Criticality Analysis of New and Spent Fuel Storage Racks

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ABSTRACT

This report presents the criticality and depletion calculation methodology that is used for the design of the new and spent fuel storage racks of the APR1400 design. The contents of this document include acceptance criteria, the description of the fuel storage racks, the computer codes, the criticality analysis of new and spent fuel storage racks, the accident analysis, and the limitations of analysis.

The new fuel storage racks provide onsite storage capacity of 112 new fuel assemblies corresponding to one (1) refueling batch plus additional margin. The spent fuel storage pit is made up of region I and region II. The fresh or partially burnt fuel assemblies are stored in region I which has a storage capacity for one (1) full core, one (1) refueling batch, and five (5) damaged fuels. Region I storage area is designed to accommodate fuel assemblies with initial enrichment up to 5.0 weight percent U-235. Region II has a storage capacity of spent fuel assemblies generated during plant operation of twenty (20) years at full power in case of an 18-month fuel cycle. Spent fuel storage racks are capable of receiving 1,792 fuel assemblies.

The SCALE 6.1.2 code package is used for the depletion and criticality calculations. Among the modules of the SCALE 6.1.2 code package, the TRITON module is used for generating cross section libraries and the ORIGEN-ARP is used for depletion calculations with cross section libraries generated using the TRITON module. The CSAS5 module with KENO-V.a is used for the criticality calculation using the isotopic content from the depletion calculation. The ENDF/B-VII 238-group library is used for the depletion and criticality calculations.

The depletion and criticality calculations are performed to evaluate criticality safety of new and spent fuel storage racks. The biases and uncertainties by the calculation methods and variations of design parameters are analyzed and the results are included to calculated k_{eff}. The postulated accident analyses such as a dropped fuel assembly, a misloaded fuel assembly and a boron dilution accident are also performed.

All the effective neutron multiplication factors for new and spent fuel storage racks meet the acceptance criteria of the 10 CFR 50.68 described as below under normal and accident conditions with some limitations on fuel, operation and spent fuel pool.

The acceptance criteria of the new fuel storage racks are such that k_{eff} including all biases and uncertainties does not exceed 0.95 with full density unborated water and 0.98 with optimum moderation in the new fuel storage racks, at a 95 percent probability and 95 percent confidence level. For spent fuel storage racks, the credit is taken for soluble boron so that the k_{eff} including all biases and uncertainties does not exceed 0.95 with borated water, at a 95 percent probability and 95 percent confidence level, and remains below 1.00 with full density unborated water, at a 95 percent probability and 95 percent confidence level.

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ACRONYMS AND ABBREVIATIONS

2-D	two dimensional
3-D	three dimensional
AFG	average energy group of neutrons causing fission
EALF	energy of average lethargy causing fission
ENDF	evaluated nuclear data file
FA	fuel assembly
G/T	guide tube
GWd	gigawatt days
HTC	Haut Taux de Combustion
I.D.	inner diameter
k_{eff}	effective neutron multiplication factor
LCO	limiting conditions for operation
METAMIC™	trademark of Metamic, LLC
MTU	metric ton uranium
MWt	megawatts thermal
NFR	new fuel storage rack
NFSP	new fuel storage pit
O.D.	outer diameter
PWR	pressurized water reactor
SFP	spent fuel pool
SFR	spent fuel storage rack
SS	stainless steel
wt%	weight percent

1 INTRODUCTION

This report documents the criticality safety analysis of the new fuel storage rack (NFR) and the spent fuel storage rack (SFR) of the APR1400 design. This report includes the acceptance criteria, the description of the fuel storage racks, computer codes, the criticality analysis of new and spent fuel storage racks, the accident analysis, and limitations of analysis.

The acceptance criteria and relevant guidance for the criticality safety evaluation for new and spent fuel storage racks are as follows: 10 CFR 50 Appendix A, General Design Criterion (GDC) 62 (Reference 1), 10 CFR Part 50.68 (Reference 2), NRC guidance (Reference 3), and NUREG/CR-6698 (Reference 4). The 10 CFR Part 50.68 (b) items (2) and (3) for new fuel storage racks and item (4) for spent fuel storage racks are applied as the criticality safety design criteria.

Chapter 2 and 3 denote the criticality analysis for new and spent fuel storage racks, respectively. Chapter 4 describes the accident analysis for a dropped fuel assembly, a misloaded fuel assembly and a boron dilution accident. Chapter 5 describes the limitations of analysis, and Chapter 6 provides the conclusions.

2 CRITICALITY ANALYSIS OF NEW FUEL STORAGE RACK

2.1 Design Input Data

New fuel assemblies are stored in the NFR in a dry fuel storage pit. The NFR consists of 2 racks, each of which has 7x8 cells array, so a total of 112 new fuel assemblies can be stored in the NFR. Figure 2.1-1 and Figure 2.1-2 show the top and front view of the NFR calculation model. As shown in the Figures the racks are designed to be located in the new fuel storage pit (NFSP) surrounded by concrete walls. To set up boundaries of SCALE model, 30 cm thickness of concrete is assumed to build bounding walls in side and bottom. Concrete with 30 cm thickness is enough to take account of back scattering effect of neutrons from the wall to the NFR.

The design input data for the criticality calculation for the NFR are summarized in Table 2.1-1 and the composition ratio of constituent nuclides for the SCALE model is shown in Table 2.1-2.

Table 2.1-1 Design Input Data of Fuel Assembly and New Fuel Storage Rack

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Table 2.1-2 Composition Ratio of Constituent Nuclide for Calculation Model

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Figure 2.1-1 Top View of Criticality Calculation Model for the NFR

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Figure 2.1-2 Front View of Criticality Calculation Model for the NFR

2.2 Key Assumptions

For a normal condition of new fuel storage rack (NFR), the effective multiplication factor for the fuel rack system is very low since the fuel assemblies are stored in dry environment. Therefore, the criticality evaluation for the NFR is performed for the postulated accident situations such as a mist environment, which provides an optimum moderation condition, and an immersion in pure water with the maximum density.

The key assumptions applied for the criticality analysis of the NFR are as follows:

a. The design maximum enrichment of 5.0 wt% is applied to all the UO_2 fuel rods in the NFR,

b. There is no zoning of enrichment and no burnable poison rod in the fuel assembly (all fuel rods have the same maximum enrichment in the fuel assembly),

c. Axial blankets in the fuel rod are not considered for conservatism (the same maximum enrichment is assumed along the entire axial length of the effective fuel region),

d. The structural materials in the upper and lower parts of fuel rod such as plenum, spring, end caps, etc., and the grids in the fuel assembly are assumed as water,

e. The structural materials beyond the effective fuel length area in the NFR are ignored and replaced by water, and

f. The temperature of a fuel assembly, a rack structure, and water are assumed as room temperature.

2.3 Design Method

The CSAS5 module in the SCALE 6.1.2 code system (Reference 5) is used to calculate the effective neutron multiplication factor (k_{eff}) for the NFR with immersion in pure water and optimum moderation conditions. The v7-238 cross section library based on ENDF/B-VII (Reference 6) is used for CSAS5 module calculations.

Calculated result of criticality analysis should reflect the bias and bias uncertainty obtained from benchmark calculations based on the criticality experiments. Moreover, additional uncertainties which account for the variations of design parameters and mechanical tolerances of the NFR and fuel assembly should be included in the criticality analysis result. To take account of the bias and uncertainties, k_{eff} is evaluated from the following expression:

$$k_{eff} = k(calc) + \Delta k(bias) + \Delta k(uncert)$$

where:

k_{eff} = effective neutron multiplication factor,

$k(calc)$ = calculated nominal k_{eff},

$\Delta k(bias)$ = sum of biases determined from critical benchmark comparisons, and

$\Delta k(uncert)$ = statistical summation of all tolerance and uncertainty components (square root of the sum of the squares).

The bias and bias uncertainty due to the criticality code are provided in Reference 7 and the uncertainties due to variations of design parameters and mechanical tolerances are calculated through the sensitivity analyses for the dimensional and material tolerances of the NFR and fuel assembly.

2.4 Criticality Analysis for New Fuel Storage Rack

2.4.1 Criticality Calculation

The criticality analyses for the NFR consider waters with an optimum moderation condition and the maximum densities, therefore criticality calculations are performed for water densities ranged from 0.01 g/cm^3 to 1.0 g/cm^3 to determine the maximum k_{eff} for an optimum moderation condition. The criticality calculation model including NFSP is shown in Figure 2.4-1.

The calculation results are presented in Table 2.4-1 as the calculated nominal k_{eff} with corresponding water densities. The nominal k_{eff} values in the Table 2.4-1 do not contain any bias or uncertainties.

2.4.2 Bias and Uncertainty

The bias and uncertainties by the calculation methods and variations of design parameters are estimated from the following items:

a. Bias and bias uncertainty of a criticality calculation method,

b. Statistical uncertainty of the Monte Carlo calculation, and

c. Uncertainty due to tolerances or variations in the design parameters.

The basis of bias and uncertainty items and their corresponding values considered for the criticality analysis of the NFR are described in below.

(1) Bias and bias uncertainty of a criticality calculation method

Bias and bias uncertainty are evaluated to validate the criticality analysis methodology through the benchmark calculations based on the criticality experiments (Reference 7).

a. Bias: [ ]TS

b. Bias uncertainty: [ ]TS

(2) Statistical uncertainty of the Monte Carlo calculation

Statistical uncertainty (2σ) of the criticality calculation for full density water model (reference model) is [ ]TS as shown in Table 2.4-2.

(3) Uncertainty due to tolerances or variations in the design parameters

To evaluate uncertainties due to the tolerances in the mechanical and material specifications of the fuel and rack structures, sensitivity analyses are performed for the fuel rack cell in various conditions including the dimensional and material tolerances of the structure. Items in the sensitivity analysis for the criticality uncertainty evaluation are as follows:

a. UO_2 pellet stack density: [ ]TS,

b. UO_2 pellet diameter: [ ]TS,

c. Fuel rod pitch: [ ]TS,

d. Fuel clad outer diameter: [ ]TS,

e. Fuel assembly position in fuel rack cell: [ ]TS,

f. NFR cell pitch: [ ]TS, and

g. NFR cell thickness: [ ]TS.

The uncertainty analyses are performed for the unit cell model of the NFR as shown in Figure 2.4-2. The calculation results for the uncertainty analysis due to the tolerances or variations in the design parameters are summarized in Table 2.4-2. To determine the reactivity difference (Δk_i) associated with a specific manufacturing tolerance, the k_{eff} calculated for the reference model is compared to that for the model with an individual tolerance. The Δk_i due to a tolerance is then calculated as follows:

$$\Delta k_i = k_i - k_R + 1.645\sqrt{\sigma_i^2 + \sigma_R^2}$$

where:

k_i = k_{eff} with the tolerance,

k_R = k_{eff} for the reference model,

σ_i = Monte Carlo standard deviation for the case with tolerance,

σ_R = Monte Carlo standard deviation for the reference model, and

1.645 = One-sided 95/95 confidence interval factor.

The resultant uncertainty due to tolerances or variations in the design parameters which is calculated as square root of the sum of the squares of individual Δk_i is presented in the last row of Table 2.4-2.

The evaluated total bias and uncertainty (Δk_{NFR}), which includes bias for the criticality analysis of the NFR is determined as follows:

[ ]TS

Table 2.4-1 Calculated Nominal k_{eff} of the NFR

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Table 2.4-2 Calculated Uncertainties due to Tolerances or Variations in the Design Parameters

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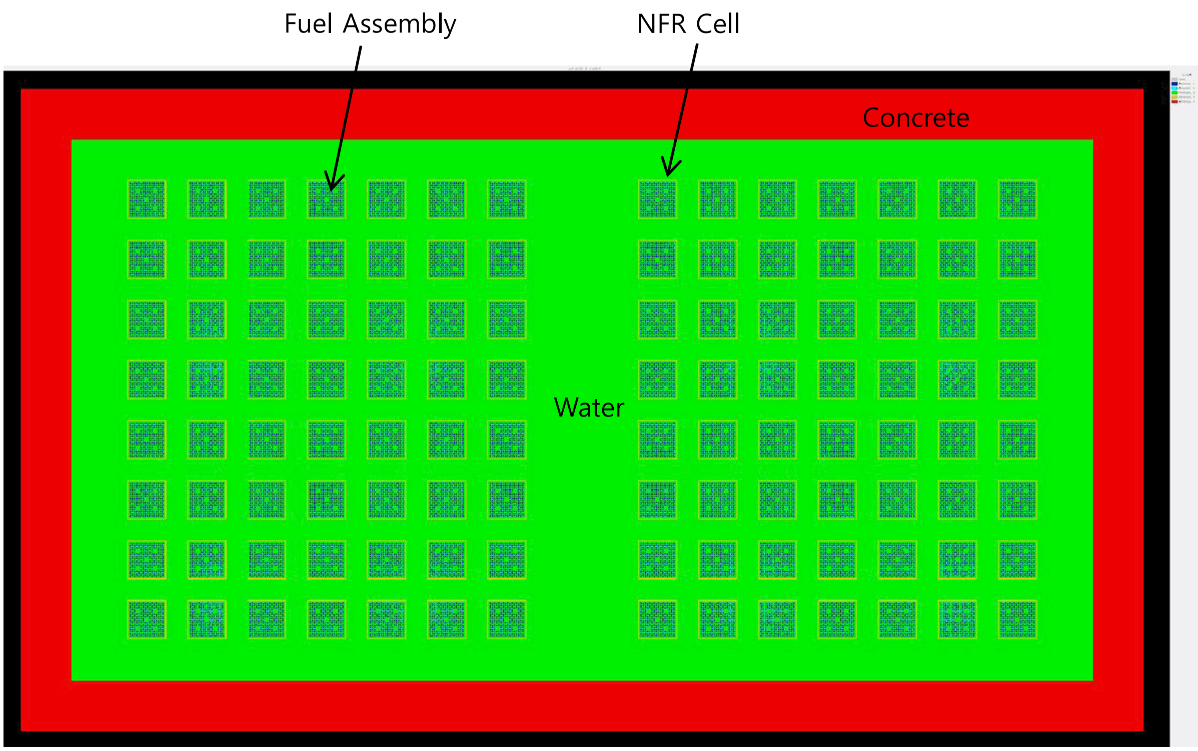

Figure 2.4-1 Criticality Calculation Model for the NFR in a New Fuel Storage Pit

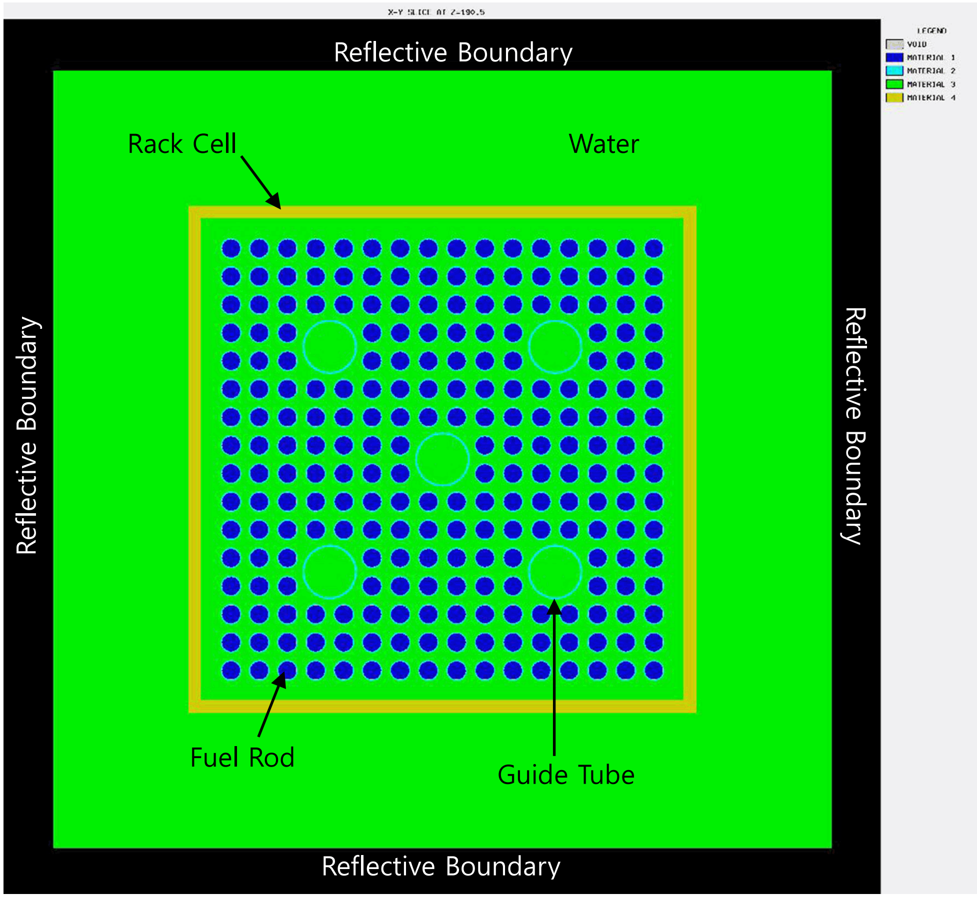

Figure 2.4-2 Unit Cell Model for Uncertainty Analysis

2.5 Results

The criticality analysis for the NFR with 5.0 wt% enrichment of the PLUS7 fuel is performed by using SCALE 6.1.2 code. Table 2.5-1 and Figure 2.5-1 show the evaluated effective neutron multiplication factors according to various water densities. The evaluated results for the criticality analysis include the evaluated total bias and uncertainty (Δk_{NFR}). An optimum moderation condition occurs at water density of 0.14 g/cm^3.

The evaluated effective neutron multiplication factors for the NFR have the additional margin due to the conservative assumptions for the criticality analysis as follows:

- The design maximum enrichment of 5.0 wt% is applied to all the UO_2 fuel rods in the NFR,
- Miscellaneous structures such as grid, spring, end caps, etc., are not included in the calculation model,
- Burnable absorber rods in fuel assembly or axial blankets in fuel rod are not considered in the calculation model, and
- Zoning to alleviate the power peak in the fuel assembly is not considered and all fuel rods are assumed to have the same maximum enrichment.

As a conclusion, the evaluated effective neutron multiplication factors for the NFR satisfy the acceptance criteria as follows:

Description	k_{eff}	Acceptance criteria
k_{eff} for flooded by pure water	0.91257	≤ 0.95
k_{eff} for optimum moderation	0.93298	≤ 0.98

Table 2.5-1 Effective Multiplication Factors for the NFR

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Figure 2.5-1 Effective Multiplication Factors for the NFR according to Water Density Changes

3 CRITICALITY ANALYSIS OF SPENT FUEL STORAGE RACK

3.1 Design Input Data

The spent fuel storage pit of the ARP1400 design is made up of region I and region II. The fresh or partially burnt fuel assemblies are stored in region I which has a storage capacity for one (1) full core, one (1) refueling batch, and five (5) damaged fuel assemblies. Region I storage area is designed to accommodate fuel assemblies with initial enrichment up to 5.0 wt% U-235. Region II has a storage capacity of spent fuel assemblies generated during plant operation of twenty (20) years at full power in case of an 18-month fuel cycle. Spent fuel storage racks are capable of receiving 1,792 fuel assemblies and the center-to-center spaces between adjacent fuel assemblies are designed to be 27.5 cm and 22.5 cm for region I and II, respectively, to maintain subcriticality.

The design input data for a criticality analysis of spent fuel storage rack are shown in Tables 3.1-1, 3.1-2, and Figure 3.1-1. The fuel assembly (PLUS7) design data and the operating data are listed in Table 3.1-3.

Table 3.1-1 Design Data for Spent Fuel Pool Region I

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Table 3.1-2 Design Data for Spent Fuel Pool Region II

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Table 3.1-3 Fuel Assembly Design and Operating Data

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Figure 3.1-1 Arrays and Dimensions of Spent Fuel Pool Regions I and II

3.2 Key Assumptions

Key assumptions for the conservative criticality calculation of spent fuel storage rack are as follows:

a. KENO-V.a model for spent fuel pool region I assumes an infinite array of one normal fuel storage cell using all reflective conditions,

b. KENO-V.a model for spent fuel pool region II assumes an infinite array of 2x2 storage cells with periodic boundary conditions for sides and reflective conditions for top and bottom,

c. KENO-V.a model assumes 30 cm of water above and below the active fuel length,

d. The assemblies are assumed as non-blanketed assemblies for conservatism (see Subsection 3.5.3.9.4),

e. Fuel rod cladding and guide tube cladding are only considered as structural material within the active fuel length,

f. Water density of 1.0 g/cm^3 is used for conservatism (see Subsections 3.4.3.5 and 3.5.3.5),

g. Soluble boron is not considered in normal conditions,

h. No burnable absorber rod is considered for conservatism,

i. No zoning around guide tube is considered for conservatism,

j. 12 actinides and 16 fission products recommended in ISG-8 (Reference 8) are considered for spent fuel pool region II. Recommended 28 nuclides are presented in Table 3.2-1,

k. A neutron absorber plate is assumed to have 75% of the minimum B-10 areal density for conservatism, and

l. A neutron absorber plate is assumed to have the minimum length, the minimum width and the maximum thickness allowed by the tolerances.

Key assumptions for the conservative depletion calculation of spent fuel storage rack are as follows:

a. The maximum fuel temperature, [ ]^{TS}, is used,

b. The maximum fuel density, [ ]^{TS}, is used,

c. The maximum moderator temperature, [ ]^{TS}, is used,

d. The maximum cycle average soluble boron concentration, [ ]^{TS}, is used,

e. The maximum power level, [ ]^{TS}, is used,

f. No decay time is considered after the fuel assembly is depleted for conservatism,

g. No burnable poison rod is considered for conservatism,

h. No zoning around guide tube is considered for conservatism,

i. The uniform axial power distribution is assumed and the end effect is considered as a bias, and

j. No burnup credit is considered for spent fuel pool region I.

Table 3.2-1 Considered Nuclides in the Criticality Calculation (Reference 8)

Recommended set of nuclides for actinide-only burnup credit			
U-234	U-235	U-238	
Pu-238	Pu-239	Pu-240	
Pu-241	Pu-242	Am-141	
Recommended set of additional nuclides for actinide and fission product burnup credit			
Mo-95	Tc-99	Ru-101	Rh-103
Ag-109	Cs-133	Sm-147	Sm-149
Sm-150	Sm-151	Sm-152	Nd-143
Nd-145	Eu-151	Eu-153	Gd-155
U-236	Am-243	Np-237	

3.3 Design Method

Design methods of the criticality analysis for spent fuel storage rack are described in following Subsections.

3.3.1 Criticality Calculation

Criticality calculations are performed using the CSAS5/KENO-V.a sequence with the ENDF/B-VII 238-energy-group library. The isotopic contents of burned fuel are generated by the depletion calculation discussed in following Subsections.

3.3.2 Depletion Calculation

The depletion calculations are performed using the ORIGEN-ARP with cross section libraries which are pre-generated by the TRITON-NEWT.

3.3.2.1 ORIGEN-ARP calculation

The ORIGEN-ARP is a depletion analysis sequence of SCALE 6.1.2 to perform point-depletion calculations with the ORIGEN-S code using problem-dependent cross sections. Problem-dependent cross section libraries are generated using the ARP (Automatic Rapid Processing) module using an interpolation algorithm that operates on pre-generated libraries created for a range of fuel properties and operating conditions.

3.3.2.2 TRITON-NEWT calculation

TRITON-NEWT calculation is used to generate cross section libraries for PLUS7 16x16 fuel assembly.

The procedures to generate ORIGEN-ARP cross section libraries by TRITON-NEWT method are described in below.

The first step is to construct a physics model of the fuel lattice of the reactor assembly under consideration. For a given initial fuel enrichment, a TRITON depletion calculation is performed using 2 dimensional depletion analysis sequence of SCALE 6.1.2. TRITON uses an explicit 2-D representation of the fuel assembly using the NEWT discrete ordinates transport code.

The depletion calculation sequences are used to simulate irradiation and depletion of the fuel over the required irradiation history. A burnup analysis is performed using a series of time intervals. During the simulation, cross sections that are representative of the mid-point of each burnup step are created and saved in the library by the depletion sequence.

Each set of burnup-dependent cross sections is stored within the single library, and is accessed sequentially by its position in the library. Position 1 contains fresh-fuel cross sections and the other positions contain cross section sets which correspond to burnup levels characterizing the midpoint of each burnup step in the depletion sequence calculation.

For fuels with multiple enrichment values, the above procedure is repeated for the multiple enrichment values. Cross section changes with enrichment are generally represented using 0.5 wt% increments (1.5, 2.0, 2.5, 3.0, 3.5, 4.0, 4.5, 5.0, 5.5 and 6.0 wt% of U-235).

3.3.3 Bias and Uncertainty Calculation

For spent fuel pool region I, the bias and uncertainties by the calculation methods and variations of design parameters are estimated from the following items:

a. Bias and bias uncertainty of the criticality calculation method,

b. Statistical uncertainty of the Monte Carlo calculation,

c. Uncertainty due to tolerances or variations in the design parameters,

d. Uncertainty due to eccentric fuel assembly positioning, and

e. Bias due to pool cooling water temperature.

For spent fuel pool region II, the bias and uncertainties by the calculation methods, variations of design parameters and the depletion calculation are estimated from the following items:

a. Bias and bias uncertainty of a the criticality calculation method,

b. Statistical uncertainty of the Monte Carlo calculation,

c. Uncertainty due to tolerances or variations in the design parameters,

d. Uncertainty due to eccentric fuel assembly positioning,

e. Bias due to pool cooling water temperature,

f. Bias due to axial burnup distribution (end effect),

g. Bias due to minor actinides and fission products,

h. Uncertainty due to burnup measurement, and

i. Uncertainty due to the depletion calculation.

3.3.4 Calculation of the Loading Curve for Spent Fuel Pool Region II

The loading curve of spent fuel pool region II is a function of burnup versus enrichment to meet the minimum burnup requirement for each initial enrichment satisfying the target k_{eff}. The curve is produced targeting k_{eff} less than 1.0 with consideration of all the biases and uncertainties associated with the analyses.

3.4 Criticality Analysis for Spent Fuel Pool Region I

Spent fuel pool region I is designed to accommodate damaged fuel assemblies and fuel assemblies with initial enrichment up to 5.0 wt% U-235. The damaged fuel assemblies are stored in the canister which is located in the damaged fuel storage cell. Fresh or partially burnt fuel assemblies are stored in the normal fuel storage cell. Therefore, the criticality analysis is conducted to evaluate criticality safety of both normal fuel storage cell and damaged fuel storage cell.

3.4.1 Normal Fuel Storage Cell

The criticality calculation model for normal fuel storage cell is modeled as an infinite array of one normal fuel storage cell with reflective boundary conditions on all sides as shown in Figure 3.4-1. The design input data for the criticality analysis are as follows:

a. Cross section library: ENDF/B-VII based 238 multi-group library,
b. Material composition,
   - Fuel pellet: Fresh 5.0 wt% U-235 with density of [ ]TS g/cm^3,
   - Cladding: ZIRLO,
   - Cooling water: non-borated pure water with density of 1.0 g/cm^3,
   - Neutron absorber: METAMIC™ with B-10 areal density of [ ]TS g B-10/cm^2,
   - Structural material: SS-304,
   - Pool wall: Concrete,
c. Fuel assembly geometric data: See detailed data in Table 3.1-3,
d. Normal fuel storage cell geometric data: See detailed data in Table 3.1-1,
e. KENO-V.a model assumes 30 cm of water above and below the active fuel length, and
f. Reflective boundary conditions are applied for all sides of the calculation model.

To evaluate the gap effect between racks on criticality, sensitive analyses are performed for the gap between racks ranged from 0 mm to 60 mm. Figure 3.4-2 shows the model for a gap effect with 0 mm gap between racks. The effective neutron multiplication factors and the statistical Monte Carlo calculation uncertainties are shown in Table 3.4-1.

3.4.2 Damaged Fuel Storage Cell Criticality

The damaged fuel is stored in a canister which is located in the damaged fuel storage cell, so the size of damaged fuel storage cell is a little bigger than the normal fuel storage cell. The damaged fuel storage cell is made with the same material as normal fuel storage cell.

The criticality calculation model for damaged fuel storage cells is modeled as a 6x8 array as shown in Figure 3.4-3. A 6x8 array consists of five damaged fuel storage cells and 43 normal storage cells**.** The design input data for criticality analysis are almost the same as those of the normal fuel storage cell criticality analysis, except for the geometric data (See detailed data in Table 3.1-1).

The effective neutron multiplication factor and the statistical Monte Carlo calculation uncertainty for the damaged fuel storage cell are shown in Table 3.4-1.

3.4.3 Bias and Uncertainty Calculations

As discussed in Subsection 3.3.3, the bias and uncertainties for criticality analysis of spent fuel pool region I are following items:

a. Bias and bias uncertainty of the criticality calculation method,

b. Statistical uncertainty of the Monte Carlo calculation,

c. Uncertainty due to tolerances or variations in the design parameters,

d. Uncertainty due to eccentric fuel assembly positioning, and

e. Bias due to pool cooling water temperature.

To estimate the reactivity difference (Δk_i) associated with a specific disturbed condition, the k_{eff} for the reference model is compared to the k_{eff} for the individual disturbed condition.

The analyses of the bias and uncertainties are described in the following Subsections.

3.4.3.1 Bias and Bias Uncertainty due to Methodology

The bias and bias uncertainty of the criticality calculation method are evaluated to validate the criticality analysis methodology through the benchmark calculations based on the criticality experiments (Reference 7). As a result of trend analysis discussed in Reference 7, only enrichment showed a statistically significant trend. Table 3.4-2 shows the bias and bias uncertainty as a function of enrichment and Table 3.4-3 shows the area of applicability for the bias and bias uncertainty.

3.4.3.2 Uncertainty due to Monte Carlo Calculation

Statistical uncertainties due to Monte Carlo calculations are listed in Table 3.4-1.

3.4.3.3 Uncertainties due to Mechanical Tolerances

The uncertainties due to mechanical tolerances of the fuel assembly and the rack are summarized in Table 3.4-4. And the detailed assessments are described in the following Subsections.

3.4.3.3.1 Fuel Assembly

The uncertainties due to mechanical tolerances for the fuel assembly including a fuel pellet enrichment, a fuel pellet stack density, a fuel pellet diameter, a fuel cladding diameter, a fuel rod pitch, and a guide tube cladding diameter are evaluated. Table 3.4-4 shows the uncertainties due to mechanical tolerances of fuel assembly:

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3.4.3.3.2 Racks

The uncertainties due to mechanical tolerances about the rack including a cell pitch, a cell wall thickness, and a sheath thickness are assessed. The bias and uncertainty analyses for neutron absorber plate are not considered because 75% of minimum B-10 areal density and bounding dimension parameters (maximum thickness, minimum width and length) are used for a criticality analysis for the purpose of conservatism.

Table 3.4-4 shows the uncertainties due to mechanical tolerances of the racks:

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3.4.3.4 Uncertainty due to Eccentric Fuel Assembly Positioning

The uncertainty due to fuel assembly positioning in the cell is evaluated. Figure 3.4-4 shows the eccentric position of fuel assembly and the evaluation results are shown in Table 3.4-5. The effective neutron multiplication factor of the eccentric fuel assembly positioning model is less than that of normal positioning model as shown in Table 3.4-5. So the uncertainty of fuel assembly positioning is not included in the total uncertainty.

3.4.3.5 Bias due to Pool Cooling Water Temperature

The bias due to the temperature of cooling water in the pool is assessed. The evaluation results are listed in Table 3.4-6 and show that the pool has a negative moderator coefficient, i.e., k_{eff} at the lower temperature is higher than those at the higher temperatures. Therefore, the bias due to cooling water density is not included in the total bias.

3.4.4 Results of Criticality Analysis of Spent Fuel Pool Region I

The criticality analysis for spent fuel pool region I with 5.0 wt% U-235 enrichment of PLUS7 fuel is performed by using SCALE 6.1.2 code. Table 3.4-7 shows the summary of bias and uncertainty. Table 3.4-8 shows the evaluated effective neutron multiplication factors including total bias and uncertainty. The evaluated effective neutron multiplication factors for spent fuel pool region I have additional margin due to the conservative assumptions included in the input parameters for the criticality analysis as follows:

a. The design maximum enrichment of 5.0 wt% is applied to all the UO_2 fuel rods in the spent fuel pool region I,

b. Miscellaneous structures such as grid, spring, end caps, etc., are not included in the calculation model,

c. Burnable absorber rods in fuel assembly or axial blankets in fuel rod are not considered in the calculation model, and

d. Zoning to alleviate the power peak in the fuel assembly is not considered and all fuel rods are assumed to have the same maximum enrichment.

The acceptance criteria of the spent fuel storage racks with soluble boron credit are as follows:

a. The k_{eff} value, including all biases and uncertainties, must not exceed 0.95 with borated water, at a 95 percent probability, 95 percent confidence level, and

b. The k_{eff} value, including all biases and uncertainties, less than 1.00 with full density unborated water, at a 95 percent probability and 95 percent confidence level.

The k_{eff} for the normal fuel storage cell is 0.92623 without applying soluble boron and the k_{eff} for the damaged fuel storage cells is 0.93655 without applying soluble boron. Therefore, the spent fuel pool region I satisfies criticality safety criteria since the k_{eff} for both normal and damaged fuel storage cells are less than the regulatory limit as follows:

Description	k_{eff}	Acceptance criteria (with soluble boron)	Acceptance criteria (without soluble boron)
K_{eff} for the Normal Fuel Storage Cell	0.92623	≤ 0.95	< 1.00
K_{eff} for the Damaged Fuel Storage Cell	0.93655		

Table 3.4-1 k_{eff} without Bias and Uncertainty for Spent Fuel Pool Region I

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Table 3.4-2 Benchmark Calculation Bias and Bias Uncertainty
as a Function of Enrichment (Reference 7)

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Table 3.4-3 Benchmark Calculation Bias and Bias Uncertainty
Area of Applicability (Reference 7)

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Table 3.4-4 Uncertainty due to Mechanical Tolerances

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Table 3.4-5 Uncertainty due to Fuel Assembly Position in the Cell

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Table 3.4-6 Bias due to Cooling Water Temperature

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Table 3.4-7 Summary of Bias and Uncertainty for Spent Fuel Pool Region I

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Table 3.4-8 Summary of k_{eff} with Bias and Uncertainty for Spent Fuel Pool Region I

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Figure 3.4-1 Reference Model of Spent Fuel Pool Region I

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Figure 3.4-2 Model for Gap Effect of Spent Fuel Pool Region I

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Figure 3.4-3 Model for Damaged Fuel Storage Cells of Spent Fuel Pool Region I

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Figure 3.4-4 Model for Eccentric Position of Fuel Assembly in Spent Fuel Pool Region I

3.5 Criticality Analysis for Spent Fuel Pool Region II

The spent fuel pool region II is designed to accommodate the fuel assemblies with the minimum burnup which satisfies the criticality acceptance criteria. The criticality analysis is performed using the CSAS5/KENO-V.a sequence and the ORIGEN-ARP with cross section libraries generated using the TRITON and the ENDF/B-VII 238 energy group library.

In order to determine the loading curve, the criticality analyses are performed to find the minimum burnup which produced a k_{eff} less than 1.0 at the each initial enrichment of fuel assemblies.

3.5.1 Depletion Calculations

As discussed in Subsection 3.3.2, the depletion calculations are performed using the ORIGEN-ARP with cross section libraries generated using the TRITON sequence. For the generation of the cross section libraries using the TRITON sequence, bounding reactor parameters described in the Subsection 3.5.1.1 are used. The isotopic concentrations are generated by the ORIGEN-ARP at each 2.25 GWd/MTU intervals from 0 to 72 GWd/MTU for the initial enrichments from 2.0 to 5.0 wt% U-235 with 0.5 wt% increments of U-235 enrichment.

No burnup credit is taken for conservatism so that the fuel assembly is not allowed to decay after depleted to a desired assembly-average burnup.

3.5.1.1 Bounding Reactor Parameters for the Depletion Calculation

The bounding reactor parameters are used for the depletion calculation for conservatism. The reactor parameters are a fuel temperature, a fuel density, a moderator temperature, a soluble boron concentration and a power level as presented in Table 3.5-1. The sensitivity analysis is performed to identify the trend with respect to a power level, and it is found that there is no trend with respect to power level as shown in Table 3.5-2. So, the maximum power level corresponding to the maximum fuel temperature is used as a bounding reactor parameter.

By using the bounding reactor parameters for the depletion calculation, there is no need to add the additional uncertainty to the calculated nominal k_{eff} for the reactor operational conditions.

3.5.1.2 ORIGEN-ARP Calculation

As discussed in Subsection 3.3.2.1, the ORIGEN-ARP code is used for the depletion calculation.

3.5.1.3 TRITON-NEWT Calculation

The TRITON sequence is used to generate libraries for the PLUS7 16x16 fuel assembly. Figure 3.5-1 shows the depletion calculation model for the PLUS7 16x16 fuel assembly. The burnup steps of 3 GWd/MTU are generally adequate to represent the cross section variations with burnup in creating LWR fuel libraries (Reference 5). But, burnup steps of 2.25 GWd/MTU are used in this calculation for better accuracy. In this calculation, the maximum burnup is 72 GWd/MTU. 32 burnup steps are used with intervals of 2.25 GWd/MTU, and one library is generated for each steps. The library generated by this analysis contains 33 sets of cross sections, which are fresh fuel cross sections and 32 burnup-dependent cross sections. Summary of parameters for the depletion calculation are listed in Table 3.5-3. The burnup values corresponding to each set are listed in Table 3.5-4.

3.5.2 Criticality Calculations

The KENO-V.a code with 238 multi-group library based on ENDF/B-VII is used for the criticality calculation. The criticality analysis model for the spent fuel pool region II is modeled as an infinite 2x2 array of the spent fuel storage cells as shown in Figures 3.5-2 (2D) and 3.5-3 (3D). The design data of the spent fuel pool region II and the fuel assembly are presented in Tables 3.1-2 and 3.1-3, respectably. The k_{eff} without bias and uncertainty and Monte Carlo standard deviations for k_{eff} calculation are summarized in Tables 3.5-5 and 3.5-6, respectively.

3.5.3 Bias and Uncertainty Calculations

The bias and uncertainty related to the criticality calculations are as follows:

a. Bias and bias uncertainty due to methodology,
b. Uncertainty due to Monte Carlo calculation,
c. Uncertainty due to mechanical tolerances,
d. Bias due to the credited minor actinides and fission products,
e. Bias due to Pool cooling water temperature, and
f. Uncertainty due to eccentric fuel assembly positioning.

And the bias and uncertainty related to the depletion calculations are as follows:

a. Burnup measurement uncertainty,
b. Depletion uncertainty, and
c. Bias due to the axial power distribution.

The analysis results of the bias and uncertainty calculations are shown in the following Subsections.

3.5.3.1 Bias and Bias Uncertainty due to Methodology

The bias and bias uncertainty of the criticality calculation method are evaluated to validate the criticality analysis methodology through the benchmark calculations based on the criticality experiments (Reference 7).

Two sets of benchmark cases are analyzed to perform trend analysis and to generate bias and bias uncertainties.

a. Fresh fuel with absorbers for region I, and
b. Fresh and depleted fuel (HTC) with absorbers for region II.

In both sets, the only statistically significant trend observed is related to enrichment. Bias and bias uncertainty due to the first set (fresh fuel only) is slightly higher than that due to the second set (with HTC). Therefore, the first set of bias and bias uncertainty is used for both region I and region II calculations. Table 3.4-2 shows the bias and bias uncertainty as a function of enrichment and Table 3.4-3 shows the area of applicability for the bias and bias uncertainty.

3.5.3.2 Uncertainty due to Monte Carlo Calculation

The statistical uncertainties due to the Monte Carlo calculation are presented in Table 3.5-6.

3.5.3.3 Uncertainty due to Mechanical Tolerances

To evaluate uncertainties due to tolerances in the mechanical and material specifications of the fuel and rack structures, sensitivity analyses are performed with various parameters as shown in Table 3.5-7.

The uncertainty due to mechanical tolerances of the fuel assembly and the rack is summarized in Table 3.5-8. And the detailed assessments are described in the following Subsections.

3.5.3.3.1 Fuel Assembly

The uncertainties due to mechanical tolerances for the fuel assembly including a fuel pellet enrichment, a fuel pellet diameter, a fuel cladding diameter, a fuel rod pitch, and a guide tube cladding diameter are evaluated. A bounding fuel pellet stack density is used in both the depletion calculations and the criticality analyses so that no tolerance calculation for density is needed.

Items in the sensitivity analysis for the criticality uncertainty evaluation are summarized as follows:

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3.5.3.3.2 Rack

The uncertainties due to mechanical tolerances for the rack including a cell pitch, a cell wall thickness, and a sheath thickness are evaluated. As discussed in Subsection 3.4.3.3.2, the bounding values are used for design parameters of the neutron absorber plate in the criticality calculation. So, the tolerance effects for these parameters are not necessary.

Items in the sensitivity analysis for the criticality uncertainty evaluation are summarized as follows:

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3.5.3.4 Bias for Minor Actinide and Fission Product

In order to analyze the bias for minor actinide and fission product, the sensitivity analysis is performed to assess the worth of the minor actinides and fission products. The reactivity differences are the worth of the minor actinides and fission products as summarized in Table 3.5-9. Table 3.5-9 shows that the credited minor actinide and fission product worth is no greater than 0.1 in k_{eff}. Although the worth ([ ]^{TS}) of 5.0 wt% and 51.75 GWd/MTU is slightly over the limit, the excess worth is negligible.

As discussed in the NUREG/CR-7109 (Reference 9), one point five percent (1.5 %) of the worth of the minor actinides and fission products conservatively covers the bias due to these isotopes under the following rage of applicability:

a. Low enriched fuel (< 5.0 wt% U-235) with ENDF/B-VII cross section library,

b. Maximum burnup is 70 GWd/MTU, and

c. Total minor actinide and fission product nuclide worth does not exceed 0.1 in k_{eff}.

So, [ ]^{TS} is used as the bias for the minor actinides and fission products.

3.5.3.5 Bias due to Pool Cooling Water Temperature

The bias due to the pool cooling water temperature is assessed. The sensitivity analyses in the ranges of water density from 0.962 g/cm^3 to 1.0 g/cm^3 are performed to evaluate the effect of pool cooling water temperature range of the spent fuel pool with water temperature from 4 ^{o}C to 95 ^{o}C.

The effective neutron multiplication factors of the disturbed models are less than those of reference model as shown in Table 3.5-10. So the bias due to cooling water density is not included in the total bias.

3.5.3.6 Uncertainty due to Eccentric Fuel Assembly Positioning

The fuel assembly is assumed to be located in the center position of the cell for the reference model. But the fuel assembly could be located eccentrically in the cell. The uncertainty due to fuel assembly positioning in the rack cell is assessed. Figure 3.5-4 shows the model of the eccentric positioning of the fuel assembly. The fresh fuel has the highest reactivity at each enrichment value compared to the burned fuel, so the fresh fuel is used to evaluate the uncertainty due to the eccentric fuel assembly positioning. The multiplication factor for the fuel assembly positioning model is less than that of the reference model as shown in Table 3.5-11. So the uncertainty of fuel assembly positioning is not included in the total uncertainty.

3.5.3.7 Uncertainty due to Burnup Measurement

The burnup measurement uncertainty is calculated by the reactivity difference due to the 5 % change in burnup based on the NUREG/CR-6998 (Reference 10). The uncertainty due to a burnup measurement is shown in Table 3.5-12.

3.5.3.8 Uncertainty due to Depletion

The depletion uncertainty is taken to be 5 % of the reactivity difference between the reactivity at the fresh fuel condition and the reactivity at the burned fuel condition of interest (Reference 11). The summary of depletion uncertainties by the enrichment is shown in Table 3.5-13.

3.5.3.9 Bias due to Axial Power Distribution

The reactivity difference between the effective neutron multiplication factors (k_{eff}) calculated with explicit representation of the axial burnup distribution and k_{eff} calculated assuming a uniform axial burnup distribution is referred to as the "end effect". The end effect is shown to be dependent on many factors, including the axial burnup profile, a total accumulated burnup, a cooling time, an initial enrichment, an assembly design, and the isotopic concentrations.

In this calculation, the fuel assembly type is only the PLUS7 16x16 fuel assembly, assumed as non-blanketed fuel, 28 actinides and fission products are considered, and no decay time is credited. And calculations are performed with and without a bounding axial burnup profile, to assess the magnitude of the end effect which is applied as a bias.

3.5.3.9.1 Selection of Bounding Axial Burnup Profile

For the axial power distribution, the uniform axial burnup is assumed and the end effect is considered as a bias. To quantify the end effect as a function of enrichment and burnup, a bounding axial burnup profile is selected by surveying 304 burnup profiles which cover all possible types of axial burnup distributions. The significantly under-burned top nodes of the fuel assemblies are the most important to quantify the end effect. So the axial burnup profile having the smallest burnup of the top fuel region is chosen as a bounding axial burnup profile among the profiles. And then the original 26 non-uniform heights (nodes) are modified into the 18 non-uniform heights (nodes) by merging flat burnup regions in the middle of fuel rod as shown in Table 3.5-14.

3.5.3.9.2 Modeling of Axial Burnup Distribution

The axial burnup profile has 18 nodes having different local burnups. The local powers for each node are assumed by multiplying a normalized burnup distribution by the assembly-averaged power as shown in Table 3.5-14.

3.5.3.9.3 End Effect of Non-Blanketed Fuel

The fuel assemblies are assumed as non-blanketed fuel for conservatism. The k_{eff} calculated with explicit representation of the axial burnup distribution for the non-blanketed fuel is shown in Table 3.5-15, and k_{eff} calculated assuming flat axial burnup distribution for the non-blanketed fuel is shown in Table 3.5-16. And the reactivity difference between them is shown in Table 3.5-17, which is the end effect of non-blanketed fuel. Only positive end effects are applied as bias.

3.5.3.9.4 End Effect of Blanketed Fuel

The PLUS7 16x16 fuel assembly has blankets (6 inches long 2 wt% U-235 pellets) at the top and bottom end of the fuel rod. Calculations are performed to assess the magnitude of the blanket effect which is considered as an additional margin.

The k_{eff} calculated for a blanketed fuel with explicit representation of the axial burnup distribution is shown in Table 3.5-18, and k_{eff} calculated for a blanketed fuel assuming flat distribution is shown in Table 3.5-19. And the reactivity difference between them is shown in Table 3.5-20 which is the end effect of blanketed fuel. By comparing the end effect without blanket (Table 3.5-17) and with blanket (Table 3.5-20) shows that assuming fuel as a non-blanketed fuel gives margin up to 0.03 in k_{eff}.

3.5.3.10 k_{eff} with Bias and Uncertainty

All biases are directly added to determine the total bias. Total bias is the sum of all the biases due to the methodology, a minor actinide and fission product, and an axial power distribution. The total bias is summarized in Table 3.5-21.

All uncertainty values are statistically combined (the square root of the sum of the squares) to determine the total uncertainty. The uncertainties are due to the methodology, a Monte Carlo calculation, a mechanical tolerance, a burnup measurement, and a depletion. The total uncertainty is summarized in Table 3.5-22. Total bias and uncertainty and the k_{eff} with bias and uncertainty are summarized in Tables 3.5-23 and 3.5-24, respectively.

3.5.4 Calculation of Minimum Burnup versus Enrichment Curve

The minimum burnup versus enrichment curve is based on the k_{eff} with bias and uncertainty in Table 3.5-24. The k_{eff} with bias and uncertainty is represented as graph in Figure 3.5-5 which shows the linear fitting equations for each enrichment value. Table 3.5-25 shows the minimum burnup for target k_{eff} (1.00) calculated by the fitting equations in Figure 3.5-5 for each enrichment value. And Figure 3.5-6 shows the minimum burnup versus enrichment curve based on Table 3.5-25. The 3rd degree polynomial is used to generate the fitting equation. Then, for conservatism the y-interception of the fitting equation is adjusted to be 99% of the raw value. Table 3.5-26 shows the adjustment result and Figure 3.5-6 shows the adjusted fitting equation.

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Figure 3.5-7 shows the final minimum burnup versus enrichment curve for spent fuel pool region II.

Table 3.5-1 Bounding Value of Reactor Parameters for the Depletion Calculation

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Table 3.5-2 Sensitivity Analysis for Power Level

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Table 3.5-3 Parameters for Generated ORIGEN-ARP Libraries of PLUS7 16x16 Fuel Assembly

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Table 3.5-4 Burnup Values of Cross Section Sets

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Table 3.5-5 k_{eff} without Bias and Uncertainty for Spent Fuel Pool Region II

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Table 3.5-6 Monte Carlo Standard Deviation for k_{eff} Calculation

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Table 3.5-7 Mechanical Tolerances

TS

Table 3.5-8 Uncertainty due to Mechanical Tolerance

TS

Table 3.5-9 Bias for Minor Actinides and Fission Products

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Table 3.5-10 Bias for Pool Cooling Water Temperature

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Table 3.5-11 Uncertainty due to Eccentric Fuel Assembly Positioning

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Table 3.5-12 Uncertainty due to Burnup Measurement

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Table 3.5-13 Depletion Uncertainty

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Table 3.5-14 Bounding Axial Burnup Profile

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Table 3.5-15 k_{eff} Calculated with Bounding Axial Burnup Distribution (Non-Blanketed)

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Table 3.5-16 k_{eff} Calculated with Flat Burnup Distribution (Non-Blanketed)

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Table 3.5-17 Reactivity difference between Flat and Axial Burnup Distribution (Non-Blanketed)

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Table 3.5-18 k_{eff} Calculated with Bounding Axial Burnup Distribution (Blanketed)

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Table 3.5-19 k_{eff} Calculated with Flat Burnup Distribution (Blanketed)

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Table 3.5-20 Reactivity difference between Flat and Axial Burnup Distribution (Blanketed)

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Table 3.5-21 Total Bias for Spent Fuel Pool Region II

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Table 3.5-22 Total Uncertainty for Spent Fuel Pool Region II

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Table 3.5-23 Total Bias and Uncertainty for Spent Fuel Pool Region II

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Table 3.5-24 k_{eff} with Bias and Uncertainty for Spent Fuel Pool Region II

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Table 3.5-25 Minimum Burnup Calculated with Raw Fitting Equation

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Table 3.5-26 Minimum Burnup versus Enrichment for Raw Fitting and Adjusted Fitting

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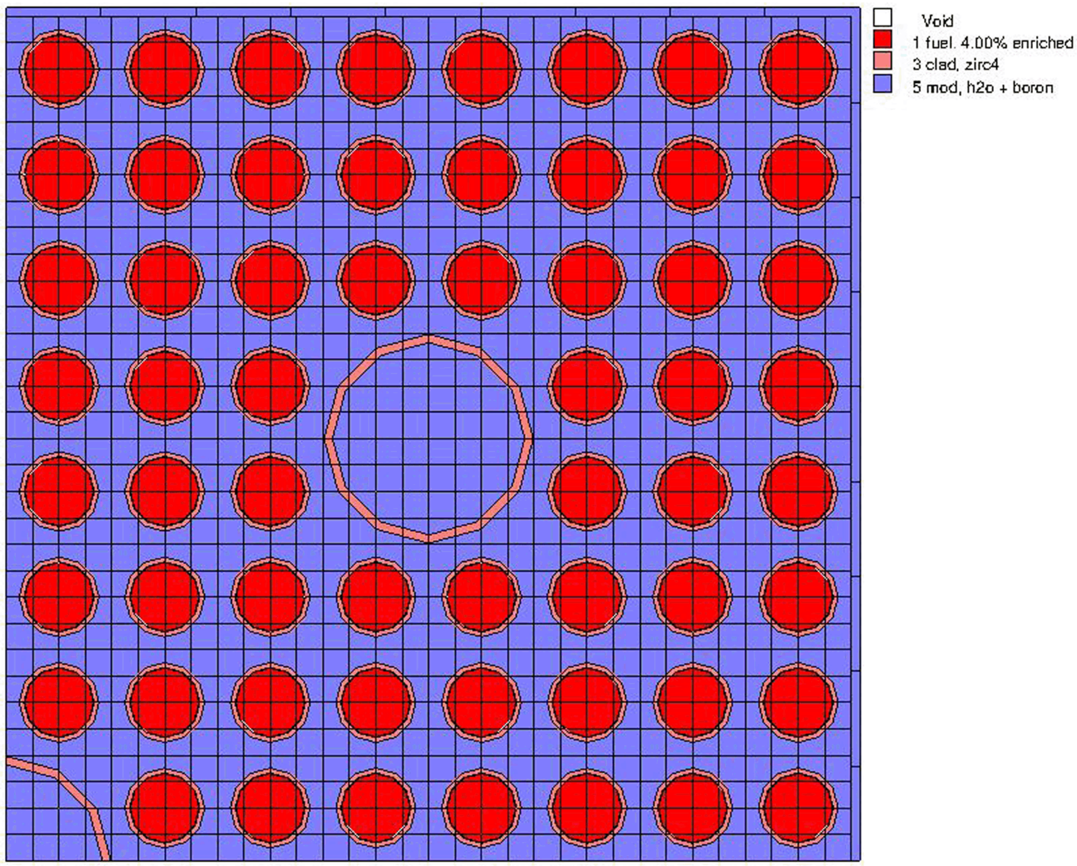
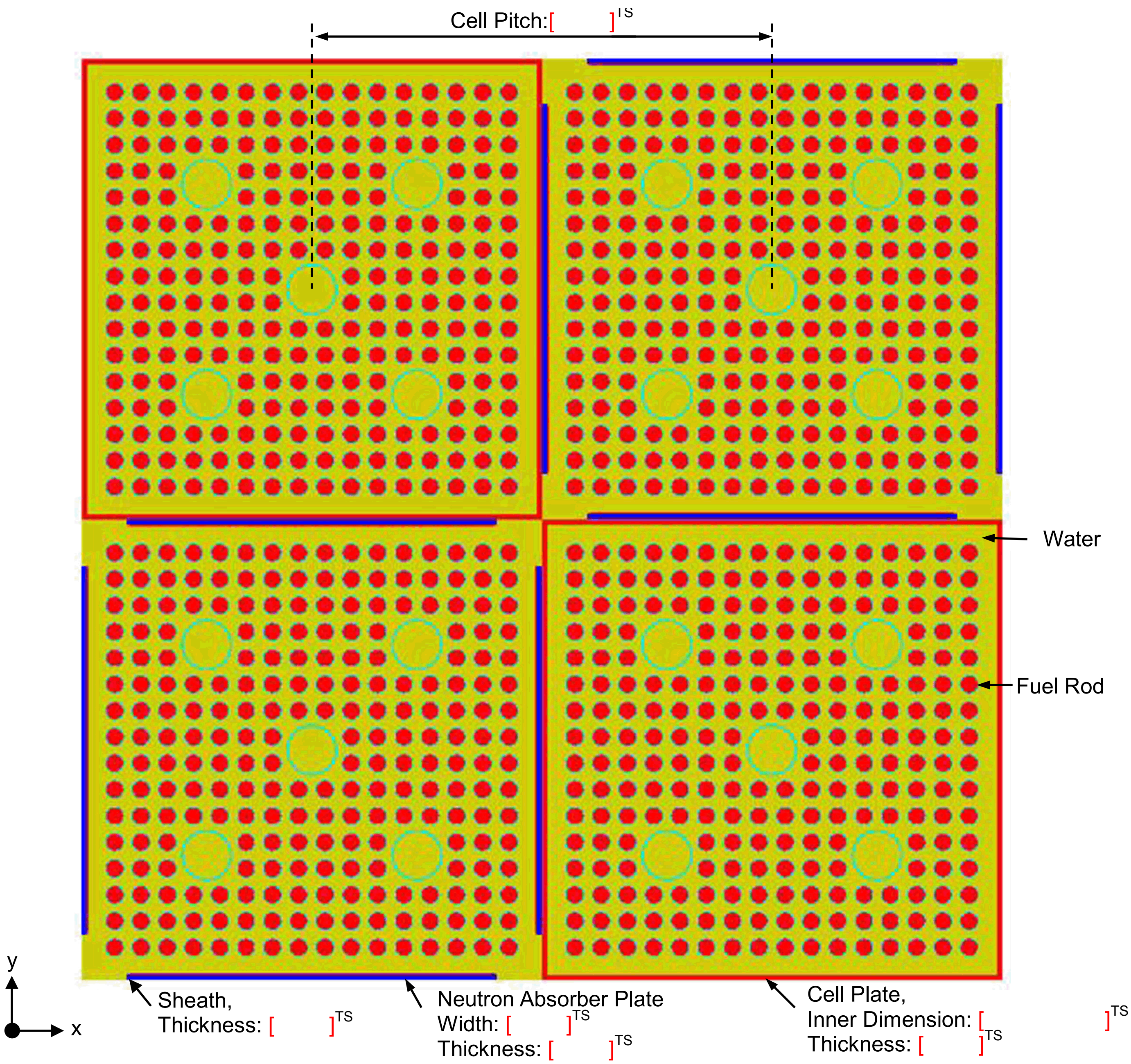


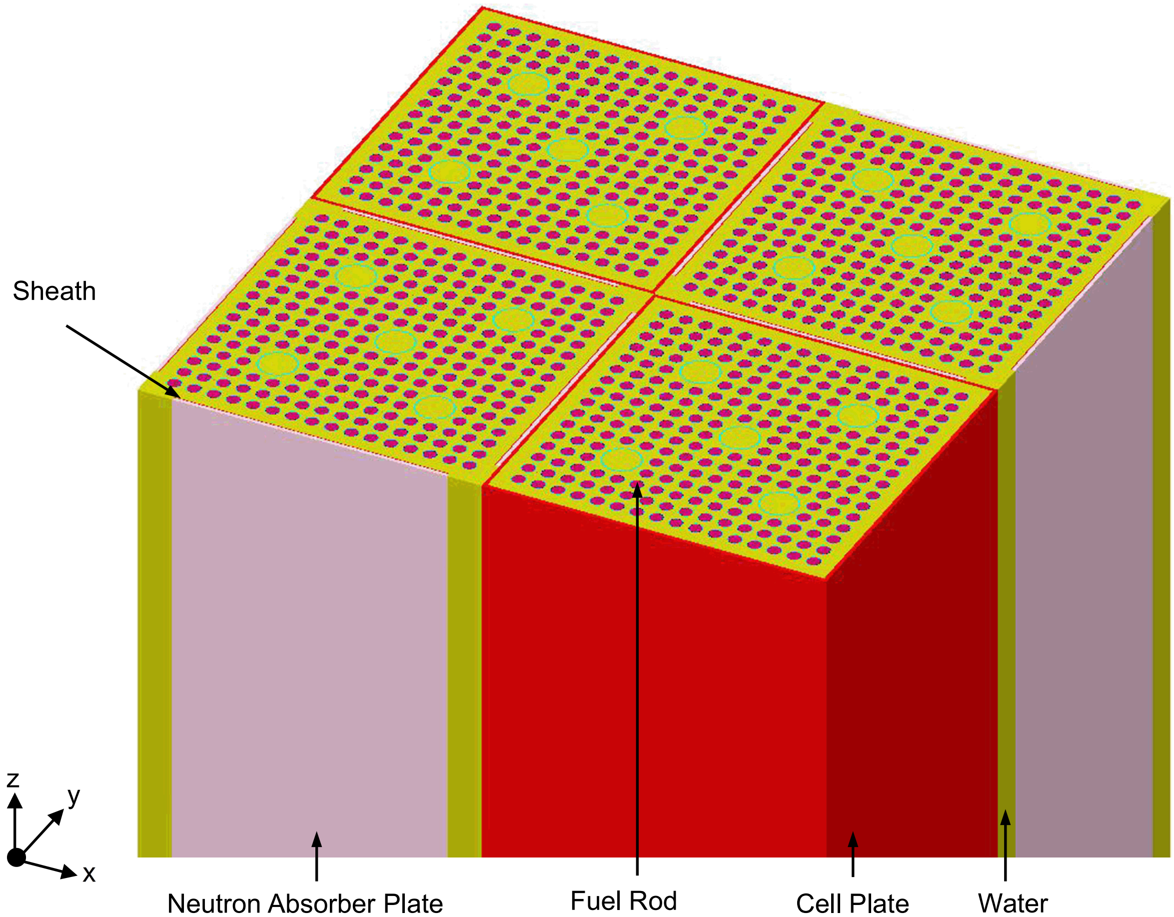
Note:
Reflective boundary conditions are applied on all sides.

Figure 3.5-1 Symmetric Depletion Calculation Model for the PLUS7 16x16 Fuel Assembly



Note:
x-y axis: periodic boundary conditions, z axis: reflective boundary conditions.

Figure 3.5-2 Cross Section View of KENO-V.a Model for 2x2 Array of Fuel Rack in Region II



Notes:
x-y axis: periodic boundary conditions, z axis: reflective boundary conditions.

Figure 3.5-3 3-D View of KENO-V.a Model for 2x2 Array of Fuel Rack in Region II

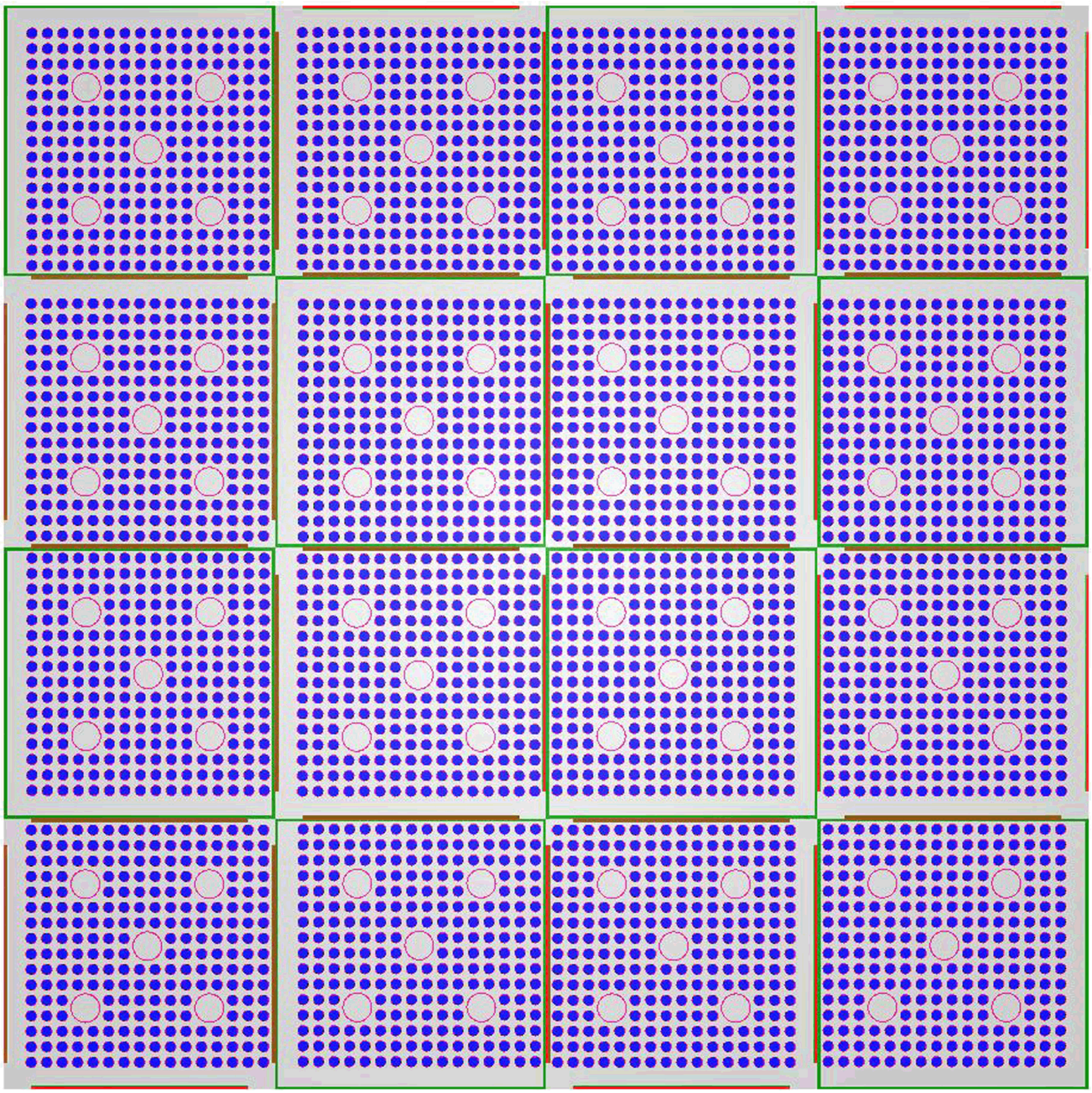


Figure 3.5-4 Model for Eccentric Position of Fuel Assemblies in Spent Fuel Pool Region II

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Figure 3.5-5 Burnup versus k_{eff} with Fitting Equations

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Figure 3.5-6 Minimum Burnup versus Enrichment with Raw Fitting and Adjusted Fitting

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Figure 3.5-7 Minimum Burnup versus Enrichment Curve

3.6 Interface Between Regions

3.6.1 Interface within Each Region

Interface within region I is less reactive than the reference case for region I, because the gap between the racks in region I is 60 cm, but the gap between the cells in region I (the reference case) is 43.8 cm.

Interface within region II is also less reactive than the reference case for region II, because the gap between the racks in region II is 30 cm, but the gap between the cells in region II (the reference case) is 0 cm.

3.6.2 Interface between Regions I and II

Interface between region I and region II is also predictable due to the same reason in Subsection 3.6.1, because the gap between region I and region II is at least 60 cm. Furthermore, racks in region I have neutron absorber panel on the exterior of the rack, so there is no local increase in reactivity at the rack interface. There are sufficient neutron absorber panels among racks so that the maximum k_{eff} is much less than the limiting k_{eff} in region I or region II.

4 ACCIDENT ANALYSIS

The following postulate accidents are considered in following Subsections:

a. Dropped fresh fuel assembly,

b. Misloaded fresh fuel assembly into incorrect storage rack location, and

c. Boron dilution accident.

4.1. Dropped Fresh Fuel Assembly

During the placement of the fuel assemblies in the spent fuel storage rack, it is possible to drop the fuel assembly between concrete wall and racks. This postulate accident is analyzed under the most severe conditions such that the dropped fuel assembly lands just beside outer-most storage cell of region I and all storage cell are occupied by fresh fuel assemblies.

Figure 4.1-1 shows the accident analysis model of a dropped fresh fuel assembly. As shown in the figure, the analysis model consists of a dropped fresh fuel assembly, a concrete wall, fuel storage cells and fuel assemblies stored in the cells. Instead of modeling whole storage cells in the region I, 1x17 arrays of storage cells with reflective boundary condition are considered. Enrichment of a dropped fuel assembly and stored fuel assemblies is assumed as 5.0 wt%. It is assumed that the soluble boron concentration in the pool water is 2,150 ppm, which is the minimum boron concentration specified in technical specification LCO 3.7.15. Additional analyses are performed to find the minimum boron concentration which is sufficient to assure the regulatory limit (k_{eff} of 0.95).

The criticality analysis results of a dropped fuel accident are summarized in Table 4.1-1. Under the minimum boron concentration, 2,150 ppm, k_{eff} with bias and uncertainty is 0.79716, much smaller than regulatory limit (k_{eff} of 0.95). It is shown that the k_{eff} reaches to 0.95 when boron concentration decreases to 698.45 ppm as demonstrated in Figure 4.1-2.

The criticality analysis information of dropped fresh fuel assembly is summarized as follows:

a. Enrichment of dropped fuel and stored fuel: 5.0 wt%,

b. Distance between inner concrete wall and outer-most storage cell: 840 mm,

c. Distance between dropped fuel assembly and stored fuel assembly: 12.7808 mm,

d. Soluble boron concentration: 2,150 ppm,

e. Thickness of concrete wall: 300 mm,

f. Boundary conditions:

+X, +Y, -Y, +Z and -Z directions: Reflective boundary condition,

-X direction: Vacuum boundary condition,

g. Design data of storage cell of SFP region I and fuel assemblies are presented in Tables 3.1-1 and 3.1-3, respectively, and

h. Bias and uncertainty discussed in Subsection 3.4.3 are applied to the critical analysis results.

Table 4.1-1 Criticality Analysis Results for Dropped Fuel Assembly Accident

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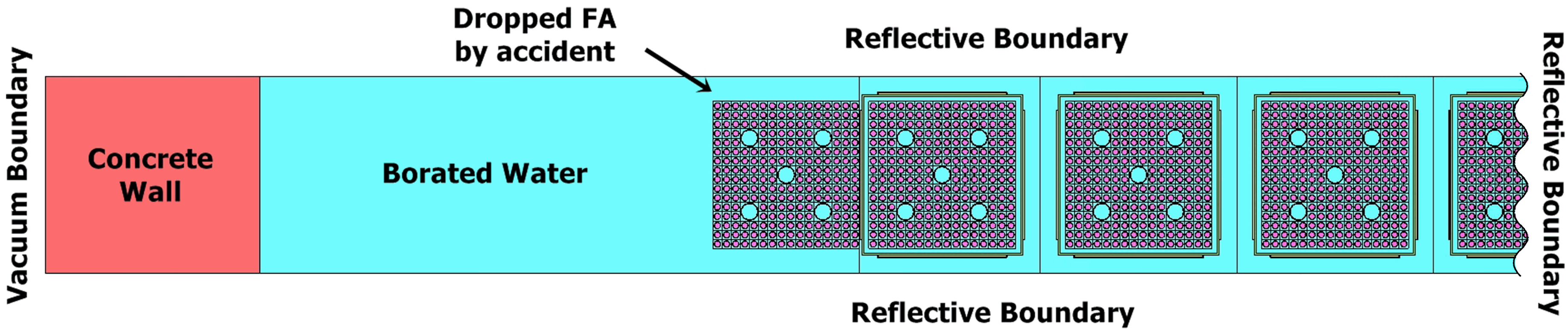

Figure 4.1-1 Accident Analysis Model for the Dropped Fresh Fuel Assembly

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Figure 4.1-2 k_{eff} versus Boron Concentration Curves for Dropped Fuel Accident and Misloaded Fresh Fuel Accident

4.2. Misloaded Fresh Fuel Assembly

The misloaded fresh fuel assembly accident is the case that a fresh fuel assembly is placed into region II storage cell intended for spent fuel assemblies. The most severe case is that misloaded fresh fuel assembly is being surrounded by the most reactive fuel allowed in region II.

As illustrated in Figure 4.2-1, the analysis model for misloaded fresh fuel assembly accident consists of 2x2 arrays of storage cell with periodic boundary conditions on all four sides. The 2x2 arrays of storage cell are occupied by a misloaded fresh fuel assembly and three spent fuel assemblies. The enrichment of fresh fuel assembly is assumed as 5.0 wt% and the burnup and the initial enrichment of the spent fuel stored in the region II are assumed 33.75 GWd/MTU and 4.5 wt%, respectively. The nuclide densities of spent fuel assembly are obtained from the depletion calculation and presented in Table 4.2-1. This assumption is conservative because the acceptable minimum burnup for initial enrichment of 4.5wt% fuel is 38.5 GWd/MTU as discussed in Subsection 3.5.4. The soluble boron concentration is assumed as 2,150 ppm, which is the same as Subsection 4.1. Additional analyses are performed to find the minimum boron concentration which is sufficient to assure the regulatory limit (k_{eff} of 0.95).

The criticality analysis results of a misloaded fresh fuel accident are summarized in Table 4.2-2. Under the boron concentration of 2,150 ppm, k_{eff} with bias and uncertainty is 0.86938, much smaller than regulatory limit as dropped fuel accident case. It is shown that the k_{eff} reaches to regulatory limit when boron concentration decreases to 1,224.46 ppm as demonstrated in Figure 4.1-2.

The criticality analysis information of the misloaded fresh fuel assembly accident is summarized as follows:

a. Enrichment of misloaded fuel: 5.0 wt%,

b. Initial Enrichment of spent fuel stored in the region II: 4.5 wt%,

c. Burnup of spent fuel stored in the region II: 33.75 GWd/MTU,

d. Soluble boron concentration: 2,150 ppm,

e. Thickness of concrete wall: 300mm,

f. Boundary conditions:

   X, Y axis: Periodic boundary condition,

   Z axis: Reflective boundary condition,

g. Design data of storage cell of region II is presented in Table 3.1-2, and

h. Bias and uncertainty discussed in Subsection 3.5.3 are applied to the critical analysis results.

Table 4.2-1 Number Densities of Nuclide in the Spent Fuel
(Initial enrichment: 4.5 wt%, Burnup: 33.75 GWd/MTU)

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Table 4.2-2 Criticality Analysis Results for Misloaded Fuel Assembly Accident

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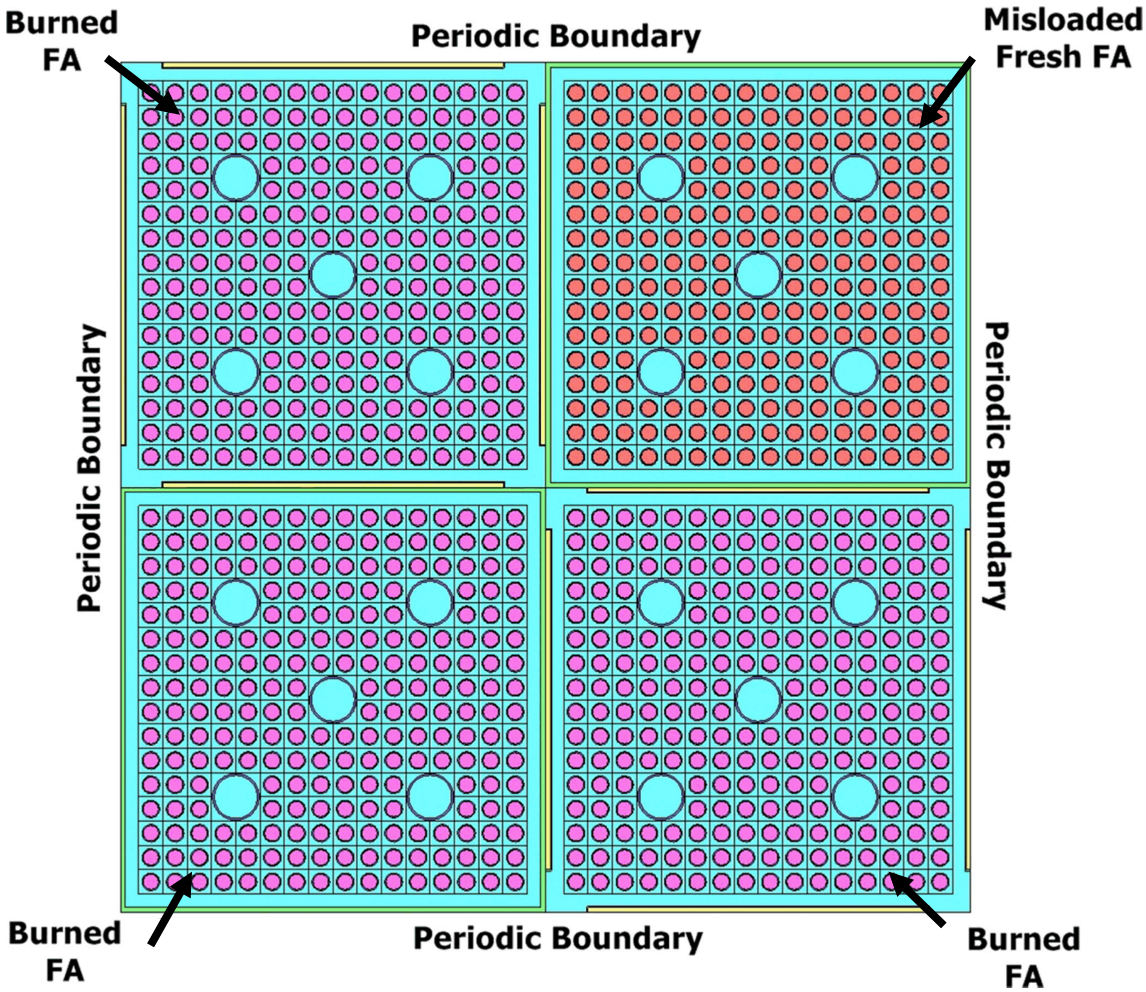

Figure 4.2-1 Accident Analysis Model for Misloaded Fresh Fuel Assembly

4.3. Boron Dilution Accident

4.3.1 Minimum Soluble Boron Concentration

The soluble boron concentration is the important factor for the critical safety since it is utilized to control the reactivity of spent fuel in the pool. Therefore, a boron dilution accident is one of the most severe accidents in the view of criticality safety.

The analysis model of a boron dilution accident for region I is illustrated in Figure 4.3-1. The model consists of 2x2 arrays of region I storage cell with reflective boundary conditions. The 2x2 arrays of storage cell are occupied by the fresh fuel assembly and its initial enrichment is assumed as 5.0 wt% for conservatism.

Figure 4.3-2 shows the analysis model of a boron dilution accident for region II. The model consists of 2x2 arrays of region II storage cell with periodic boundary conditions. The 2x2 arrays of storage cell are occupied by spent fuel assemblies. The burnup and the initial enrichment of the spent fuel stored in the region II are assumed as 38.25 GWd/MTU and 4.5 wt%, respectively. The burnup of 38.25 GWd/MTU is conservative assumption since the acceptable minimum burnup for initial enrichment of 4.5 wt% fuel is 38.5 GWd/MTU. The nuclide number densities of spent fuel are obtained from the depletion calculation and presented in Table 4.3-1.

The criticality analysis results of boron dilution accident for region I and region II are summarized in Tables 4.3-2 and 4.3-3, respectively. As demonstrated in Figure 4.3-3, k_{eff} of region I doesn't exceed regulatory limit even though boron concentration is 0 ppm. In case of region II, the minimum boron concentration to assure the regulatory limit is 422.92 ppm as shown in Figure 4.3-3.

The criticality analysis information of boron dilution accident is summarized as follows:

For region I

a. Enrichment of fresh fuel: 5.0 wt%,
b. Boundary conditions:
   All axis: Reflective boundary condition,
c. Design data of storage cell of region I and fuel assemblies are presented in Tables 3.1-1 and 3.1-3, respectively, and
d. Bias and uncertainty discussed in Subsection 3.4.3 are applied to the critical analysis results.

For region II

a. Initial Enrichment of spent fuel stored in the region II: 4.5 wt%,
b. Burnup of spent fuel stored in the region II: 38.25 GWd/MTU,
c. Boundary conditions:
   X, Y axis: Periodic boundary condition,
   Z axis: Reflective boundary condition,
d. Design data of storage cell of region II is presented in Table 3.1-2, and
e. Bias and uncertainty discussed in Subsection 3.5.3 are applied to the critical analysis results.

4.3.2 Evaluation of Boron Dilution Accidents in the Spent Fuel Pool

This Subsection provides analyses of potential boron dilution accidents if credit for soluble boron is taken for demonstrating spent fuel storage rack subcriticality for the APR1400 spent fuel pool (SFP) design.

There are various systems within the SFP vicinity which contain unborated water and under accident conditions could potentially result in some degree of boron dilution for the SFP. Based on the systems and the associated maximum unborated water flow rates of such postulated unborated water addition, the amount of time takes for the postulated maximum flows considered to dilute the boron concentration to the prescribed limit of 423 ppm can be calculated. This calculation utilizes the following boron dilution equation.

$$C(t)=C_o\times e^{-\left(\frac{t}{\tau_2}\right)}$$

Where:

$C(t)$ is the boron concentration at time t,

C_o is the initial boron concentration,

τ_2 is V/Q,

V is the control volume (SFP volume), and

Q is the volumetric flow rate of the unborated water.

For this boron dilution analysis, the following input data are utilized regarding the SFP:

$C(t)$ = 423 ppm (Minimum Soluble Boron Concentration (accident)),

C_o = 2,150 ppm (Boron Concentration specified in technical specification LCO 3.7.15), and

V = 429,407 gallons (SFP Volume at a level specified in technical specification LCO 3.7.14)

For the conservative calculation of the boron dilution time in SFP, instead of utilizing the normal (operating) concentration of boron in the SFP of 4,000 to 4,400 ppm and the normal operation volume of SFP of 446,138 gallons, the boron concentration and the SFP volume listed above are utilized.

Utilizing the equation, the SFP input data, the above conservative assumptions, and the maximum unborated water flow rates, the time required to dilute the SFP from a boron concentration of 2,150 ppm to a boron concentration of 423 ppm is calculated. Additionally, utilizing the volumetric flow rate of unborated water and the SFP volume at the high level alarm set point, the time to SFP high level alarm and the required time values for boron dilution to 423 ppm after SFP high level alarm are also calculated. In this evaluation, the SFP volume at the high level alarm set point is 447,996 gallons. The results of these calculations are provided in Table 4.3-4.

As a result of this evaluation, it is concluded that an event which would result in the dilution of the SFP boron concentration from 2,150 ppm to 423 ppm is not a credible event. This conclusion is supported by all of the followings.

- In order to dilute the SFP from a boron concentration of 2,150 ppm to 423 ppm resulting in a k_{eff} of 0.95, a substantial amount of water (greater than 698,152 gallons) is needed. Since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via high level alarms, or by normal operator rounds through the SFP area.

- The requirement of the minimum soluble boron concentration to assure the k_{eff} is less than 0.95 was set to 423 ppm. This is far less than the normal operating conditions of 4,000 ppm. In the case of a boron dilution event, the calculated dilution times in Table 4.3-4 are long enough to allow corrective actions to be made and to disrupt the dilution event.

- The existence of high level alarms in the main control room would be readily detected by plant personnel. As provided in Table 4.3-4, the sufficient time after SFP high level alarm is available to respond to a dilution event.

From the evaluation of boron dilution accidents in the spent fuel pool, it is confirmed that the design criteria 10 CFR 50.68 are met and that subcriticality is maintained.

Table 4.3-1 Number Densities of Nuclide in the Spent Fuel
(Initial enrichment: 4.5 wt%, Burnup: 38.25 GWd/MTU)

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Table 4.3-2 Analysis Results of Boron Dilution Accident in the Region I

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Table 4.3-3 Analysis Results of Boron Dilution Accident in the Region II

TS

Table 4.3-4 Critical Time Values for Boron Dilution from 2,150 ppm to 423 ppm within the SFP

TS

Table 4.3-4 Critical Time Values for Boron Dilution from 2,150 ppm to 423 ppm within the SFP (Cont.)

TS

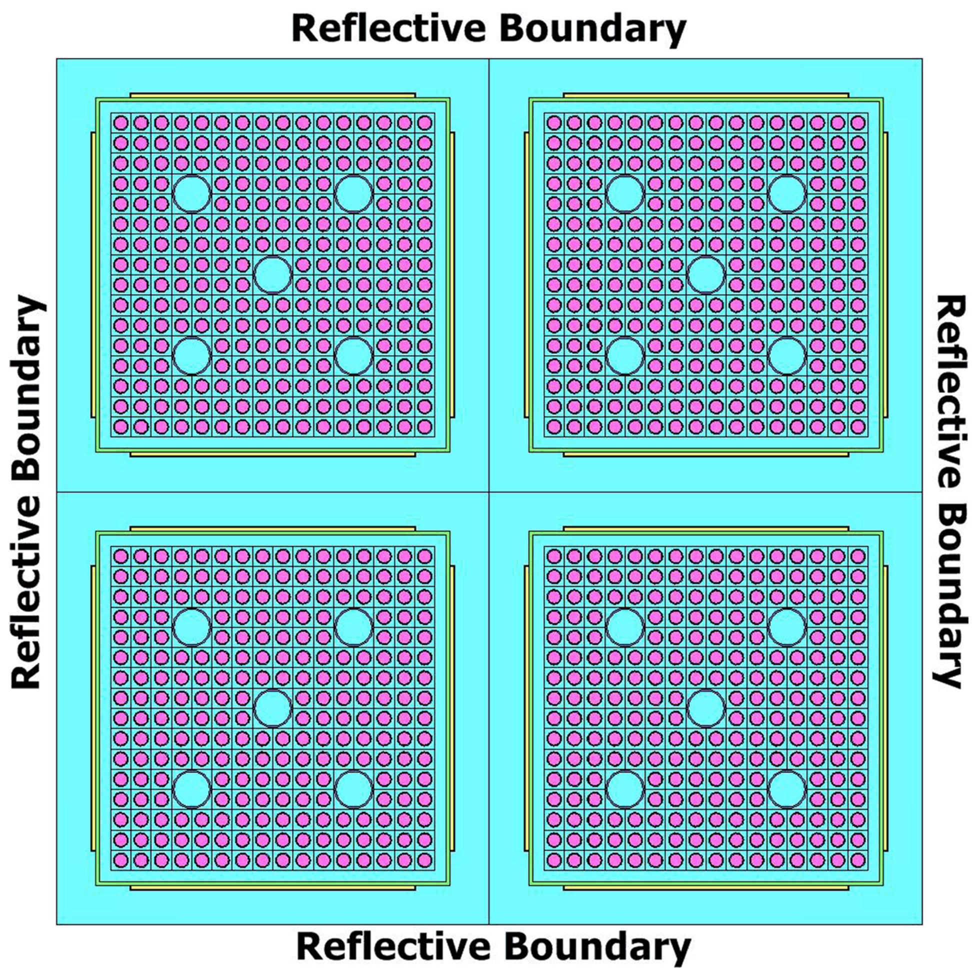

Figure 4.3-1 Analysis Model of Boron Dilution Accident in the Region I

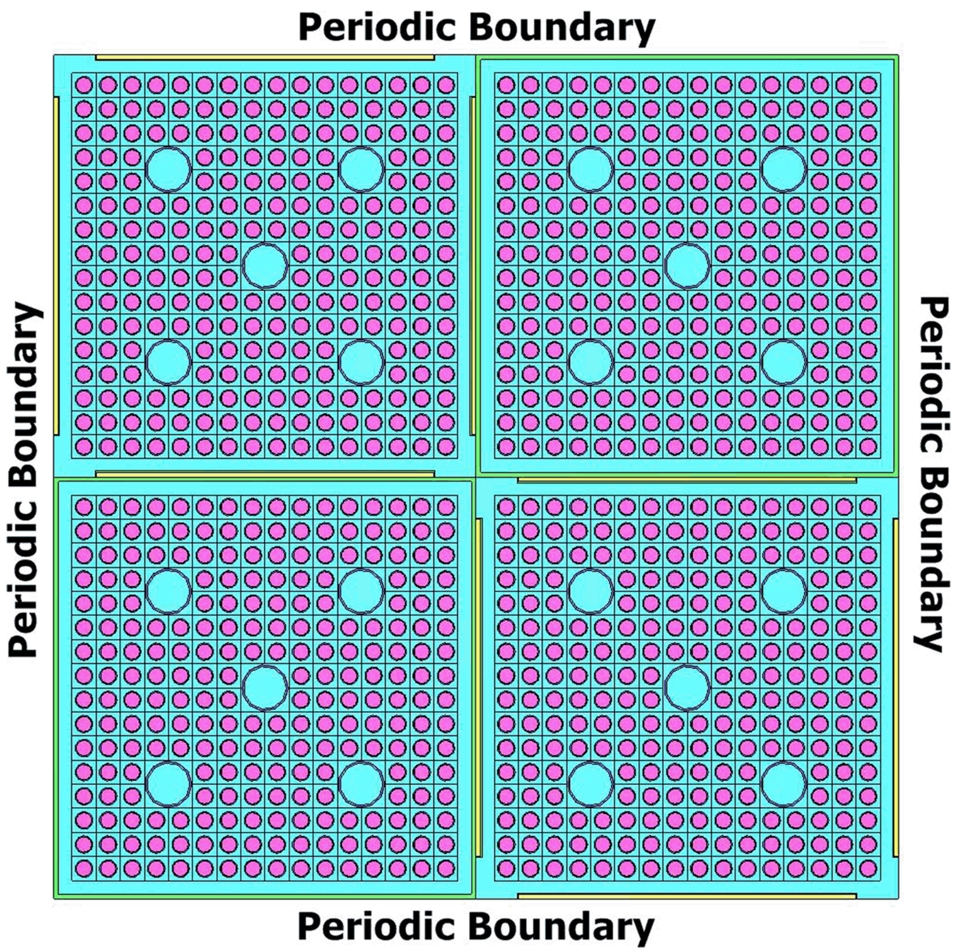

Figure 4.3-2 Analysis Model of Boron Dilution Accident in the Region II

TS

Figure 4.3-3 k_{eff} versus Boron Concentration Curves for Boron Dilution Accident

5 LIMITATIONS OF ANALYSIS

The APR1400 design is an advanced PWR design that is functionally similar to existing plants. The following design input data to this analysis will be checked in order to ensure compliance with the criticality safety design basis.

5.1 Fuel Limitations

1. This analysis is applicable to the PLUS7 16x16 fuel design.

2. The initial stack density shall be less than [ ]^{TS} of the theoretical density of uranium dioxide ([ ]^{TS} g/cm^3).

5.2 Operational Limitations

1. The cycle averaged soluble boron concentration for all fuel assemblies shall be less than [ ]^{TS} ppm.

2. Fuel assemblies that do not meet operational limits and assumptions will be specifically evaluated and classified following the same methodology used in this report.

5.3 Spent Fuel Pool Limitations

1. An areal density of each neutron absorber material (METAMIC^{TM}) shall be greater than or equal to [ ]^{TS} B-10 g/cm^2 for spent fuel pool region I.

2. An areal density of each neutron absorber material (METAMIC^{TM}) shall be greater than or equal to [ ]^{TS} B-10 g/cm^2 for spent fuel pool region II.

3. The center to center spacing of region I shall be greater than or equal to 27.5 cm and the center to center spacing of region II shall be greater than or equal to 22.5 cm.

6 CONCLUSIONS

The effective neutron multiplication factors, k_{eff}, are calculated for the new and spent fuel storage racks of the APR1400 design. The analysis covers both a normal and accident conditions. The proper set of bias and uncertainty is applied for each case in order to assure the conservatism in the analysis.

From the evaluation results described in Chapter 2 (NFR), Chapter 3 (SFR), and Chapter 4 (accident analyses), it is confirmed that the design criteria are met and the subcriticality is maintained in the new and spent fuel storage racks.

7 REFERENCES

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