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

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

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

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

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

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

Undamaged Fuel Assembly is defined as a) fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial 3.3 wt% U-235, without known or suspected GROSSLY BREACHED SPENT FUEL RODS, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

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

ZPA is an acronym for zero period acceleration.

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

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

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

1.2.1.4.1 Neutron Absorber – Metamic HT

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

[Withheld in Accordance with 10 CFR 2.390]

[Withheld in Accordance with 10 CFR 2.390]

[Withheld in Accordance with 10 CFR 2.390]

[Withheld in Accordance with 10 CFR 2.390]

mechanistic tipover event. To satisfy this criterion, the primary membrane stresses in the lid components are compared against the material yield strength. The most heavily loaded component is the upper shim plate closest to the point of impact (Figure 3.4.21). In order to determine the primary membrane stress in the upper shim plate, the stresses are linearized along a path that follows the outside vertical edge of the upper shim plate (see Figure 3.4.21 for path definition). Figure 3.4.22 shows the linearized stress results. Since the membrane stress is less than the yield strength of the material at 300ºF (Table 3.3.6), it is concluded that the lid will not suffer any gross loss of shielding as a result of the non-mechanistic tipover event. The complete details of the lid tipover analysis are provided in [3.4.13].

Finally, to evaluate the potential for crack propagation and growth for the MPC fuel baskets under the non-mechanistic tipover event, a boundingconservative crack propagation analysis is carried out for both MPC-37 and MPC-89all of the fuel baskets using the same methodology utilized in Attachment D of [1.2.6] to evaluate the HI-STAR 180 F-37 fuel basket in support of the HI-STAR 180 SAR [3.1.10]. The crack propagation analysis demonstrates that a through-thickness linear flaw measuring 1/32 inches in length (i.e., maximum undetectable flaw size per inspection criteria) remains stable under the most severe accident loading conditions.is bounding since the maximum tensile strength of the basket material (28.2 ksi) documented in Table 1.2.8 is conservatively considered as the maximum tensile stress experienced by the Metamic fuel baskets in the tip-over accident and used as input to the following crack propagation analysis.

Per [1.2.6] the critical stress intensity factor of Metamic-HT panels is estimated to be

$$K_{IC} = 30 ksi\sqrt{in}$$

based on Charpy V-notch absorbed energy (CVE) correlations for steels. The estimated value is consistent with the range for aluminum alloys, which is 20 to 50 $MPa\sqrt{m}$ or 18.2 to 45 $ksi\sqrt{in}$ per Table 3 of [3.4.19]. Next the minimum crack size, a_{min}, for crack propagation to occur is calculated below using the formula for a through-thickness edge crack given in [3.1.5]. Although the formula is derived for a straight edge specimen, the use of the maximum tensile strength of the fuel basket material as the maximum tensile stress experienced by the basket well compensates for the geometric difference between the basket panel and the specimen. Moreover, the maximum size of a pre-existing crack (1/16") in the fuel basket panel is less than 1/59th of the minimum basket panel thickness (0.5935"). Thus, the assumption of a through-thickness edge crack is very conservative. The result is

$$a_{min} = \frac{\left(\frac{K_{IC}}{1.12\sigma_{max}}\right)^2}{\pi} = \frac{\left[\frac{30ksi\sqrt{in}}{1.12(28.2ksi)}\right]^2}{\pi} = 0.287in$$

And the safety factor against crack propagation (based on a 1/16" minimum detectable flaw size) is

$$SF = \frac{a_{\min}}{a_{\det}} = \frac{0.287in}{0.0625in} = 4.595$$

~~The calculated minimum crack size is about 4.6 times the maximum possible pre-existing crack size in the fuel basket (based on 100% surface inspection of each panel). The large safety factor ensures that crack propagation in the HI-STORM FW fuel baskets will not occur due to the non-mechanistic tipover event.~~

3.4.4.1.5 Load Case 5: Design Internal Pressure

The MPC Enclosure Vessel, which is designed to meet the stress intensity limits of ASME Subsection NB [3.4.4], is analyzed for design internal pressure (Table 2.2.1) using the ANSYS finite element code [3.4.1]. Except for the applied loads and the boundary conditions, the finite element model of the MPC Enclosure Vessel used for this load case is identical to the model described in Subsections 3.1.3.2 and 3.4.3.2 for the MPC lifting analysis.

The only load applied to the finite element model for this load case is the MPC design internal pressure for normal conditions (Table 2.2.1). All internal surfaces of the MPC storage cavity are subjected to the design pressure. The center node on the top surface of the MPC upper lid is fixed against translation in all directions. Symmetric boundary conditions are applied to the two vertical symmetry planes. This set of boundary conditions allows the MPC Enclosure Vessel to deform freely under the applied pressure load. Figure 3.4.31 graphically depicts the applied pressure load and the boundary conditions for Load Case 5.

The stress intensity distribution in the MPC Enclosure Vessel under design internal pressure is shown in Figure 3.4.23. Figures 3.4.32 and 3.4.33 plot the thru-thickness variation of the stress intensity at the baseplate center and at the baseplate-to-shell juncture, respectively. The maximum primary and secondary stress intensities in the MPC Enclosure Vessel are compared with the applicable stress intensity limits from Subsection NB of the ASME Code. The allowable stress intensities are taken at 450°F for the MPC shell and MPC lids, 300°F for the baseplate, and 250°F at the baseplate-to-shell juncture. The maximum calculated stress intensities in the MPC Enclosure Vessel, and their corresponding allowable limits, are summarized in Table 3.4.7 for Load Case 5.

Since the stress intensity distribution in the MPC Enclosure Vessel is a linear function of the internal pressure, and the stress intensity limits for normal and off-normal conditions are the same (Table 3.1.7), the minimum calculated safety factor from Table 3.4.7 is used to establish the internal pressure limit for off-normal conditions (Table 2.2.1).

3.4.4.1.6 Load Case 6: Maximum Internal Pressure Under Accident Conditions

The maximum pressure in the MPC Enclosure Vessel under accident conditions is specified in Table 2.2.1. The stress analysis under this pressure condition uses the same model as the one described in the preceding subsection. The only change is the magnitude of the applied pressure. Figure 3.4.34

indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (the increase in dose rate on the radial surface due to CRAs and APSRs are virtually negligible). For the surface dose rate at the bottom, the value for the CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

While the evaluations described above are based on conservative assumptions, the conclusions can vary slightly depending on the number of CRAs and their operating conditions.

5.4.5 Effect of Uncertainties

The design basis calculations presented in this chapter are based on a range of conservative assumptions, but do not explicitly account for uncertainties in the methodologies, codes and input parameters, that is, it is assumed that the effect of uncertainties is small compared to the numerous conservatisms in the analyses. To show that this assumption is valid, calculations have previously been performed as "best estimate" calculations and with estimated uncertainties added [5.4.9]. In all scenarios considered (e.g., evaluation of conservatisms in modeling assumptions, uncertainties associated with MCNP as well as the depletion analysis (including input parameters), etc.), the total dose rates long with uncertainties are comparable to, or lower than, the corresponding values from the design basis calculations. This provides further confirmation that the design basis calculations are reasonable and conservative.

5.4.6 MPC-32ML with Regionalized Loading Patterns

As discussed in Section 5.2, there is only one heat load pattern with a uniform loading configuration for MPC-32ML. However, different burnup, enrichment and cooling time combinations may produce same decay heat, but different source terms. Additional regionalized shielding analysis is provided in this subsection by dividing the MPC-32ML basket cells into Regions 1 to 3, where Region 1 is the innermost cells, and Region 3 is the outermost cells. The fuel burnup, enrichment and cooling time combinations in Table 5.4.9 are used to calculate the adjacent and 1-m dose rates for HI-STORM FW with MPC-32ML. The heat load of each combination is either more than the decay heat limit per cell, or for the minimum cooling time of 3 years. Conservative enrichments are also considered for all combinations. Using very low enrichments (e.g. 1 wt%) with highly burned (e.g. 65 GWd/mtU) fuel would be unrealistically conservative since no such fuel exists. Hence the lower bound enrichment is selected as a function of the burnup, based on a review of actual fuel assemblies in the industry and other

Holtec approved cask systems (e.g., HI-STAR 190 SAR [5.4.10] Appendix 7.C). Based on this approach, the source terms used in the analyses are reasonably bounding for all realistically expected assemblies.- Each burnup, enrichment, and cooling time combination can be in Region 1, Region 2, and/or Region 3 cells. The maximum adjacent and 1 m dose rates are provided in Tables 5.4.10 and 5.4.11, respectively. Higher concrete density may be used in site specific shielding analysis to further lower the occupational dose rates.

[5.4.4] J. C. Wagner, et al., "MCNP: Criticality Safety Benchmark Problems," LA-12415, Los Alamos National Laboratory, October 1992.

[5.4.5] S. E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks," SAND-89-0018, Sandia National Laboratory, Oct. 1989.

[5.4.6] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.

[5.4.7] B.L. Broadhead, "Recommendations for Shielding Evaluations for Transport and Storage Packages," NUREG/CR-6802 (ORNL/TM-2002/31), Oak Ridge National Laboratory, May 2003.

[5.4.8] HI-2012610, Rev. 3, "Final Safety Analysis Report for the HI-STAR 100 Cask System", USNRC Docket 72-1008.

[5.4.9] HI-2073681, Rev. 3, "Safety Analysis Report on the HI-STAR 180 Package", USNRC Docket 71-9325.

[5.4.10] HI-2146214, Rev. 1, "Safety Analysis Report on the HI-STAR 190 Package," USNRC Docket 71-9373.

$$k_{eff}^{max} = k_c + K_c \sigma_c + Bias + \sigma_B$$

where:

⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;

⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.5]. Each final k_{eff} value is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;

⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code;

⇒ ***Bias*** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and

⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

6.4.4 Damaged Fuel and Fuel Debris

6.4.4.1 Generic Approach

All MPCs are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into DFCs. The number and permissible location of DFCs is provided in Table 2.1.1 and the licensing drawing in Section 1.5, respectively. Because the entire height of the fuel basket contains the neutron absorber (Metamic-HT), axial movement of DFC's does not have any reactivity consequence to MPC.

Damaged fuel assemblies, for the most part, are considered to be assemblies with known or suspected cladding defects greater than pinholes or hairline cracks, or with missing rods, but excluding fuel assemblies with gross defects (for the exact definition see the Glossary). Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

To identify the configuration or configurations leading to the highest reactivity, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following subsections.

Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term 'damaged fuel' as used throughout this chapter designates both damaged fuel and fuel debris.

Note that the modeling approach for damaged fuel and fuel debris is identical to that used in the HI-STORM 100 and HI-STAR 100.

Bounding Undamaged Assemblies

The undamaged assemblies assumed in the basket in those cells not filled with DFCs are those that show the highest reactivity for each group of assemblies, namely

- 9x9E for BWR 9x9E/F, 8x8F and 10x10G assemblies
- 10x10F for BWR 10x10F assemblies
- 10x10A for all other BWR assemblies;
- 16x16A for all PWR assemblies with 14x14 and 16x16 arrays; and
- 15x15F for all PWR assemblies with 15x15 and 17x17 arrays; and
- 16x16D for all PWR assemblies qualified for MPC-32ML..

Since the damaged fuel modeling approach results in higher reactivities, requirements of soluble boron for PWR fuel and maximum enrichment for BWR fuel are different from those for undamaged fuel only. Those limits are listed in Table 6.1.4 (PWR) and Table 6.1.5 (BWR) in Section 6.1. Note that for the calculational cases for damaged and undamaged fuel in the MPC-89, the same enrichment is used for the damage and undamaged assemblies.

Note that for the first group of BWR assemblies listed above (9x9E/F, 8x8F and 10x10G), calculations were performed for both 9x9E and 10x10G as undamaged assemblies, and assembly class 9x9E showed the higher reactivity, and is therefore used in the design basis analyses. This may seem contradictory to the results for undamaged assemblies listed in Table 6.1.2, where the 10x10G shows a higher reactivity. However, the cases in Table 6.1.2 are not at the same enrichment between those assemblies.

All calculations with damaged and undamaged fuel are performed for an active length of 150 inches. There are two assembly classes (17x17D and 17x17E) that have a larger active length for the undamaged fuel. However, the calculations for undamaged fuel presented in Table 6.1.1 show that the reactivity of those undamaged assemblies is at least 0.0050 delta-k lower than that of the assembly class 15x15F selected as the bounding assembly for the cases with undamaged and damaged fuel. The effect of the active fuel length is less than that, with a value of 0.0026 reported in Table 6.2.1 for a much larger difference in active length of 50 Inches. The difference in active length between the 17x17D/E and 15x15F is therefore more than bounded, and the 15x15F assembly class is therefore appropriate to bound all undamaged assemblies with 15x15 and 17x17 arrays.

Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e., all cladding and other structural material in the DFC is replaced by water.

- For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.

- The active length of these rods is assumed to be the same as for the intact fuel rods in the basket, even for more densely packed bare fuel rod arrays where it results in a total amount of fuel in the DFC that exceeds that for the intact assembly.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 16 (4x4) and 324 (18x18) for BWR fuel, ~~and~~ between 64 (8x8) and 576 (24x24) for PWR fuel, and between 289 (17x17) and 676 (26x26) for 16x16D. Note that the various arrays of fuel rods in the criticality model are hypothetical. Table 2.1.1 in Chapter 2 provides the maximum allowable fuel assembly weight (including DFC) authorized for storage in each MPC.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

Further, this is a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 150 inch (BWR and PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

All calculations are performed for full cask models, containing the maximum permissible number of DFCs together with undamaged assemblies.

As an example of the damaged fuel model used in the analyses, Figure 6.4.1 shows the basket cell of an MPC-37 with a DFC containing a 14x14 array of bare fuel rods.

Principal results are listed in Table 6.4.6, ~~and~~ 6.4.7~~,~~ ~~6.4.11~~ and 6.4.11~~2~~2 for the MPC-37, MPC-89, ~~MPC-31C~~ and MPC-~~89~~32ML, respectively. In all cases, the maximum k_{eff} is below the regulatory limit of 0.95.

9.2 PROCEDURE FOR LOADING THE HI-STORM FW SYSTEM IN THE SPENT FUEL POOL

9.2.1 Overview of Loading Operations

The HI-STORM FW system is used to load, transfer, and store spent fuel. Specific steps, required to prepare the HI-STORM FW system for fuel loading, to load the fuel, to prepare the system for storage, and to place it in storage at an ISFSI are described in this chapter. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC VW and/or HI-STORM FW may be moved between the ISFSI and the fuel loading facility using any load handling equipment designed for such applications. Users of the HI-STORM FW system are required to develop detailed written procedures to control on-site transport operations. Instructions for general lifting, handling, and placement of the HI-STORM FW overpack, MPC, and HI-TRAC VW vary by site and are provided on a site-specific basis in Holtec-approved procedures and instructions.

The broad operational steps are explained below and illustrative figures are provided at the end of this section. At the start of loading operations, an empty MPC is upended. The empty MPC is raised and inserted into the HI-TRAC VW. The annulus is filled with plant demineralized water[1] and an inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC VW to prevent spent fuel pool water from contaminating the exterior surface of the MPC when it is submerged in the pool. The MPC is filled with either spent fuel pool water or plant demineralized water (borated as required)[2]. The HI-TRAC VW top flange is outfitted with the lift blocks and the HI-TRAC VW and MPC are then raised and lowered into the spent fuel pool[3] for fuel loading using the lift yoke. Pre-selected assemblies[4] are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielded lid (the MPC lid) is installed. The lift yoke remotely engages to the HI-TRAC VW lift blocks to lift the HI-TRAC VW and loaded MPC close to the spent fuel pool surface. When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC VW from the spent fuel pool, the cask is removed from the spent fuel pool. The lift yoke and HI-TRAC VW are decontaminated, in accordance with instructions from the site's radiological protection personnel, as they are removed from the spent fuel pool.

HI-TRAC VW is placed in the designated preparation area and the lift yoke is removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC VW are decontaminated. The neutron shield water jacket is filled with water (if drained). The inflatable annulus seal is removed and an annulus shield is installed. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

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1 Users may substitute domestic water or radiologically clean borated water in each step where demineralized water is specified.

2 Users may also fill the MPC with water during HI-TRAC placement in the spent fuel pool.

3 Spent Fuel Pool as used in this chapter generically refers to the users designated cask loading location.

4 Damaged fuel assemblies are loaded and stored in Damaged Fuel Containers in the MPC basket.

The MPC water level and annulus water level are lowered slightly, the MPC is vented, and the MPC lid is welded on using the automated welding system. Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. A progressive PT examination as described in the Code Alternatives listed in the CoC is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The MPC welds are then pressure tested followed by an additional liquid penetrant examination performed on the MPC Lid-to-Shell weld to verify structural integrity. To calculate the helium backfill requirements for the MPC (if the backfill is based upon helium mass or volume measurements), the free volume inside the MPC must first be determined. This free volume may be determined by measurement or determined analytically. The remaining bulk water in the MPC is drained.

Caution:

Inert gas must be used any time the fuel is not covered with water to prevent oxidation of the fuel cladding. The fuel cladding is not to be exposed to air at any time during loading operations.

Depending on the burn-up or decay heat load of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system (VDS) or forced helium dehydration (FHD) system. For MPCs without high burn-up fuel or with high burnup fuel and with sufficiently low decay heat, the vacuum drying system may be connected to the MPC and used to remove all liquid water from the MPC. The annular gap between the MPC and HI-TRAC is filled with water during vacuum drying to promote heat transfer from the MPC and maintain lower fuel cladding temperatures. The internal pressure is reduced and held in accordance with Technical Specifications to ensure that all liquid water is removed.

An FHD system is required for high-burn-up fuel at higher decay heat (it can be used as an alternative to vacuum drying) to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. The temperature of the gas exiting the system demoisturizer is maintained in accordance with Technical Specification requirements to ensure that all liquid water is removed.

Following MPC moisture removal, by VDS or FHD, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, and provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes (for multi-pass welds). The cover plate welds are then leak tested.

The MPC closure ring is then placed on the MPC and aligned, tacked in place, and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

9.2.3 MPC Fuel Loading

Note:
When loading an MPC 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.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC. Perform a verification of the types, amounts, and location of non-fuel hardware using plant fuel records to ensure that only non-fuel hardware that meet the conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC.

Caution:
In accordance with the definition of "Undamaged Fuel," some low-enriched channeled fuel must be shown to be without known or suspected grossly breached spent fuel rods. This determination can be made based on review of records, fuel sipping, or other method.

~~16.~~2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern[6].

~~17.~~3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

~~18.~~4. If required, install fuel shims where necessary in the fuel cells.

9.2.4 MPC Closure

1. Install MPC lid and remove the HI-TRAC VW from the spent fuel pool as follows:

   a. Rig the MPC lid for installation in the MPC in accordance with site-approved rigging procedures.

   b. Install the drain line to the underside of the MPC lid.

   c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC for installation.

   d. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

[6] Damaged fuel must be loaded into Damage Fuel Containers in the MPC basket.