APP-GW-GLR-029
Revision 0

June 2006

AP1000 Standard Combined License Technical Report

Spent Fuel Storage Racks Criticality Analysis

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1 INTRODUCTION

This report summarizes the criticality analysis for the AP1000 Spent Fuel Storage Racks. The AP1000 Spent Fuel Storage Racks are used to store fresh fuel assemblies prior to loading them into the reactor core and spent fuel assemblies after they have been discharged from the reactor core. The requirements for this analysis are identified in the AP1000 Design Control Document (DCD), subsections 4.3.2.6 and 9.1.2.3 (Reference 1). The completion of this analysis is identified as Combined Operating License (COL) Information Item 9.1-4 (Final Safety Evaluation Report [Reference 2] Action Item 9.1.6-4) in DCD subsection 9.1.6 to be completed by the Combined License applicant:

> COL Information Item 9.1-4: "Perform a confirmatory criticality analysis for the spent fuel rack, as described in AP1000 DCD Subsection 9.1.2.3. This analysis should address the degradation of integral neutron absorbing material; in the spent fuel pool storage racks as identified in GL-96-04, and assess the integral neutron absorbing material capability to maintain a 5-percent subcriticality margin."

Associated with the closure of this COL information item, the descriptions of the criticality analysis as discussed in DCD subsections 4.3.2.6.1, 4.3.2.6.2, 4.3.5, 9.1.2.3, 9.1.6, and 9.1.7 are updated.

This COLA Technical Report closes this COL Information Item. The calculation "AP1000 Spent Fuel Storage Racks Criticality Analysis" (Reference 3) is available for U.S. Nuclear Regulatory Commission (U.S. NRC) audit. The structural/seismic analysis of the AP1000 Spent Fuel Storage Racks is presented in AP1000 COLA Technical Report, APP-GW-GLR-033, "Spent Fuel Storage Rack Structural/Seismic Analysis" (Reference 4).

2 TECHNICAL BACKGROUND

This report presents the results of criticality analyses for the AP1000 spent fuel pool racks with credit for assembly burnup, ^{241}Pu decay, and soluble boron. The primary objectives of this calculation are as follows:

- To demonstrate that fresh 5.0 weight-percent U-235 fuel assemblies may be safely stored in the Region 1 "All-Cell" fuel assembly storage configuration.
- To determine the minimum fuel assembly burnup versus initial enrichment limits required for safe storage of fuel assemblies in the Region 2 "All-Cell" and "1-out-of-4 5.0 weight-percent Fresh" storage configurations with credit for 5, 10, 15, and 20 years of ^{241}Pu decay.
- To determine the assembly loading requirements at the interface between storage configurations.
- To determine the amount of soluble boron required to maintain k-effective (K_{eff}) less than or equal to 0.95 in the spent fuel pools, including all biases and uncertainties, assuming the most limiting plausible reactivity accident.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 5 and uses analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation (Reference 6). The methodology cited in Reference 5 was reviewed and approved by the U.S. NRC. The methodology used in this analysis and in cited in Reference 5 uses axially distributed burnups to represent discharged fuel assemblies.

Westinghouse has performed the criticality analysis of the AP1000 spent fuel racks and not the manufacturing vendor for the spent fuel racks as stated in the DCD.

2.1 DESIGN CRITERIA

The design criteria are consistent with General Design Criterion (GDC) 62, Reference 7, and U.S. NRC guidance given in Reference 8. The applicable 10 CFR Part 50.68 requirements are as follows:

1. The maximum K_{eff} value, including all biases and uncertainties must be less than 0.95 with soluble boron credit and less than 1.0 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 4 of Paragraph b. Note that the specific terminology is as follows:

   "If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

2. The maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent U-235. Note this design criterion is provided in 10 CFR Part 50.68, Item 7 of Paragraph b. Note that the specific terminology is as follows:

   "The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight."

Section 2.2 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria is as follows:

1. Determine the fresh and spent fuel storage configurations using no soluble boron conditions such that the 95/95 upper tolerance limit value of K_{eff}, including applicable biases and uncertainties, is less than 0.995. This is accomplished with infinite arrays of either fresh or spent fuel assembly configurations. Note that the actual U.S. NRC K_{eff} limit for this condition is unity. Therefore, an additional margin of 0.005 ΔK_{eff} units is included in the analysis results.

2. Determine the amount (ppm) of soluble boron necessary to reduce the K_{eff} value of all storage configurations by at least 0.05 ΔK_{eff} units. This is accomplished by constructing a KENO model

for the entire spent fuel pool, which includes the storage configurations that are least sensitive to changes in soluble boron concentration. As an example, storage configurations that contain depleted fuel assemblies (and represented by depleted isotopics) are less reactivity-sensitive to changes in soluble boron concentration than a fuel assembly represented by zero burnup and relatively low initial fuel enrichment.

3. Determine the amount of soluble boron necessary to compensate for 5% of the maximum burnup credited in any storage configuration. In addition, determine the amount of soluble boron necessary to account for a reactivity depletion uncertainty of 1.0% ΔK_{eff} per 30,000 MWD/MTU of credited fuel burnup. This is accomplished by multiplying this derivative by the maximum burnup credited in any storage configuration and converting to soluble boron using the data generated in Step 2.

4. Determine the largest increase in reactivity caused by postulated accidents and the corresponding amount of soluble boron needed to offset this reactivity increase.

An alternative form of expressing the soluble boron requirements is given in Reference 6. The final soluble boron credit (SBC) requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where,

SBC_{TOTAL} = total soluble boron credit requirement (ppm)

$SBC_{95/95}$ = soluble boron requirement for 95/95 K_{eff} less than or equal to 0.95 (ppm)

SBC_{RE} = soluble boron required to account for burnup and reactivity uncertainties (ppm)

SBC_{PA} = soluble boron required to offset accident conditions (ppm)

For the analyses, minimum burnup limits established for fuel assemblies to be stored in the storage configurations racks include burnup credit established in a manner that takes into account approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

The soluble boron credit methodology provides additional reactivity margin in the spent fuel storage analyses, which may then be used to implement added flexibility in storage criteria.

The fuel assembly type used for all the analyses is the Westinghouse AP1000 17x17 assembly design. The most reactive spent fuel pool temperature (with full moderator density of 1 g/cc) is used for each fuel assembly storage configuration so that the analysis results are valid over the nominal spent fuel temperature range (50° to 140°F) (Reference 9).

The reactivity characteristics of the storage racks were evaluated using infinite lattice analyses; this environment was used in the evaluation of the burnup limits versus initial enrichment as well as the

evaluation of physical tolerances and uncertainties. A full spent fuel pool model was also used to evaluate soluble boron worth, the reactivity worth of postulated accidents, and the multiplication factor for the zero soluble boron condition.

2.2 METHODOLOGY AND ASSUMPTIONS

2.2.1 Methodology

This subsection describes the methodology used to ensure the criticality safety of the AP1000 spent fuel pool and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology uses: (1) SCALE-PC, a personal computer version of the SCALE-4.4a code system, as documented in Reference 10 with the updated SCALE-4.4a version of the 44-group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross-section library, and (2) the two-dimensional Discrete Integral Transport (DIT) code (Reference 11) with an Evaluated Nuclear Data File, Version 6 (ENDF/B-VI) neutron cross-section library.

SCALE-PC was used for calculations involving infinite arrays for the storage configurations in the spent fuel pool. In addition, it was used in a full pool representation of the storage racks to evaluate soluble boron worth and postulated accidents.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. All references to KENO mean the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

2.2.1.1 SCALE-PC

The SCALE system was developed for the U.S. NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on personal computers.

2.2.1.2 Validation of SCALE-PC

Validation of SCALE-PC for fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs: the Babcock & Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel (Reference 12), and the Pacific Northwest Laboratory (PNL) Program in support of the design of Fuel Shipping and Storage Configurations. References 13 and 14, as well as several of the relevant thermal experiment evaluations in Reference 15, were useful in updating pertinent experimental data for the PNL experiments.

The validation of SCALE-PC was limited to the 44-group library provided with the SCALE-PC version 4.4a package (Reference 16). The 238-group library, which is used for the off-nominal temperature cases, was further validated by comparing the results from identical cases performed with the 44-group library and confirming that the results agreed within the statistical uncertainty.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIA. These analyses used measured critical data, rather than the extrapolated configurations to a fixed critical water height reported in Reference 12, to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations used variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or borated aluminum (B/Al) plates of different boron contents in the water channels between rod clusters were measured. Table 2-1 summarizes the results of these analyses performed with both the 44-group and 238-group libraries.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blackness. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 2-2 summarizes the results of these analyses performed with both the 44-group and 238-group libraries.

The approach used for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 17. For a given KENO-calculated value of K_{eff} and associated one sigma uncertainty, the magnitude of k95/95 is computed by the equation below. By this definition, there is a 95-percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than k95/95.

$$k_{95/95} = k_{keno} + \Delta k_{bias} + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{1/2}$$

where,

k_{keno}	is the KENO-calculated multiplication factor.
Δk_{bias}	is the mean calculational method bias.
$M_{95/95}$	is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses, and is obtained from the tables of Reference 18.
σ_m^2	is the mean calculational method variance deduced from the validation analyses.
σ_{KENO}^2	is the square of the KENO standard deviation.

The equation for the mean calculational methods bias is as follows:

$$\Delta k_{bias} = \frac{1}{n}\sum_{i=1}^{n}\left(1 - k_i\right)$$

Where,

k_i is the ith value of the multiplication factor for the validation lattices of interest.

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows:

$$\sigma_m^2 = \frac{n\sum_1^n \frac{(k_i - k_{ave})^2}{\sigma_i^2}}{(n-1)\sum_1^n \frac{1}{\sigma_i^2}} - \sigma_{ave}^2$$

where k_{ave} is given by the following equation:

$$k_{ave} = \frac{\sum_1^n \frac{k_i}{\sigma_i^2}}{\sum_1^n \frac{1}{\sigma_i^2}},$$

σ_{ave}^2 is given by the following equation:

$$\sigma_{ave}^2 = \frac{\sum_1^n \sigma_i^2 G_i}{\sum_1^n G_i},$$

G_i is the number of generations.

For this bias evaluation, the data points of Table 2-1 and Table 2-2 are pooled into a single group from the 44-group library calculations. With this approach, the mean calculational methods bias, Δk_{bias}, and the mean calculational variance, σ_m^2, calculated by the equations given above, were determined to be 0.00310 and $(0.00285)^2$, respectively. The magnitude of $M_{95/95}$ is obtained from Reference 18 for the total number of pooled data points, 30.

The magnitude of k95/95 is given by the following equation for SCALE 4.4a KENO analyses using the 44-group ENDF/B-V neutron cross-section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance:

$$k_{95/95} = k_{KENO} + 0.00310 + 2.22\left[(0.00285)^2 + \sigma_{KENO}^2\right]^{\frac{1}{2}}$$

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier for the 44-group library were deduced as 0.00310, $(0.00285)^2$ and 2.22, respectively.

2.2.1.3 Application to Fuel Storage Pool Calculations

As noted previously, the CSAS25 control module was used to execute the functional modules within SCALE-PC. The CSAS25 control module was used to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were used in the SCALE-PC analyses consistent with the design input given in Section 2.2; these data are listed in Table 2-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data in Table 2-3. For burnt fuel representations, the fuel isotopics were derived from the DIT code as described in the following subsection.

2.2.1.4 DIT Code

The DIT code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the ENDF/B-VI. Cross sections have been collapsed into an 89-group structure, which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually four broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the assembly in- or out-leakage. Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains; 171 fission products; the gadolinium, erbium, and boron depletable absorbers; and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density, and boron concentration dependencies needed for three-dimensional (3D) calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross-section library are used in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each complete set of fuel isotopics is reduced to a smaller set of burnt fuel isotopics at specified time points after discharge. The latter burnt fuel representation includes the following nuclides: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{149}Sm,

^{16}O, and ^{10}B. The DIT code lists the ^{149}Sm isotopics for both ^{149}Sm and ^{149D}Sm (a metastable isomer). Since ^{149}Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based on pin cell averaged values. The input to KENO calculations requires that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of B^{10} supplied to KENO is such that the KENO and DIT fuel assembly k_∞ values (at room temperature and unborated conditions) agree to within one sigma of the KENO calculation.

2.2.2 Assumptions

The assumptions for the AP1000 Spent Fuel Storage Racks Criticality Analysis are as follows:

- The Westinghouse AP1000 17x17 assembly was modeled as the design basis fuel assembly to conservatively represent all fresh and depleted fuel assemblies.

- Fresh and depleted fuel assemblies were conservatively modeled with a UO_2 density equal to 10.686 g/cm^3 (97.5% of theoretical density). This translates into a pellet density equal 98.6% of theoretical density with a 1.1% dishing (void) fraction.

- All fuel assemblies, fresh and depleted, were conservatively modeled as containing solid right cylindrical pellets and as uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs that incorporate lower enrichment blanket or annular pellets.

- The design basis limit for K_{eff} at the zero soluble boron condition was conservatively reduced from 1.0 to 0.995 for this analysis.

- Metamic® poison panels were conservatively modeled with the minimum B-10 areal density. In addition, the Metamic panels were modeled as 168 inches long. Note that the Metamic panels will be manufactured with an overall length equal to 172 inches and overlap the ends of the active fuel by 2 inches.

2.3 DESIGN INPUT

This section provides a brief description of the AP1000 spent fuel pool and storage racks with the objective of establishing a basis for the analytical models used in the criticality analyses described in Section 2.4.

2.3.1 17x17 Fuel Assembly Description

The fuel assembly modeled in this analysis is the Westinghouse AP1000 17x17 design as Figure 2-1 depicts in cross section. Table 2-4 lists the Westinghouse AP1000 17x17 fuel assembly parameters as modeled in this analysis. Note that this last entry in Table 2-4 provides the relative elevation of the

bottom of the active fuel to the top of the base plate. This value was used to set the bottom elevation of the Metamic poison panel (172 inches) so that the active fuel length (168 inches) was conservatively covered with a 4-inch overlap (2-inch overlap on each end of the active fuel).

2.3.2 Design Input from HOLTEC

Design data related to the AP1000 Spent Fuel Pool and storage racks required to develop the KENO models were obtained from Holtec International Drawing 4743, Revision 0 (Page 2 of 6) (Reference 19). This drawing is summarized in Figure 2-2. In addition, Holtec supplied Westinghouse a "PWR Rack Data Sheet" (Reference 19), which provides detailed storage cell design information concerning the Region 1 and Region 2 spent fuel storage racks and the five Defective Fuel Assembly Storage Cells. Tables 2-5, 2-6, and 2-7 contain the design information from the "PWR Rack Data Sheet" that pertains to the Region 1 storage cells, Region 2 storage cells, and the Defective Fuel Assembly Storage Cells respectively.

2.3.3 AP1000 Spent Fuel Pool and Storage Racks Description

Figure 2-2 presents the layout of the AP1000 spent fuel pool. The total storage capacity is 889 locations. The AP1000 spent fuel pool contains three Region 1 rack modules, five Region 2 rack modules, and five individual Defective Fuel Assembly Storage Cells.

The Region 1 modules are all 9x9 arrays of storage cells. They are designated Modules A1, A2, and A3. Note that the Region 1 modules are located along the west wall of the AP1000 spent fuel pool.

There are four Region 2 modules which are 12x11 arrays of storage cells. They are designated Modules B1, B2, B3, and B4. These modules are located along the east wall of the AP1000 spent fuel pool.

There is a single 12x10 (-7) Region 2 module. It is designated Module C1. (Note that the term "12x10 (-7)" means a 12x10 array that is missing seven storage cells. The seven storage cells removed from the 12x10 array provide space for the five Defective Fuel Assembly Storage Cells.)

The five Defective Fuel Assembly Storage Cells are located between the Region 2 12x10 (-7) module and the west wall of the AP1000 spent fuel pool.

2.3.3.1 Region 1 Storage Cell Description

The Region 1 storage cells are centered on a pitch of 10.9 inches. Each storage cell consists of a stainless steel canister, which has a nominal inside dimension of 8.8 inches and is 0.075 inch thick. Metamic panels are attached to the outside surfaces of the canister in all Region 1 storage cells except for the surfaces directly facing the west wall of the spent fuel pool. Each Metamic poison panel is held in place and is centered on the surface of the stainless steel canister by an outer stainless steel sheathing panel. There is a small void space (0.012 inch) between the sheathing and the Metamic panel. The dimensions of the Metamic poison panel are 7.5 inches wide by 0.106 inch thick. The sheathing panels on interior storage canisters are 0.035 inch thick.

Each Region 1 storage cell is 199.5 inches long, and each rests on top of a base plate whose top is 5 inches above the concrete floor. Note that each Metamic poison panel is 172 inches long and has a bottom elevation that is 6.23 inches above the top of the base plate. The bottom elevation of the Metamic poison panel was positioned to be 2 inches lower than the bottom elevation of the active fuel. The Metamic poison material used in the Region 1 storage cells is a mixture of B_4C and Al with a nominal B_4C concentration equal to 31.0 weight-percent, and uses natural Boron isotopics (i.e., not enriched B^{10}). The Region 1 storage cell dimensions and tolerances are summarized in Table 2-5.

2.3.3.2 Region 2 Storage Cell Description

The Region 2 storage cells are formed by welding open stainless steel canisters together at the corners. Therefore, the Region 2 storage cells are a combination of individual canister storage cells and "developed" storage cells. The "developed" storage cells result from the welding process. As an example, the welding of four canisters at the corners of each canister produces a single "developed" storage cell at the center of the four canisters. Each Region 2 stainless steel canister has an inside dimension of 8.8 inches and is 0.075 inch thick. The center-to-center spacing between storage cells is 9.028 inches.

Metamic panels are attached to the outside surfaces of each stainless steel canister except for the surfaces directly facing the walls of the spent fuel pool. Each Metamic poison panel is held in place and is centered on the surface of the stainless steel canister by an outer stainless steel sheathing panel. There is a small void space (0.012 inch) between the sheathing and the Metamic panel. The dimensions of the Metamic poison panel are 7.5 inches wide by 0.106 inch thick. The sheathing panels on interior storage canisters are 0.035 inch thick.

Each Region 2 storage cell is 199.5 inches long and rests on top of a base plate whose top is 5 inches above the concrete floor. Note that each Metamic poison panel is 172 inches long and has a bottom elevation that is 6.23 inches above the top of the base plate. The bottom elevation of the Metamic poison panel was positioned to be 2 inches lower than the bottom elevation of the active fuel. The Metamic poison material used in the Region 2 storage cells is a mixture of B_4C and Al with a nominal B_4C concentration equal to 31.0 weight-percent, and uses natural Boron isotopics (i.e., not enriched B^{10}). The Region 2 storage cell dimensions are summarized in Table 2-6.

2.3.3.3 Defective Fuel Assembly Storage Cell

The Defective Fuel Assembly Storage Cells consist of open stainless canisters with an inside dimension of 10.25 inches and a thickness of 0.075 inch. The center-to-center spacing between storage cells is 10.478 inches. Metamic panels are attached to the surfaces of the canister, which face another canister. Each Metamic poison panel is held in place and is centered on the surface of the stainless steel canister by an outer stainless steel sheathing panel. There is a small void space (0.012 inch) between the sheathing and the Metamic panel. The dimensions of the Metamic poison panel are 7.5 inches wide by 0.106 inch thick. The sheathing panels on interior facing walls are 0.035 inch thick.

Each Defective Fuel Assembly Storage Cell is 199.5 inches long, and rests on top of a base plate whose top is 5 inches above the concrete floor. Note that each Metamic poison panel is 172 inches long, and has a bottom elevation that is 6.23 inches above the top of the base plate. The bottom elevation of the

Metamic poison panel was positioned to be 2 inches lower than the bottom elevation of the active fuel. The Metamic poison material employed in the Defective Fuel Assembly Storage Cells is a mixture of B_4C and Al with a nominal B_4C concentration equal to 31.0 weight-percent and uses natural Boron isotopics (i.e., not enriched B^{10}). The Defective Fuel Assembly Storage Cell dimensions are summarized in Table 2-7.

2.4 ANALYSIS

2.4.1 KENO Models for Spent Fuel Pool Storage Configurations

The AP1000 Spent Fuel Storage Racks use three different fuel assembly storage configurations: the "All-Cell" storage configuration in Region 1, and the "All-Cell" and "1-out-of-4 5.0 weight-percent Fresh" storage configurations in Region 2. Note the Region 1 and Region 2 storage racks use different storage cell configurations. The KENO models of these storage configurations are provided in the following subsections. Modeling of the axial burnup distributions for the spent fuel is described in subsection 2.4.3.

The fuel assemblies modeled by KENO represent the Westinghouse AP1000 17x17 design. This assembly design is neutronically similar to the Westinghouse Standard 17x17 assembly design with the exception of the fuel assembly length. The Westinghouse AP1000 17x17 fuel assembly has an active fuel length of 168 inches, and the Westinghouse Standard 17x17 design has an active fuel length of 144 inches. Table 2-4 provides the fuel assembly parameters as modeled in this analysis.

The enrichment of fresh fuel pellets is modeled as 5.0 weight-percent U-235, and the UO_2 density is modeled equal to 10.686 g/cc (97.5% of theoretical density). The fuel pellets in a fuel rod are modeled as a fully enriched right solid cylinder which is 168 inches tall. This assumption conservatively bounds fuel rod designs that incorporate annular and or lower enrichment fuel pellets such as those used for axial blankets.

2.4.1.1 KENO Model for Region 1 "All-Cell" Storage Configuration

The Region 1 "All-Cell" fuel assembly storage configuration is modeled in KENO as a square cell with a pitch of 10.9 inches. The fuel assembly inside the Region 1 storage cell is a fresh Westinghouse AP1000 17x17 design with an enrichment equal to 5.0 weight-percent U-235. The fresh fuel assembly is positioned inside a stainless steel canister with an inside dimension of 8.8 inches and is 0.075 inch thick. Metamic poison panels are modeled as attached to the outside of the stainless steel canister. The Metamic panels are 7.5 inches wide and 0.106 inch thick. Stainless steel sheathing panels hold the Metamic panels in place and are modeled as 0.035 inch thick. Note that there is a small gap between the Metamic poison panels and the stainless steel sheathing that is modeled as a void 0.012 inch thick. The stainless steel canister, fuel assembly, Metamic panels, and stainless steel sheathing are all modeled as 168 inches tall and are elevated above the concrete floor by 13.23 inches. Note that the Metamic panels will be manufactured with an overall length of 172 inches and will overlap the ends of the fuel assembly by 2 inches.

A 2-foot water reflector is modeled above the storage cell geometry and a 13.23-inch water reflector is modeled below the storage cell geometry. Reflective boundary conditions are applied to the outside

surfaces of the overall Region 1 storage cell geometry. This simulates an infinitely repeating array. The pool water is simulated to be full density (1 g/cm^3) at room temperature (20°C).

2.4.1.2 KENO Model for Region 2 "All-Cell" Storage Configuration

The Region 2 "All-Cell" fuel assembly storage configuration is modeled in KENO as a 12x12 array of "stainless steel canister" storage cells and "developed" storage cells. The pitch modeled between storage cells is 9.028 inches.

The KENO model for a 12x12 array of Region 2 storage cells was developed to accurately represent the construction of these storage cells. Note that the largest Region 2 module is a 12x11 array of storage cells. Therefore, the KENO results for a 12x12 array of storage cells (with periodic boundary conditions) will bound the expected results for the largest Region 2 module.

The fuel assembly inside each storage cell is a depleted Westinghouse AP1000 17x17 assembly. The stainless steel canister is modeled with an inside dimension of 8.8 inches and is 0.075 inch thick. Metamic poison panels are modeled as attached to the outside of the stainless steel canister. The Metamic panels are modeled as 7.5 inches wide and 0.106 inch thick. Stainless steel sheathing panels hold the Metamic panels in place and are modeled as 0.035 inch thick. Note that there is a small gap between the Metamic poison panels and the stainless steel sheathing that is modeled as a void 0.012 inches thick. The Metamic poison material is modeled with the minimum B-10 areal loading. The stainless steel canister, fuel assembly, Metamic panels, and stainless steel sheathing are all modeled as 168 inches tall and are elevated above the concrete floor by 13.23 inches.

A 2-foot water reflector is modeled above each storage cell, and a 13.23-inch water reflector is modeled below each storage cell. Reflective boundary conditions are applied to the top of the top water reflector and bottom of the bottom water reflectors respectively. Periodic boundary conditions are applied on the lateral surfaces of the 12x12 array. This simulates an infinitely repeating 12x12 array of storage cells. The pool water is simulated to be full density (1 g/cm$^{3)}$ at room temperature (20°C).

2.4.1.3 KENO Model for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

The Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration is modeled in KENO as a 12x12 array of "stainless steel canister" storage cells and "developed" storage cells. The pitch modeled between storage cells is 9.028 inches.

The KENO model for a 12x12 array of Region 2 storage cells was developed to accurately represent the construction of these storage cells. Note that the largest Region 2 module is a 12x11 array of storage cells. Therefore, the KENO results for a 12x12 array of storage cells (with periodic boundary conditions) will bound the expected results for the largest Region 2 module.

The fuel assembly inside each storage cell is either a fresh or depleted Westinghouse AP1000 17x17 assembly. Note the arrangement of fresh and depleted fuel assemblies within the 12x12 array is illustrated in Figure 2-3. As Figure 2-3 illustrates, there are two fresh fuel assemblies and six depleted fuel assemblies within each 2 x 4 configuration of storage cells. Therefore, it would be appropriate to call

this arrangement a "2-out-of-8 5.0 weight-percent Fresh" fuel assembly storage configuration. For simplicity, it is labeled a "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration.

The stainless steel canister is modeled with an inside dimension of 8.8 inches and is 0.075 inch thick. Metamic poison panels are modeled as attached to the outside of the stainless steel canister. The Metamic panels are modeled as 7.5 inches wide and 0.106 inch thick. Stainless steel sheathing panels hold the Metamic panels in place and are modeled as 0.035 inch thick. Note that there is a small gap between the Metamic poison panels and the stainless steel sheathing that is modeled as a void 0.012 inch thick. The Metamic poison material is modeled with the minimum B-10 areal loading. The stainless steel canister, fuel assembly, Metamic panels, and stainless steel sheathing are all modeled as 168 inches tall and are elevated above the concrete floor by 13.23 inches.

A 2-foot water reflector is modeled above each storage cell and a 13.23-inch water reflector is modeled below each storage cell. Reflective boundary conditions are applied to the top of the top water reflector and bottom of the bottom water reflectors respectively. Periodic boundary conditions are applied on the lateral surfaces of the 12x12 array. This simulates an infinitely repeating 12x12 array of storage cells. The pool water is simulated to be full density (1 g/cm^3) at room temperature (20°C).

2.4.1.4 KENO Model for Entire Spent Fuel Pool

The AP1000 spent fuel pool is modeled in KENO with an inside dimension of 444 inches in the north-to-south direction and 204 inches in the east-to-west direction. The AP1000 spent fuel pool is modeled containing three Region 1 modules, five Region 2 modules, and five individual Defective Fuel Assembly Storage Cells.

The Region 1 modules are modeled as 9x9 arrays of storage cells and are labeled "A1," "A2," and "A3" in Figure 2-2. Note that the Region 1 modules are modeled along the west wall of the AP1000 spent fuel pool. The Region 1 storage cells are modeled exactly as described in subsection 2.4.1.1.

There are four Region 2 modules that are modeled as 12x11 arrays of storage cells. The 12x11 Region 2 modules are labeled "B1," "B2," "B3," and "B4" in Figure 2-2. Theses modules are modeled along the east wall of the AP1000 spent fuel pool. The single Region 2 module labeled "C1" in Figure 2-2 is modeled as a 12x10 (-7) array of storage cells. (Note that the term "12x10 (-7)" means a 12x10 array that is missing seven storage cells. The seven storage cells removed from the 12x10 array provide space for the five Defective Fuel Assembly Storage Cells.) The Region 2 storage cells are modeled exactly as described in subsections 2.4.1.2 and 2.4.1.3.

The five individual Defective Fuel Assembly Storage Cells are modeled between module "C1" and the west wall of the AP1000 spent fuel pool and use the dimensions provided in Table 2-7.

2.4.2 Design Basis Fuel Assembly

Figure 2-1 shows the Westinghouse AP1000 17x17 fuel assembly, and Table 2-4 provides the fuel assembly parameters as modeled in this analysis. The Westinghouse AP1000 17x17 fuel assembly was modeled as the design basis fuel assembly to represent fresh and depleted fuel assemblies residing in all

of the fuel assembly storage configurations. Note that the fresh Westinghouse AP1000 17x17 fuel assembly does not contain any credit for a burnable absorber.

The fuel pellets in a fuel rod as modeled as a solid right cylinder with a UO_2 density equal to 10.686 g/cm^3 (97.5% of theoretical density). No credit is taken for the nominal 1.1 void fraction percentage that is associated with dishing or chamfering. In addition, no credit is taken for any natural or reduced enrichment pellets at the ends of the active fuel. This assumption results in conservative calculations of reactivity for all fuel assemblies stored in the racks. No credit is taken for any spacer grids or sleeves.

2.4.3 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology used in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the depleted fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the majority of spent fuel assemblies stored in the pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to represent the burnt fuel assembly with a representative axial burnup profile.

For any given spent fuel assembly, the fuel burnup is a continuous function of axial position. However, from a calculational point of view, this function can be discretized so that the axial "end-effect" is adequately captured. It is common practice to divide the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Moreover, the size of the top and bottom axial zones must be small (typically 6 to 8 inches) to capture the steep burnup gradient with axial position, while that of the central zone may be larger. In spent fuel pool calculations, a four-zone axial model is found to be adequate (Reference 20) to represent the spent fuel assembly. Such a four-zone model would have three zones with fine mesh spacing (three at the top of the fuel assembly), and the fourth zone is the remainder of the fuel assembly. Figure 2-4 shows the axial zone layout of the four-zone model used in the calculations.

The individual power fractions of each zone are modeled to give the same volume averaged burnup when compared to a uniform burnup model. This model is validated because the relative contribution of the bottom zones of the fuel assembly to the K_{eff} value is negligible. A benchmarking comparison of the assembly K_{eff} value in the spent fuel pool environment, of the four- and seven-zone models performed in Reference 20, provides adequate validation for the use of such a simplified model for 3D representation of depleted fuel assemblies.

Input to this analysis is based on the limiting axial burnup profile data provided in the Department of Energy (DOE) Topical Report, as documented in Reference 21. The burnup profile in the DOE Topical Report is based on a database of 3,169 axial-burnup profiles for pressurized water reactor (PWR) fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. The axial burnup profile in the DOE report is based on the most limiting axial burnup shape found in the database. The four-zone model is constructed based on this limiting axial burnup profile.

The DIT code was used to generate the isotopic concentrations for each segment of the axial burnup profile. Table 2-8 lists the fuel and moderator temperatures used in the spectral calculations for the fuel assembly average burnup model and each node of the four-zone axial burnup models. The fuel temperatures for each axial zone are calculated based on a representative fuel temperature correlation, while the moderator temperatures are based on a linear relationship with axial position. These node dependent moderator and fuel temperature data and power profile data were used in DIT to deplete the fuel to the desired burnup for each initial enrichment and each axial zone.

The most important reactivity aspect of the axial burnup profile is the difference between the average fuel burnup and the burnup in the top section of the fuel assembly. A secondary effect is the difference between the average fuel temperature and moderator density and the values at the top of the fuel assembly.

The values of assembly average burnups and feed enrichment for which isotopic number densities were generated with DIT are presented in Table 2-9.

A constant soluble boron concentration of 800 ppm was used in all the burnup steps. This value is representative of a cycle average soluble boron concentration in the core. For the purpose of extracting the number densities, the DIT computer code was executed in two modes. First, a normal depletion was continued in steps of 1000 MWD/MTU (with respect to the assembly average case) until the desired burnup was reached. Then, a restart was performed at cold, spent fuel pool conditions and the fuel assembly was allowed to decay for 100 hours, at which time the reactivity of the depleted fuel assembly is at its highest. The k_∞ and the isotopic number densities were then extracted for the KENO model development at these assembly conditions.

2.4.4 Tolerance/Uncertainty Calculations

Using the input described above, analytical models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than 0.995 with zero soluble boron present in the pool water. Applicable biases factored into this evaluation are: 1) the methodology bias deduced from the validation analyses of pertinent critical experiments, and 2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50° to 140°F. Note that cases for nominal conditions were run with a full moderator density (1 g/cc), which actually corresponds to 40°F, which is less than the normal operating range and more conservative.

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties. The following are included in the summation of variances:

a. 95/95 confidence level methods variance
b. 95/95 confidence level calculational uncertainty
c. Fuel rod manufacturing tolerances
d. Storage rack fabrication tolerances
e. Tolerance due to positioning the fuel assembly in the storage cell
f. Assembly depletion and burnup uncertainty

Items a and b are based on the calculational methods validation analyses described in subsection 2.2.1.2.

For item c, the following fuel rod manufacturing tolerances were evaluated:

- An increase in fuel enrichment from 5.0 to 5.05weight-percent U-235 (+0.05 weight-percent U-235)
- An increase in UO2 density form 97.5 to 98.5% theoretical density (+1%)
- An increase in pellet diameter from 0.3225 to 0.3230 inches (+0.0005 inches)
- A decrease in cladding outside diameter from 0.374 to 0.3725 inches (-0.0015 inches)

For item d, the following storage rack fabrication tolerances were evaluated for Region 1:

- The storage cell pitch was decreased from 10.9 to 10.86 inches.
- The canister thickness was increased from 0.075 to 0.082 inches.
- The canister inside diameter was decreased from 8.8 to 8.76 inches.
- The sheathing thickness was increased from 0.035 to 0.039 inches.
- The Metamic width was decreased from 7.5 to 7.4375 inches.

For item d, the following storage rack fabrication uncertainties were evaluated for Region 2:

- The storage cell pitch was decreased from 9.028 to 8.988 inches.
- The canister thickness was increased from 0.075 to 0.082 inches.
- The canister inside diameter was decreased from 8.8 to 8.76 inches.
- The sheathing thickness was increased from 0.035 to 0.039 inches.
- The Metamic width was decreased from 7.5 to 7.4375 inches.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e), all nominal calculations were carried out with fuel assemblies conservatively centered in the storage cells. Cases were run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells.

For item f, a 5% burnup measurement uncertainty was considered. The 5% burnup measurement uncertainty was applied to all the fuel assembly storage configurations that contain depleted fuel assemblies.

Tables 2-10, 2-11, and 2-12 provide a summary of the KENO results used in the calculation of biases and uncertainties for the fuel assembly storage configurations.

2.4.5 No Soluble Boron 95/95 K_{eff} Calculational Results

The following subsections present the KENO-calculated multiplication factors for the fuel assembly storage configurations analyzed for the AP1000 spent fuel storage racks.

The KENO calculations reported in this subsection were performed at 20°C, with maximum water density of 1.0 g/cm^3, to maximize the array reactivity, for both a uniform and axially distributed burnup profile. The relative axial burnup profile used for these latter calculations is discussed in subsection 2.4.3. The limiting K_{eff} data were then used to determine the burnup versus initial enrichment limits for a target K_{eff} value at zero soluble boron. The target value of K_{eff} was selected to be less than 0.995 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses.

2.4.5.1 Region 1 "All-Cell" Storage Configuration

As described in subsection 2.4.1.4, the Region 1 "All-Cell" fuel assembly storage configuration consists of an infinitely repeating array of Region 1 storage cells each loaded with a fresh fuel assembly. The fresh fuel assembly is modeled with the design parameters of the Westinghouse AP1000 17x17 design and enriched to 5.0 weight-percent U-235.

The nominal KENO calculated K_{eff} value for the Region 1 "All-Cell" fuel assembly storage configuration, with zero soluble boron, is 0.90977 ± 0.00067 as shown in Table 2-10. The total biases and uncertainties for this fuel assembly storage configuration is equal to 0.01530 delta K_{eff} units, also shown in Table 2-10. Therefore, the final 95/95 K_{eff} value for the Region 1 "All-Cell" fuel assembly storage configuration is 0.92507. This value is well below the design basis limit of 0.995 at zero soluble boron.

2.4.5.2 Region 2 "All-Cell" Storage Configuration

As described in subsection 2.4.1.2, the Region 2 "All-Cell" fuel assembly storage configuration consists of an infinitely repeating array of Region 2 storage cells each loaded with a depleted fuel assembly. The depleted fuel assembly is modeled with the design parameters of the Westinghouse AP1000 17x17 design.

The K_{eff} values were calculated for an infinite array of Region 2 "All-Cell" storage configurations over a range of initial enrichment values up to 5.0 weight-percent U-235 and average burnups up to 35,000 MWD/MTU. From Table 2-11, the sum of the biases and uncertainties for the Region 2 "All-Cell" fuel assembly storage configuration is 0.01919. Therefore, the target K_{eff} value for the Region 2 "All-Cell" fuel assembly storage configurations is 0.97581 (0.995-0.01919).

Table 2-13 lists the K_{eff} values for the Region 2 "All-Cell" fuel assembly storage configuration versus initial enrichment and average burnup with a uniform axial burnup profile. Table 2-14 lists the K_{eff} values for the Region 2 "All-Cell" fuel assembly storage configuration versus initial enrichment and average burnup with an axially distributed burnup profile. Table 2-15 lists the limiting K_{eff} values for the Region 2 "All-Cell" fuel assembly storage configuration. The first entry in Table 2-15 lists the initial enrichment for no burnup. Based on the target K_{eff} value, the interpolated enrichment for no burnup is 2.09 weight-percent U-235. The derived burnup limits, for enrichments greater than 2.09 weight-percent U-235, are based on the limiting K_{eff} values for 3.0, 4.0, and 5.0 weight-percent U-235. For each of these three enrichments, KENO calculations were performed at three assembly average burnup values. A second degree fit of the burnup versus limiting K_{eff} data was then used to determine the burnup required to meet the target K_{eff} value of 0.97581. The resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time are provided in Table 2-16. The limiting burnups as a function of initial enrichment were fitted to a second degree polynomial. These polynomials are given below Table 2-16 and will be used to determine the minimum burnup as a function of initial enrichment for the

Region 2 "All-Cell" fuel assembly storage configuration. The data in Table 2-16 are plotted in Figure 2-5.

2.4.5.3 Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

As described in subsection 2.4.1.3, the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration consists of a repeating 2x4 array, with a fresh 5.0 weight-percent fuel assembly occupying two storage locations and depleted fuel assemblies in the remaining locations.

K_{eff} values were calculated for an infinite array of the described fuel assembly storage configuration over a range of initial enrichment values up to 5.0 weight-percent U-235 and average burnups up to 55,000 MWD/MTU. From Table 2-12, the sum of the biases and uncertainties is 0.01588 delta K_{eff} units. Therefore, the target K_{eff} value for this fuel assembly storage configuration is 0.97912 (0.995-0.01588).

Table 2-17 lists the K_{eff} values for an infinite array of this fuel assembly storage configuration versus initial enrichment and average burnup with a uniform axial burnup profile. Table 2-18 lists the K_{eff} values for this fuel assembly storage configuration versus initial enrichment and average burnup with an axially distributed burnup profile. Table 2-19 lists the limiting K_{eff} values for the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration. The first entry in Table 2-19 lists the initial enrichment for no burnup. Based on the target K_{eff} value, the interpolated enrichment for no burnup is 1.44 weight-percent U-235. The derived burnup limits, for enrichments greater than 1.44 weight-percent U-235, are based on the K_{eff} values for 3.0, 4.0, and 5.0 weight-percent U-235. For each of these three enrichments, KENO calculations were performed at three assembly average burnup values. A second degree fit of the burnup versus limiting K_{eff} data was then used to determine the burnup required to meet the target K_{eff} value of 0.97912. The resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time are provided in Table 2-20. The limiting burnups as a function of initial enrichment were fitted to a second degree polynomial. These polynomials are given below Table 2-20 and will be used to determine the minimum burnup as a function of initial enrichment for the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration. The data in Table 2-20 are plotted in Figure 2-6.

2.4.5.4 Entire Spent Fuel Pool

A KENO model for the entire AP1000 spent fuel pool was constructed for this analysis. The spent fuel pool KENO model is described in subsection 2.4.1.4. This KENO model assumes that Region 2 is half filled with the "All-Cell" fuel assembly storage configuration and half filled with the "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration. Region 1 of this KENO model contains fresh 5.0 weight-percent U-235 fuel assemblies. Therefore, the interface between these different configurations was directly simulated.

The interface between the two Region 2 fuel assembly storage configurations was loaded with the requirements provided in Table 3-21. Table 3-21 indicates that any fuel assembly meeting the requirements of the "All-Cell" fuel assembly storage configuration may be loaded at the interface and that only depleted fuel assemblies from the "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration may be loaded at the interface. Figure 2-7 illustrates the allowable fuel assemblies at the interface with these two fuel assembly storage configurations.

The KENO calculated multiplication factors for infinite arrays of the Region 2 "All-Cell" and "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration are greater than the calculated multiplication factor for the entire spent fuel pool. This indicates that the overall geometry is less reactive than either of these two infinite arrays and that the interfaces meet all criticality requirements. Note also that the KENO calculated multiplication factor for the entire spent fuel pool is "driven" by the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration and that Region 1 is the least reactive configuration in the model.

2.4.5.5 Burnup Requirements for Intermediate Decay Time Points

For all the storage configurations in the AP1000 spent fuel pool crediting ^{241}Pu decay, burnup requirements for intermediate decay time points should be determined using at least a second order polynomial.

2.4.5.6 Empty Cells

An empty cell is permitted in any fuel assembly storage configuration of the spent fuel pool to replace an assembly since the water cell will not cause any increase in reactivity of the spent fuel pool. Non-fissile material and debris canisters may be stored in empty cells provided that the canister does not contain fissile materials.

2.4.5.7 Non-Fissile Equipment

Non-fissile equipment, such as ultrasonic cleaning equipment, is permitted on top of the fuel storage racks, because this equipment will not cause any increase in reactivity in the spent fuel pool.

2.4.6 Soluble Boron

The U.S. NRC Safety Evaluation Report (SER) for Westinghouse report WCAP-14416-P is given in Reference 6. Page 9 of the enclosure to Reference 6 defines the total soluble boron requirement as the sum of three quantities:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where,

SBC_{TOTAL} is the total soluble boron credit requirement (ppm),

$SBC_{95/95}$ is the soluble boron requirement for 95/95 K_{eff} less than or equal to 0.95 (ppm),

SBC_{RE} is the soluble boron required to account for burnup and reactivity uncertainties (ppm),

SBC_{PA} is the soluble boron required to offset accident conditions (ppm).

Each of these terms is discussed in the following subsections.

2.4.6.1 Soluble Boron Requirement to Maintain K_{eff} Less Than or Equal to 0.95

Table 2-22 contains the KENO-calculated K_{eff} values for the spent fuel pool from 0 to 600 ppm of soluble boron, in increments of 200 ppm. These KENO models assume that the pool is filled with the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration containing depleted fuel assemblies at 55,000 MWD/MTU with 5.0 weight-percent U-235 initial enrichment. (Note that the reactivity of the Region 1 "All-Cell" and Region 2 "All-Cell" fuel assembly storage configurations is more sensitive to soluble boron concentration increases.) The initial enrichment and assembly burnup chosen to represent this storage configuration was based on minimizing the soluble boron worth. The soluble boron worth decreases as burnup increases. The reactivity worth, ΔK_{eff}, of the soluble boron was determined by subtracting the K_{eff} value, for a given soluble boron concentration, from the K_{eff} value for zero soluble boron. The soluble boron concentration and reactivity worth data was then fitted to a third degree polynomial, which is shown on the bottom of Table 2-22. This polynomial was then used to determine the amount of soluble boron required to reduce K_{eff} by 0.05 ΔK_{eff} units, which is equal to 500.1 ppm.

2.4.6.2 Soluble Boron Requirement for Reactivity Uncertainties

The soluble boron credit, in units of ppm, required for reactivity uncertainties was determined by converting the uncertainty in fuel assembly depletion and the uncertainty in absolute fuel burnup values to a soluble boron concentration, in units of ppm, necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly depletion, is calculated by using a fuel assembly depletion uncertainty of 0.010 ΔK_{eff} units per 30,000 MWD/MTU of burnup (obtained from Reference 6) and multiplying by the maximum amount of burnup credited in a storage configuration. Table 2-23 contains the calculated fuel assembly depletion uncertainties for the two Region 2 fuel assembly storage configurations. The maximum fuel assembly depletion uncertainty is 0.01642 ΔK_{eff} units and was calculated for the Region 2 "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration.

The uncertainty in absolute fuel burnup values is conservatively calculated as 5% of the maximum fuel burnup credited in a storage configuration analysis. The maximum fuel burnup credited in the various storage configurations, the 5% uncertainty in these burnup values, and the corresponding reactivity values are given in Table 2-24. Note that the resulting ΔK_{eff} values were obtained by evaluating the derivative of K_{eff} with respect to assembly burnup at the maximum credited assembly burnup, and then multiplying by the 5% burnup uncertainty. The maximum burnup reactivity uncertainty is 0.00847 ΔK_{eff} units and was calculated for the Region 2 "All-Cell" fuel assembly storage configuration. Table 2-25 contains a summary of the reactivity uncertainties for both fuel assembly storage configurations.

The maximum depletion reactivity uncertainty was added to the maximum burnup reactivity uncertainty and the total is 0.02489 ΔK_{eff} units. Note that this value is slightly conservative compared to the maximum value calculated in Table 2-25. By applying the polynomial at the bottom of Table 2-22, the soluble boron concentration (ppm) necessary to compensate for this reactivity is found to be 246.7 ppm.

2.4.6.3 Soluble Boron Required to Mitigate Accidents

The soluble boron concentration, in units of ppm, to mitigate accidents is determined by first surveying all possible events that increase the K_{eff} value of the spent fuel pool. The accident event that produced the

largest increase in spent fuel pool Keff value is used to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered includes:

- Dropped fresh fuel assembly on top of the storage racks
- Intramodule water gap reduction due to seismic event
- Spent fuel pool temperature greater than 140°F
- Misloaded (or dropped) fuel assembly into an incorrect storage rack location, or outside the racks

Several fuel mishandling events were simulated using the KENO model to assess the possible increase in the K_{eff} value of the spent fuel pool. The fuel mishandling events all assumed that a fresh Westinghouse standard 17x17 fuel assembly enriched to 5.0 weight-percent U-235 (and no burnable poisons) was misloaded into a storage rack or between the racks and the walls of the spent fuel pool. These cases were simulated with the KENO model for the entire spent fuel pool.

It is possible to drop a fresh fuel assembly on top of the spent fuel pool storage racks. In this case the physical separation between the fuel assemblies in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the accident. In other words, dropping the fresh fuel assembly on top of the storage racks does not produce a positive reactivity increase. Note that the design of the spent fuel racks and fuel handling equipment precludes the insertion of a fuel assembly between the rack modules.

The water gaps between modules in the KENO models representing the entire spent fuel pool are less than 0.5 inch. The nominal water gaps between modules, as displayed in Figure 2-2, are at least 1.0 inch. Further reduction of the water gaps between modules would produce a reactivity increase, which is small compared to the reactivity increase due to fuel mishandling events. Therefore, the reactivity increase due to eliminating the water between modules as a result of a seismic event is bounded by the fuel mishandling reactivity increases.

For an accident condition in which the spent fuel pool temperature exceeds 140°F, the reactivity of the spent fuel pool will decrease. This conclusion is based upon the observation that over the nominal spent fuel pool temperature range, the reactivity decreased as the water temperature was increased.

The following fuel mishandling scenarios were considered:

- Fuel mishandling scenario 1: A fresh fuel assembly is misloaded outside of Region 1 on the south side of module A1.

- Fuel mishandling scenario 2: A fresh fuel assembly is misloaded outside of Region 2 on the west side of module C1.

- Fuel mishandling scenario 3: A depleted fuel assembly in the "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration is replaced with a fresh fuel assembly along the interface with modules B2 and A2. Note that this fuel assembly misloading scenario results in three fresh fuel assemblies in adjacent Region 2 storage cells at the interface with Region 1.

- Fuel mishandling scenario 4: A fuel assembly (with a control rod and attached to the fuel assembly handling tool) is dropped and impacts the base plate as discussed in subsection 2.8.5 of Reference 4. The analysis in subsection 2.8.5 of Reference 4 indicates that the base plate will be deformed as a result of this drop. For conservatism, the fuel assemblies in nine storage cell locations have been modeled as vertically dropped by 3.5 inches to bound the consequences of the base plate deformation.

The fuel mishandling event that produced the largest increase in the spent fuel pool K_{eff} value is the misloaded fresh fuel assembly outside Region 2 on the west side of module C1 (between module C1 and the west pool wall). This limiting fuel assembly mishandling event was run with soluble boron concentrations of 0, 200, and 400 ppm and the results are contained in Table 2-26. The required soluble boron concentration necessary to mitigate this fuel mishandling event (and all less severe accident events) was calculated to be 201.3 ppm.

2.4.6.4 Total Soluble Boron Requirement

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents.

The magnitude of each soluble boron requirement is as follows:

$SBC_{95/95}$ = 500.1 ppm

SBC_{RE} = 246.7 ppm

SBC_{PA} = 201.3 ppm

Therefore, without considering an additional accident, the soluble boron (with 19.9% B^{10} abundance) necessary to maintain K_{eff} less than or equal to 0.95 (including all biases and uncertainties) is:

$$SBC_{95/95} + SBC_{RE} = 500.1 \text{ ppm} + 246.7 \text{ ppm} = 747 \text{ ppm}.$$

The soluble boron concentration required for a B^{10} atom percent equal to 19.6 (to account for variations in B^{10} atom percent) is 758 ppm.

A total of 948 ppm of soluble boron (with 19.9% B^{10} abundance) is required to maintain K_{eff} less than or equal to 0.95 (including all biases and uncertainties) assuming the most limiting single accident. The soluble boron concentration required for a B^{10} atom percent equal to 19.6 is 963 ppm. The recommended minimum boron level is 963 ppm and is sufficient to accommodate all the design requirements. Note that this value (963 ppm) is well below the Technical Specification limit equal to 2300 ppm and that a boron dilution accident does not to be assumed concurrent with a fuel mishandling accident.

2.4.7 Boron Dilution Evaluation

2.4.7.1 Introduction

A boron dilution evaluation has been performed in order to support soluble boron reactivity credit in the spent fuel pool as part of the criticality analysis. The evaluation looks at possible sources of unborated water in the AP1000 plant systems, which can enter the spent fuel pool and dilute it from a concentration of 2,300 ppm boron to 758 ppm. The 758 ppm minimum boron concentration is discussed in subsection 2.4.6.4. Once the amount of unborated water has been determined from the individual AP1000 plant systems, the time to accumulate this volume of unborated water is determined. This is the duration that an operator has to detect the boron dilution event and mitigate the dilution before the spent fuel rack criticality analysis 0.95 K_{eff} design basis is exceeded.

The spent fuel pool provides safe storage for fresh and irradiated fuel assemblies. The spent fuel pool is filled with borated water. The water removes decay heat, provides shielding for personnel handing the fuel, and reduces the amount of radioactive gases released during a postulated accident. Pool water evaporation takes place on a continual basis; therefore, it is necessary to periodically deliver makeup water to the pool. The makeup sources can be unborated. Since the evaporation process does not carry off boron, an increase in the boron concentration in the pool can occur. However, continuing to deliver makeup flow of unborated water to the spent fuel pool above the initial water level would begin to dilute the spent fuel pool water. Continuing to input flow above a water elevation of 134.25 feet would trigger the spent fuel pool high level alarm. If dilution flow continues, the spent fuel pool, fuel transfer canal, and upender pit level will continue to rise until the flow reaches the operating deck level of 135.25 feet. The pool would then over flow onto the operating deck and into associated floor drains. Dilution of the spent fuel pool will continue as water overflows onto the operating deck and is drained to the waste holdup tank. These water levels are monitored by three safety-related (1E) level channels (LT 019A, ST 019B, and LT 019C). Each of these channels provides three level alarms. These are: a low-low level alarm at 132.25 feet, a low level alarm at 132.75 feet, and a high level alarm at 134.25 feet.

2.4.7.2 Calculation Approach/Methodology

The amount of boron in the spent fuel pool at its initial concentration is computed to determine the volume of water necessary to dilute the spent fuel pool boron concentration to 758 ppm. Since boron is not lost during evaporation, the total amount of boron in the spent fuel pool and connected pits/canals remains constant; therefore, the total volume of water to lower the boron concentration to 758 ppm can be calculated. This volume can be compared to calculated volumes of water that can be added to the spent fuel pool, refueling canal, and upender pit before the water level of each one reaches the operating deck. This determines how much water will spill out of the spent fuel pool onto the operating deck. The relevant spent fuel pool volumes are as follows:

- 214,073 gallons – Low-Low alarm water volume in the spent fuel pool, refueling canal, and refueling upending pit
- 229,164 gallons – Volume of spent fuel pool, refueling canal, and refueling upending pit at the high alarm level

- 235,453 gallons – Volume of spent fuel pool, refueling canal, and refueling upending pit at the operating deck

- 450,278 gallons – Total volume of water to dilute the spent fuel pool from 2300 ppm to 758 ppm

The time it takes to fill and dilute the spent fuel pool with unborated water is calculated based on the flow rates from the individual sources. There are only four sources capable of diluting the spent fuel pool below its minimum concentration. Each of these unborated water sources is characterized as follows:

- Chemical and Volume Control System (CVS)

  The CVS connects to the spent fuel pool via the discharge of the CVS makeup pumps to the spent fuel pool pump suction piping. The connection is normally isolated and is used to transfer water from the boric acid tank or demineralized water storage tank to the spent fuel pool. The line from the CVS contains an orifice to limit the flow to 100 gpm to compensate in the event of a spent fuel pool leak or evaporation of the spent fuel pool water.

- Passive Containment Cooling System (PCS) via the Ancillary Water Storage Tank

  The ancillary water storage tank can deliver unborated water to the spent fuel pool. The associated pumps have a design flow of 135 gpm. The tank contains 780,000 gallons of water. The ancillary water storage tank delivers water to the spent fuel pool via a connection to the passive core cooling water storage tank connection.

- PCS via the Passive Core Cooling Water Storage Tank (PCCWST)

  The PCCWST contains unborated water and is connected to the spent fuel pool via the Normal Residual Heat Removal System suction piping and the spent fuel pool spray nozzles. Each path is normally isolated by at least two normally closed manual valves, and each contains a flow instrument to monitor any flow in the line. The lines are normally isolated at the tank outlet and have a normally open drain valve to ensure that the lines are dry and do not freeze closed in cold weather. As a result, inadvertent dilution from this source is highly unlikely from an operation standpoint; however, the PCCWST contains 780,695 gallons of water and, thus, represents a dilution source capable of potentially flooding the pool above the operating deck. The maximum potential flow rate to the spent fuel pool from either of these connections is less than 270 gpm.

- Fire Protection System (FPS)

  The FPS also supplies unborated water to the spent fuel spray nozzles. The FPS line to the spent fuel pool is isolated by a normally-closed gate valve and it contains a flow instrument to monitor any flow in the line. Similar to the PCCWST, the maximum possible flow rate to the spent fuel pool is less than 270 gpm.

The spent fuel pool is capable of receiving water from other plant sources. However, these sources either deliver borated water or have an insufficient capacity to flood the spent fuel pool above the operating deck with unborated water. These sources are as follows:

- Demineralized Water Transfer and Storage System
- Cask Washdown Pit Hose Station
- Spent Fuel Pool Cooling System–borated water
- In-Containment Refueling Water Storage Tank (IRWST)–borated water
- CVS Boric Acid Storage Tank–borated water

2.4.7.3 Results of Boron Dilution Evaluation

Results of the evaluation are shown in Table 2-27 for the four systems evaluated for boron dilution in the spent fuel pool.

The dilution source capable of diluting the spent fuel pool the fastest is the Fire Protection System. However, it still takes over 26 hours to reach the minimum boron concentration of 758 ppm once the high level alarm is indicated. As mentioned earlier, there are three safety-related instruments with high level alarms to alert an operator that a dilution event is underway. If an inadvertent dilution were to be initiated, administrative procedures are in place to address a high level alarm in the spent fuel pool. These procedures instruct an operator to add borated water to the spent fuel pool to recover from the event. The procedures also require the operator to isolate the source of dilution water. Borated water is available from the boric acid tank in the CVS and the IRWST. The IRWST is even available during a loss of offsite power accident since it is gravity fed to the pool. As a last resort, boric acid powder can be added directly to the pool. In summary there is sufficient time for an operator to diagnose and mitigate the postulated boron dilution events for the AP1000.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required to dilute the spent fuel pool from 2300 ppm to 758 ppm. The 758 ppm endpoint was used to ensure that K_{eff} for the spent fuel racks would remain less than or equal to 0.95. As part of the criticality analysis for the spent fuel racks, a calculation has been performed on a 95/95 basis to show that the spent fuel rack K_{eff} remains less than 0.995 with non-borated water in the pool. Thus, even if the spent fuel pool were diluted to zero ppm, which would take significantly more water than evaluated above, the spent fuel would remain subcritical and the health and safety of the public would be ensured.

2.4.8 Maintenance of Subcriticality Margin for Metamic Material

This subsection addresses the capability of Metamic poison to maintain a 5-percent subcriticality margin. The topic is discussed in U.S. NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Storage Racks" (Reference 22). GL 96-04 specifically addresses degradation of Boraflex neutron absorbing material. The AP1000 racks do not use Boraflex. However, this subject is addressed in this subsection. Information is provided in the Holtec Generic Report on Metamic (Reference 23) describing the tests performed on Metamic to demonstrate its suitability for use as a neutron absorber in the spent fuel pool environment and its capability to maintain a 5-percent subcriticality margin. Also cited is the U. S. NRC Safety Evaluation on the application of Metamic at the Entergy Arkansas Nuclear One Units 1 and 2 spent fuel pools (Reference 24).

Metamic is a fully dense, discontinuously reinforced, metal matrix composite material. It consists of high-purity Type 6061 aluminum (Al 6061) alloy matrix reinforced with Type 1 American Society for Testing and Materials (ASTM) C 750 isotopically-graded boron carbide (B_4C). Table 2-28 shows the relationship among the weight percentages of B_4C and Al 6061, the volume percentage of B_4C, and the composite density of B_4C in grams per centimeters cubed (gm/cm^3).

An application for Metamic in plate form is its use in high-density fuel storage racks. In this application, the Metamic plates, or sheets, are placed between the cells of the storage racks and function as efficient thermal neutron absorbers. To ensure its satisfactory performance in the environments that exist in spent fuel pools, Metamic has been subjected to many tests that mimic the conditions to which the material will be exposed in actual service. The testing to which Metamic has been subjected includes the following:

- Short-term (48-hour) elevated temperature (900°F) testing
- Long-term elevated temperature (750°F) testing for times in excess of 1 year
- Accelerated corrosion testing (195°F) for times in excess of 1 year
- Accelerated radiation testing at exposures up to $1.5x10^{11}$ rads gamma
- Mechanical properties testing at temperatures up to 900°F
- Neutron transmission testing of coupons before and after corrosion testing

The U. S. NRC Office of Nuclear Reactor Regulation concluded in Section 3 Technical Evaluation of Reference 24 "Metamic, a metal matrix composite, is comprised primarily of B_4C and AL6061. B_4C is the main constituent in materials known to perform effectively as neutron absorbers. In addition, AL6061 is a marine-qualified material known for its resistance to corrosion. The staff finds that these characteristics support the assertion that Metamic is a desirable neutron absorber for wet storage applications. However, the distribution of B_4C in Metamic is also an important factor in determining the overall effectiveness of this material for spent fuel pool applications. Therefore, the staff reviewed the technical assessment of the B_4C distribution in Metamic. The technical assessment of the B_4C distribution of Metamic concluded that there were no significant local nonuniformities observed in the B_4C loading or in the areal density of the 15 weight-percent and 31 weight-percent B_4C plates tested. There were small global nonuniformities in B_4C loading and areal density, but were concluded to be of no consequence."

The U.S. NRC Office of Nuclear Regulatory Regulation found that the overall properties of the material (Metamic) are suitable for application in the spent fuel pool environment. The U.S. NRC Staff also limited the B_4C content to 31 weight-percent. The AP1000 Spent Fuel Racks will have a nominal B_4C content equal to 31 weight-percent.

Westinghouse concludes that based on Holtec extensive testing, the use of Metamic in spent fuel pool applications, and the U.S. NRC Staff review that Metamic has the capability to maintain a 5-percent subcriticality margin in the spent fuel pool environment.

2.4.9 Hypothetical Fuel Assembly Drop and Impact on Criticality Analysis

It is possible to drop a fuel assembly with a control rod and the handling tool into or on top of a storage cell in the AP1000 Spent Fuel Storage Racks as described in subsection 9.1.2.2.1 C of the DCD (Reference 1). In the event that the dropped load (fuel assembly with a control rod and the handling tool)

hits the top of a storage cell, the analyses in subsection 2.8.5 of Reference 4 indicate that both the Metamic and active fuel materials are not adversely impacted in any storage cell (Region 1, Region 2, or Defective Fuel Storage Cells). There is no degradation of the criticality safety margin as a result of dropping the described load on top of a storage cell and therefore no additional criticality analyses are required.

It is also possible to drop the described load straight through an empty cell and impact the storage rack base plate. The analysis in subsection 2.8.5 of Reference 4 indicates that the base plate will be deformed as a result of this drop. This accident scenario is bounded by the other accident scenarios discussed in subsection 2.4.6.3.

2.5 RESULTS

The following sections contain the criticality analysis results for the AP1000 spent fuel pool with assembly burnup, Pu-241 decay time, and soluble boron credit.

2.5.1 Allowable Storage Configurations

Figure 2-8 displays the allowable fuel assemblies in the Region 1 "All-Cell" fuel assembly storage configuration. The Region 1 "All-Cell" fuel assembly storage configuration will be employed to store fresh or depleted fuel assemblies with an initial enrichment less than or equal to 5.0 weight-percent U-235. Note that any location may be left empty. Figure 2-9 displays the allowable fuel assemblies in the Region 2 "All-Cell" fuel assembly storage configuration. The Region 2 "All-Cell" fuel assembly storage configuration will be employed to store depleted fuel assemblies which meet the requirements of Figure 2-5. Note that any location may be left empty. Figure 2-3 displays the allowable fuel assemblies in the "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration. The "1-out-of-4 5.0 weight-percent Fresh" fuel assembly storage configuration will be employed to store fresh fuel assemblies with enrichments less than or equal to 5.0 weight-percent U-235 and depleted fuel assemblies which meet the requirements of Figure 2-6.

2.5.2 Interface Requirements in Spent Fuel Pool Storage Racks

Fuel assemblies stored at the interface of storage configurations shall comply with the assembly loading instructions provided in Table 2-21. As indicated in Table 2-21, the interface loading instructions do not allow a fresh fuel assembly at the interface of a Region 2 "All-Cell" and a Region 2 "1-out-of-4 5.0 weight-percent Fresh" storage configuration. Note that it is acceptable to leave a storage cell empty.

2.5.3 Empty Cells

An empty cell is permitted in any fuel assembly storage configuration of the spent fuel pool to replace an assembly since the water cell will not cause any increase in reactivity of the spent fuel pool. Non-fissile material and debris canisters may be stored in empty cells provided that the canister does not contain fissile materials.

2.5.4 Non-Fissile Equipment

Non fissile equipment, such as ultrasonic cleaning equipment, is permitted on top of the AP1000 spent fuel storage racks, as this equipment will not cause any increase in reactivity of the spent fuel pool.

2.5.5 Total Soluble Boron Requirement

The soluble boron (with 19.9% B^{10} abundance) necessary to maintain K_{eff} less than or equal to 0.95 (including all biases and uncertainties) is 747 ppm. The soluble boron concentration required for a B^{10} atom percent equal to 19.6 (to account for variations in B-10 atom percent) is 758 ppm. This is the concentration of soluble boron which would need to be defended in a boron dilution accident analysis.

A total of 948 ppm of soluble boron (with 19.9% B^{10} abundance) is required to maintain K_{eff} less than or equal to 0.95 (including all biases and uncertainties) assuming the most limiting single accident. The soluble boron concentration required for a B^{10} atom percent equal to 19.6 is 963 ppm. The recommended minimum boron level is 963 ppm and is sufficient to accommodate all the design requirements.

Note that the above soluble boron concentrations are well below the Technical Specification limit equal to 2300 ppm for the AP1000 spent fuel pool.

Table 2-1 Calculational Results for Cores X Through XXI of B&W Close Proximity Experiments

Core	Run No.	KENO K_{eff} with 44-group library	KENO K_{eff} with 238-group library	Plate Type[1]	Spacing[2]
X	2348	0.99648 ± 0.00074	0.99295 ± 0.00078	none	3
XI	2355	1.00016 ± 0.00072	0.99770 ± 0.00078	SS-304	1
XI	2359	0.99937 ± 0.00072	0.99723 ± 0.00071	SS-304	1
XI	2360	1.00030 ± 0.00076	0.99692 ± 0.00065	SS-304	1
XI	2361	0.99931 ± 0.00070	0.99693 ± 0.00064	SS-304	1
XI	2362	1.00129 ± 0.00070	0.99885 ± 0.00066	SS-304	1
XI	2363	1.00043 ± 0.00059	0.99951 ± 0.00069	SS-304	1
XI	2364	1.00186 ± 0.00071	0.99875 ± 0.00079	SS-304	1
XII	2370	0.99847 ± 0.00076	0.99343 ± 0.00073	SS-304	2
XIII	2378	0.99653 ± 0.00082	0.99628 ± 0.00083	B/Al	1
XIIIA	2423	0.99537 ± 0.00077	0.99266 ± 0.00076	B/Al	1
XIV	2384	0.99559 ± 0.00073	0.99159 ± 0.00077	B/Al	1
XV	2388	0.99045 ± 0.00068	0.98882 ± 0.00068	B/Al	1
XVI	2396	0.99078 ± 0.00068	0.98784 ± 0.00081	B/Al	2
XVII	2402	0.99397 ± 0.00065	0.99155 ± 0.00076	B/Al	1
XVIII	2407	0.99527 ± 0.00068	0.99107 ± 0.00081	B/Al	2
XIX	2411	0.99614 ± 0.00068	0.99365 ± 0.00064	B/Al	1
XX	2414	0.99428 ± 0.00074	0.99160 ± 0.00081	B/Al	2
XXI	2420	0.99384 ± 0.00082	0.99044 ± 0.00068	B/Al	3

Notes:
1. Entry indicates metal separating unit assemblies.
2. Entry indicates spacing between unit assemblies in units of fuel rod pitch.

Table 2-2 Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping, and Storage Configurations

Experiment	KENO K_{eff} with 44-group library	KENO K_{eff} with 238-group library	Comments
043	0.99757 ± 0.00083	0.99321 ± 0.00084	Uniform rectangular array, no poison
044	0.99945 ± 0.00084	0.99526 ± 0.00080	Uniform rectangular array, no poison
045	0.99875 ± 0.00085	0.99433 ± 0.00080	Uniform rectangular array, no poison
046	0.99826 ± 0.00080	0.99441 ± 0.00076	Uniform rectangular array, no poison
061	0.99681 ± 0.00072	0.99183 ± 0.00089	2 x 2 array of rod clusters, no poison
062	0.99588 ± 0.00087	0.99346 ± 0.00093	2 x 2 array of rod clusters, no poison
064	0.99591 ± 0.00088	0.99368 ± 0.00077	2 x 2 array of rod clusters, 0.302-cm thick SS-304 cross
071	0.99760 ± 0.00081	0.99411 ± 0.00082	2 x 2 array of rod clusters, 0.485-cm thick SS-304 cross
079	0.99500 ± 0.00086	0.99293 ± 0.00086	2 x 2 array of rod clusters, cross of 0.3666 g boron/cm^2
087	0.99582 ± 0.00085	0.99174 ± 0.00082	2 x 2 array of rod clusters, cross of 0.1639 g boron/cm^2
093	0.99604 ± 0.00078	0.99260 ± 0.00094	2 x 2 array of rod clusters, cross of 0.1425 g boron/cm^2

Table 2-3 Standard Material Compositions Used in Criticality Analysis of AP1000 Spent Fuel Storage Racks

Material	Element	Weight Fraction
Zircaloy[1] Density = 6.578 g/cm^3 @ 293.15 K	Zr	0.9824
	Sn	0.0145
	Fe	0.0021
	Cr	0.0010
Water	SCALE Standard Composition Library Density = 1.0 g/cm^3 @ 293.15 K	
Stainless Steel	SCALE Standard Composition Library Density = 7.94 g/cm^3 @ 293.15 K	
Fresh UO_2	Fraction of Theoretical Density = 0.975 Enrichment = 5.0 weight-percent U-235 @ 293.15 K	
Regular Concrete	SCALE Standard Composition Library Density = 2.3 g/cm^3 @ 293.15 K	

Note:

1. The fuel rod, guide tube, and instrumentation tube claddings are modeled with Zircaloy in this analysis. This is conservative with respect to the Westinghouse ZIRLO product, which is a zirconium alloy containing additional elements including niobium. Niobium has a small absorption cross section, which causes more neutron capture in the cladding regions resulting in a lower reactivity. Therefore, this analysis is conservative with respect to fuel assemblies containing ZIRLO cladding in the fuel rods, guide tubes, and instrumentation tube.

Table 2-4 Westinghouse AP1000 17x17 Fuel Assembly Parameters Used for AP1000 Spent Fuel Pool Criticality Analysis (Tolerances are not shown because they are Westinghouse Proprietary Information.)

Parameter	Value
Rods per Assembly	264
Guide Tubes per Assembly	24
Instrument Tubes per Assembly	1
Rod Pitch (in)	0.4960
Pellet OD (in)	0.3225
UO_2 Density (% of TD)	97.5
Enrichment, weight-percent U-235	5.0
Active Fuel Length (in)	168
Burnable Absorbers	None
Clad OD (in)	0.3740
Clad Thickness (in)	0.0225
Guide Tube OD (in)	0.482
Guide Tube Thickness (in)	0.02
Instrument Tube OD (in)	0.482
Instrument Tube Thickness (in)	0.02
Bottom Elevation of Active Fuel Relative to Top of Base Plate	+8.23 inches

Table 2-5 Design Input for Region 1 Storage Racks (All dimensions are in inches; tolerances are not shown because they are Westinghouse Proprietary Information.)

Parameter	Value
Storage Cell Center-to-Center Pitch	10.9
Storage Cell Inner Dimension (Width)	8.8
Inter-Cell Flux Trap Gap	1.644
Storage Cell Length Region 1 Spent Fuel Storage Rack New Fuel Storage Rack	 199.5 193.5
Storage Cell Wall Thickness	0.075
Neutron Absorber Material	Metamic
Neutron Absorber Length	172
Neutron Absorber Width	7.5
Neutron Absorber Thickness	0.106
Distance from Top of Rack Baseplate to Bottom of Neutron Absorber	6.23
Neutron Absorber B_4C Loading	31 weight-percent
Neutron Absorber Sheathing Thickness Internal Walls Periphery Walls	 0.035 0.075
Baseplate Thickness	0.75
Baseplate Flow Hole Diameter	6
Rack Pedestal Type (fixed or adjustable)	Adjustable
Rack Pedestal Height (female + male)	2.75
Rack Female Pedestal Dimensions	20 x 20 x 2.25
Rack Male Pedestal Diameter	4.5
Rack Bearing Pad Thickness	1.5

Table 2-6 Design Input for Region 2 Storage Racks (All dimensions are in inches; tolerances are not shown because they are Westinghouse Proprietary Information.)

Parameter	Value
Storage Cell Center-to-Center Pitch	9.028
Storage Cell Inner Dimension (Width)	8.8
Inter-Cell Flux Trap Gap	N/A
Storage Cell Length	199.5
Storage Cell Wall Thickness	0.075
Neutron Absorber Material	Metamic
Neutron Absorber Length	172
Neutron Absorber Width	7.5
Neutron Absorber Thickness	0.106
Distance from Top of Rack Baseplate to Bottom of Neutron Absorber	6.23
Neutron Absorber B_4C Loading	31 weight-percent
Neutron Absorber Sheathing Thickness Internal Walls Periphery Walls	 0.035 0.075
Baseplate Thickness	0.75
Baseplate Flow Hole Diameter	6
Rack Pedestal Type (fixed or adjustable)	Adjustable
Rack Pedestal Height (female + male)	2.75
Rack Female Pedestal Dimensions	18 x 18 x 2.25
Rack Male Pedestal Diameter	4.5
Rack Bearing Pad Thickness	1.5

Table 2-7 Design Input for Damaged Fuel Assembly Storage Cells (All dimensions are in inches; tolerances are not shown because they are Westinghouse Proprietary Information.)

Parameter	Value
Storage Cell Center-to-Center Pitch	12.35
Storage Cell Inner Dimension (Width)	10.25
Inter-Cell Flux Trap Gap	0.91
Storage Cell Length	199.5
Storage Cell Wall Thickness	0.075
Neutron Absorber Material	Metamic
Neutron Absorber Length	172
Neutron Absorber Width	7.5
Neutron Absorber Thickness	0.106
Distance from Top of Rack Baseplate to Bottom of Neutron Absorber	6.23
Neutron Absorber B_4C Loading	31 weight-percent
Neutron Absorber Sheathing Thickness Internal Walls Periphery Walls	 0.035 0.075

Table 2-8 Relative Power and Fuel/Moderator Temperatures for the Average and Four-Zone Models

Zone Number	Height	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
Average	168	1.0	944.12	579.95
Four-Zone Model				
1	150.0	1.033	823.72	574.14
2	6.0	0.971	1036.77	605.12
3	6.0	0.738	1226.51	608.99
4	6.0	0.462	1151.87	612.86

Table 2-9 Burnup and Initial Enrichment Combinations Used to Determine Isotopic Number Densities

3.0 Weight-Percent U-235 (MWD/MTU)	4.0 Weight-Percent U-235 (MWD/MTU)	5.0 Weight-Percent U-235 (MWD/MTU)
0	0	0
5,000	5,000	5,000
15,000	15,000	15,000
25,000	25,000	25,000
35,000	35,000	35,000
45,000	45,000	45,000
55,000	55,000	55,000
65,000	65,000	65,000

Table 2-10 Determination of Final K_{eff} Value for the Region 1 "All-Cell" Storage Configuration

Case Description	K_{eff}	ΔK_{eff}
Nominal Case	0.90977 ± 0.00067	N/A
Increase in U-235 Enrichment (+0.05 weight-percent)	0.91091 ± 0.00063	0.00244
Increase in UO_2 Density (+1%)	0.91095 ± 0.00068	0.00253
Increase in Pellet OD (+0.0005 in)	0.91026 ± 0.00068	0.00184
Decrease in Clad OD (-0.0015 in)	0.91164 ± 0.00070	0.00324
Decrease in Cell Pitch (-0.04 in)	0.91285 ± 0.00063	0.00438
Increase in Cell Wall Thickness (+0.007 in)	0.91085 ± 0.00065	0.00240
Increase in Cell ID (-0.04 in)	0.91541 ± 0.00061	0.00692
Increase in Sheathing Thickness (+0.004 in)	0.90932 ± 0.00078	0.00100
Decrease in Metamic Width (-0.0625 in)	0.91080 ± 0.00067	0.00237
Asymmetric Assembly Positioning	0.90869 ± 0.00067	0.00026
Methodology Uncertainty(1)		0.00657
Statistical Sum of Uncertainties		0.01220
Methodology Bias(2)		0.00310
Temperature Bias(3)		0.0 (Negative)
Sum of Uncertainties and Biases		0.01530
Final K_{eff} Value		0.92507

Notes:

1. Methodology uncertainty is the statistical sum of the methodology calculational variance and the KENO uncertainty, using a maximum KENO uncertainty of all the bias and uncertainty calculations for this configuration equal to 0.00080.
2. Methodology bias or the mean calculational methods bias is evaluated to be 0.00310 (see subsection 2.2.1 for background).
3. Pool temperature bias is calculated with the 238-group library at a burnup equal to 25,000 MWD/MTU and an initial enrichment equal to 4.0 weight-percent U-235.

Table 2-11 K_{eff} Values for the Tolerance/Uncertainty Cases for the Region 2 "All-Cell" Storage Configuration

Case Description	K_{eff}	ΔK_{eff}
Nominal Case(1)	0.97064 ± 0.00051	N/A
Increase in U-235 Enrichment (+0.05 weight-percent)	0.97840 ± 0.00053	0.00880
Increase in UO_2 Density (+1%)	0.97216 ± 0.00048	0.00251
Increase in Pellet OD (+0.0005 in.)	0.97239 ± 0.00049	0.00275
Decrease in Clad OD (-0.0015 in.)	0.97266 ± 0.00048	0.00301
Decrease in Cell Pitch (-0.04 in.)	0.97184 ± 0.00049	0.00220
Increase in Cell Wall Thickness (+0.007 in.)	0.97135 ± 0.00056	0.00178
Decrease in Cell ID (-0.04 in.)	0.97069 ± 0.00052	0.00108
Increase in Sheathing Thickness (+0.004 in.)	0.97219 ± 0.00054	0.00260
Decrease in Metamic Width (-0.0625 in.)	0.97289 ± 0.00053	0.00329
Asymmetric Assembly Positioning	0.96833 ± 0.00056	-0.00124 (set to zero)
Burnup Uncertainty(2)		0.00941
Methodology Uncertainty(3)		0.00657
Statistical Sum of Uncertainties		0.01609
Methodology Bias(4)		0.00310
Temperature Bias(5)		0.0 (Negative)
Sum of Uncertainties and Biases		**0.01919**

Notes:

1. The nominal case for the Region 2 "All-Cell" storage configuration contains standard fuel assembly at the fresh enrichment of 2.07 weight-percent U-235.
2. Burnup uncertainty is found by evaluating the differential burnup worth at the conservative maximum burnup credit value of 35,000 MWD/MTU.
3. Methodology uncertainty is the statistical sum of the methodology calculational variance and the KENO uncertainty, using a maximum KENO uncertainty of all the bias and uncertainty calculations for this configuration equal to 0.00080.
4. Methodology bias or the mean calculational methods bias is evaluated to be 0.00310 (see subsection 2.2.1 for background).
5. Pool temperature bias is calculated with the 238-group library at a burnup equal to 25,000 MWD/MTU and an initial enrichment equal to 4.0 weight-percent U-235.

Table 2-12 K_{eff} Values for the Tolerance/Uncertainty Cases for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

Case Description	K_{eff}	ΔK_{eff}
Nominal Case[1]	0.97041 ± 0.00060	N/A
Increase in U-235 Enrichment (+0.05 weight-percent)	0.97659 ± 0.00060	0.00738
Increase in UO_2 Density (+1%)	0.97158 ± 0.00054	0.00231
Increase in Pellet OD (+0.0005 in.)	0.97049 ± 0.00064	0.00132
Decrease in Clad OD (-0.0015 in.)	0.97159 ± 0.00061	0.00239
Decrease in Cell Pitch (-0.04 in.)	0.97052 ± 0.00062	0.00133
Increase in Cell Wall Thickness (+0.007 in.)	0.97009 ± 0.00054	0.00082
Decrease in Cell ID (-0.04 in.)	0.96886 ± 0.00065	-0.00030 (set to zero)
Increase in Sheathing Thickness (+0.004 in.)	0.96966 ± 0.00058	0.00043
Decrease in Metamic Width (-0.0625 in.)	0.97188 ± 0.00055	0.00262
Asymmetric Assembly Positioning	0.96826 ± 0.00057	-0.00098 (set to zero)
Burnup Uncertainty[2]		0.00652
Methodology Uncertainty[3]		0.00657
Statistical Sum of Uncertainties		0.01278
Methodology Bias[4]		0.00310
Pool Temperature Bias[5]		0.0
Sum of Uncertainties and Biases		**0.01588**

Notes:

1. The nominal case for the "1-out-of-4 5.0 weight-percent at Fresh" storage configuration contains standard fuel assembly at the fresh enrichment of 1.375 weight-percent U-235.
2. Burnup uncertainty is found by evaluating the differential burnup worth at the conservative maximum burnup credit value of 55,000 MWD/MTU.
3. Methodology uncertainty is the statistical sum of the methodology calculational variance and the KENO uncertainty, using a maximum KENO uncertainty of all the bias and uncertainty calculations for this configuration equal to 0.00080.
4. Methodology bias or the mean calculational methods bias is evaluated to be 0.00310 (see subsection 2.2.1 for background).
5. Pool temperature bias is calculated with the 238-group library at a burnup equal to 35,000 MWD/MTU and an initial enrichment equal to 4.0 weight-percent U-235.

Table 2-13 K_{eff} Values versus Initial Enrichment and Assembly Burnup (Uniform) for Region 2 "All-Cell" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
2.147	0					
3.000	5,000	1.02070 ± 0.00054	1.02059 ± 0.00052	1.01966 ± 0.00049	1.02008 ± 0.00048	1.01848 ± 0.00047
3.000	15,000	0.93835 ± 0.00048	0.93076 ± 0.00046	0.92612 ± 0.00045	0.92110 ± 0.00048	0.91800 ± 0.00046
3.000	25,000	0.86395 ± 0.00052	0.84912 ± 0.00041	0.83723 ± 0.00040	0.82857 ± 0.00042	0.82123 ± 0.00041
4.000	15,000	1.01452 ± 0.00052	1.00981 ± 0.00054	1.00573 ± 0.00049	1.00281 ± 0.00054	1.00130 ± 0.00052
4.000	25,000	0.94375 ± 0.00051	0.93335 ± 0.00043	0.92451 ± 0.00043	0.91796 ± 0.00044	0.91218 ± 0.00051
4.000	35,000	0.87613 ± 0.00045	0.85975 ± 0.00040	0.84581 ± 0.00042	0.83623 ± 0.00038	0.82769 ± 0.00041
5.000	15,000	1.07124 ± 0.00051	1.06740 ± 0.00050	1.06590 ± 0.00050	1.06323 ± 0.00048	1.06183 ± 0.00051
5.000	25,000	1.00565 ± 0.00050	0.99747 ± 0.00049	0.99014 ± 0.00048	0.98549 ± 0.00047	0.98101 ± 0.00049
5.000	35,000	0.94289 ± 0.00046	0.92954 ± 0.00044	0.92029 ± 0.00044	0.91127 ± 0.00045	0.90423 ± 0.00043

Table 2-14 K_{eff} Values versus Initial Enrichment and Assembly Burnup (Distributed) for Region 2 "All-Cell" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
2.147	0					
3.000	5,000	1.01882 ± 0.00050	1.01800 ± 0.00051	1.01863 ± 0.00060	1.01713 ± 0.00054	1.01636 ± 0.00049
3.000	15,000	0.93337 ± 0.00045	0.92638 ± 0.00051	0.92130 ± 0.00046	0.91858 ± 0.00047	0.91480 ± 0.00048
3.000	25,000	0.86720 ± 0.00059	0.85695 ± 0.00051	0.84915 ± 0.00061	0.84247 ± 0.00059	0.83819 ± 0.00056
4.000	15,000	1.01410 ± 0.00049	1.00930 ± 0.00045	1.00563 ± 0.00051	1.00332 ± 0.00051	1.00020 ± 0.00053
4.000	25,000	0.94267 ± 0.00053	0.93450 ± 0.00050	0.92805 ± 0.00057	0.92226 ± 0.00060	0.91899 ± 0.00055
4.000	35,000	0.88925 ± 0.00065	0.87850 ± 0.00058	0.86813 ± 0.00058	0.86336 ± 0.00075	0.85731 ± 0.00067
5.000	15,000	1.06775 ± 0.00052	1.06415 ± 0.00050	1.06116 ± 0.00050	1.05929 ± 0.00054	1.05777 ± 0.00049
5.000	25,000	1.00327 ± 0.00046	0.99625 ± 0.00053	0.99112 ± 0.00047	0.98653 ± 0.00050	0.98308 ± 0.00046
5.000	45,000	0.94929 ± 0.00048	0.94038 ± 0.00051	0.93203 ± 0.00054	0.92708 ± 0.00059	0.92267 ± 0.00060

Table 2-15 Limiting K_{eff} Values versus Initial Enrichment and Assembly Burnup for Region 2 "All-Cell" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
2.147	0					
3.000	5,000	1.02070 ± 0.00054	1.02059 ± 0.00052	1.01966 ± 0.00049	1.02008 ± 0.00048	1.01848 ± 0.00047
3.000	15,000	0.93835 ± 0.00048	0.93076 ± 0.00046	0.92612 ± 0.00045	0.92110 ± 0.00048	0.91800 ± 0.00046
3.000	25,000	0.86720 ± 0.00059	0.85695 ± 0.00051	0.84915 ± 0.00061	0.84247 ± 0.00059	0.83819 ± 0.00056
4.000	15,000	1.01452 ± 0.00052	1.00981 ± 0.00054	1.00573 ± 0.00049	1.00332 ± 0.00051	1.00130 ± 0.00052
4.000	25,000	0.94375 ± 0.00051	0.93450 ± 0.00050	0.92805 ± 0.00057	0.92226 ± 0.00060	0.91899 ± 0.00055
4.000	35,000	0.88925 ± 0.00065	0.87850 ± 0.00058	0.86813 ± 0.00058	0.86336 ± 0.00075	0.85731 ± 0.00067
5.000	15,000	1.07124 ± 0.00051	1.06740 ± 0.00050	1.06590 ± 0.00050	1.06323 ± 0.00048	1.06183 ± 0.00051
5.000	25,000	1.00565 ± 0.00050	0.99747 ± 0.00049	0.99112 ± 0.00047	0.98653 ± 0.00050	0.98308 ± 0.00046
5.000	35,000	0.94929 ± 0.00048	0.94038 ± 0.00051	0.93203 ± 0.00054	0.92708 ± 0.00059	0.92267 ± 0.00060

Table 2-16 Fuel Assembly Burnup versus Initial Enrichment for Region 2 "All-Cell" Storage Configuration

Initial Enrichment (Weight-Percent U-235)	Limiting Burnup (MWD/MTU)				
	0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
2.09	0	0	0	0	0
3.000	10,241	9,688	9,394	9,117	8,893
4.000	20,051	19,024	18,455	17,905	17,688
5.000	30,133	28,600	27,414	26,658	26,096

Note:

The assembly burnups as a function of initial enrichment (e) for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $-2.8457E+02\ e^2 + 1.2312E+04\ e - 2.4395E+04$

Assembly Burnup (5 yr decay) = $-2.5995E+02\ e^2 + 1.1617E+04\ e - 2.3059E+04$

Assembly Burnup (10 yr decay) = $-3.3850E+02\ e^2 + 1.1779E+04\ e - 2.3076E+04$

Assembly Burnup (15 yr decay) = $-3.1297E+02\ e^2 + 1.1337E+04\ e - 2.2262E+04$

Assembly Burnup (20 yr decay) = $-3.4366E+02\ e^2 + 1.1383E+04\ e - 2.2255E+04$

Table 2-17 K_{eff} Values versus Initial Enrichment and Assembly Burnup (Uniform) for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
1.361	0					
3.000	15,000	1.02307 ± 0.00052	1.01879 ± 0.00050	1.01523 ± 0.00049	1.01249 ± 0.00047	1.01073 ± 0.00049
3.000	25,000	0.98174 ± 0.00047	0.97446 ± 0.00051	0.96991 ± 0.00058	0.96426 ± 0.00056	0.96156 ± 0.00052
3.000	35,000	0.95019 ± 0.00047	0.94173 ± 0.00055	0.93450 ± 0.00054	0.92984 ± 0.00051	0.92631 ± 0.00054
4.000	25,000	1.02552 ± 0.00052	1.01967 ± 0.00049	1.01470 ± 0.00048	1.01166 ± 0.00051	1.00905 ± 0.00051
4.000	35,000	0.98900 ± 0.00045	0.97987 ± 0.00048	0.97346 ± 0.00054	0.96731 ± 0.00050	0.96514 ± 0.00047
4.000	45,000	0.95814 ± 0.00055	0.94853 ± 0.00055	0.94203 ± 0.00055	0.93677 ± 0.00052	0.93227 ± 0.00049
5.000	35,000	1.02531 ± 0.00046	1.01842 ± 0.00051	1.01257 ± 0.00045	1.00830 ± 0.00053	1.00407 ± 0.00046
5.000	45,000	0.99129 ± 0.00048	0.98297 ± 0.00051	0.97627 ± 0.00052	0.96918 ± 0.00049	0.96503 ± 0.00051
5.000	55,000	0.96434 ± 0.00051	0.95402 ± 0.00049	0.94485 ± 0.00051	0.93872 ± 0.00043	0.93557 ± 0.00053

Table 2-18 K_{eff} Values versus Initial Enrichment and Assembly Burnup (Distributed) for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
1.361	0					
3.000	15,000	1.01850 ± 0.00048	1.01507 ± 0.00049	1.01162 ± 0.00047	1.01018 ± 0.00048	1.00801 ± 0.00050
3.000	25,000	0.97868 ± 0.00050	0.97020 ± 0.00051	0.96602 ± 0.00055	0.96075 ± 0.00051	0.95850 ± 0.00053
3.000	35,000	0.94770 ± 0.00053	0.93904 ± 0.00050	0.93200 ± 0.00053	0.92785 ± 0.00057	0.92380 ± 0.00052
4.000	25,000	1.02224 ± 0.00050	1.01559 ± 0.00052	1.01064 ± 0.00048	1.00683 ± 0.00050	1.00406 ± 0.00049
4.000	35,000	0.98494 ± 0.00045	0.97674 ± 0.00053	0.97088 ± 0.00054	0.96646 ± 0.00053	0.96310 ± 0.00049
4.000	45,000	0.95653 ± 0.00050	0.94727 ± 0.00055	0.94151 ± 0.00053	0.93412 ± 0.00056	0.93287 ± 0.00051
5.000	35,000	1.02188 ± 0.00047	1.01495 ± 0.00043	1.00833 ± 0.00044	1.00468 ± 0.00045	1.00025 ± 0.00044
5.000	45,000	0.98892 ± 0.00055	0.98191 ± 0.00052	0.97428 ± 0.00052	0.97012 ± 0.00050	0.96519 ± 0.00056
5.000	55,000	0.96285 ± 0.00051	0.95399 ± 0.00058	0.94725 ± 0.00053	0.94155 ± 0.00056	0.93756 ± 0.00059

Table 2-19 Limiting K_{eff} Values versus Initial Enrichment and Assembly Burnup for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (Weight-Percent U-235)	Burnup (MWD/MTU)	K_{eff} Value				
		0 Year Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
1.361	0					
3.000	15,000	1.02307 ± 0.00052	1.01879 ± 0.00050	1.01523 ± 0.00049	1.01249 ± 0.00047	1.01073 ± 0.00049
3.000	25,000	0.98174 ± 0.00047	0.97446 ± 0.00051	0.96991 ± 0.00058	0.96426 ± 0.00056	0.96156 ± 0.00052
3.000	35,000	0.95019 ± 0.00047	0.94173 ± 0.00055	0.93450 ± 0.00054	0.92984 ± 0.00051	0.92631 ± 0.00054
4.000	25,000	1.02552 ± 0.00052	1.01967 ± 0.00049	1.01470 ± 0.00048	1.01166 ± 0.00051	1.00905 ± 0.00051
4.000	35,000	0.98900 ± 0.00045	0.97987 ± 0.00048	0.97346 ± 0.00054	0.96731 ± 0.00050	0.96514 ± 0.00047
4.000	45,000	0.95814 ± 0.00055	0.94853 ± 0.00055	0.94203 ± 0.00055	0.93677 ± 0.00052	0.93287 ± 0.00051
5.000	35,000	1.02531 ± 0.00046	1.01842 ± 0.00051	1.01257 ± 0.00045	1.00830 ± 0.00053	1.00407 ± 0.00046
5.000	45,000	0.99129 ± 0.00048	0.98297 ± 0.00051	0.97627 ± 0.00052	0.97012 ± 0.00050	0.96519 ± 0.00056
5.000	55,000	0.96434 ± 0.00051	0.95402 ± 0.00049	0.94725 ± 0.00053	0.94155 ± 0.00056	0.93756 ± 0.00059

Table 2-20 Fuel Assembly Burnup versus Initial Enrichment for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

Initial Enrichment (Weight-Percent U-235)	Limiting Burnup (MWD/MTU)				
	0 Years Decay	5 Years Decay	10 Years Decay	15 Years Decay	20 Years Decay
1.440	0	0	0	0	0
3.000	25,740	23,747	22,704	21,410	20,893
4.000	38,034	35,206	33,408	31,804	31,355
5.000	49,274	46,222	44,101	42,285	40,859

Note:

The assembly burnups as a function of initial enrichment (e) for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $-1.2226E+03\ e^2 + 2.1648E+04\ e - 2.8559E+04$

Assembly Burnup (5 yr decay) = $-9.9944E+02\ e^2 + 1.9346E+04\ e - 2.5697E+04$

Assembly Burnup (10 yr decay) = $-9.3905E+02\ e^2 + 1.8346E+04\ e - 2.4366E+04$

Assembly Burnup (15 yr decay) = $-7.9427E+02\ e^2 + 1.6913E+04\ e - 2.2613E+04$

Assembly Burnup (20 yr decay) = $-8.9385E+02\ e^2 + 1.7194E+04\ e - 2.2859E+04$

Table 2-21 Assembly Loading Requirements at Interface between Different Storage Configurations	
Configuration	**Assembly that Must be Loaded at Interface with Another Configuration[1]**
"All-Cell"	Any
"1-out-of-4 5.0 Weight-Percent Fresh"	Only Depleted Fuel Assemblies

Note:
1. An empty storage location is always permitted.

Instructions:
1. Identify which storage configurations will be interfaced.
2. Look up the assembly loading requirements for both storage configurations.

Table 2-22 K_{eff} Values as a Function of Soluble Boron Concentration for Spent Fuel Pool Loaded with "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration				
	K_{eff}			
Configuration	**0 ppm**	**200 ppm**	**400 ppm**	**600 ppm**
Depleted Fuel (5.0 Weight-Percent, 55,000 MWD/MTU)	0.97102 ± 0.00032	0.93533 ± 0.00031	0.90529 ± 0.00032	0.87873 ± 0.00030

Note:
The following polynomial describes an amount of boron as a function of ΔK_{eff} for the entire spent fuel pool:

$$ppm = -6747.975\Delta K_{eff}^3 + 16719.356\Delta K_{eff}^2 + 5015.692\Delta K_{eff}$$

Table 2-23 Summary of Fuel Assembly Depletion Uncertainties for Storage Configurations		
Configuration	**Maximum Burnup (MWD/MTU)**	**Fuel Assembly Depletion Uncertainty, ΔK_{eff}**
"All-Cell"	30,132.7	0.01004
"1-out-of-4 5.0 Weight-Percent Fresh"	49,273.6	0.01642

Table 2-24 Summary of Absolute Burnup Uncertainties for the Storage Configurations			
Configuration	Maximum Burnup (MWD/MTU)	5% Burnup Uncertainty	ΔK_{eff}
"All-Cell"	30,132.7	1,506.6	0.00847
"1-out-of-4 5.0 Weight-Percent Fresh"	49,273.6	2463.7	0.00677

Table 2-25 Summation of Reactivity Uncertainties for the Storage Configurations			
Configuration	Fuel Assembly Depletion Uncertainty, ΔK_{eff}	Absolute Burnup Uncertainty, ΔK_{eff}	Total ΔK_{eff}
"All-Cell"	0.01004	0.00847	0.01851
"1-out-of-4 5.0 Weight-Percent Fresh"	0.01642	0.00677	0.02319

Table 2-26 K_{eff} Values for Limiting Fuel Mishandling Accident Scenario in Spent Fuel Pool (Nominal K_{eff} equal to 0.95929 ± 0.00059)		
Description of Accident	K_{eff}	ΔK_{eff}
Accident Scenario 2, 0 ppm	0.99171	+0.03242
Accident Scenario 2, 200 ppm	0.95948	+0.00019
Accident Scenario 2, 400 ppm	0.93276	-0.02653

Table 2-27 Time to Dilute Spent Fuel Pool from 2300 ppm to 758 ppm Boron			
Source of Unborated Water	Flow Rate (gpm)	Time to Reach Minimum Boron Concentration (hours)	Time from High Alarm to Minimum Boron Concentration (hours)
Chemical and Volume Control System (CVS) Makeup Pumps	100	73.4	71.3
Ancillary Water Storage Tank	135	54.4	54.0
PCCWST	270	27.2	26.4
Fire Protection System (FPS)	270	27.2	26.4

Table 2-28 Properties of Metamic at Different Weight-Percent B_4C

Weight-Percent B_4C	Weight-Percent Al 6061	Volume-Percent B_4C	Composite Density, gm/cm^3
14.14	85.86	15	2.673
18.92	81.08	20	2.664
28.57	71.43	30	2.646
38.36	61.64	40	2.628

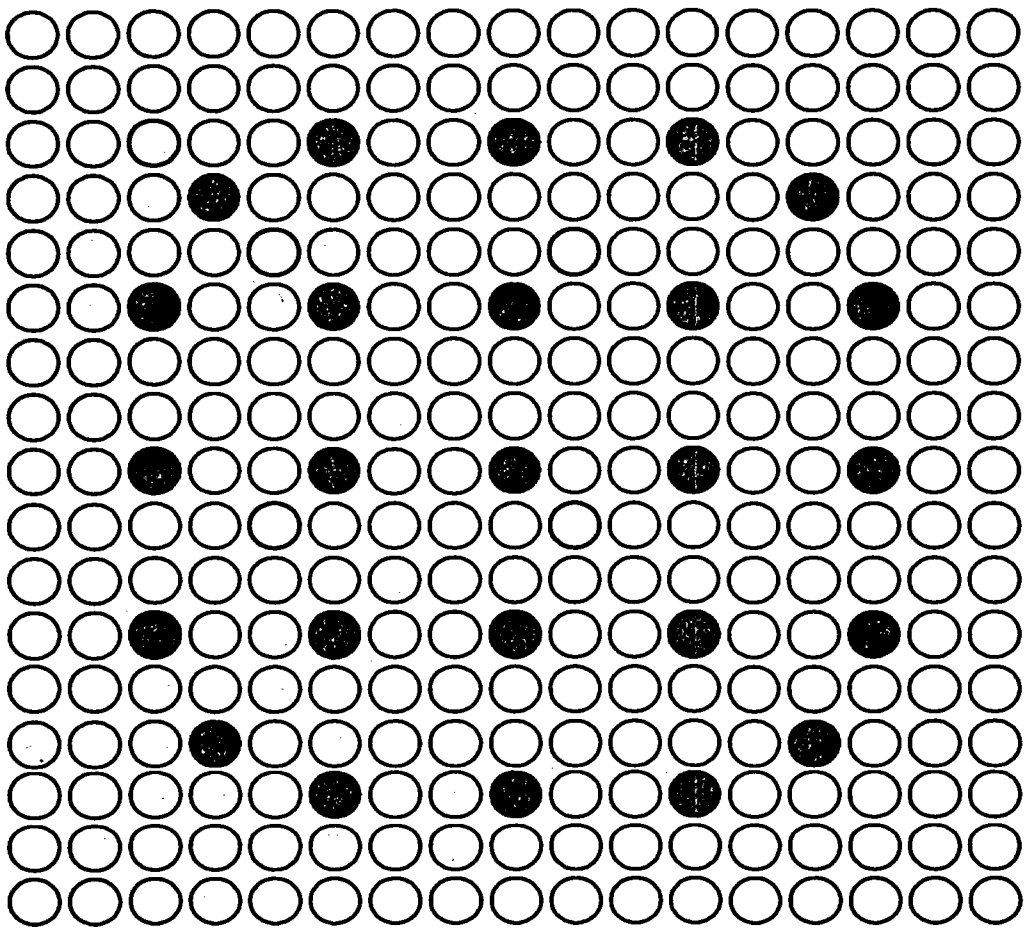


Figure 2-1 Westinghouse AP1000 17x17 Fuel Assembly Cross Section (Black circles represent guide tube locations.)

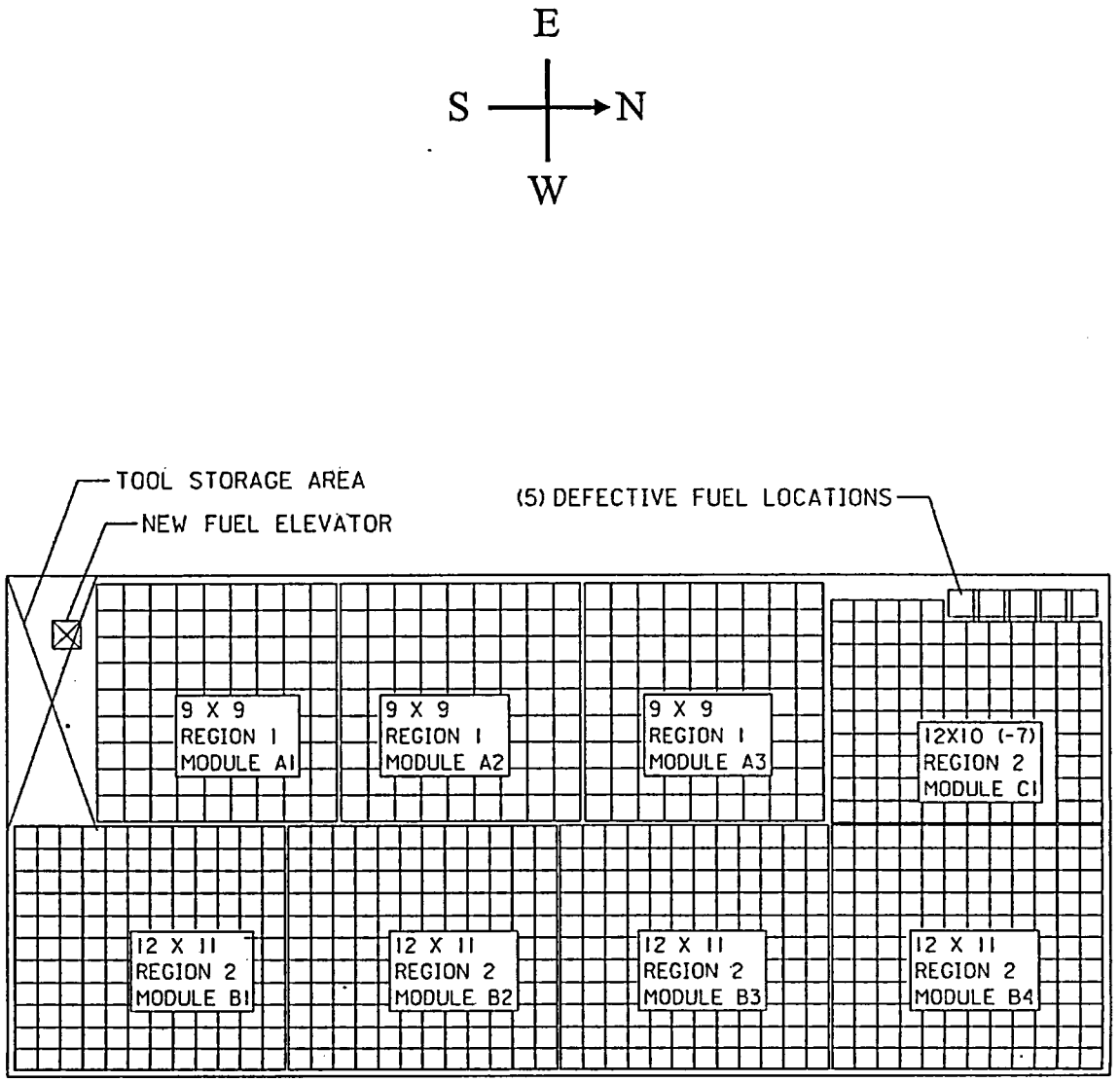
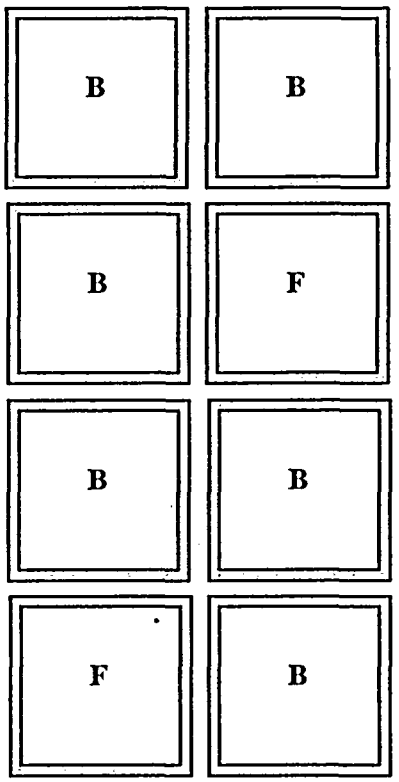


Figure 2-2 AP1000 Spent Fuel Pool Layout
(889 Total Storage Locations)



"F" represents:

- Fresh or depleted fuel assembly with an initial enrichment less than or equal to 5.0 weight-percent U-235.

<u>or</u>

- An empty location.

"B" represents:

- Depleted fuel assembly which meets the requirements of Figure 2-6.
- An empty location.

Note: This figure depicts a repeating array. The arrangement of fresh fuel assemblies within the repeating array must not be changed. Note, however, that a depleted fuel assembly that meets the requirements of Figure 2-6 may be stored in any fresh fuel assembly location.

Figure 2-3 Allowable Fuel Assemblies in Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

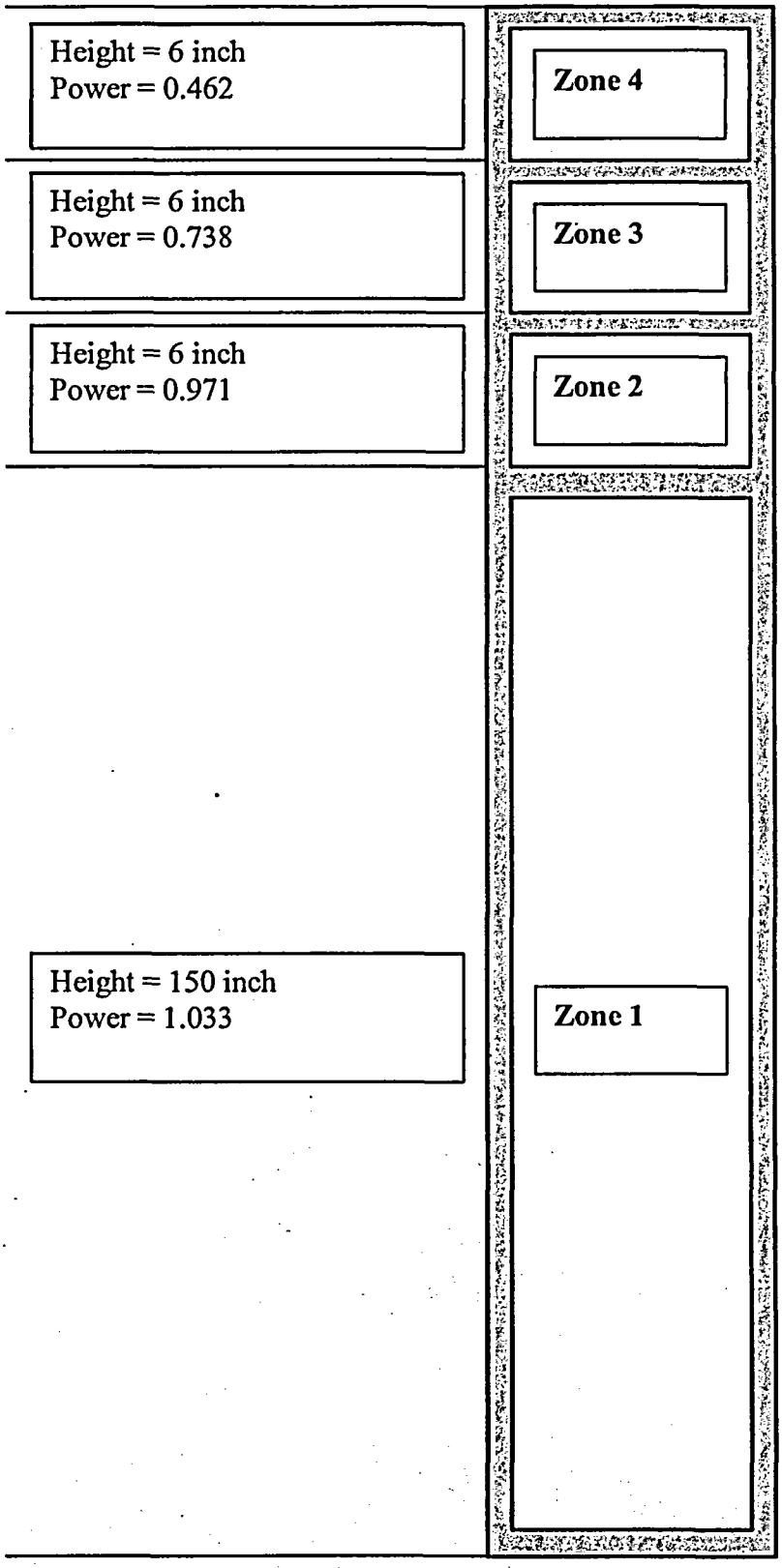


Figure 2-4 Sketch of Axial Zones Used in Fuel Assembly: Four-Zone Model

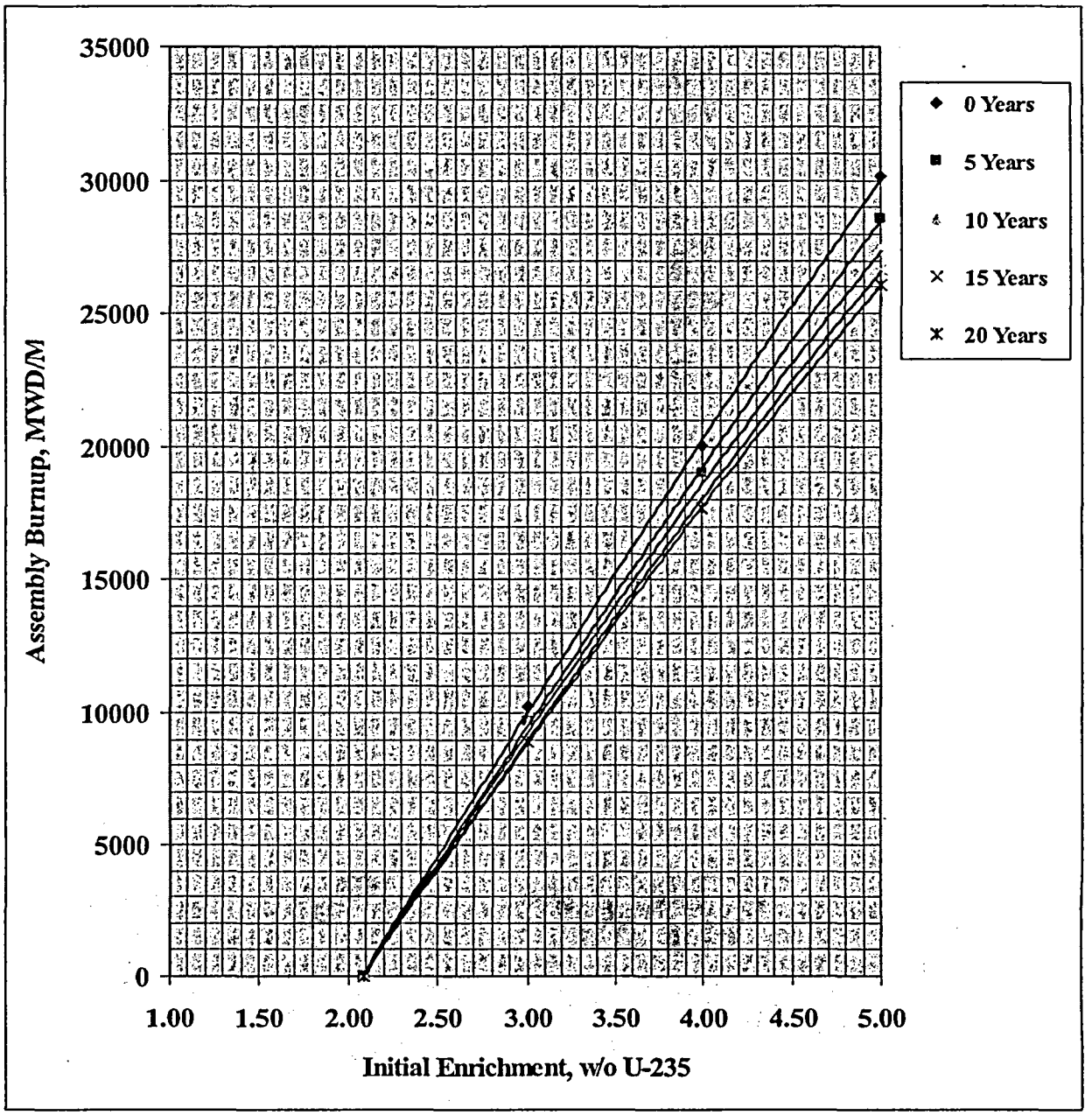


Figure 2-5 Minimum Fuel Assembly Burnup versus Initial Enrichment for Region 2 "All Cell" Storage Configuration

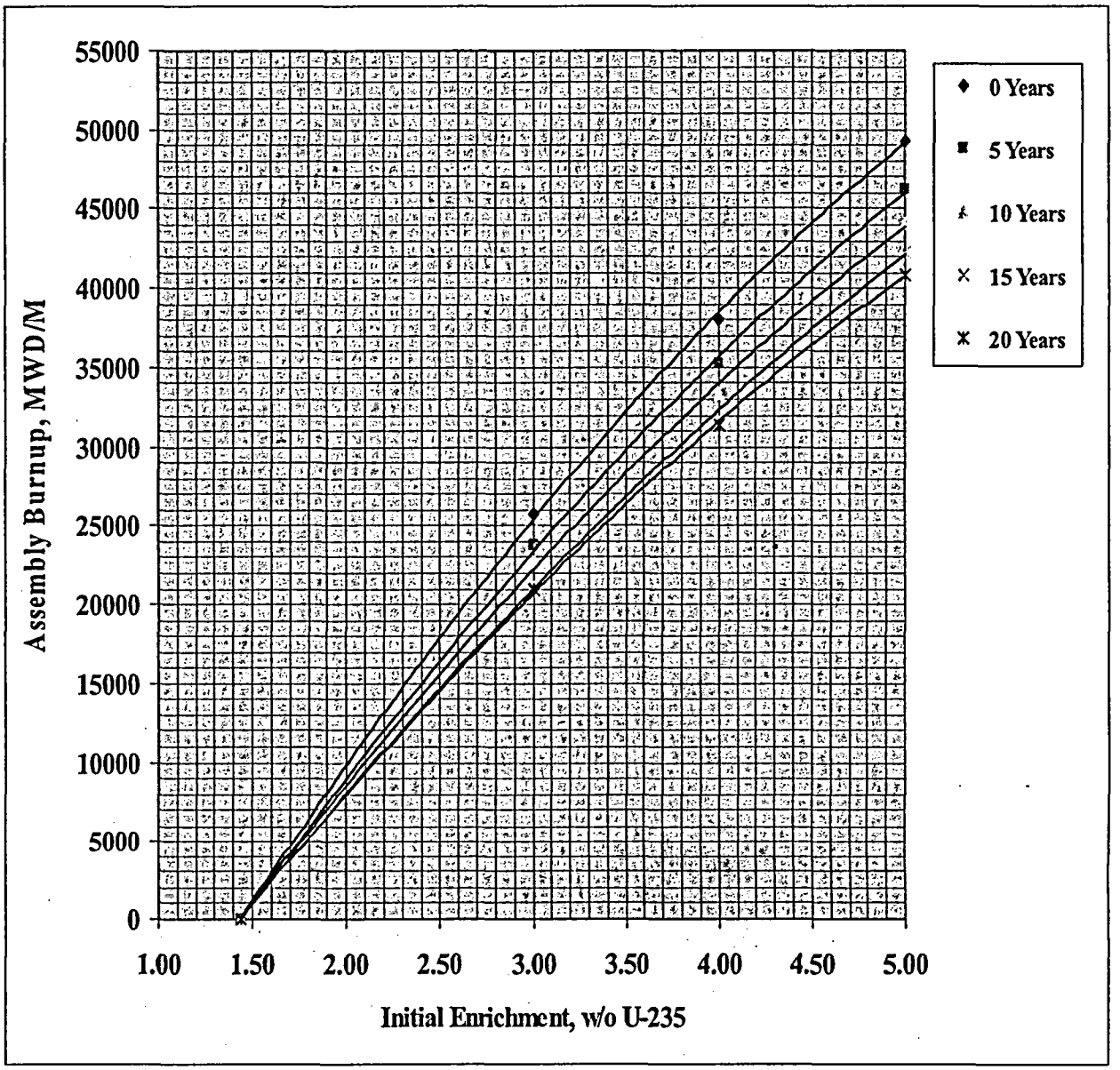


Figure 2-6 Minimum Fuel Assembly Burnup Versus Initial Enrichment for Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configuration

1-out-of-4 5.0 weight-percent Fresh

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
B	B	B	B	A	A	A
F	B	F	B	A	A	A
B	B	B	B	A	A	A
B	F	B	B	A	A	A

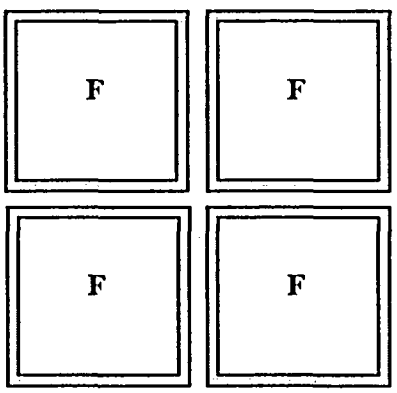
All-Cell

A : Depleted fuel assembly in the Region 2 "All-Cell" storage configuration which meets the requirements of Figure 2-5

B : Depleted fuel assembly in the Region 2 "1-out-of 4 5.0 weight-percent Fresh" storage configuration which meets the requirements of Figure 2-6

F : Fresh or depleted fuel assembly with an initial enrichment less than or equal to 5.0 weight-percent U-235

Figure 2-7 Allowable Interface between "All-Cell" and "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configurations in Region 2



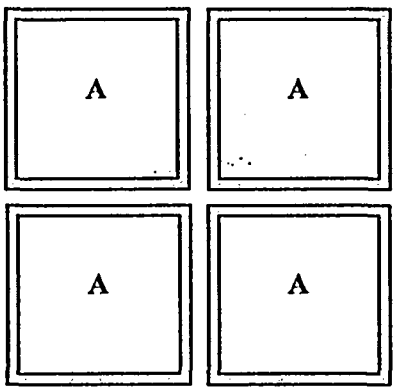
"F" represents:

- Fresh or depleted fuel assembly with an initial enrichment less than or equal to 5.0 weight-percent U-235.

<u>or</u>

- An empty location.

Figure 2-8 Allowable Fuel Assemblies in Region 1 "All-Cell" Storage Configuration



"A" represents:

- Depleted fuel assembly which meets the requirements of Figure 2-5.

<u>or</u>

- An empty location.

Figure 2-9 Allowable Fuel Assemblies in Region 2 "All-Cell" Storage Configuration

3 REGULATORY IMPACT

The design of the AP1000 Spent Fuel Storage Rack is addressed in subsection 9.1.2, "Spent Fuel Storage," of the U.S. NRC Final Safety Evaluation Report (Reference 2). The completion of the criticality analysis for the AP1000 Spent Fuel Storage Racks is identified in the Final Safety Evaluation Report as COL Action Item 9.1.6-4.

The changes to the DCD presented in this report do not represent an adverse change to the design functions of the AP1000 Spent Fuel Storage Racks, or to how design functions are performed or controlled. The criticality analysis of the spent fuel racks is consistent with the description of the analysis in subsection 4.3.2.6.1, "Criticality Design Methods Outside the Reactor," and the description of the design presented in subsection 9.1.2, "Spent Fuel Storage," of the DCD. The changes to the DCD do involve revising a DCD-described evaluation methodology. This methodology is the application of soluble boron credit in the spent fuel pool. Although new to the AP1000, Westinghouse has used this methodology for several years on operating plants spent fuel pools including Diablo Canyon and Millstone Unit 2. The Westinghouse methodology for soluble boron credit has been accepted by U.S. NRC for these plants. Subsection 4.3.2.6.1 of the DCD has been revised accordingly to include a summary description of the soluble boron credit methodology. The changes to the DCD do not involve a test or experiment not described in the DCD. The updating of the discussion of the criticality analysis in subsection 4.3.2.6.1 and associated reference changes primarily is due to providing additional information and refinement and does not represent a departure from an evaluation methodology in the DCD used in establishing design bases. The DCD change does not require a license amendment per the criteria of VIII. B. 5.b. of Appendix D to 10 CFR Part 52.

The changes described do not involve design features used to mitigate severe accidents. Therefore, a license amendment based on the criteria of VIII. B. 5.c of Appendix D to 10 CFR Part 52 is not required.

The closure of the COL Information Item will not alter barriers or alarms that control access to protected areas of the plant. The closure of the COL Information Item will not alter requirements for security personnel. Therefore, the closure of the COL Information Item does not have an adverse impact on the security assessment of the AP1000.

Two new Technical Specifications will be created to control the spent fuel pool boron concentration and the spent fuel pool loading configuration. These proposed changes to the Technical Specifications will be included in a future COLA Technical Report submittal which will include these and other proposed Technical Specification changes.

4 REFERENCES

1. APP-GW-GL-700 Rev. 15, "AP1000 Design Control Document", Westinghouse Electric, November 2005.

2. Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, U. S. Nuclear Regulatory Commission, September 2004.

3. Westinghouse Calculation, APP-FS02-N1C-002 Rev. 0, "AP1000 Spent Fuel Storage Racks Criticality Analysis," June 2006. (Westinghouse Proprietary)

4. AP1000 Standard Combined License Technical Report, APP-GW-GLR-033, Rev. 0, "Spent Fuel Storage Racks Structural/Seismic Analysis," June 2006.

5. Letter, G. S. Vissing (NRC) to R. C. Mecredy (RGE), "R. E. Ginna Nuclear Power Plant – Amendment Re-Revision to the Storage Configuration Requirements within the Existing Storage Racking and Taking Credit for a Limited Amount of Soluble Boron," December 7, 2000.

6. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)," October 25, 1996.

7. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

8. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," February 1998.

9. APP-SFS-M3-001 Rev. 0 "Spent Fuel Pool Cooling System Specification Document", Westinghouse Electric, March 2006. (Westinghouse Proprietary)

10. "SCALE 4.4a- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC CODE PACKAGE CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000.

11. "DIT: Discrete Integral Transport Assembly Design Code," CE-CES-11, Revision 4-P, April 1994.

12. M. N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report," BAW-1484-7, July 1979.

13. S. R. Bierman and E.D. Clayton, "Critical Experiments with Subcritical Clusters of 2.35 Wt% U-235 Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6," NUREG/CR-1547, PNL-3314, July 1980.

14. S.R. Bierman and E.D. Clayton, "Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 Wt% U-Enriched UO2 Rods in Water with Steel Reflecting Walls," Nuclear Technology, Vol. 54, pg. 131, August 1981.

15. International Handbook of Evaluated Criticality Safety Benchmark Experiments, Nuclear Energy Agency and Organization for Economic Cooperation and Development, NEA/NSC/DOC (95) 03, September 2005 Edition.

16. M. G. Anness, P. F. O'Donnell, and M. A. Cunningham, "Unix SCALE 4.4 Criticality Validation and Benchmarking Analysis," CN-CRIT-206, 2004. (Westinghouse Proprietary)

17. W. Marshall, et al., "Criticality Safety Criteria," TANS Vol. 35, pg. 278, 1980.

18. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandia Corporation Monograph, March 1963.

19. Holtec Letter Number 1540001, "AP 1000 Spent Fuel, New Fuel and In-Containment Racks Data Sheet", April 27, 2006

20. P. A. Narayanan, "Isotopic Number Densities for Discharged Westinghouse 17x17 Fuel Assemblies," A-GEN-FE-0118, Revision 0, September 26, 2000.

21. Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages, DOE/RW-0472, Rev. 2, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, September 1998.

22. "Boraflex Degradation in Spent Fuel Storage Racks," Generic Letter 96-04 U.S. Nuclear Regulatory Commission, June 1996.

23. "Use of Metamic® in Fuel Pool Applications," Holtec Report No. HI-2022871, Holtec International, August 2002. (Holtec Proprietary)

24. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding the Use of Metamic® in Fuel Pool Applications Facility Operating License NOS DPR-51 and NPF-6 Entergy Operations, Inc Arkansas Nuclear One, Unit Nos. 1 and 2 Docket Nos. 50-313 and 50-368, U. S. Nuclear Regulatory Commission, June 17, 2003.

5 DCD MARKUP

The following DCD markup identifies how COL Application Final Safety Analysis Reports should be prepared to incorporate the subject change.

Revise subsection 4.3.2.6.1 as follows:

4.3.2.6.1 Criticality Design Method Outside the Reactor

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping, and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

~~The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI 57.2 (Reference 19) and ANSI 57.3 (Reference 20).~~ The design criteria are consistent with General Design Criterion (GDC) 62, Reference 19, and NRC guidance given in Reference 20. The applicable 10 CFR Part 50.68 requirements are as follows:

1) The maximum K-effective value, including all biases and uncertainties, must be less than 0.95 with soluble boron credit and less than 1.0 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 4 of Paragraph b. Note that the specific terminology is:

"If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

2) The maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent U-235. Note this design criterion is provided in 10 CFR Part 50.68, Item 7 of Paragraph b. Note that the specific terminology is:

"The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight."

The following conditions are assumed in meeting this design bases:

- The fuel assembly contains the highest enrichment authorized without any control rods or non-integral burnable absorber(s) and is at its most reactive point in life.
- For flooded conditions, the moderator is pure water at the temperature within the design limits which yields the largest reactivity.
- The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design.
- Mechanical uncertainties are treated either by using worst-case conditions or by performing sensitivity studies and obtaining appropriate uncertainties.
- Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.
- ~~Where borated water is present, credit for the dissolved boron is not taken, except under postulated accident conditions, where the double contingency principle of ANSI N16.1-1975 is~~

~~applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident.~~

- ~~For fuel storage application, water is usually present. However, the design methodology also prevents accidental criticality when fuel assemblies are stored in the dry condition. For this case, possible sources of moderation such as those that arise during fire fighting operations are included in the analysis. The design basis k_{eff} is 0.98 as recommended in the Standard Review Plan.~~

The design method which determines the criticality safety of fuel assemblies outside the reactor uses the SCALE 4.4a system~~,~~ (Reference 21) ~~Rev. 4,~~ which includes the BONAMI and NITAWL-II codes for cross sections generation and the KENO-V.a code for reactivity determination.

The 238~~18~~ groups library obtained from ENDF/B-~~I~~V data is the origin of the 442~~7~~ groups library used in these analyses and in the modeling of the critical experiments which are the basis for the qualification of the SCALE/KENO-V.a (Reference 26~~1~~) calculation system.

A set of 304~~1~~ critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials and enrichments, all of them adequate for qualifying methods to analyze light water reactor lattices (References 22 to 25, 27~~6~~).

The analysis of the 304~~1~~ critical experiments results in an average Keff of 0.996938. Comparison with the measured values results in a method bias of 0.00316~~2~~. The standard deviation of the set of reactivities is 0.00285~~396~~. The 95/95 tolerance factor is 2.22~~118~~.

~~The total uncertainty (TU) to be added to criticality calculations:~~

$$\cancel{TU =} \left[(ks)^2_{method} + (ks)^2_{KENO} + \sum_i (ks)^2_{mech} \right]^{1/2}$$

~~where:~~

~~$(ks)_{method}$ = method uncertainty as discussed above.~~

~~$(ks)_{KENO}$ = the statistical uncertainty associated with the particular KENO calculation being used.~~

~~$(ks)_{mech}$ = a series of statistical uncertainties associated with mechanical tolerances, such as thicknesses and spacings. If worst-case assumptions are used for tolerances, this term will be zero.The criticality design criteria are met when the calculated effective multiplication factor plus the total uncertainty is less than 0.95 or, in the special case defined above, 0.98.~~

The analytical methods employed herein conform with ANSI N18.2 (Reference 3), Section 5.7, Fuel Handling System; ANSI N16.9 (Reference 29)~~, ANSI 57.2 (Reference 19), subsection 6.4.2, ANSI 57.3 (Reference 20), Section 6.2.4;~~ NRC Standard Review Plan, subsection 9.1.2, the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 30).

4.3.2.6.2 Soluble Boron Credit Methodology

The methodology used in this analysis for soluble boron credit is analogous to that of Reference 62, and it uses analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 63. Reference 62 was reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC). The methodology used in this analysis and in Reference 62 uses axially distributed burnups to represent discharged fuel assemblies.

A brief summary of the analysis approach and criteria follows:

25. Determine the fresh and spent fuel storage configurations using no soluble boron conditions so that the 95/95 upper tolerance limit value of K_{eff}, including applicable biases and uncertainties, is less than 0.995. This is accomplished with infinite arrays of either fresh or spent fuel assembly configurations. Note that the actual NRC K_{eff} limit for this condition is unity. Therefore, an additional margin of 0.005 ΔK_{eff} units is included in the analysis results.

26. Determine the amount (ppm) of soluble boron necessary to reduce the K_{eff} value of all storage configurations by at least 0.05 ΔK_{eff} units. This is accomplished by constructing a KENO model for the entire spent fuel pool, which includes the storage configurations that are least sensitive to changes in soluble boron concentration. As an example, storage configurations that contain depleted fuel assemblies (and represented by depleted isotopics) are less reactivity-sensitive to changes in soluble boron concentration than a fuel assembly represented by zero burnup and relatively low initial fuel enrichment.

27. Determine the amount of soluble boron necessary to compensate for 5 percent of the maximum burnup credited in any storage configuration. In addition, determine the amount of soluble boron necessary to account for a reactivity depletion uncertainty of 1.0% ΔK_{eff} per 30,000 MWD/MTU of credited fuel burnup. This is accomplished by multiplying this derivative by the maximum burnup credited in any storage configuration and converting to soluble boron using the data generated in Step 2.

28. Determine the largest increase in reactivity caused by postulated accidents and the corresponding amount of soluble boron needed to offset this reactivity increase.

An alternative form of expressing the soluble boron requirements is given in Reference 6. The final soluble boron credit (SBC) requirement is determined from the following summation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

Where,

SBC_{TOTAL} = total soluble boron credit requirement (ppm).

$SBC_{95/95}$ = soluble boron requirement for 95/95 K_{eff} less than or equal to 0.95.

SBC_{RE} = soluble boron required to account for burnup and reactivity uncertainties.

SBC_{PA} = soluble boron required to offset accident conditions.

For the analyses, minimum burnup limits established for fuel assemblies to be stored in the storage configurations racks include burnup credit established in a manner that takes into account approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

Revise the references in subsection 4.3.5 as follows:

4.3.5 References

19. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."~~ANSI N57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations."~~

20. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," February 1998.~~ANSI N57.3-1983, "Design Requirements for New Fuel Storage Facilities at Light-Water Reactor Plants."~~

21. "SCALE 4.4a- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC CODE PACKAGE CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000.~~NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Oak Ridge National Laboratory; Rev. 4, January 1990~~.

26. M. G. Anness, P. F. O'Donnell, and M. A. Cunningham, "Unix SCALE 4.4 Criticality Validation and Benchmarking Analysis," CN-CRIT-206, 2004.~~NEDO-32028 "MCNP Light Water Reactor Critical Benchmarks," S. Sitaraman, March 1992.~~

62. Letter, G. S. Vissing (NRC) to R. C. Mecredy (RGE), "R. E. Ginna Nuclear Power Plant – Amendment Re-Revision to the Storage Configuration Requirements within the Existing Storage Racking and Taking Credit for a Limited Amount of Soluble Boron," December 7, 2000.

63. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)," October 25, 1996.

Revise the second paragraph of subsection 9.1.2.3 as follows:

9.1.2.3 Safety Evaluation

The design of the racks is such that K_{eff} remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Because of the close spacing of the cells, it is impossible to insert a

fuel assembly in other than design locations. Inadvertent insertion of a fuel assembly between the rack periphery and the pool wall or placement of a fuel assembly across the top of a fuel rack is considered a postulated accident, and as such, realistic initial conditions such as boron in the pool water are assumed. These accident conditions have an acceptable K_{eff} of less than 0.95. The spent fuel storage racks are purchased equipment. The purchase specification for the spent fuel storage racks ~~will~~ requires ~~the vendor to perform~~ a criticality analysis of the spent fuel storage racks. The criticality evaluation will consider the inherent neutron absorbing effect of the materials of construction, including fixed neutron absorbing "poison" material.

Revise the second paragraph of subsection 9.1.6 as follows:

9.1.6 Combined License Information for Fuel Storage and Handling

Complete. ~~The Combined License applicant is responsible for a~~ A confirmatory criticality analysis for the spent fuel racks, as described in subsection 9.1.2.3 ~~.~~ is provided in APP-GW-GLR-029 Rev. 0, AP1000 Standard Combined License Technical Report (Reference 168). This ~~analysis should~~ report addresses the degradation of integral neutron absorbing material in the new fuel pool storage racks as identified in GL-96-04, and assesses the integral neutron absorbing material capability to maintain a 5-percent subcriticality margin.

Revise subsection 9.1.7 as follows:

9.1.7 References

168. APP-GW-GLR-029 Rev. 0 "AP1000 Standard Combined License Technical Report Spent Fuel Storage Racks Criticality Analysis", Westinghouse Electric Company LLC, June 2006.