance with 10CFR72 [10.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the overarching regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10.4 REFERENCES

[10.0.1] U.S. Code of Federal Regulations, Title 10, "Energy", Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste".

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

[10.1.1] American Society of Mechanical Engineers, " Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 2007 Edition.

[10.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.

[10.1.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.

[10.1.4] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.

[10.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5, January 1997.

[10.1.6] "Final Safety Analysis Report for HI-STORM 100 Cask Storage System", Holtec Report No. HI-2002444 (latest revision).

[10.1.7] "Metamic-HT Qualification Sourcebook", Holtec Report No. HI-2084122, Latest Revision (Holtec Proprietary)[2]

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

[10.1.9] "HI-STORM FW MPC Thermal Test Procedure", Holtec Procedure HPP-5018-1, Rev. 0.

[2]Supporting document submitted with the HI-STAR 180 License Application (Docket 71-9325).

CHAPTER 11: RADIATION PROTECTION†

11.0 INTRODUCTION

This chapter discusses the design considerations and operational features that are incorporated in the HI-STORM FW 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 presented. Since the determination of off-site doses is necessarily site-specific, similar dose assessments shall be prepared by the licensee, as part of implementing the HI-STORM FW system in accordance with 10CFR72.212 [11.0.1]. The information provided in this chapter meets the requirements of NUREG-1536 [11.0.3].

11.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

11.1.1 Policy Considerations

The HI-STORM FW has been designed in accordance with 10CFR72 [11.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [11.1.1] and the guidance provided in Regulatory Guides 8.8 [11.1.2] and 8.10 [11.1.3]. Licensees using the HI-STORM FW system will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [11.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM FW system, and be familiarized with the expected dose rates around the MPC, HI-STORM overpack and HI-TRAC VW during all phases of loading, storage, and unloading operations. Chapter 13 provides dose rate limits at the HI-TRAC VW and HI-STORM overpack surfaces to ensure that the HI-STORM FW system is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings will be held with workers and radiological protection personnel prior to work on or around the system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [11.1.1] standards for radiation protection are met in accordance with the site's written commitments.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61[11.0.2]. However, the material content of this chapter also fulfills the requirements of NUREG 1536[11.0.3]. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1 in this SAR. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

It is noted that although Loading Pattern B for the MPC-37 allows assemblies with higher heat loads and therefore higher source terms in the outer region (Region 3) of the MPC, the guiding principle in selecting fuel loading should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

11.1.2 Radiation Exposure Criteria

The radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM FW system are as follows:

1. 10CFR72.104 [11.0.1] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements. As discussed below, the design features of the HI-STORM FW system components are configured to meeting this and other criteria cited below without undue burden to the user (discussed in Subsection 11.1.2).

2. 10CFR72.106 [11.0.1] requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.

3. 10CFR20 [11.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.

4. Regulatory Position 2 of Regulatory Guide 8.8 [11.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM FW storage system as described below:

   - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [11.0.1]. Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a loaded HI-STORM FW overpack is placed in an ISFSI. Due to the passive nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.

- Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 and in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM FW system design include:
  - system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;
  - system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;
  - system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;
  - system designs that minimize planned maintenance requirements;
  - system designs that minimize decontamination requirements at ISFSI decommissioning;
  - system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;
  - thick walled overpack that provides gamma and neutron shielding;
  - thick MPC lid which provides effective shielding for operators during MPC loading and unloading operations;
  - multiple welded barriers to confine radionuclides;
  - smooth surfaces (that come in contact with pool water) to reduce decontamination time;
  - minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
  - capability of maintaining uncontaminated water in the MPC during welding to reduce dose rates;
  - capability of maintaining water in the transfer cask annulus space and water jacket to reduce dose rates during closure operations;
  - MPC penetrations located and configured to reduce neutron streaming paths;
  - HI-TRAC VW designed to reduce streaming paths;
  - streaming paths in the HI-STORM FW overpack are limited to the air vent passages.
  - MPC vent and drain ports with resealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;

- use of a bottom lid, annulus seal, and Annulus Overpressure System to prevent contamination of the MPC shell outer surfaces during in-pool activities;
- maximization of shielding around the top region of HI-TRAC VW where the most human activities occur during loading operations; and
- low-maintenance design to reduce occupational dose during long-term storage.

- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.
- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM FW storage system is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this SAR, and no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean water in the HI-TRAC VW-MPC annulus and by using a proven inflatable annulus seal design.
- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM FW system ISFSI since there are no radioactive systems at an ISFSI that could transport crud.
- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the HI-TRAC VW transfer cask is designed for ease of decontamination. In addition, an inflatable annulus seal is used to prevent fuel pool water from contacting and contaminating the exterior surface of the MPC.
- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.
- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins.
- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC Enclosure Vessel. This material is resistant to the damaging effects of radiation and is well proven in the SNF cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

11.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STORM FW system include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated welding system, lift yoke, mating device and moisture removal systems to reduce time operators spend in the vicinity of the loaded MPC;
- use of a well-shielded base for staging the welding system;
- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;
- low fuel assembly lift-over height over the HI-TRAC VW maximizes water coverage over assemblies during fuel assembly loading;
- a water-filled neutron shield jacket allows filling after removal of the HI-TRAC VW from the spent fuel pool. This maximizes the shielding on the HI-TRAC VW without exceeding the crane capacity;
- descriptive operating procedures that provide guidance to reduce equipment contamination, obtain survey information, minimize dose and alert workers to possible changing radiological conditions;
- preparation and inspection of the HI-STORM FW overpack and HI-TRAC VW in low-dose areas;
- MPC lid fit tests and inspections prior to actual loading to ensure smooth operation during loading;
- gas sampling of the MPC and HI-STAR 100 annulus (receiving from transport) to assess the condition of the cladding and MPC Confinement Boundary;
- HI-STORM FW overpack temperature monitoring equipment allows remote monitoring of the vent operability surveillance;
- Use of proven ALARA measures such as wetting of component surfaces prior to placement in the spent fuel pool to reduce the need for decontamination;
- decontamination practices which consider the effects of weeping during HI-TRAC VW transfer cask heat up and surveying of HI-TRAC VW prior to removal from the fuel handling building;

- Use of non-porous neutron absorber (Metamic-HT) to preclude waterlogging of the neutron absorber to minimize basket drying time. Specifically, Boral (a sandwich of aluminum sheets containing a mixture of boron carbide and aluminum powder which tends to hold the pool water in the porous space of the mixture extending canister drying times) is prohibited from use in HI-STORM FW MPCs);
- a sequence of short-term operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations

11.1.4 Auxiliary/Temporary Shielding

In addition to the design and operational features built into the HI-STORM FW system components, a number of ancillary shielding devices can be deployed to mitigate occupational dose. Ancillaries are developed on a site-specific basis that further reduce radiation at key work locations and/or allow for operations to be performed faster to reduce the time personnel spend in close proximity in the radiation field. Licensees are encouraged to use such ALARA-friendly ancillaries and practices.

11.2 RADIATION PROTECTION FEATURES IN THE SYSTEM DESIGN

The design of the HI-STORM FW components has been principally focused on maximizing ALARA during the short-term operations as well as during long-term storage. Some of the key design features engineered in the system components to minimize occupational dose and site boundary dose are summarized in Table 11.2.1. The design measures listed in Table 11.2.1 have been incorporated in the HI-STORM FW system to effectively reduce dose in fuel storage applications.

Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE

	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
1.	HI-STORM FW Overpack	Use of the steel weldment structure permits the density of concrete (set at a minimum of 150 lb/cubic feet) to be increased to as high as 200 lb/cubic feet.	A
2.	HI-STORM FW Overpack	The lid of the HI-STORM FW overpack contains the outlet ventilation ducts (Holtec Patent No. 6,064,710) in the overpack's closure lid. This eliminates the need for temporary shielding that will otherwise be needed if the ducts were located in the cask body for MPC transfer operations.	B
3.	HI-STORM FW Overpack	Use of multiple curved inlet ducts maximize radiation blockage (Holtec Patent No. 6,519,307B1).	B
4.	HI-STORM FW Overpack	Cask's vertical disposition and use of a thick lid (see drawing package in Section 1.5) and high density concrete minimizes skyshine.	B
5.	HI-TRAC VW/ MPC	The height of the MPC minimized for each site so that the height of HI-TRAC VW can be minimized and thus the maximum amount of lateral shielding in the cask can be incorporated consistent with the plant's crane capacity limits.	B
6.	HI-TRAC VW	The geometry of the HI-TRAC is configured to maximize shielding in areas with penetrations and discontinuities, such as lifting trunnions, or eliminating trunnions and replacing them with TALs (see Glossary).	B
7.	MPC	Use of Metamic-HT in the fuel basket reduces the weight of the fuel basket (in comparison to stainless steel). Thus additional shielding can be incorporated in the transfer cask whose total weight is limited by the plant's crane capacity.	B

Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE

	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
8.	HI-STORM FW overpack/MPC	The dose from a HI-STORM FW storage system is minimized because of the following advantages: a. Regionalized storage of fuel (cold fuel in the peripheral storage cells) possible because of the Metamic-HT fuel basket and the thermosiphon action-enabled MPC provides self-shielding. (Note that while loading hotter fuel in the inner cells is a requirement for some but not all loading configurations, it is preferred from an ALARA perspective.) b. Tight packing of overpacks on the ISFSI (that maximizes self-shielding) is possible because a large spacing between the modules is not necessary.	A,B
9.	MPC, HI-TRAC VW	The occupational dose from loading a HI-STORM FW overpack is minimized because of: a. A well-shielded HI-TRAC VW transfer cask. b. Regionalized fuel loading. (Note that while loading hotter fuel in the inner cells is a requirement for some but not all loading configurations, it is preferred from an ALARA perspective) c. A short water draining time (less than 2 hours) for the MPC. d. Reduced overall MPC welding time because the welding machine does not have to be removed and replaced to weld the secondary lid. e. Reduced time and personnel needed to install the MPC in the HI-STORM FW overpack due to vertical (gravity-aided) insertion. f. Reduced drying time because of use of porosity-free Metamic-HT.	B

Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE

	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
10.	MPC	HI-STORM FW has been designed to accommodate high burnup and a maximum number of PWR or BWR fuel assemblies in each MPC to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site.	A,B
11.	HI-STORM FW overpack	HI-STORM FW overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance degradation (e.g., corrosion).	A
12.	MPC	HI-STORM FW has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the Confinement Boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or credible accident conditions.	A,B
13.	HI-TRAC VW	HI-TRAC VW transfer cask utilizes a mating device (Holtec Patent No. 6,625,246) which reduces streaming paths and simplifies operations.	B
14.	HI-TRAC VW	The HI-TRAC VW cask and mating device are designed for quick alignment with HI-STORM.	B
15.	HI-STORM FW overpack	HI-STORM FW has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel (see Section 1.4).	A
16.	HI-STORM FW overpack	The HI-STORM FW overpack features narrow and tall optimized inlet duct shapes to minimize radiation streaming.	A

Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE

	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
17.	HI-STORM FW overpack/MPC	The combination of a Metamic-HT (highly conductive) basket, a thermosiphon capable internal basket geometry, and a high profile inlet ducts enables the HI-STORM FW system to reject heat to the ambient to maintain the fuel cladding temperature below short-term limits in the scenario where the ISFSI is flooded and the floodwaters are just high enough to block off the ventilation airflow. This feature eliminates the need for human intervention to protect the fuel from damage from an adverse flood event and reduces occupational dose.	A,B
18.	HI-STORM FW overpack	The steel structure of the HI-STORM FW overpack gives it the fracture resistance properties that protect the overpack from developing streaming paths in the wake of the impact from a projectile such as a tornado missile strike or handling incident.	A,B

11.3 ESTIMATED ON-SITE CUMULATIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM FW system. This section uses the shielding analysis provided in Chapter 5, the operations procedures provided in Chapter 9 and the experience from the loading of many MPCs to develop a realistic estimate of the occupational dose.

The dose rates from the HI-STORM FW overpack, MPC lid, HI-TRAC VW, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the fuel loading and unloading operations. No assessment is made with respect to background radiation since background radiation can vary significantly by site.

The estimated occupational dose is governed by three principal parameters, namely:

i. The dose rate emanating from the MPC.
ii. Average duration of human activity in the radiation elevated space.
iii. Relative proximity of humans to the radiation source.

The dose rate accreted by the MPC depends on its contents. Regionalized storage has been made mandatory in the HI-STORM FW MPC to reduce its net radiation output. The duration of required human activity and the required human proximity, on the other hand, are dependent on the training level of the personnel, and user friendliness of ancillary equipment and the quality of fit-up of parts that need to be assembled in the radiation field.

To provide a uniform basis for the dose estimates presented in this chapter, the reference MPC contents data, available HI-TRAC VW weight, etc., are set down in Table 11.3.1.

Using Table 11.3.1 data, the dose data for fuel loading (wet to dry storage) is provided in Table 11.3.2. The dose for the reverse operation (dry to wet storage) is summarized in Table 11.3.3.

For each step in Table 11.3.2, the task description, average number of personnel in direct radiation field, exposure duration in direct radiation field and average dose rate are identified. The relative locations refer to all HI-STORM FW overpacks. The dose rate location points around the transfer cask and overpack were selected based on actual experience in loading HI-STORM 100 Overpacks. Cask operators typically work with workers entering and exiting the immediate cask area. To account for this, an average number of workers and average dose rates are used. The tasks involved in each step presented in Table 11.3.2 are not provided in any specific order.

11.3.1 Estimated Exposures for Loading and Unloading Operations

Exposures estimates presented in Tables 11.3.2 is expected to bound those for unloading operations. This assessment is based on the similarity of many loading versus operations with the elimination of several of the more dose intensive operations (such as weld inspections and leakage testing). Therefore, loading estimates should be viewed as bounding values for the contents considered for unloading operations.

11.3.2 Estimated Exposures for Surveillance and Maintenance

Table 11.3.3 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM FW modules or remote viewing methods instead of performing direct visual observation of the modules. The security surveillances can be performed from outside the ISFSI, and the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area. Although the HI-STORM FW system requires only minimal maintenance during storage (e.g., touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair, hence most of the maintenance is expected to occur outside the actual cask array.

Table 11.3.1 ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE		
	Item	**Value**
1.	**MPC-Contents (MPC-37)[1]**	**45,000 MWD/MTU and 4.5 years**
2.	**Weight of HI-TRAC VW Full of Fuel and Water**	**125 tons**
3.	**HI-STORM Concrete Density**	**150 lb/cubic feet**

[1] The case of MPC-37 is used but similar results are expected for the MPC-89.

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
Fuel loading and removal of the transfer cask and MPC from the spent fuel pool (includes: fuel loading, fuel assembly identification check, MPC lid installation, Lift Yoke attachment to the HI-TRAC VW, HI-TRAC VW removal from the spent fuel pool, preliminary decontamination, HI-TRAC VW movement to the DAS, Lift Yoke removal and decontamination. Background radiation of 1 mrem/hr assumed.	3	800	1.0	40.0
MPC preparation for closure (includes: HI-TRAC VW and MPC decontamination, radiation surveys, partial MPC pump down, annulus seal removal, partial lowering of annulus water level, annulus shield ring installation, weld system installation); workers assumed to be on scaffolding near the top of the HI-TRAC.	3	30	55.7	83.5
MPC Closure (includes MPC lid to shell welding, weld inspection). Assumes welding machine uses standard Holtec pedestal which provides additional shielding. Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	2	185	55.7	34.3

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
MPC Preparation for Storage (includes: MPC hydrostatic testing, draining, drying and backfill, vent and drain port cover plate installation, welding, weld inspection and leakage testing). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 20% of the total duration.	2	170	175.4	198.7
MPC Closure Ring Installation (includes: closure ring to MPC shell welding, weld inspection and leakage testing of the MPC primary closure). Holtec auxiliary shielding methods and equipment assumed (lead blankets, water shields, etc.) Assumes operators are present for 10% of the total duration.	2	80	229.4	61.2
HI-STORM FW system preparation for receiving MPC (includes: HI-STORM FW overpack positioning at transfer location, HI-STORM lid removal, Mating Device installation on HI-STORM FW overpack).	3	160	0	0

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
MPC Transfer (attachment of MPC lifting device, movement of HI-TRAC VW to transfer location, placement of HI-TRAC VW in Mating Device, bottom lid removal, MPC lowering, HI-TRAC VW removal, MPC lift device removal). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	3	120	148	88.8
HI-STORM FW overpack movement to the ISFSI (will include: movement of the HI-STORM FW overpack from the fuel building to placement of the HI-STORM FW overpack on the ISFSI pad, disconnecting transporter, attachment of HI-STORM FW lid, attachment of thermal monitoring system). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 50% of the total duration.	3	220	37.3	205.2
TOTAL EXPOSURE (person-mrem)	711.6			

Table 11.3.3

ESTIMATED EXPOSURES FOR HI-STORM FW SURVEILLANCE AND MAINTENANCE

ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 11.3.2, 11.3.3, AND 11.3.4:

1. Refer to Chapter 9 for detailed description of activities.
2 Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.

11.4 ESTIMATED CONTROLLED AREA BOUNDARY DOSE ASSESSMENT

11.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [11.0.1] 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 FW 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 [11.0.1]. 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 5.1.3 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 [11.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this is the minimum distance analyzed in Chapter 5. One hundred percent (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. Table 5.1.3 and Figure 5.1.3 in Chapter 5 show the annual dose rates for an array of HI-STORM FWs with MPC-37. These results are presented only as an illustration to demonstrate that the HI-STORM FW system is in compliance with 10CFR72.104 [11.0.1]. 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 [11.0.1] contributors and 10CFR20 [11.1.1].

Chapter 7 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the Confinement Boundary has been rendered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

11.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Chapter 12, 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 VW) do not result in the degradation of the HI-STORM FW

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.

11.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [11.0.1] specifies the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 11.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 12 presents the results of the evaluations performed to demonstrate that the HI-STORM FW system can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [11.0.1]. The accident events addressed in Chapter 12 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, and blockage of MPC basket air inlets.

The worst-case shielding consequence of the accidents evaluated in Chapter 12 for the loaded HI-STORM FW overpack shows that as a result of a fire, all of the concrete remains below the short term temperature limit and therefore there is no adverse effect on shielding or the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Chapter 12 for the loaded HI-TRAC VW transfer cask assumes that as a result of a fire, tornado missile, or handling accident, that all the water in the water jacket is lost. The shielding analysis of the HI-TRAC VW with complete loss of the water from the water jacket is discussed in Subsection 5.1.1. The results in that subsection show the resultant dose rate at the 100-meter controlled area boundary during the accident condition. At the calculated dose rate, Table 5.1.9 shows the calculated time to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 12. Therefore, the dose requirement of 10CFR72.106 [11.0.1] is satisfied. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 11.4.1
(Intentionally Deleted)

11.5 REFERENCES

[11.0.1] *U.S. Code of Federal Regulations,* Title 10, "Energy" Part 72 "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[11.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989

[11.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.

[11.1.1] *U.S. Code of Federal Regulations,* Title 10, "Energy" Part 20 "Standards for Protection Against Radiation."

[11.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.

[11.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May 1997.

CHAPTER 12†: ACCIDENT ANALYSIS

12.0 INTRODUCTION

This chapter presents the evaluation of the HI-STORM FW System for the effects of off-normal and postulated accident conditions; and other scenarios that warrant safety analysis (such as MPC reflood during fuel unloading operations), pursuant to the guidelines in NUREG-1536. The design basis off-normal and postulated accident events, including those based on non-mechanistic postulation as well as those caused by natural phenomena, are identified. For each postulated event, the event cause, means of detection, consequences, and corrective actions are discussed and evaluated. For other miscellaneous events (i.e., those not categorized as either design basis off-normal or accident condition events), a similar outline for safety analysis is followed. As applicable, the evaluation of consequences includes the impact on the structural, thermal, shielding, criticality, confinement, and radiation protection performance of the HI-STORM FW System due to each postulated event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM FW System under the short-term operations and various conditions of storage are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and analyses reported therein.

Chapter 12 is in full compliance with NUREG-1536; no exceptions are taken.

† 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.

12.1 OFF-NORMAL CONDITIONS

Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions, which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are described in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM FW:

1. Off-Normal Pressure
2. Off-Normal Environmental Temperatures
3. Leakage of One Seal
4. Partial Blockage of Air Vents
5. Malfunction of FHD System

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of off-normal events and remain in compliance with the applicable acceptance criteria. The following subsections present the evaluation of the HI-STORM FW System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses meet the requirements of 10CFR72.104(a) and 10CFR20, with appropriate margins.

12.1.1 Off-Normal Pressure

The sole pressure boundary in the HI-STORM FW System is the MPC enclosure vessel. The off-normal pressure condition is specified in Subsection 2.2.2. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the MPC cavity under normal storage. The MPC internal pressure under the off-normal condition is evaluated with 10% of the fuel rods ruptured and with 100% of ruptured rods fill gas and 30% of ruptured rods fission gases released to the cavity.

12.1.1.1 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 extremely low. Nonetheless, the event is postulated and evaluated.

12.1.1.2 Detection of Off-Normal Pressure

The HI-STORM FW 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 or safety imperative for detection of off-normal pressure and, therefore, no monitoring is required.

12.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off-normal internal pressure is reported in Subsection 4.6.1 for the following conditions: limiting fuel storage scenario, tech. spec. maximum helium backfill pressure with a 10% rod rupture that causes a 100% of the ruptured rod fill gas and 30% of the ruptured rod gaseous fission products released into the MPC cavity along with off-normal ambient temperature. The analysis shows that the MPC pressure remains below the design MPC internal pressure (given in Table 2.2.1). The corresponding fuel cladding temperature is provided in Table 4.6.1. It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would reduce the temperature rise, therefore the calculated pressure is higher than that would actually occur.

i. Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

ii. Thermal

The MPC internal pressure for off-normal conditions is reported in Subsection 4.6.1. The design basis internal pressure used in the structural evaluation (Table 2.2.1) bounds the off-normal condition pressure.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. 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 pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. 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 FW System.

12.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM FW System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. Therefore, there is no corrective action requirement for off-normal pressure.

12.1.1.5 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.

12.1.2 Off-Normal Environmental Temperatures

The HI-STORM FW System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (loaded HI-STORM FW overpack) and 0°F to 100°F (loaded HI-TRAC VW transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM FW System and must be evaluated against the allowable component design temperatures. The off-normal temperatures are evaluated against the off-normal condition temperature limits for HI-STORM FW components listed in Table 2.2.3.

12.1.2.1 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 FW System to achieve thermal equilibrium. Because of the large mass of the HI-STORM FW System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

12.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM FW 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 FW overpack and MPC. Chapter 2 provides operational limitations on the use of the HI-TRAC VW transfer cask at temperatures ≤32°F and prohibits use of the HI-TRAC VW transfer cask below 0°F.

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

The off-normal event considers an environmental temperature of 100°F with insolation for sufficient duration to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The Off-Normal ambient temperature condition is evaluated in Subsection 4.6.1. The results are in compliance with off-normal pressure and temperature limits in Tables 2.2.1 and 2.2.3, respectively.

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a sufficient duration to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM FW overpack. The HI-STORM FW overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC VW transfer cask is 0°F and the HI-TRAC VW is conservatively assumed to reach 0°F throughout the structure. Subsection 3.1.2, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM FW System.

i. Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 4.6.1, the resultant pressure is below the off-normal design pressure (Table 2.2.1). The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits. The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsection 3.1.2.

ii. Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Subsection 4.6.1 for the HI-STORM FW overpack and MPC. The evaluation in Subsection 4.6.1 indicates that all temperatures for the off-normal

environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.2.3.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. 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 pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. 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 FW System.

12.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM FW System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. For ambient temperatures from 0° to 32°F, ethylene glycol fortified water must be used in the water jacket of the HI-TRAC VW transfer cask to prevent freezing. There are no corrective actions required for off-normal environmental temperatures.

12.1.2.5 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.

12.1.3 Leakage of One Seal

The HI-STORM FW System has a high integrity welded boundary to contain radioactive fission products within the Confinement Boundary. The Confinement Boundary is defined by the MPC shell, baseplate, MPC lid, vent and drain port cover plates, closure ring, and associated welds. The closure ring provides a redundant welded closure to further protect against 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. The fabrication shop welds for the confinement boundary are tested for helium leakage. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. The welds on the vent and drain port cover plates are helium leakage tested. Additionally, the MPC lid weld is subjected to a pressure test to verify its integrity. There are no seals present in the design of the MPC confinement boundary.

Section 7.1 provides the narrative that demonstrates that the MPC design, welding, testing and inspection meet the requirements such that leakage from the Confinement Boundary is considered non-credible.

12.1.4 Partial Blockage of Air Vents

The HI-STORM FW System is designed with debris screens on the inlet and outlet air openings. These screens ensure the openings are protected from the incursion of foreign objects. There are multiple inlet and outlet openings and an axisymmetric outlet and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet and outlet openings. As required by the design criteria presented in Chapter 2, it is conservatively assumed that 50% of the air inlet openings are completely blocked. The scenario described in Chapter 4 of the partial blockage of air vents is evaluated with a normal 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 thermal performance of the HI-STORM FW System during this event.

12.1.4.1 Postulated Cause of Partial Blockage of Air Vents

The presence of screens prevents foreign objects from entering the openings and the screens are either inspected periodically or the outlet air temperature is monitored per the technical specifications. It is, however, possible that blowing debris may partially block the inlet and/or outlet openings for a short time until the openings are cleared of debris.

12.1.4.2 Detection of Partial Blockage of Air Vents

The detection of the partial blockage of air inlet and/or outlet openings will occur during the routine visual inspection of the screens or temperature monitoring of the outlet air required by the technical

specifications. The frequency of inspection is based on an assumed complete blockage of all air inlet openings. There is no inspection requirement as a result of the postulated partial inlet and/or outlet blockage, because the complete blockage of all air inlet openings is bounding.

12.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Vents

i. Structural

There are no structural consequences as a result of this off-normal event since the HI-STORM FW components do not exceed the off-normal temperature limits (Table 2.2.3).

ii. Thermal

The thermal analysis for the 50% blocked inlet openings off-normal condition is performed in Subsection 4.6.1. The analysis demonstrates that under bounding (steady-state) conditions, no system components exceed the off-normal temperature limits in Table 2.2.3. Subsection 4.6.1 also describes the effect of partially blocked outlets on the temperatures.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

vi. 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 openings does not affect the safe operation of the HI-STORM FW System.

12.1.4.4 Corrective Action for Partial Blockage of Air Vents

The corrective action for the partial blockage of air inlet and/or outlet openings 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 vent openings does not affect the safe operation of the HI-STORM FW System.

Periodic inspection of the HI-STORM FW overpack air opening screens is required per the technical specifications. Alternatively, per the technical specifications, the outlet air temperature is monitored. The frequency of inspection is based on an assumed blockage of all air inlet openings analyzed in Section 12.2.

12.1.4.5 Radiological Impact of Partial Blockage of Air Vents

The off-normal event of partial blockage of the air inlet openings has no radiological impact because the confinement barrier is not breached and the system's shielding effectiveness is not diminished.

12.1.5 Malfunction of FHD System

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

12.1.5.1 Postulated Cause of FHD Malfunction

Likely causes of FHD malfunction are (i) a loss of external power to the FHD System and (ii) an active component trips the FHD System. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in helium leakage from the MPC or the FHD.

12.1.5.2 Detection of FHD Malfunction

The FHD System is monitored during its operation. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

12.1.5.3 Analysis of Effects and Consequences of FHD Malfunction

i. Structural

The FHD System is required to be equipped with safety relief devices* to prevent the MPC structural boundary pressures from exceeding the normal condition pressure limits. Consequently there is no adverse effect.

* The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.

ii. Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit (see Table 2.2.3) must not be exceeded. The FHD System malfunction event is evaluated assuming the following bounding conditions:

a. Steady state maximum temperatures have been reached
b. Design maximum heat load in the limiting MPC-37
c. Air in the HI-TRAC VW annulus
d. The helium pressure in the MPC is at the minimum possible value from the technical specification.

The results of a steady state analysis (which implies an extended period of FHD unavailability) are provided in Section 4.6. The results provide the assurance that the peak fuel cladding temperature in the MPC will remain below the ISG-11 limit (see Table 2.2.3) in the event of a prolonged unavailability of the FHD system under the most thermally adverse conditions (highest possible heat load absence of any forced heat removal measures and minimum system helium pressure).

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, the MPC structural boundary internal pressures cannot exceed the normal condition pressure limits, assuring Confinement Boundary integrity.

vi. Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the FHD malfunction does not affect the safe operation of the HI-STORM FW System.

12.1.5.4 Corrective Action for FHD Malfunction

The HI-STORM FW System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently no corrective action is required.

12.1.5.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

12.2 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 FW System or events postulated because their consequences may affect the public health and safety. Subsection 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM FW System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM FW 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. In the following, the evaluation of the design basis postulated accident conditions and natural phenomena is presented. The evaluations 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 2.2.13. The accident load combination evaluations are provided in Section 3.4.

Table 12.2.1 provides a listing of the accident events considered in this section and their probability of occurrence.

12.2.1 HI-TRAC VW Transfer Cask Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-TRAC VW transfer cask is typically lifted and handled in a vertical orientation (except as described in Subsection 4.5.1). A drop of the loaded HI-TRAC VW transfer cask is not a credible accident as the loaded HI-TRAC VW transfer cask shall be lifted and handled in either the vertical or non-vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-TRAC VW transfer cask in the realm of Part 72 operations is non-credible.

12.2.2 HI-STORM FW Overpack Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-STORM FW overpack is lifted and handled in a vertical orientation at all times. A vertical drop of the loaded HI-STORM FW is not a credible accident as the loaded HI-STORM FW shall be lifted and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-STORM FW in the Part 72 space is non-credible.

12.2.3 HI-STORM FW Overpack Non-Mechanistic Tip-Over

The freestanding HI-STORM FW storage overpack, containing a loaded MPC, cannot tip over as a result of postulated natural phenomenon events, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a *non-mechanistic* tip-over scenario per NUREG-1536 is analyzed (Subsection 2.2.3) in Chapter 3.

12.2.3.1 Cause of Tip-Over

The tip-over accident is stipulated as a non-mechanistic accident because a credible mechanism for the cask to tip over cannot be identified. Detailed discussions are provided in Subsections 3.1.2 and 3.4.4.

However, it is recognized that the mechanical loadings at a specific ISFSI may be sufficiently strong to cause a tip-over event, even though such a scenario is determined to be counterfactual under the loads treated in this FSAR. To enable the safety evaluation of a postulated tip-over scenario, it is necessary to set down an analysis methodology and the associated acceptance criteria. In Sections 2.2 and 3.4, the methodology and acceptance criteria are presented and a reference tip-over problem is solved. The reference tip-over problem corresponds to a free rotation of the HI-STORM FW overpack from the condition of rest at the incipient tipping point (i.e., C.G.-over-corner). The evaluations presented below refer to the above non-mechanistic tip-over scenario.

12.2.3.2 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 Subsection 3.4.4. The structural analysis demonstrates the following:

(i) The lateral plastic deformation of the basket panels in the active fuel region is less than the limiting value in Table 2.2.11.
(ii) The impact between the MPC guide tubes and the MPC does not cause a thru-wall penetration of the MPC shell.

The side impact will cause some localized damage to the concrete and outer shell of the overpack in the local area of impact. However, there is no significant adverse effect on the structural, confinement, thermal, or criticality performance.

As mentioned earlier the non-mechanistic tip-over accident has been addressed to demonstrate the defense-in-depth features of the design.

12.2.3.3 Tip-over Accident Corrective Actions

Corrective action after a tip-over would include a radiological and visual inspection to determine the extent of the damage to the overpack and the contained MPC. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

12.2.4 Fire

12.2.4.1 Cause of Fire

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

12.2.4.2 Fire Analysis

The HI-STORM FW System must withstand elevated temperatures due to a fire event. The HI-STORM FW overpack and HI-TRAC VW transfer cask fire accidents for storage are conservatively postulated as described in Subsection 4.6.2. The acceptance criteria for the fire accident are provided in Subsection 2.2.3.

12.2.4.2.1 Fire Analysis for HI-STORM FW Overpack

The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2. The transport vehicle fuel tank fire has been analyzed to evaluate the outer layers of the storage overpack heated by the incident thermal radiation and forced convection heat fluxes and to evaluate fuel cladding and MPC temperatures.

i. Structural

As discussed in Section 3.4, there are no structural consequences as a result of the fire accident condition since the short-term temperature limit of the concrete is not exceeded and all component temperatures remain within applicable temperature limits (Table 2.2.3). The MPC structural boundary remains within normal condition internal pressure and temperature limits.

ii. Thermal

Based on a conservative analysis discussed in Subsection 4.6.2, of the HI-STORM FW System response to the 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 FW System to maintain cooling of the spent nuclear fuel within temperature limits (Table 2.2.3) during and after fire is not compromised.

iii. Shielding

All the concrete remains below its short-term temperature limit. There is no adverse effect on the shielding function of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event since the structural integrity of the confinement boundary is unaffected.

vi. Radiation Protection

Since there is minimal reduction, if any, 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 FW System.

12.2.4.2.2 Fire Analysis for HI-TRAC VW 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 HI-TRAC VW transfer cask is performed. The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2.

i. Structural

As discussed in Section 3.4, there are no adverse structural consequences as a result of the fire accident condition.

ii. Thermal

The thermal analysis of the MPC in the HI-TRAC VW transfer cask under a fire accident is performed in Subsection 4.6.2. The analysis shows that the MPC internal pressure and fuel temperature increases as a result of the fire accident. The fire accident MPC internal pressure and peak fuel cladding temperature are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).

As can be concluded from the analysis, the temperatures for fuel cladding and components are below the accident temperature limits.

iii. Shielding

The conservatively 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 shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10CFR72.106 are not exceeded.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. Increases in the local dose rates adjacent to the water jacket are evaluated in Chapter 5. Immediately after the fire accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit exposure to site personnel.

12.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC VW transfer cask neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC VW transfer cask in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC VW transfer cask following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The high temperatures experienced by the HI-STORM FW overpack concrete are limited to the outermost layer and remain below the short term temperature limit. Therefore, there is no overall reduction in neutron shielding capabilities. . The loaded HI-STORM FW 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.

12.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC VW transfer cask or HI-STORM FW overpack, the ISFSI owner shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC VW transfer cask as the water jacket rupture discs may open with resulting water loss and increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC VW transfer cask. If damage to the HI-TRAC VW transfer cask is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced. If damage to the HI-TRAC VW transfer cask is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

If damage to the HI-STORM FW storage overpack as the result of a fire event is widespread and/or as radiological conditions require (based on dose rate measurements), the MPC shall be removed from the HI-STORM FW overpack in accordance with Chapter 9. However, the thermal analysis described herein demonstrates that the radial concrete which is behind the carbon steel outer shell remains below its design temperature. The HI-STORM FW overpack may be returned to service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the shielding effectiveness of the overpack is confirmed) and if the visual inspection is satisfactory.

12.2.5 Partial Blockage of MPC Basket Flow Holes

Each MPC basket fuel cell wall has flow holes near the bottom to allow thermosiphon action to assist the cooling of MPC internals. The flow holes in the bottom of the fuel basket in each MPC are located to ensure that the amount of crud listed in Table 2.2.8 does not block the internal helium circulation. Therefore the partial blockage of the HI-STORM FW MPC basket flow holes is not credible.

12.2.6 Tornado

12.2.6.1 Cause of Tornado

The HI-STORM FW System will be stored on an unsheltered ISFSI concrete pad and thus will be subject to ambient environmental conditions throughout the storage period. Additionally, the transfer of the MPC between the HI-TRAC VW transfer cask and the storage overpack may be performed at the unsheltered ISFSI concrete pad. It is therefore possible that the HI-STORM FW System (and/or the HI-TRAC VW transfer cask) may experience the extreme environmental conditions, resulting in the impact from a tornado-borne projectile.

12.2.6.2 Tornado Analysis

A tornado event is characterized by high wind velocities and tornado-generated missiles. The reference missiles considered in this FSAR (see Section 2.2) are of three sizes: small, medium, and large. A small projectile, upon collision with a cask, would tend to penetrate it. A large projectile, such as an automobile, on the other hand, would tend to destabilize a free-standing cask. Accordingly, the tornado event has two distinct effects on the HI-STORM FW System. First, the tornado winds and/or tornado missile attempt to tip-over the loaded HI-STORM FW overpack or HI-TRAC VW 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 HI-STORM FW overpack or HI-TRAC VW transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC VW transfer cask shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC VW transfer cask while it is being handled. Penetration by a small missile, however, is credible. The tornado wind and missile are assumed to act synergistically in the safety evaluation in Section 3.4 to determine the kinematic stability of the HI-STORM FW overpack.

i. Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the HI-STORM FW 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 large tornado missiles.

Analyses provided in Section 3.4 also show that there is a potential for a tornado missile (8 inch steel cylinder) to penetrate the water jacket of the HI-TRAC VW transfer cask. The HI-STORM FW overpack will suffer minor local damages due to the missile impact with no significant damage in the shielding and there will be no damage to the MPC.

ii. Thermal

The thermal consequences of the complete loss of water due to rupture of the water jacket from a tornado missile has been analyzed in Section 4.6. It has been demonstrated that the consequences are within the short term fuel cladding and material temperature limits.

iii. Shielding

Since the structural evaluation shows that the tornado missiles may penetrate the HI-TRAC VW water jacket and cause loss of water, for a conservative estimate of the dose rates a complete loss of water in the water jacket is assumed and is bounded by the fire condition

assumptions. This assumption results in an increase in the radiation dose rates however the shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

HI-STORM FW overpack: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Since there is only a possibility of minimal reduction in localized shielding and there is 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.

HI-TRAC VW transfer cask: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Increases in the local dose rates as a result of the possible loss of water in the HI-TRAC VW transfer cask water jacket are evaluated in Chapter 5. Immediately after the tornado accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit the exposure to the site personnel.

12.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded HI-STORM FW overpack; damage the shielding materials of the HI-STORM FW overpack or HI-TRAC VW transfer cask; 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 HI-STORM FW overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC VW transfer cask water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC VW 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.

12.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM FW System to a tornado, the ISFSI owner shall perform a visual and radiological inspection of the overpack and/or HI-TRAC VW transfer cask.

Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and may be repaired, if required, while in-service. The HI-STORM FW overpack may continue its service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the final visual inspection is satisfactory.

Damage sustained by the HI-TRAC VW transfer cask shall be inspected and repaired. As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. If damage to the HI-TRAC VW transfer cask water jacket or HI-TRAC VW transfer cask body is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

12.2.7 Flood

12.2.7.1 Cause of Flood

Many ISFSIs are located in flood plains susceptible to floods. Therefore, it is necessary for such ISFSIs to define a Design Basis Flood (DBF). 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.

A flood event is characterized by two parameters:

a. flood water velocity
b. flood height over the ISFSI pad as a function of time

The design basis flood (DBF) event for an ISFSI site should provide the maximum flood water velocity. The highest flood height, on the other hand, is not the governing condition for the flood event because of the vented construction of the overpack. The bottom vents in the HI-STORM FW overpack ensure that the flood level inside and outside the overpack will be equal. When the flood waters are high and the MPC is fully submerged then there is no short-term threat to the storage system because the MPC's heat rejection to water is far more efficient than the (normal condition) heat rejection to air. The most adverse flood condition, therefore, exists when the flood waters are high enough to block the inlet openings but no higher. In this scenario, the MPC surface has minimum submergence in water and the ventilation air is completely blocked. In fact, as the flood water begins to accumulate on the ISFSI pad, the air passage size in the inlet vents begins to get smaller. Therefore, the rate of floodwater rise with time is necessary to analyze the thermal-hydraulic problem. For the reference design basis flood (DBF) analysis in this FSAR, the flood waters are assumed to rise instantaneously to the height to block the inlet vents and stay at that precise elevation for 32 hours. For each ISFSI site subject to a DBF, the flood time-history will be incorporated in determining the acceptability of the flood event. The acceptance criteria are provided in Section 2.2.

The analysis results for the reference DBF are presented below.

12.2.7.2 Flood Analysis

i. Structural

The flood accident affects the HI-STORM FW overpack structure in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood accident affects the MPC by applying an external pressure.

Section 3.4 provides the analysis of the flowing floodwater 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.

ii. Thermal

As stated above, for a flood of sufficient height to allow the water to come into extensive 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 submerged flood will enter the annulus between the MPC and the overpack. The water would provide cooling that would be an order of magnitude greater than that available in the air filled annulus (due to water's higher heat transfer coefficient).

The reference DBF that is most adverse to heat rejection is treated for 100% blockage of air inlets in Subsection 12.2.13. If the duration of the flood blockage exceeds the DBF blockage specified in Subsection 4.6.2 then a site specific evaluation shall be performed in accordance with the methodology presented in Chapter 4 and evaluated for compliance with Subsection 2.2.3 criteria.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water provides additional shielding that reduces radiation doses.

iv. 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.

v. 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.

vi. 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 FW System.

12.2.7.3 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.

12.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM FW System sustains no damage as a result of the flood, which is a short-duration event. At the completion of the flood, exposed surfaces may need debris and adherent foreign matter removal.

12.2.8 Earthquake

12.2.8.1 Cause of Earthquake

The HI-STORM FW 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 FW System, the ISFSI may experience an earthquake. The earthquake event postulated for the ISFSI is referred to as the "Design Basis Earthquake" (DBE). The DBE is defined in Subsection 2.2.3.

12.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM FW overpack based on a static stability criteria discussed in Subsection 2.2.3.

Some ISFSI sites will have earthquakes that exceed the static stability limit specified in Subsection 2.2.3. For these high-seismic sites, a dynamic analysis shall be performed based on the methodology provided in Subsection 3.4.4.

i. Structural

The methodology for the evaluation of the earthquake consequences has been presented in Subsection 3.4.4. An earthquake is a vibratory event, which is fully described by an acceleration time-history. However, for "weak" earthquakes, a static equilibrium based calculation suffices. However, for stronger DBEs, a dynamic analysis is required. The results from the reference DBE analyzed in Subsection 3.4.4 are used to evaluate the safety case for the earthquake using the acceptance criteria in Section 2.2.

The following conclusions can be reached from the structural analysis of the HI-STORM FW System under the reference DBE event:

a. The MPC Confinement Boundary remains unbreached.
b. The HI-STORM FW overpack structure remains intact; i.e., the lid is not displaced.
c. There is no physical damage to the HI-STORM FW overpack shielding concrete.
d. The HI-STORM FW overpack does not tip over.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 reference DBE does not affect the continued safe operation of the HI-STORM FW System.

12.2.8.3 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. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates at the site boundary.

12.2.8.4 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 displaced from their installed position or tipped over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 12.2.3.

12.2.9 100% Fuel Rod Rupture

This accident event postulates the non-mechanistic condition that all the fuel rods rupture and that the quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity consistent with ISG-5, Revision 1.

12.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM FW 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. Therefore, there is no credible cause for 100% fuel rod rupture. This accident is presumably postulated in NUREG-1536 to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

12.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no containment consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, the shielding capability of the system, or the criticality control features of the fuel basket; and does not challenge the structural integrity of the MPC.

i. Structural

The structural analysis provided in Chapter 3 evaluates the MPC Confinement Boundary under the accident condition design internal pressure limit set in Table 2.2.1. Calculations in Chapter 4 show that the accident internal pressure limit bounds the pressure from 100% fuel rod rupture.

ii. Thermal

The determination of the maximum accident pressure is provided in Chapter 4. The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.5, which is bounded by the design basis accident condition MPC internal pressure limit set in Table 2.2.1.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 event does not affect the safe operation of the HI-STORM FW System.

12.2.9.3 100% Fuel Rod Rupture Dose Calculations

The breach of fuel cladding postulated in this accident event does not result in any physical change to the storage system other than some release of gases and a limited quantity of solids (particulates) into the gaseous helium space. The amount of the radiation source remains unaffected. Hence, the radiation dose at the site boundary will not change perceptibly, i.e., there are no consequences to the site boundary dose.

12.2.9.4 100% 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 FW 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.

12.2.10 Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified would cause failure of the confinement boundary. The MPC uses redundant confinement closures to assure that there is no release of radioactive materials. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The information contained in Chapter 7 demonstrates that MPC is designed, fabricated, tested and inspected to meet the guidance of ISG-18 such that unacceptable leakage from the Confinement Boundary is non-credible.

12.2.11 Explosion

12.2.11.1 Cause of Explosion

An explosion within the protected area of an ISFSI is improbable since there are no explosive materials permitted within the site boundary. However, an explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity the effects of an explosion on a reinforced structure are minimal. Explosions that are credible for a specific ISFSI would require a site hazards evaluation under the provisions of 72.212 regulations by the ISFSI owner using the methodology set forth in Section 3.1.

12.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external design pressure (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 12.2.7 and the tornado missile accident of Subsection 12.2.6, because explosive materials are not stored within close proximity to the casks. The bounding analysis shows that the MPC and the overpack can withstand the effects of substantial accident external pressures without collapse or rupture.

An ISFSI where a credible explosion event produces a pressure wave greater than analyzed in Subsection 12.2.7 or an impactive load greater than considered in Subsection 12.2.6, shall be evaluated within the purview of §72.212. The results of the safety evaluation of the postulated limiting explosion in this subsection are as follows:

i. 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.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. 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 reference explosion accident does not affect the safe operation of the HI-STORM FW System.

12.2.11.3 Explosion Dose Calculations

The reference bounding external pressure load has no effect on the HI-STORM FW Overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM FW System is experienced as a result of the explosion pressure load. The effects of explosion generated (reference) missiles on the HI-STORM FW System structure is bounded by the analysis of tornado generated missiles.

12.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by an 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 FW System. Following an explosion, the ISFSI owner 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 overpack will be repaired as necessary.

12.2.12 Lightning

12.2.12.1 Cause of Lightning

As the HI-STORM FW 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.

12.2.12.2 Lightning Analysis

The HI-STORM FW 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 FW overpack is struck 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, provides a direct path to the ground through the optional grounding cable.

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.

i. Structural

There is no structural consequence as a result of this event.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 FW System.

12.2.12.3 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.

12.2.12.4 Lightning Accident Corrective Action

The HI-STORM FW System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

12.2.13 100% Blockage of Air Vents

12.2.13.1 Cause of 100% Blockage of Air Vents

This event is defined as a complete blockage of all vents. A blockage of all of the circumferentially arrayed vents cannot be realistically postulated to occur at most sites. However, a flood, blizzard snow accumulation, tornado debris, or volcanic activity, where applicable, can cause a significant blockage.

12.2.13.2 100% Blockage of Air Vents Analysis

The immediate consequence of a complete blockage of the air inlet and/or outlet openings 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 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 HI-STORM FW 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 HI-STORM FW overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for an extended duration. This accident condition is, however, a short duration event that will be identified by the ISFSI staff, at worst, during scheduled periodic surveillance at the ISFSI site and corrected using the site's emergency response process.

i. Structural

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

ii. Thermal

A thermal analysis is performed in Subsection 4.6.2 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and component temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is evaluated in Subsection 4.6.2 and is bounded by the design basis internal pressure for accident conditions (Table 2.2.1). As described in Subsection 4.6.2, the analysis of 100% blockage of the inlet vents bounds 100% blockage of the outlet vents. Additionally as described in Subsection 4.6.2, in the unlikely event where both the inlet and outlet vents are blocked, the temperatures remain below their short-term temperature limits.

iii. 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.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 the above evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the blockage is the first priority activity.

12.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets and/or outlets accident, the shielding capabilities of the HI-STORM FW overpack 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.

12.2.13.4 100% Blockage of Air Vents Accident Corrective Action

Analysis of the 100% blockage of air inlet and/or outlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within the maximum elapsed period between scheduled surveillance inspections. Upon detection of the complete blockage of the air inlet and/or outlet openings, the ISFSI owner shall activate its emergency response procedure to remove the blockage with mechanical and manual means as necessary. After clearing the overpack openings, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for the assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air openings for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to whether adequate heat removal for the duration of the event would occur. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain below their short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet openings 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 opening using pumps or fire-hoses or blowing air into the air outlet opening, to directly cool the MPC.

12.2.14 Burial Under Debris

12.2.14.1 Cause of Burial Under Debris

Complete burial of the HI-STORM FW System under debris is not a credible accident. During storage at the ISFSI, there are no structures above the casks that may collapse and surround them. The minimum regulatory distance(s) from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM FW System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM FW overpack air inlet openings has already been considered in Subsection 12.2.12.

12.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM FW 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. A perverse effect of the overlaid debris would be to provide additional shielding to reduce radiation doses. The accident external pressure considered in this FSAR 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 FW 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, assuming that the debris has the consistency of a typical pile of rocks (pebbles).

i. Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions set in Table 2.1.1 bounds the pressure calculated for this event. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

ii. Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised even if the burial event lasts for a substantial duration.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no deleterious effect on the site boundary dose a result of this event.

Based on the above evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the debris is the first priority activity.

12.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM FW 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 adverse radiological impact.

12.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures are not exceeded even for an extended duration of burial. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC VW 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 implementation of this corrective action.

12.2.15 Extreme Environmental Temperature

12.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated (see Table 2.2.2) as a 3-day average temperature caused by extreme weather conditions.

12.2.15.2 Extreme Environmental Temperature Analysis

To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration (3 days) to allow the HI-STORM FW overpack to achieve thermal equilibrium.

The accident condition considering an extreme environmental temperature (Table 2.2.2) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3.

i. 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.

ii. Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in evaluation performed in Subsection 4.6.2. As concluded from this evaluation, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

iii. 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.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. 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 FW System.

12.2.15.3 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

FW System for the extreme environmental temperature and the dose calculations are to the same as those for normal condition dose rates.

12.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

Table 12.2.1

ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	Event	Probability of Occurrence	Subsection Where Addressed	Comments
1.	HI-TRAC VW Transfer Cask Handling Accident	Non-Credible	12.2.1	This FSAR mandates the use of high integrity handling equipment and single-failure-proof lifting devices for handling loaded HI-TRAC VWs within the Part 72 jurisdictional boundary.
2.	HI-STORM FW Overpack Handling Accident	Non-Credible	12.2.2	This FSAR mandates the use of high integrity handling equipment and single-failure-proof lifting devices for handling loaded HI-STORM FWs within the Part 72 jurisdictional boundary.
3.	HI-STORM FW Non-Mechanistic Tip-Over	Non-Credible	12.2.3	The HI-STORM FW tip-over event is more properly referred to as a "non-mechanistic" tip-over, meaning that no physical loading considered in this FSAR leads to a tip-over event.
4.	Fire	Very small probability but credible	12.2.4	Although there are no ignition sources in the ISFSI area, combustible material (motive fuel) is present. Therefore, the potential of a fire event cannot be ruled out categorically.
5.	Partial Blockage of MPC Basket Vent Holes	Non-Credible	12.2.5	An impactive event may jolt the stored fuel and cause its crud to fall off. However, as explained in Subsection 12.2.5, there is no realistic mechanism for the blockage of the flow holes.
6.	Tornado	Credible	12.2.6	Because a HI-STORM FW ISFSI can be deployed in any state within the U.S., the potential of a tornado event at a generic ISFSI must be considered.
7.	Flood	Credible	12.2.7	Flood, like tornado, must be categorized as a credible event at a generic ISFSI site.

Table 12.2.1

ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	Event	Probability of Occurrence	Subsection Where Addressed	Comments
8.	**Earthquake**	**Credible**	**12.2.8**	**The Design Basis Earthquake for an ISFSI is a specified event for a nuclear facility.**
9.	**100% Fuel Rod Rupture**	**Non-Credible**	**12.2.9**	**"100% rod rupture" is a non-mechanistic event; no specific loading event has been identified to cause 100% rod rupture.**
10.	**Confinement Boundary Leakage**	**Non-Credible**	**12.2.10**	**The Confinement Boundary has been determined to be invulnerable to leakage in Chapter 7.**
11.	**Explosion**	**Credible**	**12.2.11**	**Explosion of gasoline is a credible event at an ISFSI.**
12.	**Lightning**	**Credible**	**12.2.12**	**Lightning is a small probability event at any ISFSI, hence, it cannot be deemed non-credible.**
13.	**100% Blockage of Air Inlets**	**Non-Credible**	**12.2.13**	**Because the air openings are along the circumference of the cask, and surveillance is at very short intervals (see Technical Specification), the assumption of blockage of all openings has no mechanistic basis.**
14.	**Burial Under Debris**	**Credible**	**12.2.14**	**Burial of a loaded system under a debris cannot be generically ruled out because a nuclear plant site may (ever so minimally) susceptible to a large adverse environment event such as a tsunami or an avalanche.**
15.	**Extreme Environmental Temperature**	**Credible**	**12.2.15**	**In certain desert areas in the country a temperature spike that reaches the accident temperature limit (Table 2.2.2) cannot be ruled out. Such areas have not been declared unfit at a nuclear plant site by the USNRC and therefore, must be factored in defining generic accident events.**

12.3 OTHER EVENTS

This section addresses the MPC reflood event which is placed in the category of "other events" since it cannot be categorized as an off-normal or accident event. The MPC reflood event occurs if an ISFSI owner needs to return the fuel in a loaded canister to wet storage in the plant's fuel pool. The MPC reflood event is presented in Subsection 9.4.3 in connection with the preparation for the MPC unloading operation. The reflooding of a loaded MPC with water results in a change to the environment around the fuel from a gaseous (low heat transfer medium) to aqueous (high heat transfer medium). This implies the generation of thermal stresses in the fuel cladding and potential for loss of cladding integrity.

The safety analysis of the reflood event in Subsection 3.4.4 focuses on the effect of strains (due to reflooding) on the integrity of the cladding. This safety analysis, which uses an appropriate thermal/structural model as well as evaluations in Subsection 4.5.5, forms the basis for the instructions in the MPC reflood procedures provided in Subsection 9.4.3. For a complete evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel, the postulated cause of the event, monitoring of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented in this section similar to the systematic evaluations presented for design basis off-normal and postulated accident condition events in the preceding sections.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of MPC reflood and remain in compliance with the applicable acceptance criteria. In particular, the integrity of the fuel cladding shall be preserved. The following subsections contain the evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel that demonstrates that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses continue to meet the requirements of 10CFR72.104(a) and 10CFR20.

12.3.1 MPC Reflood

MPC reflood is performed during the preparation for the unloading operations as described in Subsection 9.4.3. The MPC is flooded with water at a controlled rate as specified in Subsection 4.5.5 with the MPC vent port open such that the generation of steam from flashing of water is not excessive and the pressure within the MPC remains below its normal condition internal design pressure. Although past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by a direct introduction of water into the canister space (as specified in Subsection 9.4.3) a structural evaluation has been performed to ensure fuel cladding integrity and is provided in Paragraph 3.4.4.1.

12.3.1.1 Postulated Cause of MPC Reflood

Likely causes to perform MPC reflood include those associated with required actions for certain limiting conditions for operation (LCO) (as specified in Appendix A of the Technical Specifications)

to unload fuel assemblies from the MPC. A reflood operation may also be carried out at the plant owner's volition to return the fuel to wet storage as a voluntary act.

12.3.1.2 Monitoring of MPC Reflood

MPC reflooding is monitored at frequent intervals by the surveillance of MPC pressure as required by SR 3.1.3.1 to Specification 3.1.3 (Appendix A of the Technical Specifications). An indication of MPC pressurization at or above normal condition MPC internal design pressure established in Section 2.2 requires the immediate action to stop reflooding operations until the MPC cavity pressure is below the required limit. See LCO 3.1.3 and the associated basis in Chapter 13 for more information.

12.3.1.3 Analysis of Effects and Consequences of MPC Reflood

i. Structural

MPC Enclosure Vessel Integrity: The MPC water reflood rate specified in Subsection 4.5.5, the essential reflooding control procedure steps established in Subsection 9.4.3, and the surveillance instructions in SR 3.1.3.1 Specification 3.1.3 (Appendix A of the Technical Specifications) ensure that the MPC is maintained below the normal condition pressure limit and well below MPC off-normal and accident condition pressure limits set down in Section 2.2, thus ensuring large margins of safety and no harmful effect on the MPC enclosure vessel integrity.

Fuel Cladding Integrity: The structural evaluation in Paragraph 3.4.4.1 ensures that the fuel cladding integrity is preserved during the reflood event.

Other Structural Related Considerations: MPC reflooding is performed under a specified maximum water injection rate and below normal condition MPC internal design pressure. The pressures and temperatures are therefore compatible with design limits of existing MPC ancillaries and standard connections such as RVOAs. Maintaining the pressure and temperature parameters well below the design basis values for the ancillaries ensures that failure of components and appurtenances during the reflooding operation is unlikely. Therefore, no credible mechanism for risk to the plant staff or general public from radiological release due to the reflood operation can be identified.

ii. Thermal

A thermal evaluation is provided in Subsection 4.5.5 to specify the maximum water reflood rate. The maximum calculated water reflood rate will prevent MPC over-pressurization and fuel cladding damage.

iii. Shielding

There is no adverse effect on the shielding performance of the system as a result of the MPC reflood event. The shielding performance of the MPC is indeed enhanced by the flooding of its contents.

iv. Criticality

There is no adverse effect on the criticality control of the system as a result of this planned plant event. The essential procedure steps in Chapter 9 and surveillance SR 3.3.1.1 to Specification 3.3.1 (Appendix A of the Technical Specifications) ensure that the water used to reflood the MPC will have the minimum required soluble boron concentration. The generation and escape of steam from the MPC will increase (not lower) the soluble boron concentration.

v. Confinement

During the reflood operation, the MPC confinement function is inoperative (and supplanted by the part 50 facility) as the canister is connected to the plant's fluid accumulation system and the source of water (such as the fuel pool). The reflooding operation, however, does not degrade the confinement capability of the MPC because the internal pressures and temperatures are procedurally controlled to remain well within the design limits.

vi. Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no adverse effect on occupational or public exposures as a result of this MPC reflood event. The vent port steam is delivered to the radwaste gas facility of the plant in accordance with the specified procedure in Subsection 9.4.3.

Based on this evaluation, it is concluded that MPC reflood has no adverse effects or consequences on the safety or operability of the HI-STORM FW System.

12.3.1.4 Corrective Action for MPC Reflood

See Specification 3.1.3 (Appendix A of the Technical Specifications) on MPC Cavity Reflooding and Specification 3.3.1 (Appendix A of the Technical Specifications) on Boron Concentration.

12.3.1.5 Radiological Impact of MPC Reflood

The event has no radiological impact because the plant's confinement barrier and shielding infrastructure are unaffected and the operation relies on no new system for the control of effluents.

12.4 REFERENCES

Currently no references listed.

CHAPTER 13†: OPERATING CONTROLS AND LIMITS

13.0 INTRODUCTION

The HI-STORM FW system provides passive dry storage of spent fuel assemblies in interchangeable MPCs with redundant multi-pass welded closure. The loaded MPC is enclosed in a single-purpose ventilated metal-concrete overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STORM FW system at an ISFSI. The information provided in this chapter is in full compliance with NUREG-1536 [13.1.1].

13.1 PROPOSED OPERATING CONTROLS AND LIMITS

13.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

This portion of the FSAR establishes the commitments regarding the HI-STORM FW system and its use. Other 10CFR72 [13.1.2] and 10CFR20 [13.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [13.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM FW system. The general license conditions governed by 10CFR72 [13.1.2] are not repeated within these Technical Specifications. Licensees are required to comply with all commitments and requirements.

The Technical Specifications provided in Appendix A to the CoC and the authorized contents and design features provided in Appendix B to the CoC are primarily established to maintain subcriticality, the confinement boundary, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 13.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 13.1.2 provides the list of Technical Specifications for the HI-STORM FW system.

† 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 13.1.1

HI-STORM FW SYSTEM CONTROLS

Condition to be Controlled	Applicable Technical Specifications†
Criticality Control	3.3.1 Boron Concentration
Confinement boundary integrity and integrity of cladding on undamaged fuel	3.1.1 Multi-Purpose Canister (MPC)
Shielding and radiological protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 MPC Reflooding 3.2.1 TRANSFER CASK Surface Contamination 5.1 Radioactive Effluent Control Program 5.3 Radiation Protection Program
Heat removal capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural integrity	5.2 Transport Evaluation Program

† Technical Specifications are located in Appendix A to the CoC. Authorized contents are specified in this FSAR in Subsection 2.1.8

Table 13.1.2 HI-STORM FW SYSTEM TECHNICAL SPECIFICATIONS	
NUMBER	**TECHNICAL SPECIFICATION**
1.0	USE AND APPLICATION
1.1	DEFINITIONS
1.2	LOGICAL CONNECTORS
1.3	COMPLETION TIMES
1.4	FREQUENCY
2.0	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SFSC Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Surface Contamination
3.3	SFSC Criticality Control
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINSTRATIVE CONTROLS
5.1	Radioactive Effluent Control Program
5.2	Transport Evaluation Program
5.3	Radiation Protection Program

13.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM FW system to assure long-term performance consistent with the conditions analyzed in this FSAR.

13.2.1 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 FW Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM FW System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important-to-Safety (overview);
4. HI-STORM FW System Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STORM FW Technical Specifications, Approved Contents, Design Features and other Conditions for Use;
8. HI-STORM FW Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Operating Experience Reviews
11. HI-STORM FW 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 staging operation
    - Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, supplemental cooling (if used), 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

13.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM FW system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. Dry run training already performed successfully for the HI-STORM 100 System can be substituted for dry run steps applicable to HI-STORM FW. The dry run shall include, but is not limited to the following:

1. Receipt inspection of the HI-STORM FW System components.

2. Moving the MPC/HI-TRAC into the spent fuel pool.

3. Preparation of the MPC/HI-TRAC for fuel loading.

4. Selection and verification of specific fuel assemblies to ensure conformance.

5. Locating specific assemblies and placing assemblies into the MPC/HI-TRAC (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. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup MPC may be used).

8 Placement of the HI-STORM FW System at the ISFSI.

13.2.3 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 FW 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.

13.2.4 Limiting Conditions for Operation (LCO)

Limiting Conditions for Operation (LCO) specify the minimum capability or level of performance that is required to assure that the HI-STORM FW system can fulfill its safety functions.

13.2.5 Equipment

The HI-STORM FW 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 FW 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.

13.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM FW system fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Subsection 2.1.8 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

13.2.7 Design Features

This subsection describes HI-STORM FW 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 the CoC, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM FW system is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 10.

13.2.8 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.

13.2.9 HI-STORM FW Overpack

a. HI-STORM overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during handling and storage operations.

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

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

13.2.10 HI-TRAC VW Transfer Cask

a. HI-TRAC transfer cask 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-TRAC transfer cask material thermal properties and dimensions for heat transfer control.

c. HI-TRAC transfer cask material composition and dimensions for dose rate control.

13.2.11 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below demonstrate how the user of the system can determine if fuel assemblies, including NFH, are acceptable for loading in either the MPC-37 or the MPC-89 in accordance with the allowable decay heat, burnup, and cooling time for the approved contents.

Example 1

In this example, it will be assumed that the MPC-37 is being loaded with array/class 17x17A fuel in its regionalized loading pattern as shown in Figure 1.2.1 with heat loads from Table 1.2.3.

Table 13.2.1 provides four hypothetical fuel assemblies in the 17x17A array/class that will be evaluated for acceptability for loading in the MPC-37. The decay heat values and the fuel classification in Table 13.2.1 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for storage in Pattern A, Region 2 of MPC-37. Fuel Assembly Number 1 is not acceptable for storage in Pattern A Region 1 or Region 3 because the total heat load of the fuel assembly and the non-fuel hardware exceeds the assembly decay heat limit for those regions. Fuel Assembly Number 1 cannot be stored in Pattern B since the total heat load of the assembly exceeds the assembly decay heat limit for all three regions.

Fuel Assembly Number 2 is not acceptable for loading. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-37 (Figure 2.1.1). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly, for both Pattern A and B. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-37.

Fuel Assembly Number 3 is acceptable for loading in Region 1 or Region 2, both Pattern A and B. The fuel assembly is limited to these locations due to the non-fuel hardware (Figure 2.1.5) and the total heat load of the fuel assembly and non-fuel hardware is less than the assembly decay heat limits for these regions.

Fuel Assembly Number 4 is not acceptable for loading in the MPC-37 because its cooling time is less than the minimum of 3 years. When the fuel assembly attains three years cooling time it can be reevaluated based on the decay heat.

Example 2

In this example, it will be assumed that the MPC-89 is being loaded with array/class 10x10A fuel in its regionalized storage pattern as shown in Figure 1.2.2 with heat loads from Table 1.2.4.

Table 13.2.2 provides four hypothetical fuel assemblies in the 10x10A array/class that will be evaluated for acceptability for loading in the MPC-89. The decay heat values and the fuel classification in Table 13.2.2 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for loading in the MPC-89. Fuel Assembly 1 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit higher than the decay heat of the assembly, therefore the

assembly is acceptable for loading in the MPC-89, but it is limited to the cells depicted in Figure 2.1.2

Fuel Assembly Number 2 is not acceptable for loading in the MPC-89. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-89.

Fuel Assembly Number 3 is acceptable for loading in Regions 1, 2 or 3 of the MPC-89.

Fuel Assembly Number 4 is acceptable for loading in Region 2 of the MPC-89 only. The fuel assembly is limited to these locations due to the total heat load of the fuel assembly.

Table 13.2.1

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 17x17A)

FUEL ASSEMBLY NUMBER	**1**	**2**	**3**	**4**
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	2.9
FUEL ASSEMBLY DECAY HEAT (KW)	1.01	1.45	0.4	2.08
NON-FUEL HARDWARE STORED WITH ASSEMBLY	BPRA	None	NSA	None
NFH DECAY HEAT (KW)	0.5	0	0.3	0
FUEL CLASSIFICATION	Undamaged	Damaged	Undamaged	Undamaged

Table 13.2.2

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 10x10A)

FUEL ASSEMBLY NUMBER	**1**	**2**	**3**	**4**
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	7
FUEL ASSEMBLY DECAY HEAT (KW)	0.43	0.55	0.2	0.61
FUEL CLASSIFICATION	Damaged	Damaged	Undamaged	Undamaged

13.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM FW system are provided in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC. Bases applicable to the Technical Specifications are provided in the FSAR Appendix 13.A. The format and content of the HI-STORM FW system Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, "Writer's Guide for the Restructured Technical Specifications" [13.3.1] was used as a guide in the development of the Technical Specifications and Bases.

13.4 REGULATORY EVALUATION

Table 13.1.2 lists the Technical Specifications for the HI-STORM FW system. The Technical Specifications are detailed in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC.

The conditions for use of the HI-STORM FW system identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical Specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM FW system will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

13.5 REFERENCES

[13.1.1] U.S. Nuclear Regulatory Commission, NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems*, Final Report, January 1997.

[13.1.2] U.S. Code of Federal Regulations, Title 10, *Energy*, Part 72, *Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*."

[13.1.3] U.S. Code of Federal Regulations, Title 10, *Energy*, Part 20, *Standards for Protection Against Radiation*."

[13.3.1] Nuclear Management and Resources Council, Inc. – *Writer's Guide for the Restructured Technical Specifications,* NUMARC 93-03, February 1993.