Criticality Analysis for US-APWR New and Spent Fuel Storage Racks

Non-Proprietary Version

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Revision History

Revision	Page	Description
0	All	Original issued
1	-	Criticality analyses were completely revised by the design change of NFR, SFR and DFR. This is because MHI determined HOLTEC as the US-Supplier.
2	 Page 2-7 Page 2-12 Page 2-13 Page 2-14 Page 2-15 Page 2-22 Page 3-4 Page 3-7 Page 3-9 Page 3-10 Page 3-11 Page 3-12 Page 3-14 Page 2-5 Page 2-16 Page 2-23 Page 3-6 Page 3-12 Page 3-6 Page 3-17 Page 2-3 Page 3-1 Page 3-6 Page 4-1	The following section, tables and figures were revised to incorporate results of the updated criticality analysis that was performed using the revised NFR and SFR structure. Table 2-2 Table 2-5 Table 2-6 Table 2-7 Table 2-8 Figure 2-4 3.3 Table 3-1 Table 3-3 Table 3-4 Table 3-5 Table 3-6 Table 3-2(1/2) The following section, table and figure were revised to incorporate results of the criticality analysis of mislocated fuel assembly in NFR. 2.3.4 Table 2-9 Figure 2-5 The following section and table were revised to incorporate results of the criticality analysis of SFR movement. 3.3.2.4 Table 3-6 The following section and figure were revised to incorporate analysis model of mislocated fuel assembly in SFR. 3.3.2.3 Figure 3-4 The following sections were revised to incorporate justification for not to consider simultaneous occurrence of accidents. 2.2.1.2 3.0 3.3.2.3 4.0

Mitsubishi Heavy Industries, Ltd.
16-5, Konan 2-chome, Minato-ku
Tokyo 108-8215 Japan

Abstract

This report summarizes the criticality analysis of the US-APWR New and Spent Fuel Storage Racks, and provides a discussion of the design method, analysis, and results obtained to establish the margins of safety.

It is confirmed that the results of the analysis satisfy the fuel racks requirements of 10 Code of Federal Regulations (CFR) 50.68.

Table of Contents

List of Tables

List of Figures

List of Acronyms

ANS	American Nuclear Society
ANSI	American National Standards Institute
CFR	Code of Federal Regulations
DFR	Damaged Fuel Storage Racks
GDC	General Design Criteria
ID	Inner Diameter
keff	effective neutron multiplication factor
MHI	Mitsubishi Heavy Industries, LTD.
NFR	New Fuel Storage Racks
NRC	U.S. Nuclear Regulatory Commission
OD	Outer Diameter
SFP	Spent Fuel Pit
SFR	Spent Fuel Storage Racks
SS	Stainless Steel
TD	Theoretical Density
US-APWR	United States - Advanced Pressurized Water Reactor
95/95	95 percent probability, 95 percent confidence level

1.0 INTRODUCTION

This technical report summarizes the criticality analysis for the New Fuel Storage Racks (NFR), the Spent Fuel Storage Racks (SFR) and the Damaged Fuel Storage Racks (DFR), which are the facilities of the United States - Advanced Pressurized Water Reactor (US-APWR) fuel storage system. The fuel assemblies stored in these racks are 17x17 fuel assemblies for the US-APWR (Reference [1]).

Criticality analyses are performed in accordance with the following acceptance criteria and relevant requirements: General Design Criterion (GDC) 62 (Reference [2]), 10 CFR 50.68 (Reference [3]), NRC guide (Reference [4]), ANSI/ANS-8.17-2004 (Reference [5]). Specifically, 10 CFR 50.68 (b) item (2) and (3) for NFR and item (4) for Spent Fuel Pit (SFP) are applied as the criticality safety design criteria, and the analysis results were evaluated referring to ANSI/ANS-8.17-2004.

All racks of the US-APWR are made from stainless steel (SS) boxes. The SFR has Metamic™ neutron absorber panels affixed to the box walls with SS sheathing. All other racks do not use any neutron absorber. The NFR can store up to 180 fuel assemblies, the SFR can store a maximum of 900 fuel assemblies. DFR are located next to the SFR, and can store up to 12 Damaged Fuel Containers.

MHI has selected Metamic™ as the neutron absorber material which has been adopted in recent rack designs in the US and has selected HOLTEC as the US-Supplier. This technical report is based on their analysis (Reference [6]).

The results of criticality analysis for NFR and SFR (including DFR) are described in Chapter 2 and 3 respectively.

1.1 Analysis Code and Validation

For the criticality analyses of NFR and SFR, the continuous-energy Monte Carlo Code MCNP, version 5.1.40, (Reference [7]) and continuous-energy neutron cross section data ENDF/B-V are used. MHI code validation result was used because of a more conservative combination of a code bias of 0.0029 and bias uncertainty of ± 0.0030 (multiplied by the one-sided tolerance limit coefficient of 1.899 for a 95% probability at the 95% confidence level) (Reference [8]).

2.0 CRITICALITY ANALYSIS OF NFR

Chapter 2 contains the criticality analysis results for US-APWR 17x17 new fuel assemblies stored in NFR. It is shown that the maximum value of effective neutron multiplication factor (keff) at both flooded and optimum moderation conditions including biases and uncertainties, satisfies the design criteria and subcriticality is maintained.

2.1 Design Method

Design criteria, evaluation method and analysis code are described in the following subsections.

2.1.1 Design Criteria

The design criteria are pursuant to 10 CFR 50.68 (b) item (2) and (3) for NFR.

"For new fuel storage racks, the maximum keff value including all biases and uncertainties must be less than or equal to 0.95 for the flooded condition with unborated water, and less than or equal to 0.98 for optimum moderation, at a 95 percent probability, 95 percent confidence level (95/95). Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity."

As noted above, evaluations are conducted for the flooded and optimum moderation conditions.

2.1.2 Evaluation

Under the design criteria mentioned above, evaluations were conducted referring to the equation described in the most recent ANSI/ANS-8.17-2004. More specifically, Section 5 of ANSI/ANS-8.17-2004 states that the calculated multiplication factor k_p shall be equal to or less than an established allowable neutron multiplication factor; i.e.,

$$k_p \leq k_c - \Delta k_p - \Delta k_c - \Delta k_m \tag{1}$$

If the various uncertainties are independent,

$$k_p \leq k_c - \left(\Delta k_p^2 + \Delta k_c^2\right)^{1/2} - \Delta k_m \tag{2}$$

Where
- k_p is the calculated keff
- k_c is the mean keff derived from the code validation
- Δk_p is the allowance for convergence*, tolerances, and modeling limitations
- Δk_c is a bias uncertainty derived from the code validation
- Δk_m is an arbitrary margin to ensure the subcriticality of k_p

(* The 2σ value of MCNP output is applied according to the 95/95 rule.)

In this evaluation, equation (2) is rearranged taking into consideration the following items:

· To compare with the design criteria of keff=1.0, 0.98, 0.95 stated in 10CFR50.68, which consider subcriticality margin, k_c is separated into critical condition keff=1.0 and analysis code bias, and (1-k_c) is moved to the left side of the equation as a symbol to denote a bias.
· The convolution term (root of sums of squares), denoting uncertainty of calculation, tolerance and uncertainty term derived from code validation, are moved to the left side of

the equation.

- Only the term ($1.0-\Delta k_m$) is left in the right side of the equation and keff=1.0, 0.98, 0.95 criteria are applied corresponding to the evaluation.

The rearranged equation becomes as follows.

$$k_p + (1 - k_c) + \left(\Delta k_p^2 + \Delta k_c^2\right)^{1/2} \le \left(1.0 - \Delta k_m\right) \quad (3)$$

Additionally, using the analysis code bias of ($1-k_c$)=0.0029 and bias uncertainty Δk_c=0.0030 multiplied by benchmarking confidence coefficient of 1.899 at 95 percent probability, 95 percent confidence level as stated in Section 4.1.1, equation (3) becomes as follows.

$$k_p + 0.0029 + \left(\Delta k_p^2 + (1.899 \times 0.0030)^2\right)^{1/2} \le \left(1.0 - \Delta k_m\right) \quad (4)$$

Consequently, the evaluation equations for NFR are as follows.

Fully flooded condition : $k_p + 0.0029 + \left(\Delta k_p^2 + (1.899 \times 0.0030)^2\right)^{1/2} \le 0.95$ (5)

Optimum moderation : $k_p + 0.0029 + \left(\Delta k_p^2 + (1.899 \times 0.0030)^2\right)^{1/2} \le 0.98$ (6)

2.1.2.1 Reactivity Uncertainty Due to Tolerances

The reactivity due independent tolerances may be statistically combined. Here, the components of the tolerance to be considered in the criticality analysis are those of the fuel, the rack, and the positioning of the fuel in the rack cells (see Table 2-5). Each of these components of the tolerance is independent.

2.1.3 Analysis Code

As stated in Section 1.1, criticality analysis uses the three-dimensional Monte Carlo code MCNP version 5.1.40 and the continuous-energy neutron data ENDF/B-V.

Additionally, for the S(α, β) thermal scattering data, “lwtr.01t” for hydrogen in light water is applied to water. Though the effect is small, scattering effect as reflector is applied to the hydrogen in floor and wall concretes.

Neutron generation histories in Monte Carlo calculation were set to four million as shown below. In this situation, 1σ is approximately 0.0004 and is sufficiently small.

- Number of neutron particles per generation : 2000
- Number of neutron generation : 2050
- Number of skipped generation : 50
- Number of total history : 4 million

2.2 Analysis

2.2.1 Analysis Conditions

Specifications of stored fuel and NFR together with the conditions included in analysis model are described in this subsection.

2.2.1.1 Fuel Assembly Description

US-APWR 17x17 fuel assembly parameters used in the criticality analysis of NFR are listed in Table 2-1. Arrangement of fuel rods, control rod guide thimbles and an in-core instrumentation guide tube in the fuel assembly is shown in Figure 2-1. Fuel cladding is made of ZIRLO™ which is a zirconium base alloy with a small amount of niobium (Nb) added for increasing corrosion resistance.

2.2.1.2 US-APWR NFR Description

The NFR has a capacity to store a maximum of 180 new fuel assemblies. Rack configuration and design parameters are shown in Table 2-2 and Figure 2-2. As shown in Figure 2-2, the NFR is composed of two modules of 9x7 = 63 cells each and one module of 6x7+6x2 = 54 cells. The rack pitch is 16.9 inch (43.0cm). The rack material is stainless steel-304 (SS304). SS supporting structures are installed to support rack modules, but are located locally and peripherally, so they do not affect the criticality analysis.

Normally, new fuel assemblies are stored in racks in a dry condition. A drain system is provided for the New Fuel Storage Pit to preclude flooding. The new fuel pit is normally dry. Per paragraphs 2 and 3 of 10 CFR 50.68(b), the presence of moderator in the pit is an accident condition. Fuel misplacement and fuel drop are also accident conditions. In accordance with the double-contingency principle as defined in Section 3 of the Kopp memorandum (19 August 1998) "two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis." Section 4 of the Kopp memorandum further clearly defines two events with moderator in the pit (the two events also defined in paragraphs 2 and 3 of 10 CFR 50.68(b)) as accidents and states "Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered." In addition, the rack is designed to have no significant deformation which affects criticality analysis. Incidentally, from the double contingency principle, a fuel handling accident condition with flooding condition does not need to be considered, as stated in Reference [4]. Therefore, the criticality analysis for NFR addresses the case of flooding with water.

2.2.1.3 Assumptions

Based on the fuel assembly and NFR parameters, criticality analyses are performed for the following conditions.

Assumptions on Fuel Assembly

- The fresh UO_2 fuel assembly without burnable absorber is assumed to have a maximum enrichment of five weight percent, which is pursuant to 10 CFR 50.68 (b) item (7).
- The fuel cladding is conservatively assumed to be 100% zirconium which has a smaller

neutron absorption cross section compared with ZIRLO.

- The fuel rods, cladding and the control rod guide thimbles and the in-core instrumentation guide tube are modeled over the active fuel length of the fuel assembly. The grid spacers made of both Zircaloy-4 and Inconel 718 are conservatively neglected.
- The structural material at both ends of fuel assembly have many holes to allow coolant flow so the effective contribution as neutron reflector is small, thus they can be replaced by water or concrete. A 30 cm water layer is placed on the top of the effective fuel length and a concrete layer of 1 m thickness is placed on the underside.

Assumptions on NFR

- Calculations are performed simulating the actual NFR system.
- Water density of 62.43 lb/ft^3 ($1.0 g/cm^3$) is used to cover the maximum value and fractional densities are treated between 0 to 100 percent of full density so as to cover both flooded and optimum moderation.
- The concrete wall on the outside of the new fuel storage pit is modeled as 100 cm thick.
- Based on the double contingency principle, the simultaneous occurrence of an accident condition (ex. misloading, drop) with flooding condition need not be considered.

2.2.2 MCNP Model for NFR

As stated in 2.2.1.3, the evaluations are carried out using a complete model of the NFR including all three individual racks. The analysis model is shown in Figure 2-3. Note that the model shown in this figure shows the eccentric fuel positioning used in the tolerance analyses, while the reference calculations are performed with assemblies centered in the rack cells.

Sensitivity analyses for independent tolerances are carried out individually utilizing the finite model by changing the dimensions of the respective parameter. The values used in the model are found on Tables 2-1 and 2-2. However, the off-center locations of the fuel assemblies are considered in a way that maximizes keff by uniformly moving each fuel assembly to the direction of the center of the entire rack configuration. The analysis model is shown in Figure 2-3.

2.2.3 Material Composition

The density, composition and atomic density for fuel, cladding, tube and thimble material used in the analysis are shown in Table 2-3. The corresponding parameters for the SS rack, water between assemblies and as reflector material, and concrete are also shown in Table 2-3. For each composition, MCNP ZAID library names are listed in Table 2-4. The temperature of the rack is near to the temperature where the built-in neutron cross section data in the MCNP library was prepared. Specifically, the temperature is 293.6K except for Zircaloy which is 300K.

2.3 Results

Analysis results of fully flooded and optimum moderation conditions are described in the following subsections.

2.3.1 Uncertainties

The reactivity effect of each tolerance, and the statistical combination of the independent tolerances are shown in Table 2-5, for a water density of 10% and 100% of the full water density.

Uncertainty of individual tolerances are obtained by differing two effective multiplication factors calculated by Monte Carlo for two points, and adding the root of sum of squares of uncertainties (2σ) as probability error.

Namely,

$$\Delta keff_i = \left| keff_i - keff_0 \right| + \sqrt{\left(2 \times \sigma_0\right)^2 + \left(2 \times \sigma_\mathrm{i}\right)^2} \quad (9)$$

$keff_0$ = keff for normal condition
$keff_i$ = keff for model considering tolerance i
σ_0 : 1σ for nominal model
σ_i : 1σ for model considering tolerance i

2.3.2 Fully Flooded

As shown in Table 2-6, the analysis result is a keff of 0.9063 including uncertainty, which satisfies the design criteria of less than 0.95.

2.3.3 Optimum Moderation

Analysis results for various water densities from 0 to 100% are shown in Table 2-7 and in Figure 2-4. Optimum moderation occurs at 10% water density. Detailed results for this water density are shown in Table 2-8. Even at this condition keff is 0.9614 including uncertainty, and satisfies the design criteria of less than 0.98. Water density of either mist or foam from fire sprinkler is known in practice to be less than 1%, and at this condition, the keff is less than 0.8, thus the system is substantially sub-critical.

2.3.4 Abnormal Location of a Fuel Assembly

2.3.4.1 Mislocated Fresh Fuel Assembly

The mislocation of a fresh fuel assembly of the highest permissible enrichment (5.0wt% ^{235}U) could possibly occur outside of the NFR storage cells. However, since the NFR is normally dry, the reactivity effect of this accident is insignificant. To demonstrate this, a calculation is performed where a fuel assembly is mislocated in the gap between the two larger rack modules, as shown in Figure 2-5. All the fuel in these two modules are eccentrically positioned to be as close as possible to the mislocated fuel assembly. The mislocated fuel assembly is located directly face adjacent to a fresh fuel assembly in each rack. The results of the calculation are presented in Table 2-9.

Table 2-1 MHI 17x17 Fuel Assembly Parameters of US-APWR for Criticality Analysis in NFR and SFR

Parameter	Design Parameters
Fuel Rod Configuration	17x17 (Figure 2-1)
Rods per Assembly	264
Control Rod Guide Thimble / In-core Instrumentation Guide Tube per Assembly	24 / 1
Rod Pitch	0.496[ ]inch
Active Fuel Length	165.4 inch
Pellet Outer Diameter (OD)	0.322[ ]inch
Enrichment	5.0 wt%U-235
UO_2 Density (% of Theoretical Density (TD))	97[ ]% of TD
Cladding OD	0.374[ ]inch
Cladding Inner Diameter (ID)	0.329[ ]inch
Cladding Material	ZIRLO (modeled as 100% Zr)
Control Rod Guide Thimble OD	0.482 inch
Control Rod Guide Thimble ID	0.450 inch
In-core Instrumentation Guide Tube OD	0.482 inch
In-core Instrumentation Guide Tube ID	0.450 inch
Control Rod Guide Thimble / In-core Instrumentation Guide Tube Material	Zircaloy-4 (modeled as 100% Zr)
[	]
Burnable Absorbers	None
Burn-up	None

Table 2-2 Design Parameters for NFR

Parameter	Design Parameters
Storage Cells	180
Cell Center-to-Center Pitch	16.9[ ] inch
Cell Inner Dimension (Width)	8.8[ ] inch
Cell Wall Thickness	0.209[ ] inch
Rack to Wall Distance East	19.47 inch
Rack to Wall Distance West	18.70 inch
Rack to Wall Distance N-S	21.69 inch
Rack to Rack Distance	9.25[ ] inch
Cell Wall Material	Stainless Steel

Table 2-3 (1/3) Materials and Compositions for NFR and SFR

(1) Fresh UO_2 Fuel Assembly (Enrichment = 5.0 wt%)

Material Condition (1)	**Isotope**	**Atom Density (atoms/barn-cm)**
a. For SFR Nominal Model Fractional TD = 97%	^{235}U	
	^{238}U	
	O	
b. For NFR Worst Case Model and SFR Tolerance Sensitivity Analysis Fractional TD =	^{235}U	
	^{238}U	
	O	
Zircaloy (6.55 g/cm^3) (2)	Zr	4.3239×10^{-2}

(1) UO_2 Pellet Density is 10.96 g/cm^3 for 100% TD.
(2) Conservatively, ZIRLO cladding and Zircaloy were treated as 100 % Zr.

Table 2-3 (2/3) Materials and Compositions for NFR and SFR

(2) Structure Material

Material	Isotope	Atom Density (atoms/barn-cm)
SS304 (7.84 g/cm^3)	Ni	8.047×10^{-3}
	Cr	1.720×10^{-2}
	Fe	5.838×10^{-2}
	Mn	1.720×10^{-3}
Metamic™, 30.5 wt% B4C (min.) (2.6358 g/cm^3) Note Used in SFR only, except for DFR	^{10}B	6.920×10^{-3}
	^{11}B	2.812×10^{-2}
	C	8.760×10^{-3}
	Al	4.089×10^{-2}
Concrete (2.35 g/cm^3)	H	8.806×10^{-3}
	O	4.623×10^{-2}
	Na	1.094×10^{-3}
	Al	2.629×10^{-4}
	Si	1.659×10^{-2}
	K	7.184×10^{-4}
	Ca	3.063×10^{-3}
	Fe	3.176×10^{-4}

Table 2-3 (3/3) Materials and Compositions for NFR and SFR

(3) Between assemblies Water with or without boron and reflector water

Material	Water Density(1) (% of full density)	Isotope	Atom Density (atoms/barn-cm)
Water (Moderator)	100	H	6.6854×10^{-2}
		O	3.3427×10^{-2}
	10	H	6.6854×10^{-3}
		O	3.3427×10^{-3}
Boric Acid Water (Boron conc. 800ppm)	100	H	6.6854×10^{-2}
		O	3.3427×10^{-2}
		^{10}B	8.8225×10^{-6}
Water (Reflector)	100	H	6.6854×10^{-2}
		O	3.3427×10^{-2}

(1) 100 % of full density : 62.43 lbm/ft^3 ($1.0 g/cm^3$)

Table 2-4 MCNP ZAIDs Used for Each Nuclide

Nuclide	ENDF/B-V
H	1001.50c
^{10}B	5010.50c
^{11}B	5011.55c
C	6012.50c
O	8016.50c
Na	11023.51c
Al	13027.50c
Si	14000.51c
K	19000.51c
Ca	20000.51c
Cr	24000.50c
Mn	25055.50c
Fe	26000.55c
Ni	28000.50c
Zr	40000.56c
^{235}U	92235.50c
^{238}U	92238.50c

Table 2-5 Results of the NFR Tolerance Calculations

Calculation Description	10% Moderator Density			100% Moderator Density		
	keff	σ	Delta keff	keff	σ	Delta keff
Reference keff	0.9448	0.0003	n/a	0.8935	0.0004	n/a
Pellet Density max						
Pellet OD max						
Clad OD max						
Clad OD min						
Clad ID max						
Clad ID min						
Pin Pitch max						
Pin Pitch min						
Cell Pitch max						
Cell Pitch min						
Cell ID max						
Cell ID min						
Wall Thickness max						
Wall Thickness min						
Rack Gap max						
Rack Gap min						
Eccentric Position	0.9474	0.0003	0.0034	0.8949	0.0004	0.0025
Square Root Sum of the Squares (positive results)			0.0124			0.0081
2 Sigma (max of all cases)			0.0006			0.0008

Note: The maximum positive tolerance value for each case was used.

Table 2-6 Results of the NFR MCNP5 Calculations, Fully Flooded Case

Parameter	Value
Moderator Density	100%
Uncertainties:	
Bias Uncertainty 1.899 x 0.003 (95%/95%)	0.0057
Calculation Statistics (95%/95%, 2σ)	0.0008
Calculated Tolerances (see Table 2-5)	0.0081
Statistical Combination of Uncertainties	0.0099
Calculated MCNP5 keff	0.8935
Calculation Bias	0.0029
Maximum keff	0.9063
Regulatory Limit	0.9500

Table 2-7 NFR Analysis Results on Surveying the Optimum Moderation Condition

% Moderator Density	Calculated keff	σ
0	0.5819	0.0003
3	0.8047	0.0003
8	0.9400	0.0003
9	0.9442	0.0003
10	0.9448	0.0003
11	0.9411	0.0003
12	0.9358	0.0003
15	0.9080	0.0003
20	0.8477	0.0003
30	0.7490	0.0003
60	0.7317	0.0004
90	0.8533	0.0004
95	0.8747	0.0004
100	0.8935	0.0004

Note: The results above indicated a Δkeff of 0.0006 between the 10% and 9% moderator density cases. This difference is within 2σ (0.0006) and the two cases are therefore statistically equivalent. The 10% case was used for determination of the tolerances as shown in Table 2-5.

Table 2-8 Results of the NFR MCNP5 Calculations, Optimum Moderation Case

Parameter	Value
Moderator Density	10%
Uncertainties:	
Bias Uncertainty 1.899 x 0.003 (95%/95%)	0.0057
Calculation Statistics (95%/95%, 2σ)	0.0006
Calculated Tolerances (see Table 2-5)	0.0124
Statistical Combination of Uncertainties	0.0137
Calculated MCNP5 keff	0.9448
Calculation Bias	0.0029
Maximum keff	0.9614
Regulatory Limit	0.9800

Table 2-9 Results of the NFR MCNP5 Calculations, Mislocated Fuel Assembly Case

Parameter	Value
Mislocated Fuel Assembly keff	0.5826
Regulatory Limit	0.9500

Note: The result of this calculation is so subcritical that it is not necessary to recalculate the tolerances and bias for the dry accident condition to determine keff

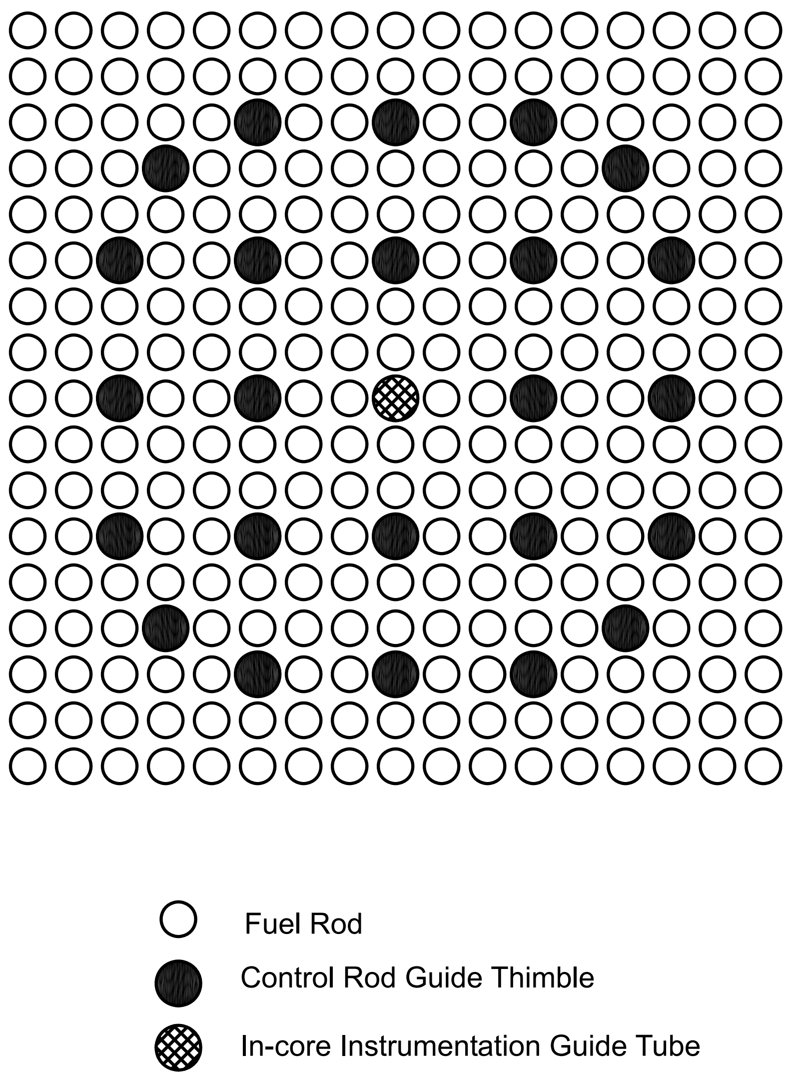


Figure 2-1 MHI US-APWR 17x17 Fuel Assembly Cross Section

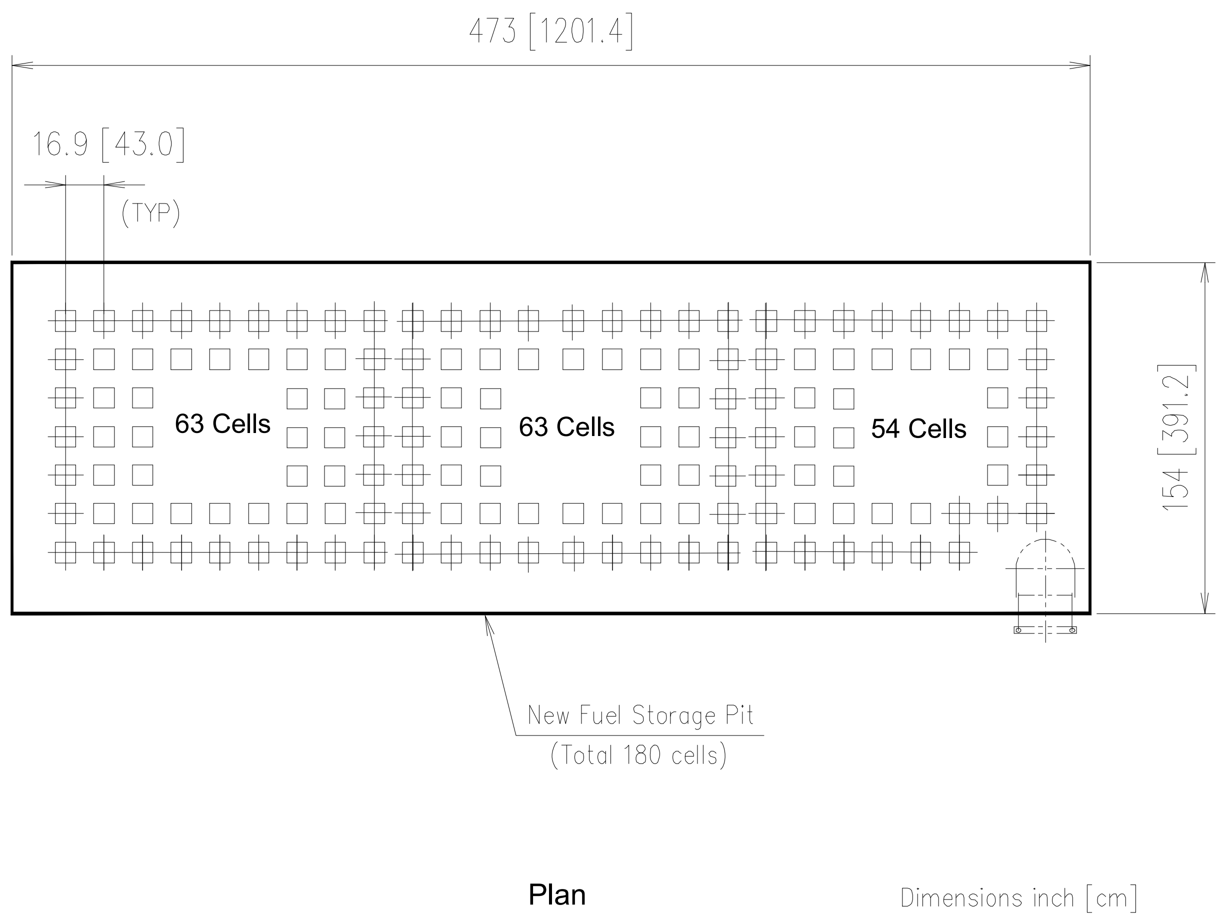
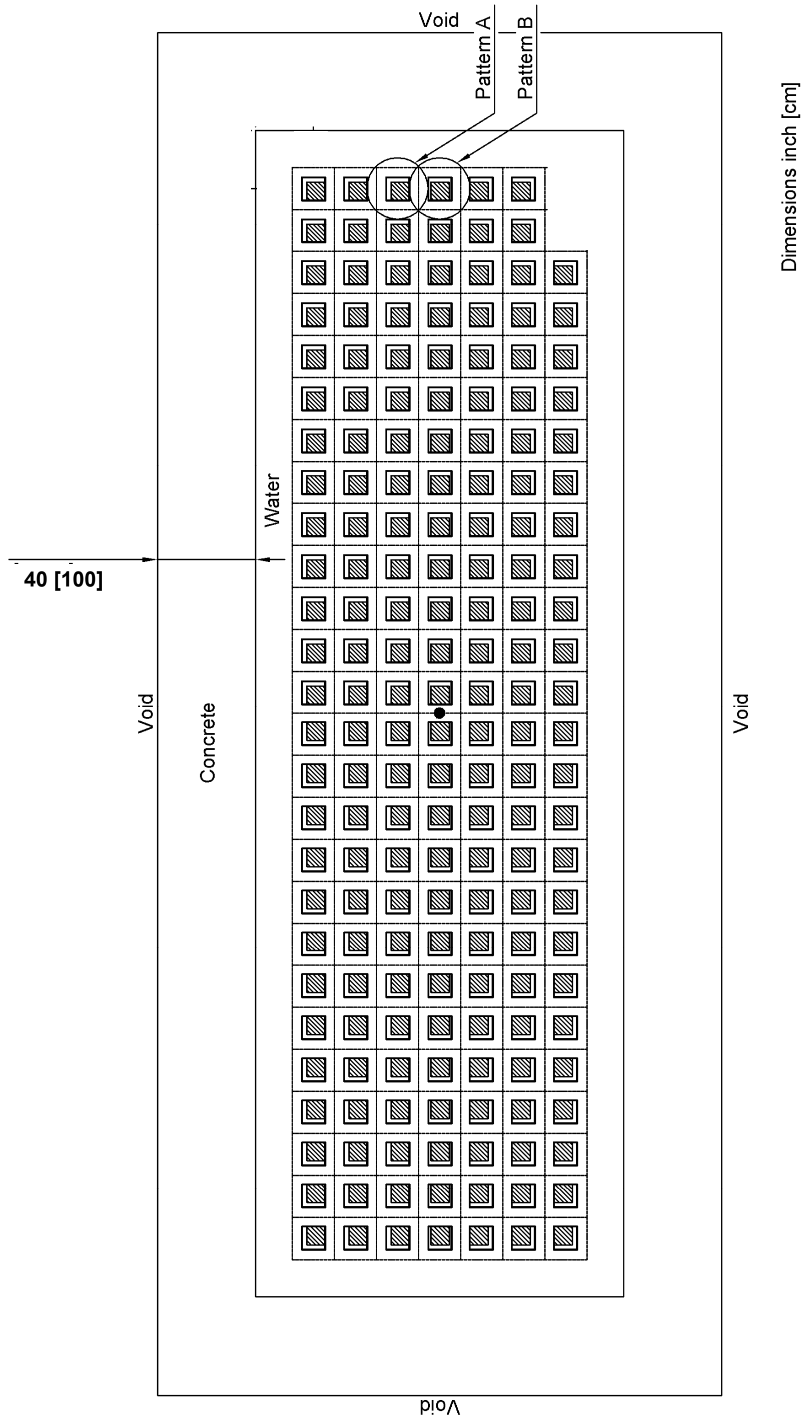
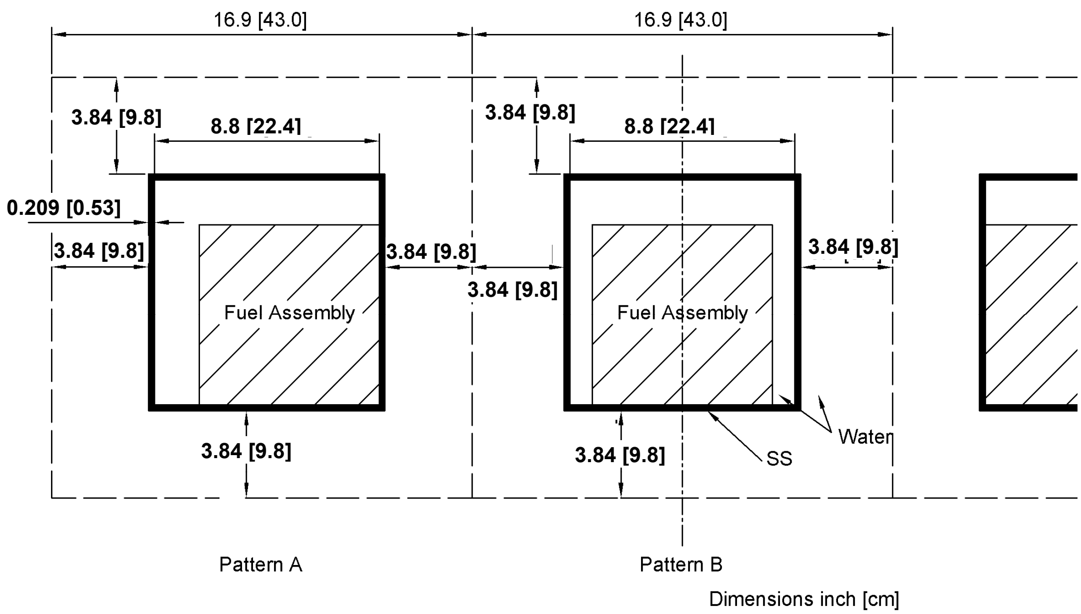


Figure 2-2 Configuration of NFR



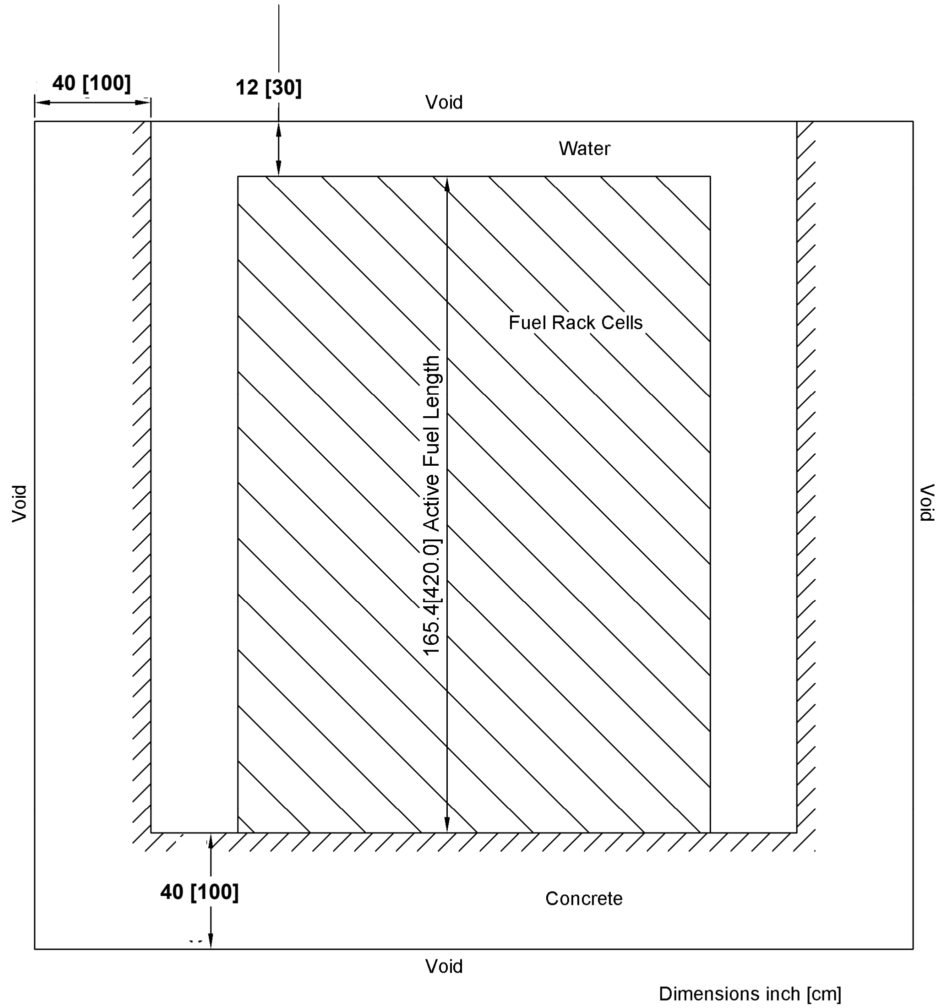
Plan for Whole
(Circle at the center of the rack means the direction of off-center arrangement of fuel assemblies.)

Figure 2-3 (1/3) NFR as modeled in MCNP (Eccentric Fuel Positioning)



Plan of detailed rack model

Figure 2-3 (2/3) NFR as modeled in MCNP (Eccentric Fuel Positioning)



Elevation

Figure 2-3 (3/3) NFR as modeled in MCNP

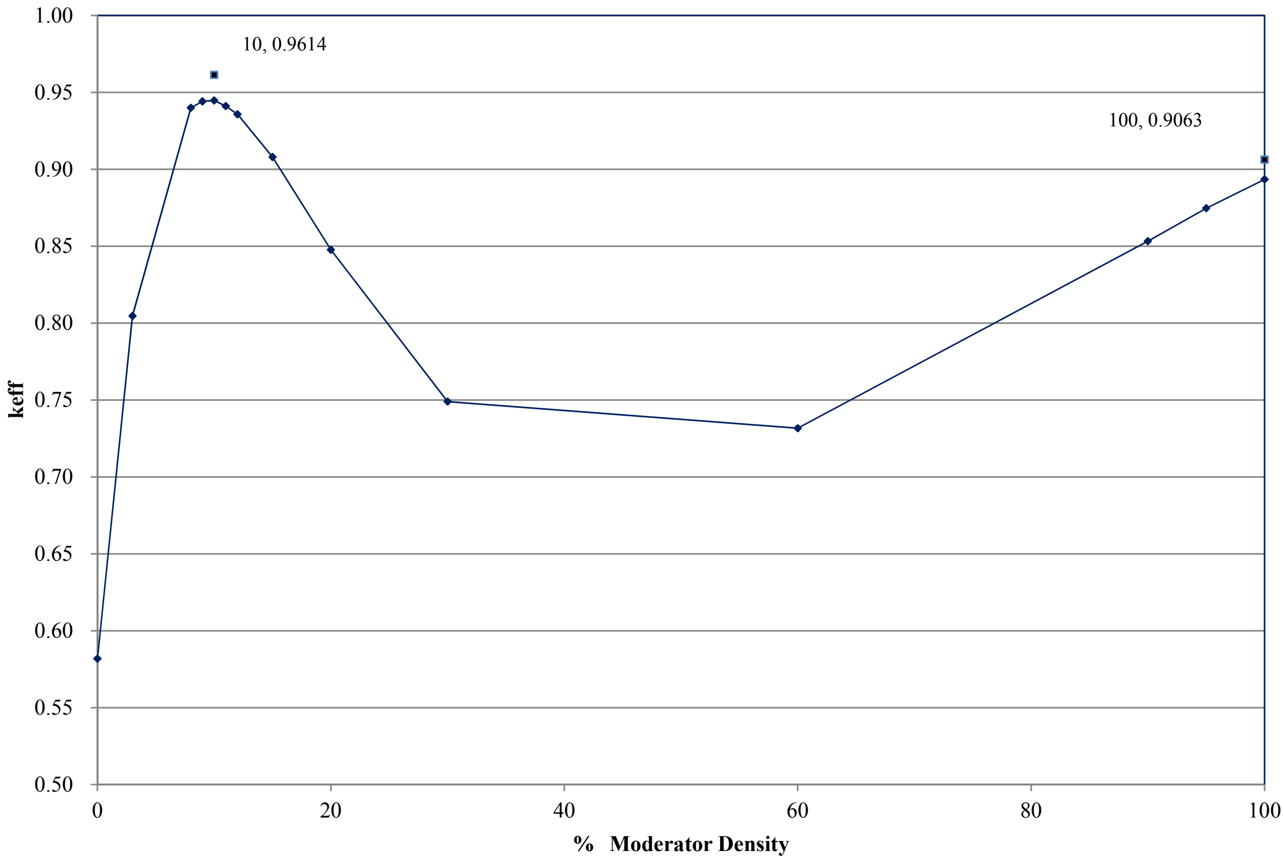


Figure 2-4 Results of Keff vs Various Water Density of NFR

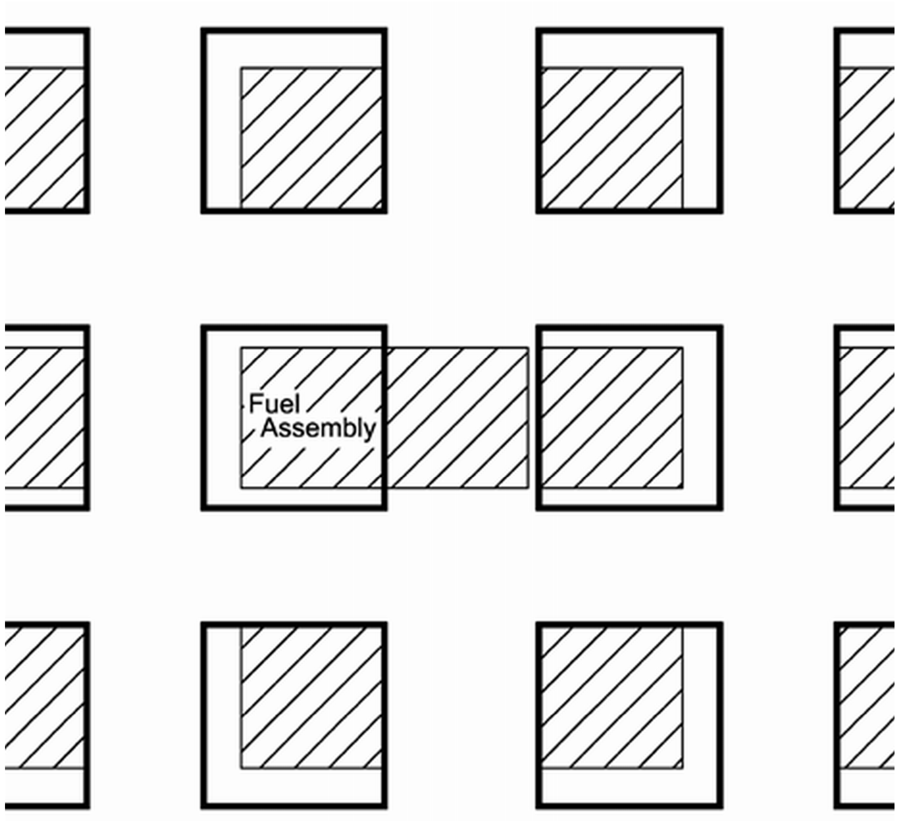


Figure 2-5 NFR as modeled in MCNP5 (Mislocated Fuel Assembly)

3.0 CRITICALITY ANALYSIS OF SFR

Chapter 3 describes the criticality analysis results for US-APWR 17×17 fuel assemblies stored in SFR. It is shown that the maximum value of keff at pure water flooded condition is less than 1.0 during normal condition, and when applying soluble boron credit, keff is less than or equal to 0.95, therefore the design criteria is satisfied and subcriticality is maintained.

3.1 Design Method

Design criteria, evaluation results and analysis code are described in the following subsections.

3.1.1 Design Criteria

The design criteria are pursuant to the 10 CFR 50.68 (b) item (4) for SFR as follows:
"For spent fuel storage racks, the maximum keff value, including all biases and uncertainties, must be less than or equal to 0.95 with partial credit for soluble boron credit and less than 1.0 with full density unborated water, at a 95 percent probability, 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity."

Therefore, an evaluation is performed to show that subcriticality is maintained at the pure water flooded condition followed, if necessary, by an evaluation to determine the boron concentration to keep keff less than or equal to 0.95.

For accident conditions other than boron dilution, the highest boron concentration to offset the reactivity increase is determined to cover the most limiting accident as a single failure.

3.1.2 Evaluation

Based on the design criteria in the previous section and equation (4) in section 2.1.2 of NFR, the evaluation equations for SFR are expressed as follows.

Pure water : $$k_p + 0.0029 + \left(\Delta k_p^2 + (1.899 \times 0.0030)^2\right)^{1/2} < 1.0 \qquad (7)$$

Borated water : $$k_p + 0.0029 + \left(\Delta k_p^2 + (1.899 \times 0.0030)^2\right)^{1/2} \leq 0.95 \qquad (8)$$

3.1.2.1 Reactivity Uncertainty Due to Tolerances

Statistical combination of the reactivity effect of independent tolerances is performed for this SFR analysis. The tolerances of the fuel assembly and rack cell tolerances are evaluated individually.

3.1.3 Analysis Code

As stated in Section 1.1 the criticality safety analysis uses the three-dimensional Monte Carlo code MCNP version 5.1.40 and the continuous-energy neutron data ENDF/B-V.
Additionally, for the S(α, β) thermal scattering data, "lwtr.01t" for hydrogen in light water is applied to water. Though the scattering effect as reflector is small, it is applied to the hydrogen in floor and wall concretes.

Neutron generation histories in Monte Carlo calculation were set to four million as shown below. In this condition, 1σ is approximately 0.0004 and is sufficiently small.

- Number of neutron particles per generation: 2000
- Number of neutron generation: 2050
- Number of skipped generation: 50
- Number of total history: 4 million

3.2 Analysis

3.2.1 Analysis Conditions

Specifications of stored fuel and SFR together with conditions to be used in analysis model are described in this subsection.

3.2.1.1 Fuel Assembly Description

US-APWR 17x17 fuel assembly parameters used in the criticality analysis of SFR are listed in Table 2-1. Arrangement of fuel rods, control rod guide thimbles and an in-core instrumentation guide tube in the fuel assembly is shown in Figure 2-1. Fuel cladding is made of ZIRLO which is a zirconium base alloy with a small amount of niobium (Nb) added for increasing corrosion resistance.

3.2.1.2 US-APWR SFR Description

The SFR storage cells are composed of stainless steel boxes separated by a water gap, with fixed neutron absorber panels centered on each side. The steel walls define the storage cells, and stainless steel sheathing supports the neutron absorber panel and defines the boundary of the flux-trap (water gap) used to augment reactivity control. Stainless steel channels (water gap flats) connect the storage cells in a rigid structure and define the flux-trap between the neutron absorber panels. Neutron absorber panels are installed on all exterior walls facing other racks, including the DFR. The SFR have a capacity to store a maximum of 900 fuel assemblies. Rack configuration and design parameters are shown in Table 3-1 and Figure 3-1. As shown in Figure 3-1, the SFR is composed of six modules of 11.1 inch (28.2 cm) cell pitch. They are three modules of 12×12=144 cells each and three modules of 12x13=156 cells each. SS supporting structures are installed to support the rack modules, but are located locally and peripherally, so they do not affect the criticality analysis.

In addition, the DFR which can store 12 Damaged Fuel Containers in a row are provided in vicinity to the SFR. Damaged fuel is inserted into Damaged Fuel Containers and stored in the DFR Cells. The rack material is stainless steel, the rack pitch is 24 inch (60.9 cm), and it is 21.7 inch (55.0 cm) apart from the SFR. Rack configuration and design parameters of DFR are shown in Table 3-2 and Figure 3-1.

3.2.1.3 Assumptions

Using the fuel and SFR parameters, analyses are performed for the following conditions:

Assumptions on Fuel Assembly

- The fresh UO_2 fuel assembly without burnable absorber is assumed to have a maximum

enrichment of five weight percent which is pursuant to 10 CFR 50.68 (b) item (7).

- The fuel cladding is conservatively assumed to be 100% zirconium which has a smaller neutron absorption cross section compared with ZIRLO.
- The fuel rods, cladding and the control rod guide thimbles and the in-core instrumentation guide tube are modeled over the active fuel length of the fuel assembly. The grid spacers made of both Zircaloy-4 and Inconel 718 are conservatively neglected.
- The structural material at both ends of fuel assembly have many holes to allow coolant flow so the effective contribution as neutron reflector is small, thus they can be replaced by water or concrete. At the flooded condition, a 30 cm water layer and a 1 m concrete layer have equivalent reflector effect and the thickness is sufficient to maximize the reflection effect. Then, 30 cm water layer is placed on the top of the effective fuel length and a concrete layer of 1 m thickness is placed on the underside.

Assumptions on SFR

- Rack calculations at nominal conditions are conducted for rack cells on a 11.1 inch (28.2 cm) pitch in an infinitely repeated array system, and the tolerances and biases are evaluated separately.
- For the uncertainties in the fuel assembly placement in each cell, analysis are carried out for centrally off-centered 4, 16, 36 fuel assembly configuration given reflective boundary condition. The maximum reactivity increase among them is selected as the reactivity uncertainty.
- Water density of 62.43 lb/ft^3($1.0 g/cm^3$) is used to cover the maximum value.
- The neutron absorber length in the SFR is 173 inches, but it is conservatively modeled to be the same length as the active region, 165.4 inch.
- The neutron absorber is modeled using worst case modeling and therefore uses the minimum boron content, width and thickness.

Assumptions on DFR

- The 12 DFR made of SS are sufficiently isolated from each other and also from the SFR racks from the neutron interaction viewpoint. The Damaged Fuel Container made of SS inserted into the DFR are conservatively neglected. Therefore only the fuel assemblies and the DFR are considered.

3.2.2 MCNP Model for SFR

3.2.2.1 Nominal Model

As stated in Subsection 3.2.1.3, the SFR MCNP5 model consists of a single rack cell (rack cell wall, neutron absorber, sheathing and water gap) with reflective boundary conditions through the centerline of the water gaps, thus simulating an infinite array of SFR storage cells. The storage rack cell is modeled the same length as the active fuel and all other storage rack materials are neglected. The neutron absorber is modeled with the worst case bounding values shown in Table 3-1, and the Metamic^{TM} panel is centered in the gap between the cell wall and sheathing. Note that the SFR has two sheathing types, boundary sheathing and inner sheathing. The boundary sheathing is along the exterior of the rack model only and is thicker than the inner sheathing to provide protection to the rack during transport. The SFR model conservatively uses the inner sheathing thickness only. Analysis model is shown in Figure 3-2.

3.2.2.2 Uncertainty Analysis Models for Tolerances

Sensitivity analyses for independent tolerances are carried out individually utilizing the above nominal model by changing the dimension of the objective parameter. However, for the fuel placement cases stated in assumptions in 3.2.1.3, simulations by this model are impracticable. The analysis models for these cases are shown in Figure 3-3.

3.2.2.3 DFR

The DFR are located in the SFP between the SFR and the SFP wall as shown in Figure 3-1. The DFR have 2 sets of 6 steel box storage cells with the dimensions given in Table 3-2. As stated in assumptions in 3.2.1.3, only the fuel assemblies and the SS rack cells are considered. The dimensions used in the model are based on worst case bounding values for the DFR (eccentric fuel positioning, minimum wall thickness, and minimum cell pitch) and nominal values for the SFR (except that the fuel in the SFR is positioned at its closest approach to the DFR). The model includes the entire SFP and was used to show by a sensitivity study that the reactivity of the SFP is less than the SFR and therefore bounded by the SFR calculations.

Sensitivity evaluation is carried out by comparing the analyses with and without DFR.

3.2.3 Material Composition

For fuel, cladding and thimble materials, the density, composition and atomic density used in the analysis are shown in Table 2-3. The corresponding parameters for the rack, neutron absorber, water, and concrete material compositions are also shown in Table 2-3. For each composition, MCNP ZAID library names are listed in Table 2-4.

The maximum bulk pool water temperature that shall be maintained to cool the SFR is 120° F (48.9° C) at normal condition, 140° F (60° C) for a single failure condition, and 200° F (93.3° C) at an accident during a full core offload. The SFP design temperature is at 200° F (93.3° C). Considering dependency of water density to temperature, the use of the library made at ambient temperature is conservative and the value of water density 62.43 lb/ft^3(1.0g/cm^3) is taken for the condition that maximizes the reactivity.

3.3 Results

As shown in Table 3-4, the final keff value is 0.9150 without applying credit for soluble boron, including uncertainties. This value is well below both design criteria of less than or equal to 0.95 and less than 1.0. Therefore, calculations with credit for soluble boron are not required.

Uncertainty of individual tolerances are obtained by differing two keffs calculated by Monte Carlo for two points, and adding the root of sum of squares of uncertainties (2σ) as probability error.

Namely,

$$\Delta keff_i = \left| keff_i - keff_0 \right| + \sqrt{(2 \times \sigma_0)^2 + (2 \times \sigma_i)^2} \tag{9}$$

$keff_0$ = keff for normal condition
$keff_i$ = keff for model considering tolerance i
σ_0 : 1σ for nominal model

σ_i : 1σ for model considering tolerance i

3.3.1 Sensitivity of DFR

Sensitivity analysis results are shown in Table 3-5. The difference between the keff with and without the DFR is within the uncertainties of the analysis.

3.3.2 Abnormal and Accident Conditions

The effects on reactivity of credible abnormal and accident conditions are examined in this section. This section identifies which if any of the credible abnormal or accident conditions would result in exceeding the limiting reactivity (keff < 0.95). The double contingency principal [4] specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

3.3.2.1 Dropped Assembly – Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 17 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in an increase in reactivity and no separate calculation is performed for the drop accident.

3.3.2.2 Dropped Assembly – Vertical

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly or that might be empty. The mechanical implications of such a drop been evaluated (Reference [9]). The results presented are conservatively bounded from a criticality perspective by assuming that the vertical drop accident results in a loss of neutron absorber at the top of the rack of 3 inches. This vertical drop accident was therefore modeled using the SFR single cell infinite array model with three inches of neutron absorber at the top of the rack replaced with water. The results of this calculation are shown in Table 3-6. The results indicate that there is no significant reactivity effect from this accident.

3.3.2.3 Abnormal Location of a Fuel Assembly

3.3.2.3.1 Misloaded Fresh Fuel Assembly

Since the fuel storage racks are qualified for storage of fresh fuel of the highest anticipated reactivity, the misloading of a fresh fuel assembly is of no concern.

3.3.2.3.2 Mislocated Fresh Fuel Assembly

The mislocation of a fresh unburned fuel assembly could, in the absence of soluble neutron absorber, result in exceeding the regulatory limit (keff < 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt% 235U) were to be accidentally mislocated outside of a storage rack adjacent to other fuel assemblies. The pool layout was examined to determine a credible worst case location for this accident, and it was determined to be in the area between the SFR and the DFR and between DFR cells. Multiple

cases were run for various distances between the rack cells for a mislocated fuel assembly. The maximum of these cases is presented in Table 3-6 and Figure 3-4. The amount of soluble boron needed to meet regulatory requirements was also determined by running the given accident cases with 800 ppm soluble boron. This boron concentration is controlled by Technical Specifications.

3.3.2.4 Rack Movement

In the event of seismic activity, there is the possibility that the SFP storage racks may move. The base plate extensions preclude the racks from moving closer together (see Section 3.2.1.2) in the horizontal direction. However, the analysis in Reference [10] indicates that the rack tops might touch due to rack tipping during seismic activity. This scenario may result the fuel at the top of the fuel assemblies being in closer proximity across the gap between racks than in the normal condition. To investigate the reactivity effect of this situation, the full SFP model (without the DFR) was divided into two sections and the gap between flux traps was nearly eliminated. This was done along the entire vertical length of the fuel to conservatively model the rack tip over situation during a seismic event and is bounding for the rack tip condition. The results are shown in Table 3-6.

Table 3-1 Design Parameters for SFR

Parameter	Design Parameters
Storage Cells	900
Cell Center-to-Center Pitch	11.1[ ]inch
Cell Inner Dimension (Width)	8.8[ ]inch
Cell Wall Thickness	0.090[ ] inch
Cell Inner Sheathing	0.024[ ]inch
Neutron Absorber Gap	0.118 inch
Neutron Absorber Thickness	[ ] inch
Neutron Absorber Width	[ ] inch
Cell Wall Material and Neutron Absorber	Stainless Steel and Metamic™
Metamic™ wt% B4C	31.0[ ]

Table 3-2 Design Parameters for DFR

Parameter	Design Parameters
Storage Cells	12
Cell Center-to-Center Pitch	24[ ] inch
Center-to-Center Pitch to near SFR Cell	21.7 inch
Cell Inner Dimension (Width)	9.25 inch
Cell Wall Thickness	0.375[ ] inch
Cell Wall Material	Stainless Steel

Table 3-3 Results of the MCNP5 SFR Tolerance Calculations

Calculation Description	keff	σ	Delta-keff
Reference keff	0.9006	0.0004	n/a
Pellet Density max			
Pellet OD max			
Pin Pitch max			
Pin Pitch min			
Clad OD max			
Clad OD min			
Clad ID max			
Clad ID min			
Cell ID max			
Cell ID min			
Wall Thk max			
Wall Thk min			
Cell Pitch max			
Cell Pitch min			
Sheathing max			
Sheathing min			
Eccentric Positioning (single cell)	0.8999	0.0004	0.0004
2x2 Cell Model Reference	0.9007	0.0004	n/a
2x2 Cell Model Eccentric Positioning	0.9007	0.0004	0.0011
3x3 Cell Model Reference	0.9009	0.0004	n/a
3x3 Cell Model Eccentric Positioning	0.8997	0.0004	-0.0001
Square Root Sum of the Squares (positive results)			0.0100
2 Sigma (max of all cases)			0.0008

Note: The maximum positive tolerance value for each case was used.

Table 3-4 Results of the SFR MCNP5 Calculations

Parameter	Value
Uncertainties:	
Bias Uncertainty 1.899 x 0.003 (95%/95%)	0.0057
Calculation Statistics (95%/95%, 2σ)	0.0008
Calculated Tolerances (see Table 3-3)	0.0100
Statistical Combination of Uncertainties	0.0115
Calculated MCNP5 keff (no soluble boron)	0.9006
Calculation Bias	0.0029
Maximum keff (no soluble boron)	0.9150
Regulatory Limit†	0.9500

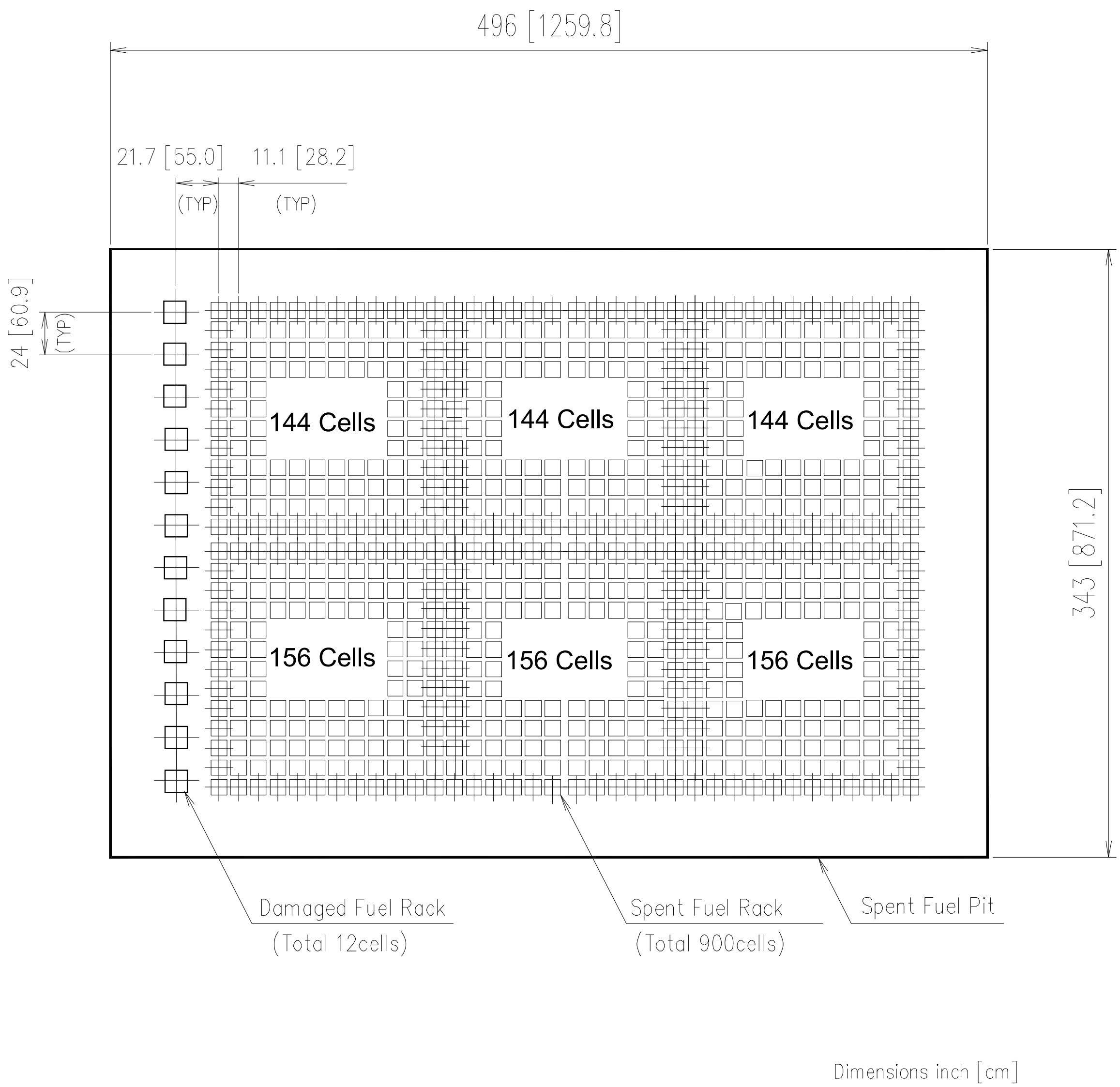
† The keff meets both requirements (1.0 and 0.95).

Table 3-5 SFR Analysis Results with and without DFR

Case	Calculated Keff
SFR Rack Cells only	0.9000 ± 0.0004
With DFR	0.8980 ± 0.0004

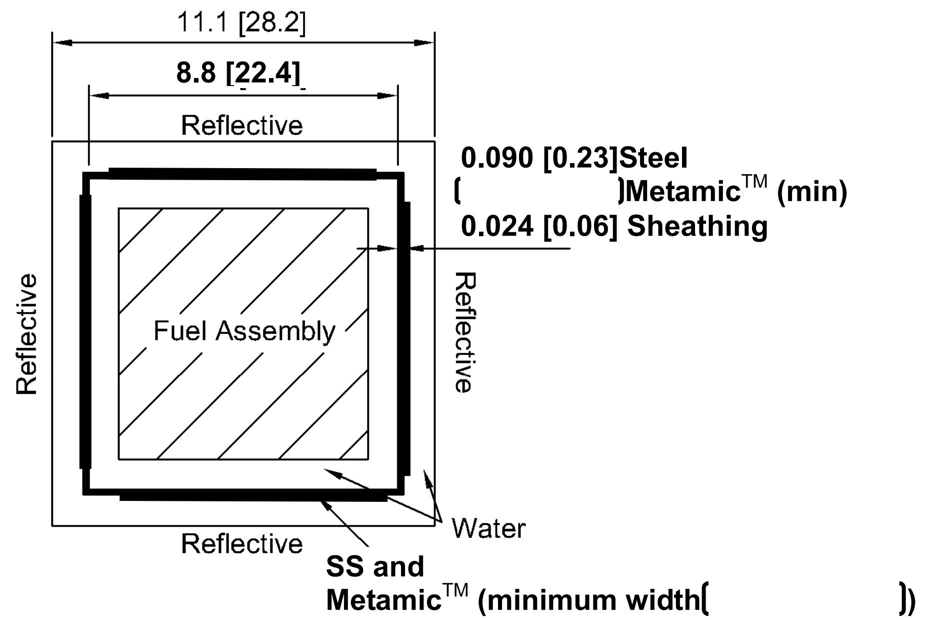
Table 3-6 Summary of SFR Accident Case Calculations

SFR Mislocated Fuel Assembly Calculations	
Parameter	**keff**
Mislocated Fuel Assembly keff (0 ppm soluble boron)	1.0283
Total Bias and Uncertainty from Table 3-4	0.0144
Maximum keff (0 ppm soluble boron)	1.0427
Mislocated Fuel Assembly keff (800 ppm soluble boron)	0.9115
Maximum keff (800 ppm soluble boron)	0.9259
Dropped Fuel Assembly Accident Results	
Parameter	**keff**
Reference Case	0.9006
3 inches of Metamic™ Loss Case	0.9014
SFR Seismic Tip Accident Calculations	
Parameter	**keff**
SFR Tip Accident Case keff (0 ppm soluble boron)	0.9528
Total Bias and Uncertainty from Table 3-4	0.0144
Maximum keff (0 ppm soluble boron)	0.9672
SFR Tip Accident Case keff (800 ppm soluble boron)	0.8728
Maximum keff (800 ppm soluble boron)	0.8872



Plan

Figure 3-1 Configuration of SFR



Dimensions inch [cm]

Plan

Figure 3-2 (1/2) Nominal MCNP Model of SFR

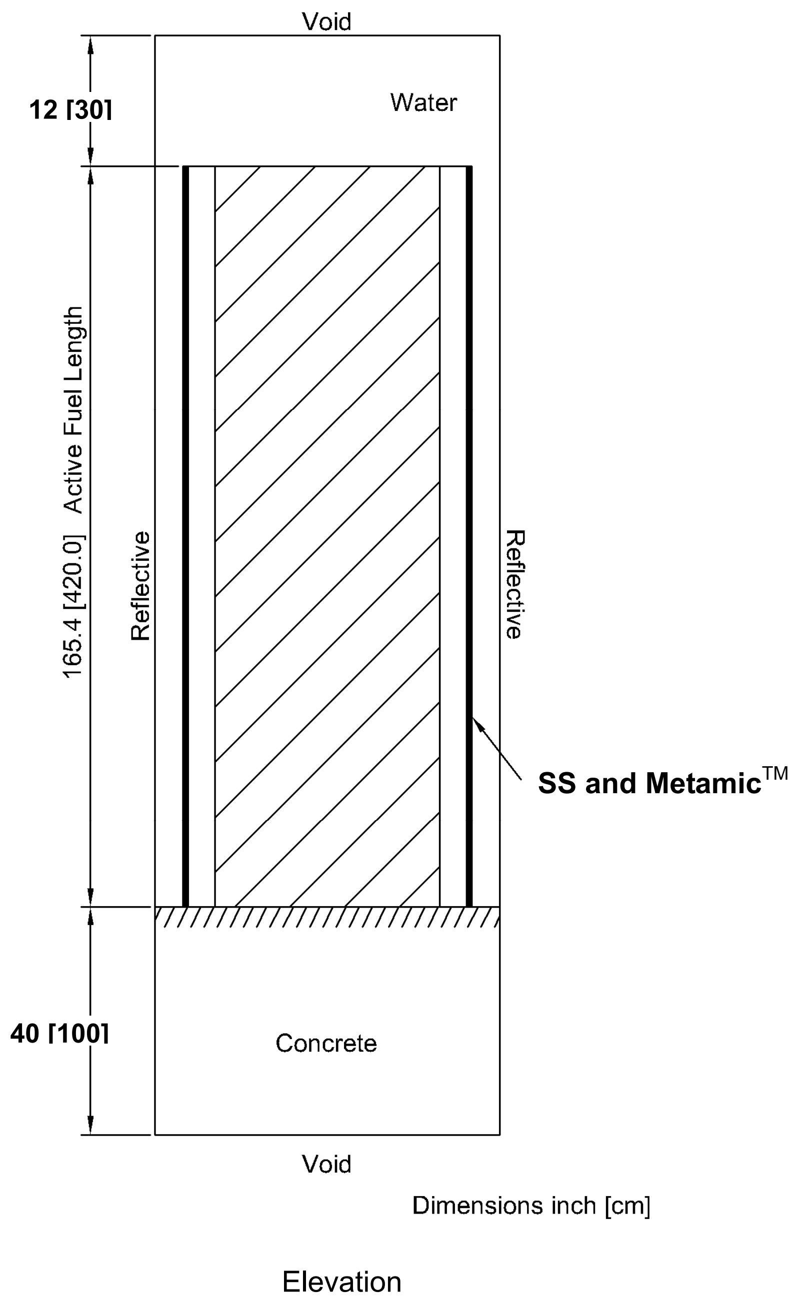
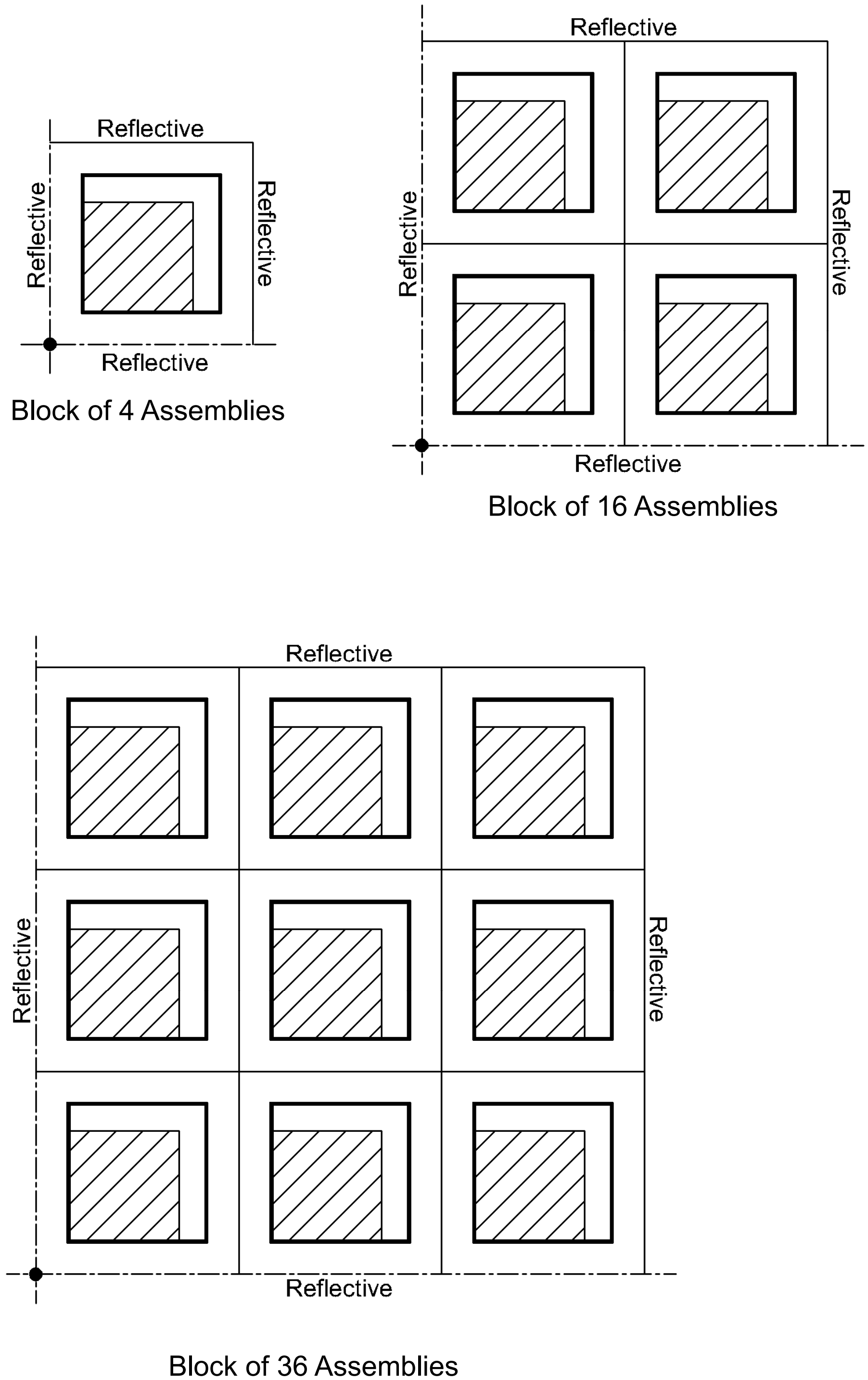


Figure 3-2 (2/2) Nominal MCNP Model of SFR



Plan
(Dotted circle means the direction of off-center arrangement of fuel assemblies.)

Figure 3-3 MCNP Model for Fuel Displacement within Cells of SFR

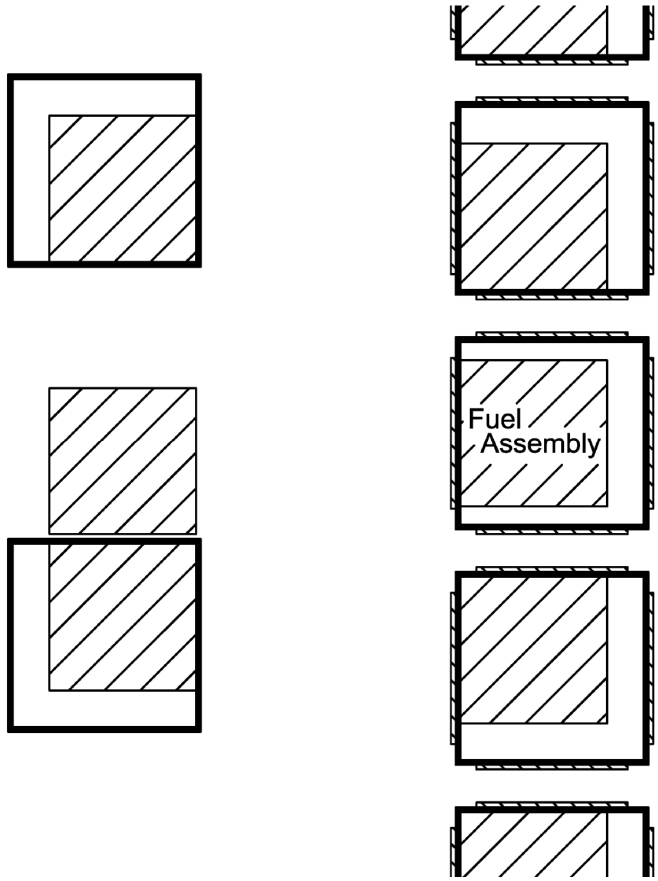


Figure 3-4 MCNP Model for Mislocated Fuel Assembly in SFP (Maximum keff)

4.0 CONCLUSIONS

From the evaluation results of NFR described in Chapter 2, of the SFR in Chapter 3, it is confirmed that the design criteria of 10CFR50.68 are met and that subcriticality is maintained.

Under normal conditions, no soluble boron is required for any of the racks. For accident conditions, the requirement of minimum soluble boron concentration to assure the keff is less than 0.95 was set to 800 ppm. This is far less than the normal operating conditions of 4000 ppm which is controlled by Technical Specifications.

5.0 REFERENCES

[1] US-APWR Fuel System Design Parameters List, MUAP-07018-P, Dec. 2007.

[2] Prevention of Criticality in Fuel Storage and Handling, 'General Design Criteria for Nuclear Power Plants,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, Criterion 62, U.S. Nuclear Regulatory Commission, Washington, DC.

[3] 'Criticality Accident Requirements,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50.68, U.S. Nuclear Regulatory Commission, Washington, DC.

[4] Kopp, L. Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants. U.S. Nuclear Regulatory Commission, Washington, DC, February 1998.

[5] Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors. ANSI/ANS-8.17-2004, American National Standards Institute/American Nuclear Society.

[6] HI-2094264 Revision 7, "Fuel Storage Racks Criticality Analysis for Comanche Peak US-APWR"

[7] X-5 Monte Carlo Team, MCNP - A General N-Particle Transport Code, Version 5, LA-UR-03-1987, Los Alamos National Laboratory, April 2003 revised Oct 3, 2005 (MCNP Team, "MCNP 5.1.40 RSICC Release Notes," LA-UR-05-8617 (Nov. 10, 2005).

[8] Validation of the MHI Criticality Safety Methodology, MUAP-07020, Dec. 2007.

[9] HI-2084212 Revision 7, "Mechanical Accident Analysis for Comanche Peak US-APWR Fuel Storage Racks"

[10] HI-2084230 Revision 3, "Structural/Seismic Analysis for US-APWR Spent and New Fuel Storage Racks"