CHAPTER 6[†]: CRITICALITY EVALUATION

6.0 INTRODUCTION

This chapter documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel in accordance with 10CFR72.124 [6.1.2]. The evaluation shows that the maximum k_{eff} value, including all applicable biases and uncertainties is below 0.95 for all normal, off-normal and accident conditions. This demonstrates that the HI-STORM FW system meets the criticality safety requirements of 10CFR72 [6.1.2] and the Standard Review Plan for Dry Cask Storage Systems (NUREG-1536) [6.1.1].

In addition, this chapter describes the HI-STORM FW system design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to provide a sufficient basis for the evaluation of the package.

Note that the analysis methodologies and modeling assumptions are identical to those utilized in the licensing of the HI-STORM 100 system in Docket No. 72-1014 ([6.0.1], Chapter 6), except for the following:

- A newer version of the Monte Carlo code MCNP, namely MCNP5, is used, together with the corresponding cross-sections. The benchmark calculations were updated accordingly.

The safety analyses summarized in this chapter demonstrate acceptable margins to the allowable limits under all design basis loading conditions and operational modes. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed safety factors may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such unquantified changes on the reduction of any of the computed safety margins cannot be deemed to be insignificant. For purposes of this determination, an insignificant loss of safety margin with reference to an

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

Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.0.2].

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. To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1] and 10CFR72.124 [6.1.2], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM FW system, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, the results demonstrate that the HI-STORM FW system is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM FW system when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM FW system depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC;

2. The fuel basket structure which is made entirely of the Metamic-HT neutron absorber material;

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and

4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel in the PWR fuel basket.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 12 have no adverse effect on the design parameters important to criticality safety, except for the non-mechanistic tip-over event, which could result in limited plastic deformation of the basket. However, a bounding basket deformation is already included in the criticality models for normal conditions, and thus, from the criticality safety standpoint, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM FW system is designed such that the fixed neutron absorber will remain effective for a storage period greater than 60 years, and there are no credible mechanisms that would cause its loss or a diminution of its effectiveness (see Chapter 8, specifically Section 8.9, and Section 10.1.6.3 for further information on the qualification and testing of the neutron absorber material). Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM FW system does not rely on the use of any of the following aids to the reduction of reactivity present in the storage system:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 90 percent of the B-10 content for the Metamic-HT fixed neutron absorber undergirded by comprehensive tests as described in Subsection 10.1.6.3.

The HI-STORM FW system consists of the HI-STORM FW storage cask, the HI-TRAC VW transfer cask and Multi-Purpose-Canisters (MPCs) for PWR and BWR fuel (see Chapter 1, Table 1.0.1). Both the HI-TRAC VW transfer cask and the HI-STORM FW storage cask accommodate the interchangeable MPC designs. The HI-STORM FW storage cask uses concrete as a shield for both gamma and neutron radiation, while the HI-TRAC VW uses lead and steel for gamma radiation and a water-filled jacket for neutron shielding. The design details can be found in the drawing packages in Section 1.5.

While the MPCs are in the HI-STORM FW cask during storage, they are internally dry (no moderator), and thus, the reactivity is very low ($k_{eff} \sim 0.6$). However, the MPCs are flooded for loading and unloading operations in the HI-TRAC VW cask, which represents the limiting case in terms of reactivity. Therefore, the majority of the analyses have been performed with the MPCs in a HI-TRAC VW cask, and only selected cases have been performed for the HI-STORM FW cask.

Confirmation of the criticality safety of the HI-STORM FW system was accomplished with the three-dimensional Monte Carlo code MCNP5 [6.1.4]. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.5].

To assess the reactivity effects due to temperature changes, CASMO-4, a two-dimensional transport theory code [6.1.6] for fuel assemblies was used. CASMO-4 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects.

Benchmark calculations were made to compare the primary code package (MCNP5) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM FW system. The most important parameters are (1) the enrichment, (2) cell spacing, (3) the ^{10}B loading of the neutron absorber panels, and (4) the soluble boron concentration in the water (for PWR fuel). The critical experiment benchmarking work is summarized in Appendix 6.A.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- No credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product.

- The fuel stack density is assumed to be at 97.5% of the theoretical density for all criticality analyses. This is a conservative value, since it corresponds to a very high pellet density of 99% or more of the theoretical density. Note that this difference between stack and pellet density is due to the necessary dishing and chamfering of the pellets.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.

- When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.

- Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water. This is conservative since studies presented in Section 6.2.1 show that all assemblies are undermoderated, and that the reduction in the amount of (borated or unborated) water within the fuel assembly always results in a reduction of the reactivity. The presence of any other structural material, which would reduce the amount of water, is therefore bounded by those studies, and neglecting this material is conservative. Additionally, the potential neutron absorption of those materials is neglected.

- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.

- Planar-averaged enrichments are assumed for BWR fuel. Analyses are presented that demonstrate that the use of planar-averaged enrichments is appropriate.

- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias and bias uncertainty, two approaches are utilized. One where the results of the benchmark calculations are used directly and one where benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000. Consistent with NUREG-1536, the larger of the combined bias and bias uncertainty of the two approaches is used.

- The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- Regarding the position of assemblies in the basket, configurations with centered and eccentric positioning of assemblies in the fuel storage locations are considered.

- For undamaged fuel assemblies, as defined in the Glossary, all fuel rod positions are assumed to contain a fuel rod. To qualify assemblies with missing fuel rods, those missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

- For DFCs, a large ID and small wall thickness is used. This is conservative, since it maximizes the area of the optimum moderated fuel, and minimizes the neutron absorption in the DFC wall.

The design basis criticality safety calculations are performed for a single internally flooded HI-TRAC VW transfer cask with full water reflection on all sides (limiting cases for the HI-STORM FW system), for fuel assemblies listed in Chapter 2, are conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and include the calculational bias, uncertainties, and calculational statistics. In addition, a few results for single internally dry (no moderator) HI-STORM FW storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are performed to confirm the low reactivity of the HI-STORM FW system in storage.

Note that throughout this chapter reactivity results are stated as maximum neutron multiplication factor values (k_{eff}) conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics, unless otherwise noted.

For undamaged fuel, and for each of the MPC designs under flooded conditions (HI-TRAC VW), minimum soluble boron concentration (if applicable) and fuel assembly classes[††], Tables 6.1.1,

[††] The assembly classes for BWR and PWR fuel are defined in Section 6.2.

6.1.2 and 6.1.6 list the bounding maximum k_{eff} value, and the associated maximum allowable enrichment. Tables 6.1.1 and 6.1.2 provide the information for undamaged fuel without known or suspected cladding defects larger than pinhole leaks or hairline cracks, while Table 6.1.6 provides the information for low-enriched, channeled BWR undamaged fuel without known or suspected grossly breached fuel rods. The maximum allowed enrichments and the minimum soluble boron concentrations are also cited in Subsection 2.1.

For MPCs in the HI-STORM FW under dry conditions, results are listed in Table 6.1.3 for selected assembly classes.

For MPCs loaded with a combination of undamaged and damaged fuel assemblies under flooded conditions, results are listed in Tables 6.1.4 and 6.1.5. For each of the MPC designs, the tables indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum k_{eff} value, the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration. Allowed enrichments are also cited in Subsection 2.1.

These results confirm that the maximum k_{eff} values for the HI-STORM FW system are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM FW system to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4. The capability of the HI-STORM FW to accommodate low enriched, channeled BWR fuel as undamaged fuel demonstrated in Subsection 6.4.9.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM FW System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM FW storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

For PWR fuel in the MPC-37, soluble boron in the water is credited. There is a strict administrative control on the soluble boron concentration during loading and unloading of the MPC, consisting of frequent and independent measurements (For details see Subsections 9.2.2, 9.2.3, 9.2.4, and 9.4.3 and the bases for LCO 3.3.1 in Chapter 13). An accidental loss of soluble boron is therefore not credible and hence not considered.

TABLE 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 (HI-TRAC VW)

Fuel Assembly Class	4.0 wt% ^{235}U Maximum Enrichment†		5.0 wt% ^{235}U Maximum Enrichment†	
	Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}	Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}
14x14A	1000	0.8946	1500	0.8983
14x14B	1000	0.9121	1500	0.9172
14x14C	1000	0.9211	1500	0.9277
15x15B	1500	0.9129	2000	0.9311
15x15C	1500	0.9029	2000	0.9188
15x15D	1500	0.9223	2000	0.9421
15x15E	1500	0.9206	2000	0.9410
15x15F	1500	0.9244	2000	0.9455
15x15H	1500	0.9142	2000	0.9325
15x15I	1500	0.9155	2000	0.9362
16x16A	1000	0.9275	1500	0.9366
17x17A	1500	0.9009	2000	0.9194
17x17B	1500	0.9181	2000	0.9380
17x17C	1500	0.9222	2000	0.9424
17x17D	1500	0.9183	2000	0.9384
17x17E	1500	0.9203	2000	0.9392

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-89 (HI-TRAC VW)

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
7x7B	4.8	0.9317
8x8B	4.8	0.9369
8x8C	4.8	0.9399
8x8D	4.8	0.9380
8x8E	4.8	0.9281
8x8F	4.5	0.9328
9x9A	4.8	0.9421
9x9B	4.8	0.9410
9x9C	4.8	0.9338
9x9D	4.8	0.9342
9x9E/F	4.5	0.9346
9x9G	4.8	0.9307
10x10A	4.8	0.9435
10x10B	4.8	0.9417
10x10C	4.8	0.9389
10x10F	4.7	0.9440
10x10G	4.6	0.9466

TABLE 6.1.3

REPRESENTATIVE k_{eff} VALUES FOR MPC-37 AND MPC-89 IN THE HI-STORM FW OVERPACK

MPC	**Fuel Assembly Class**	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
MPC-37	17x17B	5.0	0.6076
MPC-89	10x10A	4.8	0.3986

TABLE 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-37
WITH UP TO 12 DFCs

Fuel Assembly Class of Undamaged Fuel	**4.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris** †		**5.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris** †	
	Minimum Soluble Boron Concentration (ppm)	**Maximum k_{eff}**	**Minimum Soluble Boron Concentration (ppm)**	**Maximum k_{eff}**
All 14x14, 16x16A	1300	0.9023	1800	0.9163
All 15x15, all 17x17	1800	0.9032	2300	0.9276

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-89
WITH UP TO 16 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum k_{eff}
All BWR Classes except 8x8F, 9x9E/F, 10x10F and 10x10G	4.8	0.9464
8x8F, 9x9E/F and 10x10G	4.0	0.9299
10x10F	4.6	0.9428

TABLE 6.1.6

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-89
WITH LOW ENRICHED, CHANNELED BWR FUEL

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum k_{eff}
All BWR Classes	3.3	0.9325

6.2 SPENT FUEL LOADING

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in defining the authorized contents could limit or complicate the applicability of the HI-STORM FW system. To resolve this limitation, a number of fuel assembly classes for both fuel types (PWR and BWR) are defined based on bounding fuel dimensions. The results of parametric studies justify using those bounding fuel dimensions for defining the authorized contents.

6.2.1 Definition of Assembly Classes

For each array size the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; and (3) number and locations of guide tubes (PWR) or water rods (BWR). The assembly classes for PWR and BWR fuel are defined in Chapter 2, Tables 2.1.2 and 2.1.3, respectively. It should be noted that these assembly classes are consistent with the class designations in the HI-STORM 100 FSAR (Docket No. 72-1014). Specifically, assembly classes with the same identifier refer to the same set of limiting dimensions. However, some classes have been removed and others have been added compared to the HI-STORM 100.

In HI-STORM 100 FSAR (Docket No. 72-1014), extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- maximum fuel rod pitch,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely 1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:

- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.

- Fuel assemblies are predominantly characterized by the amount of fuel and the fuel-to-water (moderator) ratio. Increasing the amount of fuel, or the enrichment of the fuel, will increase the amount of fissile material, and therefore increase reactivity. Regarding the fuel-to-water ratio, it is important to note that commercial PWR and BWR assemblies are undermoderated, i.e. they do not contain enough water for a maximum possible reactivity.
- The neutron poison in the basket walls uses B-10, which is an absorber of thermal neutrons. This poison therefore also needs water (moderator) to be effective. This places a specific importance on the amount of water between the outer rows of the fuel assemblies and the basket cell walls. Note that this explains some of the differences in reactivity between the different assembly types in the same basket, even for the same enrichment, where assemblies with a smaller cross section, i.e. which have more water between the periphery of the assembly and the surrounding wall, generally have a lower reactivity.

Based on these characteristics, the following conclusions can be made:

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536 (see section 6.4.2.3), a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not

by a positive increase in reactivity from the reduction in fuel (which would be counter-intuitive). Therefore, in order not to overstate the conservative effect of the flooded fuel-to-clad gap, the calculations for the variation of the fuel pellet diameter are performed for a flooded gap of constant thickness by also changing the clad ID.

Since all assemblies have the same principal design, i.e. consist of bundles of cladded fuel rods, most of them with embedded guide/instrument tubes or water rods or channels, the above conclusions apply to all of them, and the bounding dimensions are therefore also common to all fuel assemblies analyzed here. Nevertheless, to clearly demonstrate that the main assumption is true, i.e. that all assemblies are undermoderated, a study was performed for all assembly types where the pellet-to-clad gap is empty instead of being flooded (a conservative assumption for the design basis calculations, see Section 6.4.2.3) The results are listed in Table 6.2.3, in comparison with the results of the reference cases with the flooded gap from Section 6.1 for those assembly types. In all cases, the reactivity is reduced compared to the reference case. This verifies that all assembly types considered here are in fact undermoderated, and therefore validates the main assumption stated above. All assembly types are therefore behaving in a similar fashion, and the bounding dimensions are therefore applicable to all assembly types. This discussion and the corresponding conclusions not only affect fuel behavior, but also other moderation effects, and is therefore further referenced in Section 6.3.1 and 6.4.2

As a result, the authorized contents in Subsection 2.1 are defined in terms of those bounding assembly parameters for each class.

Nevertheless, to further demonstrate that the aforementioned characteristics are in fact bounding for the HI-STORM FW, parametric studies were performed on reference PWR and BWR assemblies, namely PWR assembly class 17x17B and BWR assembly class 10x10A. The results of these studies are shown in Table 6.2.1 and 6.2.2, and verify the bounding parameters listed above. Note that in the studies presented in Tables 6.2.1 and 6.2.2, the fuel pellet diameter and cladding inner diameter are changed together. This is to keep the cladding-to-pellet gap, which is conservatively flooded with pure water in all cases (see Section 6.4.2.3), at a constant thickness, to ensure the studies evaluate the fuel parameters rather than the moderation conditions, as discussed above.

In addition to those dimensions, additional fuel assembly characteristics important to criticality control are the location of guide tubes, water rods, part length rods, and rods with differing dimensions (classes 9x9E/F only). These are identified in the assembly cross sections provided in Appendix 6.B, Section B.4.

In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel, and Integral Fuel Burnable Absorbers (IFBA) used in PWR fuel was conservatively neglected.
Some assembly classes contain partial length rods. There are differences in location of those partial length rods within the assembly that influence how those rods affect reactivity: Assembly classes 9x9A, 10x10A, 10x10B and 10x10F have partial length rods that are completely

surrounded by full length rods, whereas assembly class 10x10G has those partial length rods on the periphery of the assembly or facing the water gap, where they directly only face two full length rods (see Appendix 6.B, Section B.4). To determine a bounding configuration for those assembly classes where partial length rods are completely surrounded by full length rods, calculations are listed in Table 6.2.2 for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. The results show that the configurations with only the full length rods present, i.e. where the partial length rods are assumed completely absent from the assembly, is bounding. This is an expected outcome, since LWR assemblies are typically undermoderated, therefore reducing the fuel-to-water-ratio within the rod array tends to increase reactivity. Consequently, all assembly classes that contain partial length rods surrounded by full-length rods are analyzed with the partial length rods absent. For assembly class 10x10G, calculations with different assumptions for the length of the part-length rods are presented in Table 6.2.7, and show that reducing the length of the part length rods reduces reactivity. This means that the reduction in the fuel amount is more dominating than the change in moderation for this configuration. For this class, all rods therefore are assumed full length. Note that in neither of the cases is the configuration with the actual part length rods bounding. The specification of the authorized contents has therefore no minimum requirement for the active fuel length of the partial length rods.

BWR assemblies are specified in Table 2.1.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in the HI-STORM FSAR ([6.0.1], Chapter 6, Appendix 6.B) show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum k_{eff}. To verify that this is applicable to the HI-STORM FW, those calculations were re-performed in the MPC-89. The results are presented in Table 6.2.4, and show that, as expected, the planar average enrichments bound or are statistically equivalent to the distributed enrichment in the HI-STORM FW as they do in the HI-STORM 100. To confirm that this is also true for the higher enrichments analyzed here, additional calculations were performed and are presented in Table 6.2.2 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt% ^{235}U is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt% ^{235}U is close to the maximum rod enrichment of 5.0 wt% ^{235}U, the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt% ^{235}U, while the balance of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results are also included in table 6.2.2 and show that in all

cases, the maximum k_{eff} calculated for the distributed enrichments are statistically equivalent to or below those for the uniform enrichments. Therefore, modeling BWR assemblies with distributed enrichments using a uniform enrichment equal to the planar-average value is acceptable and conservative. The assumed enrichment distributions analyzed are shown in Appendix 6.B.

Note that for some BWR fuel assembly classes, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. For these cases, the bounding water rod thickness is listed as zero.

Two BWR classes (8x8B and 8x8D) are specified with slight variation in the number of fuel and/or water rods (see Section 6.B.4). The results listed in Section 6.1 utilize the minimum number of fuel rods, i.e. maximizing the water-to-fuel ratio. To show that this is appropriate and bounding, calculations were also performed with the alternative configurations, and are presented in Table 6.2.5. The results show that the reference conditions used for the calculations documented in Section 6.1 are in fact bounding.

For BWR assembly class 9x9E/F, two patterns of water rods were analyzed (see Section 6.B.4). The comparison is also presented in Table 6.2.5 and shows that the condition with the larger water rod spacing is bounding.

For PWR assembly class 15x15I (see Section 6.B.4), calculations with and without guide rods were performed. The comparison is also presented in Table 6.2.5. The case without the guide rods is used as the design basis case for this assembly type, therefore, no specific restrictions on the location and number of guide rods exists.

Typically, PWR fuel assemblies are designed with solid fuel pellets throughout the entire active fuel length. However, some PWR assemblies contain annular fuel pellets in the top and bottom 6 to 8 inches of the active fuel length. This changes the fuel to water ratio in these areas, which could have an effect on reactivity. However, the top and bottom of the active length are areas with high neutron leakage, and changes in these areas typically have no significant effect on reactivity. Studies with up to 12 inches of annular pellets at the top and bottom performed for the HI-STORM FW with various pellet IDs (see Table 6.2.6) confirm this, i.e., shown no significant reactivity effects, even if the annular region of the pellet is flooded with pure water. All calculations for PWR fuel assemblies are therefore performed with solid fuel pellets along the entire length of the active fuel region, and the results are directly applicable to those PWR assemblies with annular fuel pellets. This is consistent with the HI-STORM 100, where the same analyzed conditions are analyzed and qualified.

TABLE 6.2.1

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-37 with 2000 ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum k_{eff}	standard deviation
17x17B (5.0 wt% Enrichment)	Reference	0.9374	0.0004
increase pellet OD and clad ID (+0.004)	0.0052	0.9426	0.0003
decrease pellet OD and Clad ID (-0.004)	-0.0058	0.9316	0.0004
increase clad OD (+0.004)	-0.0014	0.9360	0.0004
decrease clad OD (-0.004)	0.0017	0.9391	0.0004
increase guide tube thickness (+0.004)	-0.0001	0.9373	0.0004
decrease guide tube thickness (-0.004)	0.0004	0.9378	0.0003
remove guide tubes (i.e., replace the guide tubes with water)	0.0009	0.9383	0.0004
reduced active length (100 Inches)	-0.0020	0.9354	0.0004
increase rod pitch (+0.004)	0.0019	0.9393	0.0004
reduce rod pitch (-0.004)	-0.0017	0.9357	0.0004

TABLE 6.2.2

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-89
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum k_{eff}	standard deviation
10x10A (Reference, full-length rods only)	Reference	0.9429	0.0004
increase pellet OD and Clad ID (+0.004)	0.0037	0.9466	0.0004
decrease pellet OD and Clad ID (-0.004)	-0.0042	0.9387	0.0004
increase clad OD (+0.004)	-0.0021	0.9408	0.0003
decrease clad OD (-0.004)	0.0032	0.9461	0.0004
increase water rod thickness (+0.004)	0.0002	0.9431	0.0004
decrease water rod thickness (-0.004)	0.0009	0.9438	0.0004
remove water rods (i.e., replace the water rod tubes with water)	0.0031	0.9460	0.0004
reduced active length (100 Inches)	-0.0026	0.9403	0.0004
remove channel	-0.0113	0.9316	0.0003
increase channel thickness (+0.020)	0.0007	0.9436	0.0003
full-length and part-length rods (real assembly)	-0.0054	0.9375	0.0004
part-length rods extended to full-length	-0.0102	0.9327	0.0004
increased rod pitch (+0.004)	0.0050	0.9479	0.0004
reduced rod pitch (-0.004)	-0.0043	0.9386	0.0003
distributed enrichment, Case 1	-0.0011	0.9418	0.0003
distributed enrichment, Case 2	+0.0004	0.9433	0.0003
distributed enrichment, Case 3	-0.0099	0.9330	0.0004
distributed enrichment, Case 4	-0.0121	0.9308	0.0003

TABLE 6.2.3

EFFECT OF THE FLOODING OF THE PELLET-TO-CLAD GAP

Fuel Assembly Class	Maximum k_{eff} at 5.0 wt% ^{235}U Maximum Enrichment		
	Flooded Pellet-to-Clad Gap	Empty Pellet-to-Clad Gap	Difference
14x14A	0.8983	0.8962	-0.0021
14x14B	0.9172	0.9134	-0.0038
14x14C	0.9277	0.9237	-0.0038
15x15B	0.9311	0.9284	-0.0027
15x15C	0.9188	0.9164	-0.0024
15x15D	0.9421	0.9386	-0.0035
15x15E	0.9410	0.9371	-0.0039
15x15F	0.9455	0.9408	-0.0047
15x15H	0.9325	0.9300	-0.0025
15x15I	0.9357	0.9305	-0.0052
16x16A	0.9366	0.9284	-0.0082
17x17A	0.9194	0.9160	-0.0034
17x17B	0.9380	0.9335	-0.0045
17x17C	0.9424	0.9375	-0.0049
17x17D	0.9384	0.9323	-0.0061
17x17E	0.9392	0.9346	-0.0046

TABLE 6.2.3 (continued)

EFFECT OF THE FLOODING OF THE PELLET-TO-CLAD GAP

Fuel Assembly Class	**Maximum k_{eff}**		
	Flooded Pellet-to-Clad Gap	**Empty Pellet-to-Clad Gap**	**Difference**
7x7B	0.9317	0.9261	-0.0056
8x8B	0.9369	0.9318	-0.0051
8x8C	0.9399	0.9331	-0.0068
8x8D	0.9380	0.9334	-0.0046
8x8E	0.9281	0.9230	-0.0051
8x8F	0.9328	0.9275	-0.0053
9x9A	0.9421	0.9370	-0.0051
9x9B	0.9410	0.9292	-0.0118
9x9C	0.9338	0.9290	-0.0048
9x9D	0.9342	0.9294	-0.0048
9x9E/F	0.9346	0.9261	-0.0085
9x9G	0.9307	0.9250	-0.0057
10x10A	0.9435	0.9391	-0.0044
10x10B	0.9417	0.9317	-0.0100
10x10C	0.9389	0.9333	-0.0056
10x10F	0.9440	0.9395	-0.0045
10x10G	0.9466	0.9408	-0.0058

Table 6.2.4

COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND DISTRIBUTED ENRICHMENTS

Case	**Planar Average Enrichment (wt%)**	**Peak Rod Enrichment (wt%)**	**Maximum k_{eff}**	
			Planar Average Enrichment (wt%)	**Peak Rod Enrichment (wt%)**
8x8C	3.01	3.80	0.8358	0.8309
8x8C	3.934	4.9	0.8975	0.8899
8x8D	3.42	3.95	0.8628	0.8636
8x8D	3.78	4.40	0.8862	0.8855
8x8D	3.90	4.90	0.8934	0.8913
9x9B	4.34	4.71	0.9195	0.9179
9x9D	3.35	4.34	0.8575	0.8456
Hypothetical #1 (48 outer rods of 3.967%E, 14 inner rods of 5.0%)	4.20	5.00	0.9104	0.9102
Hypothetical #2 (48 outer rods of 4.354%E, 14 inner rods of 5.0%)	4.50	5.00	0.9258	0.9247

Table 6.2.5

VARIATIONS OF NUMBER OF FUEL AND/OR WATER RODS FOR ASSEMBLY CLASSES 8x8B AND 8x8D (see Appendix B)

Case	**Maximum k_{eff}**
Assembly Class 8x8B	
63 Fuel Rods (Reference)	0.9369
64 Fuel Rods	0.9342
Assembly Class 8x8D	
60 Fuel Rods, no water rods modeled (Reference)	0.9380
60 Fuel Rods, 2 larger, 2 smaller water rods	0.9362
60 Fuel Rods, 4 larger water rods	0.9347
60 Fuel Rods, 4 smaller water rods	0.9359
60 Fuel Rods, 1 large water rods	0.9343
61 Fuel Rods, 3 water rods	0.9354

VARIATION OF WATER ROD LOCATIONS FOR ASSEMBLY CLASS 9x9E/F (see Appendix B)

Case	**Maximum k_{eff}**
Adjacent Water Rods (Reference)	0.9346
Water Rods separated by a Fuel Rod	0.9313

VARIATION OF GUIDE RODS FOR ASSEMBLY CLASS 15x15I (see Appendix B)

Case	**Maximum k_{eff}**
No Guide Rods (Reference)	0.9362
8 Guide Rods	0.9259

Table 6.2.6

EFFECT OF ANNULAR PELLETS IN THE TOP AND BOTTOM 12 INCHES OF THE ACTIVE REGION

Diameter of Annulus in Pellets	**Maximum k_{eff}**
Assembly Class 17x17B, 5% Enrichment, Undamaged Fuel	
None (Reference)	0.9380
0.1 Inches	0.9382
0.2 Inches	0.9379
0.3 Inches	0.9371
0.4 Inches	0.9371
0.5 Inches	0.9363
0.6 Inches	0.9368
Assembly Class 16x16A, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9163
0.2 Inches	0.9162
Assembly Class 15x15F, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9276
0.2 Inches	0.9266

Table 6.2.7

EFFECT OF PARTIAL LENGTH RODS FOR ASSEMBLY CLASS 10x10G

Length of Partial Length Rods as a Percentage of Full Length Rods	**Maximum k_{eff}**
100%	0.9466
75%	0.9455
50%	0.9404
25%	0.9286
0%	0.9208

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1 through 6.3.5 show representative cross sections of the criticality models for the two baskets. Figures 6.3.1 and 6.3.2 show a single cell from each of the two baskets. Figures 6.3.3 and 6.3.4 show the entire MPC-37 and MPC-89 basket, respectively. Figure 6.3.5 shows a sketch of the calculational model in the axial direction.

Full three-dimensional calculational models were used for all calculations. The calculational models explicitly define the fuel rods and cladding, the guide tubes, water rods and the channel (for the BWR assembly), the neutron absorber walls of the basket cells, and the surrounding MPC shell and overpack. For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel rod pellet-to-clad gaps, since this represents the bounding condition as demonstrated in Section 6.4.2.3. Appendix 6.B provides sample input files for typical MPC basket designs

Note that the water thickness above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR and PWR assembly type, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion in Section 6.2.1 are listed below. Note that this approach is consistent with that used for the HI-STORM 100.

Basket Manufacturing Tolerance: The two aspects of the basket tolerance that are evaluated are the cell wall thickness and the cell ID. The reduced cell wall thickness results in a reduced amount of poison (since the material composition of the wall is fixed), and therefore in an increase in reactivity. The reduced cell ID reduces amount of water between the fuel the poison, and therefore the effectiveness of the poison material. Both effects are simply a function of the geometry, and are independent of the fuel type.

Panel Gaps: Similar to the basket manufacturing tolerance for the cell wall thickness, this tolerance has a small effect on the overall poison amount of the basket, which would affect the reactivity of the system independent of the fuel type.

Eccentric positioning (see Section 6.3.3): When a fuel assembly is located in the center of a basket cell, it is surrounded by equal amounts of water on all sides, and hence the thermalization of the neutrons that occur between the assembly and the poison in the cell wall, and hence the effectiveness of the poison, is also equal on all sides. For an eccentric positioning, the

effectiveness of the poison is now reduced on those sides where the assembly is located close to the cell walls, and increased on the opposite sides. This creates a compensatory situation for a single cell, where the net effect is not immediately clear. However, for the entire basket, and for the condition where all assemblies are located closest to the center of the basket, the four assemblies at the center of the basket are now located close to each other, separated by poison plates with a reduced effectiveness since they are not surrounded by water on any side. This now becomes the dominating condition is terms of reactivity increase. This effect is also applicable to all assembly types, since those assemblies are all located close to the center of the basket, i.e. the eccentric position with all assemblies moved towards the center will be bounding regardless of the assembly type.

The basket geometry can vary due to manufacturing tolerances and due to potential deflections of basket walls as the result of accident conditions. The basket tolerances are defined on the drawings in Chapter 1. The structural acceptance criteria for the basket during accident conditions is that the permanent deflection of the basket panels is limited to a fraction of 0.005 (0.5%) of the panel width (see Chapter 3). The analyses in Chapter 3 demonstrate that permanent deformations of the basket walls during accident conditions are far below this limit. In fact, the analyses show that the vast majority of the basket panels remain elastic during and after an accident, and therefore show no permanent deflection whatsoever, and that any deformation is limited to small localized areas. Nevertheless, it is conservatively assumed that 2 adjacent cell walls in each cell are deflected to the maximum extent possible over their entire length and width, i.e. that the cell ID is reduced by 0.5% of the cell width, or 0.045" for the MPC-37 cells and 0.030" for the MPC-89 cells. Stated differently, the minimum cell ID based on tolerances was further reduced by the amounts stated above for all cells in each basket to account for the potential deflections of basket walls during accident conditions. Assuming that all cell sizes are reduced is a simplifying, but very conservative assumption, since cell walls are shared between neighboring cells, so while the deflection of a basket wall would reduce the cell size on one side, it necessarily increases that on the other side of the wall. MCNP5 was used to determine the manufacturing tolerances and deflections that produced the most adverse effect on criticality. After the reactivity effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case conditions in the direction which would increase reactivity. For simplification, the same worst case conditions are used for both normal and accident conditions. For all calculations, fuel assemblies were assumed to be eccentrically located in the cells, since this results in higher reactivities (see Section 6.3.3). Maximum k_{eff} results (including the bias, uncertainties, or calculational statistics), along with the selected dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for both baskets. The cell ID is evaluated for minimum (tolerance only), minimum with deformation, nominal and an increased value. The wall thickness is evaluated for nominal and minimum values.

Based on the calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined for the basket designs. Because the reactivity effect (positive or negative) of the

manufacturing tolerances is not assembly dependent, these dimensional assumptions were employed for all criticality analyses.

The basket is manufactured from individual slotted panels. The panels are expected to be in direct contact with each other (see Drawings in Chapter 1). However, to show that small gaps between panels would have essentially no effect on criticality, calculations are performed with a postulated 0.06" gap between panels, repeated in the axial direction every 10" in all panels. Since it is expected that the effect of these gaps would be small, these calculations were performed with a larger number of particles per cycle, larger number of inactive cycles, and a larger total number of cycles to improve the statistics of each run, so the real reactivity effect could be better separated from the statistical "noise". The results are summarized in Tables 6.3.6 and show that the METAMIC gap has a very small effect. Therefore, all calculations are performed without any gaps between panels.

Variations of water temperature in the cask were analyzed using CASMO-4. The analyses were performed for the assembly class 10x10A in the MPC-89, and for the assembly class 17x17B with 2000 ppm soluble boron in the water in the MPC-37. These are the same assemblies and conditions used for the fuel dimension studies in Section 6.2, and shown there to be representative of all assemblies qualified for those baskets. The results are presented in Table 6.3.1, and show that the minimum water temperature (corresponding to a maximum water density) are bounding. This condition is therefore used in all further calculations. This is expected since an increased temperature results in a reduced water density, a condition that is shown in Section 6.4 to result in reduced reactivities.

Calculations documented in Chapter 3 show that the baskets stay within the applicable structural limits during all normal and accident conditions. Furthermore, the neutron poison material is an integral and non-removable part of the basket material, and its presence is therefore not affected by the accident conditions. Except for the potential deflection of the basket walls that is already considered in the criticality models, damage to the cask under accident conditions is limited to possible loss of the water in the water jacket of he HI-TRAC VW. However, this condition is already considered in the calculational models. Other parameters important to criticality safety are fuel type and enrichment, which are not affected by the hypothetical accident conditions. The calculational models of the cask and basket for the accident conditions are therefore identical to the models for normal conditions, and no separate models need to be developed for accident conditions.

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM FW system are listed in Table 6.3.4. The cross section set for each nuclide is listed in Table 6.3.8, and is consistent with the cross section sets used in the benchmarking calculations documented in Appendix A. Note that these are the default cross sections chosen by the code.

The HI-STORM FW system is designed such that the fixed neutron absorber will remain effective for a storage period greater than 60 years, and there are no credible means to lose it.

The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Subsection 10.1.6.3, to validate the ^{10}B (poison) concentration in the fixed neutron absorber. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an evaluation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 60 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of ^{10}B atoms destroyed is less than 10^{-7} in 60 years. Thus, the reduction in ^{10}B concentration in the fixed neutron absorber by neutron absorption is negligible. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

The potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection The calculations in this subsection serve to identify the eccentric positioning of assemblies in the fuel storage locations, which results in a higher maximum k_{eff} value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum k_{eff} value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- Cell Center Configuration: All assemblies centered in their fuel storage cell;
- Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and
- Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical phenomenon that could move all assemblies within a basket consistently to the center or periphery. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.5, results are presented for all representative conditions. The table shows the maximum k_{eff} value for centered and the two eccentric configurations for each condition, and the

difference in k_{eff} between the centered and eccentric positioning. In all cases, moving the assemblies and DFCs to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position, and moving the assemblies and DFCs towards the center results in an increase in reactivity, compared to the cell centered position. All calculations are therefore performed with assemblies/DFCs moved towards the center of the basket.

TABLE 6.3.1

CASMO-4 CALCULATIONS FOR EFFECT OF TEMPERATURE

Change in Nominal Parameter	**Δk Maximum Tolerance**		**Action/Modeling Assumption**
	MPC-37, 17x17B, 5.0 wt%, Borated Water with 2000 ppm Soluble Boron	**MPC-89, 10x10A, 4.8 wt%, Fresh Water**	
Increase in Temperature 20°C 40°C 70°C 100°C	 Ref. -0.0008 -0.0023 -0.0042	 Ref. -0.0035 -0.0100 -0.0180	Assume 20°C
10% Void in Moderator 20°C with no void 20°C 100°C	 Ref. -0.0036 -0.0096	 Ref. -0.0282 -0.0463	Assume no void

TABLE 6.3.2

EVALUATION OF BASKET MANUFACTURING TOLERANCES

Box I.D.	**Box Wall Thickness**	**Maximum k_{eff}**
MPC-37 (17x17B, 5.0% Enrichment)		
nominal (8.94")	nominal (0.59")	0.9332
nominal (8.94")	minimum (0.57”)	0.9346
increased (8.96")	minimum (0.57”)	0.9350
minimum (8.92")	minimum (0.57”)	0.9352
minimum, including deformation (8.875")	minimum (0.57”)	0.9374
MPC-89 (10x10A 4.8% Enrichment)		
nominal (6.01")	nominal (0.40")	0.9365
nominal (6.01")	minimum (0.38”)	0.9403
increased (6.03")	minimum (0.38”)	0.9396
minimum (5.99")	minimum (0.38”)	0.9417
minimum, including deformation (5.96")	minimum (0.38”)	0.9428

TABLE 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Box I.D.	Box Wall Thickness
MPC-37	minimum, including deformation(8.875")	minimum (0.57”)
MPC-89	minimum, including deformation (5.96")	minimum (0.38”)

TABLE 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

UO_2, DENSITY 10.686 g/cm^3 (97.5% of theoretical density of 10.96 g/cm^3)					
Nuclide	**Wgt. Fraction, 4.0 wt%**	**Wgt. Fraction, 4.5 wt%**	**Wgt. Fraction, 4.7 wt%**	**Wgt. Fraction, 4.8 wt%**	**Wgt. Fraction, 5.0 wt%**
8016	0.1185	0.1185	0.1185	0.1185	0.1185
92235	0.03526	0.03967	0.04143	0.04231	0.04408
92238	0.84624	0.84183	0.84007	0.83919	0.83742

WATER (unborated and borated), DENSITY 1.0 g/cm^3							
Nuclide	**Wgt. Fraction, 0 ppm**	**Wgt. Fraction, 1000 ppm**	**Wgt. Fraction, 1300 ppm**	**Wgt. Fraction, 1500 ppm**	**Wgt. Fraction, 1800 ppm**	**Wgt. Fraction, 2000 ppm**	**Wgt. Fraction, 2300 ppm**
5010	0.000E+00	1.800E-04	2.340E-04	2.700E-04	3.240E-04	3.600E-04	4.140E-04
5011	0.000E+00	8.200E-04	1.066E-03	1.230E-03	1.476E-03	1.640E-03	1.886E-03
1002	0.11190	0.11179	0.11175	0.11173	0.11170	0.11167	0.11164
8016	0.88810	0.88721	0.88695	0.88677	0.88650	0.88633	0.88606

METAMIC HT, 9% B_4C, DENSITY 2.6 g/cm^3	
Nuclide	**Wgt. Fraction**
13027	0.91
6000	0.01956
5010	0.01289
5011	0.05755

TABLE 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

ZR CLAD, DENSITY 6.55 g/cm^3	
Nuclide	**Wgt. Fraction**
40000	1.0
STAINLESS STEEL, DENSITY 7.84 g/cm^3	
Nuclide	**Wgt. Fraction**
24000	0.190
25055	0.020
26000	0.695
28000	0.095
ALUMINUM, DENSITY 2.7 g/cm^3	
Nuclide	**Wgt. Fraction**
13027	1.0
CONCRETE, DENSITY 2.35 g/cm^3	
Nuclide	**Wgt. Fraction**
1001	0.006
8016	0.500
11000	0.017
13027	0.048
14000	0.315
19000	0.019
20000	0.083
26000	0.012
LEAD, DENSITY 11.34 g/cm^3	
Nuclide	**Wgt. Fraction**
82000	1.0

TABLE 6.3.5

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT (FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

CASE	Contents centered (Reference)	Content moved towards center of basket		Content moved towards basket periphery	
	Maximum k_{eff}	**Maximum k_{eff}**	**k_{eff} Difference to Reference**	**Maximum k_{eff}**	**k_{eff} Difference to Reference**
MPC-37, Undamaged Fuel	0.9327	0.9380	0.0053	0.9143	-0.0184
MPC-37, Undamaged Fuel and Damaged Fuel/Fuel Debris (12 DFCs)	0.9260	0.9276	0.0016	0.9158	-0.0102
MPC-89, Undamaged Fuel	0.9369	0.9435	0.0066	0.9211	-0.0158
MPC-89, Undamaged Fuel and Damaged Fuel/Fuel Debris (16 DFCs)	0.9415	0.9451	0.0036	0.9301	-0.0114

TABLE 6.3.6

REACTIVITY EFFECTS GAPS IN BASKET CELL PLATES

Gaps in Metamic-HT	**MPC-37 (17x17B, 5.0% ENRICHMENT)**	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
None	0.9380	0.9435
0.06” every 10”	0.9382	0.9439

TABLE 6.3.7

RADIAL AND AXIAL DIMENSIONS OF THE HI-TRAC VW IN THE MCNP MODELS

Component / Material	**Thickness (Inches)**
Radial Direction (Inside to Outside)	
MPC Shell (Steel)	0.5
Water between MPC and HI-TRAC VW	0.125
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Lead	2.75
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Water Jacket	4.75
HI-TRAC VW Shell (Steel)	0.5
External Water Reflector	12
Axial Direction (Bottom to Top)	
External Water Reflector	12
HI-TRAC VW Bottom Lid (Steel)	5.5
MPC Base Plate (Steel)	3
Water	2
Active Fuel Region	150
Water	6
MPC Lid (Steel)	9
External Water Reflector	12

TABLE 6.3.8

MCNP CROSS SECTION SETS USED IN THE ANALYSES

Nuclide	MCNP Cross Section Set [6.1.4]
1001	62c
5010	66c
5011	66c
6000	66c
8016	62c
13027	62c
24000	50c
25055	62c
26000	55c
28000	50c
40000	66c
82000	50c
92235	69c
92238	69c

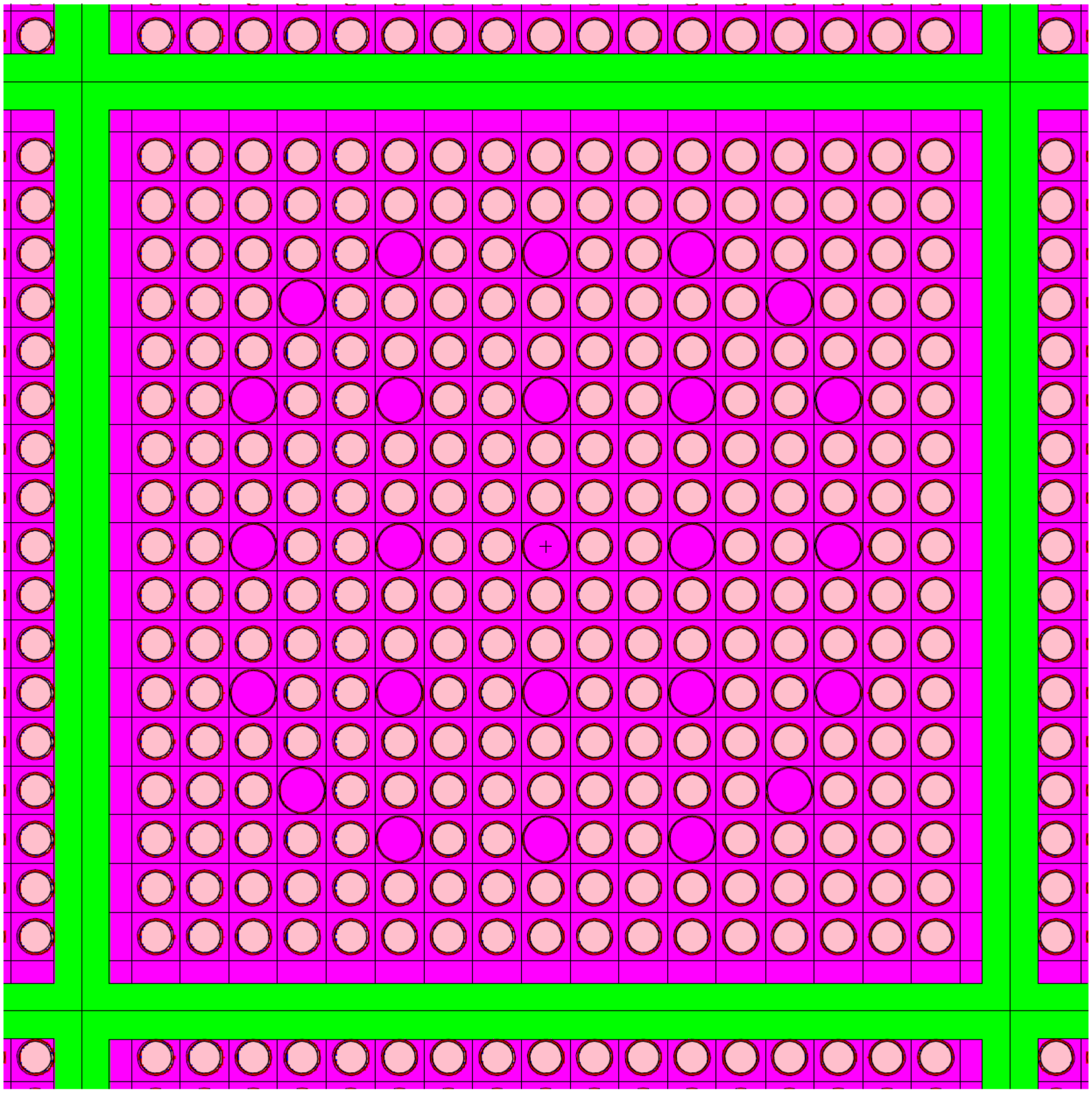

Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.1: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-37

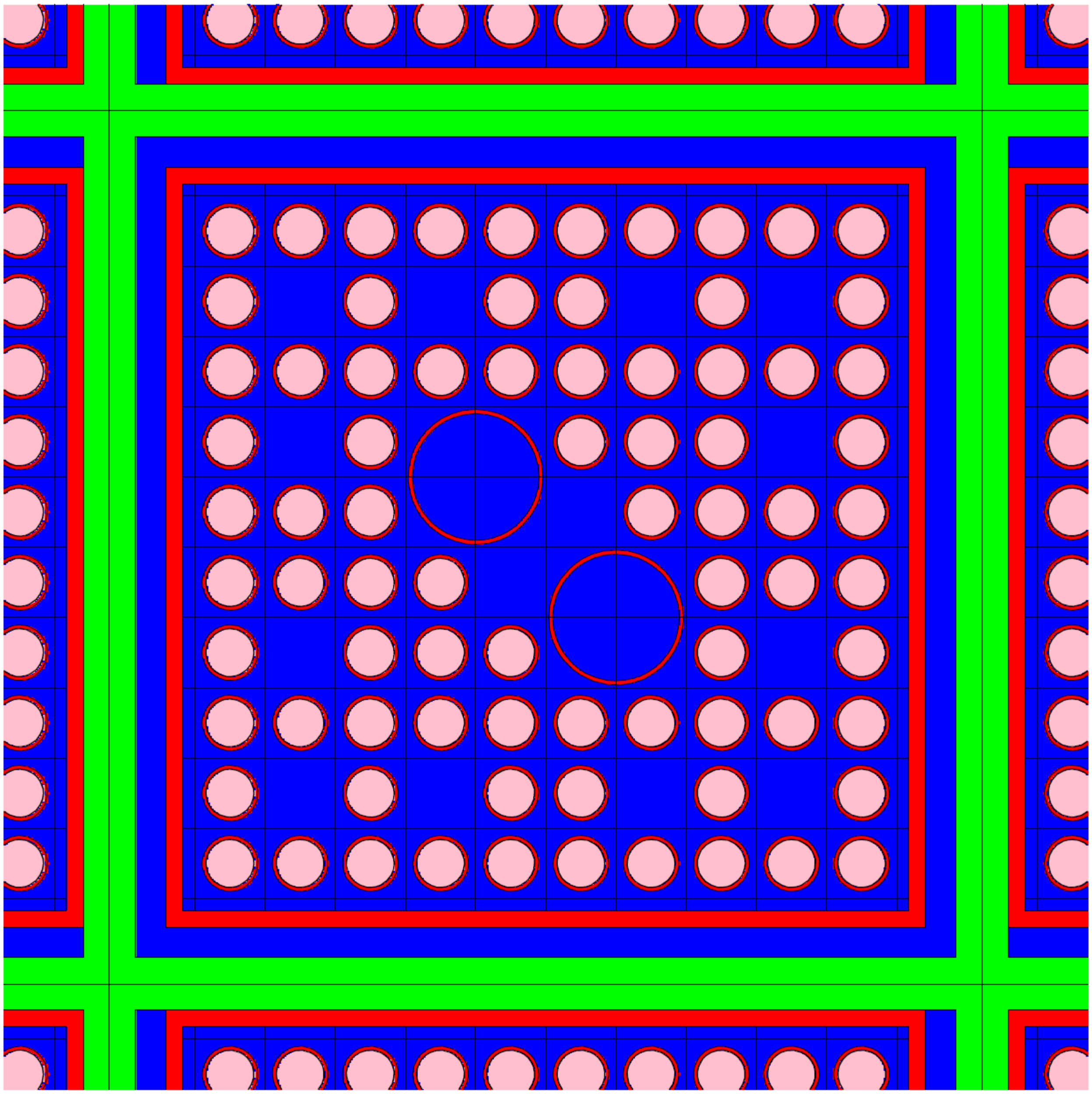

Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.2: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-89

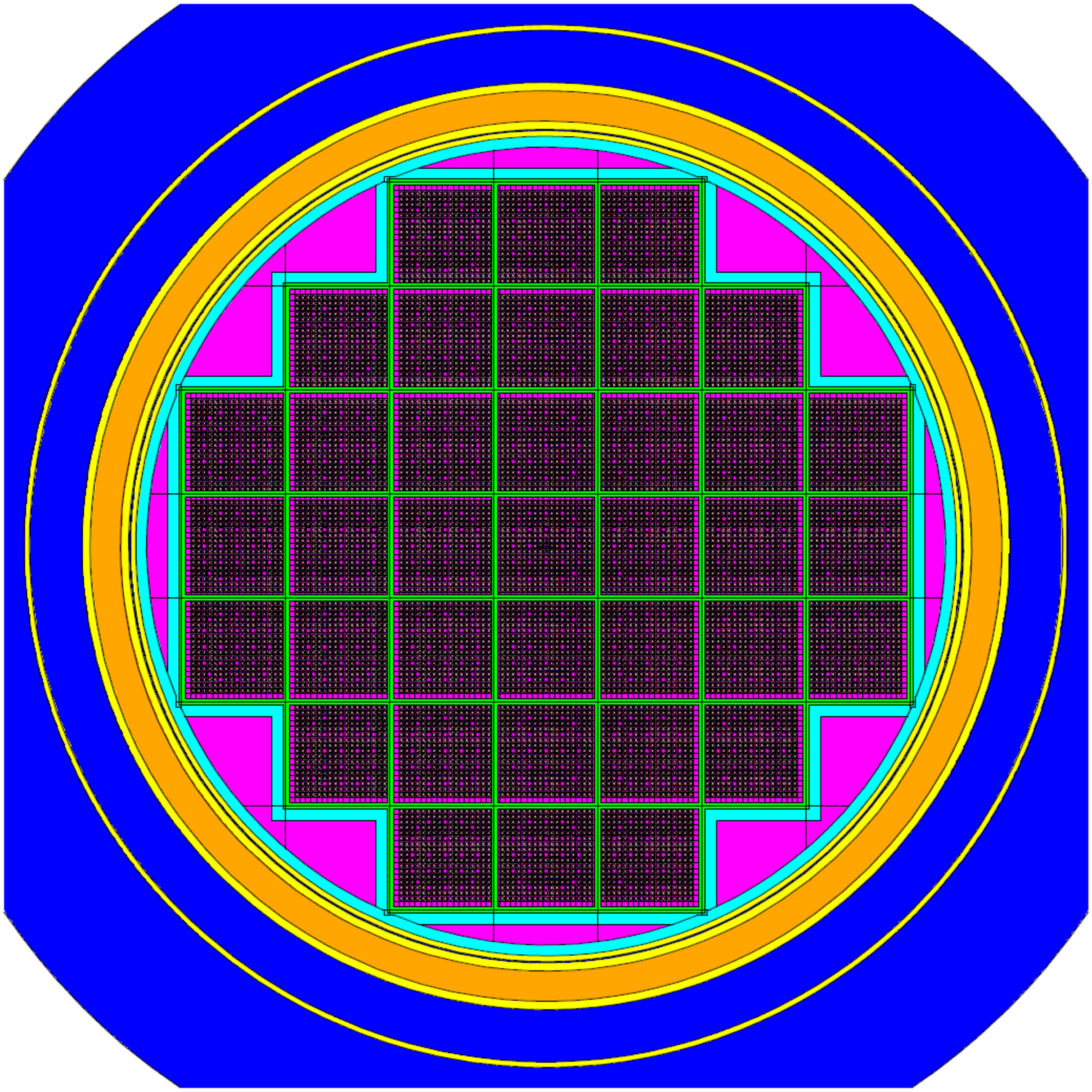

Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.3: Calculational Model (planar cross-section) of the MPC-37

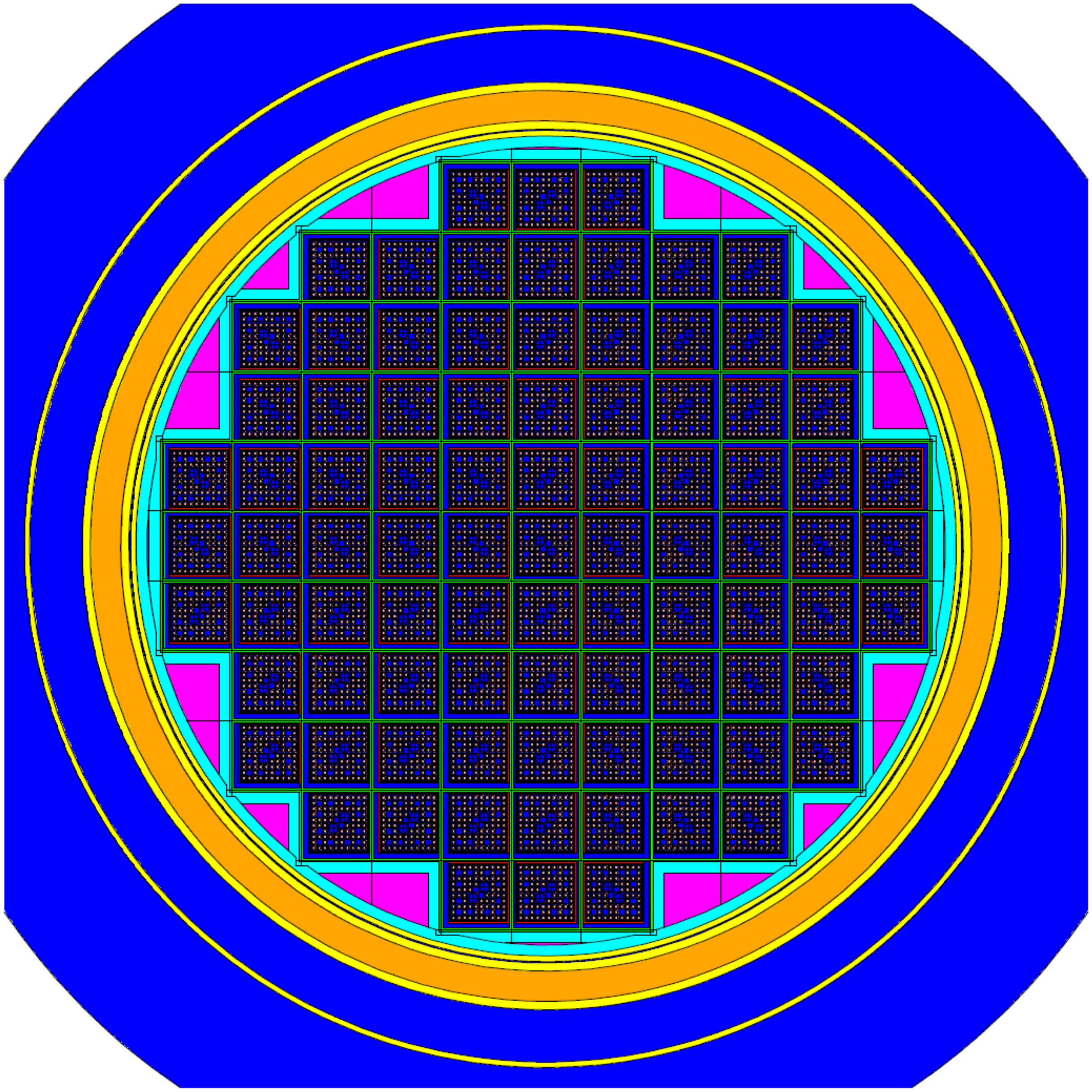

Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.4: Calculational Model (planar cross-section) of the MPC-89

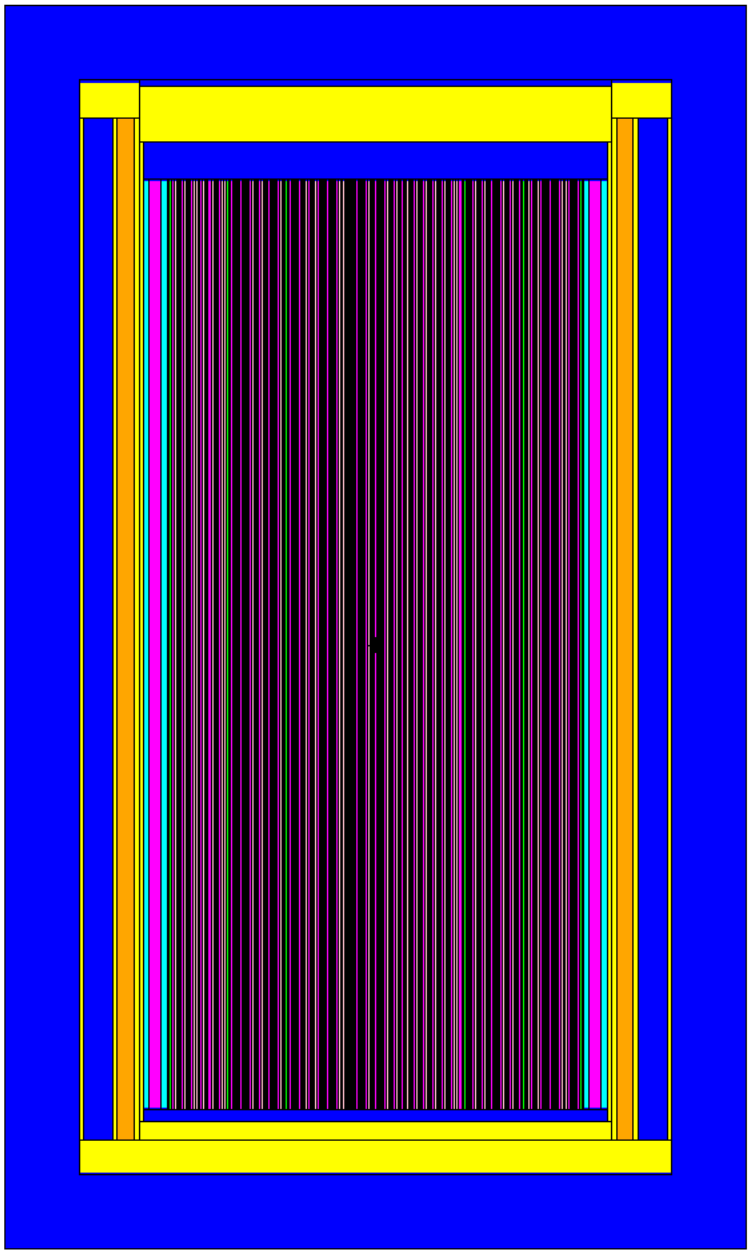

Figure generated directly from MCNP input file using the MCNP plot function. For axial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.5: Calculational Model in Axial Direction

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational Methodology

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP5 [6.1.4] developed at the Los Alamos National Laboratory. MCNP5 was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP5 calculations used continuous energy cross-section data distributed with the code [6.1.4].

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP5 criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. Based on this information, a minimum of 20,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). To verify that these parameters are sufficient, studies were performed where the number of particles per cycle and/or the number of skipped cycles were increased. The calculations are presented in Table 6.4.9, and show only small differences between the cases, with the statistical tolerance of those calculations. All calculations are therefore performed with the parameters stated above, except for some studies that are performed with 50000 neutrons per cycle for improved accuracy, and except for the calculations for the HI-STORM, which need less particles for convergence. Appendix 6.D provides sample input files for the MPC-37 and MPC-89 basket in the HI-STORM FW system.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Section 2.1. The calculations were based on the assumption that the HI-STORM FW system (HI-TRAC VW transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel assemblies and basket poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR and PWR assembly types, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion in Section 6.2.1 are listed below. Note that this approach is consistent with that used for the HI-STORM 100.

Internal and External Moderation (Section 6.4.2.1): The studies presented in Table 6.2.3 show that all assemblies essentially behave identical in respect to water moderation, specifically, that all assemblies are undermoderated. The principal effect of changes to the internal and external moderation would therefore be independent of the fuel type.

Partial Flooding (Section 6.4.2.2): The partial flooding of the basket, either in horizontal or vertical direction, reduces the amount of fuel that partakes effectively in the thermal fission process, while essentially maintaining the fuel-to-water ratio in the volume that is still flooded. This will therefore result in a reduction of the reactivity of the system (similar to that of the reduction of the active length), and due to the similarity of the fuel assemblies is not dependent on the specific fuel type.

Pellet-to-clad Gap (Section 6.4.2.3): As demonstrated by the studies shown in Section 6.2.1, all assemblies are undermoderated. Flooding the pellet-to-clad gap will therefore improve the moderation and therefore increase reactivity for all assembly types.

Preferential Flooding (Section 6.4.2.4): The only preferential flooding situation that may be credible is the flooding of the bottom section of the DFCs while the rest of the MPC internal cavity is already drained. In this condition, the undamaged assemblies have a negligible effect on the system reactivity since they are not flooded with water. The dominating effect is from the damaged fuel model in the DFCs. However, the damaged fuel model is conservatively based on an optimum moderated array of bare fuel rods in water, and therefore representative of all fuel types. The results are therefore applicable to all fuel types.

6.4.2.1 Internal and External Moderation

Calculations in this section demonstrate that the HI-STORM FW system remains subcritical for all credible conditions of moderation.

6.4.2.1.1 External Moderator Density

Calculations for the MPC designs with external moderators of various densities are shown in Table 6.4.1, all performed for the HI-TRAC VW and the MPC fully flooded. The results show that the maximum k_{eff} is essentially independent from the external water density. Nevertheless, all further evaluations are performed with full external water density.

6.4.2.1.2 Internal Moderator Density

In a definitive study, Cano, et al. [6.4.1] have demonstrated that the phenomenon of a peak in reactivity at low moderator densities (sometimes called "optimum" moderation) does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations were made to confirm that the phenomenon does not occur with low density water inside the casks.

Calculations for the MPC designs with internal moderators of various densities are shown in Table 6.4.5. Results show that in all cases the reactivity reduces with reducing water density, with both filled and voided guide and instrument tubes for PWR assemblies (see Section 6.4.7). All further calculations are therefore performed with full water density inside the MPCs.

6.4.2.2 Partial Flooding

Calculations in this section address partial flooding in the HI-STORM FW system and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cm^3) water and the remainder of the cask is filled with steam consisting of ordinary water at a low partial density (0.002 g/cm^3 or less), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

6.4.2.3 Clad Gap Flooding

As recommended by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e., different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell at the bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 12). For damaged fuel assemblies and fuel debris, the assemblies or debris are loaded into stainless steel Damaged Fuel Containers fitted with mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Preferential flooding of the MPC basket is therefore not possible.

However, when DFCs are present in the MPC, a condition could exist during the draining of the

MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. As a simplifying and conservative approach to model this condition it is assumed that the DFCs are completely flooded while the remainder of the MPC is only filled with steam consisting of ordinary water at partial density (0.002 g/cm^3 or less). Assuming this condition, the case resulting in the highest maximum k_{eff} for the fully flooded condition (see Subsection 6.4.4) is re-analyzed assuming the preferential flooding condition. Table 6.4.4 lists the maximum k_{eff} in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{eff} than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC Confinement Boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Subsection 6.4.4.

6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 12 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC VW transfer cask and minor damage to the concrete radiation shield for the HI-STORM FW storage cask, which have no adverse effect on the design parameters important to criticality safety, and to minor deformation of the basket geometry, which is already considered in the analyses for the normal conditions.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM FW system is in full compliance with the requirement of 10CRF72.124, which states that “before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.”

6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) and summarized in Section 6.1. To demonstrate the applicability of the HI-TRAC VW analyses, results of the design basis criticality safety calculations for the HI-TRAC VW cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-4) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

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

where:

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

⇒ $\boldsymbol{K_c}$ 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;

⇒ $\boldsymbol{\sigma_c}$ 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

⇒ $\boldsymbol{\sigma_B}$ is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

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

6.4.4 Damaged Fuel and Fuel Debris

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), the DFCs are covered by the neutron absorber even if they were to move axially.

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.

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.

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 for the MPC-37 and MPC-89, respectively. The highest maximum k_{eff} values correspond to a 16x16 array of bare fuel rods for the MPC-37,

and for an 11x11 array for the MPC-89. In all cases, the maximum k_{eff} is below the regulatory limit of 0.95.

For the HI-STORM 100, additional studies for damaged fuel assemblies were performed to further show that the above approach using arrays of bare fuel rods are bounding. The studies considered conditions including

- Fuel assemblies that are undamaged except for various numbers of missing rods
- Variations in the diameter of the bare fuel rods in the arrays
- Consolidated fuel assemblies with cladded rods
- Enrichment variations in BWR assemblies

Results of those studies were shown in the HI-STORM 100 FSAR, Table 6.4.8 and 6.4.9 and Figure 6.4.13 and 6.4.14 (undamaged and consolidated assemblies); HI-STORM 100 FSAR Table 6.4.12 and 6.4.13 (bare fuel rod diameter); and HI-STORM 100 FSAR Section 6.4.4.2.3 and Table 6.4.13 (enrichment variations). In all cases the results of those evaluations are equivalent to, or bounded by those for the bare fuel rods arrays. Since the generic approach of modeling damaged fuel and fuel debris is unchanged from the HI-STORM 100, these evaluations are still applicable and need not be re-performed for the HI-STORM FW.

6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of undamaged fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

6.4.6 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

6.4.7 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware as defined in Table 2.1.1 are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies, except for ITTRs, which are placed into the instrument tube.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide and instrument tubes, i.e., any absorption of the hardware is neglected. Table 6.4.10 shows results for all PWR assembly classes at 5% enrichment with filled and voided guide and instrument tubes. These results show that for all classes, the condition with filled guide and instrument tubes bound those, or are statistically equivalent to those, with voided guide and instrument tubes. For the higher soluble boron concentration required in the presence of damaged fuel, the same is shown in Table 6.4.5 (two columns on the right). In this case, only the bounding case (Assembly class 15x15F as undamaged fuel) was analyzed.

In summary, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

6.4.8 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM FW system. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a k_{eff} less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e., they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

6.4.9 Low Enriched, Channeled BWR fuel

The calculations in this subsection show that low enriched, channeled BWR fuel with indeterminable cladding condition is acceptable for loading in all storage locations of the MPC-89 without placing those fuel assemblies into DFCs, hence classifying those assemblies as undamaged. The main characteristics that must be assured are:

- The channel is present and attached to the fuel assembly in the standard fashion; and
- The channel is essentially undamaged; and
- The maximum planar average enrichment of the assembly is less than or equal to 3.3 wt% ^{235}U

This analysis covers older assemblies, where the cladding integrity is uncertain, and where a verification of the cladding condition is prohibitive. An example of this type of fuel is the so-called CILC fuel, which has potential corrosion-induced damaged to the cladding but does not have grossly breached spent fuel rods.

The presence of the essentially undamaged and attached channel confines the fuel rods to a limited volume and the low enrichment, required for all assemblies in the MPC, limits the reactivity of the fuel even under optimum moderation conditions. Due to the uncertain cladding condition, the analysis of this fuel follows essentially the same approach as that for the Damaged Fuel and Fuel Debris, i.e. bare fuel rod arrays of varying sizes are analyzed within the confines of the channel. This is an extremely conservative modeling approach for this condition, since reconfiguration is not expected and cladding would still be present. The results of this conservative analysis are listed in Table 6.4.8 and show that the system remains below the regulatory limit with these assemblies in all cells of the MPC-89, without DFCs.

These results confirm that even with unknown cladding condition the maximum k_{eff} values are below the regulatory limit when fully flooded and loaded with any of the BWR candidate fuel assemblies, therefore if the cladding is not grossly breached and the fuel assembly structurally sound it can be considered undamaged when loading in an MPC-89.

TABLE 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED EXTERNAL WATER DENSITIES

Water Density		Maximum k_{eff}	
Internal	**External**	**MPC-37 (17x17B, 5.0%)**	**MPC-89 (10x10A, 4.8%)**
100%	100%	0.9380	0.9435
100%	70%	0.9377	0.9432
100%	50%	0.9399	0.9439
100%	20%	0.9366	0.9428
100%	10%	0.9374	0.9437
100%	5%	0.9376	0.9435
100%	1%	0.9383	0.9435

TABLE 6.4.2

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

MPC-37 (17x17B, 5.0% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9175	0.8306
50	0.9325	0.9093
75	0.9357	0.9349
100	0.9380	0.9380
MPC-89 (10x10A, 4.8% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9204	0.8345
50	0.9382	0.9128
75	0.9416	0.9392
100	0.9435	0.9435

TABLE 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	**Maximum k_{eff}**	
	MPC-37 (17x17B, 5.0% ENRICHMENT)	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
dry	0.9335	0.9391
flooded with unborated water	0.9380	0.9435

TABLE 6.4.4

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	**Maximum k_{eff}**	
	Preferential Flooding	**Fully Flooded**
MPC-37 with 12 DFCs (5% Enrichment, Undamaged assembly 15x15F, 20x20 Bare Rod Array)	0.8705	0.9276
MPC-89 with 16 DFCs (4.8 % Enrichment, Undamaged assembly 10x10A, 9x9 Bare Rod Array)	0.8296	0.9464

TABLE 6.4.5

MAXIMUM k_{eff} VALUES WITH REDUCED WATER DENSITIES

Internal Water Density† in g/cm³	**Maximum k_{eff}**						
	MPC-89 10x10A, 4.8%	**MPC-37 (1500ppm) 17x17B, 4.0 %**		**MPC-37 (2000ppm) 17x17B, 5.0 %**		**MPC-37† (2300ppm) 15x15F and Damaged Fuel 5.0 %**	
Guide Tubes	N/A	filled	void	filled	void	filled	void
1.00	0.9435	0.9181	0.9071	0.9380	0.9292	0.9276	0.9265
0.99	0.9415	0.9181	0.9059	0.9367	0.9296	0.9271	0.9264
0.98	0.9391	0.9162	0.9054	0.9368	0.9279	0.9271	0.9257
0.97	0.9370	0.9166	0.9035	0.9364	0.9272	0.9265	0.9242
0.96	0.9345	0.9147	0.9005	0.9360	0.9265	0.9265	0.9232
0.95	0.9304	0.9148	0.9010	0.9356	0.9243	0.9253	0.9217
0.94	0.9280	0.9133	0.8995	0.9335	0.9238	0.9255	0.9225
0.93	0.9259	0.9128	0.8986	0.9355	0.9237	0.9263	0.9214
0.92	0.9232	0.9120	0.8955	0.9327	0.9203	0.9237	0.9204
0.91	0.9183	0.9105	0.8947	0.9335	0.9208	0.9229	0.9194
0.90	0.9169	0.9090	0.8934	0.9303	0.9189	0.9226	0.9169
0.85	0.9013	0.9042	0.8840	0.9272	0.9109	0.9190	0.9127
0.80	0.8850	0.8973	0.8733	0.9222	0.9022	0.9138	0.9040
0.70	0.8462	0.8813	0.8477	0.9068	0.8780	0.9000	0.8851
0.60	0.7980	0.8565	0.8132	0.8866	0.8478	0.8806	0.8571
0.40	0.6762	0.7876	0.7195	0.8244	0.7585	0.8192	0.7735
0.20	0.5268	0.6827	0.5806	0.7284	0.6298	0.7237	0.6517
0.10	0.4649	0.6206	0.5112	0.6698	0.5639	0.6669	0.5889

† External moderator is modeled at 100%.

† With undamaged and damaged fuel. All other cases with undamaged fuel only

TABLE 6.4.6

MAXIMUM k_{eff} VALUES IN THE MPC-37 WITH UNDAMAGED (15x15F) AND DAMAGED FUEL

Bare Rod Array inside the DFC	Maximum k_{eff}, 4.0 wt%	Maximum k_{eff}, 5.0 wt%
8x8	0.8883	0.9122
10x10	0.8899	0.9135
12x12	0.8910	0.9152
14x14	0.8945	0.9177
15x15	0.8966	0.9198
16x16	0.8982	0.9224
17x17	0.9003	0.9238
18x18	0.9027	0.9262
20x20	0.9032	0.9276
22x22	0.9023	0.926
24x24	0.9008	0.9239

TABLE 6.4.7

MAXIMUM k_{eff} VALUES IN THE MPC-89 WITH UNDAMAGED (10x10A) AND DAMAGED FUEL

Bare Rod Array inside the DFC	**Maximum k_{eff}, 4.8 wt% (planar average)**
4x4	0.9389
6x6	0.9411
8x8	0.9432
9x9	0.9464
10x10	0.9454
11x11	0.9451
12x12	0.9460
13x13	0.9453
14x14	0.9444
16x16	0.9429
18x18	0.9423

TABLE 6.4.8
MAXIMUM k_{eff} VALUES IN THE MPC-89 WITH LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED, BWR FUEL

Rod Array inside the Channel	Maximum k_{eff}
4x4	0.4018
6x6	0.7320
8x8	0.8999
9x9	0.9294
10x10	0.9325
11x11	0.9131
12x12	0.8762
13x13	0.8237
14x14	0.7606
16x16	0.6664
18x18	0.6334

TABLE 6.4.9

COMPARISON OF MCNP CONVERGENCE PARAMETERS

Calculation Parameters		**Maximum k_{eff}**	
Particles per Cycle	Skipped Cycles	**MPC-37 (17x17B, 5.0% ENRICHMENT)**	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
20,000	20	0.9380	0.9435
50,000	20	0.9376	0.9428
20,000	100	0.9387	0.9436
50,000	100	0.9379	0.9434

TABLE 6.4.10

COMPARISON OF MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 WITH CONDITIONS OF FILLED AND VOIDED GUIDE AND INSTRUMENT TUBES AT 5 % ENRICHMENT

Fuel Assembly Class	**Maximum k_{eff}, Filled Tubes**	**Maximum k_{eff}, Voided Tubes**
14x14A	0.8983	0.8887
14x14B	0.9172	0.9015
14x14C	0.9275	0.9277
15x15B	0.9311	0.9251
15x15C	0.9188	0.9134
15x15D	0.9421	0.9379
15x15E	0.9410	0.9365
15x15F	0.9455	0.9404
15x15H	0.9325	0.9317
15x15I	0.9357	0.9362
16x16A	0.9366	0.9320
17x17A	0.9194	0.9135
17x17B	0.9380	0.9292
17x17C	0.9424	0.9345
17x17D	0.9384	0.9293
17x17E	0.9392	0.9314

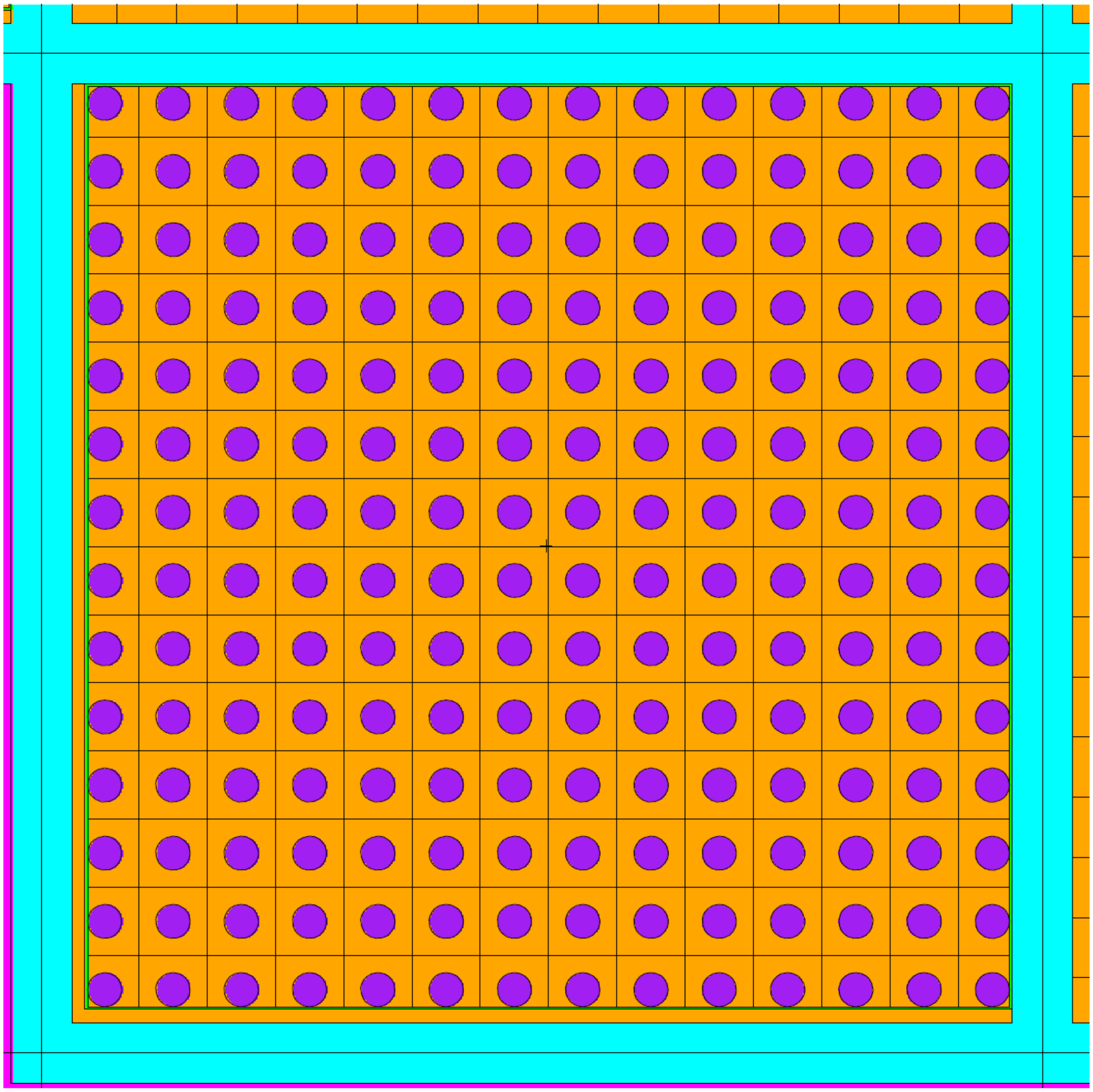

Figure 6.4.1: Calculational Model (planar cross-section) of a DFC in a MPC-37 cell with a 14x14 array of bare fuel rods

6.5 CRITICALITY BENCHMARK EXPERIMENTS

Benchmark calculations have been made on selected critical experiments, chosen, insofar as possible, to bound the range of variables in the cask designs. The most important parameters are (1) the enrichment, (2) cell spacing, and (3) the ^{10}B loading of the neutron absorber panels. Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included. No significant trends were evident in the benchmark calculations or the derived bias. Detailed benchmark calculations are presented in Appendix 6.A.

The benchmark calculations were performed with the same computer codes and cross-section data, described in Section 6.4, that were used to calculate the k_{eff} values for the cask. Further, all calculations were performed on the same computer hardware (personal computers).

6.6 REGULATORY COMPLIANCE

This section documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STORM FW system is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety, as well as the limiting fuel characteristics, are described in sufficient detail in this section to enable an evaluation of their effectiveness.

The HI-STORM FW system is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poison has shown that they will remain effective for a storage period greater than 60 years, and there is no credible way to lose it; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10CFR72.124(b).

The criticality evaluation has demonstrated that the cask will enable the storage of spent fuel for a minimum of 60 years with an adequate margin of safety. Further, the evaluation has demonstrated that the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, the HI-STORM FW system is in full compliance with the double contingency requirements of 10CRF72.124. Therefore, it is concluded that the criticality design features for the HI-STORM FW system are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The criticality evaluation provides reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel.

6.7 REFERENCES

[6.0.1] HI-STORM 100 FSAR, NRC Docket 72-1014, Holtec Report HI-2002444, Latest revision

[6.0.2] "Criticality Analyses for the HI-STORM FW System", Holtec Report HI-2094432 Rev.0 (proprietary)

[6.1.1] NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington, D.C., January 1997.

[6.1.2] 10CFR72.124, "Criteria For Nuclear Criticality Safety."

[6.1.3] not used

[6.1.4] "MCNP - A General Monte Carlo N-Particle Transport Code, Version 5"; Los Alamos National Laboratory, LA-UR-03-1987 (2003).

[6.1.5] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

[6.1.6] "CASMO-4 Methodology", Studsvik/SOA-95/2, Rev. 0, 1995.

"CASMO-4 A Fuel Assembly Burnup Program, Users Manual," SSP-01/400, Rev. 1, Studsvik Scandpower, Inc., 2001.

"CASMO-4 Benchmark Against Critical Experiments", Studsvik/SOA-94/13, Studsvik of America, 1995.

[6.4.1] J.M. Cano, R. Caro, and J.M Martinez-Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," Nucl. Technol., 48, 251-260, (1980).

APPENDIX 6.A: BENCHMARK CALCULATIONS

Withheld in Accordance with 10 CFR 2.390

APPENDIX 6.B: MISCELLANEOUS INFORMATION

6.B.1 Sample Input File MPC-37

Withheld in Accordance with 10 CFR 2.390

6.B.2 Sample Input File MPC-89

Withheld in Accordance with 10 CFR 2.390

6.B.3 Analyzed Distributed Enrichment Patterns

Withheld in Accordance with 10 CFR 2.390

6.B.4 Assembly Cross Sections

Withheld in Accordance with 10 CFR 2.390